DOE/EIS-0310
December 2000

FINAL Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility

Summary

United States Department of Energy
Office of Nuclear Energy, Science and Technology
Washington, DC

Cover photograph and illustration identification, beginning at top center and going clockwise:

- Radioisotope tagged monoclonal antibodies, “smart bullets,” target malignant cells for diagnosis and treatment of diseases
- The Fast Flux Test Facility at the Hanford Site near Richland, Washington
- Illustration of a satellite that could use radioisotope power systems
- The High Flux Isotope Reactor at the Oak Ridge National Laboratory near Oak Ridge, Tennessee
- The Advanced Test Reactor at the Idaho National Engineering and Environmental Laboratory near Idaho Falls, Idaho
- Tip of a remote-handling arm, used for work in developing industrial and medical isotopes

AVAILABILITY OF THE FINAL NI PEIS

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This PEIS is accessible on the Office of Nuclear Energy, Science and Technology web site at www.nuclear.gov.

Printed with soy ink on recycled paper

Department of Energy
Washington, DC 20585

November 28, 2000

Dear Interested Party:

The *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* [NI PEIS]) (DOE/EIS-0310) has now been completed. This document has been prepared in accordance with the National Environmental Policy Act and reflects consideration of comments received on the draft NI PEIS released in July 2000.

The Department of Energy (DOE) is responsible for undertaking research and development activities related to development of nuclear power for civilian use, meeting the nuclear material needs of other Federal agencies, and ensuring the availability of isotopes for medical, industrial, and research applications. The NI PEIS presents an evaluation of the potential environmental impacts associated with the proposed expansion of the nuclear irradiation capabilities for accomplishing civilian nuclear energy research and development activities, accommodating the projected growth in demand for medical and industrial isotopes, and production of plutonium-238 to support future National Aeronautics and Space Administration space exploration missions. In addition to the "No Action" alternative, DOE evaluated other alternatives that include using operating facilities within the DOE complex, building a new research reactor, building one or two accelerators, and restarting the Fast Flux Test Facility (FFTF) that is currently in standby status. In addition, the NI PEIS includes an alternative to permanently deactivate FFTF.

After careful consideration of public comments, environmental impacts, and programmatic objectives, DOE's preferred alternative is to use its existing nuclear facility infrastructure to the extent possible to pursue the missions outlined in the PEIS, i.e., Alternative 2, Option 7. DOE would reestablish domestic production of plutonium-238, as needed, using the Advanced Test Reactor in Idaho and the High Flux Isotope Reactor in Tennessee and would process irradiated plutonium-238 targets at the Radiochemical Engineering Development Center in Tennessee. DOE would permanently deactivate FFTF under the "Preferred Alternative." Lack of clear commitments from likely users discouraged the Department from planning to build new facilities or to restart the FFTF. Further details on the Preferred Alternative can be found in the summary and in section 2.8 of volume 1 of this NI PEIS.

We appreciate your continued participation in this decision-making process.

Sincerely,

for William D. Magwood, IV, Director
Office of Nuclear Energy, Science
and Technology

Cover Sheet

Responsible Agency: United States Department of Energy (DOE)

Title: *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (NI PEIS)

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Abstract: Under the authority of the Atomic Energy Act of 1954, as amended, the DOE is responsible for ensuring the availability of isotopes for medical, industrial and research applications, meeting the nuclear material needs of other Federal agencies, and undertaking research and development activities related to development of nuclear power for civilian use. To meet these responsibilities, DOE maintains nuclear infrastructure capabilities that support various missions. Current estimates for the future needs of medical and industrial isotopes, plutonium-238, and research requirements indicate that the current infrastructure may soon be insufficient to meet the projected demands. DOE proposes to enhance these capabilities to provide for: (1) production of isotopes for medical and industrial uses, (2) production of plutonium-238 for use in advanced radioisotope power systems for future National Aeronautics and Space Administration (NASA) space exploration missions, and (3) the Nation's nuclear research and development needs for civilian application.

This NI PEIS evaluates the environmental impacts of a No Action Alternative (maintaining status quo), four alternative strategies to accomplish this mission, and an alternative to permanently deactivate the Fast Flux Test Facility (FFTF), with no new missions. Alternatives 2, 3, and 4 also include permanent deactivation of FFTF. The alternatives are:

No Action
1. Restart FFTF at Hanford, Washington
2. Use only existing operational facilities
3. Construct one or two new accelerators
4. Construct a new research reactor
5. Permanently deactivate FFTF (with no new missions)

The Preferred Alternative is Alternative 2, Option 7, Use Only Existing Operational Facilities. DOE would reestablish domestic production of plutonium-238, as needed, using the Advanced Test Reactor in Idaho and the High Flux Isotope Reactor in Tennessee, and would process irradiated plutonium-238 targets at the Radiochemical Engineering Development Center in Tennessee. DOE would permanently deactivate FFTF under the Preferred Alternative.

Public Comments: The Draft NI PEIS was issued for public review and comment on July 21, 2000. The comment period ended on September 18, 2000, although late comments were considered to the extent practicable. Public hearings were held to obtain comments on the Draft NI PEIS in Oak Ridge, Tennessee; Idaho Falls, Idaho; Hood River and Portland, Oregon; Seattle and Richland, Washington; and Arlington, Virginia. All comments were considered by DOE in preparing the Final NI PEIS, which also incorporates any new information received since issuance of the Draft NI PEIS. In response to comments on the Draft NI PEIS and as a result of information that was unavailable at the time of the issuance of the Draft PEIS, the Final PEIS contains revisions and new information, indicated by a sidebar in the margin. Volume 3 contains the comments received during the public review period for the Draft NI PEIS and DOE's responses to these comments. DOE will use the analyses presented in the Final NI PEIS as well as other information, including public input, costs, nonproliferation impacts, schedules, technical assurance, and other policy and programmatic objectives, in preparing the Record of Decision for accomplishing expanded civilian nuclear energy research and development and isotope production missions in the United States, including the role of FFTF. DOE will issue the Record of Decision no sooner than 30 days after the U.S. Environmental Protection Agency publishes a notice of availability of the Final NI PEIS in the Federal Register.

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AAA	Advanced Accelerator Applications
AGS	Alternating Gradient Synchrotron
ATLAS	Argonne Tandem - LINAC Accelerator System
ATR	Advanced Test Reactor
ATW	Accelerator Transmutation of Waste
BLIP	Brookhaven LINAC Isotope Producer
BNL	Brookhaven National Laboratory
CANDU	Canadian Deuterium Uranium
CEQ	Council on Environmental Quality
CLWR	commercial light water reactor
CPP	Chemical Processing Plant
DOE	U.S. Department of Energy
EA	environmental assessment
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
FDPF	Fluorinel Dissolution Process Facility
FFTF	Fast Flux Test Facility
FMEF	Fuels and Materials Examination Facility
FONSI	finding of no significant impact
Hanford	Hanford Site
HFIR	High Flux Isotope Reactor
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
IPF	Isotope Production Facility
LANL	Los Alamos National Laboratory
LANSCE	Los Alamos Neutron Science Center
LINAC	linear accelerator
NASA	National Aeronautics and Space Administration
NE	DOE's Office of Nuclear Energy, Science and Technology
NEPA	National Environmental Policy Act
NEPO	Nuclear Energy Plant Optimization
NERAC	Nuclear Energy Research Advisory Committee
NERI	Nuclear Energy Research Initiative
NI PEIS	*Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility*
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
PCAST	President's Committee of Advisors on Science and Technology
REDC	Radiochemical Engineering Development Center
RPL	Radiochemical Processing Laboratory
SRPS	Stirling radioisotope power system
SRS	Savannah River Site
SRTG	Small Radioisotope Thermoelectric Generator
SST/SGT	safe, secure trailer/SafeGuards Transport
TRIGA	training, research, isotopes General Atomics (reactor)

Summary

S.1 Purpose and Need for Agency Action

Under the authority of the Atomic Energy Act of 1954, as amended, the U.S. Department of Energy (DOE) is responsible for ensuring the availability of isotopes for medical, industrial and research applications, meeting the nuclear material needs of other Federal agencies, and undertaking research and development activities related to development of nuclear power for civilian use.

To meet these responsibilities, DOE maintains nuclear infrastructure capabilities that support various missions in areas such as nuclear materials production and testing, research, and development activities related to civilian applications of nuclear power. These infrastructure capabilities include research and test facilities such as research reactors and accelerators used for steady-state neutron irradiation of materials to produce radionuclides, as well as shielded "hot cell" and glovebox facilities used to prepare materials for testing and/or to handle postirradiation materials. An additional component of this infrastructure is the highly trained workforce that specializes in performing complex tasks that have been learned and mastered over the life of these facilities.

Over the years, DOE's nuclear facility infrastructure has diminished because of the shutdown of facilities, recent examples being the High Flux Beam Reactor at Brookhaven National Laboratory (BNL), New York, and the Cyclotron Facility at Oak Ridge National Laboratory (ORNL), Tennessee. This, in turn, has hampered DOE's ability to satisfy increasing demands in various mission areas. To continue to maintain sufficient irradiation facilities to meet its obligations under the Atomic Energy Act, DOE has assessed the need for expansion of its existing nuclear infrastructure in light of its commitments to ongoing programs, its commitments to other agencies for nuclear materials support, and its role in supporting civilian nuclear energy research and development programs to maintain the viability of civilian nuclear power as one of the major energy sources available to the United States.

The Nuclear Energy Research Advisory Committee (NERAC) was established in 1998 by DOE in accordance with the Federal Advisory Committee Act to provide independent, expert advice on complex science and technical issues that arise in the planning, management, and implementation of DOE's civilian nuclear energy research programs. The chairman of NERAC has informed the Secretary of Energy that:

- "There is an urgent sense that the nation must rapidly restore an adequate investment in basic and applied research in nuclear energy if it is to sustain a viable United States capability in the 21st Century."

- "[T]he most important role for DOE [Office of Nuclear Energy, Science and Technology] in the nuclear energy area at the present time is to ensure that the education system and its facility infrastructure are in good shape."

- "Of particular need over the longer term are dependable sources of research isotopes and reactor facilities providing high volume flux irradiation for nuclear fuels and materials testing" (Duderstadt 2000).

Under the guidance of NERAC, DOE has completed an internal assessment of its existing nuclear facility infrastructure capabilities. This *Nuclear Science and Technology Infrastructure Roadmap* evaluates the existing DOE infrastructure, and identifies gaps in that infrastructure for meeting projected demands (DOE 2000a). The basic finding of this assessment also concluded that the capabilities of currently operating

DOE facilities will not meet projected U.S. needs for nuclear materials production and testing, research, and development.

Consistent with these findings, DOE recognizes that adequate nuclear research reactor, accelerator, and associated support facilities must be available to implement and maintain a successful nuclear energy program. As demand continues to increase for steady-state neutron sources needed for isotope production and civilian nuclear energy research and development, DOE's nuclear infrastructure capabilities to support this demand have not improved. To continue meeting its responsibilities under the Atomic Energy Act and to satisfy projected increases in the future demand for isotope products and irradiation services, DOE proposes to enhance its existing nuclear facility infrastructure to provide for: (1) production of isotopes for medical, research, and industrial uses, (2) production of plutonium-238 for use in advanced radioisotope power systems for future National Aeronautics and Space Administration (NASA) space exploration missions, and (3) support of the Nation's civilian nuclear energy research and development needs.

To evaluate the potential environmental impacts associated with this proposed enhancement, DOE has prepared the *Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]). The NI PEIS evaluates impacts from new facility construction, modification, startup, and 35 years of operation, followed by decommissioning when applicable. For analysis purposes, a 35-year operating period was established based on the projected availability of existing DOE irradiation facilities to potentially support these missions. This timeframe also accommodates current projections that indicate the demand for radioisotopes and civilian nuclear energy research and development will extend for at least the next 20 years (Wagner et al. 1998; NERAC 2000a; DOE 2000a).

Medical and Industrial Isotope Production

Over the past few decades, isotopes have become vital tools for use in medicine, industry, and scientific research. Isotopes, including both radioisotopes and stable isotopes, play a particularly important role in medical diagnosis, treatment, and research. Currently, more than 12 million nuclear medicine procedures are performed each year in the United States, and approximately one-third of all patients admitted to U.S. hospitals undergo at least one medical procedure that employs the use of medical isotopes (NERAC 2000a). Many medical isotopes are produced in the United States by DOE in nuclear reactors and particle accelerators. In limited cases, some medical isotopes can also be produced by extracting them from existing radioactive materials, such as thorium-229 obtained from DOE's existing stockpile of uranium-233. Radioisotopes are used for both diagnosis and therapy. Diagnostic radioisotopes are used for imaging internal organs. Unlike conventional radiology, imaging with radioisotopes reveals organ function and structure, which provides additional data for a more accurate diagnosis, and assists in the early detection of abnormalities. In ongoing clinical testing, therapeutic isotopes have proven effective in treating cancer and other illnesses by cell-directed localized radiation therapy (i.e., deploying antibodies or carriers of radioactive isotopes to seek and destroy invasive cancer cells). This directed therapy can minimize adverse side effects (e.g., healthy tissue damage, nausea, hair loss), making it an effective, attractive alternative to traditional chemotherapy or radiation treatments.

For nearly 50 years, DOE has actively promoted the use of radioisotopes to improve the health and well-being of U.S. citizens. DOE's use of its unique technologies and capabilities to develop isotopes for civilian purposes has enabled the widespread application of medical and industrial isotopes seen today. DOE must provide an adequate supply of isotopes to keep pace with the growing and changing needs of the research community if it is to continue to serve this key role.

An Expert Panel convened by DOE in 1998 reviewed several industry projections for growth in demand for medical isotopes. The Expert Panel concluded that the growth rate in medical isotope use will be significant over the next 20 years (Wagner et al. 1998). Specifically, the Expert Panel estimated that the expected growth rate of medical isotope use during the next 20 years will range from 7 to 14 percent per year for therapeutic applications, and from 7 to 16 percent per year for diagnostic applications. The panel noted that these growth rates are attainable only if basic research in nuclear medicine is supported and modern, reliable isotope production facilities are available. In the period since the initial estimates were made, the actual growth of medical isotope use has tracked at levels consistent with the Expert Panel findings. DOE and NERAC have agreed with the following findings and recommendations provided by the Expert Panel.

- Several isotopes have proven their clinical efficacy, but supply and cost concerns could dramatically affect the use of these isotopes in the practice of nuclear medicine.

- Although commercial and research applications for certain isotopes have been developed or are being developed, their limited availability and high prices are inhibiting their use in clinical applications.

- Research isotopes that have shown promise as diagnostic and therapeutic materials are not being explored because of their lack of availability or high price.

- At present, there is no domestic production facility to guarantee the continued supply of many of these isotopes.

- To meet current and future needs of the biomedical sciences community, the Expert Panel recommended:

  > ". . . the United States develop a capability to produce large quantities of radionuclides [radioisotopes] to maintain existing technologies and to stimulate future growth in the biomedical sciences. The successful implementation of such a program would help insure our position as an international leader in the biomedical sciences well into the twenty-first century. The panel recommends that the U.S. Government build this capability around a reactor, an accelerator, or a combination of both technologies as long as isotopes for clinical and research applications can be supplied reliably, with diversity in adequate quantity and quality" (Wagner et al. 1998).

In its recent report from the Subcommittee for Isotope Research and Production Planning, NERAC further identified that:

> "It is now widely conceded that limited availability of specific radionuclides is a constraint on the progress of research. The problem is especially apparent in a number of medical research programs that have been terminated, deferred, or seriously delayed by a lack of isotope availability . . . The lack of radionuclides significantly inhibits progress in evaluating a host of promising diagnostic and therapeutic drugs in patients with debilitating and fatal diseases, examining fundamental basic science questions, studying human behavior and normal growth and development, and exploring the aging process and the products of transgene expression . . . the DOE long-term goal to have a reliable isotope supply system in place that would enable scientists to bring their creative ideas into practical use safely, quickly and efficiently is appropriate, be it basic science research, clinical medicine, or industrial endeavors. The discovery and dissemination of new knowledge should continue to be a core mission, and basic science and the application of basic science to clinical research discoveries

> to improve the diagnosis and treatment outcomes should be a crucial component of that mission. [DOE], in providing a federal system for the reliable supply of stable and radioactive isotopes for research, will be an important aspect of fulfilling the federal responsibility to support biomedical research" (NERAC 2000a).

Current domestic and global producers of radioisotopes include governments that operate reactors and accelerators at national laboratories or institutes, and private sector companies that own and operate accelerators. There are also many partnership arrangements where companies lease irradiation space in government reactors or operate processing facilities in coordination with the government. A few universities also produce radioisotopes, but their ability to provide reliable and diverse supplies is generally limited by the small-scale capabilities or operating schedules of their facilities.

DOE's production and sale of radioisotopes fall into two categories: "commercial" and "research". Commercial radioisotopes are those that are produced in large, bulk quantities and sold to pharmaceutical companies or distributors, or to equipment or sealed source manufacturers. DOE only produces commercial isotopes when there is no U.S. private sector capability or when foreign sources do not have the capacity to meet U.S. needs reliably.

In contrast, research radioisotopes are typically produced and sold in small quantities in response to specialty orders from researchers preparing experiments in the field of medicine, with small quantities of these radioisotopes also purchased by industrial researchers. Because small-quantity production of research isotopes is not financially attractive to private-sector producers, it is generally not undertaken. DOE attempts to provide all research radioisotopes that are requested, subject to production capability, inventory, and financial constraints. As successful application of a specific research isotope is established, the production and sales of that radioisotope may shift from research to commercial status. In recent years, over 95 percent of DOE's sales of radioisotopes by dollar volume were commercial, and 5 percent were for research.

DOE produces radioisotopes using the High Flux Isotope Reactor (HFIR) at ORNL, the Advanced Test Reactor (ATR) at Idaho National Engineering and Environmental Laboratory (INEEL), and the Annular Core Research Reactor at Sandia National Laboratories. DOE also produces radioisotopes using accelerators, namely the Isotope Production Facility (IPF) at Los Alamos National Laboratory (LANL) and the Brookhaven LINAC (Linear Accelerator) Isotope Producer (BLIP) at BNL. At each of these DOE sites, the radioisotope production mission shares the reactor or accelerator with other basic energy sciences or defense missions that are generally much larger and exercise considerable influence on facility schedules and priorities. As such, radioisotope production is often relegated to fulfilling a secondary mission that is dependant on the operating constraints of these larger, primary missions. Currently, approximately 50 percent of DOE's isotope production capability is being utilized. Assuming a midpoint growth curve for future isotope demand and ensuring a diversity and redundancy of isotope supply, DOE estimates that its isotope production facilities would be fully used within a 5- to 10-year timeframe if no enhancements to the existing nuclear facility infrastructure are implemented. This projection is made in the context of a worldwide market for radioisotopes. Although DOE's market share is a small fraction of the overall total, it is very significant for some radioisotopes and particularly important for a large number of radioisotopes that are used in relatively small quantities for research. These isotopes, which are used almost exclusively by researchers at universities and hospitals, are not purchased in quantities that would attract private industry to take over their production. However, DOE may need to significantly increase the production levels of these radioisotopes as world demand changes and promising research developments in their medical use are brought to commercialization.

Recent analyses indicate that the greatest challenge to meeting projected isotope market requirements over the next 20 years will be in the area of therapeutic medical isotopes, several of which are currently unavailable or are available only in limited quantities (Battelle 1999). For the purpose of analysis in the NI PEIS, a

representative set of isotopes was selected on the basis of the recommendations of the Expert Panel, medical market forecasts (Frost & Sullivan 1997), reviews of medical literature, and more than 100 types of ongoing clinical trials that use radioisotopes for the treatment of cancer and other diseases. Currently, these medical applications primarily involve the diagnosis and treatment of three major classes of disease—cancer, vascular disease, and arthritis. Although these isotopes are a representative sample of possible isotopes that could be produced, DOE expects that the actual isotopes produced as a result of the proposed action would vary from year to year in response to the focus of clinical research and the specific market needs occurring at that time.

The United States currently purchases approximately 90 percent of its medical isotopes from foreign producers, most notably Canada. However, Canada only supplies a limited number of economically attractive commercial isotopes (primarily molybdenum-99), and it does not supply research isotopes or the diverse array of medical and industrial isotopes considered in the NI PEIS. As such, reliance on Canadian sources of isotopes to satisfy projected U.S. isotope needs would not meet DOE's mission requirements.

Industrial isotope applications fall into three broad categories: nucleonic instrumentation, irradiation and radiation processing, and technologies that use radioactive tracers. Examples of nucleonic instrumentation include gauges for measuring physical parameters, e.g., detection systems for pollutants, explosives, drugs, ores, petroleum, and natural gases; nondestructive testing by gamma radiography; and smoke detectors. Irradiation and radiation processing technologies include radiation sterilization of food and medical products and the curing of plastics. Radioactive tracer applications include studies of chemical synthesis reactions; mass transfer monitoring in industrial plants; analysis of the transport and uptake of nutrients, fertilizers, herbicides, and waste materials in plants, soils, and groundwater; and laboratory-based studies of the properties of materials.

In proposing to expand its radioisotope production capability, DOE intends to continue to complement the commercial availability of these radioisotopes. Consistent with current isotope production activities, DOE will continue to make its facilities available to the private sector to support production and sales of isotopes.

Plutonium-238 Production for Space Missions

As part of its charter under the Atomic Energy Act, DOE and its predecessor agencies have been developing and supplying radioisotope power systems to NASA for space exploration for more than 30 years. These radioisotope power systems include radioisotope thermoelectric generators used to power electrical components and radioisotope heater units used to keep spacecraft instruments warm. Previous NASA space missions that have used radioisotope power systems include the Apollo lunar scientific packages and the Pioneer, Viking, Voyager, Galileo, and Ulysses deep space probes. More recent missions include the Mars Pathfinder mission launched in 1996 and the Cassini mission launched in 1997. These radioisotope power systems have repeatedly demonstrated their performance, safety, and reliability in various NASA space missions. Without these power systems, these types of space exploration missions could not have been performed by NASA.

The radioisotope used in these power systems is plutonium-238. Through a Memorandum of Understanding with NASA, DOE provides these radioisotope power systems, and the plutonium-238 that fuels them, for space missions that require or would be enhanced by their use (DOE and NASA 1991). In addition, under the National Space Policy issued by the Office of Science and Technology Policy in September 1996, and consistent with DOE's charter under the Atomic Energy Act, DOE is responsible for maintaining the capability to provide the plutonium-238 needed to support these missions. The Intersector Guidelines section of the National Space Policy states that, "The Department of Energy will maintain the necessary capability to support space missions which may require the use of space nuclear power systems" (The White House 1996). Although research to identify other potential fuel sources to support these space exploration missions has been conducted, no viable alternative to using plutonium-238 has been established.

Historically, the reactors and chemical processing facilities at DOE's Savannah River Site (SRS) were used to produce plutonium-238; however, downsizing of the DOE nuclear weapons complex resulted in the shutdown of the last remaining SRS operating reactor, K-Reactor, in early 1996. Also, in 1992 then-Secretary of Energy Watkins issued a decision to phase out operations at the two chemical processing facilities (F-Canyon and H-Canyon) at SRS. In accordance with that decision, the separation facilities are planned to be shut down following completion of their current missions to stabilize and prepare for the disposition of Cold War legacy nuclear materials and certain spent nuclear fuel, and a determination that a new nonchemical processing technology is capable of preparing aluminum-based research reactor spent nuclear fuel for ultimate disposition.

In order to obtain a source of plutonium-238 to support NASA space missions, DOE signed a 5-year contract in 1992 to purchase plutonium-238 from Russia, authorizing the United States to purchase up to 40 kilograms (88.2 pounds) of plutonium-238, with the total available for purchase in any one year limited to 10 kilograms (22 pounds).[1] Under this contract, DOE purchased approximately 9 kilograms (19.8 pounds) of plutonium-238[2]. This material constitutes the only available U.S. inventory that has been reserved for space missions, an amount that is expected to be depleted by approximately 2005. DOE's practice of purchasing on an as-needed basis has avoided the costs from processing the plutonium-238 to remove the decay products that would result from storing it for an extended period of time. In 1997, DOE extended the contract for another 5 years; therefore, it is set to expire in 2002. Any purchases beyond 2002 would likely require the negotiation of a new contract and may require additional NEPA review. The long-term viability of pursuing additional contract extensions or entering into a new contract is unclear.

The political and economic climate in Russia creates uncertainties that could affect its reliability as a source of plutonium-238 to satisfy future NASA space mission requirements. Reestablishing a domestic plutonium-238 production capability would ensure that the United States has a long-term, reliable supply of this material. In doing so, the United States would have greater control over the available supply, plans for satisfying future demand, and the nuclear safety and nonproliferation implications of the material. As such, DOE's preference is to reestablish a domestic plutonium-238 production capability rather than to rely on Russia as the sole long-term supplier. A plutonium-238 production rate of 2 to 5 kilograms (4.4 to 11 pounds) per year is expected to be sufficient to meet NASA's estimated long-term requirements.

DOE is planning to provide radioisotope heater units for several NASA Mars Exploration missions over the next 10 years. Each heater unit would require approximately 2 grams (0.07 ounce) of plutonium-238. The number of heater units varies depending on the spacecraft. Each of the two Mars missions in 2003 is projected to require up to 11 heater units. In May 2000, NASA provided preliminary guidance to DOE to also plan for the potential use of radioisotope power systems for the Pluto/Kuiper Express mission scheduled for launch in 2004, the Europa Orbiter mission scheduled for launch in 2006, and the Solar Probe mission scheduled for launch in 2007 (NASA 2000a). The amount of plutonium-238 needed for these missions was approximately 7.4 kilograms (16.3 pounds) for the Pluto/Kuiper Express mission, which would use an existing spare radioisotope thermoelectric generator, and approximately 3 kilograms (6.6 pounds) each for the Europa Orbiter and Solar Probe missions, which would use the Stirling radioisotope power system (SRPS). With NASA's current emphasis on smaller and less expensive spacecraft, the SRPS is being developed as a new, more efficient and lighter weight power system requiring one-third less plutonium-238 as its fuel source. However,

[1] The NI PEIS presents the weight of plutonium-238 in terms of kilograms of isotope. In contrast, NASA documentation expresses this weight in terms of plutonium oxide. The equivalent plutonium oxide weight can be approximated by multiplying the isotope kilogram weight by 1.134.

[2] The environmental impacts of purchasing plutonium-238 from Russia are evaluated and documented in the *Environmental Assessment of the Import of Russian Plutonium-238* (DOE 1993), prepared by DOE's Office of Nuclear Energy.

the technology is developmental, and NASA has requested that the plutonium-238 needed for a large radioisotope thermoelectric generator be maintained as backup.

A plutonium-238 production goal of 2 to 5 kilograms (4.4 to 11 pounds) per year could produce sufficient quantities of plutonium-238 to theoretically yield a SRPS every 8 months if production were maintained at the high end of the range. However, DOE chose the 5-kilogram (11-pound) per year production rate as an upper bound due to uncertainties in the SRPS technology development requirements for backup units, and variability in the amount of plutonium-238 that may be needed for each of the units to meet NASA's power requirements.

In updated mission planning guidance provided in September 2000, NASA indicated that for programmatic and technical reasons, implementation of the Pluto/Kuiper Express mission as currently conceived was being deferred, and that the SRPS generators were candidate power systems for the Europa Orbiter and Solar Probe missions (NASA 2000b, 2000c). NASA also requested that the spare radioisotope thermoelectric generator and assembling and fueling a spare thermoelectric converter be maintained as backups for the Europa Orbiter mission in the event the SRPS technology was not ready in time. If NASA chooses to use the SRPS to support the Europa Orbiter and Solar Probe missions, there would be no change in NASA's requirements regarding the plutonium-238 needed for these two missions (i.e., approximately 3 kilograms [6.6 pounds] each, as described above), although the remaining quantity of plutonium-238 would not be sufficient to support additional deep space or long-lived exploration missions. Should NASA decide to use the backup radioisotope thermoelectric generators rather than the SRPS to support the Europa Orbiter mission, approximately 8 kilograms (17.6 pounds) of plutonium-238 would be needed, which would effectively expend all of DOE's available plutonium-238 inventory prior to supporting the Solar Probe mission. While this latest NASA guidance modifies the specific radioisotope power systems and missions for which DOE needs to plan, it does not fundamentally change NASA's overall potential plutonium-238 requirements, or the expectation that the available U.S. inventory of this material would effectively be depleted by approximately 2005.[3]

Although future space mission schedules over a long-term planning horizon of 10 to 35 years cannot be specified at this time, DOE anticipates that NASA space exploration missions conducted during this period will continue to require plutonium-238-fueled power systems. For example, NASA announced in a recent press conference (October 26, 2000) that mission launches in 2014 and 2016 for the long-term exploration of Mars would involve long-life rover vehicles. Radioisotope power systems would be required to provide the long-life capability.

Therefore, DOE proposes to reestablish a domestic capability for producing and processing this material. Because the SRS facilities previously used for plutonium-238 production are no longer available, DOE needs to evaluate other DOE irradiation and chemical processing facilities, as well as potential commercial light water reactors (CLWR), for this mission. Unless an assured domestic supply of plutonium-238 is established, DOE's ability to provide radioisotope power systems to support future NASA space exploration missions may be lost.

Civilian Nuclear Energy Research and Development

Nuclear energy is an important contributor in reducing greenhouse gas emissions in the United States, Asia, and Europe. Globally, nuclear energy produces 17 percent of the world's electricity. In the United States, nuclear energy generated 20 percent of all electricity consumed in 1999. In view of energy and environmental contributions, there is a renewed interest in nuclear power to meet an equivalent portion of the Nation's future expanding energy requirements.

[3] Applicable NASA mission planning correspondence is presented in Appendix R, Volume 2 of the NI PEIS.

In January 1997, President Clinton tasked his Committee of Advisors on Science and Technology (PCAST) to evaluate the current national energy research and development portfolio and to provide a strategy that ensures the United States has a program to address the Nation's energy and environmental needs for the next century. In its November 1997 report responding to this request, the PCAST Energy Research and Development Panel determined that restoring a viable nuclear energy option to help meet our future energy needs is important and that a properly focused research and development effort to address the potential long-term barriers to expanded use of nuclear power (e.g., nuclear waste, proliferation, safety, and economics) was appropriate. The PCAST panel further recommended that DOE reinvigorate its nuclear energy research and development activities to address these potential barriers.

Clean, safe, reliable nuclear power has a role today and in the future for our national energy security. Recognizing this need, two significant new nuclear energy research and development programs have been initiated: the Nuclear Energy Research Initiative (NERI) and Nuclear Energy Plant Optimization (NEPO). The NERI program, initiated in fiscal year 1999, sponsors new and innovative scientific and engineering research and development to address the potential long-term barriers identified by the PCAST Panel affecting the future use of nuclear energy. The NEPO program, a cost-shared program with industry, initiated in fiscal year 2000, sponsors applied research and development to ensure that current nuclear plants can continue to deliver adequate and affordable energy supplies up to and beyond their initial 40-year license period by resolving open issues related to plant aging, and by applying new technologies to improve plant reliability, availability, and productivity.

The NERAC Subcommittee on Long-Term Planning for Nuclear Energy Research has set forth a recommended 20-year research and development plan to guide DOE's nuclear energy programs in areas of materials research, nuclear fuel, and reactor technology development (NERAC 2000b). This plan stresses the need for DOE facilities to sustain the nuclear energy research mission in the years ahead. Such civilian nuclear energy research and development initiatives requiring an enhanced DOE nuclear facility infrastructure fall into three basic categories: materials research, nuclear fuel research, and advanced reactor development.

Materials Research. The high radiation fields, high temperatures, and corrosive environments in nuclear reactors (terrestrial or space) and other complex nuclear systems (e.g., accelerator transmutation of waste [ATW] systems) can accelerate the degradation of pressure vessels and structural material, component materials, material interfaces and joints between materials (e.g., welds). Radiation effects in materials can cause a loss of mechanical integrity (fracture toughness and ductility) by embrittlement, dimensional changes (creep and swelling), and fatigue and cracking (irradiation-assisted stress corrosion cracking). Acquiring a fundamental understanding of radiation effects in current and future reactor materials (engineered steel alloys, ceramics, composites, and refractory metals), as well as the experimental validation of analytical models and computational methods, would require material irradiation testing over a range of neutron energies (thermal and fast flux) and doses. Material testing under simulated reactor conditions would be required to ensure the compatibility of advanced materials with the various moderators/coolants of future reactor concepts. In addition, the thermophysical properties and behaviors of liquid metal coolants being considered for advanced reactor (terrestrial or space) and ATW systems require further irradiation testing. One key area of materials research that is important to plant safety and the license renewal of existing nuclear power plants is the accelerated aging of materials to simulate radiation effects over a plant lifetime. Researchers from the United States and many foreign countries use DOE's high flux research reactors for materials testing and experimentation. These facilities have the capability to maintain a high density of neutrons in a given test volume for materials testing; shorten the time needed for such testing; tailor the neutron flux to simulate the different reactor types and conditions; and instrument the core for close monitoring of the test conditions.

Nuclear Fuel Research. Increasing demands are being placed on nuclear fuel and cladding material performance as the fuel burnup limits are extended in existing light water reactors to maximize plant

performance and economic benefits. New fuel types and forms are being investigated that offer potential benefits such as enhanced proliferation resistance (uranium-thorium fuel), higher burnup, and improved waste forms for the new reactor concepts being researched and developed by DOE. In addition, plutonium-uranium mixed oxide fuels are being developed for the disposition of surplus weapons material, and high temperature, long-life fuels may be required for space reactors. Each of the various fuel and cladding types, forms, and material compositions would require research and irradiation testing under prototypical reactor conditions to fully understand fuel performance, cladding performance, cladding/fuel interaction, and cladding/coolant material compatibility. Fuel research includes a variety of thermal and fast spectrum power reactor fuel forms (ceramic, metal, hybrids such as cermet) and various fuel types (oxides, nitrides, carbides, and metallics). Irradiation experiments to characterize fuel performance would require the capability to test fuel pellets, pins, and fuel assemblies under steady-state and transient conditions in the higher temperature environments expected in future reactor designs. Reactor physics and criticality safety data for benchmarking computational codes and analytical methods used in fuel design and performance analysis would also be required.

Advanced Reactor Development. Certification and licensing of advanced reactor and complex nuclear systems will require the demonstration and validation of reactor and safety system thermal and fluid dynamic properties under steady-state and transient conditions. Typically, nonnuclear test loops are used to perform this research. However, because of the unique nature of some proposed advanced reactor concepts, test loop operation under prototypical temperature and neutron flux conditions would be necessary to adequately test and demonstrate coolant/moderator physics and thermal properties, heat transfer, fluid flow, and fuel-moderator performance.

S.2 SCOPE OF THE NI PEIS

Public Scoping Process

On October 5, 1998, DOE published in the Federal Register (63 FR 53398) a Notice of Intent to prepare an environmental impact statement (EIS) on the proposed production of plutonium-238 for use in advanced radioisotope power systems for future space missions. With that announcement, DOE began preparing the *Environmental Impact Statement for the Proposed Production of Plutonium-238 for Use in Advanced Radioisotope Power Systems for Future Space Missions* (*Plutonium-238 Production EIS*). The scope of the *Plutonium-238 Production EIS* was established through a public scoping process conducted from November 4, 1998 through January 4, 1999. As part of the scoping process for that draft, DOE announced that the Fast Flux Test Facility (FFTF) would not be considered a reasonable alternative for the plutonium-238 production mission unless restart of the facility was proposed for other reasons.

Since then, the Secretary of Energy announced on August 18, 1999, that DOE would prepare the NI PEIS. Because plutonium-238 production would be among the missions considered in the NI PEIS, the scope of the *Plutonium-238 Production EIS* in its entirety was incorporated within the scope of the NI PEIS, and preparation of the *Plutonium-238 Production EIS* as a separate National Environmental Policy Act (NEPA) review was terminated.

On September 15, 1999, DOE published in the Federal Register a Notice of Intent to prepare the NI PEIS (64 FR 50064). In this Notice of Intent, DOE invited the public to comment on the proposed actions during the 45-day NI PEIS scoping period that ended October 31, 1999. During this period, DOE held public scoping meetings at seven locations: Oak Ridge, Tennessee; Idaho Falls, Idaho; Richland and Seattle, Washington; Hood River and Portland, Oregon; and Washington, D.C. The written and oral comments received at these meetings and the additional comments received via U.S. mail, electronic mail, and toll-free faxes and telephone calls during the public scoping period were reviewed and considered by DOE in preparing

the NI PEIS. Similarly, DOE reviewed and considered all comments and input originally received from the public during the *Plutonium-238 Production EIS* scoping period in the preparation of the NI PEIS.

For the *Plutonium-238 Production EIS*, approximately 750 scoping comments were received by DOE. At the scoping meetings on the *Plutonium-238 Production EIS*, the following general issues and concerns were raised:

- Additional irradiation service alternatives, such as CLWRs and accelerators
- Additional storage, target fabrication, and target processing alternatives, such as Argonne National Laboratory's Hot Fuels Examination Facility and the SRS H-Canyon and HB-Line
- Generation of additional waste
- Costs of implementing the various alternatives

In general, the people who attended the meetings in Idaho and Tennessee were supportive of DOE's proposed plans to produce plutonium-238 domestically for future space missions. However, in Richland, Washington, the meeting was attended by several stakeholder and environmental groups who voiced considerable opposition to DOE's consideration of FFTF for plutonium-238 production.

At the meeting in Richland, Washington, the main concern was that DOE should not consider restarting FFTF, that DOE has worked hard over the years to change the Hanford Site's (Hanford) mission from "production" to "cleanup," and that DOE should continue to honor its commitment to cleanup. There were concerns about the generation of additional waste at the site and the operational safety of FFTF. There was strong opposition to restart of FFTF for any mission.

For the NI PEIS, approximately 7,000 scoping comments were received by DOE. At the scoping meetings on the NI PEIS, the most prevalent concerns were:

- Status of and commitment to cleanup at Hanford and the impact of FFTF restart on the existing waste cleanup at Hanford
- Lack of justification for the identified missions
- Costs of implementing the various alternatives
- Need for an additional alternative calling for the permanent deactivation of FFTF coupled with the No Action alternative elements, that is, no plutonium-238 production and no additional research and development or medical isotope production beyond existing operating levels

The number of people who commented at the scoping meetings conducted in Oak Ridge, Tennessee; Idaho Falls, Idaho; and Washington, D.C., was smaller in comparison to the meetings held in the Pacific Northwest. At the scoping meeting in Oak Ridge, Tennessee, a commentor was concerned with the relationship of the NI PEIS to other DOE programs and the relative merits of accelerator and reactor performance. The commentor stated that the NI PEIS should include an explanation of mixed oxide fuel disposition. In addition, the commentor supported medical isotope production in Oak Ridge because it is near a transportation hub and some medical isotopes are short-lived; therefore, transportation is key.

At the scoping meeting in Idaho Falls, Idaho, most commentors supported siting the new missions at INEEL. The commentors also stated that the socioeconomic impacts of the alternatives need to be considered in the NI PEIS. A commentor stated that decisions in regard to medical isotope production should be based on the needs of the Nation as a whole and not on perceived commercial needs. The commentor also stated that incremental DOE and commercial investments in ATR would be sufficient to enhance reactor radioisotope production needs and meet the requirements of the nuclear medicine industry.

At the scoping meetings held in the states of Washington and Oregon, many of the comments concerned using FFTF to accomplish the proposed action. Many who attended the meetings in Seattle, Washington; Portland, Oregon; and Hood River, Oregon, were strongly opposed to restart of FFTF. Many commentors stated that the Hanford cleanup mission would be jeopardized, especially when DOE has not met the Hanford cleanup milestones. Many of the comments received at the Richland, Washington, meeting supported restarting FFTF, stated that restart would not hamper Hanford's cleanup mission, and further stated that operation of FFTF could help save the lives of many people by producing isotopes to be used in new ways to treat cancer, heart disease, and other illnesses. Commentors were also concerned about the potential generation of radioactive and hazardous wastes as a result of the proposed action, as well as DOE's commitment to ongoing cleanup programs, particularly at Hanford.

At the scoping meeting in Washington, D.C., the commentors supported the need for medical isotope production. Several commentors were against the restart of FFTF and others stated that DOE needs to consider partnerships with private industry to generate necessary funds for restart. Some commentors thought a cost study should be prepared and include avoided future health care costs and cost savings to the national Medicare and Medicaid programs that could be realized by using nuclear isotopes in medical applications. Proliferation concerns were also raised as some commentors stated that: (1) the United States would be sending the wrong message by restarting FFTF; (2) a change in the U.S. nonproliferation policy will be required to import German mixed oxide fuel; and (3) the use of highly enriched uranium is contrary to existing U.S. nonproliferation policy. Other concerns included waste generation, Hanford cleanup, and safety at FFTF.

Comments received during the scoping periods were systematically reviewed by DOE. As a means of summarizing the issues raised during scoping, those comments with similar or related topics were grouped into categories to identify specific issues of public concern. After these issues were identified, they were further evaluated to determine whether they fell within or outside the proposed scope of the NI PEIS. In several instances, the original scope was expanded to accommodate additional issues resulting from the public scoping process.

Comments received that contributed to expansion of the scope concerned the following general areas:

- Deactivate FFTF: Alternative 5, Permanently Deactivate FFTF with no new missions at existing facilities, has been added to the scope of the NI PEIS.

- Cleanup at Hanford: Although not within the scope of the NI PEIS, information is included about the cleanup mission at Hanford and land-use planning efforts.

- Environmental contamination at Hanford: Information is included about the groundwater quality at the Hanford Site.

- Nonproliferation issues: The import of German SNR–300 fuel is addressed, and a separate *Nuclear Infrastructure Nonproliferation Impact Assessment for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including*

the Role of the Fast Flux Test Facility (*NI Nonproliferation Impact Assessment*) report was prepared and distributed to the public in September 2000.

- Transition of FFTF stewardship after it is deactivated: The appropriate transition information is included.

- Restart of FFTF and budget constraints: DOE has made a commitment that implementation of the Record of Decision will not divert or reprogram budgeted funds designated for Hanford cleanup.

- Tri-Party Agreement at Hanford: Information about the Tri-Party Agreement and its relationship to the NI PEIS is included.

The public comments and materials submitted during the public scoping periods for both the *Plutonium-238 Production EIS* and the NI PEIS were logged and placed in the Administrative Record for the NI PEIS. Appendix N of the NI PEIS summarizes the comments received during both public scoping periods.

Issues Raised During the Public Comment Period on the Draft NI PEIS

DOE published the Draft NI PEIS in July 2000. In accordance with Council on Environmental Quality (CEQ) and DOE NEPA regulations, DOE announced the availability of the Draft NI PEIS in the Federal Register (65 FR 46443) and invited interested parties to provide comments on the Draft NI PEIS analysis and results. The Draft NI PEIS or Summary was distributed to approximately 6,000 individuals.

NEPA regulations mandate a minimum 45-day comment period after the U.S. Environmental Protection Agency's (EPA) Notice of Availability of a draft EIS to provide an opportunity for the public to comment on the EIS analysis and results. The original 45-day comment period on the Draft NI PEIS began on July 28, 2000. To provide interested parties with additional time to comment, the deadline for transmittal of comments was changed from September 11, 2000 (as stated in the transmittal letter of the Draft PEIS and the Summary) to September 18, 2000. During the 52-day comment period, DOE held seven hearings to discuss the proposed action and to receive oral and written comments on the Draft NI PEIS. These hearings were held at Oak Ridge, Tennessee; Idaho Falls, Idaho; Hood River, Oregon; Portland, Oregon; Seattle, Washington; Richland, Washington; and Arlington, Virginia. In addition, the public was encouraged to submit comments via U.S. mail, e-mail, a toll-free phone line, and a toll-free fax line. During the public comment period, DOE received approximately 3,400 submittals containing over 6,200 comments. DOE has responded to all comments received during the public comment period. These comments are presented in Volume 3 of the Final NI PEIS. DOE considered comments received after the close of the public comment period to the extent practicable.

The public comments received on the Draft NI PEIS addressed a wide range of issues. The following discusses the major issues raised, and DOE's responses to these issues. Changes made in response to comments received on the Draft NI PEIS are described in the next section.

Major issues raised addressed purpose and need for the proposed action; impact of FFTF on Hanford cleanup; waste management and spent nuclear fuel; cost of the various alternatives; nuclear nonproliferation policy; public involvement; and environmental impacts. Aside from comments on the proposed action and its environmental impacts, many commentors expressed support for or opposition to FFTF restart, the major point of public controversy associated with the NI PEIS.

Purpose and Need for the Proposed Action. Many commentors expressed the opinion that DOE failed to demonstrate a compelling argument for the projected need for medical isotopes, and that such medical isotopes

could be produced or purchased elsewhere, particularly in Canada. In contrast, a large number of commentors expressed support for expanded isotope production by sharing personal stories of how medical isotopes had either saved a relative or friend, or could have saved them had isotopes been available. As presented in Section 1.2.1 of Volume 1 of the NI PEIS, DOE sought independent analysis of trends in the use of medical isotopes, and established two advisory bodies, the Expert Panel and the NERAC. DOE has adopted these growth projections as a planning tool for evaluating the potential capability of the existing nuclear facility infrastructure to meet programmatic requirements. In the period since the initial estimates were made, the actual growth of medical isotope use has tracked at levels consistent with the Expert Panel findings. While Canada currently provides a large amount of the medical radioisotopes used in the United States, it only supplies a limited number of economically attractive commercial isotopes (primarily molybdenum-99), and it does not supply research isotopes or the diverse array of medical and industrial isotopes considered in the NI PEIS.

A number of commentors also questioned the suitability of using FFTF for producing research isotopes in light of findings presented in the NERAC Subcommittee for Isotope Research and Production Planning Report (NERAC 2000a). While it would not be cost effective to restart FFTF for the singular purpose of producing small quantities of various research isotopes, sustained operation of FFTF for the production of larger quantities of both research and commercial isotopes would be viable if FFTF were operated in concert with producing plutonium-238 and conducting nuclear energy research and development for civilian applications. In recognition of these constraints on its operational feasibility, the NI PEIS only evaluates the use of FFTF for isotope production when coupled with these other missions.

Commentors also questioned the need for the United States to reestablish domestic production of plutonium-238. In particular, commentors pointed to the availability of plutonium-238 that could be purchased from Russia, and recent guidance from NASA stating that DOE no longer needed to support certain radioisotope power systems. As discussed in Section 1.2.2 of Volume 1, DOE could purchase plutonium-238 from Russia. However, for supply reliability reasons and concern of nuclear nonproliferation, DOE's preference is to establish a domestic plutonium-238 production capability. Current NASA guidance to DOE is also discussed in Section 1.2.2. The May 22, 2000, correspondence from NASA identifies that it no longer has a planned requirement for Small Radioisotope Thermoelectric Generator (SRTG) power systems (NASA 2000a). This does not mean that NASA no longer requires DOE to provide the necessary plutonium-238 to support deep space missions. Rather, SRTG development efforts were stopped in order to permit reprogramming of funds to support development of a new radioisotope power system based on an SRPS technology. This new radioisotope power system, referred to in the subject correspondence, requires one-third less plutonium as its fuel source. Because the SRPS technology is developmental, NASA has requested in a September 22, 2000, letter to DOE that the plutonium-238 needed for a large radioisotope thermoelectric generator be maintained as a backup (NASA 2000b).

Impact of FFTF Restart on Hanford Cleanup. A number of commentors expressed concern that DOE's primary mission at Hanford needs to be cleanup, including compliance with the Tri-Party Agreement. Although beyond the scope of the NI PEIS, ongoing Hanford cleanup activities are high priority to DOE. Hanford environmental restoration activities are conducted in accordance with the Tri-Party (i.e., DOE's Richland Operations Office, EPA, and the State of Washington Department of Ecology) Agreement. This agreement specifies milestones and schedules for restoration of all parts of Hanford. FFTF milestones in the Tri-Party Agreement were placed in abeyance (suspension) by agreement of the three parties until a decision is made on the future of FFTF. Public meetings were held on this formal milestone change. DOE is fully committed to honoring this agreement.

A number of commentors also expressed concern that funding for Hanford cleanup would be diverted for FFTF restart and hamper the progress of cleanup activities. The U.S. Congress funds Hanford cleanup through

the Office of the Assistant Secretary for Environmental Management. Congress also funds FFTF through the Office of Nuclear Energy, Science and Technology (NE). The nuclear infrastructure missions described in Section 1.2 of Volume 1 would also be funded through NE, which has no funding connection to Hanford cleanup activities. As stated in Section N.3.2 of Volume 2, implementation of the nuclear infrastructure alternatives would not divert or reprogram budgeted funds designated for Hanford cleanup, regardless of the alternative(s) selected.

Waste Management and Spent Nuclear Fuel. A number of commentors expressed concern over the generation and disposition of waste resulting from the proposed action. In particular, commentors pointed to past DOE waste management practices and questioned whether wastes resulting from proposed NI PEIS activities would be properly managed. The NI PEIS addresses wastes produced for each alternative, as well as cumulative impacts related to waste production. Waste minimization programs at each of the alternative sites are also addressed. These programs would be implemented for the alternative selected in the Record of Decision. The waste generated from any of the alternatives considered in the NI PEIS would be managed (i.e., treated, stored, and disposed of) in a safe and environmentally protective manner and in compliance with all applicable Federal and state laws and regulations and applicable DOE orders.

A number of commentors expressed specific concern over the generation and disposition of waste resulting from FFTF restart and operation, and how this would impact Hanford's existing waste management infrastructure. Management of wastes that would be generated under implementation of Alternative 1 (Restart FFTF) is discussed in Section 4.3 of Volume 1 (e.g., see Section 4.3.1.1.13). Section 4.3.1.1.13 of the NI PEIS was revised to clarify that the Hanford waste management infrastructure is analyzed in the PEIS for the management of waste resulting from FFTF restart and operation. This analysis is consistent with policy and DOE Order 435.1, *Radioactive Waste Management,* that DOE radioactive waste shall be treated, stored, and in the case of low-level waste, disposed of at the site where the waste is generated, if practical, or at another DOE facility. However, if DOE determines that use of the Hanford waste management infrastructure or other DOE sites is not practical or cost effective, DOE may issue an exemption under DOE Order 435.1 for the use of non-DOE facilities (i.e., commercial facilities) to store, treat, and dispose of such waste generated from the restart and operation of FFTF. In addition, Sections 4.3.3.1.13 and 4.4.3.1.13 also address the potential impacts associated with the waste generated from the target fabrication and processing in the Fuels and Materials Examination Facility (FMEF) and how this waste would be managed at the site.

A number of commentors also raised concern that processing of irradiated targets for production of plutonium-238 would generate high-level radioactive waste. DOE Manual 435.1, *Radioactive Waste Management,* defines high-level radioactive waste as "the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation." DOE has prepared an implementation guide to M 435.1 to assist in implementing the requirements contained in that manual. For this particular "requirement," the definition of high-level radioactive waste, the guide is intended to facilitate the classification of indefinite waste as to whether or not it is high-level radioactive waste. It is recognized that the definition of high-level radioactive waste is not precise and is essentially a source-based definition that also alludes to concentrations of a given waste stream. Page II-8 of the guide notes that "For the purpose of managing high-level waste under DOE M 435.1 [sic], spent nuclear fuel includes spent driver elements and/or irradiated target elements that contain transuranium elements." This statement was included in the guide because the concentrations of long-lived isotopes are likely to be somewhat high during reprocessing and it also meets the source-based definition. As a result of reviewing this guide and to address the comments raised, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. As a result, the Waste Management sections (i.e., Sections 4.3.1.1.13, 4.3.2.1.13, 4.3.3.1.13, and 4.4.3.1.13) of the

NI PEIS have been revised to reflect this different classification from what was assumed in the Draft NI PEIS. As discussed in these revised sections, irrespective of how the waste is classified (i.e., transuranic or high-level radioactive waste), the composition and characteristics are the same and the waste management (i.e., treatment and onsite storage) for the NI PEIS would be the same. In addition, even if the waste is managed as high-level radioactive waste it would have no impact on the existing high-level radioactive waste management infrastructure (e.g., high-level waste storage tanks) because the high-activity waste from processing of the targets would be initially stored and vitrified within the processing facility (i.e., FMEF, the Radiochemical Engineering Development Center (REDC) or the Fluorinel Dissolution Process Facility [FDPF]).

Commentors also expressed concern over the potential impacts of spent nuclear fuel generation from FFTF restart and operation, particularly regarding human health risk. The NI PEIS estimates that about 16 metric tons of heavy metal spent nuclear fuel would be generated over 35 years of operation of FFTF. Hanford is currently managing about 2,000 metric tons of heavy metal spent nuclear fuel. The radiation risk to a maximally exposed individual from normal operational activities during management of the current stored spent nuclear fuel over 35 years is 1.4×10^{-8} latent cancer fatality. The risk to the maximally exposed individual that would be associated with the new nuclear infrastructure operations to restart FFTF and operate FMEF or the Radiochemical Processing Laboratory is 9.5×10^{-8} latent cancer fatality. Furthermore, only a small fraction of this risk would be attributable to management of the additional spent nuclear fuel at FFTF. The annual dose to the maximally exposed individual from all current and reasonably foreseeable activities at Hanford is less than 0.2 millirem. The dose is well within the DOE dose limits given in DOE Order 5400.5, *Radiation Protection of the Public and the Environment*. As discussed in that order, the dose limit from airborne emissions is 10 millirem per year, as required by EPA regulations under the Clean Air Act; the dose limit from drinking water is 4 millirem per year, consistent with the EPA drinking water criteria under the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. The risk to the population from all activities at Hanford would be 0.21 latent cancer fatality over 35 years. DOE has committed to remove the spent nuclear fuel at Hanford for ultimate disposition in a geologic repository.

Cost of the Various Alternatives. Commentors expressed opinions about the costs related to the stated missions. Commentors stated that a cost-benefit analysis was necessary to show the value of production of medical isotopes balanced against facility costs, in particular, the restart of FFTF, and noted that perhaps facilities would be able to pay for themselves. There were concerns that FFTF restart would take funds away from the cleanup of Hanford. Commentors noted that the decommissioning costs were not included for the restart FFTF option in the *Cost Report for Alternatives Presented in the Draft Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (NI Cost Report)*. Several commentors remarked that the expense of plutonium-238 production cannot be justified when DOE needs to clean up existing problems at its sites.

Although the costs of proposed actions are not required by NEPA and CEQ regulations to be included in a PEIS, DOE prepared a separate *NI Cost Report*. This report would provide additional pertinent information to the Secretary of Energy so that he may make an informed decision with respect to the alternatives presented in the Final NI PEIS. Pursuant to CEQ regulations (40 CFR Section 1505.1(e)), such documents comparing alternatives should be made available to the public prior to any decision being made. DOE mailed this document to more than 730 interested parties on August 24, 2000. This report was made available immediately upon release on the NE web site (http://www.nuclear.gov) and in the public reading rooms. DOE has also provided the summary of the *NI Cost Report* in Appendix P, of the Final NI PEIS.

Nuclear Nonproliferation Policy. Commentors expressed opinions about the nuclear nonproliferation implications of the proposed action. Commentors were concerned about keeping plutonium-238 out of the hands of third parties, and it was suggested that the purchase of plutonium-238 from Russia would stop

proliferation of the material and the United States would know the disposition of the quantity purchased. Several commentors raised concerns about specific facilities described in the NI PEIS, including FDPF and FFTF. The use of highly enriched uranium fuel in FFTF was questioned related to possible violation of U.S. nuclear nonproliferation policy. Conversely, the shutdown of FFTF that occurred previously was characterized as being done to discourage proliferation of nuclear weapons worldwide, but had instead weakened the U.S. position as a world leader in nuclear technology. There were comments about the timeliness of release of the *NI Nonproliferation Impact Assessment*, that no nonproliferation information was included in the Draft NI PEIS, and that nuclear nonproliferation policy should be considered by DOE in selection of its preferred alternative.

The plutonium being considered for production in the NI PEIS is plutonium-238, which is not the same isotope of plutonium that is used in nuclear weapons. The production of plutonium-238 does not present a nonproliferation concern. DOE developed the separate *NI Nonproliferation Impact Assessment*, published in September 2000, that analyzed the nonproliferation impacts of the actions considered in the NI PEIS and found that there are no U.S. nonproliferation policies, laws, regulations, or international agreements that preclude the use of any of the facilities in the manner described in the Draft NI PEIS. Although this policy analysis is not required under NEPA, it is an essential element in the decision-making process for the DOE nuclear infrastructure. A summary of the *NI Nonproliferation Impact Assessment* is included in Appendix Q of the Final NI PEIS. It is also available on the DOE NE web site (http://www.nuclear.gov).

Public Involvement. Commentors expressed opinions about the length of the comment period on the Draft NI PEIS, and said they wanted additional time to obtain and review relevant documents, including the *NI Cost Report* and *NI Nonproliferation Impact Assessment*. The deadline for transmittal of comments was changed from September 11, 2000 (as stated in the transmittal letter of the Draft PEIS and the Summary) to September 18, 2000. While the official comment period ended on September 18, 2000, DOE addressed late comments to the extent practicable and considered all comments received through October 31, 2000, in preparing the Final NI PEIS. Comments that were received through September 25, 2000, along with corresponding responses, have been included in Chapter 2 of the comment response volume (Volume 3). Direct responses are not included to comments that were received after September 25, 2000. However, all of these comments were considered and are characterized by other comments received during the comment period (for which a response has been provided).

Many commentors expressed the opinion that public input is intended for "show only," and that DOE has already made its decisions. Commentors also stated that they had given the same comments over and over again and that DOE representatives were not listening. DOE policy encourages effective public participation in its decision-making process. In compliance with NEPA and CEQ regulations, DOE provided opportunity to the public to comment on the scope of the NI PEIS and the environmental impact analysis of DOE's proposed alternatives. DOE gave equal consideration to all comments. In preparing the Final NI PEIS, DOE carefully considered all comments received from the public.

Some commentors expressed opinions about the conduct of the hearings, both positive and negative. The public hearing format was designed to be fair. The public hearing format used was based on stakeholder input and was presented in the Notice of Availability (65 FR 46443 et seq.) for the Draft NI PEIS. This format was intended to encourage public participation, regardless of the motivation for attending the hearing. It provided an opportunity for the participants to meet, exchange information, and share concerns with DOE personnel available throughout the course of each hearing to answer questions. The meetings were facilitated by an independent moderator to ensure that all persons wishing to speak had an opportunity to do so. Persons wishing to comment were selected at random from the audiences rather than according to the order in which they registered. This was accomplished by a random number drawing. In addition to the comment recorder stationed at the main hearing, a second recorder was available in an adjacent room to receive comments

without the need to await selection at the main proceeding. The hearing format promoted open and equal representation by all individuals and groups.

Environmental Impacts. A number of commentors questioned the results of the environmental impact analysis and cumulative impacts, specifically at Hanford. Many of these comments focused on concerns that the proposed action would result in negative impacts to the health of individuals residing in the Hanford region. The NI PEIS analyzes the impacts of the various alternatives, and the environmental impacts associated with all proposed nuclear infrastructure activities are addressed in detail in Chapter 4 of Volume 1. Specifically, the environmental impacts associated with operation of the Hanford facilities during normal operations and from postulated accidents are presented in Section 4.3. These assessments were made using well-established and accepted analytical methods, as described in Appendixes G through L in Volume 2. The analytical methodology is conservative by nature; the actual impacts to the environment would be expected to be less than calculated. All impacts have been shown to be small. No fatalities among workers or the general public would be expected over the 35-year operational period. The impacts to the biosphere (air, water, and land) were also evaluated and determined to be small.

Some commentors raised specific concern over potential contamination of the Columbia River resulting from the restart of FFTF. However, FFTF is approximately 4.5 miles from the Columbia River. There are no discharges to the river from FFTF and no radioactive or hazardous discharges to groundwater. As indicated in analyses presented in Chapter 4 of Volume 1 (e.g., Sections 4.3.1.1.4, 4.3.3.1.4, 4.4.3.1.4, 4.5.3.2.4, and 4.6.3.2.4), there would be no discernible impacts to groundwater or surface water quality at Hanford from operation of Hanford facilities that would support the nuclear infrastructure missions described in Section S.1.

A number of commentors also expressed concern that DOE would expose individuals in the Pacific Northwest to risks associated with the importing weapons-grade plutonium. None of the proposed alternatives involve the shipment of any weapons-grade plutonium to any port in the United States. Alternative 1 does postulate that DOE might decide at some point to import mixed oxide fuel from Europe to fuel FFTF. At this time, however, DOE has not proposed to import this fuel through any specific port. If DOE ultimately decides to import fuel from Europe, it would perform a separate NEPA analysis to select a port. This review would address all relevant potential impacts of overseas and inland water transportation, shipboard fires, package handling, land transportation, as well as safeguards and security associated with the import of SNR–300 mixed oxide fuel through a variety of specific candidate ports on the west and east coasts. It would take into account all public comments, including local resolutions, concerning the desirability of bringing mixed oxide fuel into the proposed alternative ports.

In the event that DOE decides to enhance its nuclear infrastructure, it would not expose any population to high, unacceptable risks under any alternative. Any transportation activities that would be conducted by DOE would comply with U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Transportation regulations. Associated transatlantic shipments would comply with International Atomic Energy Agency requirements. In Section J.6.2 of Volume 2, DOE reviewed the potential maximum impacts from the marine transportation of mixed oxide fuel from Europe to a representative military port (Charleston, South Carolina), and overland transportation to Hanford. Also in that section, the results of a bounding analysis show that the maximum potential radiological risks to the surrounding public from mixed oxide fuel shipments would be extremely small (e.g., less than 1 chance in a trillion for a latent cancer fatality per shipment from severe accidents at docks and in channels and less than 1 chance in 50 billion for a latent cancer fatality per shipment from overland highway accidents).

Changes from the Draft NI PEIS

In response to comments on the Draft NI PEIS and as a result of information that was unavailable at the time of its issuance, the Final NI PEIS contains revisions and new information. These revisions and new information are indicated by sidebars. A brief discussion of the most important changes included in the Final NI PEIS is provided in the following paragraphs.

Chapter 1. ***Purpose and Need for Agency Action:*** As a result of public comments, additional discussion was incorporated to address DOE's production of medical, research, and industrial isotopes relative to global isotope production and availability. In addition, the discussion of the need for plutonium-238 production for space missions was expanded and updated to reflect the most recent planning guidance provided by NASA to DOE.

Issues Raised During the Public Comment Period on the Draft NI PEIS: Section 1.5, Issues Raised During the Public Comment Period on the Draft NI PEIS, was added to the Final NI PEIS.

Related NEPA Reviews: The Final NI PEIS was revised to add descriptions of the *Final Environmental Impact Statement, Management of Spent Nuclear Fuel from the K Basins at the Hanford Site, Richland, Washington* (DOE/EIS-0245F), and the *Environmental Assessment, Management of Hanford Site Non-Defense Production Reactor Spent Nuclear Fuel* (DOE/EA-1185). The impacts of these NEPA actions were factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action. The Final NI PEIS was also revised to reflect recent Records of Decision that have been issued for the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (DOE/EIS-0218F), the *Final Environmental Impact Statement for Treating Transuranic (TRU)/Alpha Low-Level Waste at the Oak Ridge National Laboratory Oak Ridge, Tennessee* (DOE/EIS-0305), and the *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306).

Changes from the Draft NI PEIS: Section 1.8, Changes from the Draft NI PEIS, was added to the Final NI PEIS.

Chapter 2. ***Transportation Requirements:*** Additional U.S. ports were named as candidates for receiving mixed oxide fuel from Europe.

Alternatives Considered and Dismissed: Information was provided to explain why the IPF at LANL, the BLIP and the Alternating Gradient Synchrotron (AGS) accelerator complex at BNL, and CLWRs were not considered reasonable alternatives for the production of medical isotopes. Information was also provided to explain why increasing the power levels at ATR and/or HFIR or installing rapid radioisotope retrieval systems would be insufficient to meet the long-term growth projection needs and therefore dismissed as reasonable alternatives.

Preferred Alternative: The discussion of DOE's preferred alternative for accomplishing the proposed action, that is, Alternative 2, Use Only Existing Operational Facilities, Option 7, is included in the Final NI PEIS.

Summary of Environmental Impacts: Section 2.7 of the NI PEIS was revised in response to comments that it was difficult to compare environmental impacts among alternatives. Although estimates of the environmental impacts that would result from implementation of the alternatives are the same as those in the Draft NI PEIS, the tables and accompanying text were reformatted for ease in comparing environmental impacts among alternatives and among options within alternatives. Section 2.7 was also revised to focus on incremental impacts that would result from implementation of the alternatives. Baseline environmental

impacts were removed from the comparisons among alternatives and options. This information is now presented in Chapter 3.

Chapter 3. ***Affected Environment:*** Additional information was provided on the environmental baseline at each site, including graphics to more clearly illustrate existing surface water and groundwater conditions.

Estimates of existing impacts for current HFIR/REDC operations were added to Sections 3.2.3.2 (Air Quality), 3.2.9.1.2 (Radiation Exposure and Risk), and 3.2.11.1 (Waste Inventories and Activities). Similarly, estimates for current ATR operations were added to Sections 3.3.3.2 (Air Quality), 3.3.9.1.2 (Radiation Exposure and Risk), and 3.3.11.1 (Waste Inventories and Activities). Information was also provided on the impacts of the range fires affecting Hanford and INEEL during the summer of 2000. In addition, site data were updated to reflect recent measurements and analyses. Estimates of existing impacts of maintaining FFTF in standby were added to Section 3.4.3.1 (Air Quality).

In response to public comments on the Draft NI PEIS, additional information on health studies conducted in the Hanford area was also incorporated.

Chapter 4. ***Air Quality:*** Stack parameters used for the air quality modeling were added. In response to public comment, estimates of the ambient air quality concentrations from FFTF sources were added to the deactivation section.

Water Resources: New water use and sanitary wastewater generation increments for REDC and FDPF were added to reflect the revised additional workforce required at these facilities and to be consistent with FMEF. Water use and waste water generation rates for the New Accelerator(s) and New Research Reactor alternatives were also revised. These changes were also incorporated into the waste management analyses.

Ecological and Cultural and Palentological Resources: These sections were updated to reflect consultations concerning threatened and endangered species and cultural resources conducted with appropriate Federal and state agencies. Consultations were also conducted with interested Native American tribes. No major issues were raised as a result of these consultations.

Socioeconomics**:** Section 4.5.1.1.8 was revised to reflect changes in the number of workers associated with FFTF operations and deactivation. The associated impacts on community services were also incorporated. In addition, the number of workers at the Oak Ridge Reservation was revised to reflect the entire site workforce rather than just the number of workers at ORNL.

Normal Operations: Based on more recent site data on occupational radiation exposure for workers at REDC, all worker health impacts for target processing at REDC, FMEF, and FDPF and for neptunium target storage at REDC, Chemical Processing Plant 651 (CPP–651), and FMEF were updated. Also, low-energy accelerator source terms were modified to properly reflect normal operational emissions resulting in modifications to the population health impacts for all options of Alternative 3.

Facility Accidents: The high-energy accelerator analysis was redone to incorporate a more accurate revised source term, and the risks for currently operating reactors were added to the tables. An additional analysis addressing industrial accidents was also performed and incorporated into Chapter 4.

Transportation: The neptunium inventory was revised to use the recently declassified actual inventory. The number of actual shipments from SRS to the processing facilities and the transportation risk estimates were modified accordingly.

Waste Management: The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. The Waste Management sections (i.e., Sections 4.3.1.1.13, 4.3.2.1.13, 4.3.3.1.13, and 4.4.3.1.13) were revised to reflect this different classification from what was assumed in the Draft NI PEIS.

Spent Nuclear Fuel Management: The sections were revised to quantify the generation of spent fuel from 35 years of operation and to state that dry spent nuclear fuel storage at the FFTF site is similar to NRC-approved methods currently being used for interim storage of commercial spent nuclear fuel. In addition, based on public comments, a K Basins spent fuel storage reference document was added.

Cumulative Impacts: Cumulative impact tables in Section 4.8 were revised to present the contributions from each of the various site actions anticipated during the course of the operational period evaluated in the NI PEIS. Air quality tables were also revised to incorporate the revised baseline from Chapter 3. In addition, waste management tables were revised to include the site's treatment, storage, and disposal capacities for easier comparison of the waste generations, by waste type, to the waste management capacities at the sites.

Chapter 5. In response to public comments, a list of organizations that DOE contacted during the consultation process was added.

Volume 2. Summaries of the *NI Cost Report* and *NI Nonproliferation Impact Assessment* were added as Appendixes P and Q, respectively. NASA mission guidance correspondence was added as Appendix R.

Volume 3. Volume 3 of the NI PEIS was added to present the comments received during the public review period for the Draft NI PEIS and DOE's responses to these comments.

S.3 ALTERNATIVES EVALUATED IN THE NI PEIS

The NI PEIS analyzes the potential environmental impacts of using various irradiation and processing facilities to meet the following projected DOE irradiation service mission needs for 35 years: (1) production of medical and industrial isotopes, (2) production of up to 5 kilograms (11 pounds) per year of plutonium-238 for use in advanced radioisotope power systems for future NASA space missions, and (3) support for U.S. civilian nuclear energy research and development activities. The proposed irradiation facilities include facilities that are currently operating, those that could be brought on line, or those that could be constructed and operated to meet DOE's nuclear infrastructure mission requirements. A No Action Alternative and five programmatic alternatives are listed below.

No Action Alternative
Alternative 1—Restart FFTF
Alternative 2—Use Only Existing Operational Facilities
Alternative 3—Construct New Accelerator(s)
Alternative 4—Construct New Research Reactor
Alternative 5—Permanently Deactivate FFTF (with No New Missions)

It is possible during the Record of Decision process that a combination of the alternatives could be selected, for example, a low-energy power accelerator in combination with the existing reactors to optimize research isotope production, or in combination with FFTF to optimize research and therapeutic isotope production.

The alternatives, their associated facility options, and their relative capabilities are described in detail in Chapter 2 of the NI PEIS. As presented in **Table S–1**, the NI PEIS evaluates 26 specific technology/siting options associated with the alternatives identified above. DOE's Preferred Alternative for accomplishing expanded civilian nuclear energy research and development and isotope production missions in the United States is Alternative 2, Use Only Existing Operational Facilities, Option 7. Under this alternative and option, DOE would reestablish domestic production of plutonium-238, as needed, using irradiation capabilities at both ATR at INEEL and HFIR at ORNL. REDC at ORNL would be used to store neptunium-237 and to fabricate and process the targets irradiated at ATR and HFIR. The production of medical and industrial isotopes and support of civilian nuclear energy research and development would continue and increase to the extent possible under current reactor operating levels. FFTF at Hanford would be permanently deactivated. The preferred alternative is discussed in more detail at the end of this section.

No Action Alternative. Under the No Action Alternative (maintain status quo), FFTF would be maintained in standby status for all or a portion of the 35-year evaluation period for operations covered in the NI PEIS. For purposes of analysis in the NI PEIS, the maximum of 35 years was assumed. Ongoing operations at existing facilities, as described in Chapter 3 of the NI PEIS, would continue under this alternative. DOE would not establish a domestic plutonium-238 production capability, but could, instead, continue to purchase Russian plutonium-238 to meet the needs of future U.S. space missions. For the purposes of analysis in the NI PEIS, DOE assumed that it would continue to purchase plutonium-238 to meet the space mission needs for the 35-year evaluation period, and has included in the NI PEIS the transportation impacts of purchasing up to 175 kilograms (385.8 pounds) from Russia. However, DOE recognizes that any purchase beyond what is currently available to the United States through the existing contract will require additional NEPA review. DOE would continue its medical and industrial isotope production and civilian nuclear energy research and development activities at the current operating levels of existing facilities. A consequence of a No Action decision would be the need to determine the future of the neptunium-237 stored at SRS. Therefore, the impacts of possible future transportation and storage of neptunium-237 are evaluated as part of the No Action Alternative.

Four options are analyzed under the No Action Alternative. If DOE decides not to establish a domestic plutonium-238 production capability in the future, the neptunium-237 would have no programmatic value and Option 1 would be selected. Under this option, DOE would follow its current stabilization strategy for the neptunium-237, currently in solution form at SRS. If, however, DOE decides to maintain the neptunium-237 inventory for future plutonium-238 production, the neptunium-237 oxide would be transported from SRS to one of three candidate DOE sites for up to 35 years of storage for possible future use: Option 2, REDC at ORNL; Option 3, Building CPP–651 at INEEL; or Option 4, FMEF at Hanford.

Alternative 1—Restart FFTF. Under Alternative 1, FFTF at Hanford would be restarted and operated at a nominal 100 megawatts for the 35-year evaluation period. FFTF would be used to irradiate targets for medical and industrial isotope production, plutonium-238 production, and research and development irradiation requirements. Ongoing operations at existing facilities as described in Chapter 3 of the NI PEIS would continue.

Targets for medical and industrial isotope production would be fabricated in one or more facilities at Hanford. Target material would typically be acquired from ORNL, where enrichment processes are conducted to produce high purity target material suitable for production of medical isotopes, and stored at Hanford. The targets would be irradiated at FFTF and then returned to the fabrication facility for postirradiation processing. From there, the isotope products would be sent directly to commercial pharmaceutical distributors.

Table S–1 NI PEIS Alternatives and Options

	Option Number	Irradiation Facility	Plutonium-238 Production Mission		Medical and Industrial Isotope Production and Civilian Nuclear Energy Research and Development Mission	
			Storage Facility	Target Fabrication and Processing Facility	Storage Facility	Target Fabrication and Processing Facility
No Action Alternative	1	–	–	–	–	–
	2	–	REDC	–	–	–
	3	–	CPP–651	–	–	–
	4	–	FMEF	–	–	–
Alternative 1: Restart FFTF	1	FFTF[a]	REDC	REDC	RPL/306–E	RPL/306–E
	2	FFTF[a]	FDPF/CPP–651	FDPF	RPL/306–E	RPL/306–E
	3	FFTF[a]	FMEF	FMEF	FMEF	FMEF
	4	FFTF[b]	REDC	REDC	RPL/306–E	RPL/306–E
	5	FFTF[b]	FDPF/CPP–651	FDPF	RPL/306–E	RPL/306–E
	6	FFTF[b]	FMEF	FMEF	FMEF	FMEF
Alternative 2: Use Only Existing Operational Facilities	1	ATR	REDC	REDC	–	–
	2	ATR	FDPF/CPP–651	FDPF	–	–
	3	ATR	FMEF	FMEF	–	–
	4	CLWR	REDC	REDC	–	–
	5	CLWR	FDPF/CPP–651	FDPF	–	–
	6	CLWR	FMEF	FMEF	–	–
	7	HFIR and ATR	REDC	REDC	–	–
	8	HFIR and ATR	FDPF/CPP–651	FDPF	–	–
	9	HFIR and ATR	FMEF	FMEF	–	–
Alternative 3: Construct New Accelerator(s)	1	New	REDC	REDC	New[c]	New[c]
	2	New	FDPF/CPP–651	FDPF	New[c]	New[c]
	3	New	FMEF	FMEF	New[c]	New[c]
Alternative 4: Construct New Research Reactor	1	New	REDC	REDC	New[c]	New[c]
	2	New	FDPF/CPP–651	FDPF	New[c]	New[c]
	3	New	FMEF	FMEF	New[c]	New[c]
Alternative 5: Permanently Deactivate FFTF (with No New Missions)	–	–	–	–	–	–

a. Hanford FFTF would operate with mixed oxide fuel for 21 years and highly enriched uranium fuel for 14 years.
b. Hanford FFTF would operate with mixed oxide fuel for 6 years and highly enriched fuel for 29 years.
c. The new facility would not be required if a DOE site with available support capability and infrastructure is selected.

Key: ATR, Advanced Test Reactor at INEEL; CLWR, commercial light water reactor; CPP–651, INEEL Building CPP–651 Storage Vault; FDPF, Fluorinel Dissolution Process Facility at INEEL; FFTF, Fast Flux Test Facility at Hanford; FMEF, Fuels and Materials Examination Facility at Hanford; HFIR, High Flux Isotope Reactor at ORNL; REDC, Radiochemical Engineering Development Center at ORNL; RPL/306–E, Radiochemical Processing Laboratory and Building 306–E at Hanford.

Targets for plutonium-238 production would be fabricated in one of three candidate facilities at ORNL, Hanford, or INEEL. The material needed for target fabrication (neptunium-237) would be transported from SRS to the fabrication facilities, where it would be stored until fabrication. The nonirradiated targets would be transported to and irradiated at FFTF and transported back to the fabricating facilities for postirradiation processing. The separated plutonium-238 would be transported to LANL for fabrication into heat sources for radioisotope power systems and heating units.

Under Alternative 1, raw materials, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for raw target material acquisition, material storage, target fabrication, target irradiation, and postirradiation processing and the final destination for the medical and industrial isotopes and the plutonium-238 product or various research and development test sites.

The six options under this alternative are associated with the type of nuclear fuel to be used for FFTF operations and the specific facilities to be used for target fabrication and processing. The first three options (Options 1 through 3) would involve operating FFTF with a mixed oxide fuel core for the first 21 years and a highly enriched uranium fuel core for the remaining 14 years. The last three options (Options 4 through 6) would involve operating FFTF with a mixed oxide fuel core for the first 6 years and a highly enriched uranium fuel core for the remaining 29 years. FFTF can provide similar irradiation services with either a mixed oxide core or a highly enriched uranium core. Potential impacts from the deactivation of FFTF at the end of its operating life are not explicitly covered under this alternative, but are addressed under Alternative 5.

The U.S. nonproliferation policy (U.S. House of Representatives 1992 and the White House 1993) strongly discourages the use of highly enriched uranium fuel in civilian research and test reactors. The Reduced Enrichment for Research and Test Reactors Program implements this policy by developing technical means to reduce and eventually eliminate the use of highly enriched uranium in research and test reactors throughout the world and in the United States, without decreasing their safety or significantly affecting their performance and operating costs.

To be in compliance with these policy directives, the most appropriate fuel supply for FFTF in the out years (beyond current Hanford mixed oxide and possible SNR–300 mixed oxide supplies) must be determined by a technical study with the preferred fuel source being low-enriched uranium. Highly enriched uranium fuel should only be considered if low-enriched uranium is not technically feasible, or if there are significant impacts on safety, performance, or cost associated with using fuels other than highly enriched uranium.

In the event that a decision is made to restart the reactor, and to support these policy directives, DOE's Office of Nonproliferation and National Security would undertake a study to consider the technical feasibility of low-enriched uranium fuel (under the Reduced Enrichment for Research on Test Reactors Program) for FFTF. If low-enriched uranium fuel is found infeasible, DOE would subsequently procure highly enriched uranium fuel in a manner consistent with U.S. nonproliferation policy. This study would be conducted, decisions implemented, and fuel made available during the time period between a Record of Decision indicating an FFTF restart and prior to the end of available Hanford mixed oxide and possible SNR–300 mixed oxide fuel supplies.

For the purposes of presenting a bounding analysis in the NI PEIS, DOE has analyzed the impacts of using highly enriched uranium fuel in FFTF after the available mixed oxide fuel supplies have been expended. These impacts would bound those of using a low-enriched uranium fuel form.

Alternative 2—Use Only Existing Operational Facilities. Under Alternative 2, DOE would use existing operating DOE reactors or U.S. commercial nuclear power plants to produce plutonium-238 for future space

missions. The production of medical and industrial isotopes and support of civilian nuclear energy research and development in DOE reactors and accelerators would continue at the No Action Alternative level.

The currently operating DOE reactors, HFIR and ATR, cannot fully meet the projected long-term need for medical isotope production and civilian nuclear energy research and development, with or without the plutonium-238 production mission. Depending on the combination of facilities used in Alternative 2, HFIR and ATR could continue their current support of the medical and industrial isotope and research and development missions, including some near-term growth, while accommodating the production of plutonium-238. Under other scenarios, some of the near-term growth in medical and industrial isotope production and civilian nuclear energy research and development possible in these reactors could be limited by the addition of the plutonium-238 production. In any case, non-DOE use of these facilities would be affected by the addition of the plutonium-238 mission. If a commercial reactor were used for plutonium-238 production, the DOE facilities would be unaffected and would continue operating as discussed under the No Action Alternative.

Another component of Alternative 2 is permanent deactivation of FFTF. Permanent deactivation of FFTF (Alternative 5) would occur in conjunction with any of the options under Alternative 2, 3, or 4. Ongoing operations at existing facilities as described in Chapter 3 of the NI PEIS would continue under Alternative 2.

Targets for plutonium-238 production would be fabricated in one of three facilities at ORNL, INEEL, or Hanford. The material needed for target fabrication (neptunium-237) would be processed and transported from SRS to the fabrication facilities where it would be stored until fabrication. The targets would be irradiated at existing reactor facilities (HFIR, ATR, CLWR, as described in Section S.4) and would be transported back to the fabricating facilities for postirradiation processing.

Under Alternative 2, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for storage, target fabrication, target irradiation, and postirradiation processing. In addition, the plutonium-238 product would be transported to LANL.

Nine options are proposed under this alternative. Options 1 through 3 involve the irradiation of targets in ATR at INEEL. Options 4 through 6 involve the irradiation of targets in a generic CLWR. Options 7 through 9 involve the irradiation of targets in both INEEL's ATR and ORNL's HFIR.

Alternative 3—Construct New Accelerator(s). Under Alternative 3, one or two new accelerators would be used for target irradiation for the evaluation period of 35 years. The new accelerator(s), which would be constructed at an existing DOE site(s), would be used to irradiate all of the targets (i.e., for production of plutonium-238, isotopes for medical and industrial uses, and materials testing for research and development). Ongoing operations at existing facilities as described in Chapter 3 of the NI PEIS would continue.

The targets for plutonium-238 production would be fabricated in one of the three alternative facilities at ORNL, INEEL, or Hanford. The material needed for the target fabrication (neptunium-237) would be transported from SRS to the fabrication facilities, where it would be stored until fabrication. The targets would be irradiated at a new high-energy accelerator facility and transported back to the target fabricating facilities for postirradiation processing.

Targets for medical and industrial isotope production would be fabricated in a new support facility located at the same site as the low-energy accelerator. Target materials would be stored on site until fabrication. The targets would be irradiated in the low-energy accelerator and returned to the new support facility for postirradiation processing. Site selection for Alternative 3 is not evaluated as part of the NI PEIS. Because Alternative 3 is evaluated at a generic DOE site, no credit was taken for any support infrastructure existing at

the site, and it was postulated that a new support facility would be required to support operation of the low-energy accelerator and its missions and the high-energy accelerator civilian nuclear energy research and development missions if both accelerators were located on the same site. While this approach bounds the environmental impact assessment for the implementation of Alternative 3, it overstates the impacts because the NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 3 or the low-energy accelerator alone is selected by the Record of Decision for subsequent consideration, follow-on NEPA reviews would evaluate potential locations for either both accelerators or one of the accelerators. It is unlikely that DOE would consider locating the new low-energy or high-energy accelerator on a DOE site that does not have existing infrastructure capable of supporting all or most of the mission requirements.

Under Alternative 3, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for storage, target fabrication, target irradiation, postirradiation processing, and the final destination of the plutonium-238. Alternative 3 also would include decontamination and decommissioning of the accelerator(s) and the processing facility when the missions are over, as well as deactivation of FFTF at Hanford.

Alternative 4—Construct New Research Reactor. Under Alternative 4, a new research reactor would be used for target irradiation for the evaluation period of 35 years. The new research reactor, to be constructed at an existing DOE site, would be used to irradiate all targets (i.e., for the production of plutonium-238, isotopes for medical and industrial uses, and materials testing for civilian nuclear energy research and development). Ongoing operations at existing facilities as described in Chapter 3 of the NI PEIS would continue.

The targets for plutonium-238 production would be fabricated in one of the three facilities at ORNL, INEEL, or Hanford. The material needed for the target fabrication (neptunium-237) would be transported from SRS to the fabrication facilities where it would be stored until fabrication. The targets would be irradiated at the new research reactor facility and transported back to the target fabrication facilities for postirradiation processing.

Targets for medical and industrial isotope production would be fabricated in a new support facility located at the same site as the new research reactor. Target materials would be stored on site until fabrication. The targets would be irradiated in the new research reactor and returned to the new support facility for postirradiation processing.

Alternative 4 site selection is not evaluated as part of the NI PEIS. Because Alternative 4 is evaluated at a generic DOE site, no credit was taken for any existing support infrastructure at the site and it was postulated that a new support facility would be required to support operation of the new research reactor and its missions. While this approach bounds the environmental impact assessment for the implementation of Alternative 4, it overstates the impacts because the NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 4 is selected by the Record of Decision for subsequent consideration, follow-up NEPA reviews would evaluate potential locations for the new research reactor. It is unlikely that DOE would consider locating the new research reactor on a DOE site that does not have existing infrastructure capable of supporting all or most of the proposed medical and industrial isotope production and civilian nuclear energy research and development mission requirements.

Under Alternative 4, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for storage, target fabrication, target irradiation, postirradiation processing, and the final destination of the plutonium-238. Alternative 4 also would include the decontamination and

decommissioning of both the research reactor and the support facility when the missions are over, as well as deactivation of FFTF at Hanford.

Alternative 5—Permanently Deactivate FFTF (with No New Missions). Under Alternative 5, DOE would permanently deactivate FFTF, with no new missions. Medical and industrial isotope production and civilian nuclear energy research and development missions, at the existing facilities described in Chapter 3, would continue. DOE's nuclear facilities infrastructure would not be enhanced.

Selection of Alternatives

In the NI PEIS Record of Decision, DOE can select any alternative or combination of alternatives or elements of alternatives. For example, DOE could select Alternative 2 in combination with the new low-energy accelerator element of Alternative 3. This combination of alternative elements would provide for the requirements of the plutonium-238 production, enhanced civilian nuclear energy research and development capability, and enhanced medical and industrial isotope production capability.

Alternatives Considered and Dismissed

In developing a range of reasonable alternatives, DOE examined the capabilities and available capacities of the existing and planned nuclear research facilities (accelerators, reactors, and processing [hot] cells) that potentially could be used to support one or all of the proposed isotope production and research missions (DOE 2000a). The following facilities were initially considered, but were subsequently dismissed as reasonable alternatives for meeting DOE's nuclear infrastructure mission requirements.

Irradiation Facilities Dismissed. DOE evaluated the irradiation capabilities of existing government, university, and commercial irradiation facilities to determine whether they could significantly support the proposed expanded nuclear infrastructure missions. **Table S–2** presents irradiation facilities that were initially considered but dismissed from further evaluation because they lacked technical capability or available capacity. Reasons for lacking technical capability include that the facility has been permanently shut down, it does not possess the capability to produce steady-state neutrons, or that it could not maintain sufficient power levels to adequately support steady-state neutron production. Facilities were similarly dismissed if existing capacity was fully dedicated to existing missions, or if use of existing capacity to support the NI PEIS proposed action would impact existing missions. Although a number of facilities shown in Table S–2 have some available capacity, their combined available capacity is a very small percentage of the capacity needed to support the missions evaluated in the NI PEIS.

Two of these facilities, IPF at the Los Alamos Neutron Science Center (LANSCE) and BLIP at BNL, were identified in the NI PEIS Notice of Intent as existing facilities that could potentially support the proposed nuclear infrastructure missions. IPF produces radioisotopes using LANSCE's half-mile accelerator that delivers medium-energy protons. IPF's three major products include germanium-68, strontium-82, and sodium-22. As a result of changing DOE missions, the production of radioisotopes at target area "A" of the LANSCE has been rendered inoperable. DOE is currently in the process of upgrading the LANSCE facility with a new 100-million-electron-volt IPF. The facility is scheduled for completion in 2001. After completion of the LANSCE upgrade, the existing capability at these two facilities will be twice the current need for accelerator-generated medical isotopes. Thus, no new accelerator capacity is needed in the short term. Should isotope demand grow consistent with the Expert Panel Report, there will be a need for expanded isotope production capacity for those isotopes generated by IPF and BLIP. IPF and BLIP were dismissed as a reasonable alternative for the production of medical isotopes because they cannot meet the projected future demand for accelerator-produced isotopes.

Table S–2 Irradiation Facilities Considered and Dismissed from Further Evaluation

Reasons for Dismissal	Facility
Facilities lacking sufficient neutron production capacity to support the NI PEIS proposed action without impacting existing missions	Neutron Radiographic Reactor Argonne National Laboratory–West
	Brookhaven Medical Research Reactor Brookhaven National Laboratory
	National Bureau of Standards Reactor National Institute of Standards and Technology
	General Atomics Training, Research, and Isotope Production Reactors
	University Small Research Reactors
	University Large Research Reactors (i.e., Massachusetts Institute of Technology and University of Missouri)
	ATLAS Heavy Ion Facility Argonne National Laboratory
	Holifield Radioactive Ion Beam Facility Oak Ridge National Laboratory
	Oak Ridge Electron Linear Accelerator Oak Ridge National Laboratory
	Heavy Ion Linear Accelerator Lawrence Berkeley National Laboratory
	Alternating Gradient Synchrotron Heavy Ion Facility Brookhaven National Laboratory
	Continuous Electron Beam Accelerator Facility Thomas Jefferson National Accelerator Facility
	Electron Linear Accelerator Lawrence Livermore National Laboratory
	University Linear Accelerators
Facilities with capacity fully dedicated to existing missions	Annular Core Research Reactor Sandia National Laboratory
	Brookhaven LINAC Isotope Producer Brookhaven National Laboratory
Facilities not capable of steady-state neutron production	Sandia Pulse Reactor II and III Sandia National Laboratory
	Transient Reactor Test Facility Argonne National Laboratory–West
	Zero Power Physics Reactor Idaho National Engineering and Environmental Laboratory
	Power Burst Facility Idaho National Engineering and Environmental Laboratory
	Intense Pulsed Neutron Source Argonne National Laboratory
	Flash X-Ray Facility Lawrence Livermore National Laboratory
Facilities with insufficient power to sustain adequate steady-state neutron production	Brookhaven Medical Research Reactor Brookhaven National Laboratory
	Los Alamos Critical Assembly Facility Los Alamos National Laboratory
	General Atomics Training, Research and Isotope Production Reactors
	University Small Research Reactors
	Booster Applications Facility Brookhaven National Laboratory

Table S–2 Irradiation Facilities Considered and Dismissed from Further Evaluation (Continued)

Reasons for Dismissal	Facility
Facilities with insufficient power to sustain adequate steady-state neutron production (continued)	Cyclotron Facility Brookhaven National Laboratory
	Low-Energy Demonstration Accelerator[a] Los Alamos National Laboratory
Facilities that jointly can meet existing accelerator-produced medical isotope demands but cannot meet projected future demands.	Los Alamos Neutron Science Center Linear Accelerator Isotope Production Facility Los Alamos National Laboratory
	Brookhaven LINAC Isotope Producer Brookhaven National Laboratory
Facilities that are under construction with capacity fully dedicated to other planned missions	Dual Axis Radiographic Hydrodynamic Test Facility Los Alamos National Laboratory
	Spallation Neutron Source Oak Ridge National Laboratory
Facilities that have been permanently shut down	High Flux Beam Reactor Brookhaven National Laboratory
	Tower Shielding Facility Oak Ridge National Laboratory
	Cyclotron Facility Oak Ridge National Laboratory

a. Not listed in source document.
Key: LINAC, linear accelerator; ATLAS, Argonne Tandem - LINAC Accelerator System.
Source: DOE 2000a.

The AGS accelerator complex at BNL was evaluated for meeting the mission requirements of medical and industrial isotope production, plutonium-238 production, and civilian nuclear energy research and development. AGS presently accelerates up to 7×10^{13} protons to 24 gigavolts (1,000 million electron volts) with a cycle time of approximately 2.5 seconds. This corresponds to a beam power of approximately 100 kilowatts. The complex was dismissed as a reasonable alternative because the potential neutron flux generated by the facility in the required configuration (i.e., with a spallation target) would not be adequate to meet the mission goals and, in addition, operating the complex in the required configuration would not be compatible with the present primary mission of the facility (Kovar 2000).

Two existing operating DOE facilities, ATR and HFIR, were evaluated as components of Alternative 2, Use Only Existing Operational Facilities. These two facilities currently provide isotope production capability, and were examined for their ability to meet the isotope production and civilian nuclear energy research and development requirements of the proposed expanded missions. In addition, DOE considered whether production from ATR and HFIR could be enhanced by increasing power levels at the reactors or through other modifications to the facilities, which included the installation of rapid radioisotope retrieval systems for the production of isotopes with a short half-life. In general, the installation of rapid radioisotope retrieval systems in reactors does not increase the ability of reactors to produce larger quantities of isotopes, it enable the reactors to produce a broader spectrum of isotopes. While some growth is possible in isotope production at ATR and HFIR, such growth would be insufficient to meet the long-term growth projections. Further growth could only be enabled by increasing reactor power levels. At ATR, increases in power levels are possible to the extent that priority DOE Office of Naval Reactor missions are not impacted. Raising ATR power would only delay the point in time at which capacity is reached. The power level at HFIR is already at 100 percent of its current Authorization Basis (85 megawatts), and modification of this Authorization Basis would be required to increase to full-design power (100 megawatts). Increasing the power levels at ATR and/or HFIR will enhance the isotope production capability of these reactors. However, the enhancement in production capability would not be adequate to meet the future demand for isotope production; it would only delay the point in time at

which the United States' reactor isotope production capacity is reached. Therefore, increasing the power levels at ATR and/or HFIR was dismissed as a reasonable alternative for meeting the requirements of the DOE missions.

Modification of CLWRs to enable online insertion and retrieval of targets for the medical and industrial isotope production missions was evaluated and dismissed as a reasonable alternative. This decision was made because the required facility modifications would be significant and would include penetrations into the reactor vessel and, potentially, the containment vessel. Additional facility modifications would be required to enable loading of the targets into a shielded cask for transport to a processing facility. Performing these facility modifications would require an extended refueling outage (with a resulting loss of power generation revenue to the CLWR owner) and could potentially extend subsequent maintenance or refueling outages to inspect, test and maintain the insertion and retrieval system, reactor vessel penetrations, and potential containment vessel penetrations. CLWRs were considered for the production of medical isotopes with moderate and long half-lives by irradiating targets in the CLWR vessel but outside the reactor core region (i.e., outside of the fuel assembly region). Only one isotope, strontium-89, was considered a potential candidate for production in the CLWR outside of the reactor core region. Strontium-89 has a half-life of 50.5 days. Irradiated targets containing strontium-89 could only be harvested from a CLWR every 18 to 24 months during a scheduled reactor refueling outage. Approximately 10 CLWRs, with refueling outages scheduled every 2 to 3 months, would be required to support a program to ensure a continuous and reliable supply of strontium-89. Due to the CLWR's ability to irradiate targets for only a very limited array of medical isotopes (only one isotope in current demand was identified), it was not considered a reasonable alternative for expanding the U.S. infrastructure to provide an overall enhancement of the medical isotope production mission. CLWRs were also considered for the DOE civilian nuclear energy research and development missions. CLWRs will continue to support the commercial industry research and development activities by providing a test bed for industry sponsored lead test assemblies and other related research. CLWRs cannot meet most of the requirements for supporting the DOE civilian nuclear energy research and development missions and were therefore dismissed as a reasonable alternative for supporting these missions.

Canadian Deuterium Uranium (CANDU) reactors, operating in Canada, were considered for supplying irradiation services for the plutonium-238 production mission. (Note: Canada is currently the major supplier of medical radioisotopes used in the United States.) Since use of the CANDU reactors does not meet the programmatic issue being addressed in the NI PEIS, that is the enhancement of the United States infrastructure to support the proposed missions, the CANDU reactors were considered and dismissed as a reasonable alternative. However, the environmental impacts associated with transporting the nonirradiated and irradiated neptunium-237 targets between the CANDU reactors and the target fabrication and processing facilities in the United States are bounded by the evaluations presented in the NI PEIS for the commercial light water reactor options of Alternative 2, Use Only Existing Operational Facilities.

Some facilities listed in Table S–2 that do not have the capacity to support the proposed action without impacting existing missions do have some existing medical or industrial isotope production or civilian nuclear energy research and development missions. These facilities will continue to support their existing missions at current levels.

Processing Facilities Dismissed. Numerous existing U.S. processing hot cell facilities possess the capabilities and capacity to support the proposed missions. Given this general availability, only existing processing facilities that are colocated at DOE's candidate irradiation facility sites (i.e., ORNL, INEEL, and Hanford) were evaluated in the NI PEIS. Although multiple processing facilities exist at each of these sites, only the most suitable facilities in terms of capability, capacity, and availability were given further consideration. The processing facilities that were dismissed from consideration are listed in **Table S–3**.

Table S–3 Processing Facilities Considered and Dismissed from Further Evaluation

Location	Facility
Argonne National Laboratory	Irradiated Materials Facility
	Alpha-Gamma Hot Cell Facility
	Building 205
Argonne National Laboratory–West	Hot Fuel Examination Facility
	Analytical Laboratory
	Fuel Conditioning Facility
Brookhaven National Laboratory	Target Processing Laboratory
	Metallurgical Evaluation Laboratory
	High Intensity Radiation Development Laboratory
Hanford Site	222-S Facility
	Postirradiation Testing Laboratory
	Shielded Material Facility
Idaho National Engineering and Environmental Laboratory	Test Area North Hot Shop and Hot Cell Facilities
	Remote Analytical Laboratory
	Fuel Processing Facility
Los Alamos National Laboratory	Chemistry and Metallurgical Research Building
	Technical Area TA–48
Oak Ridge National Laboratory	Radioactive Materials Analytical Laboratory
	Building 4501
	Irradiated Materials Examination and Testing Facility
	Radioisotope Development Laboratory
	Irradiated Fuels Examination Laboratory
Sandia National Laboratories	Hot Cell Facility
Savannah River Site	Defense Waste Processing Facility
	High-level cells
	Intermediate-level cells
	Californium shipping/receiving facility
	Californium processing facility

Source: DOE 2000a.

Based on comments on the scope of the *Plutonium-238 Production EIS*, the H-Canyon and HB-Line facilities at SRS that previously performed the processing for the plutonium-238 production mission were reconsidered as potential processing facilities for the proposed plutonium-238 production mission even though the facilities are not colocated with a proposed irradiation facility. After reviewing the plutonium-238 production target fabrication and processing requirements, the capabilities and capacities of the facilities, and the modifications and resources required to support the plutonium-238 production mission, use of the H-Canyon and HB-Line facilities was dismissed as a reasonable alternative because:

1. DOE plans to shut down these facilities following completion of their current missions to stabilize and prepare for disposition of Cold War legacy nuclear materials and certain spent nuclear fuel, and a determination that a new nonchemical processing technology is capable of preparing aluminum-clad research reactor spent nuclear fuel for ultimate disposition.

2. The cost to extend the operating lives of these facilities to support plutonium-238 production for the proposed 35-year evaluation period would be approximately one order of magnitude higher than the costs associated with the processing facilities evaluated in the NI PEIS.

A commentor also proposed using the H-Canyon and HB-Line for a short campaign to produce all of the required plutonium-238. Based on prior production rates, it would take approximately 7 years to produce 175 kilograms (385 pounds) of plutonium-238, the total plutonium-238 production goal. The target fabrication and irradiation requirements to support this processing campaign to produce 25 kilograms (55 pounds) per year of plutonium-238 would be significant but feasible. The irradiation requirements could be supported by operating five CLWRs or operating FFTF at the 400-megawatt power level. However, a concern about the short campaign option is that the plutonium-238 would be stored a long time before use and because of natural decay may not meet the specification requirements when finally needed. This alternative was dismissed because of the uncertainty that, over time, the plutonium-238 produced may not meet the required specification for NASA missions.

Preferred Alternative

CEQ regulations require an agency to identify its preferred alternative(s) in the final programmatic environmental impact statement (40 CFR 1502.14(e)). The preferred alternative is the alternative that the agency believes would fulfill its statutory mission, giving consideration to environmental, economic, technical, and other factors. Consequently, to identify a preferred alternative, DOE has developed information on potential environmental impacts, costs, policy issues, technical risks, and schedule risks for the alternatives under consideration. The NI PEIS provides information on the environmental impacts. Cost, nonproliferation policy, and various technical reports have also been prepared and are available in the appropriate DOE Reading Rooms for public review.

Based on the analysis discussed above, DOE's Preferred Alternative is to apply its existing infrastructure to the extent possible to pursue the missions outlined in the NI PEIS, that is, Alternative 2, Option 7. Under this approach, DOE proposes to consider opportunities to enhance its existing facilities to maximize the agency's ability to address future mission needs.

The Preferred Alternative also addresses the future of FFTF. While DOE recognizes that this facility has unique capabilities, the Department did not receive the commitments from the private sector or other governments that would clearly justify the restart of the facility. Lacking such commitment, DOE would permanently deactivate FFTF under the Preferred Alternative.

Finally, under the Preferred Alternative, DOE proposes to reestablish domestic production of plutonium-238, as needed, to support U.S. space exploration. ATR in Idaho and HFIR in Tennessee would be used, as appropriate, to irradiate targets for this purpose without interfering with either reactor's primary mission. The Preferred Alternative includes processing the irradiated plutonium-238 targets at REDC at ORNL.

In view of the lack of commitments that would justify the restart of FFTF or the construction of new facilities as proposed under Alternatives 3 and 4, DOE anticipates that its current infrastructure will serve the needs of the research and isotope communities for the next several years. In particular, DOE will consider opportunities to enhance its effort to provide medical and research isotopes. If significantly larger amounts of isotopes are required in the future, DOE would rely on the private sector to fulfill these needs.

As a potential option for the longer-term future, DOE proposes to work over the next 2 years to establish a conceptual design for an Advanced Accelerator Applications (AAA) facility. Such a facility, which would be used to evaluate spent fuel transmutation, conduct various nuclear research missions, and ensure a viable backup technology for the production of tritium for national security purposes, was proposed and initial work funded in the fiscal year 2001 Energy and Water Appropriation. If DOE proposes specific enhancements of existing facilities or development of the AAA facility, further NEPA review would be conducted.

S.4 OVERVIEW OF NUCLEAR INFRASTRUCTURE FACILITIES AND TRANSPORTATION

The following is a brief description of the facilities involved in target fabrication and postirradiation processing and target irradiation. Detailed descriptions of these facilities and the processes associated with them are provided in Appendixes A through F of the NI PEIS. Also provided is a summary of the transportation required by each alternative.

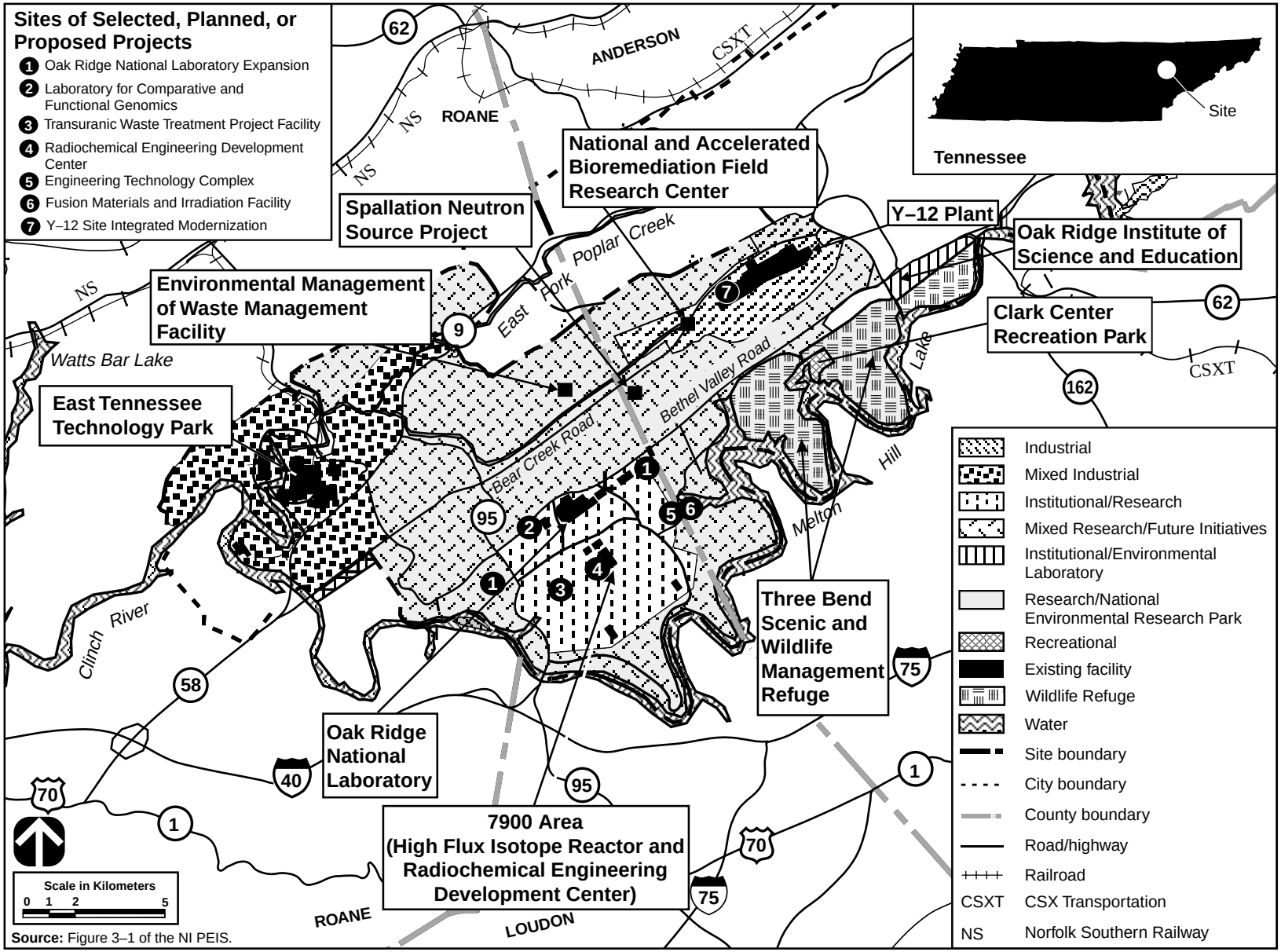
Target Fabrication and Postirradiation Processing Facilities

Radiochemical Engineering Development Center. REDC at ORNL is a companion facility to the HFIR. REDC's two buildings house heavily shielded hot cells and analytical laboratories that are used for remote fabrication of rods and targets (for irradiation in HFIR) and processing of irradiated rods and targets for the separation and purification of transuranic elements, process development, and product purification and packaging. ORNL's REDC Building 7930 is proposed for the storage of neptunium-237 in one option of the No Action Alternative. It also is proposed for the storage of neptunium-237, fabrication of neptunium-237 targets, and processing of irradiated neptunium-237 targets for two irradiation options in Alternative 1 (Restart FFTF), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and for one irradiation option in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor). REDC's current radiochemical missions would not be impacted by the addition of the proposed storage of neptunium-237, fabrication of neptunium-237 targets, and the processing of irradiated neptunium-237 targets activities. REDC would have no role in support of Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). **Figure S–1** presents a map of the Oak Ridge Reservation (ORR) that depicts the location of REDC.

REDC Building 7930 is divided into four major areas: (1) a cell complex with seven cells, six shielded and one unshielded; (2) maintenance and service areas surrounding the cell complex; (3) an operating control area; and (4) an office area adjacent to, but isolated from, the operating areas. Utility services, ventilation systems, crane and manipulator systems, and liquid-waste systems also are included. The proposed plutonium-238 processing and storage activities would require equipment installation in three main areas of the second floor of REDC Building 7930. The REDC hot cell facilities that would be used for the proposed action have never been used. The activities required for target fabrication would take place in shielded gloveboxes. (Appendix A of the NI PEIS provides a description of the target fabrication process.) The mechanical operations involved in the final target fabrication may present lesser hazards that permit them to be carried out in open boxes. Cell E would contain processing equipment to purify the separated plutonium-238 product, prepare the plutonium oxide, and transfer the oxide into shipping containers. Cell E would also contain vertical storage wells for dry storage of neptunium and other actinides.

Cell D activities would include receipt of irradiated targets, as well as target dissolution, chemical separation of neptunium and plutonium from fission products, and partitioning and purification of neptunium. Cell D also contains process equipment to remove transuranic elements from the aqueous waste streams and vitrifying waste.

Fluorinel Dissolution Process Facility. FDPF is in the Idaho Nuclear Technology and Engineering Center (INTEC), which is located northeast of the Central Facilities Area at INEEL and approximately 3.2 kilometers (2 miles) southeast of ATR. **Figure S–2** presents a map of the INEEL site that depicts the location of FDPF. FDPF is proposed for fabrication of neptunium-237 targets, and processing of irradiated neptunium-237 targets for two irradiation options in Alternative 1 (Restart FFTF), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and one irradiation option in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor).



Source: Figure 3–1 of the NI PEIS.

Figure S–1 Generalized Land Use at Oak Ridge Reservation and Vicinity

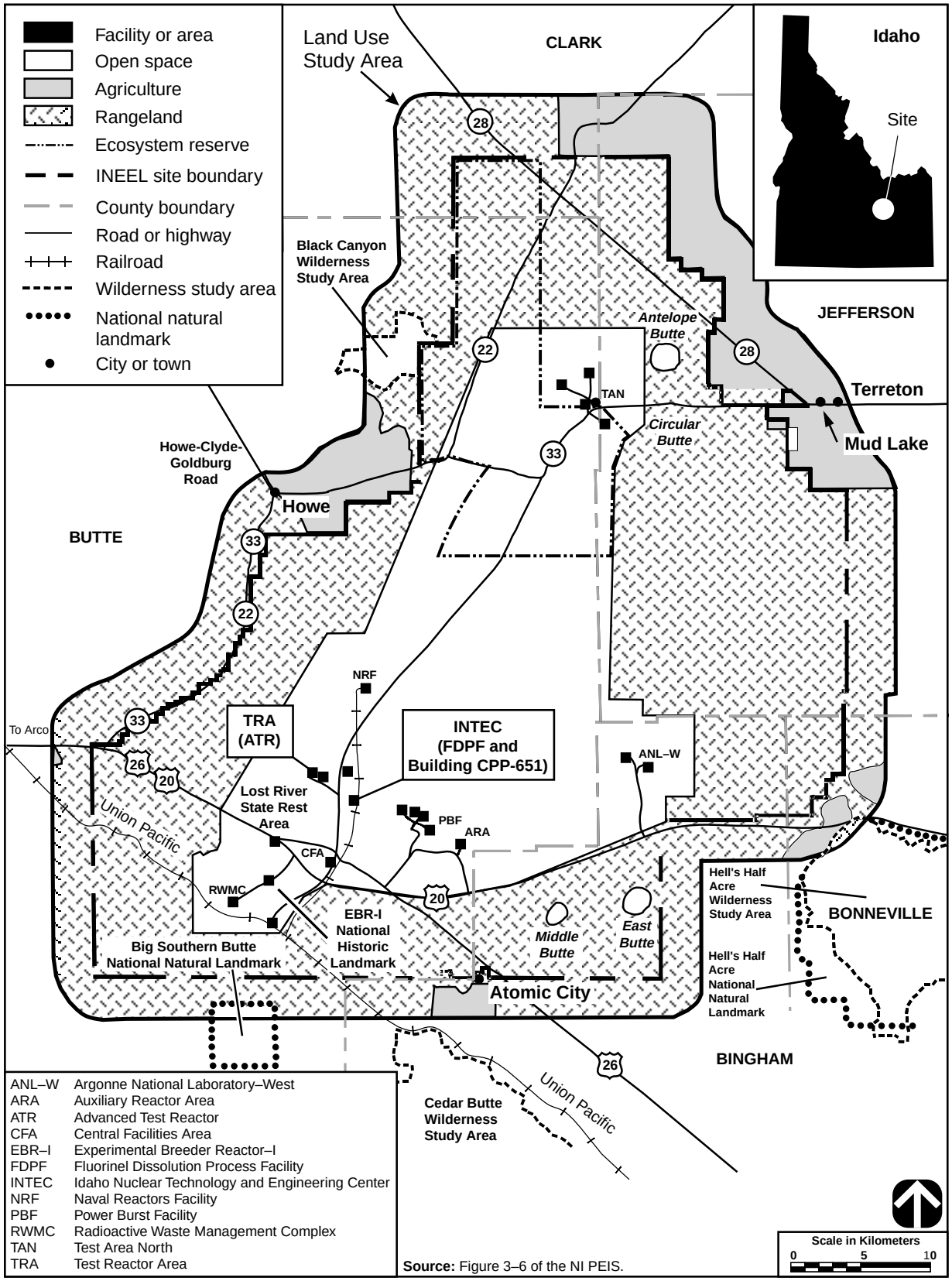


Figure S–2 Generalized Land Use at Idaho National Engineering and Environmental Laboratory and Vicinity

FDPF has no current mission. Historically, INTEC reprocessed spent nuclear fuel from U.S. Government reactors to recover reusable highly enriched uranium. After DOE announced in April 1992 that it would no longer reprocess spent fuel, reprocessing operations at INTEC ended. Two buildings at INTEC are candidate storage and processing sites for plutonium-238 production: Building CPP–651, the Unirradiated Fuel Storage Facility, and Building CPP–666, FDPF.

Building CPP–651 was originally designed for the storage of special nuclear materials to support Defense Programs and is flexible in terms of the size and shape of special nuclear materials that it can receive and store. The 100 storage positions in the vault use the existing structural barriers of Building CPP–651 (earth and concrete) and provide supplemental security protection via their in-ground concrete storage silo design. Each storage position houses a rack that holds seven highly enriched uranium product cans. Racks are raised and lowered in their storage positions via an overhead 1-ton hoist.

Building CPP–666 is divided into two parts, the Fuel Storage Facility and FDPF. The Fuel Storage Facility consists of receiving and unloading areas, a fuel unloading pool, and six storage pools for storing nuclear fuel. FDPF was designed and built to process Navy fuel via three dissolver trains. When fuel reprocessing was discontinued, uranium and hazardous materials were flushed from FDPF, and the facility is currently under consideration for new missions. FDPF consists of a large hot cell and supporting areas with a total area of approximately 3,700 square meters (40,000 square feet). The facility is divided into five levels identified by their elevation relative to ground level.

The chemical separation would take place in the FDPF cell using small centrifugal contactors installed for that purpose. Storage of neptunium-237 would be performed in Building CPP–651, which is located within 100 meters (328 feet) of FDPF. There are 100 in-ground concrete-shielded storage well positions in this vault. Each storage well contains a rack that can be modified to house cans of neptunium-237.

Fuels and Materials Examination Facility. Use of Hanford's FMEF is proposed for storage of neptunium-237 in one option of the No Action Alternative. It is also proposed for storage of neptunium-237, fabrication of neptunium-237 targets, and processing of irradiated neptunium-237 targets for two irradiation options in Alternative 1 (Restart FFTF), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and for one irradiation option in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor). In addition to the support of the plutonium-238 production mission activities in Alternative 1, FMEF would also support medical and industrial production mission and civilian nuclear energy research and development mission activities at the Hanford Site. FMEF would have no role in supporting Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). FMEF is adjacent to the west of FFTF in the 400 Area of Hanford. **Figure S–3** presents a map of Hanford that depicts the location of FMEF. FMEF was built during the late 1970s and early 1980s as a major addition to the breeder reactor technology development program at Hanford. Although it has never been used, the facility was constructed to perform fuel fabrication and development and postirradiation examination of breeder reactor fuels.

FMEF is currently being maintained in a condition suitable for a future mission. In 1998, FMEF was placed into a partial layup condition in order to reduce the cost of maintaining the facility. Many systems were shut down and most hazardous materials were removed from the building. However, FMEF is considered clean and uncontaminated because no nuclear materials have been introduced. Some critical systems remain in operation, e.g., the fire detection and protection systems. In order to avoid freezing of the fire protection water systems, limited heating and ventilation remains available. For example, the heating, ventilating, and air conditioning system has been modified to simplify its operation by clocking automatic dampers in appropriate configurations. Also, although the chillers have been laid up, including removal of the refrigerant, the chilled water system (containing an ethylene glycol-water mixture) remains available to help distribute heat within

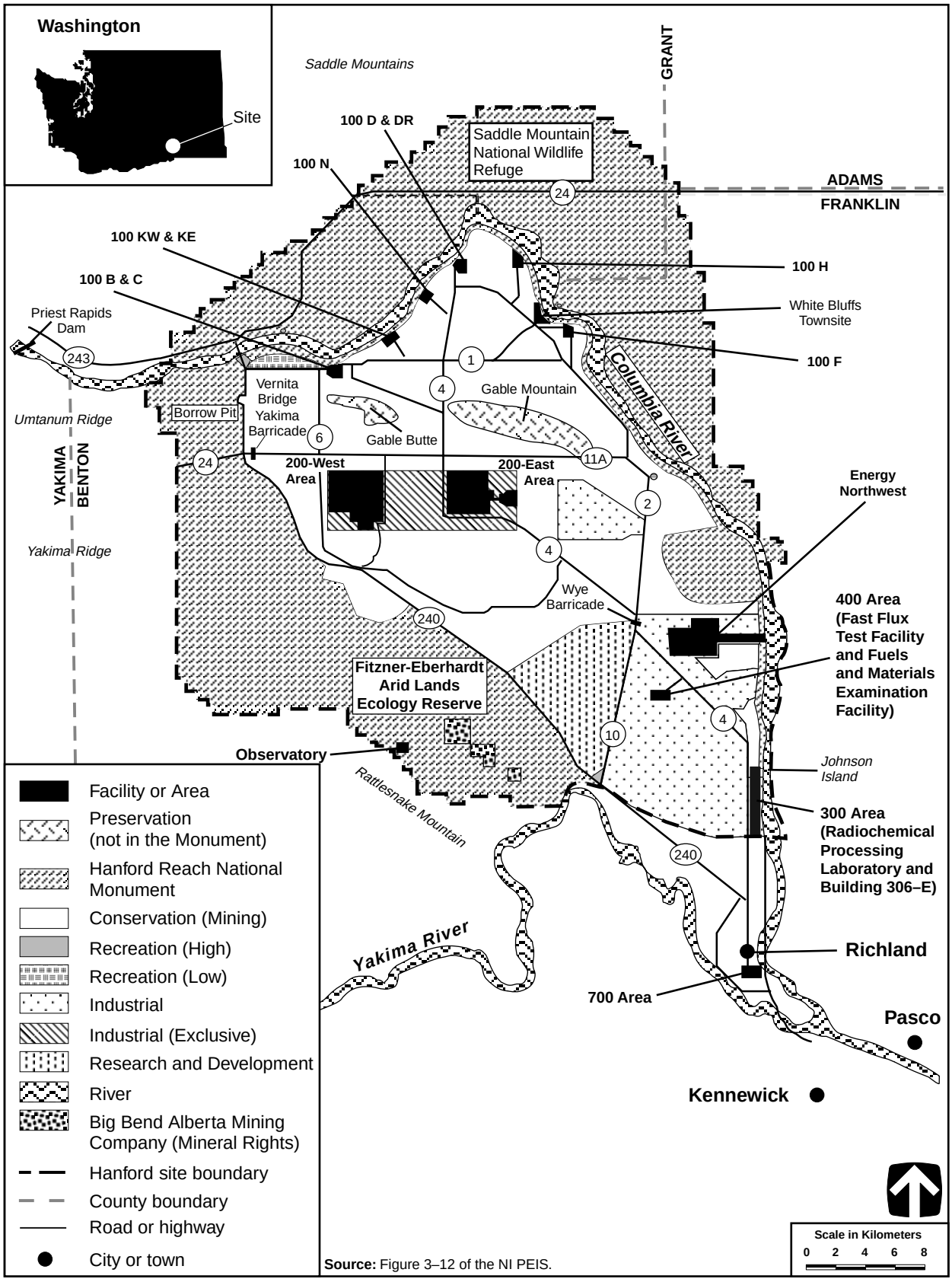


Figure S–3 Generalized Land Use at the Hanford Site and Vicinity

the building. Electrical power and lighting remain available, and the freight elevator remains in service to support routine facility walkdowns and any required maintenance. FFTF staff conducts surveillance and maintenance of FMEF.

FMEF consists of a 30-meter (98-foot) high Process Building, which has an attached Mechanical Equipment Wing on the west side and an Entry Wing on the south (front) side. The Mechanical Equipment Wing houses utility and support equipment, including water treatment equipment, air compressors, and a portion of the air conditioning equipment. The Entry Wing contains space for reactor fuel assembly (recently used as a training facility in support of Hanford's cleanup mission), lunchroom and change rooms, and heating and air conditioning equipment associated with the Entry Wing. Office space and administrative support areas are also housed on the second floor of the Entry Wing.

The Process Building is approximately 53.3 meters (175 feet) wide by 82.3 meters (270 feet) long, and extends from around 10.7 meters (35 feet) below grade to 30 meters (98 feet) above grade. Total potential operating space is approximately 17,470 square meters (188,000 square feet). The Process Building contains several large interconnected hot cells and many smaller connected hot cells. Major cranes are available, but some cranes, windows, and manipulators were not installed because construction of FMEF was halted prior to completing work on the hot cell complex. Nevertheless, the building is divided into six operating floors or levels, which are identified by their elevation relative to ground level and their primary function. The use of FMEF for neptunium-237 target material storage, target fabrication, and postirradiation processing would require the construction of a new 76-meter (250-foot) stack. The neptunium dioxide containers will be stored in specially designed storage vaults to provide secure, safe storage for the materials.

Hanford 300 Area Facilities (Radiochemical Processing Laboratory/Building 306–E). Two Hanford 300 Area facilities are proposed to support medical and industrial isotope target fabrication and postirradiation: RPL and the Development Fabrication Test Laboratory (Building 306–E). The facilities support the four irradiation options in Alternative 1 (Restart FFTF) that are not supported by FMEF. RPL/306–E would be used to support medical and industrial isotope production and civilian nuclear energy research and development activities. These activities would not impact current missions at the facilities. RPL/306–E have no role in support of the No Action Alternative, Alternative 2 (Use Only Existing Operational Facilities), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), and Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). Figure S–3 presents a map of Hanford that depicts the location of RPL/306–E.

Radiochemical Processing Laboratory: The research and development activities of the Radiochemical Processing Group are conducted at RPL in the 300 Area of Hanford. RPL consists of a central area that contains general purpose laboratories designed for low-level radioactive work, a front wing that contains office space and shops, and two annexes that provide shielded enclosures with remote manipulators for high-level radiochemical work. The facility also contains laboratories and specialized facilities designed for work with nonradioactive materials, microgram-to-kilogram quantities of fissionable materials, and up-to-megacurie quantities of radionuclides. RPL would be the primary site for fabricating the radioactive targets (i.e., targets containing radium-226 or recycled materials from previous irradiations).

Total space within RPL is 13,350 square meters (143,700 square feet), of which 4,140 square meters (44,500 square feet) are occupied by general chemistry laboratories. A recent space utilization survey of RPL indicated that 646 square meters (6,950 square feet), representing 15.6 percent of the laboratory area, are presently unoccupied. All of the occupied and nearly all of the unoccupied laboratories are functional and are fully equipped with standard utilities. Several of the laboratories, especially those used for radioanalytical work, have been renovated during the past few years. Upgrading and modernization of the equipment within the chemistry laboratories has been given a high priority during the past 2 years. During the space utilization

survey at RPL, an assessment was made of the number of fume hoods and shielded gloveboxes (including several small hot cells) that are available in the chemistry laboratories for additional programmatic work. Of the 79 functional fume hoods and 23 shielded gloveboxes, 50 fume hoods and 15 gloveboxes are available for additional work.

A special feature of RPL is the existence of two heavily shielded hot cell facilities located in annexes on the east and west sides of the building. These shielded facilities are the High-Level Radiochemistry Facility and the Shielded Analytical Laboratory. These two hot cell complexes are heavily used because they provide capabilities for conducting bench-scale to pilot-scale work with a wide variety of highly radioactive materials. Their capabilities include those required to conduct radiochemical separation and purification procedures, irradiated fuel or target sectioning and processing, metallography, physical properties testing of activated metals, thermal processing (including waste vitrification), and radioanalytical and preparatory chemistry operations.

The High-Level Radiochemistry Facility contains three large, interconnected hot cells designated as A-Cell, B-Cell, and C-Cell. Each of the three cells is 4.6 meters (15 feet) high and 2.1 meters (7.0 feet) deep. The A-Cell is 4.6 meters (15 feet) wide, and the B-Cell and C-Cell are each 1.8 meters (6.0 feet) wide. In-cell operations are performed using medium-duty electromechanical manipulators, and operators view their work through leaded-glass, oil-filled windows. Closed-circuit television cameras and videocassette recorders have been installed for detailed inspection work within the hot cells. The A-Cell and C-Cell also have overhead bridges that contain hoists with a 2,200-kilogram (4,840-pound) capacity. The hot cells are fully equipped with utilities and have shielded service penetrations at the front wall to allow insertion of special instruments. Each hot cell contains several process vessels located below the work deck that range in capacity from 4.0 to 320 liters (1.1 to 84.5 gallons). A large shielded door and a shielded double-door transfer port located in the rear wall of the cell provide access to each hot cell in the High-Level Radiochemistry Facility. Cask payloads weighing up to 2,200 kilograms (4,840 pounds) can be transferred into and out of the hot cells using a bridge crane located in the canyon behind the cells.

The Shielded Analytical Laboratory contains six interconnecting hot cells, each of which is 1.7 meters (5.5 feet) wide, 1.7 meters (5.5 feet) deep, and 2.9 meters (9.5 feet) high. Each hot cell is equipped with a pair of medium-duty manipulators. Turntables built into the rear walls of the hot cells provide rapid transfers of radioactive samples into and out of the cells. The Shielded Analytical Laboratory hot cells are equipped to perform a wide variety of analytical chemistry operations with highly radioactive samples.

Building 306–E: Building 306–E was constructed in 1956 as part of the nuclear material production program at Hanford, and was used to develop the co-extrusion process for N-Reactor fuel. Major upgrades and renovations were completed in the late 1960s and early 1970s to support the civilian reactor development program (Liquid Metal Reactor Program–FFTF). The building has 4,273 square meters (46,000 square feet) of floor space, with a 36.5-meter by 61-meter by 6.4-meter high (120-foot- by 200-foot- by 21-foot-high) bay containing three 10-ton, one 5-ton, and one 1.5-ton cranes. The facility has electron beam laser welding, certified nondestructive testing, a 3.7-meter by 3.7-meter (12-foot by 12-foot) vertical assembly and test station with 24.4-meter (80-foot) hook height, a machine shop, and an instrument development laboratory.

The building is serviced by three 1,416-cubic-meter-per-minute (50,000-cubic-feet-per-minute) supply units complete with filters, steam coils and spray chambers. Two of the units have refrigeration coils for summer time cooling. Two ceiling mounted 1,012-cubic-meter-per-minute (35,750-cubic-feet-per-minute) recirculation fans with freon compressors provide additional cooling and air movement. Fume hoods have individual exhaust fans. Chemical and acid tanks exhaust through two 340-cubic-meter-per-minute (12,000-cubic-feet-per-minute) fume scrubbers to a 12.2-meter-high 7.6-centimeter diameter (40-foot-high 3-inch diameter) stainless steel exhaust stack. Equipment exhaust collects through a grid that leads to two 566-cubic-meter-per-

minute (20,000-cubic-feet-per-minute) exhaust fans. Plastic hoods and duct work are provided for highly corrosive service. Major equipment includes three industrial x-ray machines, a 6-kilowatt Hamilton Standard electron beam welder, five open face hoods, two inert gas welding chambers and one electrolytic cutoff saw.

Utilities include hot and cold water, deionized water, propane, helium, compressed air, argon, steam, and sanitary and process sewers as well as a special acid drain and neutralizing tank. Normal power is provided by a 1500-kilovolt-ampere transformer with 15-kilovolt-ampere backup power from an adjoining building, and a 30-kilovolt-ampere emergency transformer. The building is protected by redundant emergency alarm systems, fire gongs, and an evacuation siren.

New Support Facility. A new generic support facility would have the mission of preparing medical and industrial isotope targets for irradiation, processing exposed targets, and housing the materials research and development activities in association with Alternatives 3 and 4. Siting of the generic support facility for medical and industrial isotope production would require that the facility be located in the same general vicinity (0.2 to 20 kilometers [0.07 to 12.4 miles]) as the new irradiation facility (accelerator or reactor). Colocation with the irradiation facility would be needed to process some irradiated target materials promptly after removal from the reactor/accelerator. Colocation would also minimize transportation time, which is desirable because some isotopes have short half-lives. Although the facility could be located within the irradiation facility security protection area, the lack of a defense mission and the lack of a fissile material presence in the generic support facility indicates that a high level of physical protection would not be warranted.

The generic support facility mission would be accommodated by a one-story, 3,345-square-meter (36,000-square-foot) above-grade building with a 1,490-square-meter (16,000-square-foot) basement area under a portion of the footprint. The facility is designed around a center area containing the highest-risk activities and the material inventories requiring the highest level of engineered controls. Irradiated materials in casks or other shielded transport containers would enter a loading dock with a straight-line access to the primary facility hot cell. The hot sample entry area would be a high bay area with a high floor loading area between the loading dock and the hot cell access port. This configuration would allow transport cask access to the hot cell. In addition, an overhead hoist would be available to facilitate handling of materials and devices in the proximity of the hot cell.

The hot cell would accept high-radiation-level samples or those difficult to shield or manipulate (e.g., reactor core components containing samples). The hot cell would have access to a conveyor that can remotely transport samples to the hot process laboratories. In addition, samples from the hot cell could be transferred to the hot research and development laboratory gloveboxes for detailed analysis and testing. Hot cell manipulators would be located on both the operating gallery and the research and development sides of the hot cell. Adjacent to that would be the central receiving station for all other radioactive and short-exposure samples not in the reactor core components. This area, while not a hot cell, would provide personnel protection (i.e., shielding and controlled ventilation) for preliminary sample preparation and examination. It would also provide interim irradiated sample storage prior to delivery to the designated processing laboratory. When needed, samples would be transported remotely to the processing laboratories by the conveyor system.

Samples requiring a lesser degree of control would be distributed for processing throughout the remaining process laboratory wing. After processing, the radiopharmaceuticals would be either stored or packaged and shipped immediately to offsite vendors. Radioactive waste would be packaged and stored for eventual disposal. Those materials containing short-lived isotopes would be delivered to a decay/holding room so that, given appropriate decay time, they could be disposed of without a radioactive component. The process and research and development areas would be considered radiologically controlled areas, but no routinely occupied areas would require control as contaminated radiological areas. Radioactive contamination would be controlled at the hood or glovebox face. Due to this configuration, protective clothing and change rooms

would be needed only for occasional maintenance activities when temporary radiological areas are established. Cold sample (nonradioactive) preparation would be accomplished in a set of three large laboratories where radiological conditions are not anticipated. Completed samples would be stored in an adjacent room along with raw sample materials (nonradioactive). Radioactive sample preparation and irradiated material recycling activities would be conducted in one of the laboratories adjacent to the conveyor. Irradiated research and development samples introduced into the hot cell could be processed or examined using manipulators within the hot cell. Samples could also enter the research and development suite of lab rooms through the hot cell port into a hot cell or glovebox. From there, they could be moved to additional research and development laboratory rooms within a controlled environment for detailed analysis and testing.

Target Irradiation Facilities

Fast Flux Test Facility. FFTF is proposed to support the three proposed missions: (1) plutonium-238 production, (2) medical and commercial isotope production, and (3) civilian nuclear energy research and development.

FFTF is a 400-megawatt thermal, liquid-cooled (sodium) nuclear test reactor that is owned by DOE and is at the Hanford Site in southeastern Washington State near Richland, Washington. Figure S–3 presents a map of Hanford that depicts the location of FFTF. Following extensive testing, FFTF was started in April 1982. During its operation, FFTF successfully tested advanced nuclear fuels, materials, components, operating protocols, and reactor safety designs. FFTF also produced a wide variety of medical isotopes and made tritium for the U.S. fusion research program.

FFTF was originally designed and operated as a science test bed for U.S. liquid metal fast reactor programs. These programs, which were canceled in 1993, were key elements both in closed fuel cycle and actinide waste disposition technology development. In December 1993, DOE decided not to operate FFTF due to a lack of economically viable missions at that time. In accordance with NEPA, DOE published an environmental assessment (EA) and Finding of No Significant Impact (FONSI) for the shutdown and deactivation of FFTF in May 1995 (DOE 1995a). The EA contained an evaluation of the environmental impacts associated with the actions necessary to place FFTF in a radiologically and industrially safe shutdown condition suitable for long-term surveillance and maintenance before final decontamination and decommissioning.

The FFTF complex includes the reactor, as well as equipment and structures for heat removal, containment, reactor safety and shutdown systems core component handling and examination, fuel off-loading and storage, utilities, and other essential services. There are 100 systems supporting various functions of FFTF during operations. The central structure of FFTF is the reactor containment building, an all-welded cylindrical steel structure 41 meters (135 feet) in diameter and 57 meters (187 feet) high. The reactor is located below grade in a shielded cell in the center of the containment structure. Heat is removed from the reactor by circulating liquid sodium under low pressure through three separate closed primary piping loops, which include pumps, piping, and intermediate heat exchangers. These loops are located within inerted cells (cells filled with inert gases) within the containment structure. Three secondary sodium loops transport reactor heat from the intermediate heat exchangers to the air-cooled tubes of the dump heat exchangers. From there, the heat dissipates into the atmosphere through the forced draft dump heat exchanger. [Commercial nuclear power reactors use reactor heat to create steam, which turns a turbine to produce electricity. FFTF, however, does not generate electricity.]

FFTF has demonstrated its capability to function as a nuclear science and irradiation services user facility. It has five distinct features: size, flux, test evaluation and irradiation capabilities, fuel type, and coolant type. In combination, these features provide a multipurpose facility suitable for medical and industrial isotopes production, plutonium-238 production, and civilian nuclear energy research and development purposes.

Although FFTF was used primarily to evaluate reactor fuels and different fuel assembly materials during its 10 years of operation, the reactor facility has also supported large and varied test programs for industry, nuclear energy (domestic and international), medical isotope applications and research, space nuclear power, and fusion research programs.

FFTF is currently defueled and is being maintained in a standby condition. Seventy-seven of the 100 systems are operational; the other 23 are in a recoverable standby state. System integrity and configuration control are being maintained. The Main Heat Transport System is being operated at approximately 200 °C (400 °F) to keep the sodium coolant in the reactor liquefied and circulating. If a decision were made to restart FFTF, several equipment upgrades are planned to return systems to operation, improve reliability, conform to current standards, improve efficiency, and minimize waste. Most of the required modifications would consist of either mechanical equipment upgrades or replacement of outdated control and computer systems.

The NI PEIS postulates that FFTF would operate at a nominal power level of 100 megawatts, one quarter of the reactor design power level, to meet the irradiation requirements of the proposed missions. Periodic increases in power level between 100 and 400 megawatts may be required to support civilian nuclear energy research and development activities. Operating FFTF at a nominal 100-megawatt power level extends the reactor life and significantly reduces the generation rate of spent fuel. FFTF is currently designed to operate using mixed oxide fuel, however, it can also be operated using highly enriched uranium fuel.

There are eight locations available in the FFTF reactor core that are termed Open Test Assembly positions. These positions are located under spool pieces in the reactor head and allow the installation of 38-foot-long assemblies that extend from the reactor head down to the reactor core. Within the 82 active core locations, there are up to 20 or more additional locations that could contain a standard length (3.6-meter or 12-foot) test assembly. In addition to the test locations within the active fueled region of the core, there are 108 locations available in the surrounding reflector region where other tests could be inserted.

The FFTF core would be modified to include an array of target assemblies and rapid radioisotope retrieval systems capable of producing a number of long- and short-lived isotopes for medical and industrial applications and plutonium-238 for space power applications. In addition, reactor space would be provided for research and development test articles.

Fifteen plutonium-238 production targets would be included in the reflector region with an annual production rate of 5 kilograms. The residence time for these targets would be three 100-day cycles with five assemblies being harvested at the end of each cycle.

Long-Term Irradiation Vehicles would be used to irradiate targets to produce long-lived isotopes, installed in the reactor during normal refueling operations, and handled using standard FFTF handling equipment. The Long-Term Irradiation Vehicle would consist of a bundle of target pins installed inside a nozzle, duct, and handling socket assembly similar in appearance to an FFTF 3.6-meter (12-foot) long fuel assembly. Rapid radioisotope retrieval systems would be installed in selected Open Test Assembly positions for the production of short-lived isotopes. There would be a maximum of eight systems in the core.

Advanced Test Reactor. ATR is a light-water-cooled and moderated reactor with a design thermal power of 250 megawatts that is owned by DOE and is in the Test Reactor Area in the southwest portion of INEEL. Figure S–2 presents a map of INEEL that depicts ATR's location. ATR would continue to operate and meet its current mission requirements including naval reactor research and development, medical and industrial isotope production, and civilian nuclear energy research and development activities, at its current operating levels under the No Action Alternative, Alternative 1 (Restart FFTF), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF

[with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities) when it is not providing irradiation services in support of the plutonium-238 production mission. When ATR is supporting the plutonium-238 production mission, it would fully support its primary mission, naval reactor research and development; however, it would support the medical and industrial isotope production and civilian nuclear energy research and development activities to the extent possible within its current reactor operating levels. Consideration must be given to the need to maintain appropriate levels of neutron flux to support ATR's primary mission. Neutron flux levels can be impacted by the placement of targets, such as neptunium-237 targets for production of plutonium-238, in the reactor core. The production planning assumption for ATR is from 3 kilograms (6.6 pounds) of plutonium-238 per year (if used in conjunction with HFIR) to 5 kilograms (11 pounds) of plutonium-238 per year (if ATR were used alone). Thus, ATR alone could meet the program goal of up to 5 kilograms (11 pounds) per year and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission.

Special features of ATR include high neutron flux levels (ranging from 1×10^{15} neutrons per square centimeter per second in the flux traps to 1×10^{13} neutrons per square centimeter per second in the outer reflector positions) and the ability to vary power to fit different experiment needs in different test positions. The primary user of ATR is the U.S. Naval Nuclear Propulsion Program. A variety of other users include foreign and domestic government programs, a commercial isotope production company, industrial customers, and research and development interests. A number of support facilities are important to the operation of ATR. Among these are the Advanced Test Reactor Critical Facility, which is used to baseline experiment impacts to ATR flux profile, and the Nuclear Materials Inspection and Storage facility, which is used to receive, store and inspect reactor fuel prior to its placement in ATR.

The reactor, its primary coolant system, control room, and much of its auxiliary and experimental support equipment are in Test Reactor Area Building 670. ATR began operation in 1967 and is expected to continue operating for several decades. The reactor vessel is entirely stainless steel and the core internals are replaced every 7 to 9 years. Buildings and structures in other parts of the Test Reactor Area provide additional support functions.

ATR is currently operating at approximately 140 megawatts or less. ATR operates with highly enriched uranium fuel. Typical operating cycles are 42 days or 49 days at power followed by a 7-day outage for refueling and changeout of experiments and isotope production targets. The core is 1.2 meters (4 feet) high and is surrounded by a 1.3-meter-diameter (4.25-foot-diameter) beryllium reflector. Beryllium is an excellent neutron reflector and is used to enhance the neutron flux essential to a test reactor. ATR has nine flux traps in its core and achieves a close integration of flux traps and fuel by means of a serpentine fuel arrangement. When viewed from above, the ATR fuel region resembles a four-leaf clover. The four flux traps positioned within the four lobes of the reactor core are almost entirely surrounded by fuel, as is the center position. Four other flux trap positions between the lobes of the core have fuel on three sides. The ATR's unique control device design permits large power shifts among the nine flux traps. Testing can be performed in test loops installed in some flux traps with individual flow and temperature control or in reflector irradiation positions with primary fluid as coolant. The curved fuel arrangement brings the fuel closer on all sides of the test loops than is possible in a rectangular grid.

Of the nine flux traps, five are configured with pressurized-water loops that allow for individual temperature, pressure, flow, and chemistry controls. The five test loops are used by the Naval Reactors program. Of the remaining four flux traps, one is dedicated to the Naval Reactors program, one is used for isotope production, one is used for low-specific-activity cobalt production, and the fourth has recently had the Irradiation Test Vehicle installed. The Irradiation Test Vehicle can be described as three small pressurized-gas test loops. The use of one of these three test loops was recently purchased by a British corporation; negotiations for use of the other two are currently under way.

In addition to the primary flux trap irradiation positions, there are some 70 irradiation positions in the beryllium reflector (and aluminum support structure) that are available for experiment irradiation and isotope production. These position diameters range from 1.6 centimeters (0.625 inch) to 12.7 centimeters (5.0 inches) with thermal neutron flux levels ranging from 1×10^{15} neutrons per square centimeter per second to 1×10^{13} neutrons per square centimeter per second.

INEEL has privatized the production of medical and industrial isotopes through contracting with a commercial entity, which specializes in producing isotope targets for irradiation in ATR and processing and distributing commercial-grade isotopes to its customers. Prior to commercialization, INEEL's isotope production operations were limited in types and quantities. Since the start of commercial activities, production has expanded. Incremental investments have been identified for ATR that would make it a more versatile and capable reactor for isotope production. Commercial companies are in the discussion phase of investing in ATR to install an isotope shuttle (or rabbit) system for the production of short-lived radioisotopes. Many of these short-lived radioisotopes are expected to be in growing demand for various cancer therapies.

High Flux Isotope Reactor. HFIR is a beryllium-reflected, light-water-moderated and -cooled reactor operating at a thermal power level of 85 megawatts. HFIR is owned by DOE and is in the 7900 Area in the southern portion of ORR. Figure S–1 presents a map of ORR that depicts the location of HFIR.

HFIR would continue to be operated to meet the primary mission of neutron science based research for DOE's Office of Science. In addition, medical and industrial isotope production and civilian nuclear energy research and development activities would be performed on a not-to-interfere basis at the current operating level in the No Action Alternative, Alternative 1 (Restart FFTF), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities). When HFIR is supporting the plutonium-238 production mission, it would fully support its primary mission, but would support the medical and industrial isotope production and civilian nuclear energy research and development activities to the extent possible within the current reactor operating levels. Consideration must be given to the need to maintain appropriate levels of neutron flux to support HFIR's primary mission. Neutron flux levels can be impacted by the placement of targets, such as neptunium-237 targets for the production of plutonium-238, in the reactor core. Under the planning assumptions for plutonium-238 production, HFIR could only produce from 1 to 2 kilograms (2.2 to 4.4 pounds) per year without impacting ongoing missions. As the program goal is to achieve a production rate of 5 kilograms (11 pounds) per year, production at HFIR would need to be augmented by the use of ATR to meet this goal. HFIR and ATR together could meet the program goal of up to 5 kilograms (11 pounds) per year, and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission.

HFIR was originally designed as both an isotope production and a research reactor with a thermal flux of 3 to 5×10^{15} neutrons per square centimeter per second and a full power level of 100 megawatts-thermal (3.4×10^{8} British thermal units per hour). It is currently operating at a maximum authorized power level of 85 megawatts-thermal (2.9×10^{8} British thermal units per hour) to extend the useful life of the reactor. Many experiment-irradiation facilities were provided for in the original design and several others have been added. The primary mission of HFIR is neutron science research. Isotope production is done on a not-to-interfere basis.

HFIR transfers its primary coolant heat load to secondary coolant through heat exchangers for dissipation to the atmosphere by an induced-draft cooling tower. The reactor uses highly enriched uranium and aluminum-clad plate fuel. The reactor vessel itself is immersed in a pool in a poured-concrete reactor building that also houses the primary coolant pumps and heat exchangers, a spent fuel pool, and experiment areas. The control and water wing of the reactor building contains the reactor control room; relay and amplifier areas; heating and

ventilating equipment; pool and fire alarm equipment; instrumentation systems; and office and support rooms. A separate electrical building adjacent to the reactor building contains switchgear, diesel generators, and associated transformers that connect the facility to offsite power. The reactor building is essentially airtight and provides dynamic confinement. A special hot exhaust system exhausts air from potentially contaminated areas of the building through filters (two high-efficiency particulate air and two charcoal filters) before being released to the atmosphere through a 76-meter (250-foot) stack. The stack serves as the exhaust point for both HFIR and REDC at ORNL.

After the reactor completed 17.2 full-power years of its 20 full-power year design life in November 1986, several measures were taken to extend the useful life of the reactor, including reducing the 100 megawatts-thermal (3.4×10^8 British thermal units per hour) rated power level to 85 megawatts-thermal (2.9×10^8 British thermal units per hour); adjusting the primary coolant temperature and pressure; conducting periodic hydrostatic tests; establishing an irradiation embrittlement surveillance program; and installing an emergency depressurization system. Subsequent life extension programs can enable HFIR to provide support during the total 35-year evaluation period for operations.

Experiment-irradiation facilities available include (1) the hydraulic tube facility, located in the very high flux region of the flux trap, which allows for insertion and removal of irradiation samples while the reactor is operating; (2) 30 target positions in the flux trap, which normally contain transuranium production rods but which can be used for the irradiation of other experiments (two are instrumented target positions provided by a recent modification); (3) six peripheral target positions located at the outer edge of the flux trap; (4) numerous vertical irradiation facilities of various sizes located throughout the beryllium reflector; (5) two pneumatic tube facilities in the beryllium reflector, which allow for insertion and removal of irradiation samples while the reactor is operating for activation analysis; (6) four horizontal beam tubes, which originate in the beryllium reflector; and (7) four slant access facilities, called "engineering facilities," located adjacent to the outer edge of the beryllium reflector. In addition, spent fuel assemblies are used for gamma irradiation in the gamma irradiation facility in the reactor pool.

The reactor core assembly is contained in a 2.44-meter (8-foot) diameter pressure vessel located in a pool of water. The top of the pressure vessel is 5.18 meters (17 feet) below the pool surface, and the reactor horizontal midplane is 8.38 meters (27.5 feet) below the pool surface. The control plate drive mechanisms are located in a subpile room beneath the pressure vessel. These features provide the necessary shielding for working above the reactor core and greatly facilitate access to the pressure vessel, core, and reflector regions.

The neutron flux within HFIR is primarily a thermal neutron flux ranging from approximately 2×10^{15} neutrons per square centimeter per second in the flux trap to approximately 4×10^{14} neutrons per square centimeter per second in the outer regions of the beryllium reflector. Specially designed neutron beam tubes provide access to neutrons that supply intense neutron beams to various specialized instruments used for neutron scattering research.

ORNL produces a variety of medical isotopes using the HFIR for irradiation and various hot cell and glovebox facilities for target fabrication and final product purification. The nine hydraulic tube positions in the central high flux region permit the insertion and removal of targets at any time during the operating cycle (22 to 24 days) and have traditionally represented a major site for the production of medical radioisotopes. In addition to providing radioisotopes for extramural research and development and commercial applications by distribution through the DOE Isotope Production and Distribution Program, there are medical radioisotope research and development programs at ORNL that depend on the availability of HFIR-produced radioisotopes.

Commercial Light Water Reactor. A CLWR would continue to operate and meet its primary mission requirement, providing steam for the generation of electrical power in the No Action Alternative, Alternative 1

(Restart FFTF), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities) when it is not providing irradiation services in support of the plutonium-238 production mission. When the CLWR is supporting the plutonium-238 production mission, it would still fully support its primary mission. The production planning assumption for the generic CLWR is 5 kilograms (11 pounds) per year of plutonium-238 or 7.5 kilograms (16.5 pounds) per 18-month operating cycle. Thus, the CLWR alone could meet the program goal of up to 5 kilograms (11 pounds) per year and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission. The use of a CLWR for the medical and industrial isotope production mission and the DOE civilian nuclear energy research and development mission were not considered practical.

A typical pressurized water reactor core consists of 170 to 200 fuel assemblies arranged in the reactor vessel in an approximately cylindrical pattern. Most pressurized water reactors operating in the United States are licensed to operate at thermal power levels of 2,500 to 3,500 megawatts (8.5×10^9 to 1.2×10^{10} British thermal units per hour) for net station electrical outputs of 800 to 1,200 megawatts electric (2.7×10^9 to 4.1×10^9 British thermal units per hour).

The nuclear steam supply system powered by the pressurized water reactor is generally arranged as two heat transport loops, each with two primary coolant circulating pumps and one steam generator in which the primary coolant dissipates heat generated in the reactor core to the secondary fluid in the steam generator. In addition to serving as a heat transport medium, the primary coolant also serves as a neutron moderator and reflector and as a solvent for the soluble boron used in chemical reactivity control. All nuclear steam supply system components are designed to withstand the effects of earthquakes and loss-of-coolant accidents.

The containment for a pressurized-water reactor plant consists of two structures: (1) a steel containment vessel and (2) a reinforced-concrete shield building. The containment, including all of its penetrations, is a low-leakage steel structure designed to withstand a postulated loss-of-coolant accident and to confine a postulated release of radioactive material. It houses the reactor pressure vessel, reactor coolant piping, pressurizer, pressurizer quench tank and coolers, reactor primary coolant pumps, steam generators, core flooding tanks, and letdown coolers. Safety systems directly associated with this vessel include the containment spray system, the containment air cooling system, and the containment isolation system. An annular space is provided between the wall of the containment vessel and the shield building. Overhead clearance from the dome of the shield building is also provided.

The shield building itself is a concrete structure surrounding the containment that is designed to provide biological shielding during both normal operations and hypothetical accident conditions. The shield building enables the collection and filtration of fission product leakage from the containment following a hypothetical accident by means of its emergency ventilation system. In addition, the shield building provides environmental protection for the containment from adverse atmospheric conditions and external missiles (e.g., tornado debris).

All fuel assemblies are identical in mechanical construction and are interchangeable in any core location. The basic fuel assembly is normally composed of 208 fuel rods, 16 control rod guide tubes, and one centrally located position for instrumentation, all within a 15×15 position square array. The fuel assembly is approximately 20.3×20.3 centimeters (8×8 inches) in cross section and has an overall length of 419 centimeters (165 inches).

The neptunium-237 targets can be placed in numerous locations within the reactor core region (i.e., fuel assembly region) and outside the reactor core region to be irradiated for the production of plutonium-238. Three potential target arrangements were considered for evaluation in the NI PEIS: (1) all targets located in the center fuel assembly position in the reactor core, (2) all targets distributed within locations in the reactor

core, and (3) all targets distributed outside the reactor core region. The center fuel assembly position was selected for evaluation in the NI PEIS because it was assumed that this would be the worst-case location during postulated beyond-design-basis accident conditions. This assumption conservatively postulated that during a beyond-design-basis core disruptive accident, temperatures in the center fuel assembly position would reach levels that would fail the cladding on all of the neptunium-237 targets located in that position, resulting in worst-case releases.

The substitution of target rods for fuel rod positions in the center fuel assembly would only minimally impact reactor operations. The fuel rods located in the center fuel assembly position would normally not be fresh fuel (i.e., fuel inserted within the first 18-month operating cycle in the reactor); instead, they would be in their second or third operating cycle. The normal power distribution within the core and reactor coolant flow and its distribution within the core would remain within existing technical specification limits.

New Accelerator(s). One or two new accelerators would be constructed and operated in Alternative 3 (Construct New Accelerator[s]). Preconceptual designs have been developed for a low-energy accelerator and a high-energy accelerator for evaluation in the NI PEIS. The low-energy accelerator would support the medical and industrial isotope production missions and the civilian nuclear energy research and development mission. This could effectively be accomplished with accelerator energies in the range of 30 to 70 million electron volts. The high-energy accelerator design would support the plutonium-238 production mission and the civilian nuclear energy research and development mission. An accelerator with an energy level of 1,000 million electron volts is required to support the plutonium-238 and civilian nuclear energy research and development missions.

The preconceptual design of the high-energy accelerator presented in Appendix F of the NI PEIS focused on supporting the plutonium-238 production mission. Although not analyzed in the NI PEIS, the design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce "tuneable" neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and materials interactions.

The accelerator(s) would be constructed and operated at one or two existing DOE sites. The low-energy accelerator would be located on the same DOE site as the new support facility or at a DOE site with an existing support facility. The high-energy accelerator could be located at a different DOE site. Alternative 3 site selection is not evaluated as part of the NI PEIS.

Because Alternative 3 is evaluated at a generic DOE site, no credit was taken for any existing support infrastructure at the site(s), and it was postulated that a new support facility would be required to support operation of the low-energy accelerator and its missions and the high-energy accelerator civilian nuclear energy research and development missions if both accelerators are located on the same site. While this approach bounds the environmental impact assessment for the implementation of Alternative 3, it overstates the impacts because the NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site(s). In the event that Alternative 3 or the low-energy accelerator alone is selected in the Record of Decision for subsequent consideration, follow-on NEPA reviews would evaluate potential locations for either both or one of the accelerators. It is unlikely that DOE would consider locating the new low-energy or high-energy accelerator on a DOE site that does not have an existing infrastructure capable of supporting all or most of the mission requirements. To determine the environmental impacts if Alternative 3 is implemented at a site with adequate support infrastructure, the environmental impacts for the construction of the support facility could be subtracted from the environmental

impacts of Alternative 3 as presented in the NI PEIS. Section 4.5 of the NI PEIS presents the environmental impacts from construction and operation of the new support facility separately.

Low-Energy Accelerator: Three low-energy accelerator options would be available for the production of medical and industrial isotopes and to support nuclear energy research and development: (1) a high-current proton linear accelerator, (2) a multiparticle cyclotron, or (3) a proton-only cyclotron. The proton-only cyclotron would have distinct technical advantages over the other two options and is described further in the section that follows.

The proton-only cyclotron can be either a positive proton or negative ion type and is referred to as a proton cyclotron H^+ or proton cyclotron H^-. The alternative of a positive proton cyclotron would offer lower vacuum requirements and, with the latest technology, high-extraction efficiency can be achieved. But obtaining variable energy output would be complicated; extraction can be into only a single port and splitting the beam would require a complicated septum magnet. In comparison, the negative ion cyclotron would offer a continuous beam with high-current capacity using very simple high-efficiency extraction, a simple method to vary the particle energy, and the possibility of simultaneous irradiation of two different target arrays at different energies. The high-extraction efficiency would be achieved simply by passing the negatively charged beam through a thin foil that strips the electrons from the ion, creating a positive proton. The proton would be directly ejected from the machine by the existing magnetic field with high efficiency (greater than 98 percent). This feature would be important to minimize the activation of the cyclotron structure and thus reduce radiation exposure to the operational staff.

A high-beam current would be advantageous because more products could be prepared in a shorter time. In addition, a much higher specific-activity radioisotope could be prepared at the higher-beam current of the cyclotron. Specific activity is often a critical parameter in many nuclear medicine applications, including research and clinical use. The cyclotron can also continuously tune the beam energy, which would be an advantage for research. The ability to tune the energy with precision can also help achieve high-purity isotope production by avoiding energies where impurity isotopes would be readily co-produced. These are important advantages for flexibility in research isotope production and are within the capabilities of commercially proven technology.

A new building, with a 43-meter (140-foot) by 43-meter (140-foot) footprint, would be constructed to house the cyclotron and the four beam lines. The walls of the facility would be 4.6 meters (15 feet) thick behind the target stations to minimize the neutron flux outside the building. The walls surrounding the cyclotron itself would be 3 meters (10 feet) thick. The mazes throughout the building in general would have walls 1.5 meters (5 feet) thick, so that the total thickness surrounding the cyclotron area would be 3 meters (10 feet). The beam would be diverted to the four target stations by switching magnets located in the cyclotron vault. The beam would be directed through focusing and steering magnets to the target. In the isotope production beam line (northwest cave), the targets would be installed and removed vertically from a hot cell, which would be located on the second floor directly above the target station. The power supplies for the magnets would be housed with the power supplies for the cyclotron. The mechanical equipment for cooling water would be housed in a shielded mechanical room adjacent to the cyclotron vault. Recirculating water for cooling of the targets and systems that could contain potentially radioactive material would be separated to prevent cross-contamination. These systems would be contained in mechanical equipment rooms near the respective target station. Piping would be contained in waterproof trenches with leak detection.

High-Energy Accelerator: In accelerator production of plutonium-238, an energetic beam of protons generated by a linear accelerator would be transported to a heavy metal target where spallation neutrons would be produced and moderated in a surrounding blanket. The blanket containing neptunium-237 would capture the slowed neutrons to produce plutonium-238 through the same nuclear sequence that occurs in a reactor. The

accelerator would be housed in a concrete tunnel, buried below ground to provide radiation shielding for operating personnel. A building housing radio frequency power systems and other equipment used to drive, monitor, and control the accelerator would be located above ground close to the accelerator tunnel. The target/blanket assembly would be housed inside a steel and concrete shield located within a multistory building that would contain appropriate service equipment. At the target, the small-diameter proton beam transported magnetically from the accelerator would be converted to a much larger cross section by a beam expander to reduce the power density to acceptable levels for the target cooling systems.

A source of neutrons produced by an accelerator can be used to produce plutonium-238 from neptunium-237 feedstock through the capture and decay nuclear processes. A 1,000-million-electron-volt proton beam produced by a radio frequency linear accelerator would bombard a heavy metal (uranium-238) target, with each proton producing about 40 neutrons.

A very preliminary target/blanket design has been developed for scoping purposes, based on the architecture employed in the accelerator production of tritium target/blanket design. It would use uranium-238 (cooled by heavy water [D_2O]) as the neutron-production target. The target would be surrounded by a blanket of neptunium-237 in a dilute mixture of aluminum and water coolant. Enclosing the blanket would be a beryllium reflector.

To meet the plutonium-238 production goal of 5 kilograms (11 pounds) per year, the high-energy accelerator facility would conduct three 4-month production campaigns. Each campaign would be divided into 100 days of production and 21 days for recycling the production blanket. A 90 percent plant availability during the scheduled operating periods is assumed. Based on operating experience at the Los Alamos Neutron Science Center Linear Accelerator, the 90 percent plant availability should be achievable.

The preconceptual design of the high-energy accelerator presented in Appendix F of the NI PEIS focused on supporting the plutonium-238 production mission. While not evaluated in the NI PEIS, the design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce "tuneable" neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and materials interactions. The accelerator designs for Alternative 3 were developed to a level of detail that was adequate to assess the environmental impacts associated with the construction and operation of the proposed facilities and the technical feasibility of meeting the mission objectives. In the event that the NI PEIS Record of Decision selects Alternative 3, DOE would prepare conceptual, preliminary, and detailed designs and optimize the facility designs to accomplish the stated missions. Additional NEPA review would be required for site selection and to evaluate the environmental impacts of integrating the more refined accelerator designs with the existing site infrastructure(s).

New Research Reactor. A new research reactor would be constructed and operated in Alternative 4 (Construct New Research Reactor). A preconceptual design for a new research reactor was developed to meet the following DOE missions: (1) producing medical and industrial isotopes, (2) producing plutonium-238 (annual production of up to 5 kilograms [11 pounds]), and (3) supporting nuclear energy research and development. In accordance with U.S. nuclear nonproliferation policy, a design limitation of this new research reactor is that it can only use low-enriched uranium with an enrichment of less than 20 percent uranium-235. This preconceptual design includes the basic elements of the research reactor facility, which are sufficient to support the NI PEIS, but does not include the design details (e.g., system and layout drawings, bill of materials, electrical and piping routing) commensurate with a complete preliminary reactor design.

The reactor design was developed to a level of detail that was adequate to assess the environmental impacts associated with the construction and operation of the proposed facilities and the technical feasibility of meeting the mission objectives. The design of the new research reactor is based on current research reactor designs that have been approved by both the NRC and the International Atomic Energy Agency, as well as nuclear regulatory authorities of many nations. Reactor core physics calculations were performed to evaluate three different nuclear fuel designs. Based on this analysis, the desired mission for this reactor, current nuclear fuel manufacturing capabilities, and safety considerations; a training, research, isotope General Atomics (TRIGA) production reactor fuel design was selected for the new research reactor. The principal distinguishing features of the TRIGA fuel are its proven safety performance during power pulsing and its demonstrated long-term irradiation integrity.

To concurrently produce medical and industrial isotopes along with the required quantity of plutonium-238 production goal of 5 kilograms (11 pounds) per year and provide irradiation services for civilian nuclear energy research and development, it was determined that a reactor core power of 50 megawatts-thermal would be necessary. Higher power levels and alternative target designs capable of meeting production requirements were also considered in the new research reactor design analysis but were not analyzed in the NI PEIS. For example, although not analyzed in the NI PEIS, operating at 100 megawatts-thermal could reduce the amount of neptunium-237 required to meet plutonium-238 production requirements. At the 50-megawatts-thermal power level, the core would require an active cooling system with forced coolant flow to maintain the fuel below its material thermal limits. The new research reactor cooling system would use a tank within a pool that is connected to primary coolant circulating pumps, heat exchangers, and an ultimate heat sink consisting of two cooling towers. The pool would be housed in a reactor building that would also enclose the pumps, heat exchangers, secondary systems, and spent nuclear fuel storage pool. The spent nuclear fuel storage pool, sized to store the reactor core's discharged spent nuclear fuel for its entire 35-year production period, could be hydraulically connected to the reactor core pool for refueling and emergency reflooding. The ultimate heat sink cooling towers, air exhaust stack, and emergency diesel generators would be located outside the reactor building.

The fuel for the new research reactor would be based on an extension of current licensed low-enriched uranium TRIGA fuel designs for 10- to 16-megawatts-thermal reactors. The new research reactor fuel design would be identical to current low-enriched uranium TRIGA fuel for higher power cores, except the new reactor fuel would have a larger assembly configuration array (i.e., 8 by 8 versus 4 by 4) and a longer active fuel length (153.7 centimeters [60.5 inches] versus 55.88 centimeters [22.0 inches]). The larger array and length were selected to meet the plutonium-238 production requirements and to maintain high safety factors with respect to fuel thermal performance.

Along with the fuel rods, the core would contain a number of medical and industrial isotope and plutonium-238 production target rods. These rods would occupy positions in a fuel assembly where a fuel rod would otherwise exist. Each of these positions would have an Incoloy-800 alloy guide tube with the same dimensions as the fuel rod cladding. The target rods would be inserted into these guide tubes for their design irradiation time period. In addition, some fuel rod positions in core fuel assemblies would be replaced with similar guide tubes to accommodate Incoloy-800-clad boron carbide control rods. Boron carbide is a widely used, proven, and accepted neutron absorber for control rods. The new research reactor core design would consist of 68 fuel assemblies, each of which would be enclosed in a square aluminum shroud for structural support and coolant flow control. The core would include eight rabbit tubes for short irradiation time production of medical or industrial isotopes and civilian nuclear energy research and development. These rabbit tubes would be located outside the fuel region of the core, but still within an area with a relatively high neutron flux.

The new research reactor would be constructed and operated at an existing DOE site. Since the potential site has not been selected, it is evaluated in the NI PEIS as a generic DOE site. Because Alternative 4 was evaluated at a generic DOE site, no credit was taken for any existing support infrastructure at the site, and it was postulated that a new support facility would be required to support operation of the new research reactor and its medical isotope production and civilian nuclear energy research and development missions. While this approach bounds the environmental impact assessment for the implementation of Alternative 4, it overstates the impacts because the NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 4 were selected in the Record of Decision for subsequent consideration, follow-on NEPA reviews would evaluate potential site locations. It is unlikely that DOE would consider locating the new research reactor on a DOE site that does not have an existing infrastructure capable of supporting all or most of the mission requirements. To determine the environmental impacts if Alternative 4 were implemented at a site with adequate support infrastructure, the environmental impacts for the construction of the support facility could be subtracted from the environmental impacts of Alternative 4 as presented in the NI PEIS. Section 4.6 of the NI PEIS presents the environmental impacts from construction and operation of the new support facility separately.

Transportation

For all alternatives, overland shipments of nuclear materials are assumed to use trucks, either commercial vehicles or DOE safe secure trailers. Transatlantic shipments of mixed oxide fuel would use purpose-built ships and certain isotopes would be shipped in aircraft. The types of packaging used to transport materials is discussed in Appendix J of the NI PEIS.

Plutonium-238 purchased from Russia under all options of the No Action Alternative would be transported from St. Petersburg to a U.S. port of entry, and from there to LANL where it would be prepared for use in radioisotope power systems and heating units. The impacts of the transportation of a total of 40 kilograms (88.2 pounds) of plutonium-238 are estimated in the *Environmental Assessment of the Import of Russian Plutonium-238* (DOE 1993) and are summarized in Section 4.2 of the NI PEIS. The impacts associated with transporting 175 kilograms (385 pounds) (5 kilograms per year for the 35-year evaluation period) of plutonium-238 have been determined by extrapolation and are included in the same section. Under Options 2 through 4 of the No Action Alternative, neptunium-237 would be shipped from SRS to the designated storage facilities at ORNL, INEEL, or Hanford for long-term storage. Under Alternatives 1 through 4, the neptunium-237 would be shipped to the same facilities for storage and subsequent processing for fabrication of targets for plutonium-238 production. Under all alternatives, medical isotopes would continue to be shipped to commercial vendors via truck and air from DOE locations throughout the country.

Under Alternative 1, targets for plutonium-238 production would be fabricated in one of three alternative facilities at ORNL, INEEL, or Hanford. The targets would be irradiated at FFTF using mixed oxide fuel currently stored at Hanford or shipped from Europe and/or highly enriched uranium fuel from a commercial fuel fabricator in the United States. The irradiated targets would be transported back to the fabricating facilities for postirradiation processing. The separated plutonium-238 would be transported to LANL for fabrication into heat sources for radioisotope power systems. Targets for medical and industrial isotope production would be fabricated in one or more facilities at Hanford. Target materials would be shipped to Hanford from other offsite facilities. The targets would be irradiated in FFTF and returned to the fabrication facilities for postirradiation processing. Medical and commercial isotopes would then be shipped to commercial vendors via truck and air.

Under Alternative 2, targets for plutonium-238 production would be fabricated in one of three facilities at ORNL, INEEL, or Hanford. The targets would be irradiated at ATR, HFIR, or a CLWR and transported back to the fabricating facilities for postirradiation processing. The separated plutonium-238 would than be shipped

to LANL following postirradiation processing. Medical isotopes would continue to be shipped to commercial vendors via truck and air from DOE locations throughout the country.

Under Alternative 3, the targets for plutonium-238 production would be fabricated in one of three facilities at ORNL, INEEL, or Hanford. The targets would be irradiated at the new high-energy accelerator and transported back to the fabricating facilities for postirradiation processing. The separated plutonium-238 would than be shipped to LANL following postirradiation processing. Targets for medical and industrial isotope production would be fabricated in a new facility at the generic DOE site. Target materials would be shipped to the new facility from offsite. The targets would be transported to the on site low-energy accelerator for irradiation and returned to the fabrication facilities for postirradiation processing. Products would then be shipped to commercial vendors via truck and air transport.

Under Alternative 4, the targets for plutonium-238 production would be fabricated in one of three facilities at ORNL, INEEL, or Hanford. The targets would be irradiated at the new reactor and transported back to the fabricating facilities for postirradiation processing. The separated plutonium-238 would then be shipped to LANL following postirradiation processing. Targets for medical and industrial isotope production would be fabricated in a new facility at the generic DOE site. Target materials would be shipped to the new facility from offsite. The targets would be transported to the new on site research reactor for irradiation and returned to the fabrication facilities for postirradiation processing. Products would then be shipped to commercial vendors via truck and air transport.

No transportation is analyzed for Alternative 5, the deactivation of FFTF, with no new missions. Medical isotopes would continue to be shipped to commercial vendors via truck and air from DOE locations throughout the country.

For alternatives that include fabrication and irradiation of targets at one site, intrasite transportation between facilities is analyzed. The shipment of fuel to the irradiation facilities is also analyzed. For Alternative 4, this includes the shipment of low-enriched uranium fuel to the new reactor. For alternatives involving irradiation at FFTF, this includes the shipment of mixed oxide fuel from Europe and/or highly enriched uranium fuels from a commercial fuel fabricator in the United States. At this time, however, DOE has not proposed to import the European fuel through any specific port. DOE did, however, review the potential maximum impacts from the marine transportation of mixed oxide fuel from Europe to a representative military port (i.e., Charleston, South Carolina). If DOE ultimately decides to import fuel from Europe, it would perform a separate NEPA analysis to select a port.

S.5 Approach to Environmental Impact Analysis

The environmental impact analysis addresses the full range of natural and human resource areas pertinent to the sites considered for the nuclear infrastructure alternatives. Impacts are assessed for land resources, noise, air quality, water resources, geology and soils, ecological resources, cultural and paleontological resources, socioeconomics, waste management, and cumulative impacts. A region of influence for each resource area is identified and analyzed for each candidate site.

Baseline conditions at the three DOE sites (ORR, INEEL, and Hanford) assessed in the NI PEIS, as well as an existing CLWR, include present and reasonably foreseeable future actions at each site. Option 1 of the No Action Alternative was used as the basis for the comparison of impacts that would occur under implementation of the other options and alternatives.

Impacts within each resource area were analyzed consistently; that is, the impact values were estimated using a consistent set of input variables and computations. Moreover, calculations in all areas used accepted

protocols and up-to-date models. The following is a brief summary of the affected resources and their impact assessment methodologies.

Land Use

Land use includes the land on and adjacent to each site, the physical features that influence current or proposed uses, pertinent land use plans and regulations, and land ownership and availability. The region of influence for land use varies due to the extent of land ownership, adjacent land use patterns and trends, and other geographic or safety considerations. The amount of land disturbed and conformity with existing land use were considered in order to evaluate impacts. Conformity with existing land use was evaluated for each alternative. Land disturbance was considered only for those alternatives involving new construction. However, because the location of one or two new accelerators or a research reactor and support facility is unknown, the acreage required is only an approximation. In order to determine the range of potential effects from new facilities, the analysis considered potential impacts from construction and operation at both a disturbed and undisturbed location at a generic DOE site.

Visual Resources

Visual resources are the natural and human-created features that give a particular landscape its character and aesthetic quality. Landscape character is determined by the visual elements of form, line, color, and texture. The region of influence for visual resources includes the geographic area from which the proposed facilities may be seen. Impacts to visual resources were determined by evaluating whether or not the Bureau of Land Management Visual Resource Management classification of the site would change as a result of the proposed action. For those alternatives involving existing facilities at known DOE sites, alterations to visual features were readily evaluated and the impact on the current Visual Resource Management classification determined. For those alternatives involving construction and operation of one or two new accelerators or a research reactor at a generic DOE site, the visual characteristics of the site are unknown. Thus, to determine the range of potential visual effects, the analysis considered potential impacts from construction and operation at both a disturbed and an undisturbed location at the generic site. Impacts associated with the use of an existing CLWR are also described in a general manner because its location is not known.

Noise

Sound results from the compression and expansion of air or some other medium when an impulse is transmitted through it. Sound requires a source of energy and a medium for transmitting the sound wave. Propagation of sound is affected by various factors, including meteorology, topography, and barriers. Noise is undesirable sound that interferes or interacts negatively with the human or natural environment. The region of influence for each site includes the site and surrounding area, including transportation corridors, where proposed activities might increase noise levels. Impacts from facility modification and operation were assessed according to the types of noise sources and the locations of the proposed facilities relative to the site boundary. Potential noise impacts from traffic were based on the likely increase in traffic volume. Possible impacts to wildlife were evaluated based on the possibility of sudden loud noises occurring during facility modification and operation. Acoustic impacts from facility construction and operation at generic sites were assessed according to the types of new noise sources and characteristics identified for a generic site. The change in traffic noise levels at a generic site could not be assessed without site-specific data.

Air Quality

Air pollution refers to the introduction, directly or indirectly, of any substance into the air that could result in harmful effects of such nature as to endanger human health and harm living resources and ecosystems, as well

as material property, and impair or interfere with the comfortable enjoyment of life and other legitimate uses of the environment. For the purpose of the NI PEIS, only outdoor air pollutants were addressed, which may be in the form of solid particles, liquid droplets, gases, or a combination of these forms. Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Potential air quality impacts of pollutant emissions from facility modification and normal operations were evaluated for those alternatives associated with FFTF restart and the use of existing facilities. This assessment included a comparison of pollutant concentrations from each alternative with applicable Federal and state ambient air quality standards. If both Federal and state standards exist for a given pollutant and averaging period, compliance was evaluated using the more stringent standard. Air quality impacts associated with a CLWR were addressed as a contribution from the facility operation. Air quality impacts from one or two new accelerators or a new research reactor were discussed for construction and operation at a generic DOE site. Emissions of potential stratospheric ozone-depleting compounds were not evaluated, as no emissions of these pollutants were identified in conceptual engineering design reports.

Water Resources

Water resources are the surface and subsurface waters that are suitable for human consumption, aquatic or wildlife propagation, agricultural purposes, irrigation, or industrial and commercial purposes. The region of influence used for water resources encompasses those surface water and groundwater systems that could be impacted by water withdrawals, effluent discharges, and/or spills or stormwater runoff associated with construction and operation of the proposed facilities. Water use analysis involved the review of engineering estimates of expected water use and effluent discharges associated with each alternative, and the impacts on local water availability and quality, including surface water and groundwater. Impacts on water use were assessed by determining changes in the volume of current water usage and effluent discharges as a result of the proposed activities. Water quality analysis consisted of determining how effluent discharges to surface water, as well as discharges reaching groundwater, from the proposed facilities would affect current water quality. A comparison of the projected water quality with relevant regulatory standards was made. Separate analyses were conducted for surface water and groundwater impacts.

Geology and Soils

Geologic resources include consolidated and unconsolidated earth materials, including mineral assets such as ore and aggregate materials, and fossil fuels such as coal, oil, and natural gas. Geologic conditions include hazards such as earthquakes, faults, volcanoes, landslides, and land subsidence. Soil resources include the loose surface materials of the earth in which plants grow, usually consisting of mineral particles from disintegrating rock, organic matter, and soluble salts. Prime farmland includes cropland, pasture land, rangeland, and forest land. The region of influence for geology and soils includes all areas subject to disturbance by construction and operation of the proposed facilities, as applicable, and those areas beneath existing or proposed new facilities that would remain inaccessible for the life of the facilities. The geology and soils impact analysis considered the risks to the existing and proposed new facilities of large-scale geologic hazards such as faulting and earthquakes, lava extrusions and other volcanic activity, landslides, and sinkholes, (i.e., conditions that tend to affect broad expanses of land). As the exact nature of the generic DOE or CLWR sites is not known, bounding assumptions were made regarding the range of potential geologic and soils conditions that could be present, coupled with the use of highly conservative estimates of expected impacts. If a DOE or CLWR site were selected, subsequent NEPA assessment would be required.

Ecological Resources

Ecological resources include terrestrial and aquatic resources (plants and animals), wetlands, and threatened and endangered species. Terrestrial resources are defined as those plant and animal species and communities

that are most closely associated with the land; for aquatic resources, a water environment. Wetlands generally include swamps, marshes, bogs, and similar areas. Endangered species are defined as those species in danger of extinction throughout all or a large portion of their range. Threatened species are defined as those species likely to become endangered within the foreseeable future. Critical habitat is defined as specific areas that contain physical and biological features essential to the conservation of species and that may require special management consideration or protection. The region of influence used for the ecological resource analysis encompassed the area potentially disturbed by construction and operation of the proposed facilities. Impacts to ecological resources may occur as a result of land disturbance, water use, air and water emissions, human activity, and noise associated with project implementation. For alternatives involving construction and operation of one or two new accelerators or a research reactor at a generic DOE site, the analysis generally considered impacts at both a disturbed and an undisturbed location at a generic DOE site. Impacts to terrestrial and aquatic ecosystems and wetlands from water use and air and water emissions were evaluated based on the results of the analysis conducted for air quality and water resources.

Cultural and Paleontological Resources

Potential impacts were assessed separately for each of the three general categories of cultural resources: prehistoric, historic, and Native American. Prehistoric resources are physical remains of human activities that predate written records. Historic resources consist of physical remains that postdate the emergence of written records; in the United States, they are architectural structures or districts, archaeological objects, and archaeological features dating from 1492 and later. Native American resources are sites, areas, and materials important to Native Americans for religious or heritage reasons. Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age. The region of influence for the cultural and paleontological resource analysis encompassed the area potentially disturbed by construction and operation of the proposed facilities. The analysis of impacts to cultural and paleontological resources addressed potential direct and indirect impacts at each site. Potential indirect impacts include those associated with reduced access to a resource site, as well as impacts associated with increased traffic and visitation to sensitive areas. Direct impacts include those resulting from groundbreaking activities associated with new construction. Because the specific location is unknown, impacts from new construction of one or two new accelerators or a research reactor, as well as operation of an existing CLWR, were addressed in a general manner. In order to determine the range of potential impacts, the analysis for new construction considered potential effects at both a disturbed and an undisturbed location at a generic DOE site.

Socioeconomics

Socioeconomic impacts are defined in terms of changes to the demographic and economic characteristics of a region. The socioeconomic environment is made up of two geographic regions, the regional economic area and region of influence. Regional economic areas are made up of regional economies and include descriptions of industrial and service sector characteristics and their linkages to the communities within a region. For each regional economic area, data were compiled on the current socioeconomic conditions, including unemployment rates, economic industrial and service sector activities, and the civilian labor force. The workforce requirements of each alternative were determined in order to measure their possible effect on these socioeconomic conditions. Similarly, potential demographic impacts were assessed for the region of influence. The region of influence could represent a smaller geographic area. For each region of influence, census statistics were compiled on population, housing demand, and community services. U.S. Census Bureau population forecasts for the regions of influence were combined with overall projected workforce requirements for each of the alternatives being considered at each of the sites to determine the extent of impacts on housing demand and levels of community services. For those alternatives involving construction and operation of one or two new accelerators or a research reactor at a generic DOE site, the socioeconomic characteristics of the site are unknown. Specific impacts cannot be measured until candidate sites are identified and therefore,

impacts were addressed in a general manner. Impacts associated with the use of an existing CLWR were also addressed in a general manner as the location is unknown.

Public and Occupational Health and Safety—Normal Operations

An individual may be exposed to ionizing radiation externally (from a radioactive source outside the body) or internally (from ingesting or inhaling radioactive material). For the analyses conducted in the NI PEIS, exposure to a radioactive source as a result of releases to air and water pathways has been considered. The dose from internal exposure was calculated over 50 years following the initial exposure. The three types of doses calculated are external dose, internal dose, and combined external and internal dose. The external dose can result from several different pathways (exposure to a radioactive source in the air, water, or ground), all having in common the fact that the radiation causing the exposure is external to the body. The appropriate measure of dose is called the effective dose equivalent. The internal dose results from a radiation source entering the human body through either ingestion of contaminated food or inhalation of contaminated air. The unit of measure for internal dose is the committed effective dose equivalent. The units used for combined external and internal dose are the rem and millirem (1/1000 of 1 rem). The corresponding unit for the collective dose to a population (the sum of the doses to members of the population, or the product of the number of exposed individuals and their average dose) is the person-rem.

The potential impacts of exposure to hazardous chemicals released to the atmosphere were also evaluated for routine operations associated with the alternatives analyzed in the NI PEIS. The receptors considered in these evaluations are the public. Impacts of exposures to hazardous chemicals for workers directly involved in the treatment process were not quantitatively evaluated because workers use personal protective equipment and engineering process controls that limit their exposure to levels within applicable limits. The health effect endpoints evaluated in this analysis include excess incidences of latent cancers for carcinogenic chemicals, and a spectrum of chemical-specific noncancer health effects expressed in terms of a hazard index. This index is a measure of the likelihood of noncancer health effects, such as headache, membrane irritation, neurotoxicity, immunotoxicity, liver toxicity, kidney toxicity, developmental toxicity, reproductive toxicity, and genetic toxicity for noncarcinogens.

Public and Occupational Health and Safety—Facility Accidents

The accidents considered in the NI PEIS for both the irradiation facilities and the processing facilities are based on a spectrum of accidents ranging from high-probability low-consequence events to extremely unlikely higher consequence events. All the facilities have been treated comparably with regard to accident evaluation, while incorporating facility-specific differences in design and mitigation features.

For each evaluated accident, radiological dose consequences are provided for the maximally exposed individual. The maximally exposed individual is typically defined as a hypothetical individual who resides at the nearest site boundary in the direction that would result in the highest dose, assuming an accident occurs. Since major highways pass through some of the sites and these are well traveled, the NI PEIS also included an evaluation of individuals assumed to be located on a highway within the site. Accident doses to individuals at the nearest site boundary and on highways within the site were evaluated and the hypothetical individual receiving the highest dose was designated as the maximally exposed individual. For the hazardous chemical accident analysis, consequences are determined by comparing estimated airborne chemical concentrations to emergency response guidelines. Hazardous chemical impact information is presented for both individuals.

While it is possible that an individual member of the public could be closer to a facility than either the site boundary or the nearest onsite highway, such individuals would be present only occasionally and for brief periods (a few hours or more). Therefore, the annual probability that an individual would be close to a facility

is relatively low, and the associated risk to that individual would be bounded by the maximally exposed individual at the site boundary or nearest onsite highway.

In addition to the maximally exposed individual, accident consequence information is also provided for a noninvolved worker. For the NI PEIS accident analysis, the noninvolved worker is a hypothetical individual located 640 meters (0.4 miles) from the affected facility. The noninvolved worker impacts are provided for each facility except the CLWR. CLWR accidents selected for analysis in the NI PEIS are severe accidents that are intended to envelop the accident risk. Due to the nature and timing of these accidents, there is sufficient time prior to a radioactive release to initiate site emergency procedures. The NI PEIS accident analysis assumes that noninvolved workers, trained in emergency procedures, would have sufficient time to evacuate without suffering any consequences.

Radiological accident impacts are also provided for the offsite population within an 80 kilometer (50 mile) radius of each facility. Additional accident analyses include the evaluation of involved worker impacts and industrial accidents. Because of the large uncertainties associated with involved worker impacts, the consequences are presented qualitatively.

Public and Occupational Health and Safety—Transportation

The transportation of any commodity involves a risk to both transportation crew members and members of the public. This risk results directly from transportation-related accidents and indirectly from the increased levels of pollution from vehicle emissions, regardless of the cargo. The transportation of certain materials, such as hazardous or radioactive substances, can pose an additional risk due to the unique nature of the material itself. To permit a complete appraisal of the environmental impacts of the proposed action and alternatives, the human health risks associated with the overland transportation of neptunium- and plutonium-bearing material are analyzed in the NI PEIS. For each alternative, radiological risks (i.e., those risks that result from the radioactive nature of the neptunium and plutonium) are assessed for both incident-free (i.e., normal) and accident transportation conditions. The radiological risk associated with incident-free transportation conditions would result from the potential exposure of people to external radiation in the vicinity of a loaded shipment. The radiological risk from transportation accidents would come from the potential release and dispersal of radioactive material into the environment during an accident and the subsequent exposure of people. All radiological impacts are calculated in terms of committed dose and associated health effects in the exposed populations.

In addition to the radiological risks posed by overland transportation activities, vehicle-related risks are also assessed for nonradiological causes (i.e., causes related to the transport vehicles and not the radioactive cargo) for the same transportation routes. The nonradiological transportation risks, which would be incurred for similar shipments of any commodity, are assessed for both incident-free and accident conditions. The nonradiological risks during incident-free transportation conditions would be caused by potential exposure to increased vehicle exhaust emissions. The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities unrelated to the shipment of cargo. National transportation fatality rates are used in the assessment. Nonradiological risks are presented in terms of estimated fatalities.

Environmental Justice

The NI PEIS provides an assessment of the potential for disproportionately high and adverse human health or environmental effects on minority and low-income populations from the implementation of each alternative. Adverse health effects are measured in risks and rates that could result in latent cancer fatalities, as well as other fatal or nonfatal adverse impacts to human health. Disproportionately high and adverse human health

effects occur when the risk or rate of exposure to an environmental hazard for a minority population or low-income population is significant and exceeds the risk or exposure rate for the general population or, where available, for another appropriate comparison group. A disproportionately high and adverse environmental impact refers to an impact (or risk of an impact) in a low-income or minority community that is significant and exceeds the adverse environmental impact on the larger community. In assessing cultural and aesthetic environmental impacts, impacts that uniquely affect geographically dislocated or dispersed or minority low-income populations are considered. Potentially affected areas examined in the NI PEIS include areas defined by an 80-kilometer (50-mile) radius centered on candidate facilities for plutonium-238 production, radioisotope production, or processing activities located at INEEL, ORR, and Hanford. Potentially affected areas used in the analysis of environmental justice are the same as those used in the analysis of radiological health effects. Potentially affected areas for the other resource areas are included in the potentially affected areas used for the analysis of radiological health effects.

Waste Management

The construction and operation of the proposed facilities, as well as the permanent deactivation of FFTF and decontamination and decommissioning of one or two accelerators, research reactor, and support facility, would generate several types of waste, depending on the alternative. Such waste may include high-level radioactive waste, transuranic waste, low-level radioactive waste, mixed low-level radioactive waste, hazardous waste and nonhazardous waste. The alternatives could have an impact on existing site facilities devoted to the treatment, storage, and disposal of these categories of waste. Impacts were assessed by comparing the projected waste stream volumes generated from the proposed activities at each site with that site's waste management capacities and generation rates. Only the impacts relative to the capacities of waste management facilities were considered; other environmental impacts of waste management facility operations (e.g., human health effects) are evaluated in other sections of the NI PEIS, or in other facility-specific or sitewide NEPA documents. Projected waste generation rates for the proposed activities were compared with site processing rates and capacities of those treatment, storage, and disposal facilities likely to be involved in managing the additional waste. Projected waste stream volumes could not be compared to site waste management capacities and generation rates for the alternatives involving the use of a generic DOE site or a CLWR site because a specific location was not identified.

Cumulative Impacts

Cumulative impacts can result from individually minor but collectively significant actions taking place over a period of time. The cumulative impact analysis for the NI PEIS involved combining the impacts of the alternatives (including No Action) with the impacts of other present and reasonably foreseeable activities in the regions of influence. The regions of influence for different resources can vary widely in extent. In general, cumulative impacts were calculated by adding the values for the baseline affected environment (i.e., conditions attributable to present actions by DOE and other public and private entities), the proposed action (or no action), and other future actions. This cumulative value was then weighed against the appropriate impact indicators (e.g., standards) to determine the potential for impact. For this cumulative impact assessment, it was conservatively assumed that all facilities would operate concurrently at the DOE sites. Decontamination and decommissioning of the proposed facilities was not addressed in the cumulative impact estimates. Given the uncertainty regarding the timing of decontamination and decommissioning, any impact estimate at this time would be highly speculative. A detailed evaluation of decontamination and decommissioning will be provided in follow-on NEPA documentation closer to the actual time of those actions.

S.6 SUMMARY OF ENVIRONMENTAL IMPACTS AND MISSION EFFECTIVENESS

The following section summarizes the environmental impacts associated with the alternatives and options and compares the impacts among the alternatives described in Chapter 4 of the NI PEIS. Chapter 4 shows construction impacts that would result from implementation of Alternatives 3 and 4, as well as operational impacts for all of the alternatives.

As discussed in Section S.2, tables and text in this section have been revised in response to comments about the difficulty of comparing environmental impacts among the alternatives in the Draft NI PEIS. Tables and figures in this section now focus on estimated environmental impacts that would result from implementation of the alternatives. Baseline environmental data for the sites and for the candidate facilities are now given in Chapter 3 of the NI PEIS. In the NI PEIS, Option 1 of the No Action Alternative is used as a basis for the comparison of impacts at candidate sites.

Numerical values are assigned to environmental impacts that include radiological and nonradiological risks to the public and workers at the candidate sites and along representative transportation routes, potential quantities of waste generated, and potential quantities of spent nuclear fuel generated. These numerical values reflect the degree to which the proposed activities would increase the environmental impacts of current activities and operations at the candidate sites. It should be noted that most of the options being considered under the various alternatives involve the use of more than one site, so the numerical values presented are the sums of the values for all of the relevant sites or transportation routes. There are two exceptions—the health risks to the maximally exposed individual and the noninvolved worker. For these two exceptions, the numerical value presented is the maximum value among all relevant sites.

Radiological and Hazardous Chemical Impacts

Radiological Impacts. Table S–4 summarizes radiological and hazardous chemical risks that could occur under implementation of the alternatives from operations at fabrication, processing, and irradiation facilities. Radiological risks to the maximally exposed individual are listed in columns 2 and 5 for normal operations and accidents, respectively. Similarly, columns 3 and 6 display radiological risks to the public for normal operations and accidents, and columns 4 and 7 show radiological risks to workers at candidate irradiation facilities and processing and fabrication facilities. As indicated in the table, Option 1 of the No Action Alternative is the basis for comparing impacts that would result from implementation of the other alternatives and options. Impact values for Option 1 of the No Action Alternative are set to zero and provide a reference point for comparing impacts that would result from implementation of the other alternatives and options. Negative values in the table indicate a decrease in risk with respect to Option 1 of the No Action Alternative.

The risk values presented are the sum of individual risk values from operational activities in the fabrication, processing, and irradiation facilities used under each alternative and option. For Alternatives 2 through 4, where FFTF would be permanently deactivated, the values presented also include the reduction in risk from FFTF deactivation, where applicable. For example, the radiological risk to the population from normal operations for Option 3 of Alternative 2 (i.e., irradiation at ATR, fabrication and processing at FMEF, and deactivation of FFTF) is given as -4.7×10^{-4} latent cancer fatality. This value was calculated by adding the population risks from fabrication and processing at FMEF and irradiation at ATR, 7.7×10^{-7} latent cancer fatality, and Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), -4.7×10^{-4} latent cancer fatality. The latter risk is the sum of the population risk associated with the activities during permanent deactivation of FFTF, 1.8×10^{-5} latent cancer fatality, and that resulting from not keeping FFTF in standby for 35 years, -4.9×10^{-4} latent cancer fatality (the negative value reflects the reduction in risk). The radiological risks for accident conditions are the sum of accident risks evaluated for each option. For each accident, the

Table S–4 Comparison Among Alternatives: Impacts on Occupational and Public Health and Safety from Baseline Conditions

Options[a]	Radiological Risks from Normal Operations over 35 Years			Radiological Risks[b] from Accidents over 35 Years			Hazardous Chemical Risks from Normal Operations over 35 Years	
	Maximally Exposed Individual (LCF Risk)	Population (LCF)	Workforce (LCF)	Maximally Exposed Individual (LCF Risk)	Population (LCF)	Workforce (LCF)	Maximum Cancer Risk[c]	Hazard Index[d]
No Action Alternative								
1[e]	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
2	3.0×10^{-12}	1.4×10^{-7}	0.017	0.00	0.00	0.00	0.00	0.00
3	4.2×10^{-13}	6.1×10^{-9}	0.017	0.00	0.00	0.00	0.00	0.00
4	7.0×10^{-13}	7.5×10^{-8}	0.017	0.00	0.00	0.00	0.00	0.00
Alternative 1: Restart FFTF								
1 or 4	9.3×10^{-8}	0.0039	0.25	4.5×10^{-4}	0.54	3.5×10^{-4}	2.6×10^{-7}	0.0064
2 or 5	9.3×10^{-8}	0.0039	0.25	4.5×10^{-4}	0.41	3.5×10^{-4}	1.3×10^{-7}	0.0031
3 or 6	9.6×10^{-9}	0.0018	0.25	6.8×10^{-6}	0.21	4.2×10^{-4}	4.7×10^{-8}	0.0011
Alternative 2: Use Only Existing Operational Facilities[f, g]								
1	3.3×10^{-11}	-4.7×10^{-4}	0.16	5.7×10^{-5}	0.16	3.5×10^{-4}	2.6×10^{-7}	0.0064
2	4.6×10^{-12}	-4.7×10^{-4}	0.16	1.5×10^{-5}	0.03	3.5×10^{-4}	1.3×10^{-7}	0.0031
3	-2.3×10^{-9}	-4.7×10^{-4}	0.16	2.9×10^{-6}	0.11	3.5×10^{-4}	4.7×10^{-8}	0.0011
4	3.3×10^{-11}	-4.7×10^{-4}	0.16	5.7×10^{-5}	0.16	3.5×10^{-4}	2.6×10^{-7}	0.0064
5	4.6×10^{-12}	-4.7×10^{-4}	0.16	1.5×10^{-5}	0.03	3.5×10^{-4}	1.3×10^{-7}	0.0031
6	-2.3×10^{-9}	-4.7×10^{-4}	0.16	2.9×10^{-6}	0.12	3.5×10^{-4}	4.7×10^{-8}	0.0011
7	3.3×10^{-11}	-4.7×10^{-4}	0.16	5.7×10^{-5}	0.16	3.5×10^{-4}	2.6×10^{-7}	0.0064
8	4.6×10^{-12}	-4.7×10^{-4}	0.16	1.5×10^{-5}	0.03	3.5×10^{-4}	1.3×10^{-7}	0.0031
9	-2.3×10^{-9}	-4.7×10^{-4}	0.16	2.9×10^{-6}	0.11	3.5×10^{-4}	4.7×10^{-8}	0.0011
Alternative 3: Construct New Accelerator(s)[f, g]								
1	6.1×10^{-8}	0.0030	0.95	9.2×10^{-5}	0.22	5.0×10^{-4}	1.6×10^{-9}	1.1×10^{-7}
2	6.1×10^{-8}	0.0030	0.95	5.0×10^{-5}	0.09	5.0×10^{-4}	1.6×10^{-9}	1.1×10^{-7}
3	6.1×10^{-8}	0.0030	0.95	3.8×10^{-5}	0.18	5.0×10^{-4}	1.6×10^{-9}	1.1×10^{-7}
Alternative 4: Construct New Research Reactor[f, g]								
1	4.5×10^{-8}	0.002	0.49	9.0×10^{-5}	0.21	4.5×10^{-4}	6.4×10^{-10}	2.3×10^{-6}
2	4.5×10^{-8}	0.002	0.49	4.8×10^{-5}	0.08	4.5×10^{-4}	6.4×10^{-10}	2.3×10^{-6}
3	4.5×10^{-8}	0.002	0.49	3.6×10^{-5}	0.17	4.5×10^{-4}	6.4×10^{-10}	2.3×10^{-6}
Alternative 5: Permanently Deactivate FFTF (with No New Missions)								
	-2.3×10^{-9}	-4.7×10^{-4}	-0.0097	-2.2×10^{-13}	-1.6×10^{-8}	-1.3×10^{-13}	0.00	0.00

a. For detailed descriptions of the options under each alternative, see Section 2.5 of the NI PEIS.
b. Accident risks include accident likelihood over 35 years and the consequences.
c. Probability that an individual would develop cancer from exposure to hazardous (carcinogenic) chemicals.
d. A measure of hazard from exposure to multiple toxic (noncarcinogenic) chemicals. If this value is less than 1, the exposure is unlikely to produce an adverse toxic effect.
e. Baseline conditions for the comparison of impacts is Option 1 of the No Action Alternative.
f. These alternatives include FFTF deactivation impacts. The deactivation would lead to negative impacts (reduced risk); see Alternative 5.
g. The reduction in impacts from deactivating FFTF would affect the impacts to the population and workforce for Alternatives 2 through 4 and to the maximally exposed individual only for those options within Alternatives 2 through 4 that use FMEF.

Note: Refer to the text for a discussion on how the risk values in this table have been generated.
Key: LCF, latent cancer fatalities.

risk value is the product of the accident consequences and its occurrence likelihood over 35 years of operation. Chapter 4, Appendix H, and Appendix I of the NI PEIS provide the details on public and occupational risk calculations.

A comparison of radiological risks estimated to result from normal operations over 35 years (columns 2 and 3 of Table S–4) shows that implementation of the alternatives would result in a small risk of a latent cancer fatality among the general public. Radiological accident risks to the public over 35 years (columns 5 and 6 of Table S–4) are estimated to be less than one latent cancer fatality. **Figure S–4** shows estimated latent cancer fatalities among the population at risk from potential accidents at candidate sites. Each bar in Figure S–4 represents the estimated latent cancer fatalities for a given option.

For example, there are six bars shown above the alternative labeled "Restart FFTF." The first of the six bars represents the estimated latent cancer fatalities for implementation of Option 1, the second bar represents the estimated latent cancer fatalities for implementation of Option 2, etc. Storage containers for neptunium-237 targets would not be expected to rupture under the most severe accident evaluated in the NI PEIS. Therefore, no latent cancer fatalities would be expected under implementation of the No Action Alternative. Deactivation of FFTF (with no new missions) would result in a small reduction in radiological accident risks in comparison with the No Action Alternative. Differences in the radiological accident risks among alternatives and among options within a given alternative are driven by accident risks at the target fabrication and processing facilities. This point is illustrated in **Figure S–5**.

Figure S–5 shows risks to the public that would result from radiological accidents at candidate fabrication and processing facilities and candidate irradiation facilities. Latent cancer fatalities estimated for candidate fabrication and processing facilities are shown to the left of the dividing line in Figure S–5, and the estimated latent cancer fatalities for candidate irradiation facilities appear on the right side of the dividing line. The estimated latent cancer fatalities for FMEF under Options 3 and 6 of Alternative 1 are labeled "FMEF (Hanford)." Under Options 3 and 6 of Alternative 1, FMEF would serve as the fabrication and processing facility for all targets. If FMEF were selected to fabricate and process neptunium-237 targets only, the radiological risk to the public would be reduced by approximately a factor of two, as shown by the bar labeled "FMEF (Hanford, neptunium-237 targets only)" in Figure S–5. Among the candidate fabrication and processing facilities, accident risks to the public range from a low of 0.029 latent cancer fatality at FDPF (INEEL) to 0.377 latent cancer fatality at RPL (Hanford). Although all of the accident risks shown in Figure S–5 are less than one latent cancer fatality, risks to the public that would be expected from radiological accidents at candidate fabrication and processing facilities are relatively large in comparison to those for candidate irradiation facilities.

Prevailing weather conditions, the geographical distribution of the population at risk, and the type of target(s) processed (neptunium-237 only, other isotopes only, or both) all contribute to variations in the radiological risk to the public. Calculations of accident consequences and risks include populations residing within 80 kilometers (50 miles) of the accident site, although the consequences and risks decrease noticeably with increasing distance from the accident site. As shown in **Figure S–6**, RPL (Hanford) and REDC (ORR) have the largest populations residing within 16 kilometers (10 miles) of candidate sites, while FDPF (INEEL) has the smallest. Because the total population residing within 16 kilometers (10 miles) of FDPF is relatively small, the curve representing populations residing near FDPF is nearly coincident with the horizontal axis in Figure S–6. Comparing Figures S–5 and S–6, it is clear that accident risks due to fabrication and processing activities are driven by both the type of processing activities and the total population residing near the facilities. In turn, variations in accident risks among the alternatives, as well as variations among options within an alternative, are driven by the selection of fabrication and processing facilities. The choice for irradiation facility would have little effect on radiological accident risks to the public.

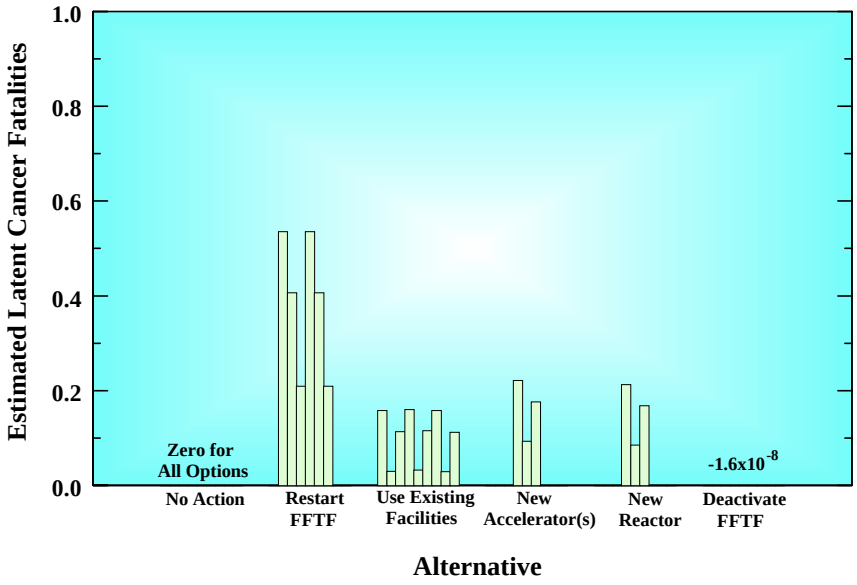


Figure S–4 Public Risks Due to Radiological Accidents at Candidate Sites (35 Years)

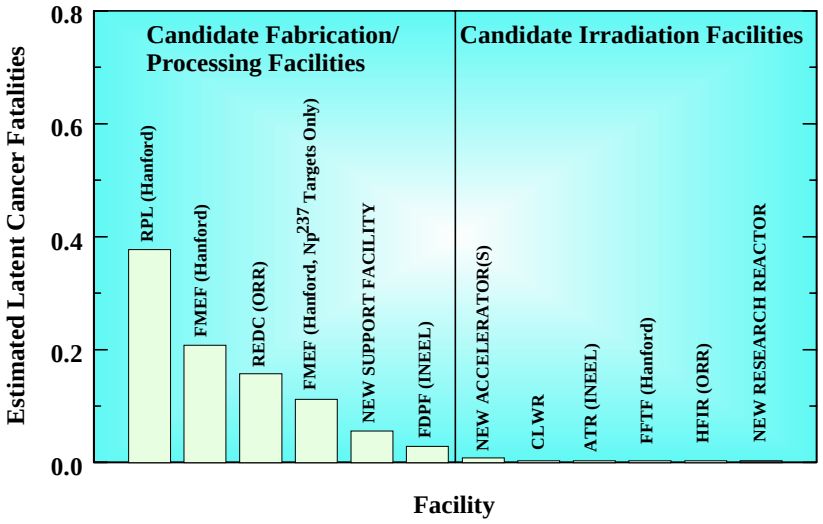


Figure S–5 Public Risks Due to Radiological Accidents at Candidate Facilities (35 Years)

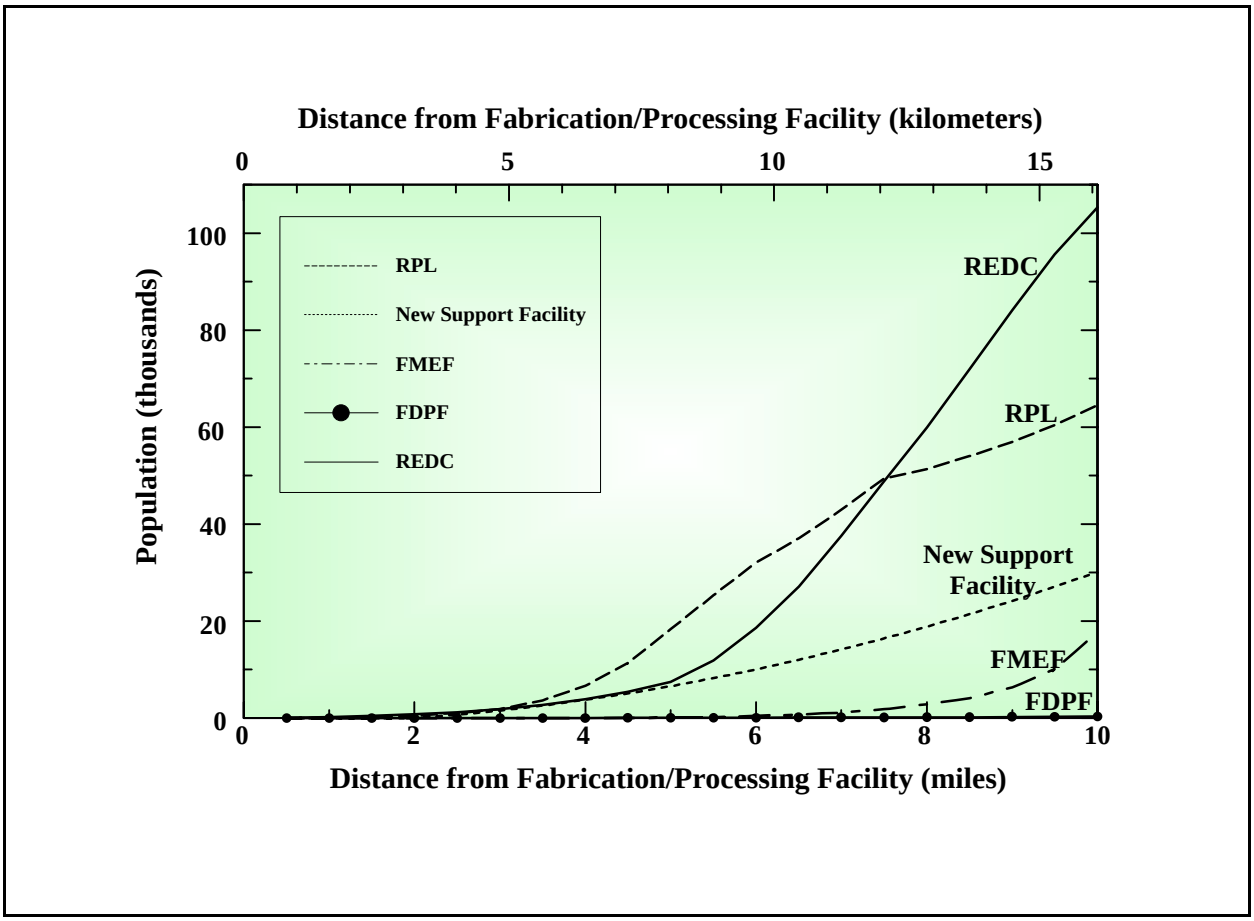


Figure S–6 Population Residing Within 16 Kilometers (10 Miles) of Candidate Fabrication and Processing Facilities

Hazardous Chemical Impacts. Columns 8 and 9 of Table S–4 display cancer risks and hazard indexes that could result from airborne emissions of hazardous chemicals from candidate processing facilities. Cancer risk factors listed in column 8 of Table S–4 are estimates of an upper-bound lifetime probability of an individual developing cancer due to exposure to carcinogenic chemicals. For all alternatives and options, the maximum cancer risk factor is 2.6×10^{-7} (or a likelihood of approximately 1 in 3,800,000) or less. Different carcinogens can cause or promote different forms of cancer. In general, cancer risk factors for different carcinogens are not additive because there are potential synergistic or antagonistic chemical interactions in multiple-substance exposures (EPA 1989). Therefore, column 8 of the table lists the maximum cancer risk factor for each alternative. Hazard indexes listed in column 9 of Table S–4 estimate the potential for adverse toxic (noncancerous) health effects due to exposure to hazardous chemicals. If the hazard index is less than one, adverse (noncancerous) health effects would not be expected. For all of the alternatives and options, hazard indexes are 0.0064 or less. The results (presented in columns 8 and 9 of Table S–4) indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under the implementation of any of the alternatives.

Generation and Disposition of Waste and Spent Nuclear Fuel

Table S–5 summarizes the estimated amount of waste and spent nuclear fuel that would be generated under implementation of the nuclear infrastructure alternatives. Waste that would result from implementation of the alternatives would be relatively small in comparison to current waste generation at the candidate sites. Current

waste management practices at the candidate sites would be sufficient to manage waste that would result from the nuclear infrastructure alternatives.

Table S–5 Comparison of Waste and Spent Nuclear Fuel Generation Among Alternatives

Options[a]	Waste Generation in Cubic Meters (35 Years)					Spent Nuclear Fuel in Metric Tons
	Transuranic/ High-Level	Low-Level	Mixed Low-Level	Hazardous	Nonhazardous	
No Action						
1	0.0	0.0	0.0	0.0	0.0	0.0
2	0.0	<10	0.0	0.0	0.0	0.0
3	0.0	<10	0.0	0.0	0.0	0.0
4	0.0	<10	0.0	0.0	0.0	0.0
Alternative 1: Restart FFTF						
1	380	5,000	320	680	943,000	16
2	240	5,200	320	680	902,000	16
3	380	5,000	320	670	1.5×10^6	16
4	380	5,000	320	680	943,000	16
5	240	5,200	320	680	902,000	16
6	380	5,000	320	670	1.5×10^6	16
Alternative 2: Use Only Existing Operational Facilities						
1	380	2,100	<180[b]	3,100[c]	105,000	0
2	240	2,300	<180[b]	3,100[c]	64,000	0
3	380	2,100	<180[b]	3,100[c]	660,000	0
4	380	2,100	<180[b]	3,100[c]	105,000	0
5	240	2,300	<180[b]	3,100[c]	64,000	0
6	380	2,100	<180[b]	3,100[c]	660,000	0
7	380	2,100	<180[b]	3,100[c]	105,000	0
8	240	2,300	<180[b]	3,100[c]	64,000	0
9	380	2,100	<180[b]	3,100[c]	660,000	0
Alternative 3: Construct New Accelerator(s)						
1	380	5,000	430[b]	3,200[c]	1.1×10^7	NA
2	240	5,200	430[b]	3,200[c]	1.1×10^7	NA
3	380	5,000	430[b]	3,200[c]	1.1×10^7	NA
Alternative 4: Construct New Research Reactor						
1	380	4,800	330[b]	3,300[c]	1.1×10^6	11
2	240	4,900	330[b]	3,300[c]	1.0×10^6	11
3	380	4,800	330[b]	3,300[c]	1.7×10^6	11
Alternative 5: Permanently Deactivate FFTF (with No New Missions)						
	0.0	0.0	(b)	2,500[d]	0.0	0

a. For detailed descriptions of the options under each alternative, see Section 2.5 of the NI PEIS.
b. The deactivation of FFTF would result in the removal of approximately 980,000 liters (260,000 gallons) of sodium. This sodium would be evaluated for alternate uses and is therefore not included in mixed low-level radioactive waste for Alternatives 2 through 5.
c. 2,500 cubic meters of these materials would be evaluated for radioactive contamination and would be reused or recycled if possible.
d. These materials would be evaluated for radioactive contamination and would be reused or recycled if possible.

Key: NA, not applicable.

Transuranic Waste/High-Level Radioactive Waste. The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste. Irrespective of how the waste is classified (i.e., transuranic or high-level radioactive), the waste composition and characteristics are the same, and the waste management (i.e., treatment and onsite storage) as described in the NI PEIS would be the same. In addition, either waste type would require disposal in a suitable repository. As shown in column 2 of Table S–5, between 240 and 380 cubic meters (314 and 497 cubic yards) of transuranic waste or high-level radioactive waste would result from implementation of Alternatives 1 through 4. This waste would result from processing irradiated neptunium-237 targets to harvest plutonium-238. Approximately 380 cubic meters (497 cubic yards) of this waste per year for 35 years would be generated for all options under Alternatives 1 through 4, except those for which target fabrication and processing would be conducted at FDPF at INEEL. If FDPF were selected for neptunium target fabrication and processing, then approximately 240 cubic meters (314 cubic yards) of waste would be generated during the program.

Low-Level and Mixed Low-Level Waste. Columns 3 and 4 of Table S–5 summarize the total low-level radioactive waste and mixed low-level radioactive waste generation that would be expected from implementation of the alternatives. Low-level radioactive waste would be generated at the irradiation facilities and at the fabrication and processing facilities. As shown, the low-level radioactive waste generation that would result under Alternative 2 would be less than half of that for Alternatives 1, 3, and 4, and mixed low-level radioactive waste generation would be almost half. This is because under Alternative 2 currently operational facilities would be used for target irradiation and these facilities would generate little additional low-level and mixed low-level radioactive waste. Also under Alternative 2, no waste generation would result from production of additional medical and industrial isotopes.

DOE's approach for managing low-level and mixed low-level radioactive waste is provided in the Record of Decision for its Waste Management Program (65 FR 10061). The Record of Decision states that for the management of low-level radioactive waste, minimal treatment will be performed at all sites, and disposal will continue to the extent practicable, on site at INEEL, LANL, ORR, and SRS. In addition, Hanford and the Nevada Test Site will be available to all DOE sites for low-level radioactive waste disposal. The Record of Decision does not preclude the use of commercially licensed low-level radioactive waste disposal facilities. Low-level radioactive waste generated at Hanford would be disposed of on site. However, if DOE determines that use of the Hanford waste management infrastructure or other DOE sites is not practical or cost effective, DOE may issue an exemption under DOE Order 435.1 for the use of non-DOE facilities (i.e., commercial facilities) to store, treat, and dispose of such waste generated from the restart and operation of FFTF.

Solid low-level radioactive waste generated at ORR eventually would have to be disposed of off site due to lack of low-level waste disposal capacity at ORR. Low-level radioactive waste generated at INEEL would be disposed of on site. At some future time, low-level radioactive waste would be disposed of off site.

In compliance with the Waste Management Program Record of Decision, DOE's mixed low-level radioactive waste will be treated at: Hanford, INEEL, ORR, and SRS, and disposed of at Hanford and the Nevada Test Site. Existing candidate sites analyzed in the NI PEIS all have treatment facilities for mixed low-level radioactive waste. Solid mixed low-level radioactive waste generated at ORR and INEEL would have to eventually be disposed of off site due to lack of onsite mixed low-level radioactive waste disposal capacity.

Hazardous Waste. Hazardous waste that would result from implementation of the nuclear infrastructure alternatives is shown in column 5 of Table S–5. The amount of hazardous waste generated under the alternatives is relatively small in comparison to hazardous waste currently generated at the candidate sites. Estimated amounts of hazardous waste that would be generated under Alternatives 2 through 4 include the hazardous waste that would be generated under Alternative 5 (Permanently Deactivate FFTF [with No New Missions]).

Based on the Record of Decision for hazardous waste issued on August 5, 1998 (63 FR 41810), nonwastewater hazardous waste would be treated and disposed of at offsite commercial facilities. Hazardous waste generated under the nuclear infrastructure alternatives would be stored in onsite facilities permitted under the Resource Conservation and Recovery Act or generator accumulation areas prior to shipment to a commercial facility permitted to manage hazardous waste.

Nonhazardous Waste. Nonhazardous waste that would be expected from implementation of the nuclear infrastructure alternatives is listed in column 6 of Table S–5. Nonhazardous waste that would be expected under implementation of Alternative 3 (Construct New Accelerator[s]) is at least a factor of six larger than the nonhazardous waste estimated for the other alternatives. Nonhazardous waste that would be produced under Alternative 3 would be driven by sanitary waste and process wastewater resulting from construction and operation of accelerators and the new support facility.

Nonhazardous solid waste that would be generated at ORR and INEEL would represent less than 0.5 percent of the generating site's onsite nonhazardous waste disposal capacity. Nonhazardous solid waste that would be generated at Hanford under the nuclear infrastructure alternatives would be recycled or sent off site for disposal as industrial waste. Nonhazardous process wastewater at the candidate sites would represent a small fraction of the generating sites capacity and would be treated on site. Sanitary wastewater would be treated on site as necessary prior to offsite disposition.

Spent Nuclear Fuel. Changes in the generation of spent nuclear fuel would occur only under implementation of Alternatives 1 (Restart FFTF) and 4 (Construct New Research Reactor). Spent nuclear fuel that would be generated under Alternative 1 would be less than 1 percent (by weight) of the current spent nuclear fuel inventory at Hanford. Spent nuclear fuel that would be generated at Hanford under implementation of Alternative 1 would be placed in facility storage vessels and onsite dry storage pending ultimate disposal in a geologic repository. Spent nuclear fuel generated under Alternative 4 would be stored on site in wet storage pending ultimate disposal in a geologic repository.

Water Use

Construction. For construction of new facilities under Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor), water is expected to be required for such uses as mixing concrete, dust control, washing activities, and potable and sanitary needs. Water use for facility construction is estimated at 22.7 million liters (6 million gallons) for the high-energy accelerator, 14 million liters (3.7 million gallons) for the low-energy accelerator, 11.7 million liters (3.1 million gallons) for the new research reactor, and 14.6 million liters (3.85 million gallons) for the new support facility on an annualized (construction-year) basis.

Operations. Figure S–7 shows the annual water use that would be expected to occur under the nuclear infrastructure alternatives. Under the No Action Alternative, FFTF would remain in standby and DOE's nuclear infrastructure would not be enhanced. In standby condition, the FFTF uses approximately 197 million liters (52 million gallons) of groundwater per year. In Figure S–7, the No Action Alternative is used as a basis for comparison of water use among the alternatives. Therefore, water use for the No Action Alternative is shown as zero. The water use shown in Figure S–7 for Alternative 1 (Restart FFTF) is the additional

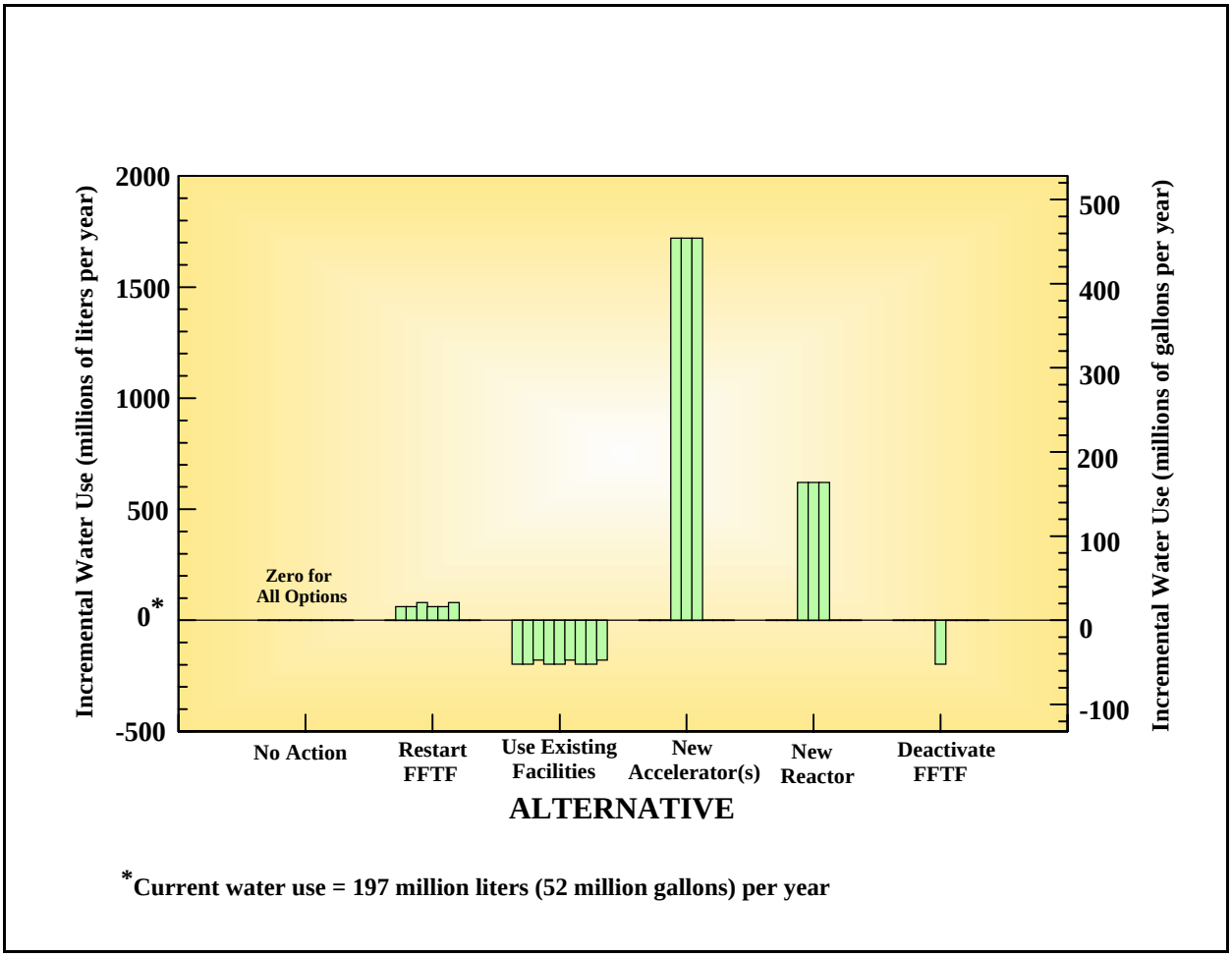


Figure S–7 Annual Water Use Under the Nuclear Infrastructure Alternatives

groundwater use that would result from operation of the FFTF. Under Alternatives 2 through 5, FFTF would be deactivated, thus saving approximately 197 million liters (52 million gallons) per year in groundwater required for maintaining FFTF in standby. As a result, the water use is negative for Alternatives 2 (Use Only Existing Operational Facilities) and 5 (Permanently Deactivate FFTF [with No New Missions]). The negative increment in water use would be more than offset by the increase in water use estimated for Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor).

Air Quality

Construction. Under Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor), new irradiation and support facilities would be constructed to support DOE's nuclear missions. Facility construction would not be required under the other alternatives. Since no specific site has yet been selected for the new accelerator[s] or the new research reactor, Federal standards are used to evaluate estimated concentrations of air pollutants. The effects of constructing the new high-energy accelerator were used to characterize air quality impacts under Alternative 3 (Construct New Accelerator[s]). Construction impacts of the low-energy accelerator and support facilities would add relatively small concentrations of air pollutants. If Alternative 3 and/or Alternative 4 were selected for implementation, site-specific environmental documentation would be prepared prior to site selection.

Construction of the new irradiation and support facilities would not be expected to exceed Federal standards and guidelines for ambient air quality. However, in comparison with air pollutant concentrations expected from facility operations, concentrations of air pollutants that would be expected during construction are relatively large. If the new facilities were constructed in an area with existing high background concentrations, construction activities could produce enough air pollutant emissions to exceed ambient air quality standards.

Operations—No Action Alternative. Under the No Action Alternative, FFTF would remain in standby and DOE's nuclear infrastructure would not be enhanced to meet the nuclear infrastructure missions. Air quality effects that would be expected from transportation of neptunium-237 oxide to REDC (Option 2), FDPF (Option 3), or FMEF (Option 4) are summarized in transportation discussion later in this section.

Operations—Alternatives 1 through 5. ***Oak Ridge Reservation*:** Under Alternatives 1 (Options 1 and 4), 2 (Options 1, 4, and 7), 3 (Option 1), and 4 (Option 1), air quality impacts at ORR would result from the production of plutonium-238 at REDC. All of the expected concentrations are small in comparison with the most stringent ambient air quality standards. Operation of REDC in support of plutonium-238 production would not be expected to significantly affect air quality or to result in air pollutant concentrations in excess of ambient air quality standards. No air quality impacts would result from operation of HFIR under Alternative 2 (Use Only Existing Operational Facilities).

***Idaho National Engineering and Environmental Laboratory*:** Under Alternatives 1 (Options 2 and 5), 2 (Options 2, 5, and 8), 3 (Option 2) and 4 (Option 2), air quality impacts at INEEL would result from the production of plutonium-238 at FDPF. All of the expected concentrations are small in comparison with the most stringent ambient air quality standards. Operation of FDPF in support of plutonium-238 production would not be expected to significantly affect air quality or to result in air pollutant concentrations in excess of ambient air quality standards. No air quality impacts would result from operation of ATR under Alternative 2 (Use Only Existing Operational Facilities).

***Hanford Site*:** If Alternative 1 were selected for implementation, impacts on air quality at Hanford would result from operation of FFTF (all options), RPL (Options 1, 2, 4, and 5), and FMEF (Options 3 and 6). FMEF could also be used for production of plutonium-238 under Alternatives 2 (Options 3, 6, and 9), 3 (Option 3), and 4 (Option 3). FFTF would be deactivated under Alternatives 2 through 5. Deactivation would, in turn, result in the shutdown of diesel-driven fire pumps, oil-fired preheaters, and a gas turbine that currently support FFTF's standby condition. If any of Alternatives 2 through 5 were selected for implementation, emissions from this supporting equipment would cease, thereby improving the air quality near FFTF. Emissions of air pollutants from FMEF are relatively small in comparison to those associated with FFTF supporting equipment.

Air quality concentrations for FFTF and FMEF were calculated with the SCREEN3 model developed by EPA. The model is intended to provide conservative estimates of the concentrations of air pollutants emitted from point or extended sources. Concentrations shown under Alternatives 2 through 5 were obtained by summing estimated emissions from the diesel-driven oil pumps, the oil-fired preheaters, and the gas turbine. Because these sources operate intermittently and do not necessarily operate at the same time, estimates of the concentrations of air pollutants are conservative because they were obtained under the assumption that all supporting equipment for FFTF would operate simultaneously, which is considered a worst-case scenario.

Generic Site for the New Accelerator(s): Under Alternative 3 (all options), air quality impacts at the site for the new accelerator(s) would result from the operation of emergency diesel generators for the high-energy accelerator and any support facilities. The low-energy accelerator would not require emergency diesel power, and it was assumed in the analysis that air quality effects of the low-energy accelerator could be ignored. Air quality impacts of the support facilities would be assessed if Alternative 3 (Construct New Accelerator[s]) were selected for implementation. In comparison with the air quality concentrations that would be expected during

construction, air quality impacts resulting from operation of the diesel generators would be relatively small. All of the expected concentrations resulting from operation of emergency generators would be small in comparison with the most stringent ambient air quality standards, and would not be expected to result in air pollutant concentrations in excess of ambient air quality standards. If the new accelerator(s) were located in an area that has high background pollutant concentrations, diesel emissions could result in pollutant concentrations in excess of the ambient standards. If Alternative 3 were selected for implementation, site-specific environmental documentation would be prepared prior to site selection.

Generic Site for the New Research Reactor: Under Alternative 4 (all options), air quality impacts at the site for the new research reactor would result from the operation of emergency diesel generators for the reactor. In comparison with the air quality concentrations that would be expected during construction, air quality impacts resulting from operation of the diesel generator would be relatively small. All of the expected concentrations resulting from operation of the emergency generator would be small in comparison with the most stringent ambient air quality standards and would not be expected to result in air pollutant concentrations in excess of ambient air quality standards. If the new research reactor were located in an area that has high background pollutant concentrations, diesel emissions could result in pollutant concentrations in excess of the ambient standards. If Alternative 4 were selected for implementation, site-specific environmental documentation would be prepared prior to site selection.

Socioeconomics

Implementation of the nuclear infrastructure alternatives would have no significant impact on regional economic areas or community services at Hanford, INEEL, and ORR. Socioeconomic impacts at the generic sites could not be evaluated in detail because areas potentially affected under Alternatives 3 and 4 could vary widely in demographic and economic composition. If Alternative 3 or 4 were selected for implementation, site-specific environmental analysis would be conducted prior to site selection. **Table S–6** shows the number of direct jobs that would be generated under implementation of the nuclear infrastructure alternatives. Deactivation of the FFTF under Alternatives 2 through 5 would result in the loss of 242 jobs that are required to keep the facility in standby condition. That loss would be offset under alternatives and options for which the FMEF would support the production of plutonium-238 (62 direct jobs).

Transportation Impacts

The transportation impacts for Option 1 of the No Action Alternative are those resulting from transporting 175 kilograms (385 pounds) (5 kilograms [11 pounds] per year for the 35-year evaluation period) of plutonium-238 from Russia to LANL. The impacts were obtained by extrapolating the impact analysis presented in the *Environmental Assessment of the Import of Russian Plutonium-238* (DOE 1993) for the purchase of 40 kilograms (88.2 pounds) of plutonium-238. The impacts presented for the other options of the No Action Alternative include those of Option 1 plus the impact from transporting neptunium oxide from SRS to the selected facilities at ORNL, INEEL, and Hanford. Because the assumptions and data used to assess the transportation impacts in the above environmental assessment are different from those used in this NI PEIS, incremental transportation impacts compared to the baseline condition (Option 1 of the No Action Alternative) can only be presented for the options under the No Action Alternative. Therefore, the transportation impacts presented in this section are not compared to the baseline condition.

Table S–6 Comparisons Among Alternatives: Change in Direct Jobs Under the Nuclear Infrastructure Alternatives

Options[a]	Oak Ridge Reservation	Idaho National Engineering and Environmental Laboratory	Hanford Site	Generic Accelerator(s) Site(Construction/ Operation)	Generic Research Reactor Site(Construction/ Operation)
No Action Alternative					
All	0	0	0	0	0
Alternative 1: Restart FFTF					
1 & 4	41	0	218	0	0
2 & 5	0	24	218	0	0
3 & 6	0	0	292	0	0
Alternative 2: Use Only Existing Operational Facilities					
1, 4, & 7	41	0	-242	0	0
2, 5, & 8	0	24	-242	0	0
3, 6, & 9	0	0	-180	0	0
Alternative 3: Construct New Accelerator(s)					
1	41	0	-242	410/225	0
2	0	24	-242	410/225	0
3	0	0	-180	410/225	0
Alternative 4: Construct New Research Reactor					
1	41	0	-242	0	160/120
2	0	24	-242	0	160/120
3	0	0	-180	0.00	160/120
Alternative 5: Permanently Deactivate FFTF (with No New Missions)					
	0	0	-242	0	0

a. For detailed descriptions of the options under each alternative, see Section 2.5 of the NI PEIS.

Radiological and nonradiological transportation impacts over the 35-year program duration are summarized in **Table S–7**. Risks to the public and workers due to incident-free transportation are shown in columns 3 through 5 of the table. Columns 6 and 7 summarize radiological and nonradiological risks to the public that could result from transportation accidents. Chapter 4 and Appendix J of the NI PEIS discuss transportation impacts in more detail.

Radiological Transportation Risks. Figure S–8 illustrates the data listed in column 6 of Table S–7. The results indicate a large risk to the public due to transportation accidents that could occur over 35 years under implementation of Alternatives 1 (Restart of FFTF), 3 (Construct New Accelerator[s]), and 4 (Construct New Research Reactor) as compared to those from implementation of Alternative 2 (Use Only Existing Operational Facilities). This large difference is due to the more than 8,000 medical isotope shipments by air transport considered under Alternatives 1, 3, and 4, and not under Alternative 2. Nearly all of the radiological and traffic accident risk are due to those involving medical and industrial isotope shipments. No enhancement of medical and industrial isotope production is considered under Alternative 2.

Implementation of Alternative 5 (Permanently Deactivate FFTF [with No New Mission]) would not result in any new transportation activities.

Table S–7 Comparison Among Alternatives: Impacts of Transportation on Occupational and Public Health and Safety

Options[a]	Transportation Distance (millions of kilometers)	Incident-Free Transportation over 35 Years			Transportation Accidents over 35 Years	
		Public: Radiological (LCF)	Workers: Radiological (LCF)	Public: Vehicle Emissions (fatalities)	Public: Radiological (LCF)	Public: Vehicle Collisions[b] (fatalities)
No Action Alternative						
1	0.11	0.010	0.0046	4.7×10^{-4}	4.4×10^{-4}	0.014
2	0.13	0.011	0.0047	5.9×10^{-4}	4.4×10^{-4}	0.014
3	0.20	0.014	0.0049	8.9×10^{-4}	4.4×10^{-4}	0.014
4	0.22	0.014	0.0050	9.2×10^{-4}	4.4×10^{-4}	0.014
Alternative 1: Restart FFTF						
1 and 4	8.0	0.149	0.012	0.030	0.53	0.19
2 and 5	6.2	0.044	0.008	0.024	0.53	0.13
3 and 6	5.6	0.009	0.007	0.023	0.53	0.12
Alternative 2: Use Only Existing Operational Facilities						
1	2.2	0.120	0.005	0.0064	4.4×10^{-5}	0.059
2	0.15	0.004	0.001	0.0007	2.1×10^{-5}	6.0×10^{-4}
3	0.83	0.040	0.002	0.0014	3.0×10^{-5}	0.017
4	2.6	0.150	0.006	0.0056	4.4×10^{-5}	0.074
5	3.1	0.179	0.007	0.0066	2.1×10^{-5}	0.088
6	3.6	0.205	0.008	0.0075	3.0×10^{-5}	0.100
7	1.8	0.096	0.004	0.0052	4.4×10^{-5}	0.048
8	0.99	0.052	0.002	0.0030	4.4×10^{-5}	0.024
9	1.6	0.084	0.004	0.0037	3.0×10^{-5}	0.039
Alternative 3: Construct New Accelerator(s)						
1	5.7	0.054	0.008	0.023	0.53	0.14
2	5.8	0.057	0.008	0.023	0.53	0.14
3	5.9	0.065	0.009	0.023	0.53	0.14
Alternative 4: Construct New Research Reactor						
1	7.5	0.154	0.011	0.026	0.53	0.19
2	7.5	0.157	0.012	0.026	0.53	0.19
3	7.9	0.177	0.012	0.027	0.53	0.19
Alternative 5: Permanently Deactivate FFTF (with No New Missions)						
	NA[c]	NA[c]	NA[c]	NA[c]	NA[c]	NA[c]

a. For detailed descriptions of the options under each alternative, see Section 2.5 of the NI PEIS.
b. No radiological spill.
c. No new transportation activities would occur under Alternative 5.
Key: LCF, latent cancer fatalities.

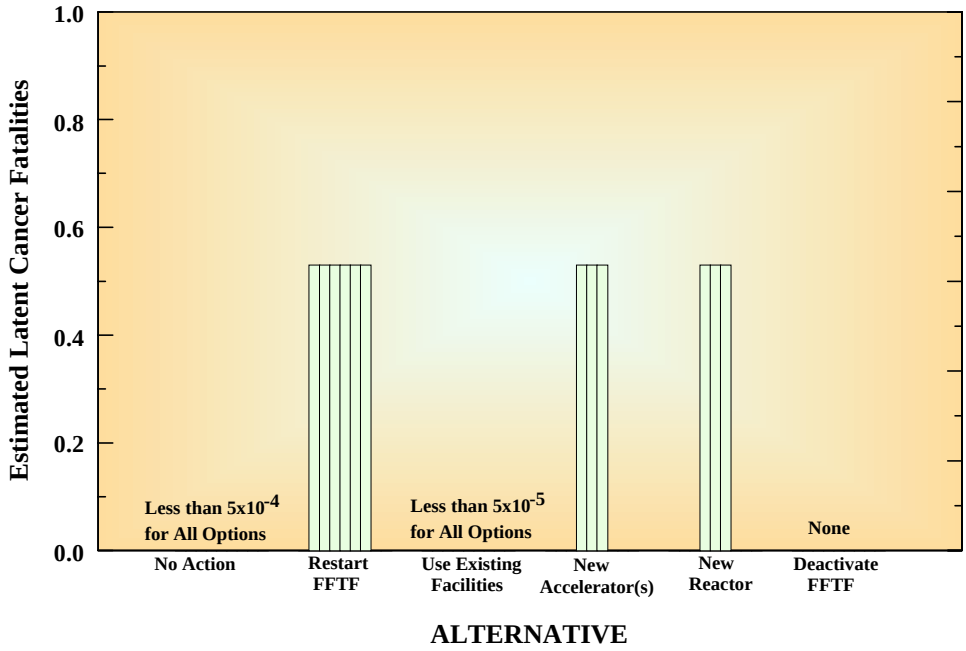


Figure S–8 Public Risks Due to Radiological Transportation Accidents (35 Years)

Figure S–9 shows the radiological risks to the public that could result from incident-free transportation over 35 years (column 3 of Table S–7). For all of the alternatives and options, incident-free radiological transportation risks are approximately 0.2 latent cancer fatality over 35 years. As shown in column 4 of Table S–7, radiological risks to workers due to incident-free transportation are less than approximately 0.012 latent cancer fatality for all alternatives and options.

Nonradiological Transportation Risks. Column 7 of Table S–7 shows the risks of traffic fatalities that would be expected to result from vehicular collisions in which there is no radiological spill. Under all alternatives and options, the expected number of traffic fatalities would be less than approximately 0.2. Data listed in column 5 of the same table indicates that less than approximately 0.03 fatality would be expected from vehicular exhaust emissions. Fatalities that would be expected to result from both vehicular collisions and exhaust emissions are closely correlated with the estimated highway mileage that would be traveled under implementation of the alternatives (see column 2 of Table S–7 and **Figure S–10**). Traffic accident rates depend on the type of carrier. Both commercial trucks and DOE's safe, secure trailer/SafeGuards Transports (SST/SGTSs) would be used for the highway transport of isotopes. Accident rates for the safe, secure trailer system are less than those for commercial trucks by at least a factor of five. As a result, expected collision fatalities for any option would increase the total distance traveled, but the impacts would also depend on relative amounts of transportation by commercial truck and the SST/SGTs.

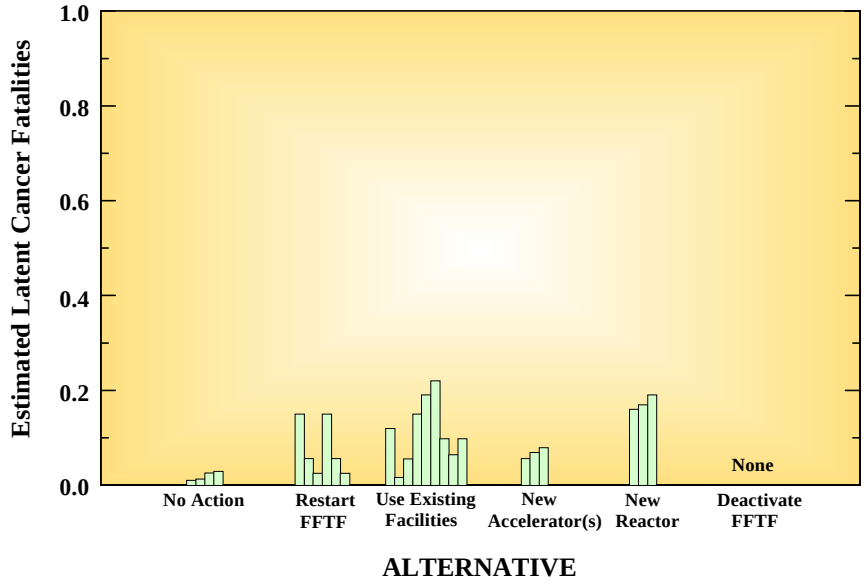


Figure S–9 Radiological Risks to the Public Due to Incident-Free Transportation (35 Years)

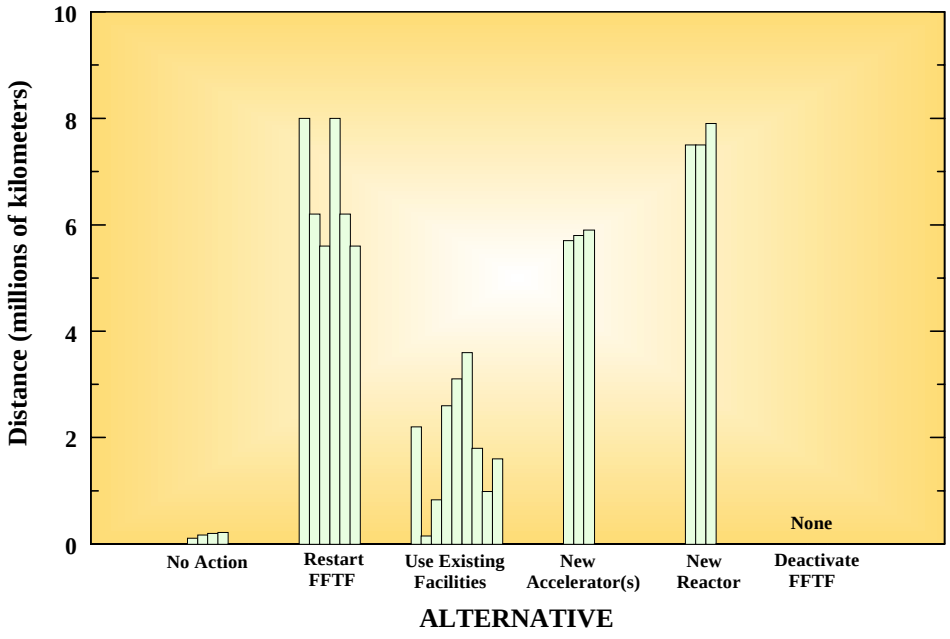


Figure S–10 Highway Distances That Would Be Traveled Under the Alternatives (35 Years)

Resource Areas Discussed in Less Detail

Implementation of the nuclear infrastructure alternatives at existing candidate sites would be expected to have little effect on land use, visual resources, noise, water quality, geology and soils, ecology, cultural resources, and environmental justice. Implementation of the alternatives at one or more generic sites could potentially result in significant impacts in one or more of these resource areas. However, these impacts are site-specific and could not be evaluated in detail in this programmatic document. If Alternative 2 (Options 4, 5, and 6), 3, or 4 were selected for implementation, site-specific environmental documentation would be prepared prior to site selection.

Land Use. Implementation of the nuclear infrastructure alternatives at existing operational candidate sites at Hanford, INEEL, and ORR would be consistent with ongoing activities and current land use at these sites. Irradiation of neptunium targets at an existing CLWR would also be consistent with the land use at the reactor site. If Alternative 3 or 4 were selected for implementation, a site-specific evaluation of land use would be conducted prior to site selection. Deactivation of the FFTF under Alternatives 2 through 5 would have no effect on ongoing land use in the 400 Area of Hanford.

Visual Resources. Existing sites that are candidates for implementation of the nuclear infrastructure alternatives are rated Class IV under the U.S. Bureau of Land Management classification guidelines for visual resources (DOI 1986). Selection of one or more of the existing candidate sites for implementation would not affect their visual resource classification as areas in which industrial development dominates the landscape. Use of a CLWR for irradiation of neptunium targets would not alter the appearance of the reactor or the surrounding landscape. Implementation of Alternative 3 (Construct New Accelerator[s]) or 4 (Construct New Research Reactor) could result in reclassification under U.S. Bureau of Land Management guidelines. If Alternative 3 or 4 were selected for implementation, a site-specific evaluation of visual resources would be conducted prior to site selection. Deactivation of FFTF under Alternatives 2 through 5 would not significantly alter the overall landscape in the 400 Area of Hanford.

Noise. Noise associated with target fabrication and processing and irradiation at existing candidate sites would be similar to currently existing onsite noise and would not be audible beyond site boundaries. These activities would not produce sudden, loud noises that would startle wildlife. Noise levels that would be generated at a CLWR under Alternative 2 (options 4, 5, and 6) would be the same as those currently existing at the reactor site. Implementation of Alternative 3 (Construct New Accelerator[s]) or 4 (Construct New Research Reactor) would result in construction activities that could disturb nearby residents or wildlife. If Alternative 3 or 4 were selected for implementation, a site-specific NEPA review would be prepared, and an evaluation of potential noise impacts would be conducted prior to site selection. Deactivation of FFTF under Alternative 5 would not significantly alter the noise levels in the 400 Area of Hanford.

Water Quality. Under Alternative 1 (Restart FFTF), there would be no liquid radiological effluent pathways to the environment from FFTF. Process wastewater from cooling tower blow-down would be ultimately discharged to the 400 Area Pond (i.e., the 4608 B/C percolation ponds). No impact on the quality of ground or surface water would be expected. Irradiation of neptunium targets at existing reactors and a generic CLWR would have no measurable effect on the quantity or quality of discharged effluents. Use of existing facilities for target fabrication and processing would not result in direct effluent discharge to the environment, and additional wastewater generation would be relatively small in comparison to existing wastewater treatment volumes at the sites. If Alternative 3 (Construct New Accelerator[s]) or 4 (Construct New Research Reactor) were selected for implementation, construction and operation of new facilities would not be anticipated to significantly impact water quality. While the water quality impacts are expected to be small, a site-specific environmental evaluation of potential water quality impacts and mitigation measures would be conducted prior to site selection. Sodium removal during deactivation of FFTF under Alternatives 2 through 5 would result

in approximately 7,600 liters (2,000 gallons) of wastewater that would be disposed of in existing wastewater treatment facilities at Hanford. Deactivation of FFTF would not be expected to impact water quality.

Geology and Soils. Except for Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor), activities conducted under the nuclear infrastructure alternatives would not require construction of new facilities. No soil would be disturbed, and there would be no impacts on the geology of potentially affected sites. Construction of new accelerators and support facilities under Alternative 3 would be expected to disturb up to approximately 27 hectares (66 acres) of soil. If Alternative 4 were selected for implementation, construction of the new reactor and support facility would be expected to disturb approximately 4 hectares (10 acres) of soil. If Alternative 3 or 4 were selected for implementation, a site-specific environmental evaluation would be conducted prior to site selection. Deactivation of FFTF under Alternatives 2 through 5 would take place on previously disturbed land. Impacts of deactivation on geology and soils would be negligible.

Ecology. Activities that would be conducted under the nuclear infrastructure alternatives at candidate existing facilities and the generic CLWR would not involve construction of new facilities or significant changes in traffic, noise, air quality, or water quality. In addition, irradiation and processing activities would take place in established industrial areas. Impacts on terrestrial resources and wetlands would be negligible. Consultations concerning threatened and endangered species were conducted with appropriate Federal and state agencies. No major issues were raised as a result of these consultations. (Chapter 4 of the NI PEIS provides detailed discussions of the results of these consultations.)

Under Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor), construction of new facilities at a yet-to-be-determined site could potentially have a significant effect on wildlife and wetlands. If Alternative 3 or 4 were selected for implementation, site-specific ecological evaluations would be conducted prior to site selection. The evaluation would include consultation with the U.S. Fish and Wildlife Service and appropriate state authorities concerning threatened and endangered species. Deactivation of FFTF under Alternatives 2 through 5 would take place on previously disturbed land in the 400 Area. No threatened or endangered species are known to reside in the 400 Area, and noise impacts on local wildlife would be temporary.

Cultural Resources. Existing candidate facilities that would host activities under the nuclear infrastructure alternatives are located within areas that contain National Historic Landmarks or structures that are eligible for nomination to the National Register of Historic Places. Several candidate facilities are eligible for nomination to the National Register, including the Reactor Containment Building and the Control Building for FFTF at Hanford, RPL at Hanford, and ATR at INEEL. Selection of these facilities to support the nuclear infrastructure missions would not alter their eligibility.

Under the nuclear infrastructure alternatives, activities at candidate existing sites and the generic CLWR would be conducted within existing facilities. Use of the FMEF at Hanford for target fabrication and processing would require construction of a 76-meter-high (250-feet-high) stack on previously disturbed land. Similarly, construction of a support facility for deactivation of the FFTF would take place on previously disturbed land in the 400 Area. Thus, except for Alternatives 3 (Construct New Accelerator[s] and 4 (Construct New Research Reactor), no disturbance of archeological resources would be expected under the nuclear infrastructure alternatives. Consultations with the State Historic Preservation Offices and potentially affected Native American tribes have been conducted for the candidate existing sites. No major issues were raised as a result of these consultations. (Chapter 4 of the NI PEIS provides detailed discussion of the results of these consultation.)

Implementation of Alternative 3 or 4 would require construction on potentially undisturbed lands. If Alternative 3 or 4 were selected for implementation, a site-specific NEPA review would be prepared, and an environmental evaluation of cultural resources would be conducted prior to site selection. The evaluation would include consultation with State Historic Preservation Offices and potentially affected Native American tribes.

Environmental Justice. The objective of the environmental justice analysis was to determine whether or not implementation of the nuclear infrastructure alternatives would result in significant environmental impacts that disproportionately affect low-income or minority populations. Normal operations at the candidate sites and incident-free transportation pose no significant radiological risks to the public or to maximally exposed offsite individuals among the public.

Portions of the Fort Hall Indian Reservation and the Yakama Indian Reservation lie within potentially affected areas surrounding INEEL and Hanford, respectively. As discussed in Appendixes H and I of the NI PEIS, calculations of radiological risks considered human exposures due to inhalation and ingestion of radioactive materials. Ingestion of contaminated fish, vegetation, and/or wildlife is an environmental justice consideration due to potential patterns of subsistence consumption for minority or low-income populations. Radiological health models used in the environmental evaluation assumed accidents at the irradiation facilities or the fabrication and processing facilities would contaminate all of the food produced in the area, and that all of the contaminated food would be consumed by persons residing in the potentially affected area. The expected risk that would result from ingestion of radiologically contaminated food for persons residing near Hanford would be approximately 0.004 latent cancer fatality and essentially zero for persons residing near the INEEL or ORR. Thus, no credible pattern of food consumption would be expected to result in a significant health risk to low-income or minority populations residing within potentially affected areas surrounding the existing candidate sites. Implementation of the alternatives would not be expected to result in significant environmental impacts in any of the environmental resource areas. Thus, no disproportionately high and adverse impacts on minority and low-income populations would be expected to result from implementation of the alternatives.

Accidents at candidate fabrication and processing facilities and during transportation of radioisotopes by aircraft were found to pose the largest risks to the public. Under conservative assumptions described in Appendix I of the NI PEIS, no latent cancer fatalities due to accidents would be expected at the existing sites. Accidents during air transport of radioisotopes could occur anywhere along the flight path and would not place any identifiable group within the general population at disproportionate risk.

The density and distribution of total, low-income, and minority populations varies from site to site, so that evaluations of environmental justice are necessarily site-specific. If Alternatives 3 (Construct New Accelerator[s]) or 4 (Construct New Research Reactor) were selected for implementation, a site-specific NEPA review would be prepared, and an evaluation of environmental justice would be conducted prior to site selection. The evaluation would include patterns of food consumption that could result in disproportionately high and adverse effects on low-income or minority populations at risk.

Industrial Safety

Estimates of potential industrial impacts to workers during construction, irradiation, fabrication and processing were evaluated based on DOE and Bureau of Labor Statistics data. Impacts are classified into two groups: total recordable cases and fatalities. A recordable case includes work-related death, illness, or injury which resulted in loss of consciousness, restriction of work or motion, transfer to another job, or required medical treatment beyond first aid. The industrial safety evaluation is discussed in more detail in Section I.3 of the NI PEIS.

The average occupational total recordable cases and fatality rates for construction and operation activities are presented in **Table S–8**.

Table S–8 Average Occupational Total Recordable Cases and Fatality Rates (per worker-year)

Labor Category	Total Recordable Cases	Fatalities
Construction	0.053	1.3×10^{-4}
Operation	0.033	1.3×10^{-5}

The expected impacts (both annual and for the duration of the activity) to workers at each facility for construction and operation are presented in **Table S–9**.

Table S–9 Industrial Safety Impacts from Construction and Operation

Facility	Estimated Number of Workers	Construction or Operation Duration (years)	Expected Annual Total Recordable Cases	Expected Activity Duration Total Recordable Cases	Annual Fatalities	Activity Duration Fatalities
Construction						
Low-energy accelerator	75	3	4.0	12	0.010	0.030
High-energy accelerator	410	5	22	110	0.057	0.285
New research reactor	160	7	8.5	59.5	0.022	0.154
Operation						
ATR[a]	0	35	–	–	–	–
HFIR[a]	0	35	–	–	–	–
CLWR[a]	0	35	–	–	–	–
FFTF	242	35	8.0	280	0.0031	0.109
Low-energy accelerator	13	35	0.4	14	1.7×10^{-4}	0.00595
High-energy accelerator	225	35	7.4	259	0.0029	0.102
New research reactor	120	35	4.0	140	0.0016	0.056
REDC	116	35	3.8	133	0.0015	0.0525
FDPF	75	35	2.5	87.5	9.8×10^{-4}	0.0343
FMEF	105	35	3.5	123	0.0014	0.049
RPL/306–E	30	35	1.0	35	3.9×10^{-4}	0.0137
New support facility	100	35	3.3	116	0.0013	0.0455

a. No additional workers would be required for the proposed activities evaluated in the NI PEIS.

No fatalities would be expected from either construction or operation of any facility.

Comparison of Mission Effectiveness Among Alternatives

This section compares the effectiveness of Alternatives 1, 2, 3, and 4 in supporting the three missions evaluated in the NI PEIS:

- Medical and industrial isotope production
- Plutonium-238 production to support NASA space missions
- Nuclear energy research and development for civilian applications

Table S–10 lists the medical isotopes that were included in the Expert Panel's forecast of future demands (Wagner et al. 1998), and identifies their means of production using accelerators, reactors, or separation from

existing stockpiles of radioisotopes. Consistent with the panel's report, the list of isotopes is presented in three categories: proven medical isotopes currently used in clinical applications, those under development for clinical applications, and radioisotopes that have shown promise during medical research. Some are most suited for production in an accelerator, some in a nuclear reactor, and some are harvested by chemical separation from existing stockpiles of long-lived radioactive isotopes. Those isotopes that can be harvested from existing stockpiles of radioactive isotopes require only hot cells for the extraction process; neither accelerators or nuclear reactors are necessary for their production.

Table S–10 Medical Isotopes and Their Means of Production

Isotope[a]	Accelerator-Produced	Reactor-Produced	Separation from Existing Stockpiles of Radioactive Isotopes
Proven Isotopes Currently Used in Clinical Applications That Face Supply and Cost Concerns			
Yttrium-90	(b)	●	
Molybdenum-99[c]	(b)	●	
Indium-111	●		
Iodine-123	●		
Rhenium-186	(b)	●	
Developmental Isotopes for Clinical Applications That Face Availability and Cost Concerns			
Fluorine-18	●		
Phosphorus-32	(b)	●	
Krypton-81m	●		
Strontium-89	(b)	●	
Palladium-103	(b)	●	
Tin-117m	(b)	●	
Xenon-127	(b)	●	
Iodine-125	(b)	●	
Iodine-131	(b)	●	
Samarium-153	(b)	●	
Promising Research Isotopes That Are Not Being Explored Due to Lack of Availability or Cost			
Scandium-47	(b)	●	
Zinc-62	●		
Copper-64	●	●	
Copper-67	●	●	
Germanium-68	●		
Gadolinium-153	(b)	●	
Holmium-166	●	●	
Lutetium-177	(b)	●	
Rhenium-188	(b)	●	
Astatine-211	α		
Bismuth-212		●	●[d]
Bismuth-213	(b)	●	●[e]
Radium-223	(b)	●	●[f]

a. Wagner et al. 1998.
b. These isotopes are produced by neutron capture and could be produced in a high-energy accelerator. However, this capability has not been included in the design, analysis, or cost estimates of Alternative 3.
c. Sufficient supplies of this isotope are available from Canadian suppliers.
d. Bismuth-212 is a progeny of thorium-232.
e. Bismuth-213 is a progeny of uranium-233.
f. Radium-233 is a progeny of protactinium-231.

Key: α, efficient means of production with an alpha particle accelerator; ●, efficient means of production.

No single production method would satisfy all of the Expert Panel's projected requirements for medical isotopes. Isotopes produced by neutron capture are typically provided by a reactor, but could be produced by a high-energy accelerator with a spallation neutron source. Accelerator production of these isotopes would be relatively inefficient, and might not be practical to provide the large quantities needed to meet clinical demands. The proposed high-energy accelerator described in the NI PEIS could be modified to provide such capability, but this would add to the design, construction, and operating complexity, would require an increase in particle energy greater than 1 gigaelectron volts, and would increase the capital and operating costs.

Bismuth and radium isotopes, which were identified as promising medical isotopes by the Expert Panel, are currently harvested from existing stockpiles of long-lived radioisotopes and can also be readily produced in a reactor.

Alternative 1—Restart FFTF. FFTF would produce high-energy neutrons and a large flux level (10^{15} neutrons per square centimeter per second) that can be tailored to nearly any desired energy level. FFTF would provide the greatest flexibility for both isotope production and nuclear-based research and development among the baseline configurations for all of the proposed alternatives. Due to its large core size, flux spectrum, demonstrated testing capability, and rated power level, it would be able to concurrently support the projected plutonium-238 needs, production of medical and industrial isotopes, and civilian nuclear energy research and development related to a broad range of materials, advanced reactors, advanced fuels, and waste transmutation.

Alternative 2—Use Only Existing Operational Facilities. Due to current mission commitments at the existing DOE facilities, a large portion of the reactor irradiation space is committed to existing users. The existing reactors are able to provide for the current plutonium-238 needs. However, fulfilling this requirement with these facilities would use most, if not all, excess capacity, and may require some non-Federal missions to be terminated. The ability to expand medical and industrial isotope production would require some current missions to be postponed or terminated. If the CLWR were used for plutonium-238 production, then the existing facilities would gain additional margin for medical and industrial isotope production and limited civilian nuclear energy research and development activities. These facilities have primary missions with sponsors who reserve the right to dictate to what degree and the times the facility could be used.

Alternative 3—Construct New Accelerator(s). Two accelerators, a low-energy accelerator and a high-energy accelerator, are proposed for Alternative 3. The low-energy accelerator would serve as a dedicated isotope production facility. Due to the nature of this type of accelerator, it could only produce a limited number of the representative isotopes discussed in Section S.1, it has no ability to satisfy the plutonium-238 needs, and a limited ability to support the proposed nuclear-based research and development needs. The preconceptual design of the high-energy accelerator focused on supporting the plutonium-238 production mission. The design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce "tuneable" neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and material interactions. The changes required to add additional capability to the high-energy accelerator could be provided, but they would increase the size of the facility, add complexity to the facility design and operation, increase the cost of construction and operation, and potentially require more time for design and construction.

Alternative 4—Construct New Research Reactor. The proposed new research reactor would provide ample neutrons for the production of plutonium-238 and for many of the representative isotopes. The thermal flux would limit the new research reactor's ability to produce a number of isotopes requiring fast or high-energy neutrons. Its lower flux levels (10^{13} neutrons per square centimeter per second) and predominantly thermal flux would limit its ability to support many of the projected nuclear-based research and development needs.

S.7 CUMULATIVE IMPACTS

The projected environmental impacts of (1) constructing (as necessary) and operating the proposed facilities to store, fabricate, irradiate, and process the various targets addressed in the NI PEIS for 35 years and (2) deactivating FFTF were added to the environmental impacts of other present and reasonably foreseeable future actions at or near the identified sites to obtain cumulative site impacts under normal conditions. The other present and reasonably foreseeable future actions at or near the candidate sites are included in the baseline impacts presented in Chapter 3 of the NI PEIS. Cumulative transportation impacts were determined by analyzing the impacts along the various routes used to transport the materials associated with nuclear infrastructure activities over the 35-year period.

In this section, cumulative site impacts are presented only for those "resources" at a site that may reasonably be expected to be affected by the storage, fabrication, irradiation, and processing of the various targets. These include site employment, electrical consumption, water usage, air quality, waste management, and public and occupational health and safety. This section also includes the cumulative impacts associated with intersite transportation.

Impacts of the following are considered in the cumulative site impact assessment:

- Current (baseline) activities at or in the vicinity of the candidate sites
- Other onsite and offsite activities that are reasonably foreseeable and documented
- Construction (as necessary), operation, and deactivation (as necessary) of the proposed nuclear infrastructure facilities to fabricate, irradiate, and process targets

Details of activities that may be implemented in the foreseeable future at any of the nuclear infrastructure candidate sites and evaluated in the cumulative impact assessment are given in the following documents:

- *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999a) (Record of Decision issued)

- *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996a) (Record of Decision issued)

- *Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement* (DOE 1996b) (Record of Decision issued)

- *Final Programmatic Environmental Impact Statement for Tritium Supply and Recycling* (DOE 1995b) (Record of Decision issued)

- *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (DOE 1997a) (Records of Decision issued for the various waste types)

- *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995c) (Record of Decision issued)

- *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (DOE 1996c) (Record of Decision issued)

- *Final Programmatic Environmental Impact Statement for Stockpile Stewardship and Management* (DOE 1996d) (Record of Decision issued)

- *Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* (DOE 1999b) (Record of Decision issued)

- *Final Environmental Impact Statement for Management of Spent Nuclear Fuel from the K-Basins at the Hanford Site, Richland, Washington*, (DOE 1996e) (Record of Decision issued)

- *Final Environmental Impact Statement for the Tank Waste Remediation System, Hanford Site, Richland, Washington* (DOE 1996f) (Record of Decision issued)

- *Hanford Reach of the Columbia River Comprehensive River Conservation Study and Environmental Impact Statement, Final* (NPS 1994) (Record of Decision issued)

- *Final Hanford Comprehensive Land-Use Plan Environmental Impact Statement* (DOE 1999c) (Record of Decision issued)

- *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE 2000b) (Record of Decision issued)

- *Final Environmental Impact Statement, Construction and Operation of the Spallation Neutron Source* (DOE 1999d) (Record of Decision issued)

- *Final Programmatic Environmental Impact Statement for Alternative Strategies for the Long-Term Management and Use of Depleted Uranium Hexafluoride* (DOE 1999e) (Record of Decision issued)

- *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement* (DOE 1999f)

- *Final Environmental Impact Statement for Treating Transuranic (TRU)/Alpha Low-Level Waste at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE 2000c) (Record of Decision issued)

- *Environmental Assessment Melton Valley Storage Tanks Capacity Increase Project - Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE 1995d) (FONSI issued)

- *Environmental Assessment for Management of Spent Nuclear Fuel on the Oak Ridge Reservation, Oak Ridge, Tennessee* (DOE 1996g) (FONSI issued)

- *Environmental Assessment - Management of Hanford Site Non-Defense Production Reactor Spent Nuclear Fuel, Hanford Site, Richland, Washington* (DOE 1997b) (FONSI issued)

- *Environmental Assessment for Transportation of Low-Level Radioactive Waste from the Oak Ridge Reservation to Off-Site Treatment or Disposal Facilities* (DOE 2000d)

- *Environmental Assessment for Transportation of Low-Level Radioactive Mixed Waste from the Oak Ridge Reservation to Off-Site Treatment or Disposal Facilities* (DOE 2000e) (Draft issued)

- *Environmental Assessment for Selection and Operation of the Proposed Field Research Centers for the Natural and Accelerated Bioremediation Research (NABIR) Program* (DOE 2000f) (FONSI issued)

The related programs included in the cumulative impact assessment for the potentially affected candidate sites are identified in **Table S–11**.

Table S–11 Other Present and Reasonably Foreseeable Actions Considered in the Cumulative Impact Assessment

Activities	ORR	INEEL	Hanford
Disposition of Surplus Plutonium	X		
Storage and Disposition of Weapons-Usable Fissile Materials	X	X	X
Disposition of Surplus Highly Enriched Uranium	X		
Waste Management PEIS	X	X	X
Spent Nuclear Fuel Management and INEL Environmental Restoration and Waste Management		X	X
Foreign Research Reactor Spent Nuclear Fuel Management		X	X
Stockpile Stewardship and Management	X		
Tank Waste Remediation			X
Radioactive Releases from WNP Nuclear Power Plant			X
Hanford Reach of the Columbia River Comprehensive River Conservation Study			X
Hanford Comprehensive Land Use Plan			X
K Basins Spent Fuel Management			X
Advanced Mixed Waste Treatment Project		X	
Treatment and Management of Sodium-Bonded Spent Nuclear Fuel		X	
Construction and Operation of the Spallation Neutron Source	X		
Long-Term Management and Use of Depleted Uranium Hexafluoride	X		
Treatment and Shipment of Transuranic Waste	X		
Management of Liquid Low-Level Radioactive Waste	X		
Management of Spent Nuclear Fuel	X		
Transportation of Low-Level Radioactive Waste to Off-Site Treatment or Disposal	X		
Transportation of Low-Level Radioactive Mixed Waste to Off-Site Treatment or Disposal	X		
Natural and Accelerated Bioremediation Field Research Center Assessment	X		
Idaho High-Level Waste and Facilities Disposition		X	

Source: Table 4–162 of the NI PEIS.

In the tables that are included in the following sections, all relevant activities at each site are identified to the extent possible. They include existing and reasonably foreseeable activities, and those associated with nuclear infrastructure operations. The impacts associated with the latter are specifically shown as "New Nuclear Infrastructure Operations." They include the impacts from construction (as necessary), operation, and deactivation (as necessary) of the proposed target fabrication, irradiation, and processing facilities assessed in the NI PEIS.

A bounding option was analyzed for each site. The bounding option is the option that would involve the greatest amounts of operational activities and associated environmental impacts at the candidate site. For example, the bounding option for ORR is Option 7 of Alternative 2, under which both HFIR and REDC operations would be involved in plutonium-238 production.

In addition to reasonably foreseeable site activities, other activities within the regions of the candidate sites were considered in the cumulative impact analysis for the selected resources. However, because of the distances between the candidate sites and these other existing and planned facilities, there is little opportunity for interactions among them.

Cumulative Impacts at ORR

For ORR, the bounding option for the NI PEIS is Option 7 of Alternative 2. This option calls for the operation of HFIR to irradiate neptunium-237 targets and operation of REDC to fabricate and process these targets and other neptunium-237 targets irradiated in ATR. The impacts associated with HFIR and REDC operations for other missions are included under "existing site activities."

Resource Requirements. Cumulative impacts on resource requirements at ORR are presented in **Table S–12**. ORR would remain within its site capacity for all major resources. If Option 7 of Alternative 2 were implemented, the proposed nuclear infrastructure facilities would require essentially no change in the site's use of electricity or water. Cumulatively, ORR would use approximately 10 percent of its electrical capacity and 37 percent of its water capacity. Site employment would increase by approximately 41 workers.

Table S–12 Maximum Cumulative Resource Use and Impacts at ORR

Activities	Site Employment	Electrical Consumption (megawatt-hours per year)	Water Usage (million liters per year)
Existing site activities[a]	14,215	726,000	14,210
Storage and Disposition PEIS	Included above	7,260	0.24
Waste Management PEIS	1,259	84,160	394
Spallation Neutron Source	744	543,120	1,592
Treatment and Shipment of Transuranic Waste	17	3,000	3.8
New nuclear infrastructure operations[b]	41[c]	Negligible[d]	2.86
Total	16,276	~1,363,540	16,203
Total site capacity	NA	13,880,000	44,348

a. Reflects current sitewide activities that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
b. Nuclear infrastructure activities from Alternative 2, Option 7.
c. Some, or all, of these worker requirements may be filled by the reassignment of the existing site workforce.
d. Additional electricity consumption associated with this option would be negligible compared to that associated with existing facility activities.

Note: To convert from liters per year to gallons per year, multiply by 0.264; to convert from megawatt-hours to British thermal units, multiply by 3.42×10^6; ~ means "approximately" and indicates that new nuclear infrastructure operations would contribute only minimally.
Key: NA, not applicable.
Source: Table 4–163 of the NI PEIS.

Air Quality. Cumulative impacts on air quality at ORR are presented in **Table S–13**. ORR is currently in compliance with all Federal and state ambient air quality standards, and would continue to be in compliance even if the cumulative effects of all activities are included. As shown in the table, the contributions of nuclear infrastructure operations to overall site concentrations would be very small.

Table S–13 Maximum Cumulative Air Pollutant Concentrations at ORR for Comparison with Ambient Air Quality Standards

Parameter	Carbon Monoxide		Nitrogen Dioxide	PM_{10}		Sulfur Dioxide		
Averaging Period	8 hours	1 hour	Annual	Annual	24 hours	Annual	24 hours	3 hours
Activities								
Existing site activities[a] (micrograms per cubic meter)	7.75	26.5	0.98	1.6	12.6	4.76	33.4	106.4
HEU disposition[b] (micrograms per cubic meter)	11.5	53	1.33	0.03	0.37	2.46	29.3	161
Waste management program (micrograms per cubic meter)	0	0	0	3	9	2.4	11	39
Spallation Neutron Source (micrograms per cubic meter)	69	99	16	1.9	23	0.1	1	2.4
New nuclear infrastructure operations[c] (micrograms per cubic meter)	0	0	1.99×10^{-4}	0	0	0.04	0.31	0.7
Total concentration (micrograms per cubic meter)	88.3	179	18.3	6.53	45	9.76	75	310
Standard								
Most stringent standard[d] (micrograms per cubic meter)	10,000	40,000	100	50	150	80	365	1,300

a. Environmental impacts associated with existing site activities (based on 1998 emissions from the *Oak Ridge Reservation Annual Site Environmental Report 1998*) that are anticipated to continue during part or all of the 35-year period evaluated for nuclear infrastructure operations. The values in this row reflect a curtailment of stockpile stewardship management activities during this time period.
b. Highly enriched uranium disposition activities.
c. Nuclear infrastructure activities from Alternative 2, Option 7.
d. The more stringent of the Federal and state standards is presented if both exist for the averaging period.

Source: Table 4–164 of the NI PEIS.

Public and Occupational Health and Safety—Normal Operations. Cumulative impacts in terms of radiation exposure to the public and workers at ORR are presented in **Table S–14**. There would be no increase expected in the number of latent cancer fatalities in the population from ORR site operations if nuclear infrastructure operations were to occur at HFIR and REDC. The dose limits for individual members of the public are given in DOE Order 5400.5. As discussed in that Order, the dose limit from airborne emissions is 10 millirem per year, as required by the Clean Air Act; the dose limit from drinking water is 4 millirem per year, as required by the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. Therefore, as is evident in Table S–14, the dose to the maximally exposed individual would be expected to remain well within the regulatory limits. Onsite workers would be expected to see an increase of approximately 0.17 latent cancer fatality due to radiation from nuclear infrastructure operations over the 35-year operational period.

Table S–14 Maximum Cumulative Radiation Impacts at ORR

Impact	Maximally Exposed Individual		Population Dose Within 80 Kilometers (50 Miles) (Year 2020)		Total Site Workforce	
	Annual Dose (millirem per year)	Risk of a Latent Cancer Fatality[a]	Dose (person-rem)	Number of Latent Cancer Fatalities[a]	Dose (person-rem per year)	Number of Latent Cancer Fatalities[a]
Existing site activities[b]	4.4	7.7×10^{-5}	60.3	1.1	103	1.4
HEU disposition	0.039	6.8×10^{-7}	0.16	0.0028	11	0.16
Stockpile stewardship and management	0.2	3.5×10^{-6}	0.6	0.011	-1.8	-0.025
Waste management	0.35	6.1×10^{-6}	1.2	0.021	0.45	0.0063
New nuclear infrastructure operations at ORR[c]	1.9×10^{-6}	3.3×10^{-11}	8.8×10^{-5}	1.5×10^{-6}	12	0.168
Total	5.0[d]	8.7×10^{-5}[(d)]	62	1.1	125	1.7

a. These values are calculated based on a 35-year exposure period.
b. Environmental impacts associated with present activities at ORR that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Impacts are bounded by Option 7 of Alternative 2.
d. The same individual would not be expected to be the maximally exposed individual for all activities at ORR. The location of the maximally exposed individual depends upon where on the site an activity is performed. However, to provide an upper bound of the cumulative impacts to the maximally exposed individual, the impacts from each activity have been summed.

Source: Table 4–165 of the NI PEIS.

Waste Management. Cumulative amounts of wastes generated at ORR are presented in **Table S–15**. It is unlikely that there would be major impacts on waste management at ORR because sufficient capacity would exist to manage the site wastes. As discussed in Section 4.3.1.1.13 of the NI PEIS, irrespective of how the waste from processing irradiated neptunium-237 targets is classified (i.e., transuranic or high-level radioactive), the waste composition and characteristics are the same, and the management (i.e., treatment and onsite storage), as described in the NI PEIS, would be the same. In addition, either waste type would require disposal in a suitable repository. None of the options assessed in the NI PEIS would generate more than a small amount of additional waste at ORR.

Cumulative Impacts at INEEL

For INEEL, the bounding option for the NI PEIS is Option 2 of Alternative 2. This option calls for the operation of ATR to irradiate neptunium-237 targets and operation of FDPF to fabricate and process these targets. The impacts associated with ATR and FDPF operations for other missions are included under "existing site activities."

Resource Requirements. Cumulative impacts on resource requirements at INEEL are presented in **Table S–16**. INEEL would remain within its site capacity for all major resources. If Option 2 of Alternative 2 were implemented, the proposed nuclear infrastructure facilities would require essentially no change in the site's use of electricity or water. Cumulatively, INEEL would use 80 percent of its electrical capacity and 13 percent of its water capacity. Site employment would increase by approximately 24 workers.

Table S–15 Cumulative Impacts on Waste Management Activities at ORR Over the 35-Year Period (cubic meters)

Waste Type	Existing Site Activities	Treatment and Shipment of Transuranic Waste[a]	Surplus Plutonium Disposition[b]	Spallation Neutron Source[c]	New Nuclear Infra-structure Operations[d]	Total	Site Capacity[e]		
							Treatment (cubic meters/ year)	Storage	Disposal
Transuranic (High-level radio-active)[f]	766 (0)	607 (0)	11 (0)	0 (0)	385 (385)	1,769 (385)	4,050/ 5 years (0)	2,845 (0)	NA (NA)
Low-level radioactive	335,755	2,778	140	612,000	<2,145	~952,818	440,405	87,776	NA
Mixed low-level radioactive	28,035	23	1	623	<175	~28,857	263,560	234,226	NA
Hazardous[g] (kilograms)	1,260,000	0	1	1,435,000	227,500	2,922,501	1,738,803	7,312	NA
Nonhazardous									
Liquid	23,845,500	1,560	1,500	2,415	99,925	23,950,900	3,395,918	NA	NA
Solid	2,590,000	5,500	130	47,215	5,180	2,648,025	NA	NA	1,219,000

a. Data from the *Final Environmental Impact Statement for Treating Transuranic (TRU)/Alpha Low-Level Waste at the Oak Ridge National Laboratory* Low-Temperature Drying Alternative was selected in the Record of Decision (65 FR 48683).
b. Data from the *Surplus Plutonium Disposition Final EIS* (DOE 1999a:4-394) postirradiation examination (2006 through 2009) and selected in Record of Decision (65 FR 1608).
c. Data from the *Spallation Neutron Source Final EIS*.
d. Option 7 of Alternative 2. This alternative would generate the most waste for all waste types.
e. Total 35-year and annual capacity derived from Table 3–13 of the NI PEIS.
f. Volumes in parentheses represent high-level radioactive waste. Section 4.3.1.1.13 of the NI PEIS provides a discussion on classification of waste from processing irradiated neptunium-237 targets.
g. Assumes for hazardous waste that 353 kilograms equal 1 cubic meter (22.0 pounds equal 1 cubic foot).

Note: To convert from cubic meters to cubic yards, multiply by1.308; < means "less than"; ~ means "approximately;" NA, not applicable (i.e., the majority of the waste is not routinely treated, stored, or disposed of on site).
Source: Table 4–166 of the NI PEIS.

Table S–16 Maximum Cumulative Resource Use and Impacts at INEEL

Activities	Site Employment	Electrical Consumption (megawatt-hours per year)	Water Usage (million liters per year)
Existing site activities[a]	7,993	232,500	4,830
SNF Management and INEL Environmental Restoration and Waste Management	–	2,200	2
Foreign Research Reactor SNF Management	–	1,000	2
Waste Management PEIS	–	13,980	194
Advanced Mixed Waste Treatment Project	–	33,000	16
High-Level Waste and Facilities Disposition	–	33,000	351
New nuclear infrastructure operations[b]	24[c]	Negligible[d]	1.68
Total	8,017	~315,680	5,397
Total site capacity	NA	394,200	43,000

a. Reflects current sitewide activities (except that "Site Employment" value also reflects projected employment from other activities) that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
b. Nuclear infrastructure activities from Alternative 2, Option 2.
c. Some, or all, of those worker requirements may be filled by the reassignment of the existing workforce.
d. Additional electricity consumption associated with this option would be negligible compared to that associated with existing facility activities.

Note: To convert from liters per year to gallons per year, multiply by 0.264; to convert from megawatt-hours to British thermal units, multiply by 3.42×10^6; ~ means "approximately," and indicates that new nuclear infrastructure operations would contribute only minimally.
Key: NA, not applicable; SNF, spent nuclear fuel.
Source: Table 4–167 of the NI PEIS.

Air Quality. Cumulative impacts on air quality at INEEL are presented in **Table S–17**. INEEL is currently in compliance with all Federal and state ambient air quality standards, and would continue to remain in compliance, even with consideration of the cumulative effects of all activities. The contributions of nuclear infrastructure operations to overall site concentrations are expected to be very small.

Table S–17 Maximum Cumulative Air Pollutant Concentrations at INEEL for Comparison with Ambient Air Quality Standards

Parameter	Carbon Monoxide		Nitrogen Dioxide	PM_{10}		Sulfur Dioxide		
Averaging Period	8 hours	1 hour	Annual	Annual	24 hours	Annual	24 hours	3 hours
Activities								
Existing site activities[a] (micrograms per cubic meter)	78	206	0.46	0.49	12	0.14	5.3	24
ANL–W contribution[b] (micrograms per cubic meter)	41	59	13	0.14	1.1	3.3	27	60
Advanced Mixed Waste Treatment Project[c] (micrograms per cubic meter)	0.85	115	0.34	0.006	4.6	0.012	4.5	25
HLW & facilities disposition[d] (micrograms per cubic meter)	4.2	10	0.19	0.02	0.28	0.57	8.9	42
New nuclear infrastructure operations[e] (micrograms per cubic meter)	0	0	3.66×10^{-4}	0	0	0.024	0.19	0.43
Total concentration (micrograms per cubic meter)	124	390	14	0.656	18	4.05	45.9	151
Standard								
Most stringent standard[f] (micrograms per cubic meter)	10,000	40,000	100	50	150	80	365	1,300

a. Environmental impacts associated with existing site activities (excluding activities at ANL–W) as shown in the *Idaho High-Level Waste and Facilities Disposition Draft EIS*, and in the *Final EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*. The activities whose concentrations are given in this row are anticipated to continue during part or all of the 35-year period evaluated for proposed nuclear infrastructure operations.
b. The contribution from existing ANL–W sources as shown the *Final EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*.
c. *Advanced Mixed Waste Treatment Project EIS* activities—proposed action with microencapsulation or vitrification.
d. High-level waste and facilities disposition site boundary contribution for planning basis option.
e. Nuclear infrastructure activities from Alternative 2, Option 2.
f. The more stringent of the Federal and state standards is presented if both exist for the averaging period.

Key: ANL–W, Argonne National Laboratory–West; HLW, high-level radioactive waste.
Source: Table 4–168 of the NI PEIS.

Public and Occupational Health and Safety—Normal Operations. Cumulative impacts in terms of radiation exposure to the public and workers at INEEL are presented in **Table S–18**. There would be no increase expected in the number of latent cancer fatalities in the population from INEEL site operations if nuclear infrastructure operations were to occur at ATR and FDPF. The dose limits for individual members of the public are given in DOE Order 5400.5. As discussed in that Order, the dose limit from airborne emissions is 10 millirem per year, as required by the Clean Air Act; the dose limit from drinking water is 4 millirem per year, as required by the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. Therefore, as is evident in Table S–18, the dose to the maximally exposed individual would be expected to remain well within the regulatory limits. Onsite workers would be expected to see an increase of approximately 0.17 latent cancer fatality due to radiation from nuclear infrastructure operations over the 35-year operational period.

Table S–18 Maximum Cumulative Radiation Impacts at INEEL

Impact	Maximally Exposed Individual		Population Dose Within 80 Kilometers (50 Miles) (Year 2020)		Total Site Workforce	
	Annual Dose (millirem per year)	Risk of a Latent Cancer Fatality[a]	Dose (person-rem)	Number of Latent Cancer Fatalities[a]	Dose (person-rem per year)	Number of Latent Cancer Fatalities[a]
Existing site activities[b]	0.008	1.7×10^{-7}	0.075	0.0013	64.9	0.91
Storage and disposition	1.6×10^{-6}	2.8×10^{-11}	1.8×10^{-5}	3.2×10^{-7}	25	0.35
Foreign research reactor spent nuclear fuel	5.6×10^{-4}	9.8×10^{-9}	0.0045	7.9×10^{-5}	33	0.46
Spent nuclear fuel	0.008	1.4×10^{-7}	0.19	0.0033	5.4	0.076
Advanced Mixed Waste Treatment Project	0.022	3.9×10^{-7}	0.009	1.6×10^{-4}	4.1	0.057
High-level waste and facilities disposition	0.002	3.5×10^{-8}	0.10	0.0018	59	0.83
Sodium-bonded spent nuclear fuel	0.002	3.5×10^{-8}	0.012	2.1×10^{-4}	22	0.31
New nuclear infrastructure operations at ATR and FDPF[c]	2.6×10^{-7}	4.6×10^{-12}	3.9×10^{-6}	6.8×10^{-8}	12	0.17
Total	0.043[d]	7.4×10^{-7}[(d)]	0.39	0.0068	225.4	3.16

a. These values are calculated based on a 35-year exposure period.
b. Environmental impacts associated with present activities at INEEL that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Impacts are bounded by Option 2 of Alternative 2.
d. The same individual would not be expected to be the maximally exposed individual for all activities at INEEL. The location of the maximally exposed individual depends upon where on the site an activity is performed. However, to provide an upper bound of the cumulative impacts to the maximally exposed individual, the impacts from each activity have been summed.

Source: Table 4–169 of the NI PEIS.

Waste Management. Cumulative amounts of wastes generated at INEEL are presented in **Table S–19**. It is unlikely that there would be major impacts on waste management at INEEL because sufficient capacity would exist to manage the site wastes. As discussed in Section 4.3.2.1.13, irrespective of how the waste from processing of irradiated neptunium-237 targets is classified (i.e., transuranic or high-level radioactive), the waste composition and characteristics are the same, and the management (i.e., treatment and onsite storage), as discussed in the NI PEIS, would be the same. In addition, either waste type would require disposal in a suitable repository. None of the alternatives assessed in the NI PEIS would generate more than a small amount of additional waste at INEEL.

Table S–19 Cumulative Impacts on Waste Management Activities at INEEL Over the 35-Year Period (cubic meters)

Waste Type	Existing Site Activities	Idaho HLW and Facility Disposition EIS[a]	Treatment and Management of Sodium-Bonded SNF[b]	New Nuclear Infrastructure Operations[c]	Total	Site Capacity[d]		
						Treatment (cubic meters/ year)	Storage	Disposal (cubic meters/ year)
Transuranic (High-level radioactive)[e]	65,000[f] (0)	110 (0)	14 (0)	245 (245)	65,369 (245)	57,794 (6,434)	190,319 (19,483)	NA (NA)
Low-level radioactive	135,600	15,325	862	<2,320	~154,107	42,363	177,493	69,530
Mixed low-level radioactive	3,767	12,837	40	<175	~16,819	157,092	187,761	NA
Hazardous	1,180	2,457	0	227,500 kilograms (644 cubic meters)[g]	4,281	NA	9,619	NA
Nonhazardous	124,905	145,262	4,960	64,015	339,142	3,200,000	NA	3,062,000

a. Data from the *Idaho HLW and Facility Disposition EIS*, Separations Alternative. Maximum quantities for any alternative.
b. Data from the *Treatment and Management of Sodium-Bonded Spent Nuclear Fuel EIS*, Alternative 1, electrometallurgically treat blanket and driver fuel at ANL–W; 12 years of operation and selected in the Record of Decision (65 FR 56565).
c. Option 2 of Alternative 2 would generate the most waste for all waste types.
d. Total 35-year and annual capacity derived from Table 3–27 of the NI PEIS.
e. Volumes in parentheses represent high-level radioactive waste. Section 4.3.2.1.13 of the NI PEIS provides a discussion on classification of waste from processing irradiated neptunium-237 targets.
f. This 65,000 cubic meters is in storage at the Radioactive Waste Management Complex.
g. Assumes for hazardous waste that 353 kilograms equals 1 cubic meter (22.0 pounds equals 1 cubic foot).

Note: To convert from cubic meters to cubic yards, multiply by 1.308; HLW means high-level radioactive waste; SNF means spent nuclear fuel; < means "less than"; ~ means "approximately;" NA, not applicable (i.e., the majority of the waste is not routinely treated, stored, or disposed of on site).
Source: Table 4–170 of the NI PEIS.

Cumulative Impacts at Hanford

For Hanford, the bounding option for the NI PEIS depends on the parameter assessed. For example, under Public and Occupational Health and Safety, the highest radiological doses and associated latent cancer fatalities to the public would be associated with Option 1 of Alternative 1, whereas the highest doses and latent cancer fatalities to workers would be associated with Option 3 of this same alternative. Processing of targets in RPL versus processing in FMEF accounts for there being different bounding options. For each of the parameters addressed in this section, a footnote is included in each of the cumulative impact tables, as necessary, to indicate the bounding alternative/option.

Resource Requirements. Cumulative impacts on resource requirements at Hanford are presented in **Table S–20**. Hanford would remain within its site capacity for all major resources. If any of the options under Alternative 1 were implemented, the proposed nuclear infrastructure facilities would require a small increase in the site's use of electricity and water. For the bounding options identified in Table S–20, this would reflect an increase of about 2 and 1 percent, respectively, over current baseline utilization for these resources. There would be no additional land disturbance or development. Cumulatively, Hanford would use approximately, 23 percent of its electrical capacity and 38 percent of its water capacity. Site employment would increase by approximately 130 workers.

Air Quality. Cumulative impacts on air quality at Hanford are presented in **Table S–21**. Hanford is currently in compliance with all Federal and state ambient air quality standards, and would continue to be in compliance even with consideration of the cumulative effects of all activities. The nuclear infrastructure contributions to overall site concentrations are expected to be very small.

Table S–20 Maximum Cumulative Resource Use and Impacts at Hanford

Activities	Site Employment	Electrical Consumption (megawatt-hours per year)	Water Usage (million liters per year)
Existing site activities[a]	16,005	323,128	2,754[b]
Tank waste remediation system	–	170,000	200
Waste Management PEIS	–	13,920	133
New nuclear infrastructure operations[c]	130[d]	55,000	80
Total	16,135	562,048	3,167
Total site capacity	NA	2,484,336	8,263[b]

a. Reflects current sitewide activities. The"Site Employment" value also reflects projected employment from other activities that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
b. Reflects domestic/potable water only and not raw water usage or availability.
c. Electrical consumption and water usage are bounded by Option 3 or 6 of Alternative 1, with the values reflecting the increase over standby operations from restart of FFTF and associated support activities in FMEF.
d. Some, or all, of these worker requirements may be filled by the reassignment of the existing site workforce.

Note: To convert from liters per year to gallons per year, multiply by 0.264; to convert from megawatt-hours to British thermal units, multiply by 3.42×10^{6}.
Key: NA, not applicable.
Source: Table 4–171 of the NI PEIS.

Table S–21 Maximum Cumulative Air Pollutant Concentrations at Hanford for Comparison with Ambient Air Quality Standards

Parameter	Carbon Monoxide		Nitrogen Dioxide	PM_{10}		Sulfur Dioxide			
Averaging Period	8 hours	1 hour	Annual	Annual	24 hours	Annual	24 hours	3 hours	1 hour
Activities									
Existing site activities[a] (micrograms per cubic meter)	27.3	63.3	0.666	0.0182	1.01	0.175	30.17	69.4	79.4
Tank waste remediation[b] (micrograms per cubic meter)	34	48	0.12	0.0079	0.75	0.020	1.6	3.6	4
Spent nuclear fuel management[c] (micrograms per cubic meter)	0	0	0.1	0	0	0	0	0	0
New nuclear infrastructure FFTF operations[d] (micrograms per cubic meter)	52.1	74.4	0.0118	8.39×10^{-4}	9.84	0.00785	9.11	20.5	22.8
New nuclear infrastructure FMEF operations[d] (micrograms per cubic meter)	0	0	4.43×10^{-5}	0	0	0.0087	0.069	0.16	0.17
Total concentration (micrograms per cubic meter)	113.4	185.7	0.90	0.027	11.6	0.212	40.9	93.7	106
Standard									
Most stringent standard[e] (micrograms per cubic meter)	10,000	40,000	100	50	150	50	260	1,300	660

a. Environmental impacts associated with existing activities. These activities are anticipated to continue during part or all of the 35-year period evaluated for proposed nuclear infrastructure operations.
b. *Hanford Tank Waste Remediation EIS* activities, vitrification facilities, Phased Implementation – Phase II Operation.
c. *Spent Nuclear Fuel Management* – regionalization alternative.
d. Nuclear infrastructure contributions are bounded by Alternative 1, Option 3. Periodic testing of emergency diesel generators would result in higher values for certain pollutants and time periods.
e. The more stringent of the Federal and State standards is presented if both exist for the averaging period.

Note: The contribution from activities in the *Final Waste Management Programmatic EIS* are small and are not shown.
Source: Table 4–172 of the NI PEIS.

Public and Occupational Health and Safety—Normal Operations. Cumulative impacts in terms of radiation exposure to the public and workers at Hanford are presented in **Table S–22**. There would be no increase expected in the number of latent cancer fatalities in the population from Hanford site operations if nuclear infrastructure operations were to occur at FMEF. The dose limits for individual members of the public are given in DOE Order 5400.5. As discussed in that order, the dose limit from airborne emissions is 10 millirem per year, as required by the Clean Air Act; the dose limit from drinking water is 4 millirem per year, as required by the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. Therefore, as is evident in Table S–22, the dose to the maximally exposed individual would be expected to remain well within the regulatory limits. Onsite workers would be expected to see an increase of approximately 0.26 latent cancer fatality due to radiation from nuclear infrastructure operations over the 35-year operational period.

Table S–22 Maximum Cumulative Radiation Impacts at Hanford

	Maximally Exposed Individual		Population Dose Within 80 Kilometers (50 Miles) (Year 2020)		Total Site Workforce	
Impact	Annual Dose (millirem per year)	Risk of a Latent Cancer Fatality[a]	Dose (person-rem)	Number of Latent Cancer Fatalities[a]	Dose (person-rem per year)	Number of Latent Cancer Fatalities[a]
Existing site activities[b]	0.02	3.5×10^{-7}	0.6	0.011	181	2.5
Waste management	0.0057	2.9×10^{-9}	0.28	0.0014	1,300	5.2
Tank remediation	(c)	2.4×10^{-6}	(c)	0.19	(c)	3.27
Spent nuclear fuel management	(c)	1.4×10^{-8}	(c)	8.0×10^{-4}	(c)	0.057
Burial of low-level waste	0	0	0	0	1,018	0.41
Plutonium Finishing Plant stabilization	0.13	3.9×10^{-7}	2.3	0.007	157	0.38
New nuclear infrastructure operations at FFTF and FMEF or RPL[d]	0.0054	9.5×10^{-8}	0.25	0.0044	18	0.26
Total	(e)	$3.3\times10^{-6(f)}$	(e)	0.21	(e)	12

a. These values are calculated based on a 35-year exposure period except for waste management (project duration for waste transfer of 10 years) and Plutonium Finishing Plant stabilization (a 6-year project).
b. Environmental impacts associated with present activities at Hanford (including activities at other non-DOE facilities at or near Hanford) that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Source document provides project total; annual values are not constant.
d. Impacts on the public are bounded by Option 1 of Alternative 1; impacts on workers are bounded by Option 3 of Alternative 1.
e. Some source documents did not provide dose values, only expected latent cancer fatalities. Therefore, no total dose estimates have been developed.
f. The same individual would not be expected to be the maximally exposed individual for all activities at Hanford. The location of the maximally exposed individual depends upon where an activity is performed on the site. However, to provide an upper bound cumulative impact for the maximally exposed individual, the impacts from each activity have been summed.

Source: Table 4–173 of the NI PEIS.

Waste Management. Cumulative amounts of wastes generated at Hanford are presented in **Table S–23**. It is unlikely that there would be major impacts on waste management at Hanford because sufficient capacity would exist to manage the site wastes. As discussed in Sections 4.3.3.1.13 and 4.4.3.1.13 of the NI PEIS, irrespective of how the waste from processing of irradiated neptunium-237 targets is classified (i.e., transuranic or high-level radioactive), the waste composition and characteristics are the same, and the management

(i.e., treatment and onsite storage), as described in the NI PEIS, would be the same. In addition, either waste type would require disposal in a suitable repository. None of the alternatives assessed in the NI PEIS would generate more than a relatively small amount of additional waste at Hanford.

Table S–23 Cumulative Impacts on Waste Management Activities at Hanford Over the 35-Year Period (cubic meters)

Waste Type	Existing Site Activities	New Nuclear Infrastructure Operations	Total	Site Capacity[a]		
				Treatment (cubic meters/ year)	Storage	Disposal
Transuranic (High-level radioactive)[b]	9,880 (0)	385[c] (385)	10,265 (385)	98,520 (50,000)	17,216 (146,000)	NA (NA)
Low-level radioactive	95,666	5,015[c]	100,681	398,112	99,910	1,970,000
Mixed low-level radioactive	46,207	315[c]	46,522	413,211	100,483	14,200
Hazardous	19,600	3,100[d]	22,700	NA	NA	NA
Nonhazardous						
Liquid	7,000,000	1,494,500[c]	8,494,500	120,000	NA	4,807,720
Solid	1,505,000	10,500[c]	1,515,500	NA	NA	NA

a. Total 35-year and annual capacity derived from Table 3–36 of the NI PEIS.
b. Volumes in parentheses represent high-level radioactive waste. Sections 4.3.3.1.13 and 4.4.3.1.13 of the NI PEIS provide a discussion on classification of waste from processing irradiated neptunium-237 targets.
c. The bounding alternative for this waste type is Alternative 1, Option 3 or 6.
d. The bounding alternative for this waste type is Alternative 2, Option 3, 6 or 9; Alternative 3, Option 3; or Alternative 4, Option 3; which all include the deactivation of FFTF and neptunium-237 target fabrication and processing at FMEF. The inventory of bulk metallic sodium is not included because alternative sponsors and/or users will be found for its disposition.

Note: To convert from cubic meters to cubic yards, multiply by1.308; < means "less than"; ~ means "approximately"; NA, not applicable.
Source: Table 4–174 of the NI PEIS.

Spent Nuclear Fuel Management. The operation of FFTF for the proposed mission at 100 megawatts for 35 years under Alternative 1 would produce a total of about 16 metric tons of heavy metal (35,200 pounds) of spent nuclear fuel. The existing spent nuclear fuel at Hanford is about 2,133 metric tons of heavy metal (4,700,000 pounds) (DOE 1995c). The management of the existing spent nuclear fuel at Hanford results in a dose of less than 0.1 millirem per year to the maximally exposed member of the public. This dose is well within the DOE dose limits cited in DOE Order 5400.5. DOE has committed to remove the spent nuclear fuel at Hanford for ultimate disposition in a geologic repository. The restart of FFTF under Alternative 1 would generate 16 metric tons of heavy metal of spent nuclear fuel, which is less than 1 weight-percent of the total spent nuclear fuel inventory presently at Hanford. Only a small fraction of the dose shown for nuclear infrastructure operations would be attributable to the management of this spent nuclear fuel at FFTF. The doses at Hanford, including those associated with spent nuclear fuel management, would remain within the DOE dose limits.

Cumulative Impacts at the Generic CLWR Site

No incremental environmental impacts at the generic site would be expected with the normal operation of a CLWR to irradiate targets. Therefore, the cumulative impacts at the generic CLWR site would not be affected by any action assessed in the NI PEIS, and are not addressed further.

Cumulative Impacts at the New Accelerator(s) Generic DOE Site

Cumulative impacts cannot be presented for a generic site. If Alternative 3 were selected for implementation, a subsequent site-specific analysis would be conducted for the DOE site chosen for the combination of new accelerator(s) and support facility or research reactor only, and appropriate NEPA documentation would be prepared to address the cumulative impacts for that site.

Cumulative Impacts at the New Research Reactor Generic DOE Site

Cumulative impacts cannot be presented for a generic site. If Alternative 4 were selected for implementation, a subsequent site-specific analysis would be conducted for the DOE site chosen for the new research reactor and support facility or research reactor only, and appropriate NEPA documentation would be prepared to address the cumulative impacts for that site.

Cumulative Impacts of Transportation

Because likely transportation routes cross many states, cumulative impacts are compared on a national basis. Under all alternatives assessed in the NI PEIS, occupational radiation exposure to transportation workers and exposure to the public are estimated to each represent less than 0.05 percent of the cumulative exposures from nationwide transportation over the 35-year period of nuclear infrastructure activities. No additional traffic fatality is expected; the increase in traffic fatalities would be less than 0.0001 percent per year.

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DOE/EIS-0310
December 2000

FINAL Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility

Volume 1 Chapters 1 through 9

United States Department of Energy
Office of Nuclear Energy, Science and Technology
Washington, DC

Cover photograph and illustration identification, beginning at top center and going clockwise:

- Radioisotope tagged monoclonal antibodies, “smart bullets,” target malignant cells for diagnosis and treatment of diseases
- The Fast Flux Test Facility at the Hanford Site near Richland, Washington
- Illustration of a satellite that could use radioisotope power systems
- The High Flux Isotope Reactor at the Oak Ridge National Laboratory near Oak Ridge, Tennessee
- The Advanced Test Reactor at the Idaho National Engineering and Environmental Laboratory near Idaho Falls, Idaho
- Tip of a remote-handling arm, used for work in developing industrial and medical isotopes

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Printed with soy ink on recycled paper

Department of Energy
Washington, DC 20585

November 28, 2000

Dear Interested Party:

The *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* [NI PEIS]) (DOE/EIS-0310) has now been completed. This document has been prepared in accordance with the National Environmental Policy Act and reflects consideration of comments received on the draft NI PEIS released in July 2000.

The Department of Energy (DOE) is responsible for undertaking research and development activities related to development of nuclear power for civilian use, meeting the nuclear material needs of other Federal agencies, and ensuring the availability of isotopes for medical, industrial, and research applications. The NI PEIS presents an evaluation of the potential environmental impacts associated with the proposed expansion of the nuclear irradiation capabilities for accomplishing civilian nuclear energy research and development activities, accommodating the projected growth in demand for medical and industrial isotopes, and production of plutonium-238 to support future National Aeronautics and Space Administration space exploration missions. In addition to the "No Action" alternative, DOE evaluated other alternatives that include using operating facilities within the DOE complex, building a new research reactor, building one or two accelerators, and restarting the Fast Flux Test Facility (FFTF) that is currently in standby status. In addition, the NI PEIS includes an alternative to permanently deactivate FFTF.

After careful consideration of public comments, environmental impacts, and programmatic objectives, DOE's preferred alternative is to use its existing nuclear facility infrastructure to the extent possible to pursue the missions outlined in the PEIS, i.e., Alternative 2, Option 7. DOE would reestablish domestic production of plutonium-238, as needed, using the Advanced Test Reactor in Idaho and the High Flux Isotope Reactor in Tennessee and would process irradiated plutonium-238 targets at the Radiochemical Engineering Development Center in Tennessee. DOE would permanently deactivate FFTF under the "Preferred Alternative." Lack of clear commitments from likely users discouraged the Department from planning to build new facilities or to restart the FFTF. Further details on the Preferred Alternative can be found in the summary and in section 2.8 of volume 1 of this NI PEIS.

We appreciate your continued participation in this decision-making process.

Sincerely,

Robert Carl Knipp

for William D. Magwood, IV, Director
Office of Nuclear Energy, Science
and Technology

Cover Sheet

Responsible Agency: United States Department of Energy (DOE)

Title: *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (NI PEIS)

Locations: Idaho, Tennessee, Washington

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Abstract: Under the authority of the Atomic Energy Act of 1954, as amended, the DOE is responsible for ensuring the availability of isotopes for medical, industrial and research applications, meeting the nuclear material needs of other Federal agencies, and undertaking research and development activities related to development of nuclear power for civilian use. To meet these responsibilities, DOE maintains nuclear infrastructure capabilities that support various missions. Current estimates for the future needs of medical and industrial isotopes, plutonium-238, and research requirements indicate that the current infrastructure may soon be insufficient to meet the projected demands. DOE proposes to enhance these capabilities to provide for: (1) production of isotopes for medical and industrial uses, (2) production of plutonium-238 for use in advanced radioisotope power systems for future National Aeronautics and Space Administration (NASA) space exploration missions, and (3) the Nation's nuclear research and development needs for civilian application.

This NI PEIS evaluates the environmental impacts of a No Action Alternative (maintaining status quo), four alternative strategies to accomplish this mission, and an alternative to permanently deactivate the Fast Flux Test Facility (FFTF), with no new missions. Alternatives 2, 3, and 4 also include permanent deactivation of FFTF. The alternatives are:

No Action
1. Restart FFTF at Hanford, Washington
2. Use only existing operational facilities
3. Construct one or two new accelerators
4. Construct a new research reactor
5. Permanently deactivate FFTF (with no new missions)

The Preferred Alternative is Alternative 2, Option 7, Use Only Existing Operational Facilities. DOE would reestablish domestic production of plutonium-238, as needed, using the Advanced Test Reactor in Idaho and the High Flux Isotope Reactor in Tennessee, and would process irradiated plutonium-238 targets at the Radiochemical Engineering Development Center in Tennessee. DOE would permanently deactivate FFTF under the Preferred Alternative.

Public Comments: The Draft NI PEIS was issued for public review and comment on July 21, 2000. The comment period ended on September 18, 2000, although late comments were considered to the extent practicable. Public hearings were held to obtain comments on the Draft NI PEIS in Oak Ridge, Tennessee; Idaho Falls, Idaho; Hood River and Portland, Oregon; Seattle and Richland, Washington; and Arlington, Virginia. All comments were considered by DOE in preparing the Final NI PEIS, which also incorporates any new information received since issuance of the Draft NI PEIS. In response to comments on the Draft NI PEIS and as a result of information that was unavailable at the time of the issuance of the Draft PEIS, the Final PEIS contains revisions and new information, indicated by a sidebar in the margin. Volume 3 contains the comments received during the public review period for the Draft NI PEIS and DOE's responses to these comments. DOE will use the analyses presented in the Final NI PEIS as well as other information, including public input, costs, nonproliferation impacts, schedules, technical assurance, and other policy and programmatic objectives, in preparing the Record of Decision for accomplishing expanded civilian nuclear energy research and development and isotope production missions in the United States, including the role of FFTF. DOE will issue the Record of Decision no sooner than 30 days after the U.S. Environmental Protection Agency publishes a notice of availability of the Final NI PEIS in the Federal Register.

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AAA	Advanced Accelerator Applications
ALARA	as low as is reasonably achievable
ALOHA	Area Locations of Hazardous Atmospheres
ATR	Advanced Test Reactor
ATW	Accelerator Transmutation of Waste
BEIR	Biological Effects of Ionizing Radiation
BLIP	Brookhaven LINAC Isotope Producer
CEQ	Council on Environmental Quality
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CLWR	commercial light water reactor
CPP–651	Chemical Processing Plant 651 (Building CPP–651)
dBA	decibels A-weighted
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
FDPF	Fluorinel Dissolution Process Facility
FFTF	Fast Flux Test Facility
FMEF	Fuels and Materials Examination Facility
FONSI	Finding of No Significant Impact
FPF	Fuel Processing Facility
FR	Federal Register
g	gravitational acceleration
Hanford	Hanford Site
HB	horizontal beam
HEPA	high-efficiency particulate air (filter)
HFIR	High Flux Isotope Reactor
HVAC	heating, ventilating, and air conditioning
I^4	International Isotopes Idaho, Inc.
IAEA	International Atomic Energy Agency
IEM	Interim Examination and Maintenance Cell
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
IPF	Isotope Production Facility
ISC3	Industrial Source Complex
ISCST3	Industrial Source Complex (short-term model)
LANL	Los Alamos National Laboratory
LANSCE	Los Alamos Neutron Science Center
LCF	latent cancer fatality
LINAC	linear accelerator
MASF	Maintenance and Storage Facility
NAAQS	National Ambient Air Quality Standards
NASA	National Aeronautics and Space Administration
NEPA	National Environmental Policy Act
NEPO	Nuclear Energy Plant Optimization

NERAC	Nuclear Energy Research Advisory Committee
NERI	Nuclear Energy Research Initiative
NESHAPs	National Emissions Standards for Hazardous Air Pollutants
NI PEIS	*Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility*
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
OSHA	Occupational Safety and Health Administration
PCAST	President's Committee of Advisors on Science and Technology
PEIS	programmatic environmental impact statement
P.L.	Public Law
$PM_{2.5}$	particulate matter with an aerodynamic diameter less than or equal 2.5 microns
PM_{10}	particulate matter with an aerodynamic diameter less than or equal to 10 microns
PNNL	Pacific Northwest National Laboratory
RCRA	Resource Conservation and Recovery Act
REDC	Radiochemical Engineering Development Center
RPL	Radiochemical Processing Laboratory
SNL	Sandia National Laboratories
SRPS	Stirling radioisotope power systems
SRS	Savannah River Site
SRTG	Small Radioisotope Thermoelectric Generator
SST/SGT	safe, secure trailer/SafeGuards Transport
TEDF	Treated Effluent Disposal Facility
TRIGA	training, research, isotopes General Atomics (reactor)
UZrH	uranium-zirconium-hydride
U.S.C.	United States Code
USGS	U.S. Geological Survey
WAG	waste area grouping
WIPP	Waste Isolation Pilot Plant
Y–12	Y–12 Plant

Chapter 1
Introduction

This *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* identifies reasonable alternatives and potential impacts associated with the proposed action to enhance the U.S. Department of Energy's existing nuclear facility infrastructure for: (1) the production of isotopes for medical, research, and industrial uses, (2) the production of plutonium-238 for use in advanced radioisotope power systems for future space missions, and (3) supporting the Nation's civilian nuclear energy research and development needs.

1.1 BACKGROUND

Under the authority of the Atomic Energy Act of 1954, as amended, the U.S. Department of Energy (DOE) is responsible for ensuring the availability of isotopes for medical, industrial, and research applications; meeting the nuclear material needs of other Federal agencies; and undertaking research and development activities related to development of nuclear power for civilian use.

To meet these responsibilities, DOE maintains nuclear infrastructure capabilities that support various missions in areas such as nuclear materials production and testing, and research and development activities related to civilian applications of nuclear power. These infrastructure capabilities include research and test facilities such as research reactors and accelerators used for steady-state neutron irradiation of materials to produce radionuclides, as well as shielded "hot cell" and glovebox facilities used to prepare materials for testing and/or to handle postirradiation materials. An additional component of this infrastructure is the highly trained workforce that specializes in performing complex tasks that have been learned and mastered over the life of these facilities.

Over the years, DOE's nuclear facility infrastructure has diminished because of the shutdown of facilities; recent examples are the High Flux Beam Reactor at Brookhaven National Laboratory, New York, and the Cyclotron Facility at Oak Ridge National Laboratory (ORNL), Tennessee. This, in turn, has hampered DOE's ability to satisfy increasing demands in various mission areas. To continue to maintain sufficient irradiation facilities to meet its obligations under the Atomic Energy Act, DOE has assessed the need for expansion of its existing nuclear infrastructure in light of its commitments to ongoing programs, its commitments to other agencies for nuclear materials support, and its role in supporting civilian nuclear energy research and development programs to maintain the viability of civilian nuclear power as one of the major energy sources available to the United States.

1.2 PURPOSE AND NEED FOR AGENCY ACTION

The Nuclear Energy Research Advisory Committee (NERAC) was established in 1998 by DOE in accordance with the Federal Advisory Committee Act to provide independent, expert advice on complex science and technical issues that arise in the planning, management, and implementation of DOE's civilian nuclear energy research programs. The chairman of NERAC has informed the Secretary of Energy that:

- "There is an urgent sense that the nation must rapidly restore an adequate investment in basic and applied research in nuclear energy if it is to sustain a viable United States capability in the 21st Century."

- "[T]he most important role for DOE [Office of Nuclear Energy, Science and Technology] in the nuclear energy area at the present time is to ensure that the education system and its facility infrastructure are in good shape."

- "Of particular need over the longer term are dependable sources of research isotopes and reactor facilities providing high volume flux irradiation for nuclear fuels and materials testing" (Duderstadt 2000).

Under the guidance of NERAC, DOE has completed an internal assessment of its existing nuclear facility infrastructure capabilities. This *Nuclear Science and Technology Infrastructure Roadmap* evaluates the existing DOE infrastructure and identifies gaps for meeting projected demands (DOE 2000a). The basic finding of this assessment also concluded that the capabilities of currently operating DOE facilities will not meet projected U.S. needs for nuclear materials production and testing or research and development.

Consistent with these findings, DOE recognizes that adequate nuclear research reactor, accelerator, and associated support facilities must be available to implement and maintain a successful nuclear energy program. As demand continues to increase for steady-state neutron sources needed for isotope production and civilian nuclear energy research and development, DOE's nuclear infrastructure capabilities to support this demand have not improved. To continue meeting its responsibilities under the Atomic Energy Act and to satisfy projected increases in the future demand for isotope products and irradiation services, DOE proposes to enhance its existing nuclear facility infrastructure to provide for: (1) production of isotopes for medical, research, and industrial uses, (2) production of plutonium-238 for use in advanced radioactive isotope (radioisotope) power systems for future National Aeronautics and Space Administration (NASA) space exploration missions, and (3) support of the Nation's civilian nuclear energy research and development needs.

To evaluate the potential environmental impacts associated with this proposed enhancement, DOE has prepared this *Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]). This NI PEIS evaluates impacts from new facility construction, modification, startup, and 35 years of operation, followed by decommissioning when applicable. For analysis purposes, a 35-year operating period was established based on the projected availability of existing DOE irradiation facilities to potentially support these missions. This timeframe also accommodates current projections that indicate the demand for radioisotopes and civilian nuclear energy research and development requiring these enhancements will extend for at least the next 20 years (Wagner et al. 1998; NERAC 2000a; DOE 2000a).

1.2.1 Medical and Industrial Isotope Production

Over the past few decades, isotopes have become vital tools for use in medicine, industry, and scientific research. Isotopes, including both radioisotopes and stable isotopes, play a particularly important role in medical diagnosis, treatment, and research. Currently, more than 12 million nuclear medicine procedures are performed each year in the United States, and approximately one-third of all patients admitted to U.S. hospitals undergo at least one medical procedure that employs the use of medical isotopes (NERAC 2000a). Many medical isotopes are produced in the United States by DOE in nuclear reactors and particle accelerators. In limited cases, some medical isotopes can also be produced by extracting them from existing radioactive materials, such as thorium-229 obtained from DOE's existing stockpile of uranium-233. Radioisotopes are used for both diagnosis and therapy. Diagnostic radioisotopes are used for imaging internal organs. Unlike conventional radiology, imaging with radioisotopes reveals organ function and structure, which provides additional data for a more accurate diagnosis, and assists in the early detection of abnormalities. In ongoing clinical testing, therapeutic isotopes have proven effective in treating cancer and other illnesses by cell-directed localized radiation therapy (i.e., deploying antibodies or carriers of radioisotopes to seek and destroy invasive cancer cells). This directed therapy can minimize adverse side effects (e.g., healthy tissue damage, nausea, hair loss), making it an effective, attractive alternative to traditional chemotherapy or radiation treatments.

For nearly 50 years, DOE has actively promoted the use of radioisotopes to improve the health and well-being of U.S. citizens. DOE's use of its unique technologies and capabilities to develop isotopes for civilian purposes has enabled the widespread application of medical and industrial isotopes seen today. DOE must provide an adequate supply of isotopes to keep pace with the growing and changing needs of the research community if it is to continue to serve this key role.

An Expert Panel convened by DOE in 1998 reviewed several industry projections for growth in demand for medical isotopes. The Expert Panel concluded that the growth rate in medical isotope use will be significant over the next 20 years (Wagner et al. 1998). Specifically, the panel estimated that the expected growth rate of medical isotope use during the next 20 years would range from 7 to 14 percent per year for therapeutic applications and from 7 to 16 percent per year for diagnostic applications. The panel noted that these growth rates are attainable only if basic research in nuclear medicine is supported and if modern, reliable isotope production facilities are available. In the period since the initial estimates were made, the actual growth of medical isotope use has tracked at levels consistent with the Expert Panel findings. DOE and NERAC have agreed with the following findings and recommendations provided by the Expert Panel:

- Several isotopes have proven their clinical efficacy, but supply and cost concerns could dramatically affect the use of these isotopes in the practice of nuclear medicine.

- Although commercial and research applications for certain isotopes have been or are being developed, their limited availability and high prices are inhibiting their use in clinical applications.

- Research isotopes that have shown promise as diagnostic and therapeutic materials are not being explored because of their lack of availability or high price.

- At present, there is no domestic production facility to guarantee the continued supply of many of these isotopes.

- To meet current and future needs of the biomedical sciences community, the Expert Panel recommended:

  > ". . . the United States develop a capability to produce large quantities of radionuclides [radioisotopes] to maintain existing technologies and to stimulate future growth in the biomedical sciences. The successful implementation of such a program would help insure our position as an international leader in the biomedical sciences well into the twenty-first century. The panel recommends that the U.S. Government build this capability around a reactor, an accelerator, or a combination of both technologies as long as isotopes for clinical and research applications can be supplied reliably, with diversity in adequate quantity and quality" (Wagner et al. 1998).

In its recent report from the Subcommittee for Isotope Research and Production Planning, NERAC further identified that:

> "It is now widely conceded that limited availability of specific radionuclides is a constraint on the progress of research. The problem is especially apparent in a number of medical research programs that have been terminated, deferred, or seriously delayed by a lack of isotope availability . . . The lack of radionuclides significantly inhibits progress in evaluating a host of promising diagnostic and therapeutic drugs in patients with debilitating and fatal diseases, examining

> fundamental basic science questions, studying human behavior and normal growth and development, and exploring the aging process and the products of transgene expression . . . the DOE long-term goal to have a reliable isotope supply system in place that would enable scientists to bring their creative ideas into practical use safely, quickly and efficiently is appropriate, be it basic science research, clinical medicine, or industrial endeavors. The discovery and dissemination of new knowledge should continue to be a core mission, and basic science and the application of basic science to clinical research discoveries to improve the diagnosis and treatment outcomes should be a crucial component of that mission. [DOE], in providing a federal system for the reliable supply of stable and radioactive isotopes for research, will be an important aspect of fulfilling the federal responsibility to support biomedical research" (NERAC 2000a).

Current domestic and global producers of radioisotopes include governments that operate reactors and accelerators at national laboratories or institutes, and private sector companies that own and operate accelerators. There are also many partnership arrangements where companies lease irradiation space in government reactors or operate processing facilities in coordination with the government. A few universities also produce radioisotopes, but their ability to provide reliable and diverse supplies is generally limited by the small-scale capabilities or operating schedules of their facilities.

DOE's production and sale of radioisotopes fall into two categories: "commercial" and "research." Commercial radioisotopes are those that are produced in large, bulk quantities and sold to pharmaceutical companies or distributors, or to equipment or sealed source manufacturers. DOE only produces commercial isotopes when there is no U.S. private sector capability or when foreign sources do not have the capacity to meet U.S. needs reliably.

In contrast, research radioisotopes are typically produced and sold in small quantities in response to specialty orders from researchers preparing experiments in the field of medicine, with small quantities of these radioisotopes also purchased by industrial researchers. Because small-quantity production of research isotopes is not financially attractive to private sector producers, it is generally not undertaken. DOE attempts to provide all research radioisotopes that are requested, subject to production capability, inventory, and financial constraints. As successful application of a specific research isotope is established, the production and sales of that radioisotope may shift from research to commercial status. In recent years, over 95 percent of DOE's sales of radioisotopes by dollar volume were commercial, and 5 percent were for research.

DOE produces radioisotopes using the High Flux Isotope Reactor (HFIR) at ORNL, the Advanced Test Reactor (ATR) at Idaho National Engineering and Environmental Laboratory (INEEL), and the Annular Core Research Reactor at Sandia National Laboratories (SNL). DOE also produces radioisotopes using accelerators, namely the Isotope Production Facility at Los Alamos National Laboratory (LANL) and the Brookhaven LINAC (Linear Accelerator) Isotope Producer at Brookhaven National Laboratory. At each of these DOE sites, the radioisotope production mission shares the reactor or accelerator with other basic energy sciences or defense missions that are generally much larger and exercise considerable influence on facility schedules and priorities. As such, radioisotope production is often relegated to fulfilling a secondary mission that is dependant on the operating constraints of these larger, primary missions. Currently, approximately 50 percent of DOE's isotope production capability is being used. Assuming a midpoint growth curve for future isotope demand and a diversity and redundancy of isotope supply, DOE estimates that its isotope production facilities would be fully used within a 5- to 10-year timeframe if no enhancements to the existing nuclear facility infrastructure are implemented. This projection was made in the context of a worldwide market for radioisotopes. Although DOE's market share is a small fraction of the overall total, it is very significant for some radioisotopes and particularly important for a large number of radioisotopes that are used in relatively

small quantities for research. These isotopes, which are used almost exclusively by researchers at universities and hospitals, are not purchased in quantities that would attract private industry to take over their production. However, DOE may need to significantly increase the production levels of these radioisotopes as world demand changes and promising research developments in their medical use are brought to commercialization.

Recent analyses indicate that the greatest challenge to meeting projected isotope market requirements over the next 20 years will be in the area of therapeutic medical isotopes, several of which are currently unavailable or are available only in limited quantities (Battelle 1999). For the purpose of analysis in this NI PEIS, a representative set of isotopes was selected on the basis of the recommendations of the Expert Panel, medical market forecasts (Frost & Sullivan 1997), reviews of medical literature, and more than 100 types of ongoing clinical trials that use radioisotopes for the treatment of cancer and other diseases. These isotopes are listed in **Table 1–1**, along with a brief description of their medical and, in some cases, industrial applications. Currently, these medical applications primarily involve the diagnosis and treatment of three major classes of disease—cancer, vascular disease, and arthritis. Although these isotopes are a representative sample of possible isotopes which could be produced, DOE expects that the actual isotopes produced as a result of the proposed action would vary from year to year in response to the focus of clinical research and the specific market needs occurring at that time.

The United States currently purchases approximately 90 percent of its medical isotopes from foreign producers, most notably Canada. However, Canada only supplies a limited number of economically attractive commercial isotopes (primarily molybdenum-99), and it does not supply research isotopes or the diverse array of medical and industrial isotopes considered in this NI PEIS. As such, reliance on Canadian sources of isotopes to satisfy projected U.S. isotope needs would not meet DOE's mission requirements.

Industrial isotope applications fall into three broad categories: nucleonic instrumentation, irradiation and radiation processing, and technologies that use radioactive tracers. Examples of nucleonic instrumentation include gauges for measuring physical parameters (e.g., detection systems for pollutants, explosives, drugs, ores, petroleum, and natural gases; nondestructive testing by gamma radiography; and smoke detectors). Irradiation and radiation processing technologies include radiation sterilization of food and medical products and the curing of plastics. Radioactive tracer applications include studies of chemical synthesis reactions; mass transfer monitoring in industrial plants; analysis of the transport and uptake of nutrients, fertilizers, herbicides, and waste materials in plants, soils, and groundwater; and laboratory-based studies of the properties of materials.

In proposing to expand its radioisotope production capability, DOE intends to continue to complement the commercial availability of these radioisotopes. Consistent with current isotope production activities, DOE will continue to make its facilities available to the private sector to support the production and sale of isotopes.

1.2.2 Plutonium-238 Production for Space Missions

As part of its charter under the Atomic Energy Act, DOE and its predecessor agencies have been developing and supplying radioisotope power systems to NASA for space exploration for more than 30 years. These radioisotope power systems include radioisotope thermoelectric generators used to power electrical components and radioisotope heater units used to keep spacecraft instruments warm. Previous NASA space missions that have used radioisotope power systems include the Apollo lunar scientific packages and the Pioneer, Viking, Voyager, Galileo, and Ulysses deep space probes. More recent missions include the Mars Pathfinder mission launched in 1996 and the Cassini mission launched in 1997. These radioisotope power systems have repeatedly demonstrated their performance, safety, and reliability in various NASA space missions. Without these power systems, these types of space exploration missions could not have been performed by NASA.

Table 1–1 Representative Radioisotopes

Radioisotope	Applications
Actinium-227	Parent of radium-223 (monoclonal antibody attachment used for cancer treatment by radioimmunotherapy)
Astatine-211	Alpha-emitting radioisotope being studied for a variety of radioimmunotherapy applications
Gold-198	Ovarian, prostate, and brain cancer; intracavity therapy
Cadmium-109	Cancer detection; pediatric imaging; industrial detection systems for pollutants, explosives, drugs, ores, petroleum, and natural gas
Copper-64	Diagnostic imaging, dosimetry studies, cerebral and myocardial blood flow, colorectal cancer therapy
Copper-67	Cancer treatment/diagnostics, cancer treatment by radioimmunotherapy, planar imaging, diagnostic imaging
Fluorine-18	Cancer detection/diagnostics
Gadolinium-153	Osteoporosis detection, diagnostic imaging
Germanium-68	Diagnostic imaging calibration, potential antibody labeling
Holmium-166	Treatment of rheumatoid arthritis, radiolabeling, and monoclonal antibody techniques
Indium-111	Cancer treatment/diagnostics
Iodine-123	Alzheimer's Disease and Schizophrenia diagnostic, breast cancer imaging, cardiac imaging, radioimmunotherapy of Parkinson's Disease
Iodine-125	Osteoporosis detection, diagnostic imaging, tracer drugs, monoclonal antibodies, brain cancer treatment (iodine-131 replacement), radiolabeling, tumor imaging, mapping of receptors in the brain, interstitial radiation therapy, brachytherapy for treatment of prostate cancer, determination of glomerular filtration rate, determination of plasma volume, detection of deep vein thrombosis of the legs
Iodine-131	Lymphoid tissue tumor/hyperthyroidism treatment; antibody labeling; brain biochemistry in mental illness; diagnosis of thyroid disorders by gamma camera imaging or counting; radioimmunotherapy; imaging; cellular dosimetry; adrenal medulla scintigraphy; treatment of Grave's disease, goiters, prostate cancer, hepatocellular carcinoma, neuroblastoma and malignant pheochromocytoma, thyroid carcinoma, and melanoma; locating metastatic lesions; internal (systemic) radiation therapy; study of kidney functions; construction of renogram; adrenal cortex imaging; investigations of hepatobillary function; determination of plasma volume
Iridium-192	Brachytherapy, brain and spinal cord tumor treatment, heart disease treatment (restenosis therapy), seed implants for breast and prostate tumors, industrial nondestructive testing by gamma radiography
Krypton-81m	Cardiac imaging
Lutetium-177	Heart disease treatment (restenosis therapy), cancer treatment by radioimmunotherapy
Molybdenum-99	Parent for technetium-99m generator used for brain, liver, lungs, heart imaging
Osmium-194	Monoclonal antibody attachment used for cancer treatment by radioimmunotherapy
Phosphorus-32	Polycythemia rubra vera (blood cell disease) and leukemia treatment, bone disease diagnosis/treatment, diagnostic imaging of tumors, pancreatic and liver cancer treatment, radiolabeling, labeling nucleic acids for in vitro research, diagnosis of superficial tumors, heart disease treatment (restenosis therapy), intracavity therapy
Phosphorus-33	Leukemia treatment, bone disease diagnosis/treatment, diagnostic imaging of tumors, radiolabeling, heart disease treatment (restenosis therapy)
Palladium-103	Prostate cancer treatment
Platinum-195m	Noninvasive monitoring of drug biodistribution and metabolism, studies with intra-arterial platinum-195m-cisplatin
Rhenium-186	Cancer treatment/diagnostics, monoclonal antibodies, bone cancer pain relief, treatment of rheumatoid arthritis, treatment of prostate cancer
Scandium-47	Bone cancer pain relief, cancer treatment by radioimmunotherapy
Selenium-75	Radiotracer used in brain studies, imaging of adrenal cortex by gamma-scintigraphy, lateral locations of steroid secreting tumors, pancreatic scanning, detection of hyperactive parathyroid glands, measuring the rate of bile acid loss from the endogenous pool
Samarium-145	Treatment of ocular cancer
Samarium-153	Cancer treatment/diagnostics, bone cancer pain relief, treatment of leukemia
Strontium-85	Detection of bone lesions, brain scans
Strontium-89	Bone cancer pain relief, treatment of prostate cancer, treatment of multiple myeloma, osteoblastic therapy, potential agent for treatment of bone metastases from prostate and breast cancer
Thorium-228	Cancer treatment by radioimmunotherapy, monoclonal antibodies, parent of bismuth-212
Thorium-229	Grandparent of bismuth-213 (alpha-emitter used in cancer treatment by radioimmunotherapy), parent of actinium-225, daughter of uranium-233

Table 1–1 Representative Radioisotopes (Continued)

Radioisotope	Applications
Tin-117m	Bone cancer pain relief
Tungsten-188	Cancer treatment by radioimmunotherapy, parent for rhenium-188 generator
Xenon-127	Neuroimaging for brain disorders, research on variety of neuropsychiatric disorders (especially schizophrenia and dementia), higher resolution diagnostic studies with lower patient dose, lung imaging evaluation of pulmonary ventilation, indicator for measurement of local cerebral blood flow
Yttrium-91	Cancer treatment by radioimmunotherapy, cellular dosimetry
Zinc-62	Parent for copper-64 generator used for diagnostic imaging

Source: Battelle 1999.

The radioisotope used in these power systems is plutonium-238. Through a Memorandum of Understanding with NASA, DOE provides these radioisotope power systems, and the plutonium-238 that fuels them, for space missions that require or would be enhanced by their use (DOE and NASA 1991). In addition, under the National Space Policy issued by the Office of Science and Technology Policy in September 1996, and consistent with DOE's charter under the Atomic Energy Act, DOE is responsible for maintaining the capability to provide the plutonium-238 needed to support these missions. The Intersector Guidelines section of the National Space Policy states that, "The Department of Energy will maintain the necessary capability to support space missions which may require the use of space nuclear power systems" (The White House 1996). Although research to identify other potential fuel sources to support these space exploration missions has been conducted, no viable alternative to using plutonium-238 has been established.

Historically, the reactors and chemical processing facilities at DOE's Savannah River Site (SRS) were used to produce plutonium-238; however, downsizing of the DOE nuclear weapons complex resulted in the shutdown of the last remaining SRS operating reactor, K-Reactor, in early 1996. Also, in 1992 then-Secretary of Energy Watkins issued a decision to phase out operations at the two chemical processing facilities (F-Canyon and H-Canyon) at SRS. In accordance with that decision, the separation facilities are planned to be shut down following completion of their current missions to stabilize and prepare for the disposition of Cold War legacy nuclear materials and certain spent nuclear fuel, and a determination that a new nonchemical processing technology is capable of preparing aluminum-based research reactor spent nuclear fuel for ultimate disposition.

In order to obtain a source of plutonium-238 to support NASA space missions, DOE signed a 5-year contract in 1992 to purchase plutonium-238 from Russia, authorizing the United States to purchase up to 40 kilograms (88.2 pounds) of plutonium-238, with the total available for purchase in any one year limited to 10 kilograms (22 pounds).[1] Under this contract, DOE purchased approximately 9 kilograms (19.8 pounds) of plutonium-238[2]. This material constitutes the only available U.S. inventory that has been reserved for space missions, an amount that is expected to be depleted by approximately 2005. DOE's practice of purchasing on an as-needed basis has avoided the costs from processing the plutonium-238 to remove the decay products that would result from storing it for an extended period of time. In 1997, DOE extended the contract for another 5 years; therefore, it is set to expire in 2002. Any purchases beyond 2002 would likely require the negotiation of a new contract and may require additional National Environmental Policy Act (NEPA) review. The long-term viability of pursuing additional contract extensions or entering into a new contract is unclear.

[1] The NI PEIS presents the weight of plutonium-238 in terms of kilograms of isotope. In contrast, NASA documentation expresses this weight in terms of plutonium oxide. The equivalent plutonium oxide weight can be approximated by multiplying the isotope kilogram weight by 1.134.

[2] The environmental impacts of purchasing plutonium-238 from Russia are evaluated and documented in the *Environmental Assessment of the Import of Russian Plutonium-238* (DOE 1993), prepared by DOE's Office of Nuclear Energy.

The political and economic climate in Russia creates uncertainties that could affect its reliability as a source of plutonium-238 to satisfy future NASA space mission requirements. Reestablishing a domestic plutonium-238 production capability would ensure that the United States has a long-term, reliable supply of this material. In doing so, the United States would have greater control over the available supply, plans for satisfying future demand, and the nuclear safety and nonproliferation implications of the material. As such, DOE's preference is to reestablish a domestic plutonium-238 production capability rather than to rely on Russia as the sole long-term supplier. A plutonium-238 production rate of 2 to 5 kilograms (4.4 to 11 pounds) per year is expected to be sufficient to meet NASA's estimated long-term requirements.

DOE is planning to provide radioisotope heater units for several NASA Mars Exploration missions over the next 10 years. Each heater unit would require approximately 2 grams (0.07 ounce) of plutonium-238. The number of heater units varies depending on the spacecraft. Each of the two Mars missions in 2003 is projected to require up to 11 heater units. In May 2000, NASA provided preliminary guidance to DOE to also plan for the potential use of radioisotope power systems for the Pluto/Kuiper Express mission scheduled for launch in 2004, the Europa Orbiter mission scheduled for launch in 2006, and the Solar Probe mission scheduled for launch in 2007 (NASA 2000a). The amount of plutonium-238 needed for these missions was approximately 7.4 kilograms (16.3 pounds) for the Pluto/Kuiper Express mission, which would use an existing spare radioisotope thermoelectric generator, and approximately 3 kilograms (6.6 pounds) each for the Europa Orbiter and Solar Probe missions, which would use the Stirling radioisotope power system (SRPS). With NASA's current emphasis on smaller and less expensive spacecraft, the SRPS is being developed as a new, more efficient and lighter weight power system requiring one-third less plutonium-238 as its fuel source. However, the SRPS technology is developmental, and NASA has requested that the plutonium-238 needed for a large radioisotope thermoelectric generator be maintained as backup.

A plutonium-238 production goal of 2 to 5 kilograms (4.4 to 11 pounds) per year could produce sufficient quantities of plutonium-238 to theoretically yield an SRPS every 8 months if production were maintained at the high end of the range. However, DOE chose the 5-kilogram (11-pound) per year production rate as an upper bound due to uncertainties in the SRPS technology development requirements for backup units, and variability in the amount of plutonium-238 that may be needed for each of the units to meet NASA's power requirements.

In updated mission planning guidance provided in September 2000, NASA indicated that for programmatic and technical reasons, implementation of the Pluto/Kuiper Express mission as currently conceived was being deferred, and that the SRPS generators were candidate power systems for the Europa Orbiter and Solar Probe missions (NASA 2000b, 2000c). NASA also requested that the spare radioisotope thermoelectric generator and assembling and fueling a spare thermoelectric converter be maintained as backups for the Europa Orbiter mission in the event the SRPS technology was not ready in time. If NASA chooses to use the SRPS to support the Europa Orbiter and Solar Probe missions, there would be no change in NASA's requirements regarding the plutonium-238 needed for these two missions (i.e., approximately 3 kilograms [6.6 pounds] each, as described above), although the remaining quantity of plutonium-238 would not be sufficient to support additional deep space or long-lived exploration missions. Should NASA decide to use the backup radioisotope thermoelectric generators rather than the SRPS to support the Europa Orbiter mission, approximately 8 kilograms (17.6 pounds) of plutonium-238 would be needed, which would effectively expend all of DOE's available plutonium-238 inventory prior to supporting the Solar Probe mission. While this latest NASA guidance modifies the specific radioisotope power systems and missions for which DOE needs to plan, it does not fundamentally change NASA's overall potential plutonium-238 requirements, or the expectation that the available U.S. inventory of this material would effectively be depleted by approximately 2005.[3]

[3] Applicable NASA mission planning correspondence is presented in Appendix R.

Although future space mission schedules over a long-term planning horizon of 10 to 35 years cannot be specified at this time, DOE anticipates that NASA space exploration missions conducted during this period will continue to require plutonium-238-fueled power systems. For example, NASA announced in a recent press conference (October 26, 2000) that mission launches in 2014 and 2016 for the long-term exploration of Mars would involve long-life rover vehicles. Radioisotope power systems would be required to provide the long-life capability.

Therefore, DOE proposes to reestablish a domestic capability for producing and processing this material. Because the SRS facilities previously used for plutonium-238 production are no longer available, DOE needs to evaluate other DOE irradiation and chemical processing facilities, as well as potential commercial light water reactors (CLWR), for this mission. Unless an assured domestic supply of plutonium-238 is established, DOE's ability to provide radioisotope power systems to support future NASA space exploration missions may be lost.

1.2.3 Civilian Nuclear Energy Research and Development

Nuclear energy is an important contributor in reducing greenhouse gas emissions in the United States, Asia, and Europe. Globally, nuclear energy produces 17 percent of the world's electricity. In the United States, nuclear energy generated 20 percent of all electricity consumed in 1999. In view of these energy and environmental contributions, there is a renewed interest in nuclear power to meet an equivalent portion of the Nation's future expanding energy requirements.

In January 1997, President Clinton tasked his Committee of Advisors on Science and Technology (PCAST) to evaluate the current national energy research and development portfolio and to provide a strategy that ensures the United States has a program to address the Nation's energy and environmental needs for the next century. In its November 1997 report responding to this request, the PCAST Energy Research and Development Panel determined that restoring a viable nuclear energy option to help meet our future energy needs is important and that a properly focused research and development effort to address the potential long-term barriers to expanded use of nuclear power (e.g., nuclear waste, proliferation, safety, and economics) was appropriate. The PCAST panel further recommended that DOE reinvigorate its nuclear energy research and development activities to address these potential barriers.

Clean, safe, reliable nuclear power has a role today and in the future for our national energy security. Recognizing this need, two new significant nuclear energy research and development programs have been initiated: the Nuclear Energy Research Initiative (NERI) and Nuclear Energy Plant Optimization (NEPO). The NERI program, initiated in fiscal year 1999, sponsors new and innovative scientific and engineering research and development to address the potential long-term barriers identified by the PCAST panel affecting the future use of nuclear energy. The NEPO program, a cost-shared program with industry initiated in fiscal year 2000, sponsors applied research and development to ensure that current nuclear plants can continue to deliver adequate and affordable energy supplies up to and beyond their initial 40-year license period by resolving open issues related to plant aging and by applying new technologies to improve plant reliability, availability, and productivity.

The NERAC Subcommittee on Long-Term Planning for Nuclear Energy Research has set forth a recommended 20-year research and development plan to guide DOE's nuclear energy programs in areas of material research, nuclear fuel, and reactor technology development (NERAC 2000b). This plan stresses the need for DOE facilities to sustain the nuclear energy research mission in the years ahead. Such civilian nuclear energy research and development initiatives requiring an enhanced DOE nuclear facility infrastructure fall into three basic categories: materials research, nuclear fuel research, and advanced reactor development.

Materials Research. The high radiation fields, high temperatures, and corrosive environments in nuclear reactors (terrestrial or space) and other complex nuclear systems (e.g., accelerator transmutation of waste [ATW] systems) can accelerate the degradation of pressure vessels and structural material, component materials, material interfaces, and joints between materials (e.g., welds). Radiation effects in materials can cause a loss of mechanical integrity (fracture toughness and ductility) by embrittlement, dimensional changes (creep and swelling), and fatigue and cracking (irradiation-assisted stress corrosion cracking). Acquiring a fundamental understanding of radiation effects in current and future reactor materials (engineered steel alloys, ceramics, composites, and refractory metals), as well as the experimental validation of analytical models and computational methods, would require material irradiation testing over a range of neutron energies (thermal and fast flux) and doses. Material testing under simulated reactor conditions would be required to ensure the compatibility of advanced materials with the various moderators/coolants of future reactor concepts. In addition, the thermophysical properties and behaviors of liquid metal coolants being considered for advanced reactor (terrestrial or space) and ATW systems would require further irradiation testing. One key area of materials research that is important to plant safety and the license renewal of existing nuclear power plants is the accelerated aging of materials to simulate radiation effects over a plant lifetime. Researchers from the United States and many foreign countries use DOE's high-flux research reactors for materials testing and experimentation. These facilities have the capability to maintain a high density of neutrons in a given test volume for materials testing; shorten the time needed for such testing; tailor the neutron flux to simulate the different reactor types and conditions; and instrument the core for close monitoring of the test conditions.

Nuclear Fuel Research. Increasing demands are being placed on nuclear fuel and cladding material performance as the fuel burnup limits are extended in existing light water reactors to maximize plant performance and economic benefits. New fuel types and forms are being investigated that offer potential benefits such as enhanced proliferation resistance (uranium-thorium fuel), higher burnup, and improved waste forms for the new reactor concepts being researched and developed by DOE. In addition, plutonium-based mixed oxide fuels are being developed for the disposition of surplus weapons material, and high temperature, long-life fuels may be required for space reactors. Each of the various fuel and cladding types, forms, and material compositions would require research and irradiation testing under prototypical reactor conditions to fully understand fuel performance, cladding performance, cladding/fuel interaction, and cladding/coolant material compatibility. Fuel research includes a variety of thermal and fast spectrum power reactor fuel forms (ceramic, metal, hybrids such as cermet) and various fuel types (oxides, nitrides, carbides, and metallics). Irradiation experiments to characterize fuel performance would require the capability to test fuel pellets, pins, and fuel assemblies under steady-state and transient conditions in the higher temperature environments expected in future reactor designs. Reactor physics and criticality safety data for benchmarking computational codes and analytical methods used in fuel design and performance analysis would also be required.

Advanced Reactor Development. Certification and licensing of advanced reactor and complex nuclear systems would require the demonstration and validation of reactor and safety system thermal and fluid dynamic properties under steady-state and transient conditions. Typically, nonnuclear test loops are used to perform this research. However, because of the unique nature of some proposed advanced reactor concepts, test loop operation under prototypical temperature and neutron flux conditions would be necessary to adequately test and demonstrate coolant/moderator physics and thermal properties, heat transfer, fluid flow, and fuel-moderator performance.

1.3 DECISIONS TO BE MADE

In reaching programmatic decisions regarding potential enhancements to its existing nuclear facility infrastructure, DOE will factor the analytical results of this NI PEIS together with the findings presented in the ancillary *Cost Report for Alternatives Presented in the Draft Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope*

Production Missions in the United States, Including the Role of the Fast Flux Test Facility (NI Cost Report) and *Nuclear Infrastructure Nonproliferation Impact Assessment for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (NI Nonproliferation Impact Assessment)*[4], the *Nuclear Science and Technology Infrastructure Roadmap* (which will be updated periodically), recommendations of NERAC and its various subcommittees, public input, and other DOE policy and programmatic considerations.

With the benefit of this broad base of information, DOE intends to make the following decisions:

- Whether to enhance its current nuclear facility infrastructure to meet projected requirements for future medical and industrial isotope production, plutonium-238 production, and civilian nuclear energy research and development.

- If a decision is made to enhance DOE's existing nuclear facility infrastructure, whether to construct new facilities (one or two accelerators or a research reactor).

- Whether to restart the Fast Flux Test Facility (FFTF) at the Hanford Site in Richland, Washington, as part of a nuclear infrastructure enhancement program and, if not, whether to remove FFTF from standby mode and permanently deactivate it in preparation for its eventual decontamination and decommissioning.

- If DOE's existing nuclear facility infrastructure is not enhanced, select from existing operating facilities those needed to support the proposed plutonium-238 mission, or decide whether to continue purchasing plutonium-238 from Russia to support future NASA space missions. Existing operating facilities performing medical, research, and/or industrial isotope production missions and/or civilian nuclear energy research and development missions would continue to support existing missions at current levels.

- Whether DOE inventories of neptunium-237 should be relocated and stored for future plutonium-238 production needs.

The programmatic decisions reached in association with this NI PEIS will address isotope production and civilian nuclear energy research and development missions which are the responsibility of the DOE Office of Nuclear Energy, Science and Technology. In addition to the range of reasonable programmatic alternatives evaluated in this NI PEIS, DOE could choose to combine components of several alternatives in selecting the most appropriate strategy. For example, DOE could select a low-energy accelerator to produce certain medical, research, and industrial isotopes, and an existing operating reactor to produce plutonium-238 and conduct civilian nuclear energy research and development. If alternatives were selected involving the siting, construction, and operation of one or two new accelerators or a new research reactor, appropriate site- and project-specific NEPA documentation, tiered from this NI PEIS, would be prepared.

1.4 ISSUES IDENTIFIED DURING THE SCOPING PROCESS

On October 5, 1998, DOE published in the Federal Register (63 FR 53398) a Notice of Intent to prepare an environmental impact statement (EIS) on the proposed production of plutonium-238 for use in advanced radioisotope power systems for future space missions. With that announcement, DOE began preparing the

[4] The findings of the *NI Cost Report* and the *NI Nonproliferation Impact Assessment* are summarized in Appendixes P and Q, respectively.

Environmental Impact Statement for the Proposed Production of Plutonium-238 for Use in Advanced Radioisotope Power Systems for Future Space Missions (*Plutonium-238 Production EIS*). The scope of the *Plutonium-238 Production EIS* was established through a public scoping process conducted from October 5, 1998, through January 4, 1999. As part of the scoping process for that draft, DOE announced that FFTF would not be considered a reasonable alternative for the plutonium-238 production mission unless restart of the facility were proposed for other reasons.

Since then, the Secretary of Energy announced on August 18, 1999, that DOE would prepare the NI PEIS. Because plutonium-238 production would be among the missions considered, the scope of the *Plutonium-238 Production EIS* in its entirety was incorporated within the scope of this NI PEIS, and preparation of the *Plutonium-238 Production EIS* as a separate NEPA review was terminated.

On September 15, 1999, DOE published in the Federal Register a Notice of Intent to prepare the NI PEIS (64 FR 50064). In the Notice, DOE invited the public to comment on the proposed actions during the 45-day NI PEIS scoping period that ended October 31, 1999. During this period, DOE held public scoping meetings at seven locations: Oak Ridge, Tennessee; Idaho Falls, Idaho; Richland and Seattle, Washington; Hood River and Portland, Oregon; and Washington, D.C. The written and oral comments, and the additional comments received via U.S. mail, electronic mail, and toll-free faxes and telephone calls, were reviewed and considered by DOE in preparing this NI PEIS. Similarly, DOE reviewed and considered all comments and input originally received from the public during the *Plutonium-238 Production EIS* scoping period in the preparation of this NI PEIS.

For the *Plutonium-238 Production EIS*, approximately 750 scoping comments were received by DOE. At the scoping meetings, the following general issues and concerns were raised:

- Additional irradiation service alternatives, such as CLWRs and accelerators
- Additional storage, target fabrication, and target processing alternatives, such as Argonne National Laboratory–West's Hot Fuels Examination Facility and the SRS H-Canyon and HB-Line
- Generation of additional waste
- Costs of implementing the various alternatives

In general, the people who attended the meetings in Idaho and Tennessee were supportive of DOE's proposed plans to produce plutonium-238 domestically for future space missions. However, in Richland, Washington, the meeting was attended by several stakeholder and environmental groups who voiced considerable opposition to DOE's consideration of FFTF for plutonium-238 production.

At the meeting in Richland, Washington, the main concern was that DOE should not consider restarting FFTF; that DOE has worked hard over the years to change Hanford's mission from "production" to "cleanup"; and that DOE should continue to honor its commitment to cleanup. There were concerns about the generation of additional waste at the site and the operational safety of FFTF. There was strong opposition to restart of FFTF for any mission.

For this NI PEIS, approximately 7,000 scoping comments were received by DOE. At the scoping meetings, the most prevalent concerns were:

- Status of and commitment to cleanup at Hanford and the impact of FFTF's restart on the existing waste cleanup at Hanford

- Lack of justification for the identified missions
- Costs of implementing the various alternatives
- Need for an additional alternative calling for the permanent deactivation of FFTF coupled with the No Action Alternative elements; that is, no plutonium-238 production and no additional research and development or medical isotope production beyond existing operating levels

The number of people who commented at the scoping meetings conducted in Oak Ridge, Tennessee; Idaho Falls, Idaho; and Washington, D.C., was smaller in comparison to the meetings held in the Pacific Northwest. At the scoping meeting in Oak Ridge, Tennessee, a commentor was concerned with the relationship of this NI PEIS to other DOE programs and the relative merits of accelerator and reactor performance. The commentor stated that the NI PEIS should include an explanation of mixed oxide fuel disposition. In addition, the commentor supported medical isotope production in Oak Ridge because it is near a transportation hub and some medical isotopes are short-lived; therefore, transportation is key.

At the scoping meeting in Idaho Falls, Idaho, most commentors supported siting the new missions at INEEL. The commentors also stated that the socioeconomic impacts of the alternatives need to be considered in this NI PEIS. A commentor stated that decisions in regard to medical isotope production should be based on the needs of the Nation as a whole and not on perceived commercial needs. The commentor also stated that incremental DOE and commercial investments in the ATR would be sufficient to enhance reactor radioisotope production needs and meet the requirements of the nuclear medicine industry.

At the scoping meetings held in the states of Washington and Oregon, many of the comments concerned using FFTF to accomplish the proposed action. Many who attended the meetings in Seattle, Washington; Portland, Oregon; and Hood River, Oregon, were strongly opposed to the restart of FFTF. Many commentors stated that the Hanford cleanup mission would be jeopardized, especially when DOE has not met the cleanup milestones. Many of the comments received at the Richland, Washington, meeting supported restarting FFTF, stating that the restart would not hamper Hanford's cleanup mission, and further stating that the operation of FFTF could help save many lives by producing isotopes to be used in new ways to treat cancer, heart disease, and other illnesses. Commentors were also concerned about the potential generation of radioactive and hazardous wastes as a result of the proposed action, as well as DOE's commitment to ongoing cleanup programs, particularly at Hanford.

At the scoping meeting in Washington, D.C., the commentors supported the need for medical isotope production. Several commentors were against the restart of FFTF; others stated that DOE needs to consider partnerships with private industry to generate necessary funds for the restart. Some commentors thought a cost study should be prepared to include avoiding future health care costs as well as cost savings to the national Medicare and Medicaid programs that could be realized by using nuclear isotopes in medical applications. Proliferation concerns were also raised as some commentors stated that: (1) the United States would be sending the wrong message by restarting FFTF; (2) a change in the U.S. nonproliferation policy would be required to import German mixed oxide fuel; and (3) the use of highly enriched uranium would be contrary to existing U.S. nonproliferation policy. Other concerns included waste generation, the Hanford cleanup, and safety at FFTF.

Comments received during the scoping periods were systematically reviewed by DOE. As a means of summarizing the issues raised during the scoping process, those comments with similar or related topics were grouped into categories to identify specific issues of public concern. After these issues were identified, they were further evaluated to determine whether they fell within or outside the proposed scope of this NI PEIS.

In several instances, the original scope was expanded to accommodate additional issues resulting from the public scoping process.

Comments received that contributed to expansion of the scope included the following general areas:

- Deactivate FFTF: Alternative 5, Permanently Deactivate FFTF with no new missions at existing facilities, has been added to the scope of this NI PEIS.

- Cleanup at Hanford: although not within the scope of this NI PEIS, information is included about the cleanup mission at Hanford and land-use planning efforts.

- Environmental contamination at Hanford: information is included about the groundwater quality at the Hanford Site.

- Nonproliferation issues: the proposed import of German SNR–300 fuel is addressed, and a separate *NI Nonproliferation Impact Assessment* report was prepared and distributed to the public in September 2000.

- Transition of FFTF stewardship after it is deactivated: the appropriate transition information is included.

- Restart of FFTF and budget constraints: DOE has made a commitment that implementation of the Record of Decision will not divert or reprogram budgeted funds designated for Hanford cleanup.

- Tri-Party Agreement at Hanford: information about the Tri-Party Agreement and its relationship to this NI PEIS is included.

Public comments and materials submitted during the public scoping periods for both the *Plutonium-238 Production EIS* and this NI PEIS were logged and placed in the Administrative Record for this NI PEIS. Appendix N summarizes the comments received during both public scoping periods.

1.5 ISSUES RAISED DURING THE PUBLIC COMMENT PERIOD ON THE DRAFT NI PEIS

DOE published the Draft NI PEIS in July 2000. In accordance with Council on Environmental Quality (CEQ) and DOE NEPA regulations, DOE announced the availability of the Draft NI PEIS in the Federal Register (65 FR 46443) and invited interested parties to provide comments on the Draft NI PEIS analysis and results. The Draft NI PEIS or Summary was distributed to approximately 6,000 individuals.

NEPA regulations mandate a minimum 45-day comment period after the U.S. Environmental Protection Agency's Notice of Availability of a draft EIS to provide an opportunity for the public to comment on the EIS analysis and results. The original 45-day comment period on the Draft NI PEIS began on July 28, 2000. To provide interested parties with additional time to comment, the deadline for transmittal of comments was changed from September 11, 2000 (as stated in the transmittal letter of the Draft PEIS and the Summary) to September 18, 2000. During the 52-day comment period, DOE held seven hearings to discuss the proposed action and to receive oral and written comments on the Draft NI PEIS. These hearings were held at Oak Ridge, Tennessee; Idaho Falls, Idaho; Hood River, Oregon; Portland, Oregon; Seattle, Washington; Richland, Washington; and Arlington, Virginia. In addition, the public was encouraged to submit comments via U.S. mail, e-mail, a toll-free phone line, and a toll-free fax line. During the public comment period, DOE received approximately 3,400 submittals containing over 6,200 comments. DOE has responded to all comments received during the public comment period. These comments are presented in Volume 3 of this Final NI PEIS.

DOE considered comments received after the close of the public comment period to the extent practicable (see Section 1.5.6).

The public comments received on the Draft NI PEIS addressed a wide range of issues. The following discusses the major issues raised, and DOE's responses to these issues. Changes made in response to comments received on the Draft NI PEIS are described in Section 1.8.

Major issues raised addressed purpose and need for the proposed action; impact of FFTF on Hanford cleanup; waste management and spent nuclear fuel; cost of the various alternatives; nuclear nonproliferation policy; public involvement; and environmental impacts. Aside from comments on the proposed action and its environmental impacts, many commentors expressed support for or opposition to FFTF restart, the major point of public controversy associated with the NI PEIS.

1.5.1 Purpose and Need for the Proposed Action

Many commentors expressed the opinion that DOE failed to demonstrate a compelling argument for the projected need for medical isotopes, and that such medical isotopes could be produced or purchased elsewhere, particularly in Canada. In contrast, a large number of commentors expressed support for expanded isotope production by sharing personal stories of how medical isotopes had either saved a relative or friend, or could have saved them had isotopes been available. As presented in Section 1.2.1, DOE sought independent analysis of trends in the use of medical isotopes, and established two advisory bodies, the Expert Panel and the NERAC. DOE has adopted these growth projections as a planning tool for evaluating the potential capability of the existing nuclear facility infrastructure to meet programmatic requirements. In the period since the initial estimates were made, the actual growth of medical isotope use has tracked at levels consistent with the Expert Panel findings. While Canada currently provides a large amount of the medical radioisotopes used in the United States, it only supplies a limited number of economically attractive commercial isotopes (primarily molybdenum-99), and it does not supply research isotopes or the diverse array of medical and industrial isotopes considered in the NI PEIS.

A number of commentors also questioned the suitability of using FFTF for producing research isotopes in light of findings presented in the NERAC Subcommittee for Isotope Research and Production Planning Report (NERAC 2000a). While it would not be cost effective to restart FFTF for the singular purpose of producing small quantities of various research isotopes, sustained operation of FFTF for the production of larger quantities of both research and commercial isotopes would be viable if FFTF were operated in concert with producing plutonium-238 and conducting nuclear energy research and development for civilian applications. In recognition of these constraints on its operational feasibility, the NI PEIS only evaluates the use of FFTF for isotope production when coupled with these other missions.

Commentors also questioned the need for the United States to reestablish domestic production of plutonium-238. In particular, commentors pointed to the availability of plutonium-238 that could be purchased from Russia, and recent guidance from NASA stating that DOE no longer needed to support certain radioisotope power systems. As discussed in Section 1.2.2, DOE could purchase plutonium-238 from Russia. However, for supply reliability reasons and concern of nuclear nonproliferation, DOE's preference is to establish a domestic plutonium-238 production capability. Current NASA guidance to DOE is also discussed in Section 1.2.2. The May 22, 2000, correspondence from NASA identifies that it no longer has a planned requirement for Small Radioisotope Thermoelectric Generator (SRTG) power systems (NASA 2000a). This does not mean that NASA no longer requires DOE to provide the necessary plutonium-238 to support deep space missions. Rather, SRTG development efforts were stopped in order to permit reprogramming of funds to support development of a new radioisotope power system based on the SRPS technology. This new radioisotope power system, referred to in the subject correspondence, requires one-third less plutonium as its

fuel source. Because the SRPS technology is developmental, NASA has requested in a September 22, 2000, letter to DOE that the plutonium-238 needed for a large radioisotope thermoelectric generator be maintained as a backup (NASA 2000b).

1.5.2 Impact of FFTF Restart on Hanford Cleanup

A number of commentors expressed concern that DOE's primary mission at Hanford needs to be cleanup, including compliance with the Tri-Party Agreement. Although beyond the scope of this NI PEIS, ongoing Hanford cleanup activities are high priority to DOE. Hanford environmental restoration activities are conducted in accordance with the Tri-Party (i.e., DOE's Richland Operations Office, EPA, and the State of Washington Department of Ecology) Agreement. This agreement specifies milestones and schedules for restoration of all parts of Hanford. FFTF milestones in the Tri-Party Agreement were placed in abeyance (suspension) by agreement of the three parties until a decision is made on the future of FFTF. Public meetings were held on this formal milestone change. DOE is fully committed to honoring this agreement.

A number of commentors also expressed concern that funding for Hanford cleanup would be diverted for FFTF restart and hamper the progress of cleanup activities. The U.S. Congress funds Hanford cleanup through the Office of the Assistant Secretary for Environmental Management. Congress also funds FFTF through the Office of Nuclear Energy, Science and Technology (NE). The nuclear infrastructure missions described in Section 1.2 would also be funded through NE, which has no funding connection to Hanford cleanup activities. As stated in Section N.3.2, implementation of the nuclear infrastructure alternatives would not divert or reprogram budgeted funds designated for Hanford cleanup, regardless of the alternative(s) selected.

1.5.3 Waste Management and Spent Nuclear Fuel

A number of commentors expressed concern over the generation and disposition of waste resulting from the proposed action. In particular, commentors pointed to past DOE waste management practices and questioned whether wastes resulting from proposed NI PEIS activities would be properly managed. This NI PEIS addresses wastes produced for each alternative, as well as cumulative impacts related to waste production. Waste minimization programs at each of the alternative sites are also addressed. These programs would be implemented for the alternative selected in the Record of Decision. The waste generated from any of the alternatives considered in this NI PEIS would be managed (i.e., treated, stored, and disposed of) in a safe and environmentally protective manner and in compliance with all applicable Federal and state laws and regulations and applicable DOE orders.

A number of commentors expressed specific concern over the generation and disposition of waste resulting from FFTF restart and operation, and how this would impact Hanford's existing waste management infrastructure. Management of wastes that would be generated under implementation of Alternative 1 (Restart FFTF) is discussed in Section 4.3 (e.g., Section 4.3.1.1.13). Section 4.3.1.1.13 was revised to clarify that the Hanford waste management infrastructure is analyzed in this NI PEIS for the management of waste resulting from FFTF restart and operation. This analysis is consistent with policy and DOE Order 435.1, *Radioactive Waste Management*, that DOE radioactive waste shall be treated, stored, and in the case of low-level waste, disposed of at the site where the waste is generated, if practical, or at another DOE facility. However, if DOE determines that use of the Hanford waste management infrastructure or other DOE sites is not practical or cost effective, DOE may issue an exemption under DOE Order 435.1 for the use of non-DOE facilities (i.e., commercial facilities) to store, treat, and dispose of such waste generated from the restart and operation of FFTF. In addition, Sections 4.3.3.1.13 and 4.4.3.1.13 also address the potential impacts associated with the waste generated from the target fabrication and processing in the Fuels and Materials Examination Facility (FMEF) and how this waste would be managed at the site.

A number of commentors also raised concern that processing of irradiated targets for production of plutonium-238 would generate high-level radioactive waste. DOE Manual 435.1, *Radioactive Waste Management,* defines high-level radioactive waste as "the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation." DOE has prepared an implementation guide to M 435.1 to assist in implementing the requirements contained in that manual. For this particular "requirement," the definition of high-level radioactive waste, the guide is intended to facilitate the classification of indefinite waste as to whether or not they are high-level radioactive waste. It is recognized that the definition of high-level radioactive waste is not precise and is essentially a source-based definition that also alludes to concentrations of a given waste stream. Page II-8 of the guide notes that "For the purpose of managing high-level waste under DOE M 435.1-1 [sic], spent nuclear fuel includes spent driver elements and/or irradiated target elements that contain transuranium elements." This statement was included in the guide because the concentrations of long-lived isotopes are likely to be somewhat high during reprocessing and it also meets the source-based definition. As a result of reviewing this guide and to address the comments raised, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. As a result, the Waste Management sections (i.e., Sections 4.3.1.1.13, 4.3.2.1.13, 4.3.3.1.13, and 4.4.3.1.13) of this NI PEIS have been revised to reflect this different classification from what was assumed in the Draft NI PEIS. As discussed in these revised sections, irrespective of how the waste is classified (i.e., transuranic or high-level radioactive waste), the composition and characteristics are the same, and the waste management (i.e., treatment and onsite storage) for this NI PEIS would be the same. In addition, even if the waste were managed as high-level radioactive waste, it would have no impact on the existing high-level radioactive waste management infrastructure (e.g., high-level waste storage tanks) because the high-activity waste from processing the targets would be initially stored and vitrified within the processing facility (i.e., FMEF, the Radiochemical Engineering Development Center [REDC], or the Fluorinel Dissolution Process Facility [FDPF]).

Commentors also expressed concern over the potential impacts of spent nuclear fuel generation from FFTF restart and operation, particularly regarding human health risk. This NI PEIS estimates that about 16 metric tons of heavy metal spent nuclear fuel would be generated over 35 years of operation of FFTF. Hanford is currently managing about 2,000 metric tons of heavy metal spent nuclear fuel. As indicated in Table 4–173, the radiation risk to a maximally exposed individual from normal operational activities during management of the current stored spent nuclear fuel over 35 years is 1.4×10^{-8} latent cancer fatality. The risk to the maximally exposed individual that would be associated with the new nuclear infrastructure operations to restart FFTF and operate FMEF or the Radiochemical Processing Laboratory is 9.5×10^{-8} latent cancer fatality. Furthermore, only a small fraction of this risk would be attributable to management of the additional spent nuclear fuel at FFTF. The annual dose to the maximally exposed individual from all current and reasonably foreseeable activities at Hanford is less than 0.2 millirem. This dose is well within the DOE dose limits given in DOE Order 5400.5, *Radiation Protection of the Public and the Environment*. As discussed in that order, the dose limit from airborne emissions is 10 millirem per year, as required by EPA regulations under the Clean Air Act; the dose limit from drinking water is 4 millirem per year, consistent with the EPA drinking water criteria under the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. The risk to the population from all activities at Hanford would be 0.21 latent cancer fatality over 35 years. DOE has committed to remove the spent nuclear fuel at Hanford for ultimate disposition in a geologic repository.

1.5.4 Cost of the Various Alternatives

Commentors expressed opinions about the costs related to the stated missions. Commentors stated that a cost-benefit analysis was necessary to show the value of production of medical isotopes balanced against facility

costs, in particular, the restart of FFTF, and noted that perhaps facilities would be able to pay for themselves. There were concerns that FFTF restart would take funds away from the cleanup of Hanford. Commentors noted that the decommissioning costs were not included for the restart FFTF option in the *NI Cost Report*. Several commentors remarked that the expense of plutonium-238 production cannot be justified when DOE needs to clean up existing problems at its sites.

Although the costs of proposed actions are not required by NEPA and CEQ regulations to be included in a PEIS, DOE prepared a separate *NI Cost Report*. This report would provide additional pertinent information to the Secretary of Energy so that he may make an informed decision with respect to the alternatives presented in this Final NI PEIS. Pursuant to CEQ regulations (40 CFR Section 1505.1(e)), such documents comparing alternatives should be made available to the public prior to any decision being made. DOE mailed this document to more than 730 interested parties on August 24, 2000. This report was made available immediately upon release on the NE web site (http://www.nuclear.gov) and in the public reading rooms. DOE has also provided the summary of the *NI Cost Report* in Appendix P of this Final NI PEIS.

1.5.5 Nuclear Nonproliferation Policy

Commentors expressed opinions about the nuclear nonproliferation implications of the proposed action. Commentors were concerned about keeping plutonium-238 out of the hands of third parties, and it was suggested that the purchase of plutonium-238 from Russia would stop proliferation of the material and the United States would know the disposition of the quantity purchased. Several commentors raised concerns about specific facilities described in the NI PEIS, including the FDPF and FFTF. The use of highly enriched uranium fuel in FFTF was questioned related to possible violation of U.S. nuclear nonproliferation policy. Conversely, the shutdown of FFTF that occurred previously was characterized as being done to discourage proliferation of nuclear weapons worldwide, but had instead weakened the U.S. position as a world leader in nuclear technology. There were comments about the timeliness of release of the *NI Nonproliferation Impact Assessment*, that no nonproliferation information was included in the Draft NI PEIS, and that nuclear nonproliferation policy should be considered by DOE in selection of its preferred alternative.

The plutonium being considered for production in the NI PEIS is plutonium-238, which is not the same isotope of plutonium that is used in nuclear weapons. The production of plutonium-238 does not present a nonproliferation concern. DOE developed the separate *NI Nonproliferation Impact Assessment*, published in September 2000, that analyzed the nonproliferation impacts of the actions considered in this PEIS and found that there are no U.S. nonproliferation policies, laws, regulations, or international agreements that preclude the use of any of the facilities in the manner described in the Draft NI PEIS. Although this policy analysis is not required under NEPA, it is an essential element in the decision-making process for the DOE nuclear infrastructure. A summary of the *NI Nonproliferation Impact Assessment* is included in Appendix Q of this Final NI PEIS. It is also available on the DOE NE web site (http://www.nuclear.gov).

1.5.6 Public Involvement

Commentors expressed opinions about the length of the comment period on the Draft NI PEIS, and said they wanted additional time to obtain and review relevant documents, including the *NI Cost Report* and *NI Nonproliferation Impact Assessment*. As identified in Section 1.5, the deadline for transmittal of comments was changed from September 11, 2000 (as stated in the transmittal letter of the Draft NI PEIS and the Summary) to September 18, 2000. While the official comment period ended on September 18, 2000, DOE addressed late comments to the extent practicable and considered all comments received through October 31, 2000, in preparing this Final NI PEIS. Comments that were received through September 25, 2000, along with corresponding responses, have been included in Chapter 2 of the comment response volume. Direct responses are not included to comments that were received after September 25, 2000. However, all of these comments

were considered and are characterized by other comments received during the comment period (for which a response has been provided).

Many commentors expressed the opinion that public input is intended for "show only," and that DOE has already made its decisions. Commentors also stated that they had given the same comments over and over again and that DOE representatives were not listening. DOE policy encourages effective public participation in its decision-making process. In compliance with NEPA and CEQ regulations, DOE provided opportunity to the public to comment on the scope of the NI PEIS and the environmental impact analysis of DOE's proposed alternatives. DOE gave equal consideration to all comments. In preparing this Final NI PEIS, DOE carefully considered all comments received from the public.

Some commentors expressed opinions about the conduct of the hearings, both positive and negative. The public hearing format was designed to be fair. The public hearing format used was based on stakeholder input and was presented in the Notice of Availability (65 FR 46443 et seq.) for the Draft NI PEIS. This format was intended to encourage public participation, regardless of the motivation for attending the hearing. It provided an opportunity for the participants to meet, exchange information and share concerns with DOE personnel available throughout the course of each hearing to answer questions. The meetings were facilitated by an independent moderator to ensure that all persons wishing to speak had an opportunity to do so. Persons wishing to comment were selected at random from the audiences rather than according to the order in which they registered. This was accomplished by a random number drawing. In addition to the comment recorder stationed at the main hearing, a second recorder was available in an adjacent room to receive comments without the need to await selection at the main proceeding. The hearing format promoted open and equal representation by all individuals and groups.

1.5.7 Environmental Impacts

A number of commentors questioned the results of the environmental impact analysis and cumulative impacts, specifically at Hanford. Many of these comments focused on concerns that the proposed action would result in negative impacts to the health of individuals residing in the Hanford region. The NI PEIS analyzes the impacts of the various alternatives, and the environmental impacts associated with all proposed nuclear infrastructure activities are addressed in detail in Chapter 4. Specifically, the environmental impacts associated with operation of the Hanford facilities during normal operations and from postulated accidents are presented in Section 4.3. These assessments were made using well-established and accepted analytical methods, as described in Appendixes G through L. The analytical methodology is conservative by nature; the actual impacts to the environment would be expected to be less than calculated. All impacts have been shown to be small. No fatalities among workers or the general public would be expected over the 35-year operational period. The impacts to the biosphere (air, water, and land) were also evaluated and determined to be small.

Some commentors raised specific concern over potential contamination of the Columbia River resulting from the restart of FFTF. However, FFTF is approximately 4.5 miles from the Columbia River. There are no discharges to the river from FFTF and no radioactive or hazardous discharges to groundwater. As indicated in analyses presented in Chapter 4 (e.g., Sections 4.3.1.1.4, 4.3.3.1.4, 4.4.3.1.4, 4.5.3.2.4, and 4.6.3.2.4), there would be no discernible impacts to groundwater or surface water quality at Hanford from operation of Hanford facilities that would support the nuclear infrastructure missions described in Section 1.2.

A number of commentors also expressed concern that DOE would expose individuals in the Pacific Northwest to risks associated with importing weapons-grade plutonium. None of the proposed alternatives involve the shipment of any weapons-grade plutonium to any port in the United States. Alternative 1 does postulate that DOE might decide at some point to import mixed oxide fuel from Europe to fuel FFTF. At this time, however, DOE has not proposed to import this fuel through any specific port. If DOE ultimately decides to import fuel

from Europe, it would perform a separate NEPA analysis to select a port. This review would address all relevant potential impacts of overseas and inland water transportation, shipboard fires, package handling, land transportation, as well as safeguards and security associated with the import of SNR–300 mixed oxide fuel through a variety of specific candidate ports on the west and east coasts. It would take into account all public comments, including local resolutions, concerning the desirability of bringing mixed oxide fuel into the proposed alternative ports.

In the event that DOE decides to enhance its nuclear infrastructure, it would not expose any population to high, unacceptable risks under any alternative. Any transportation activities that would be conducted by DOE would comply with U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Transportation regulations. Associated transatlantic shipments would comply with International Atomic Energy Agency requirements. In Section J.6.2, DOE reviewed the potential maximum impacts from the marine transportation of mixed oxide fuel from Europe to a representative military port (Charleston, South Carolina), and overland transportation to Hanford. Also in that section, the results of a bounding analysis show that the maximum potential radiological risks to the surrounding public from mixed oxide fuel shipments would be extremely small (e.g., less than 1 chance in a trillion for a latent cancer fatality per shipment from severe accidents at docks and in channels and less than 1 chance in 50 billion for a latent cancer fatality per shipment from overland highway accidents).

1.6 ALTERNATIVES EVALUATED IN THIS NI PEIS

This NI PEIS analyzes the potential environmental impacts using various irradiation and processing facilities to meet the following projected DOE irradiation service mission needs for 35 years: (1) production of medical, research, and industrial isotopes; (2) production of up to 5 kilograms (11 pounds) per year of plutonium-238 for use in advanced radioisotope power systems for future NASA space missions; and (3) support for U.S. civilian nuclear energy research and development activities. The proposed irradiation facilities include those that are currently operating, those that could be brought online, or those that could be constructed and operated to meet DOE's nuclear facility infrastructure requirements. This NI PEIS evaluates a No Action Alternative and five programmatic alternatives:

No Action Alternative
Alternative 1—Restart FFTF
Alternative 2—Use Only Existing Operational Facilities
Alternative 3—Construct New Accelerator(s)
Alternative 4—Construct New Research Reactor
Alternative 5—Permanently Deactivate FFTF (with No New Missions)

Each of the alternatives in this NI PEIS would contribute to fulfilling some of the proposed missions. However, none of the alternatives can completely meet all of the projected nuclear infrastructure needs. It is possible during the Record of Decision process that a combination of the alternatives could be selected, for example, a low-energy accelerator in combination with the existing reactors to optimize research isotope production, or in combination with FFTF to optimize research and therapeutic isotope production. The alternatives, their associated facility options, and their relative capabilities are described in detail in Chapter 2. DOE's Preferred Alternative for accomplishing expanded civilian nuclear energy research and development and isotope production missions in the United States is Alternative 2, Use Only Existing Operational Facilities, Option 7. Under this alternative and option, DOE would reestablish domestic production of plutonium-238, as needed, using irradiation capabilities at both ATR at INEEL and HFIR at ORNL. REDC at ORNL would be used to store neptunium-237 and to fabricate and process the targets irradiated at ATR and HFIR. The production of medical and industrial isotopes and support of civilian nuclear energy research and development

would continue and increase to the extent possible under current reactor operating levels. FFTF at Hanford would be permanently deactivated.

1.7 RELATED NEPA REVIEWS

This section provides brief summaries of NEPA documents related to ongoing DOE programs, including documents that address other aspects of DOE's nuclear facility infrastructure, the management of various waste types across the DOE complex, and activities currently under way or planned at candidate sites that are analyzed in this NI PEIS.

The *Environmental Assessment, Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington,* (DOE/EA-0993, May 1995; Finding of No Significant Impact [FONSI], May 1995) analyzes the environmental impacts associated with permanent shutdown (deactivation) of FFTF and the activities required to support it. Based on the environmental assessment, DOE determined that the proposed shutdown would not result in significant environmental impacts as defined under NEPA. This NI PEIS summarizes the impacts presented in the environmental assessment for those alternatives that consider permanent deactivation of FFTF (i.e., Alternatives 2, 3, 4, and 5) (DOE 1995a).

The *Environmental Assessment of the Import of Russian Plutonium-238* (DOE/EA-0841, June 1993; FONSI, June 1993) addresses the environmental impacts of importing plutonium-238 from Russia to augment the U.S. inventory for NASA space missions. The proposed action considers shipping up to 40 kilograms (88.2 pounds) of plutonium-238 fuel from Russia to a U.S. port, transporting the plutonium-238 within the United States to either SRS or LANL in Los Alamos, New Mexico, and if necessary, processing the material at SRS to remove impurities from the fuel. The impacts analyzed in the environmental assessment are summarized in this NI PEIS for those alternatives that do not consider re-establishing a domestic plutonium-238 production capability (i.e., the No Action Alternative and Alternative 5) (DOE 1993).

The *Final Environmental Impact Statement, Interim Management of Nuclear Materials* (DOE/EIS-0220, October 1995) analyzes the potential environmental impacts of managing certain nuclear materials at SRS pending decisions on future use or ultimate disposition, as well as the impacts of constructing the SRS Actinide Packaging and Storage Facility. Five related Records of Decision have been issued since this Final EIS was published. On December 12, 1995, DOE issued a Record of Decision and a Notice of Preferred Alternatives (60 FR 65300) concerning interim management of several categories of nuclear materials at SRS. On February 8, 1996, DOE issued a Supplemental Record of Decision (61 FR 6633) concerning stabilization of two of the remaining categories of nuclear materials (Mark-16 and Mark-22 fuel and other aluminum-clad targets) analyzed in the Final EIS. After considering a DOE staff study and recommendation on canyon facility use, DOE issued a second Supplemental Record of Decision on September 6, 1996 (61 FR 48474), concerning stabilization of the neptunium-237 solutions, obsolete neptunium targets, and plutonium-239 solutions. On April 2, 1997, DOE issued a third Supplemental Record of Decision (62 FR 17790) related to stabilization in the F-Canyon and FB-Line facilities of the remaining Taiwan Research Reactor spent nuclear fuel. On October 31, 1997, DOE issued a fourth Supplemental Record of Decision (62 FR 61099) to add another method, processing and storage for vitrification in the Defense Waste Processing Facility, to those being used to manage plutonium and uranium stored in vaults and to amend its September 6, 1996, Supplemental Record of Decision to now enable use of the SRS H-Canyon facilities to stabilize the plutonium-239 and neptunium-237 solutions stored in H-Canyon and the obsolete neptunium-237 targets stored in K-Reactor into oxide forms. This neptunium-237 oxide serves as the target material for the plutonium-238 production mission analyzed in this NI PEIS (DOE 1995b).

The *Final Environmental Impact Statement, Medical Isotopes Production Project: Molybdenum-99 and Related Isotopes* (DOE/EIS-249, April 1996; Record of Decision, September 1996 [61 FR 48921]) analyzes

the proposed establishment of a domestic capability to produce molybdenum-99 (a short-lived isotope that decays into technetium-99, an isotope used extensively for medical imaging) and related medical isotopes such as iodine-131, xenon-133, and iodine-125. At the time of this review, the U.S. supply of molybdenum-99 depended on the production capacity of one aging reactor in Canada, so DOE proposed this action to ensure a reliable domestic source for this vital isotope. In the Record of Decision, DOE selected the Annular Core Research Reactor and the Hot Cell Facility at SNL, New Mexico, for the production of molybdenum-99 and the related isotopes. Since that time, the diversity and reliability of the world supply of molybdenum-99 have increased. DOE has determined that, because the vulnerability in supplies of molybdenum-99 has sufficiently diminished, the selected SNL facilities should be further developed for molybdenum-99 production using private funds. Negotiations toward that end are ongoing. Until an agreement is reached, the reactor and hot cell facilities are available for emergency molybdenum-99 production should that need arise. The reactor is also being used for the production of other isotopes such as iodine-125, and has been made available on a services basis to serve defense missions. Any nuclear facility infrastructure enhancements analyzed in this NI PEIS would be separate from, and in addition to, the existing capabilities of these facilities (DOE 1996a).

The *Final Environmental Impact Statement, Construction and Operation of the Spallation Neutron Source* (DOE/EIS-0247, April 1999; Record of Decision, June 1999 [64 FR 35140]) analyzes the potential environmental impacts of constructing and operating a state-of-the-art spallation neutron source facility at one of four sites: ORNL (Preferred Alternative); Argonne National Laboratory–East in Argonne, Illinois; Brookhaven National Laboratory in Upton, New York; and LANL. The spallation neutron source facility is designed to provide a high-flux, short-pulsed neutron source that would give the United States' scientific and industrial research communities a much more intense source of pulsed neutrons than is currently available. Construction of this new facility would also ensure the future availability of a state-of-the-art facility as currently existing sources reach the end of their useful operating lives. In the associated Record of Decision, DOE designated ORNL as the chosen site for construction and operation of the spallation neutron source. The spallation neutron source is currently under construction, and the facility's full capacity has been dedicated to support planned missions. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action. However, the nuclear facility infrastructure enhancements analyzed in this NI PEIS would be separate from, and in addition to, the capabilities of this facility (DOE 1999a).

The *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (*Waste Management PEIS*) (DOE/EIS-0200-F, May 1997; Transuranic Waste Record of Decision, January 1998 [63 FR 3629]; Hazardous Waste Record of Decision, August 1998 [63 FR 41810]; High-Level Radioactive Waste Record of Decision, August 1999 [64 FR 46661]; Low-Level Radioactive Waste and Mixed Low-Level Radioactive Waste Record of Decision, February 2000 [65 FR 10061] examines the potential environmental and cost impacts of selected strategic alternatives for managing five types of radioactive and hazardous waste that result from nuclear defense and research activities at sites throughout the United States. The *Waste Management PEIS* provides information on the impacts of the various siting configurations DOE will use to decide where to locate additional treatment, storage, and disposal capacity for each waste configuration. In the transuranic waste Record of Decision, DOE determined that those sites that currently have or will generate transuranic waste will prepare it for storage and store it on site, except SNL, which will transfer its transuranic waste to LANL. The hazardous waste Record of Decision states that DOE will continue the use of offsite facilities to treat nonwastewater hazardous waste, with the exception that Oak Ridge Reservation (ORR) and SRS will treat some of their own nonwastewater hazardous waste on site. The high-level radioactive waste Record of Decision states that immobilized high-level radioactive waste will be stored at Hanford, INEEL, SRS, and the West Valley Demonstration Project in New York until a geologic repository is licensed by the NRC. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision states that DOE will minimally treat low-level radioactive waste at the generator sites, and that Hanford and the Nevada Test Site will be made available to

all DOE sites for low-level radioactive waste disposal. As part of this decision, DOE will treat mixed low-level radioactive waste at INEEL, ORR, and SRS; dispose of mixed low-level radioactive waste at the Nevada Test Site; and both treat and dispose of mixed low-level radioactive waste at Hanford. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action. This NI PEIS analysis also assumes that waste generated as part of the NI PEIS proposed action would be managed in accordance with these decisions (DOE 1997a).

The *Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada*, (DOE/EIS-0250D, July 1999) analyzes the construction, operation and monitoring, and eventual closure of a potential geologic repository at Yucca Mountain to dispose of commercial and DOE spent nuclear fuel, high-level radioactive waste, and materials that NRC determines by rule require the same degree of isolation. National transportation, Nevada transportation, and waste packaging are evaluated as part of the analysis. Three implementing design alternatives based on thermal load—low, intermediate, and high—are examined. This NI PEIS assumes for analysis purposes that Yucca Mountain is a potential geologic repository site for spent nuclear fuel produced as a result of enhancing the U.S. nuclear facility infrastructure (DOE 1999b).

The *Tank Waste Remediation System, Hanford Site, Richland, Washington, Final Environmental Impact Statement* (DOE/EIS-0189, August 1996; Record of Decision, February 1997 [62 FR 8693]) satisfies the DOE commitment made in the *Disposal of Hanford Defense High-Level, Transuranic and Tank Waste Final Environmental Impact Statement* (DOE/EIS-0113, December 1987; Records of Decision March and April 1988) to prepare a supplemental NEPA analysis. The *Tank Waste Remediation System EIS* was prepared in response to several important changes subsequent to the Record of Decision, including a revised strategy for managing and disposing of tank waste and encapsulated cesium and strontium. As part of the proposed action, the *Tank Waste Remediation System EIS* evaluates continued operation and management of the tank farms, waste transfer system upgrades, and retrieval and treatment of the tank waste, including construction and operation of a facility to vitrify high-level radioactive waste and to vitrify or similarly immobilize low-level radioactive waste. DOE decided to implement the Preferred Alternative for retrieval, treatment, and disposal of tank waste and to defer a decision on the disposition of cesium and strontium capsules. Two supplemental analyses were prepared for the *Tank Waste Remediation System EIS*. The first was *Proposed Upgrades to the Tank Farm Ventilation, Instrumentation, and Electrical Systems under Project W-314 in Support of Tank Farm Restoration and Safe Operations* (DOE/EIS-0189-SA1, June 1997). Based on these supplemental analyses, it was determined that upgrades or planned upgrades to the tank farm do not pose any additional potential environmental impacts and, therefore, no additional NEPA analysis is required. The second supplemental analysis was for the *Tank Waste Remediation System* (DOE/EIS-0189-SA2, May 1998). This analysis provides information on the most recent inventory of chemical and radiological constituents in the tanks and the new waste to be sent to the tanks for treatment. Based on the new data, it was concluded that there would be minimal changes from the impacts identified in the *Tank Waste Remediation System EIS* and, therefore, no additional NEPA analysis is required. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1996b).

The *Final Hanford Comprehensive Land Use Plan Environmental Impact Statement* (DOE/EIS-0222-F, September 1999; Record of Decision, November 1999 [64 FR 61615]) focuses on developing an overall strategy for future land use at Hanford and includes a proposed comprehensive land-use plan. The Preferred Alternative, which DOE selected in the Record of Decision, is to consolidate waste management operations in the Central Plateau; allow industrial development in the eastern and southern portions of the site; increase recreational access to the Columbia River; and expand Saddle Mountain National Refuge to include all of the Wahluke Slope, McGee Ranch, and Fitzner-Eberhardt Arid Lands Ecology Reserve. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1999c).

The *Hanford Reach of the Columbia River Comprehensive River Conservation Study and Environmental Impact Statement* (National Park Service June 1994; Record of Decision, July 1996) evaluates impacts related to protection of the Hanford Reach of the Columbia River as a Wild and Scenic River, increased recreation access, and visitor interpretation and education. In the Record of Decision, the National Park Service recommended that Congress designate the Hanford Reach of the Columbia River, public land within one-quarter mile of the river, and all public land on the Wahluke Slope as a new National Wildlife Refuge and National Wild and Scenic River. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (NPS 1994).

The *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE/EIS-0203-F, April 1995; Record of Decision, May 1995 [60 FR 28680]) is a complex-wide evaluation of the alternatives for managing the existing and projected amounts of spent nuclear fuel within the DOE inventory through 2035. The EIS contains an analysis of the impacts of transporting spent nuclear fuel, as well as sitewide alternatives for environmental restoration and waste management programs at INEEL. In the associated Record of Decision, DOE designated Hanford, INEEL, and SRS for regional spent fuel storage and management and made decisions about environmental restoration and waste management activities at INEEL. In March 1996, DOE issued an amendment to the May 1995 Record of Decision to include a decision to regionalize the management of DOE-owned spent nuclear fuel by fuel type, including spent fuel currently stored at Hanford, INEEL, and SRS. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1995c).

The *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (DOE/EIS-0218F, February 1996; Record of Decision, July 1996 [61 FR 38720]) evaluates the adoption of a joint DOE/Department of State policy to manage spent nuclear fuel from foreign research reactors, including highly enriched uranium provided by the United States to other countries for research reactors. Management alternatives include a number of implementation options for port selection, transportation, and storage at DOE sites. In the Record of Decision, DOE selected a management policy that returned spent nuclear fuel from various foreign research reactors to the United States using two designated U.S. ports and the management at INEEL and SRS. A supplement analysis (DOE/EIS-0218-SA-2, August 1998) was prepared to examine acceptance of foreign research reactor spent nuclear fuel under three scenarios not specifically examined in the EIS: (1) accepting spent nuclear fuel not included in EIS-estimated inventories, (2) accepting spent nuclear fuel from countries in quantities greater than the quantities identified in the EIS, and (3) transporting more than eight casks of spent fuel on a single oceangoing vessel. The supplement analysis concluded that the potential environmental impacts of these actions are bounded by the analysis performed in the EIS and, therefore, no supplement to the EIS is needed. In turn, DOE issued a revision to the Record of Decision on July 19, 2000 (65 FR 44767) to allow the shipment of up to 16 casks of spent nuclear fuel on a single ocean-going vessel transporting foreign research reactor spent nuclear fuel to the United States. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1996c).

The DOE INEEL *Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* (DOE/EIS-0290, January 1999; Record of Decision, April 1999 [64 FR 16948]) evaluates four alternatives: (1) a No Action Alternative under which existing waste management operations, facilities, and projects would continue; (2) the proposed action/Preferred Alternative under which BNFL, Inc., would build and operate an advanced mixed waste treatment project facility using proposed thermal and nonthermal treatment technologies for certification and shipment to the Waste Isolation Pilot Plant or to another acceptable disposal facility; (3) a nonthermal treatment alternative under which some treatment of transuranic, alpha, and low-level mixed waste would occur at an advanced mixed waste treatment project facility at the same location as the proposed action, and waste that requires thermal treatment would be repackaged for storage; and (4) a treatment and storage

alternative that would include the same processes as the proposed action/Preferred Alternative, except the treated waste would be placed in Resource Conservation and Recovery Act–permitted storage units at the onsite Radioactive Waste Management Complex for long-term storage. In the Record of Decision, DOE selected the Preferred Alternative, although construction of the thermal treatment component of this alternative has been deferred pending the recommendation of a blue ribbon panel of experts assessing possible technology alternatives. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1999d).

The *Site-Wide Environmental Impact Statement for the Continued Operation of the Los Alamos National Laboratory* (DOE/EIS-0238, January 1999; Record of Decision, September 1999 [64 FR 50797]) evaluates ongoing and projected new operations and facilities at LANL in support of DOE missions, including the storage of plutonium-238. A decision was made in the LANL Record of Decision to implement the Preferred Alternative, which includes expansion of operations as necessary, increases in existing operations to the greatest reasonably foreseeable levels, and full implementation of the mission elements assigned to LANL. Because the remaining U.S. inventory of usable plutonium-238 is stored at LANL, the NI PEIS evaluates the transport to LANL of the plutonium-238 product resulting from an enhanced nuclear facility infrastructure (DOE 1999e).

The *Final Programmatic Environmental Impact Statement for Tritium Supply and Recycling* (DOE/EIS-0161, October 1995; Record of Decision, December 5, 1995 [60 FR 63878]) evaluates the siting, construction, and operation of tritium supply technology alternatives and recycling facilities at five candidate sites, as well as the use of a CLWR for producing tritium, a gaseous radioactive isotope of hydrogen considered essential to the operation of U.S. thermonuclear weapons. In the Record of Decision, DOE selected a dual-track approach. One track explores the purchase of an operating or partially complete CLWR, or the purchase of irradiation services from such a reactor. A second track would design, build, and test critical components of an accelerator system for production of tritium. The Record of Decision stated that DOE would select one of the alternatives at a later date to serve as the primary tritium source for the U.S. nuclear weapons stockpile, while the other alternative will be developed as a backup source, if feasible. The Record of Decision also stated that DOE would determine whether the operation of FFTF might be able to play any role in future tritium requirements. On December 22, 1998, the Secretary of Energy announced his selection of the CLWR as the primary tritium supply and that an accelerator would be developed but not constructed. In addition, DOE decided that FFTF would have no role in tritium supply plans. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action. Since the accelerator for production of tritium would not be built, it was not considered as a reasonable alternative in the NI PEIS (DOE 1995d).

The *Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement* (DOE/EIS-0240, June 1996; Record of Decision, July 1996 [61 FR 40619) addresses the disposition of a nominal 220 tons of highly enriched uranium declared surplus to the national security needs of the United States. Alternatives include several approaches to blending down the highly enriched material to make it non-weapons-usable and suitable for fabrication into fuel for commercial nuclear reactors. The Record of Decision identifies DOE's intent to blend, over time, as much material as possible (up to 85 percent) for commercial use and blending the remainder for disposal as low-level radioactive waste. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1996d).

The *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE/EIS-0229, December 1996; Record of Decision, January 1997 [63 FR 43386]) analyzes the environmental impacts of alternatives considered for the long-term storage of weapons-usable fissile materials (highly enriched uranium and plutonium) and for the disposition of weapons-usable plutonium that has been declared surplus to national security needs. The Record of Decision encompasses two categories of plutonium

decisions: (1) the sites and facilities for the storage of nonsurplus plutonium and the storage of surplus plutonium pending disposition; and (2) the programmatic strategy for disposition of surplus plutonium. It also announces the decision to store surplus and nonsurplus highly enriched uranium in upgraded facilities at ORR. DOE studies indicated that significant cost savings could be realized from the transfer of nonpit materials from the Rocky Flats Environmental Technology Center and Hanford earlier than indicated in the *Storage and Disposition PEIS* Record of Decision. DOE issued an amended Record of Decision (August 1998) that supports the early closure of the Rocky Flats Environmental Technology Center and the early deactivation of plutonium storage facilities at Hanford. The amended Record of Decision includes decisions to accelerate shipment of all nonpit surplus plutonium from the Rocky Flats Environmental Technology Center to SRS and the relocation of all Hanford surplus plutonium to SRS, if SRS were selected as the immobilization site. A supplement analysis to the *Storage and Disposition PEIS*, the *Supplement Analysis for Storing Plutonium in the Actinide Packaging and Storage Facility and Building 105–K at the Savannah River Site*, was issued in July 1998. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1996e).

The *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE/EIS-0283, November 1999; Record of Decision, January 2000 [65 FR 1608]) was tiered from the *Storage and Disposition PEIS* and evaluated reasonable alternatives for the siting, construction, and operation of facilities required to implement DOE's disposition strategy for up to 50 metric tons of surplus plutonium. The disposition facilities analyzed in this EIS include a pit disassembly and conversion facility, a plutonium conversion and immobilization facility, and a mixed oxide fuel fabrication facility. The analyses also considered the potential impacts of fabricating a limited number of lead fuel assemblies for testing in a reactor. In the Record of Decision, DOE decided to provide for the safe, secure disposition using a hybrid approach of immobilizing approximately 17 metric tons (19 tons) and using up to 33 metric tons (36 tons) as mixed oxide fuel. DOE also decided to construct and operate each of the three disposition facilities at SRS, fabricate the lead assemblies at LANL, and conduct postirradiation examination of the lead assemblies at ORNL. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1999f).

The *Final Programmatic Environmental Impact Statement for Alternative Strategies for the Long-Term Management and Use of Depleted Uranium Hexafluoride* (DOE/EIS-0269, April 1999; Record of Decision, August 1999 [64 FR 43358]) evaluates the environmental impacts of six alternative strategies for the long-term management of DOE-owned depleted uranium hexafluoride currently stored at the East Tennessee Technology Park in Oak Ridge, Tennessee; the Paducah Gaseous Diffusion Plant near Paducah, Kentucky; and the Portsmouth Gaseous Diffusion Plant near Piketon, Ohio. These alternatives involve cylinder technology and design; conversion of depleted uranium hexafluoride to another chemical form; and materials use, storage, disposal, and transportation. As indicated in the Record of Decision, DOE selected the Preferred Alternative—to begin conversion of the depleted uranium hexafluoride as soon as possible, either to uranium oxide, uranium metal, or a combination of both, while allowing for future use of as much of this inventory as possible. This NI PEIS analyzes the conversion of depleted uranium hexafluoride from a representative site (Portsmouth) to uranium dioxide, which would be used as feedstock for immobilization and mixed oxide fuel and lead assembly fabrication. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1999g).

The *Final Programmatic Environmental Impact Statement for Stockpile Stewardship and Management* (DOE/EIS-0236, September 1996; Record of Decision, December 1996 [61 FR 68014]) evaluates the potential environmental impacts resulting from activities associated with nuclear weapons research, design, development, and testing, as well as the assessment and certification of their safety and reliability. The stewardship portion of the document analyzes the development of three new facilities to provide enhanced experimental capabilities. The stockpile management portion of the EIS concerns producing, maintaining,

monitoring, refurbishing, and dismantling the nuclear weapons stockpile at eight sites, including Pantex and SRS. In the Record of Decision, DOE selected to downsize a number of facilities for stockpile dismantlement and to build experimental facilities at Lawrence Livermore National Laboratory. A draft supplement analysis (DOE/EIS-0236-SA, June 1999) was prepared to examine the plausibility of a building-wide fire at LANL's plutonium facility and to examine new studies regarding seismic hazards at LANL. The draft supplement analysis was issued for public comment and a final supplemental analysis was issued on September 2, 1999. The supplement analysis concluded that there is no need to prepare a supplemental EIS. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1996f).

The *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306, July 2000; Record of Decision, September 2000 [65 FR 56565]) evaluates strategies to remove or stabilize the reactive sodium contained in a portion of DOE's spent nuclear fuel inventory to prepare the spent nuclear fuel for disposal in a geologic repository. The EIS analyzes, under the proposed action, six alternatives that employ one or more of the following technology options at nuclear fuel management facilities at SRS or INEEL: electrometallurgical treatment; the plutonium-uranium extraction process; packaging in high-integrity cans; and the melt and dilute treatment process. In the Record of Decision, DOE decided to implement the preferred alternative of electrometallurgically treating the Experimental Breeder-II spent nuclear fuel and miscellaneous small lots of sodium-bonded spent nuclear fuel at Argonne National Laboratory–West at INEEL. Because of the different physical characteristics of the Fermi-1 sodium-bonded blanket spent nuclear fuel also analyzed in the EIS, DOE decided to continue to store this material while alternative treatments are evaluated. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 2000b).

The *Final Environmental Impact Statement (EIS) for Treating Transuranic (TRU)/Alpha Low-Level Waste at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/EIS-0305-F, June 2000; Record of Decision, August 2000 [65 FR 48683]) evaluates alternatives to construct, operate, and decontaminate and decommission a transuranic waste treatment facility in Oak Ridge, Tennessee. Four types of waste would be treated at the proposed facility including remote-handled transuranic mixed waste sludge, liquid low-level radioactive waste associated with the sludge, contact-handled transuranic/alpha low-level radioactive waste solids, and remote-handled transuranic/alpha low-level radioactive waste solids. The EIS analyzes the potential environmental impacts associated with five alternatives—No Action; the Low-Temperature Drying Alternative, which is DOE's Preferred Alternative; the Vitrification Alternative; the Cementation Alternative; and the Treatment and Waste Storage at ORNL Alternative. In the Record of Decision, DOE selected the preferred alternative of constructing and operating a Transuranic (TRU) Waste Treatment Facility that will use a low-temperature drying process for treating TRU mixed waste sludge and associated low-level waste supernate, and that will treat TRU solid waste by sorting and compacting. Any solid waste containing hazardous constituents regulated under RCRA will be encapsulated. DOE will dispose of the treated TRU waste at the Waste Isolation Pilot Plant near Carlsbad, New Mexico, and treated low-level waste at DOE's Nevada Test Site. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from this NI PEIS proposed action (DOE 2000c).

The *Environmental Assessment of Melton Valley Storage Tanks Capacity Increase Project at the Oak Ridge National Laboratory in Oak Ridge, Tennessee* (DOE/EA-1044, April 1995) evaluates the potential impacts of the construction and maintenance of additional storage capacity at ORNL in Oak Ridge, Tennessee, for liquid low-level radioactive waste. New capacity would be provided by a facility partitioned into six individual tank vaults containing one 100,000-gallon liquid low-level radioactive waste storage tank each. Alternatives

considered include the No Action Alternative, ceasing generation, storage at other ORR storage facilities, source treatment, pretreatment, and storage at other DOE facilities. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1995e).

The *Environmental Assessment and FONSI for the Management of Spent Nuclear Fuel on the Oak Ridge Reservation Oak Ridge, Tennessee* (DOE/EA-1117, February 1996) evaluates the potential impacts of the management of spent nuclear fuel on the DOE Oak Ridge Reservation. Spent nuclear fuel would be retrieved from storage; transferred by truck to a hot cell facility, if segregation by fuel type and/or repackaging is required; loaded into containers/transport casks that meet regulatory requirements; and shipped via truck to offsite storage at either SRS or INEEL. The proposed action may also include construction and operation of a dry cask spent nuclear fuel storage facility on ORR to enable reactor operations to continue in the event of an interruption of offsite spent nuclear fuel shipment. In the No Action Alternative, neither construction of a dry cask storage facility nor shipment of spent nuclear fuel from ORR would occur. Spent nuclear fuel would remain at present storage locations on ORR. Due to space limitations, operations on ORR that generate spent nuclear fuel would have to cease, including operation of the High Flux Isotope Reactor. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1996g).

The *Final Environmental Impact Statement, Management of Spent Nuclear Fuel from the K Basins at the Hanford Site, Richland, Washington* (DOE/EIS-0245F, January 1996; Record of Decision, March 1996 [61 FR 10736]) analyzes the potential environmental impacts of alternatives for managing the spent nuclear fuel located in the K-East and K-West spent nuclear fuel storage basins at Hanford. In the Record of Decision, DOE decided to implement the preferred alternative evaluated in the FEIS with two modifications. The preferred alternative consists of removing the spent nuclear fuel from the basins, vacuum drying, conditioning and sealing it in inert-gas filled canisters for dry vault storage in a new facility, to be built at Hanford, for up to 40 years pending decisions on ultimate disposition. The K Basins will continue to be operated during the period over which the preferred alternative is implemented. The preferred alternative also includes transfer of the basin sludge to Hanford's double-shell tanks for management, disposal of non-spent-nuclear-fuel basin debris in a low-level burial ground at Hanford, disposition of the basin water, and deactivation of the basins pending decommissioning. The two modifications to the preferred alternative that were presented in the Record of Decision addressed the management of the sludge, and the timing of placement of the spent nuclear fuel into the transportation casks: (1) should it not be possible to put the sludge into the double-shell tanks, the sludge will either continue to be managed as spent nuclear fuel, or disposed of as solid waste; and (2) to reduce the radiation exposure to workers, the multicanister overpacks will be placed inside the transportation casks before the spent nuclear fuel is loaded into them, instead of loading the spent nuclear fuel into the multicanister overpacks prior to placing them inside the transportation casks. A supplement analysis (DOE/EIS-0245-SA-01, August 1998) was prepared to examine the potential impacts of deleting the hot conditioning/passivation step from the preferred alternative in the ROD. Based on this supplement analysis, DOE determined that there would be minimal changes from the impacts previously identified in the EIS, and therefore no additional NEPA analysis is required. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1996h).

The *Environmental Assessment, Management of Hanford Site Non-Defense Production Reactor Spent Nuclear Fuel (*DOE/EA-1185, March 1997; FONSI, March 1997) assesses the environmental impacts associated with the management of nondefense production reactor spent nuclear fuel at Hanford, and associated activities to support this work. Under the proposed action, DOE would consolidate the site's inventory of this material, stored in various facilities throughout the site, in a cost-effective, radiologically and industrially safe and passive storage condition, pending final disposition. Alternatives considered in the review process included the No Action alternative; the preferred alternative to consolidate Hanford's inventory of nondefense

production reactor spent nuclear fuel in aboveground and vault storage in the 200 Area; and alternatives addressing aboveground dry cask storage in the 400 Area and vault storage in the 200 Area. The impacts of this action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 1997b).

The *Environmental Assessment for Transportation of Low-Level Radioactive Waste from the Oak Ridge Reservation to Off-Site Treatment or Disposal Facilities* (DOE/EA-1315, April 2000) evaluates the potential impacts of the transportation of low-level radioactive waste from ORR in Tennessee for treatment or disposal at various locations in the United States. Low-level radioactive waste from three ORR facilities, ORNL, Oak Ridge Y–12 Plant, and East Tennessee Technology Park, is proposed to be loaded and transported to destinations representative of other DOE sites and licensed commercial nuclear waste treatment or disposal facilities. The treatment and/or disposal facilities include Envirocare of Utah, Inc.; Waste Control Specialists; commercial treatment or disposal facilities near SRS in Aiken, South Carolina; commercial facilities near ORR; commercial facilities near Hanford; and facilities at DOE sites such as the Nevada Test Site, the Hanford Reservation, and SRS. In the No Action Alternative, DOE would not ship and dispose of the existing and projected large quantities of ORR low-level radioactive waste at offsite radioactive waste disposal facilities. Relatively small volumes of ORR low-level radioactive waste would continue to be shipped to DOE or commercial disposal facilities under existing and previously approved categorical exclusions. Low-level radioactive waste would continue to be stored on the ORR site, eventually requiring additional low-level radioactive waste storage facilities. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from this NI PEIS proposed action (DOE 2000d).

The *Environmental Assessment for Selection and Operation of the Proposed Field Research Centers for the Natural and Accelerated Bioremediation Research (NABIR) Program* (DOE/EA-1196, March 2000; FONSI, April 2000) evaluates the potential impacts of adding a Field Research Center component to the existing NABIR Program. The NABIR Program is a 10-year fundamental research program designed to increase the understanding of fundamental biogeochemical processes that would allow the use of bioremediation approaches for cleaning up DOE's contaminated legacy waste sites. The Field Research Center would be integrated with the existing and future laboratory and field research and would provide a means of examining the fundamental biogeochemical processes that influence bioremediation under controlled small-scale field conditions. The environmental assessment analyzes the No Action Alternative and two alternative sites under the proposed action: ORNL/Y–12 Site, Oak Ridge, Tennessee, and Pacific Northwest National Laboratory/ DOE Hanford 100-H Area, Richland, Washington. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from this NI PEIS proposed action (DOE 2000e).

The *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement* (DOE/EIS-0287D, December 1999) analyzes the potential environmental consequences of managing two waste types at INEEL. The two waste types are high-level radioactive waste in a calcine form and liquid mixed transuranic waste, historically known as sodium-bearing waste and newly generated liquid waste. The disposition of existing and proposed high-level radioactive waste facilities after their missions have been completed is also analyzed. The waste processing alternatives are No Action, Continued Current Operations, Separations, Non-Separations, and Minimum INEEL Processing. The facilities' disposition alternatives are No Action, Clean Closure, Performance-Based Closure, Closure to Landfill Standards, Performance-Based Closure with Class A Grout Disposal, and Performance-Based Closure with Class C Grout Disposal. The impacts of the proposed action are factored into the assessment of potential impacts resulting from the NI PEIS proposed action (DOE 1999h).

The *Draft Environmental Assessment for Transportation of Low-Level Radioactive Mixed Waste from the Oak Ridge Reservation to Off-Site Treatment or Disposal Facilities* (DOE/EA-1317, July 2000) evaluates the potential impacts of transportation of low-level radioactive mixed waste from Oak Ridge, Tennessee, to

treatment and disposal facilities in various locations in the United States. Low-level radioactive mixed waste from three ORR facilities, East Tennessee Technology Park, ORNL, and the Y–12 Plant, is proposed to be packaged as required, loaded, and shipped to licensed, commercial nuclear waste treatment or disposal facilities. The treatment and/or disposal facilities include Envirocare of Utah, Inc.; Waste Control Specialists; Nevada Test Site; and commercial treatment or disposal facilities near SRS in Aiken, South Carolina, ORR, and the Hanford Site in Richland, Washington. The No Action Alternative of continuing to store most low-level radioactive mixed waste on site, and eventually requiring additional low-level radioactive mixed waste storage facilities is also analyzed. The impacts of the proposed action are factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action (DOE 2000f).

1.8 CHANGES FROM THE DRAFT NI PEIS

In response to comments on the Draft NI PEIS and as a result of information that was unavailable at the time of its issuance, this Final NI PEIS contains revisions and new information. These revisions and new information are indicated by sidebars. A brief discussion of the most important changes included in this Final NI PEIS is provided in the following paragraphs.

Chapter 1

Purpose and Need for Agency Action

As a result of public comments, additional discussion was incorporated to address DOE's production of medical, research, and industrial isotopes relative to global isotope production and availability. In addition, the discussion of the need for plutonium-238 production for space missions was expanded and updated to reflect the most recent planning guidance provided by NASA to DOE.

Issues Raised During the Public Comment Period on the Draft NI PEIS

Section 1.5, Issues Raised During the Public Comment Period on the Draft NI PEIS, was added to this Final NI PEIS.

Related NEPA Reviews

The Final NI PEIS was revised to add descriptions of the *Final Environmental Impact Statement, Management of Spent Nuclear Fuel from the K Basins at the Hanford Site, Richland, Washington* (DOE/EIS-0245F), and the *Environmental Assessment, Management of Hanford Site Non-Defense Production Reactor Spent Nuclear Fuel (*DOE/EA-1185). The impacts of these NEPA actions were factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action.

This Final NI PEIS was also revised to reflect recent Records of Decision that have been issued for the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (DOE/EIS-0218F), the *Final Environmental Impact Statement for Treating Transuranic (TRU)/Alpha Low-Level Waste at the Oak Ridge National Laboratory Oak Ridge, Tennessee* (DOE/EIS-0305), and the *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306).

Changes from the Draft NI PEIS

Section 1.8, Changes from the Draft NI PEIS, was added to this Final NI PEIS.

Chapter 2

Transportation Requirements

Additional U.S. ports were named as candidates for receiving mixed oxide fuel from Europe.

Alternatives Considered and Dismissed

Information was provided to explain why the Isotope Production Facility at LANL, the Brookhaven LINAC Isotope Producer and the Alternating Gradient Synchrotron accelerator complex at Brookhaven National Laboratory, and CLWRs were not considered reasonable alternatives for the production of medical isotopes.

Information was also provided to explain why increasing the power levels at ATR and/or HFIR or installing rapid radioisotope retrieval systems would be insufficient to meet the long-term growth projection needs and therefore were dismissed as reasonable alternatives.

Preferred Alternative

The discussion of DOE's preferred alternative for accomplishing the proposed action, i.e., Alternative 2, Use Only Existing Operational Facilities, Option 7, is included in this Final NI PEIS.

Summary of Environmental Impacts

Section 2.7 was revised in response to comments that it was difficult to compare environmental impacts among alternatives. Although estimates of the environmental impacts that would result from implementation of the alternatives are the same as those in the Draft NI PEIS, the tables and accompanying text were reformatted for ease in comparing environmental impacts among alternatives and among options within alternatives. Section 2.7 was also revised to focus on incremental impacts that would result from implementation of the alternatives. Baseline environmental impacts were removed from the comparisons among alternatives and options. This information is now presented in Chapter 3.

Chapter 3

Affected Environment

Additional information was provided on the environmental baseline at each site, including graphics to more clearly illustrate existing surface water and groundwater conditions. Estimates of existing impacts for current HFIR/REDC operations were added to Sections 3.2.3.2 (Air Quality), 3.2.9.1.2 (Radiation Exposure and Risk), and 3.2.11.1 (Waste Inventories and Activities). Similarly, estimates for current ATR operations were added to Sections 3.3.3.2 (Air Quality), 3.3.9.1.2 (Radiation Exposure and Risk), and 3.3.11.1 (Waste Inventories and Activities). Estimates of existing impacts of maintaining FFTF in standby were added to Section 3.4.3.1 (Air Quality). Information was also provided on the impacts of the range fires affecting Hanford and INEEL during the summer of 2000. In addition, site data were updated to reflect recent measurements and analyses.

In response to public comments on the Draft NI PEIS, additional information on health studies conducted in the Hanford area was also incorporated.

Chapter 4

Air Quality

Stack parameters used for the air quality modeling were added. In response to public comment, estimates of the ambient air quality concentrations from FFTF sources were added to the deactivation section.

Water Resources

New water use and sanitary wastewater generation increments for REDC and FDPF were added to reflect the revised additional workforce required at these facilities and to be consistent with FMEF. Water use and wastewater generation rates for the new accelerator(s) and new research reactor alternatives were also revised. These changes were also incorporated into the waste management analyses.

Ecological and Cultural and Palentological Resources

These sections were updated to reflect that consultations concerning threatened and endangered species and cultural resources were conducted with appropriate Federal and state agencies. Consultations were also conducted with interested Native American tribes. No major issues were raised as a result of these consultations.

Socioeconomics

Section 4.3.1.1.8 was revised to reflect changes in the number of workers associated with FFTF operations and deactivation. The associated impacts on community services were also incorporated. In addition, the number of workers at the Oak Ridge Reservation was revised to reflect the entire site workforce rather than just the number of workers at the Oak Ridge National Laboratory.

Normal Operations

Based on more recent site data on occupational radiation exposure for workers at REDC, all worker health impacts for target processing at REDC, FMEF, and FDPF and for neptunium target storage at REDC, Chemical Processing Plant–651, and FMEF were updated. Also, low-energy accelerator source terms were modified to properly reflect normal operational emissions resulting in modifications to the population health impacts for all options of Alternative 3.

Facility Accidents

The high-energy accelerator analysis was redone to incorporate a more accurate revised source term, and the incremental risks for currently operating reactors were added to the tables. An additional analysis addressing industrial accidents was also performed and incorporated into Chapter 4.

Transportation

The neptunium inventory was revised to use the recently declassified actual inventory. The number of actual shipments from SRS to the processing facilities and the transportation risk estimates were modified accordingly.

Waste Management

The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. The Waste Management sections (i.e., Sections 4.3.1.1.13, 4.3.2.1.13, 4.3.3.1.13, and 4.4.3.1.13) were revised to reflect this different classification from what was assumed in the Draft NI PEIS.

Spent Nuclear Fuel Management

These sections were revised to quantify the generation of spent fuel from 35 years of operation and to state that dry spent nuclear fuel storage at the FFTF site is similar to NRC-approved methods currently being used for interim storage of commercial spent nuclear fuel. In addition, based on public comments, reference was added about the K Basins spent fuel storage.

Cumulative Impacts

Cumulative impact tables in Section 4.8 were revised to present the contributions from each of the various site actions anticipated during the course of the operational period evaluated in this NI PEIS.

The air quality tables were also revised to incorporate the revised baseline from Chapter 3. In addition, waste management tables were revised to include the sites' treatment, storage, and disposal capacities for easier comparison of the waste generations by waste type to the waste management capacities at the sites.

Chapter 5

In response to public comments, a list of organizations that DOE contacted during the consultation process was added.

Volume 2

Summaries of the *NI Cost Report* and *NI Nonproliferation Impact Assessment* were added as Appendixes P and Q, respectively. NASA mission guidance correspondence was added as Appendix R.

Volume 3

Volume 3 of the NI PEIS was added to present the comments received during the public review period for the Draft NI PEIS and DOE's responses to these comments.

1.9 STRUCTURE OF THIS NI PEIS

This NI PEIS contains 9 chapters and 18 appendixes. The main analyses are included in the chapters in Volume 1, with additional project information provided in the appendixes in Volume 2. Comments received during the public comment period and DOE's associated responses are presented in Volume 3. A Summary of this NI PEIS is also included.

The nine chapters provide the following information:

Chapter 1—Introduction: Background; purpose and need for agency action; decisions to be made; issues identified during the public scoping process; issues raised during the public comment period on the Draft NI PEIS; alternatives evaluated; related NEPA documents; and changes from the Draft NI PEIS

Chapter 2—Program Description and Alternatives: Program missions; candidate facilities and proposed options to enhance U.S. nuclear infrastructure and provide the capabilities needed to meet DOE's mission requirements; operations required to implement DOE program missions and the candidate sites and facilities for these activities; transportation activities associated with the program missions; alternatives considered reasonable for detailed evaluation; alternatives and facilities considered and dismissed from evaluation; a summary of the environmental impacts; implementation schedules associated with the alternatives evaluated; a comparative evaluation of alternatives; and the description of the Preferred Alternative.

Chapter 3—Affected Environment: Aspects of the environment that could be affected by the NI PEIS alternatives

Chapter 4—Environmental Consequences: Analyses of the potential impacts of the NI PEIS alternatives on the environment

Chapter 5—Applicable Laws, Regulations, and Other Requirements: Environmental, safety, and health regulations that would apply to the alternatives of this NI PEIS, and agencies consulted for their expertise

Chapter 6—List of Preparers

Chapter 7—Distribution List

Chapter 8—Glossary

Chapter 9—Index

The 18 appendixes provide the following information:

Appendix A—Plutonium-238 Target Fabrication and Processing Operations

Appendix B—Neptunium-237 Target Irradiation Operations in Currently Operating Reactors for Plutonium-238 Production

Appendix C—Medical and Industrial Isotope Target Fabrication and Processing Operations and Civilian Nuclear Research and Development Targets

Appendix D—Fast Flux Test Facility Operations

Appendix E—Research Reactor Operations

Appendix F—New Accelerator(s)

Appendix G—Methods for Assessing Environmental Impacts

Appendix H—Evaluation of Human Health Effects from Normal Facility Operations

Appendix I—Evaluation of Human Health Effects from Facility Accidents

Appendix J—Evaluation of Human Health Effects of Transportation

Appendix K—Environmental Justice Analysis

Appendix L—Socioeconomics Analysis

Appendix M—Ecological Resources

Appendix N—Public Scoping Process

Appendix O—Contractor Disclosure Statement

Appendix P—*Nuclear Infrastructure Cost Report* Summary

Appendix Q—*Nuclear Infrastructure Nonproliferation Impact Assessment* Summary

Appendix R—NASA Mission Planning Correspondence

1.10 REFERENCES

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Chapter 2
Program Description and Alternatives

Chapter 2 describes the program missions and the logic behind the structure of the program alternatives. Section 2.1 introduces the facility alternatives and options proposed to enhance the U.S. nuclear infrastructure and provide the capabilities needed to meet DOE's mission requirements. This section also presents the No Action Alternative and an alternative suggested by some of the public scoping comments. Sections 2.2 and 2.3, respectively, describe the operations necessary to implement DOE's program missions and the candidate sites and facilities where the operations would take place. Section 2.4 discusses the transportation activities associated with the program missions. Section 2.5 describes the alternatives that were considered reasonable for detailed evaluation. Section 2.6 explains why some other alternatives and facilities were considered and dismissed from evaluation in this NI PEIS. Section 2.7 summarizes the environmental impacts and implementation schedules associated with the alternatives that were evaluated and provides a comparative evaluation of these alternatives in terms of impacts and mission effectiveness. The chapter concludes with the description of the preferred alternative.

2.1 INTRODUCTION

As discussed in Chapter 1, the U.S. Department of Energy (DOE) is proposing to enhance its existing nuclear facility infrastructure to accommodate new and expanding missions in the areas of civilian nuclear energy research and development and isotope production. DOE currently does not have sufficient steady-state neutron sources to meet its projected irradiation needs for: (1) isotopes for medical and industrial uses, (2) plutonium-238 for potential use in advanced radioisotope (radioactive isotope) power systems and heating units for future U.S. National Aeronautics and Space Administration (NASA) space exploration missions, and (3) other irradiation services to meet the Nation's civilian nuclear energy research and development needs.

The programmatic alternatives focus on the use of irradiation facilities that are currently operating, could be brought on line, or could be constructed and operated to meet DOE's irradiation needs. Thus, this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]) evaluates the following alternatives:

- **Alternative 1**, resuming operation of the Fast Flux Test Facility (FFTF) at the Hanford Site (Hanford) in Washington State
- **Alternative 2**, using existing irradiation facilities (the Advanced Test Reactor [ATR] at Idaho National Engineering and Environmental Laboratory [INEEL], and the High Flux Isotope Reactor [HFIR] at Oak Ridge National Laboratory [ORNL], or a generic commercial light water reactor [CLWR])
- **Alternative 3**, constructing and operating one or two new accelerators at an existing DOE site
- **Alternative 4**, constructing and operating a new research reactor at an existing DOE site

This NI PEIS also evaluates a **No Action Alternative** in which the status quo would be maintained; that is, DOE's existing facilities would continue to meet their current mission requirements within their operating levels, but DOE would not enhance existing U.S. nuclear facility infrastructure or expand its current missions to accommodate new missions. This NI PEIS also includes an additional alternative that would permanently deactivate Hanford's FFTF without enhancing U.S. nuclear facility infrastructure to accommodate new or expanded missions. **Alternative 5**, permanently deactivate FFTF, although a component of all alternatives

except No Action and Alternative 1, is included as a stand-alone alternative in response to numerous public comments received during the scoping period.

This NI PEIS evaluates several options under each alternative. These options primarily involve DOE facilities that could be used for the fabrication, storage, and postirradiation processing of the targets necessary for the program missions. Among the facilities proposed are: (1) the Radiochemical Engineering Development Center (REDC) at ORNL, (2) the Fluorinel Dissolution Process Facility (FDPF) and/or Building CPP–651 (storage only) at INEEL, (3) the Fuels and Materials Examination Facility (FMEF) at Hanford, (4) the Radiochemical Processing Laboratory (RPL) and Building 306–E at Hanford, and (5) a new facility to be constructed and operated at an existing DOE site to support the one or two new accelerators or new research reactor alternatives. **Table 2–1** provides an overview of the alternatives and 26 specific options for this NI PEIS.

Sections 2.2 through 2.4 describe: (1) the operations needed to implement the program missions, (2) the candidate sites and facilities where the operations would take place, and (3) the transportation activities associated with the program missions. In describing the facilities, the sections refer to the specific alternatives for which the individual facilities are proposed. Detailed descriptions of alternatives are provided in Section 2.5. Alternatives considered and dismissed are discussed in Section 2.6. Section 2.7 summarizes the environmental impacts and implementation schedules and provides a comparison of mission effectiveness between alternatives, and Section 2.8 presents the preferred alternative.

2.2 DESCRIPTION OF OPERATIONS

2.2.1 Medical Isotopes Production

Production of medical and industrial isotopes involves: (1) fabricating specially designed targets at a target fabrication facility, (2) irradiating the targets in an irradiation facility to generate specific medical isotopes, and (3) processing the targets at a target fabrication facility to prepare the medical isotopes for shipment to customers.

2.2.1.1 Target Fabrication

Each medical isotope would be produced using a target that is enriched in the appropriate target material for neutron irradiation. With the exception of the radium-226 target and those that use recycled materials, all of the targets use nonradioactive materials. Appendix C lists the types and forms of the target material used to produce each medical isotope.

After irradiation of the original target, radioactive impurities may remain with the target material after the medical isotope product is removed. Because of these impurities, reuse of the material can in some cases create targets that are radioactive. As a result, fabrication of targets from recycled target materials would require special handling and shielding.

Before beginning fabrication of a target for production of a particular medical isotope, a significant quantity of the element that makes up the target would be required. For nonradioactive targets, this material typically would be acquired from ORNL, where enrichment processes are conducted to produce target material that is sufficiently pure to support the generation of medical isotopes. The target form may be a metal, metallic oxide, or other chemical compound suitable for high-temperature irradiation, depending on engineering considerations such as material heat transfer characteristics, melting points, and metallurgical properties. The nonradioactive target material would be transported by truck from ORNL to the target fabrication facility. One candidate radioactive target material, radium-226, would not be supplied by ORNL. However, radium-226 could be supplied by a variety of sources (no decision on a proposed supplier or suppliers has been made at

Table 2–1 NI PEIS Alternatives and Options

	Option Number	Irradiation Facility	Plutonium-238 Production Mission		Medical and Industrial Isotopes Production and Nuclear Research and Development Mission	
			Storage Facility	Target Fabrication and Processing Facility	Storage Facility	Target Fabrication and Processing Facility
No Action Alternative	1	–	–	–	–	–
	2	–	REDC	–	–	–
	3	–	CPP–651	–	–	–
	4	–	FMEF	–	–	–
Alternative 1: Restart FFTF	1	FFTF[a]	REDC	REDC	RPL/306–E	RPL/306–E
	2	FFTF[a]	FDPF/CPP–651	FDPF	RPL/306–E	RPL/306–E
	3	FFTF[a]	FMEF	FMEF	FMEF	FMEF
	4	FFTF[b]	REDC	REDC	RPL/306–E	RPL/306–E
	5	FFTF[b]	FDPF/CPP–651	FDPF	RPL/306–E	RPL/306–E
	6	FFTF[b]	FMEF	FMEF	FMEF	FMEF
Alternative 2: Use Only Existing Operational Facilities	1	ATR	REDC	REDC	–	–
	2	ATR	FDPF/CPP–651	FDPF	–	–
	3	ATR	FMEF	FMEF	–	–
	4	CLWR	REDC	REDC	–	–
	5	CLWR	FDPF/CPP–651	FDPF	–	–
	6	CLWR	FMEF	FMEF	–	–
	7	HFIR and ATR	REDC	REDC	–	–
	8	HFIR and ATR	FDPF/CPP–651	FDPF	–	–
	9	HFIR and ATR	FMEF	FMEF	–	–
Alternative 3: Construct New Accelerator(s)	1	New	REDC	REDC	New[c]	New[c]
	2	New	FDPF/CPP–651	FDPF	New[c]	New[c]
	3	New	FMEF	FMEF	New[c]	New[c]
Alternative 4: Construct New Research Reactor	1	New	REDC	REDC	New[c]	New[c]
	2	New	FDPF/CPP–651	FDPF	New[c]	New[c]
	3	New	FMEF	FMEF	New[c]	New[c]
Alternative 5: Permanently Deactivate FFTF (with No New Missions)	–	–	–	–	–	–

a. Hanford FFTF would operate with mixed oxide fuel for 21 years and highly enriched uranium fuel for 14 years.
b. Hanford FFTF would operate with mixed oxide fuel for 6 years and highly enriched fuel for 29 years.
c. The new facility would not be required if a DOE site is selected with available support capability and infrastructure.

Key: ATR, Advanced Test Reactor at INEEL; CLWR, commercial light water reactor; CPP–651, INEEL Building CPP–651 Storage Vault; FDPF, Fluorinel Dissolution Process Facility at INEEL; FFTF, Fast Flux Test Facility at Hanford; FMEF, Fuels and Materials Examination Facility at Hanford; HFIR, High Flux Isotope Reactor at ORNL; REDC, Radiochemical Engineering Development Center at ORNL; RPL/306–E, Radiochemical Processing Laboratory at Hanford, Building 306–E.

this time). Once materials for the targets arrive on the site, they would be stored at the target fabrication facility until needed for fabrication into medical isotope targets.

Solid targets would be fabricated in gloveboxes using a series of mechanical and thermal processes. For the solid targets based on a powder, it is unknown at this time whether the powder would be loose or would be pressed and sintered into pellets. If the latter method were preferred, separate equipment would be required to press and sinter each type of solid target material to reduce the risk of cross-contaminating other target materials.

If pellets were used, the first major step in their preparation would be powder conditioning and pressing, which includes weighing, blending, and pressing the powder and binder into slugs. The slugs would be granulated, blended with binder addition, and pressed into pellets. The pellets would be transferred to the sintering/debind station, weighed, and subjected to a series of thermal processes to debind and sinter the pellets. The sintered pellets would be subject to characterization to ensure that specifications were met.

Acceptable pellets would be transferred to the loading and welding station to be visually inspected before inclusion into a capsule or pin. For both powder or pellet target materials, capsules and pins would be cleaned before final closure. The capsules would be leak-tested and inspected before being cleared for use.

2.2.1.2 Target Irradiation

Production of medical or industrial isotopes is accomplished by irradiating target materials in the neutron flux of an irradiation facility such as a nuclear reactor. The desired isotopes are produced by neutron-induced reactions such as activation or transmutation.

Activation is the most common neutron-induced reaction and involves the capture of a neutron with the subsequent emission of a gamma ray. Because there is no change in the number of protons, the chemical identity of the target remains the same. For example, holmium-166 is produced by irradiating target material enriched in holmium-165 by activation.

Transmutation involves the capture of a neutron and the subsequent ejection of a proton or other particle that would change the chemical identity of the product. For example, phosphorus-32 is produced by irradiating a target material enriched in sulfur-32 by transmutation (proton ejection).

2.2.1.3 Postirradiation Target Processing

Processing of irradiated targets to recover medical- and industrial-grade isotopes can be broken down into distinct steps: (1) transport of irradiated targets to a chemical separation facility and receipt at that facility; (2) chemical processing of the targets (using hot cells, shielded gloveboxes and appropriate open-faced hoods); (3) waste handling; (4) analysis of the products; (5) recycling of some of the target materials; and (6) shipment of the isotope products to customers.

Each of the medical isotope products evaluated for production in this NI PEIS is unique. Some targets would produce an isotope of the same element and would not require separation. Some targets would produce the same element, but would require some processing to remove impurities. Other target materials would produce different elements and would require chemical separation of the target material, the desired isotope product, and unwanted impurities. Details on postirradiation processing of the targets for medical and industrial isotope production are provided in Appendix C.

2.2.2 Plutonium-238 Production

Production of plutonium-238 involves (1) storing neptunium-237, (2) fabricating neptunium-237 targets, (3) irradiating the targets in an irradiation facility, and (4) processing the targets to separate the plutonium-238 and prepare the product for shipment to Los Alamos National Laboratory (LANL) where it would be fabricated into heat sources for radioisotope power systems. As stated in Section 1.2.2, a plutonium-238 production rate of 2 to 5 kilograms (4.4 to 11 pounds) per year would be sufficient to meet the projected need based on NASA space exploration missions. Evaluations presented in this NI PEIS are based on a plutonium-238 production goal of 5 kilograms (11 pounds) per year to bound the environmental impacts of the proposed plutonium-238 production mission.

2.2.2.1 Target Fabrication

The facility designated to fabricate neptunium-237 target elements for plutonium-238 production would receive the neptunium-237 oxide from the Savannah River Site (SRS) and would dissolve it in an acid solution prior to removal of protactinium-233, a decay daughter of neptunium-237. Protactinium-233 reaches 90 percent of its equilibrium activity approximately 90 days after purification and contributes significantly to radiation doses in the target fabrication line. The best approach for the removal of the protactinium-233, and possibly the easiest to implement, is to pass the neptunium solution through a column containing silica gel adsorbent (Wham et al. 1998). After protactinium-233 removal, the purified neptunium solution can be transferred to a target-fabrication glovebox line and reconversion of the neptunium to the oxide form can be initiated. The desired form of the oxide (microspheres) is obtained by loading the neptunium on a cation-exchange resin of the selected particle size range, washing the loaded resin, and using heated air to oxidize the resin and form the neptunium dioxide microspheres.

Current target designs for the ATR and HFIR reactors consist of neptunium dioxide blended with aluminum powder, pressed into a target core, and clad with aluminum. This type of target has been used in nearly all of the DOE production and research reactors (except for fast neutron flux reactors, e.g., FFTF) to produce isotopes in general and plutonium-238 specifically.

Three different techniques can be employed to fabricate such targets:

1. The neptunium dioxide and aluminum powders are blended and pressed into pellets. The pellets are then loaded into aluminum target tubes, which are seal-welded and hydrostatically compressed.

2. The neptunium dioxide and aluminum powders are blended and pressed into compacts. The compacts are then roll-milled between aluminum cladding, after which the aluminum-clad neptunium dioxide is seal-welded.

3. The neptunium dioxide and aluminum powders are blended and pressed into billets and assembled into welded and evacuated aluminum containers. The billets and containers are then coextruded to produce target tubes.

All three techniques have advantages and disadvantages. The coextrusion technique has been used successfully by SRS in its plutonium-238 program and other special isotope programs. Demonstrations of the fabrication techniques would be required to determine which techniques are best for the proposed irradiation facilities.

The target blanket for the high-energy accelerator consists of neptunium dioxide blended with aluminum powder, pressed into the required configuration, and clad with aluminum.

The CLWR targets would have stainless steel or Zircaloy cladding because of the higher operating temperatures. Targets for the new research reactor would also have stainless steel (Incoloy-800) cladding for material compability. The postirradiation processing of these targets would be different from the postirradiation processing of the aluminum-clad targets.

The targets used for production of plutonium-238 in FFTF would be similar to target concepts developed in studies performed in 1992 and 1993. The proposed target assembly would consist of 19 large-diameter pins that combine alternating thin pellets or wafers of neptunium dioxide and yttrium hydride moderator within a ferritic alloy steel cladding. The steel cladding is required for compatibility with the reactor sodium.

The conversion efficiency of neptunium-237 to plutonium-238 during target irradiation at FFTF, ATR, HFIR, or CLWR would be approximately 10 to 15 percent. Approximately 50 kilograms (110 pounds) per year of neptunium-237 would be fabricated into targets to meet the plutonium-238 production goal of 5 kilograms (11 pounds) per year. Following processing of the irradiated targets, approximately 40 kilograms (88 pounds) per year of the unconverted neptunium-237 would be stored in liquid form and recycled for the fabrication of new targets. The remainder would be process waste. Based on the current preconceptual designs, the conversion efficiency of neptunium-237 to plutonium-238 during target irradiation at the new high-energy accelerator and new research reactor would be significantly lower, approximately 2.8 percent and 1.4 percent respectively. To meet the plutonium-238 production goal of 5 kilograms (11 pounds) per year, approximately 180 kilograms (396 pounds) would be fabricated into targets annually for irradiation in the high-energy accelerator, and 380 kilograms (836 pounds) would be fabricated into targets for irradiation in the new research reactor. The neptunium-237 target fabrication requirements for the new research reactor could be reduced by a factor of 4 to 8 by refining the current target and reactor preconceptual designs presented in Appendix E.

Neptunium must be treated like uranium-235 under DOE safeguards and security requirements, which are based on the mass of neptunium and the attractiveness level of the physical and chemical form of the neptunium. This would require special security clearances for persons with access to the neptunium, as well as ongoing security reviews and audits during the time of possession of the neptunium (McCallum 1999). Safe, secure trailer/SafeGuards Transports (SST/SGTs) would be required to transport any significant quantity of neptunium. The neptunium containers would be stored in specially designed storage vaults to provide a secure, safe storage for the materials. DOE guidelines concerning safeguards and security would be followed whenever materials were being stored or processed.

2.2.2.2 Target Irradiation

Irradiation of neptunium-237 targets in neutron flux produces plutonium-238 according to the following equations:

$$_{93}Np^{237} + {}_{0}n^{1} \rightarrow {}_{93}Np^{238}$$

$$_{93}Np^{238} \rightarrow {}_{-1}\beta^{0} + {}_{94}Pu^{238}$$

The neptunium-237 target nuclide absorbs a neutron to become neptunium-238 (first equation), which in turn decays with a half-life of 2.1 days and emits a beta particle (or electron) to form plutonium-238 (second equation).

Irradiation of the neptunium-237 targets generates fission products in the targets. The irradiated targets would be cooled for at least 120 days to allow time for the decay of short-lived fission products (e.g., iodine-131). Following the cool-down period, the irradiated targets would be loaded into a shielded cask for transport to

the chemical processing facility. They would then be ready for chemical processing to separate the plutonium-238 content and unconverted neptunium-237 from radioactive waste products.

2.2.2.3 Postirradiation Target Processing

The flowsheet for processing irradiated neptunium-237 targets, recovering the unconverted neptunium-237, refabricating target elements, separating the plutonium-238 product, and shipping the plutonium-238 for fabrication into heat sources for radioisotope power systems is shown in **Figure 2–1**. Processing the irradiated neptunium-237 targets would be conducted inside heavily shielded hot cells to protect workers from high radiation doses. Hot cells are specially designed shielded vaults or areas used for the remote handling and manipulation of some radioactive materials. Certain chemical processing steps would be required to recover the plutonium-238 as product and to recover the neptunium-237 for recycle. At ORNL and Hanford this process would be accomplished in two steps:

- For targets irradiated in ATR, HFIR, or the accelerator, caustic-nitrate solution would be used to dissolve the cladding, thereby separating the bulk of the aluminum and caustic-soluble fission products from the actinide products, including the neptunium and plutonium. For targets irradiated in FFTF, the new research reactor, or a CLWR, the first step in the target processing would be to chop the targets into small pieces for dissolution in acid (in the next step).

- Next, acid would be used to dissolve the actinide products and remaining fission products to prepare the feed for the first mainline separation process. The feed would be filtered prior to pH (acidity/alkalinity) adjustment to remove any solids that could complicate the solvent extraction process. For the FFTF, new research reactor, or CLWR targets, the undissolved cladding would be discarded as waste.

Dissolution of the irradiated targets at INEEL would be accomplished using a one-step target dissolution process in a nitric acid-fluoroboric acid solution instead of the two-step process for the ATR and HFIR targets that would be used at ORNL and Hanford. It would still be necessary to shear or chop the FFTF or CLWR targets at INEEL before the acid leach process.

Subsequent to target dissolution, a tributyl phosphate-based solvent extraction process would be used for three cycles of purification. The first cycle would decontaminate the neptunium and plutonium products from fission product wastes. The second cycle solvent extraction process would separate the neptunium from the plutonium, and the third-cycle process would remove trace plutonium from the neptunium product. The plutonium product would undergo further purification using anion exchange if the product did not meet specification.

Chemical conversion of the plutonium to an oxide would start with its precipitation from solution as an oxalate. The precipitate would be filtered and calcined (heated at high temperature) to an oxide product. The plutonium dioxide product would be further treated in an oxygen-exchange process to exchange its oxygen-17 and oxygen-18 components with oxygen-16, thereby reducing the neutron emission rate. The resulting oxide product would be packaged and shipped to LANL for fabrication into heat sources for radioisotope power systems.

The purified neptunium nitrate from the third-cycle solvent extraction process would be stored as a solution. A small quantity of neptunium oxide (6 to 8 kilograms [12 to 16 pounds]) would be removed from storage, dissolved, and purified to replace the neptunium-237 that was converted to plutonium-238. This material would be added to the neptunium solution recovered during postirradiation target processing, loaded onto a cation-exchange resin, and then calcined to produce oxide microspheres for re-use in target assemblies for

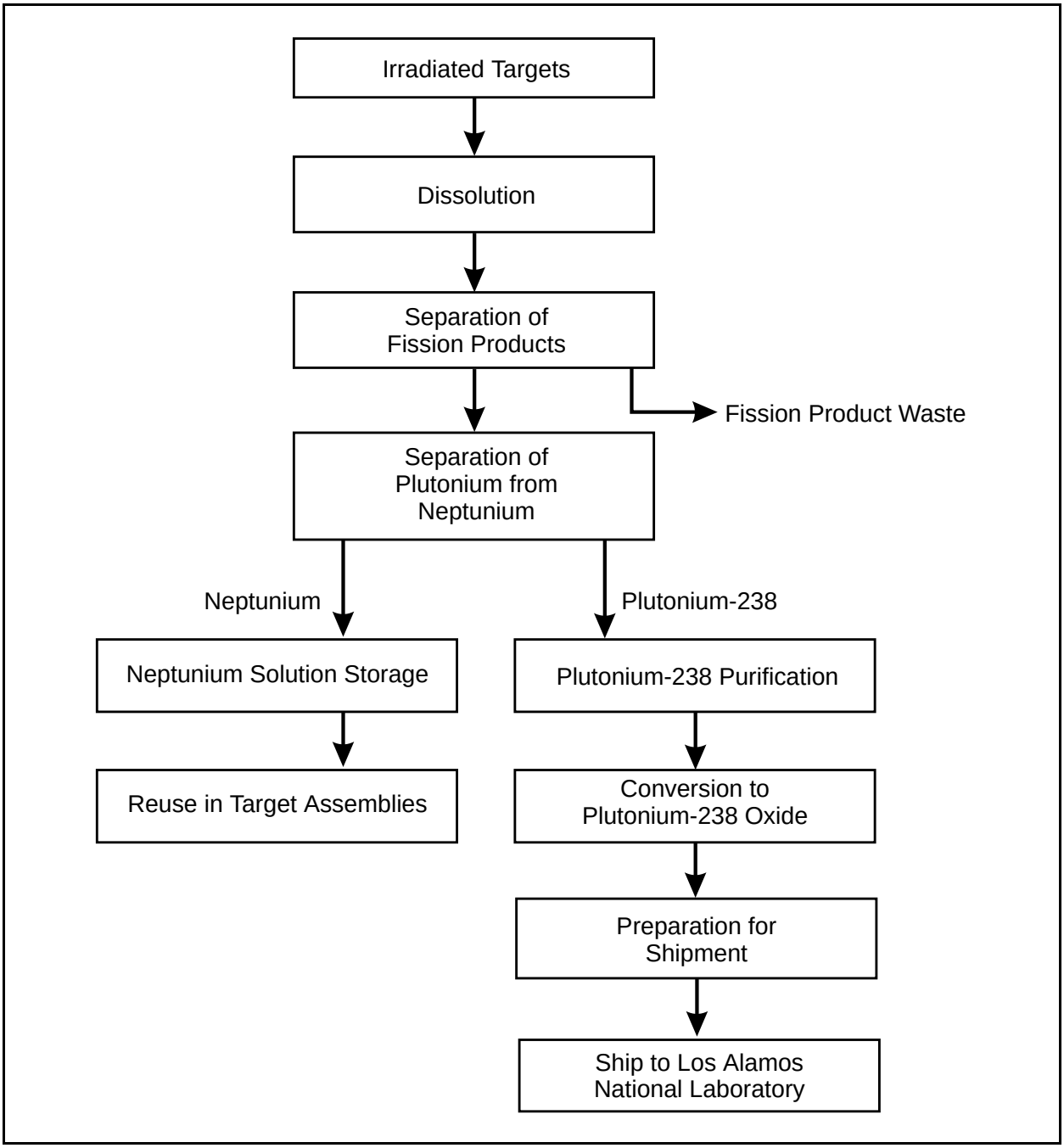


Figure 2–1 Chemical Processing Flowsheet for Irradiated Neptunium-237 Target Processing

irradiation. Waste-handling equipment would be used to minimize the activity in low-level radioactive liquid waste and to stabilize solid wastes into an acceptable waste form. The equipment would be included in the hot cells used for the chemical processing of irradiated targets.

2.2.3 Civilian Nuclear Energy Research and Development

As discussed in Section 1.2.3, civilian nuclear energy research and development initiatives requiring an enhanced DOE facility infrastructure fall into three basic categories: materials research, nuclear fuel research, and advanced reactor development.

2.3 DESCRIPTION OF FACILITIES

This section describes the facilities proposed by DOE for the production of medical and industrial isotopes, plutonium-238, and civilian nuclear energy research and development. Because the programmatic alternatives are structured around the use of the candidate irradiation facilities, they are discussed first in Section 2.3.1, followed by a discussion of the proposed target fabrication and processing facilities in Section 2.3.2. The proposed irradiation facility alternatives are (1) FFTF at Hanford, (2) ATR at INEEL, (3) HFIR at ORNL, (4) a generic CLWR, (5) one or two new accelerators at an existing DOE site, or (6) a new research reactor at an existing DOE site.

2.3.1 Target Irradiation Facilities

2.3.1.1 Fast Flux Test Facility

FFTF is a 400-megawatt thermal, liquid-cooled (sodium) nuclear test reactor (**Figure 2–2**) owned by DOE and located at Hanford in southeastern Washington State near Richland, Washington. Figure 3–12 presents a map of Hanford that depicts the location of FFTF. In May 1972, the U.S. Atomic Energy Commission published an environmental statement for FFTF (AEC 1972). That document provided information on the potential environmental impacts associated with the construction and operation of FFTF. In the late 1970s, the Safety Analysis Report prepared for FFTF was reviewed by the U.S. Nuclear Regulatory Commission (NRC) and the Advisory Committee for Reactor Safeguards. Comments from both organizations were addressed in the FFTF Final Safety Analysis Report. The construction of FFTF was completed in 1978.

Following extensive testing, FFTF was started up in April 1982. During its operation, FFTF successfully tested advanced nuclear fuels, materials, components, operating protocols, and reactor safety designs. It also produced a wide variety of medical isotopes and made tritium for the U.S. fusion research program.

Figure 2–2 Fast Flux Test Facility

FFTF was originally designed and operated as a science test bed for U.S. liquid metal fast reactor programs. These programs, which were canceled in 1993, were key elements both in closed fuel cycle and actinide waste disposition technology development. In December 1993, DOE decided not to operate FFTF because of a lack of

economically viable missions at that time. In accordance with the National Environmental Policy Act (NEPA), DOE published an environmental assessment and Finding of No Significant Impact for the shutdown and deactivation of FFTF in May 1995 (DOE 1995a). The environmental assessment contained an evaluation of the environmental impacts associated with the actions necessary to place FFTF in a radiologically and industrially safe shutdown condition suitable for long-term surveillance and maintenance before final decontamination and decommissioning.

The FFTF complex includes the reactor, as well as equipment and structures for heat removal, containment, reactor safety and shutdown systems core component handling and examination, fuel offloading and storage, utilities, and other essential services. There are 100 systems supporting various functions of FFTF during operation. The central structure of FFTF is the reactor containment building, an all-welded cylindrical steel structure 41 meters (135 feet) in diameter and 57 meters (187 feet) high. The array of buildings and equipment that surround the containment building and comprise the FFTF complex is shown in **Figure 2–3.** The reactor is below grade in a shielded cell in the center of the containment structure. Heat is removed from the reactor by circulating liquid sodium under low pressure through three separate closed primary piping loops, which include pumps, piping, and intermediate heat exchangers. These loops are located within inerted cells (cells filled with inert gases) within the containment structure. **Figure 2–4** is a cutaway of the containment building showing the location of the reactor, primary pumps, and intermediate heat exchangers. Three secondary sodium loops transport reactor heat from the intermediate heat exchangers to the air-cooled tubes of the dump heat exchangers. From there, the heat dissipates into the atmosphere through the forced draft dump heat exchanger. Commercial nuclear power reactors use reactor heat to create steam, which turns a turbine to produce electricity. FFTF, however, does not generate electricity.

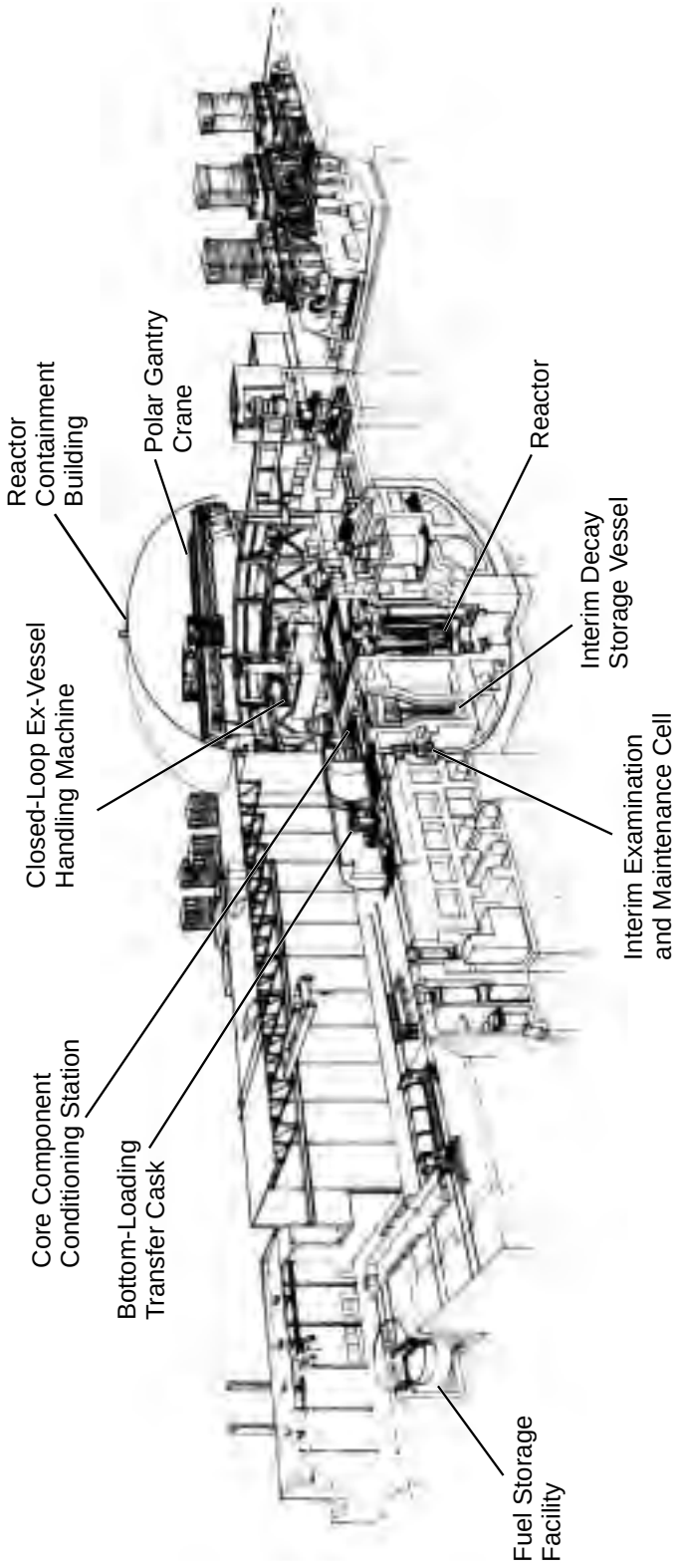
FFTF has demonstrated its capability to function as a nuclear science and irradiation services user facility. It has five distinct features: size, flux, test evaluation and irradiation capabilities, fuel type, and coolant type. In combination, these features provide a multipurpose facility suitable for medical and industrial isotopes production, plutonium-238 production, and civilian nuclear energy research and development purposes. Although FFTF was used primarily to evaluate reactor fuels and different fuel assembly materials during its 10 years of operation, the reactor facility has also supported large and varied test programs for industry, nuclear energy (domestic and international), medical isotope applications and research, space nuclear power, and fusion research programs. A more detailed description of FFTF and its capabilities is included in Appendix D.

2.3.1.1.1 Maintenance of FFTF in Standby

FFTF is currently defueled and is being maintained in a safe standby condition. FFTF would be maintained in the standby condition under the No Action Alternative. Seventy-seven of the 100 systems are operational; the other 23 are in a recoverable standby state. System integrity and configuration control are being maintained. The Main Heat Transport System is being operated at approximately 200 °C (400 °F) to keep the sodium coolant in the reactor liquefied and circulating. Essential systems, staffing, and support services are being maintained in a manner that would support potential restart.

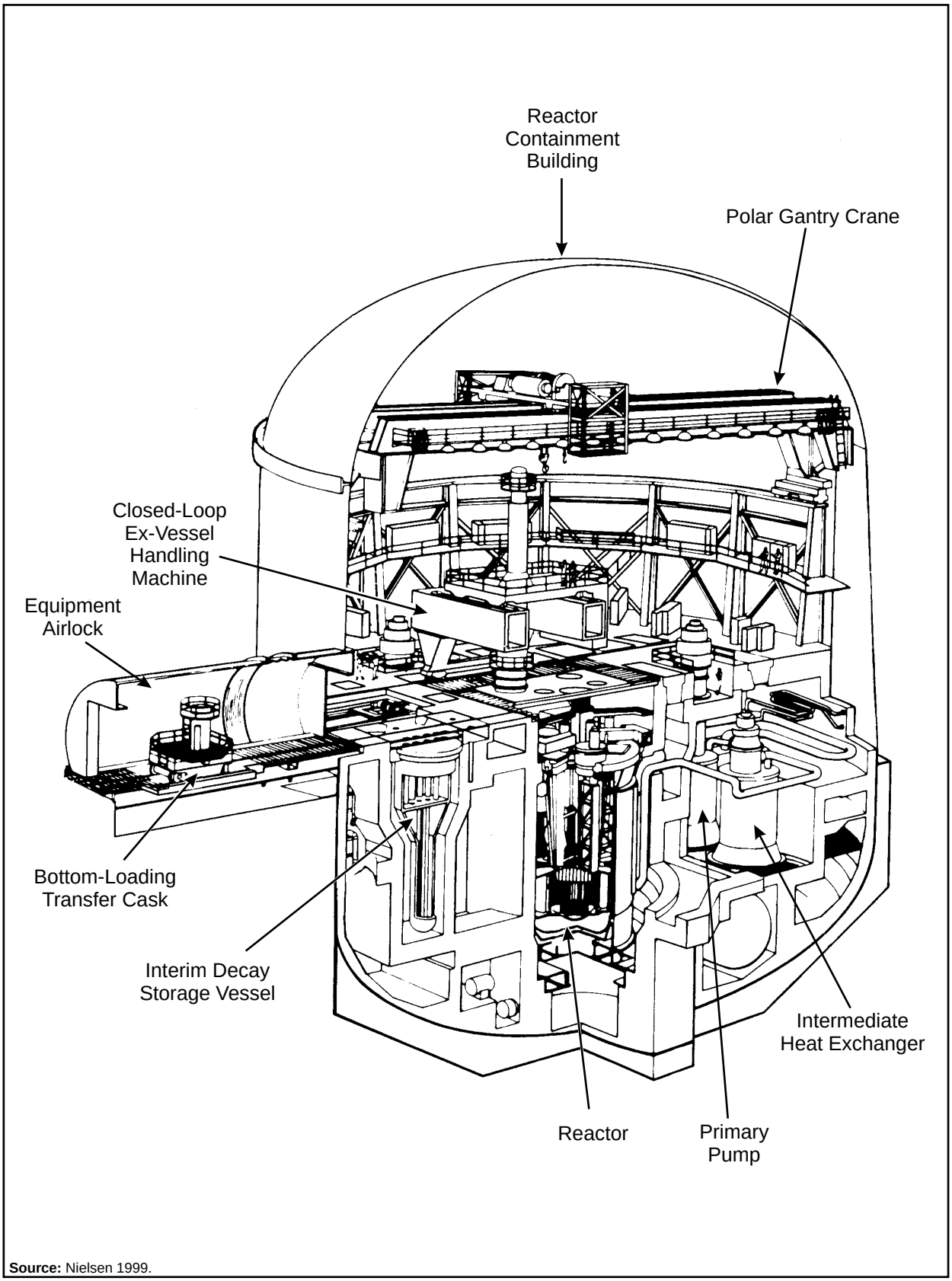
2.3.1.1.2 FFTF Restart and Operation

FFTF is proposed to be restarted and operated under Alternative 1, FFTF Restart. If a decision were made to restart FFTF, several equipment upgrades are planned to return systems to operation, improve reliability, conform to current standards, improve efficiency, and minimize waste. Most of the required modifications would consist of either mechanical equipment upgrades or replacement of outdated control and computer



Source: Nielsen 1999.

Figure 2–3 FFTF Complex



Source: Nielsen 1999.

Figure 2–4 Cutaway of the Reactor Containment Building

systems and would have minimal environmental consequences. The following is a brief list of the planned modifications if FFTF would be restarted (Battelle 1999):

- Upgrade of plant protection system (scram breakers, power supplies, and signal conditioners)
- Replacement of zero-time-outage motor generator sets with solid-state electronic units
- Upgrades of plant data systems computers
- Upgrade of conductivity metering system on three cooling towers and replacement of electronic sensors and controls
- Installation of two new electrical distribution transformers to replace the polychlorinated biphenyl-filled units that were removed during standby operations
- Establishment of a program to assess and replace elastomer seals during the startup period to take advantage of advancements in seal technology
- Upgrades of the plant simulator (A program to upgrade the existing simulator to reach commercial simulator standards was in progress, but was discontinued when FFTF was placed in standby.)

2.3.1.1.3 FFTF Fuel Use Option

This NI PEIS postulates that FFTF would operate at a nominal power level of 100 megawatts, one quarter of the reactor design power level, to meet the irradiation requirements of the proposed missions. Periodic increases in power level between 100 and 400 megawatts may be required to support civilian nuclear energy research and development activities. Operating FFTF at a nominal 100-megawatt power level would extend the reactor life and significantly reduce the generation rate of spent fuel. FFTF is currently designed to operate using mixed oxide fuel (i.e., plutonium-uranium); however, it can also be operated using highly enriched uranium fuel. FFTF has an onsite supply of mixed oxide fuel for approximately 6 years of operation at the 100-megawatt level proposed for the mission. When this onsite fuel is depleted, FFTF may continue to use mixed oxide fuel or may switch to a reactor core of highly enriched uranium fuel. DOE believes that an additional 15-year supply of mixed oxide fuel would be available from Germany under favorable economic terms, (i.e., no charge for the fuel). The fuel would be reconfigured into assemblies suitable for irradiation at FFTF before shipment to the United States. That is why this NI PEIS evaluates the operation of FFTF for two reactor core configurations for the 35-year evaluation period of operation common to all alternatives: (1) operation with a mixed oxide core for approximately 21 years followed by 14 years of operation with a highly enriched uranium core, and (2) operation with a mixed oxide core for approximately 6 years followed by 29 years of operation with a highly enriched uranium core.

In this NI PEIS, DOE has not evaluated the possibility of using low-enriched uranium fuel for operation of the FFTF because it makes programmatic and economic sense to use available mixed oxide fuel supplies before using uranium. U.S. nonproliferation policy (U.S. House of Representatives 1992 [Schumer Amendment]), strongly discourages the use of highly enriched uranium fuel in civilian research and test reactors. The Reduced Enrichment for Research and Test Reactors Program implements this policy by developing technical means to reduce and eventually eliminate the use of highly enriched uranium in research and test reactors throughout the world and in the United States without decreasing their safety or significantly affecting their performance and operating costs.

To be in compliance with these policy directives, the most appropriate fuel supply for FFTF in the out-years (beyond current Hanford mixed oxide and possible German SNR–300 mixed oxide supplies) must be determined by a technical study with the preferred fuel source being low-enriched uranium. Highly enriched uranium fuel should only be considered if low-enriched uranium is not technically feasible, or if there are significant impacts on safety, performance, or cost associated with using fuels other than highly enriched uranium.

In the event that a decision is made to restart the reactor, and to support these policy directives, DOE's Office of Nonproliferation and National Security would undertake a study to consider the technical feasibility of using low-enriched uranium fuel (under the Reduced Enrichment for Research and Test Reactors Program) for FFTF. If low-enriched uranium fuel is found infeasible, DOE would subsequently procure highly enriched uranium fuel in a manner consistent with U.S. nonproliferation policy. This study would be conducted, decisions would be implemented, and fuel would be made available during the time period between a Record of Decision indicating an FFTF restart and prior to the end of available Hanford mixed oxide and possible SNR–300 mixed oxide fuel supplies.

For the purposes of presenting a bounding analysis in this NI PEIS, DOE has analyzed the impacts of using highly enriched uranium fuel in FFTF after the available mixed oxide fuel supplies have been expended. These impacts would bound those of using a low-enriched uranium fuel form.

2.3.1.1.4 FFTF Irradiation Operations

There are eight locations available in the reactor core that are termed Open Test Assembly positions. These positions are located under spool pieces in the reactor head and allow the installation of 38-foot-long assemblies that extend from the reactor head down to the reactor core. These eight locations are unique from the rest of the reactor in that they allow direct contact instrumentation for remote monitoring during reactor operation. Within the 82 active core locations, there are up to 20 or more additional locations that could contain a standard length (3.6-meter or 12-foot) test assembly. These locations also have specific online outlet temperature and flow measurements from installed plant instrumentation. In addition to the test locations within the active fueled region of the core, there are 108 locations available in the surrounding reflector region where other tests could be inserted. These three basic testing configurations enable irradiation of large and/or very diverse quantities. The target designs vary according to the test requirements and the location of the test within the reactor.

To fulfill the mission, the FFTF core would be modified to include an array of target assemblies and rapid radioisotope retrieval systems capable of producing a number of long- and short-lived isotopes for medical and industrial applications and plutonium-238 for space power applications. In addition, reactor space would be provided for research and development test articles.

Fifteen plutonium-238 production targets would be included in the reflector region with an annual production rate of 5 kilograms. The residence time for these targets would be three 100-day cycles; five assemblies would be harvested at the end of each cycle.

Long-Term Irradiation Vehicles would be used to irradiate targets to produce long-lived isotopes. The Long-Term Irradiation Vehicles would be installed in the reactor during normal refueling operations and would be handled using standard FFTF component handling equipment. The Long-Term Irradiation Vehicle would consist of a bundle of target pins installed inside a nozzle, duct, and handling socket assembly similar in appearance to an FFTF 3.6-meter-long (12-foot-long) fuel assembly. Depending on the isotopes to be produced, the pin bundle could contain moderator pins and neutron shield pins. A design that would allow re-use of the long-term irradiation assembly nozzle, duct, and handling socket hardware would be considered

during the design process to reduce both costs and waste generation. It is assumed that 12 Long-Term Irradiation Vehicle assemblies for the production of long-lived medical and industrial isotopes would be installed. A detailed description of the Long-Term Irradiation Vehicles and their proposed use is included in Appendix D.

Rapid radioisotope retrieval systems would be installed in selected Open Test Assembly positions for the production of short-lived isotopes. These systems would extend from above the spool pieces in the reactor head down into and slightly below the active core region and would allow target materials to be inserted and withdrawn from the reactor core region while the reactor is operating. Systems for routinely inserting and removing irradiation targets, nuclear instrumentation, and research hardware have been in use for years at various research reactors throughout the world. Most of these systems use either a pneumatic rabbit-type system or a mechanical cable-type system for insertion and retrieval. There would be a maximum of eight systems in the core. One of the systems would be configured as a gas target to produce iodine-125 from xenon-124. The other seven systems would be used to produce solid short-lived medical isotopes. A detailed description of the rapid radioisotope retrieval systems and their proposed use is included in Appendix D.

FFTF would operate at a nominal power level of 100 megawatts. However, the accident analyses provided in this NI PEIS are based on the FFTF design power level of 400 megawatts and provide conservative estimates of operation at 400 megawatts-thermal and lower power levels.

Testing programs would be conducted for new materials and target designs to be irradiated in the reactor. A discussion of the types of testing that would be associated with the medical isotope and plutonium-238 production missions is included in Appendix D.

2.3.1.1.5 FFTF Deactivation

FFTF would be permanently deactivated in Alternative 2 (Use Only Existing Operational Facilities), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), and Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). This would require placement of FFTF in a radiologically and industrially safe shutdown condition that is suitable for a long-term surveillance and maintenance phase prior to final decontamination and decommissioning. An *Environmental Assessment, Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington*, issued by DOE in 1995, addressed the environmental impacts associated with permanently deactivating FFTF (DOE 1995a).

If a decision were made to proceed with permanent deactivation of FFTF, the molten sodium (radioactive) would be removed from the reactor systems and transferred to an existing sodium storage facility that was specially constructed for this purpose. The sodium would be drained by pressure transfer to the maximum practical extent into tanks located in the sodium storage facility. Residual sodium would be accommodated to a stabilized condition so that long-term monitoring and surveillance could be conducted in a safe and environmentally sound manner. The current concept for accommodating residuals would be to maintain an inert gas atmosphere that prevents any chemical reactions during long-term surveillance and maintenance.

2.3.1.2 Advanced Test Reactor

ATR is a light-water-cooled and moderated reactor with a design thermal power of 250 megawatts that is owned by DOE and is in the Test Reactor Area in the southwest portion of INEEL. Figure 3–6 presents a map of INEEL that depicts ATR's location.

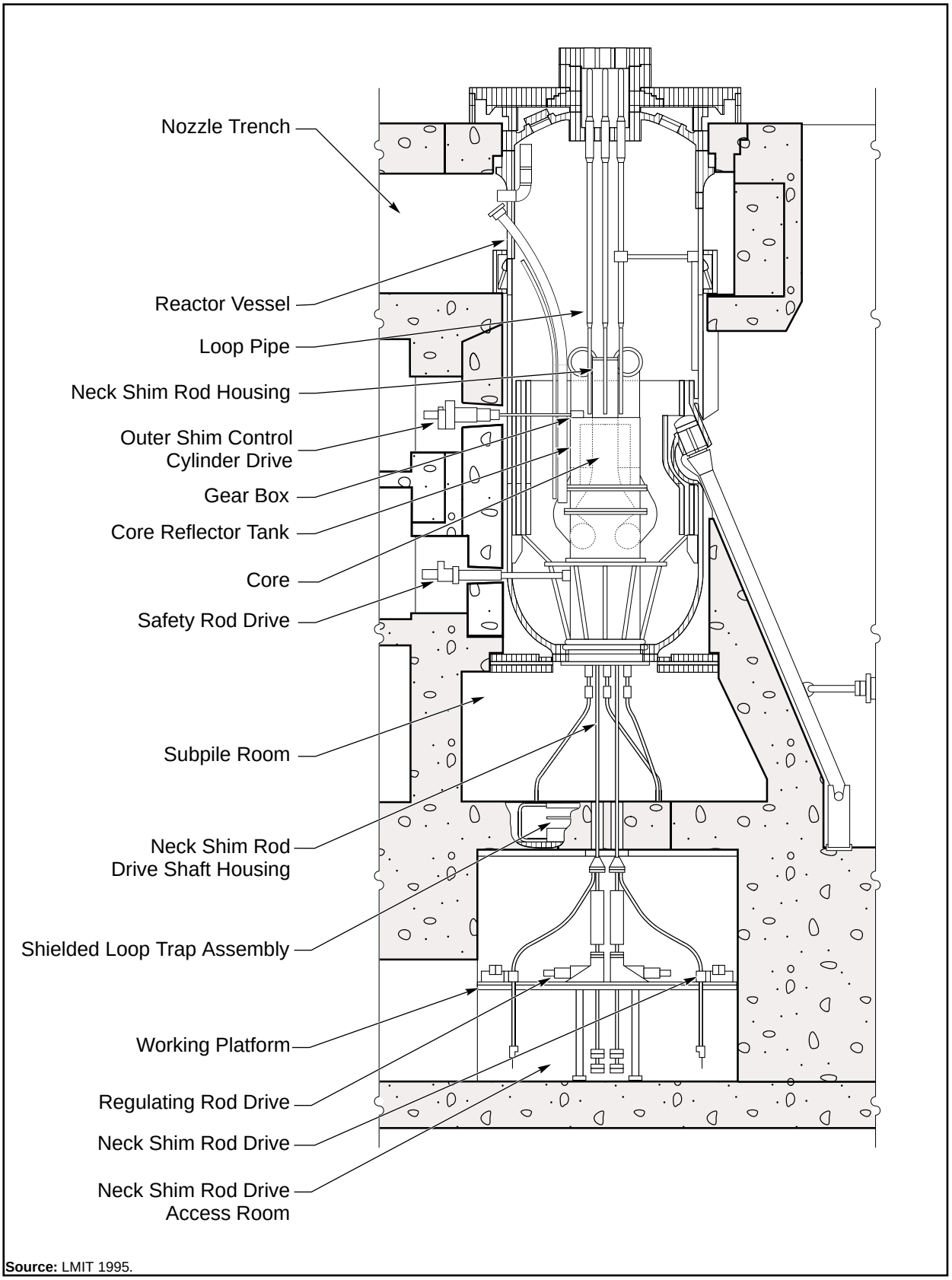
ATR would continue to operate and meet its current mission requirements, including naval reactor research and development, medical and industrial isotope production, and civilian nuclear energy research and

development activities, at its current operating levels under the No Action Alternative, Alternative 1 (FFTF Restart), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities) when it is not providing irradiation services in support of the plutonium-238 production mission. When ATR is supporting the plutonium-238 production mission, it would fully support its primary mission—naval reactor research and development; however, it would support the medical and industrial isotope production and civilian nuclear energy research and development activities to the extent possible within its current reactor operating levels. Consideration must be given to the need to maintain appropriate levels of neutron flux to support ATR's primary mission. Neutron flux levels can be impacted by the placement of targets, such as neptunium-237 targets for the production of plutonium-238, in the reactor core. The production planning assumption for ATR is from 3 kilograms (6.6 pounds) of plutonium-238 per year (if used in conjunction with HFIR) to 5 kilograms (11 pounds) of plutonium-238 per year (if ATR were used alone). Thus, ATR alone could meet the program goal of up to 5 kilograms (11 pounds) per year and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission.

Special features of ATR include high neutron flux levels (ranging from 1×10^{15} neutrons per square centimeter per second in the flux traps to 1×10^{13} neutrons per square centimeter per second in the outer reflector positions) and the ability to vary power to fit different experiment needs in different test positions. The primary user of ATR is the U.S. Naval Nuclear Propulsion Program. A variety of other users include other foreign and domestic government programs, a commercial isotope production company, industrial customers, and research and development interests. This facility description is based on information provided in the *Advanced Test Reactor Upgraded Final Safety Analysis Report* (LMIT 1997) and *Capabilities of the Test Reactor Area Featuring the Advanced Test Reactor* (LMIT 1995). A number of support facilities are important to the operation of ATR (LMIT 1997). Among these are the Advanced Test Reactor Critical Facility, which is used to baseline experiment impacts to the ATR flux profile, and the Nuclear Materials Inspection and Storage Facility, which is used to receive, store, and inspect reactor fuel prior to its placement in ATR (INEEL 1999, 2000; LMIT 1995).

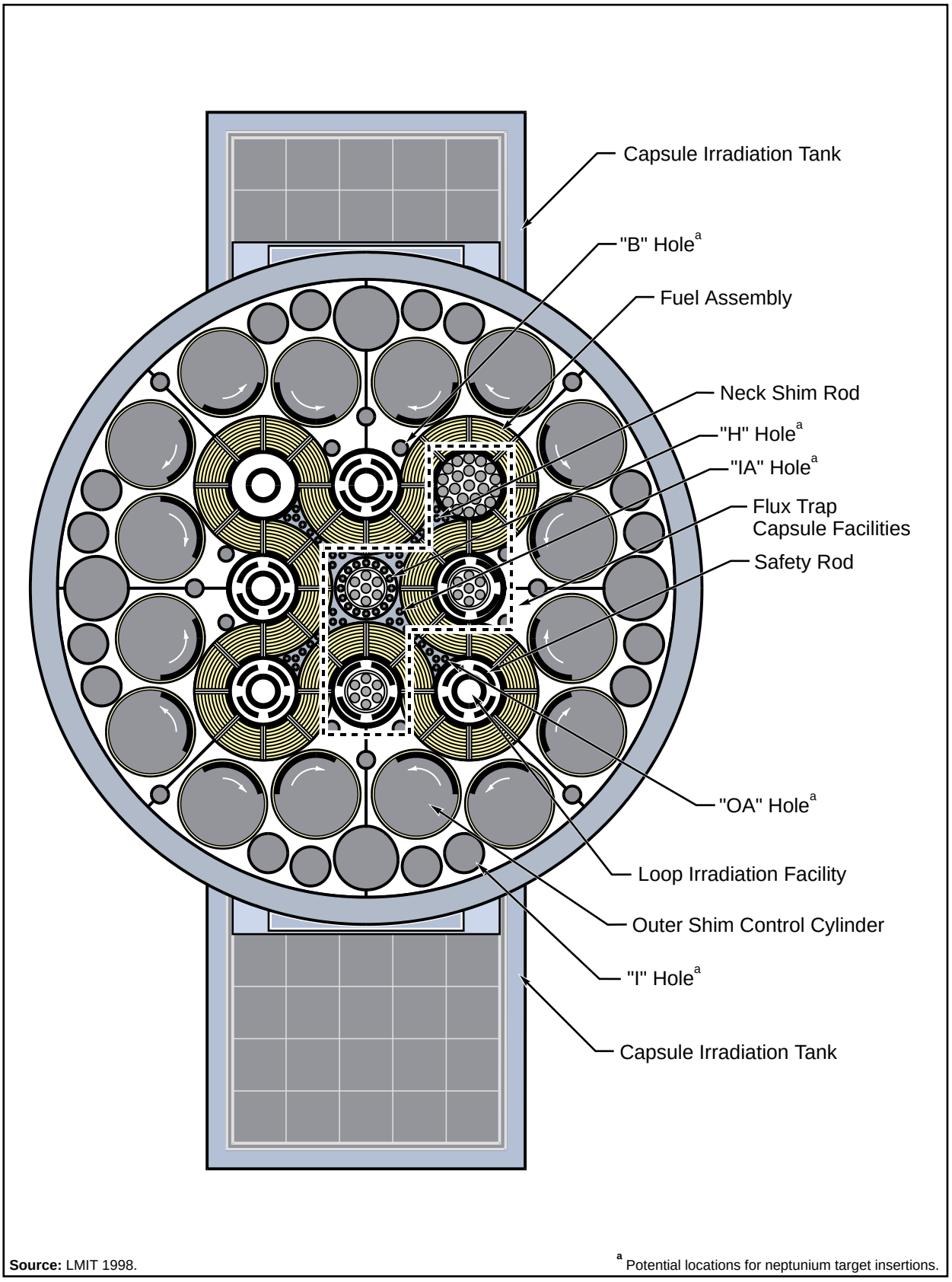
The reactor, its primary coolant system, control room, and much of its auxiliary and experimental support equipment are in Test Reactor Area Building 670. ATR began operation in 1967 and is expected to continue operating for several decades. The reactor vessel is constructed entirely of stainless steel, and the core internals are replaced every 7 to 9 years. The most recent changeout was completed in 1994 (LMIT 1995). Buildings and structures in other parts of the Test Reactor Area provide additional support functions.

ATR is currently operating at approximately 140 megawatts or less. ATR operates with highly enriched uranium fuel. Typical operating cycles are 42 days or 49 days at power followed by a 7-day outage for refueling and changeout of experiments and isotope production targets. The core is 1.2 meters (4 feet) high and is surrounded by a 1.3-meter-diameter (4.25-foot-diameter) beryllium reflector. Beryllium is an excellent neutron reflector and is used to enhance the neutron flux essential to a test reactor. The location of the core in the ATR vessel is shown in **Figure 2–5**. ATR has nine flux traps in its core and achieves a close integration of flux traps and fuel by means of a serpentine fuel arrangement (**Figure 2–6**). When viewed from above, the ATR fuel region resembles a four-leaf clover. The four flux traps positioned within the four lobes of the reactor core are almost entirely surrounded by fuel, as is the center position. Four other flux trap positions between the lobes of the core have fuel on three sides. ATR's unique control device design permits large power shifts among the nine flux traps. Testing can be performed in test loops installed in some flux traps with individual flow and temperature control or in reflector irradiation positions with primary fluid as coolant. The curved fuel arrangement brings the fuel closer on all sides of the test loops than is possible in a rectangular grid.



Source: LMIT 1995.

Figure 2–5 Vertical Cross Section of the ATR Vessel



Source: LMIT 1998.

[a] Potential locations for neptunium target insertions.

Figure 2–6 Plan View (Cross Section) of ATR

Five out of the nine flux traps are configured with pressurized-water loops that allow for individual temperature, pressure, flow, and chemistry controls. The five test loops are used by the Naval Reactors program. Of the remaining four flux traps, one is dedicated to the Naval Reactors program, one is used for isotope production, one is used for low-specific-activity cobalt production, and the fourth has recently had the Irradiation Test Vehicle installed. The Irradiation Test Vehicle can be described as three small pressurized-gas test loops. The use of one of these three test loops was recently purchased by a British corporation; negotiations for use of the other two are currently under way.

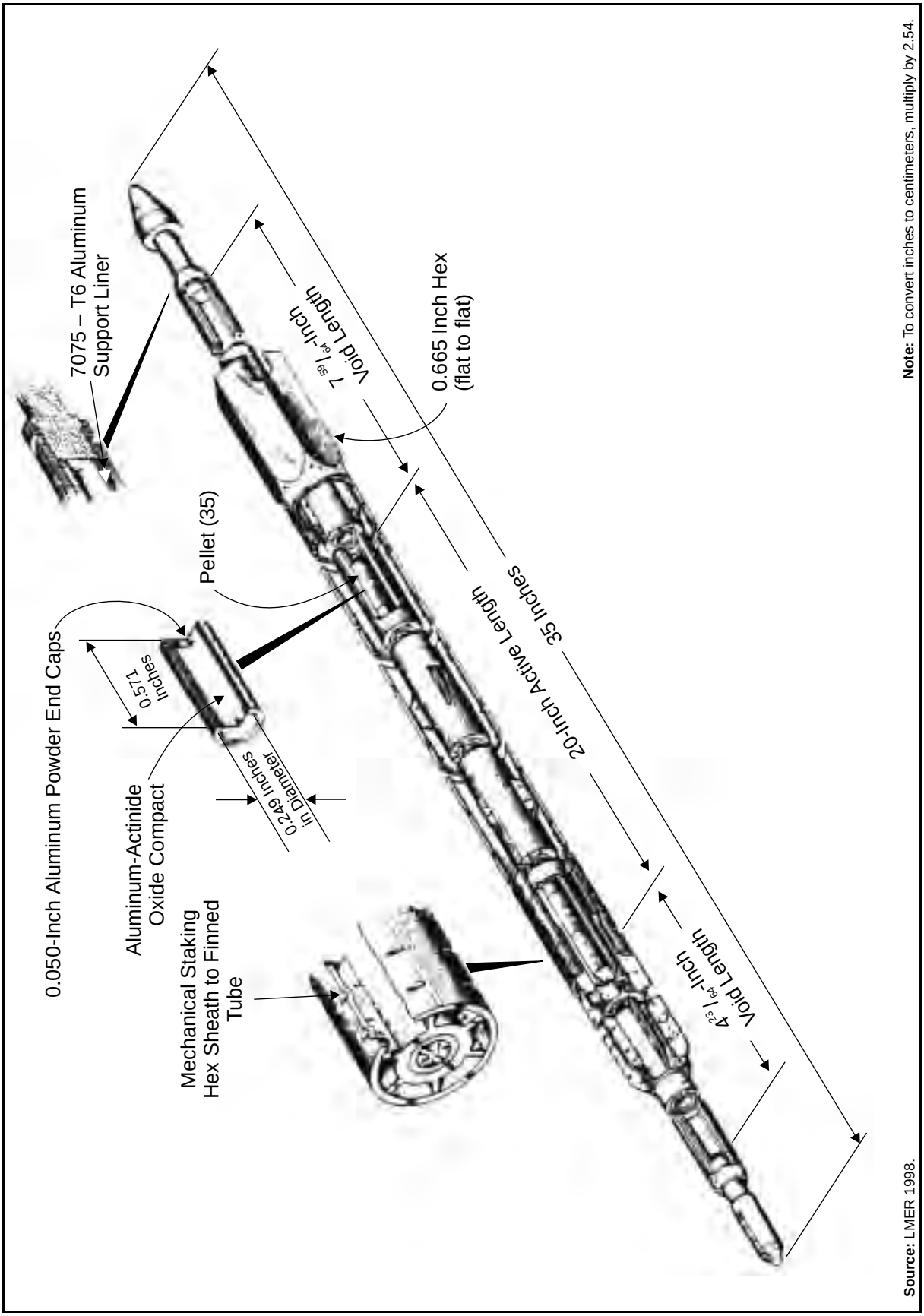
In addition to the primary flux trap irradiation positions, there are some 70 irradiation positions in the beryllium reflector (and aluminum support structure) that are available for experiment irradiation and isotope production. These position diameters range from 1.6 centimeters (0.625 inch) to 12.7 centimeters (5 inches) with thermal neutron flux levels ranging from 1×10^{15} neutrons per square centimeter per second to 1×10^{13} neutrons per square centimeter per second. Approximately 25 percent of the high-flux test positions (A holes, B holes, and H holes) are currently used for iridium-192 production. The majority of the remaining high-flux test positions are used for cobalt-60 production. Occasionally, additional isotopes (e.g., strontium-89, nickel-63) are generated in small quantities. A private company leases the space for the production of these isotopes. A small number of positions are used by other companies or government programs for other materials irradiation projects. For the production of plutonium-238, neptunium-237 targets would be placed in the beryllium reflector positions. The proposed target design consists of neptunium dioxide blended with aluminum powder, pressed into a target core, and clad with aluminum. The basic ATR target should be similar in appearance to, but longer than, the typical transuranic isotope production target shown in **Figure 2–7**. The ATR target length would be sized for the 1.2-meter (4-foot) active core length of ATR. Beryllium reflector position sizes range from 1.6 centimeters (0.625 inch) in diameter to 12.7 centimeters (5 inches) in diameter.

ATR is equipped with numerous safety features, including extensive plant protective systems, standby power sources, experiment interlocks, computerized surveillance, confinement systems, safety rods, and an emergency firewater injection system. ATR's six safety rods provide fast shutdown of the reactor if potentially damaging conditions develop. A sudden rise in power or coolant temperature, a sudden drop in coolant flow or pressure, or the overheating of a test sample are examples of approximately 360 conditions that would automatically drop the safety rods into the core. The firewater injection system provides emergency core cooling and flooding of the reactor vessel in the event of a loss of primary coolant. ATR is connected by a water canal to the ATR Critical Facility. The ATR Critical Facility is a low-power, full-size nuclear duplicate of ATR that is used to provide data as needed for experiment loadings prior to irradiation of the actual experiments in ATR.

INEEL has privatized the production of medical and industrial isotopes through contracting with a commercial entity. International Isotopes Idaho, Inc. (I^4), was selected in October 1996 as the commercial business for conducting these business operations. I^4 specializes in producing isotopes targets for irradiation in ATR and processing and distributing commercial-grade isotopes to their customers. Prior to commercialization, INEEL's isotope production operations were limited in types and quantities. Since the start of commercial activities, I^4 has expanded its commercial production to become a major world supplier of several important isotopes. I^4 has doubled the use of ATR irradiation positions for this purpose.

The major isotopes currently produced by INEEL and I^4 are iridium-192, 70 percent of the total U.S. demand; cobalt-60, 95 percent of the U.S. medical market; strontium-89, only U.S. supplier; and nickel-63, only U.S. supplier and producer of 50 percent of the world market.

Incremental investments have been identified for ATR that would make it a more versatile and capable reactor for isotope production. I^4 and another commercial company are in the discussion phase of investing in ATR to install an isotope rabbit (shuttle) system for the production of short-lived radioisotopes. Many of these



Source: LMER 1998.

Note: To convert inches to centimeters, multiply by 2.54.

Figure 2–7 Typical Transuranic Isotope Production Target

short-lived radioisotopes are expected to be in growing demand for various cancer therapies. I^4 has also committed to keep part of the rabbit system available for other users.

2.3.1.3 High Flux Isotope Reactor

HFIR is a beryllium-reflected, light-water-moderated and -cooled reactor operating at a thermal power level of 85 megawatts. HFIR is owned by DOE and is in the 7900 Area of the ORNL site in the southern portion of the Oak Ridge Reservation (ORR). Figure 3–1 presents a map of ORR that depicts HFIR's location.

HFIR would continue to be operated to meet its primary mission of neutron science-based research for the DOE Office of Science. In addition, medical and industrial isotope production and civilian nuclear energy research and development activities would be performed on a not-to-interfere basis at the current operating level in the No Action Alternative, Alternative 1 (FFTF Restart), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities). When HFIR is supporting the plutonium-238 production mission, it would fully support its primary mission, but would support the medical and industrial isotope production and civilian nuclear energy research and development activities to the extent possible within the current reactor operating levels. Consideration must be given to the need to maintain appropriate levels of neutron flux to support HFIR's primary mission. Neutron flux levels can be impacted by the placement of targets (such as neptunium-237 targets for the production of plutonium-238) in the reactor core. Under the planning assumptions for plutonium-238 production, HFIR could only produce from 1 to 2 kilograms (2.2 to 4.4 pounds) per year without impacting ongoing missions. As the program goal is to achieve a production rate of 5 kilograms (11 pounds) per year, production from HFIR would need to be augmented by the use of ATR to meet this goal. HFIR and ATR together could meet the program goal of up to 5 kilograms (11 pounds) per year and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission.

HFIR was originally designed as both an isotope production and a research reactor with a thermal flux of 3 to 5×10^{15} neutrons per square centimeter per second and a full power level of 100 megawatts-thermal (3.4×10^8 British thermal units per hour). It is currently operating at a maximum authorized power level of 85 megawatts-thermal (2.9×10^8 British thermal units per hour) to extend the useful life of the reactor. Many experiment-irradiation facilities were provided for in the original design and several others have been added. The primary mission of HFIR is neutron science research. Isotope production is done on a not-to-interfere basis.

HFIR transfers its primary coolant heat load to secondary coolant through heat exchangers for dissipation to the atmosphere by an induced-draft cooling tower. The reactor uses highly enriched uranium and aluminum-clad plate fuel. The reactor vessel itself is immersed in a pool in a poured-concrete reactor building that also houses the primary coolant pumps and heat exchangers, a spent fuel pool, and experiment areas. The control and water wing of the reactor building contains the reactor control room; relay and amplifier areas; heating and ventilating equipment; pool and fire alarm equipment; instrumentation systems; and office and support rooms. A separate electrical building adjacent to the reactor building contains switchgear, diesel generators, and associated transformers that connect the facility to offsite power. The reactor building is essentially airtight and provides dynamic confinement. A special hot exhaust system exhausts air from potentially contaminated areas of the building through filters (two HEPA filters and two charcoal filters) before being released to the atmosphere through a 76-meter (250-foot) stack. The stack serves as the exhaust point for both HFIR and REDC at ORNL.

After the reactor completed 17.2 full-power years of its 20 full-power year design life in November 1986, several measures were taken to extend the useful life of the reactor, including reducing the

100 megawatts-thermal (3.4×10^8 British thermal units per hour) rated power level to 85 megawatts-thermal (2.9×10^8 British thermal units per hour); adjusting the primary coolant temperature and pressure; conducting periodic hydrostatic tests; establishing an irradiation embrittlement surveillance program; and installing an emergency depressurization system. Subsequent life extension programs can enable HFIR to provide support during the total 35-year evaluation period for operations.

A plan view of the reactor (**Figure 2–8**) provides a cross section of the reactor vessel depicting experiment irradiation capabilities. Available experiment irradiation facilities include (1) the hydraulic tube facility in the very high flux region of the flux trap that allows for insertion and removal of irradiation samples while the reactor is operating; (2) 30 target positions in the flux trap that normally contain transuranium production rods but can be used for the irradiation of other experiments (two instrumented target positions were provided by a recent modification); (3) six peripheral target positions at the outer edge of the flux trap; (4) numerous vertical irradiation facilities of various sizes located throughout the beryllium reflector; (5) two pneumatic tube facilities in the beryllium reflector that allow insertion and removal of irradiation samples while the reactor is operating for activation analysis; (6) four horizontal beam tubes that originate in the beryllium reflector; and (7) four slant access facilities, called "engineering facilities," located adjacent to the outer edge of the beryllium reflector. In addition, spent fuel assemblies are used for gamma irradiation in the gamma irradiation facility in the reactor pool.

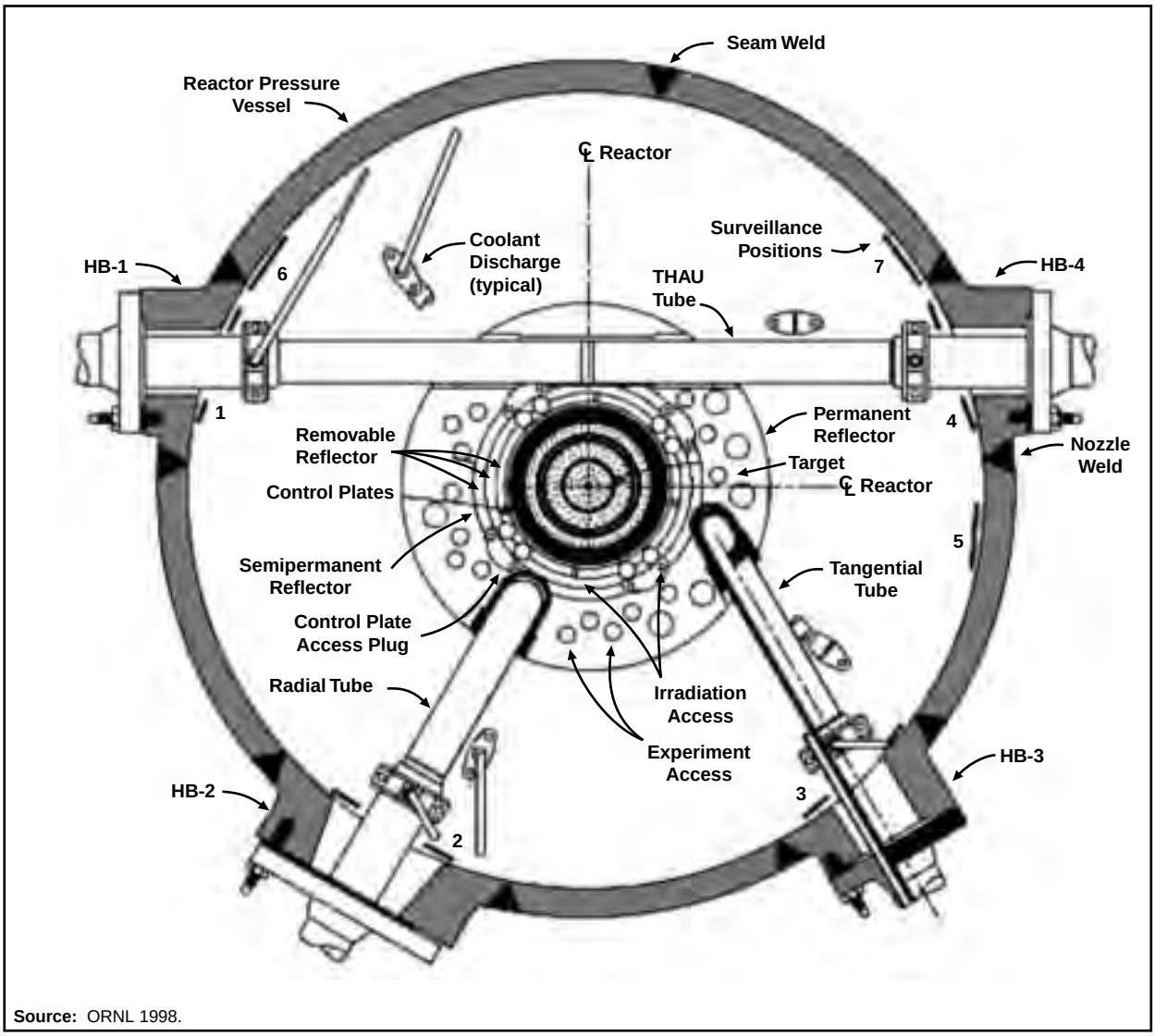
The reactor core assembly is contained in a 2.44-meter (8-foot) diameter pressure vessel in a pool of water. The top of the pressure vessel is 5.18 meters (17 feet) below the pool surface, and the reactor horizontal midplane is 8.38 meters (27.5 feet) below the pool surface. The control plate drive mechanisms are in a subpile room beneath the pressure vessel. These features provide the necessary shielding for working above the reactor core and greatly facilitate access to the pressure vessel, core, and reflector regions.

The neutron flux within HFIR is primarily a thermal neutron flux ranging from approximately 2×10^{15} neutrons per square centimeter per second in the flux trap to approximately 4×10^{14} neutrons per square centimeter per second in the outer regions of the beryllium reflector. Specially designed neutron beam tubes provide access to neutrons that supply intense neutron beams to various specialized instruments used for neutron scattering research.

ORNL produces a variety of medical isotopes using HFIR for irradiation and various hot cell and glovebox facilities for target fabrication and final product purification. **Table 2–2** presents a listing of HFIR-produced therapeutic radioisotopes. Key examples of the therapeutic radioisotopes currently produced in HFIR for distribution include dysprosium-166, rhenium-186, tin-117m, and tungsten-188 (parent of rhenium-188). The nine hydraulic tube positions in the central high flux region permit the insertion and removal of targets at any time during the operating cycle (22 to 24 days) and have traditionally been a major site for the production of medical radioisotopes.

In addition to providing radioisotopes for extramural research and development and commercial applications by distribution through the DOE Isotope Production and Distribution Program, there are medical radioisotope research and development programs at ORNL that depend on the availability of HFIR-produced radioisotopes.

The Isotopes Program at ORNL is totally funded by the DOE Office of Nuclear Energy, Science and Technology's Isotope Production and Distribution Program. It provides enriched stable isotopes, selected radioisotopes, and related technical services for use in a wide variety of research, industrial, and especially medical applications. The scope of work not only includes the production of radioisotopes, but also the development of new methods and equipment to produce, recover, and purify isotope products.



Source: ORNL 1998.

Figure 2–8 Plan View (Cross Section) of HFIR

Table 2–2 Examples of HFIR-Produced Radioisotopes of Current Interest for Therapy

Radioisotope	Half-Life	Target	Comment
Palladium-103	17 days	Palladium-102	Therapy of prostatic carcinoma
Rhenium-186	3.77 days	Rhenium-185	Therapy of prostatic carcinoma
Samarium-153	1.93 days	Samarium-152	Antibodies/bone pain palliation
Tin-117m	13.6 days	Tin-116 or tin-117	Bone pain palliation
Arsenic-77 (from germanium-77)	1.62 days	Germanium-76	Bone pain palliation
Gold-199 (from platinum-199)	3.14 days	Platinum-198	Phosphorus analogue
Tungsten-188 (rhenium-188 daughter)	69 days	Tungsten-186	Bone pain/antibodies/ synovectomy
Dysprosium-166 (holmium-166 daughter)	3.4 days	Dysprosium-164	Synovectomy/bone pain

2.3.1.4 Commercial Light Water Reactor

A CLWR would continue to operate and meet its primary mission requirement, providing steam for the generation of electrical power, in the No Action Alternative, Alternative 1 (FFTF Restart), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), and Alternative 2 (Use Only Existing Operational Facilities) when it is not providing irradiation services in support of the plutonium-238 production mission. When the CLWR is supporting the plutonium-238 production mission, it would fully support its primary mission. The production planning assumption for the generic CLWR is 5 kilograms (11 pounds) per year of plutonium-238 or 7.5 kilograms (16.5 pounds) per 18-month operating cycle. Thus, the CLWR alone could meet the program goal of up to 5 kilograms (11 pounds) per year and could be used in combination with any one of the three processing facilities for the plutonium-238 production mission. The use of a CLWR for the medical and industrial isotope production mission and the DOE civilian nuclear energy research and development mission was not considered practical, as discussed in Section 2.6.1.

A typical pressurized water reactor core consists of 170 to 200 fuel assemblies arranged in the reactor vessel in an approximately cylindrical pattern. Most pressurized water reactors operating in the United States are licensed to operate at thermal power levels of 2,500 to 3,500 megawatts (8.5×10^9 to 1.2×10^{10} British thermal units per hour) for net station electrical outputs of 800 to 1,200 megawatts-electric (2.7×10^9 to 4.1×10^9 British thermal units per hour).

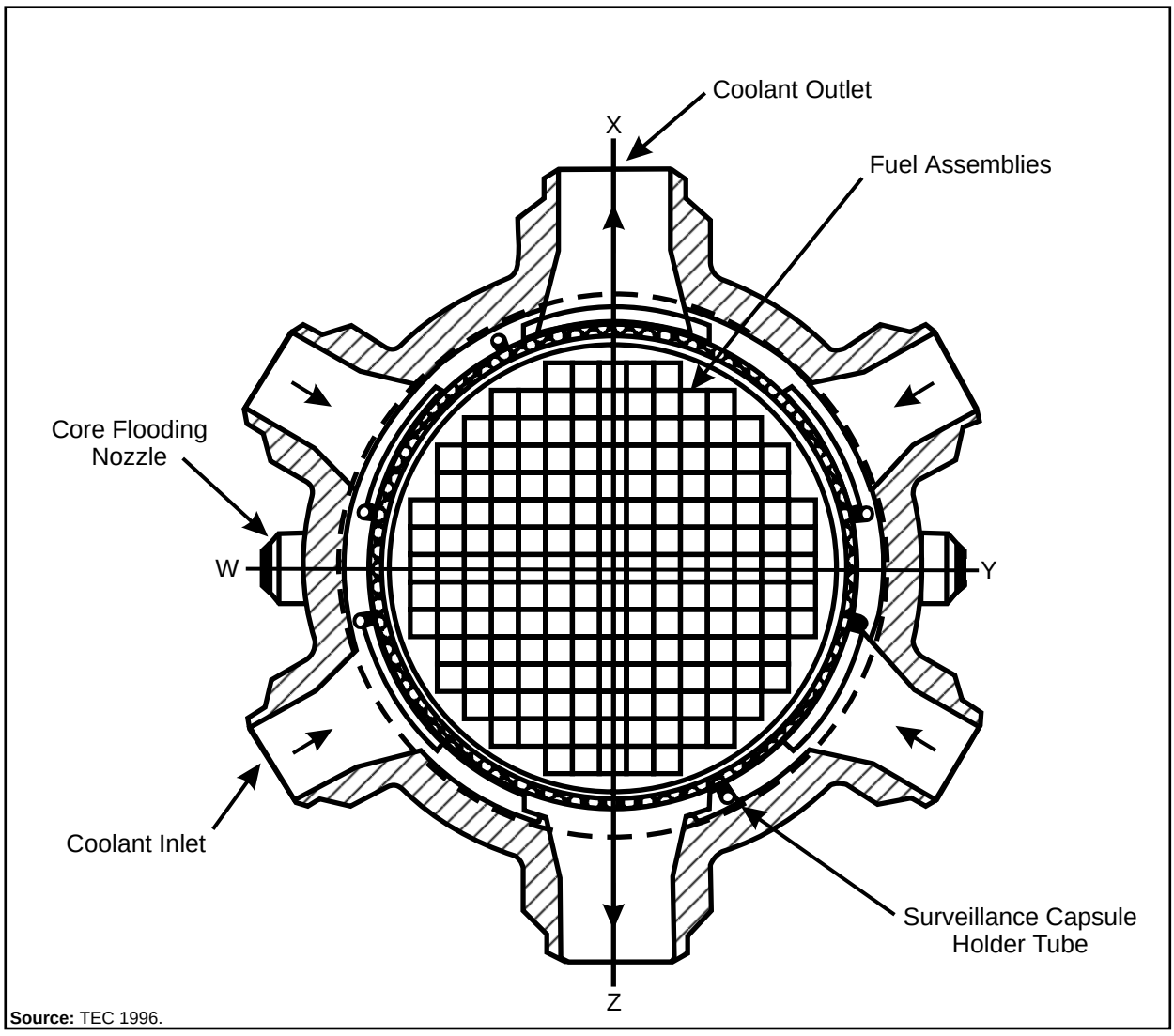
The nuclear steam supply system powered by the pressurized water reactor is generally arranged as two heat transport loops, each with two primary coolant circulating pumps and one steam generator in which the primary coolant dissipates heat generated in the reactor core to the secondary fluid in the steam generator. In addition to serving as a heat transport medium, the primary coolant also serves as a neutron moderator and reflector and as a solvent for the soluble boron used in chemical reactivity control. All nuclear steam supply system components are designed to withstand the effects of earthquakes and loss-of-coolant accidents.

The containment for a pressurized-water reactor plant consists of two structures: (1) a steel containment vessel and (2) a reinforced-concrete shield building.

The containment, including all of its penetrations, is a low-leakage steel structure designed to withstand a postulated loss-of-coolant accident and to confine a postulated release of radioactive material. It houses the reactor pressure vessel, reactor coolant piping, pressurizer, pressurizer quench tank and coolers, reactor primary coolant pumps, steam generators, core flooding tanks, and letdown coolers. Safety systems directly associated with this vessel include the containment spray system, the containment air cooling system, and the containment isolation system. An annular space is provided between the wall of the containment vessel and the shield building. Overhead clearance from the dome of the shield building is also provided.

The shield building itself is a concrete structure surrounding the containment that is designed to provide biological shielding during both normal operations and hypothetical accident conditions. The shield building enables the collection and filtration of fission product leakage from the containment following a hypothetical accident by means of its emergency ventilation system. In addition, the shield building provides environmental protection for the containment from adverse atmospheric conditions and external missiles (e.g., tornado debris).

A complete reactor core of 177 fuel assemblies, arranged in a square lattice that approximates a cylinder, is shown in **Figure 2–9**. All fuel assemblies are identical in mechanical construction and are interchangeable in any core location. The basic fuel assembly (**Figure 2–10**) is normally composed of 208 fuel rods, 16 control rod guide tubes, and one centrally-located position for instrumentation—all within a 15- by 15-position square

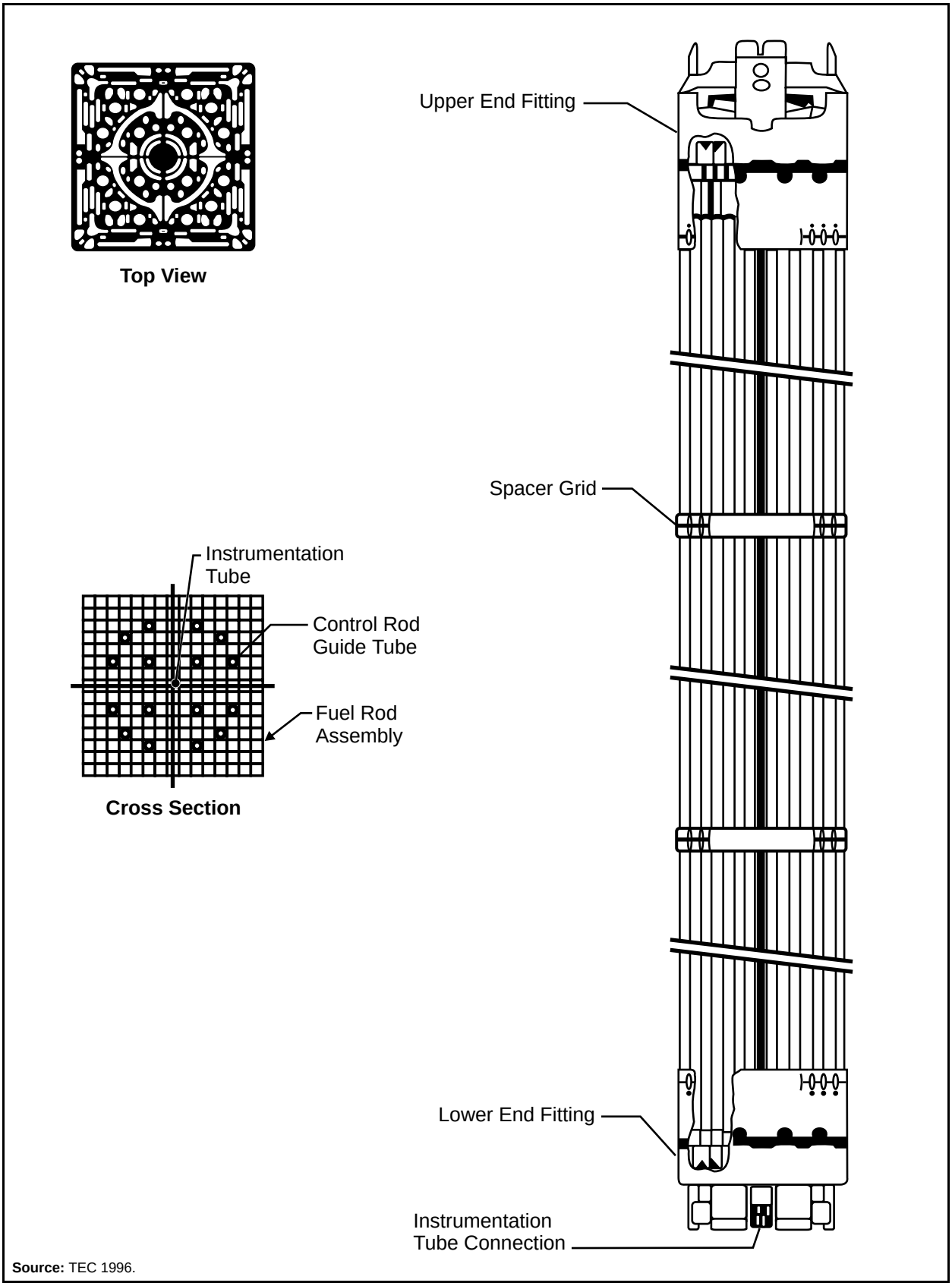


Source: TEC 1996.

Figure 2–9 Plan View (Cross Section) of a Generic CLWR

array. The fuel assembly is approximately 20.3 by 20.3 centimeters (8 by 8 inches) in cross section and has an overall length of 419 centimeters (165 inches).

The neptunium-237 targets can be placed in numerous locations within the reactor core region (i.e., fuel assembly region) and outside of the reactor core region to be irradiated for the production of plutonium-238. Three potential target arrangements were considered for evaluation in this NI PEIS: (1) all targets located in the center fuel assembly position in the reactor core, (2) all targets distributed within the reactor core region, and (3) all targets distributed outside the reactor core region. The center fuel assembly position was selected for evaluation in this NI PEIS because it was assumed that this would be the worst-case location during postulated beyond-design-basis accident conditions. This assumption conservatively postulated that during a beyond-design-basis core disruptive accident, temperatures in the center fuel assembly position would reach levels that would fail the cladding on all the neptunium-237 targets located in that position, resulting in worst-case releases.



Source: TEC 1996.

Figure 2–10 CLWR Fuel Assembly

The substitution of target rods for fuel rod positions in the center fuel assembly would only minimally impact reactor operations. The fuel rods located in the center fuel assembly position normally would not be fresh fuel (i.e., fuel inserted within the first 18-month operating cycle in the reactor); instead, they would be in their second or third operating cycle. The normal power distribution within the core and reactor coolant flow and its distribution within the core would remain within existing technical specification limits.

2.3.1.5 New Accelerator(s)

One or two new accelerators would be constructed and operated in Alternative 3 (Construct New Accelerator[s]). Preconceptual designs have been developed for a low-energy accelerator and a high-energy accelerator for evaluation in this NI PEIS (TechSource 2000). The low-energy accelerator would support the medical and industrial isotope production missions and the civilian nuclear energy research and development mission. This could effectively be accomplished with accelerator energies in the range of 30 to 70 million electron volts. The high-energy accelerator design would support the plutonium-238 production mission and the civilian nuclear energy research and development mission. An accelerator with an energy level of 1,000 million electron volts is required to support the plutonium-238 and nuclear research and development missions. The preconceptual design of the high-energy accelerator presented in Appendix F focused on supporting the plutonium-238 production mission. Although not analyzed in this NI PEIS, the design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce "tuneable" neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and materials interactions.

The accelerators would be constructed and operated at one or two existing DOE sites. The low-energy accelerator would be located on the same DOE site as the new support facility or at a DOE site with an existing support facility. The high-energy accelerator could be located at a different DOE site. Alternative 3 site selection was not evaluated as part of this NI PEIS. Because Alternative 3 was evaluated at a generic DOE site, no credit was taken for any existing support infrastructure at the site, and it was postulated that a new support facility would be required to support operation of the low-energy accelerator and its missions and the high-energy accelerator civilian nuclear energy research and development missions if both accelerators were located on the same site. While this approach bounds the environmental impact assessment for the implementation of Alternative 3, it overstates the impacts because this NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 3 or the low-energy accelerator alone is selected in the Record of Decision for subsequent consideration, follow-on NEPA reviews would evaluate potential locations for either both or one of the accelerators. It is unlikely that DOE would consider locating the new low-energy or high-energy accelerator on a DOE site that does not have an existing infrastructure capable of supporting all or most of the mission requirements. To determine the environmental impacts if Alternative 3 were implemented at a site with adequate support infrastructure, the environmental impacts for the construction of the support facility could be subtracted from the environmental impacts of Alternative 3 as presented in this NI PEIS. Section 4.5 of this NI PEIS presents the environmental impacts from construction and operation of the new support facility separately.

2.3.1.5.1 Low-Energy Accelerator

Three low-energy accelerator options would be available for the production of medical and industrial isotopes and to support nuclear energy research and development: (1) a high-current proton linear accelerator, (2) a

multiparticle cyclotron, or (3) a proton-only cyclotron. The proton-only cyclotron would have distinct technical advantages over the other two options and is described further in the sections that follow.

The proton-only cyclotron can be either a positive proton or negative ion type and is referred to as a proton cyclotron H^+ or proton cyclotron H^-. A positive proton cyclotron would offer lower vacuum requirements and, with the latest technology, a high-extraction efficiency. Obtaining variable energy output would be complicated, however, because extraction can be done using only a single port and splitting the beam would require a complicated septum magnet. In comparison, the negative ion cyclotron would offer a continuous beam with high-current capacity using very simple high-efficiency extraction, a simple method to vary the particle energy, and the possibility of simultaneous irradiation of two different target arrays at different energies. The high-extraction efficiency would be achieved simply by passing the negatively charged beam through a thin foil that strips the electrons from the ion, creating a positive proton. The proton would be directly ejected from the machine by the existing magnetic field with high efficiency (greater than 98 percent). This feature would be important to minimize the activation of the cyclotron structure and thus reduce radiation exposure to the operational staff.

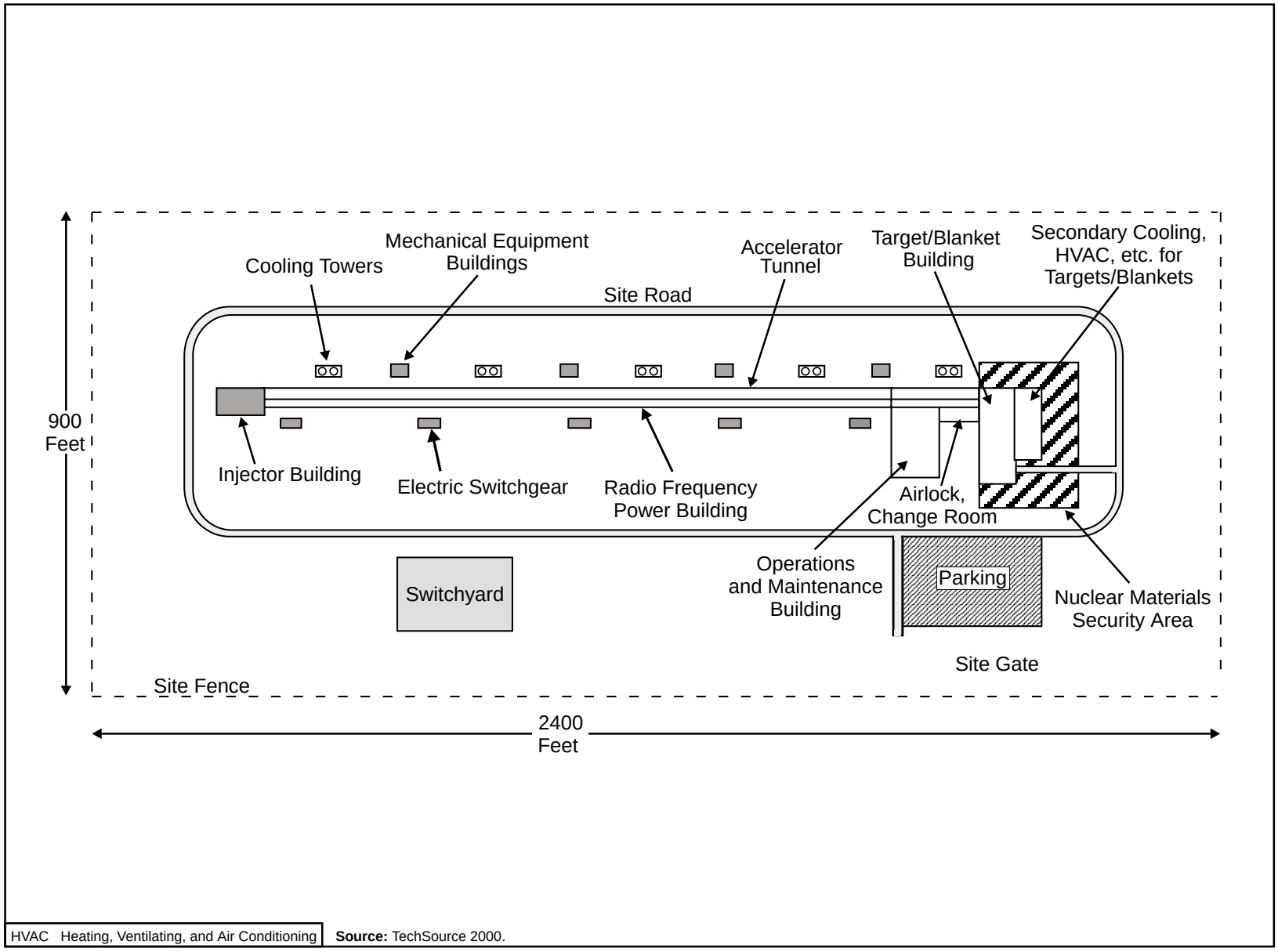
A high-beam current would be advantageous because more products could be prepared in a shorter time. In addition, a much higher specific-activity radioisotope could be prepared at the higher-beam current of the cyclotron. Specific activity is often a critical parameter in many nuclear medicine applications, including research and clinical use. The cyclotron could continuously tune the beam energy, which would be an advantage for research. The ability to tune the energy with precision could also help achieve high-purity isotope production by avoiding energies where impurity isotopes would be readily coproduced. These are important advantages for flexibility in research isotope production and are within the capabilities of commercially proven technology.

A new building with a 43-meter (140-foot) by 43-meter (140-foot) footprint would be constructed to house the cyclotron and the four beam lines. The walls of the facility would be 4.6 meters (15 feet) thick behind the target stations to minimize the neutron flux outside the building. The walls surrounding the cyclotron itself would be 3 meters (10 feet) thick. The mazes throughout the building would have walls 1.5 meters (5 feet) thick so that the total thickness surrounding the cyclotron area would be 3 meters (10 feet). The beam would be diverted to the four target stations by switching magnets located in the cyclotron vault. The beam would be directed through focusing and steering magnets to the target. In the isotope production beam line (northwest cave), the targets would be installed and removed vertically from a hot cell, which would be located on the second floor directly above the target station. The power supplies for the magnets would be housed with the power supplies for the cyclotron. The mechanical equipment for cooling water would be housed in a shielded mechanical room adjacent to the cyclotron vault. Recirculating water for cooling the targets and systems that could contain potentially radioactive material would be separated to prevent cross-contamination. These systems would be contained in mechanical equipment rooms near the respective target station. Piping would be contained in waterproof trenches with leak detection.

See Appendix F for additional details.

2.3.1.5.2 High-Energy Accelerator

In accelerator production of plutonium-238, an energetic beam of protons generated by a linear accelerator would be transported to a heavy metal target where spallation neutrons would be produced and moderated in a surrounding blanket. The blanket containing neptunium-237 would capture the slowed neutrons to produce plutonium-238 through the same nuclear sequence that occurs in a reactor. The accelerator would be housed in a concrete tunnel and buried below ground to provide radiation shielding for operating personnel. **Figure 2–11** presents the layout of the accelerator. A building to house radio frequency power systems and



HVAC Heating, Ventilating, and Air Conditioning **Source:** TechSource 2000.

Figure 2–11 Accelerator Production of Plutonium Plant Layout

other equipment used to drive, monitor, and control the accelerator would be above ground close to the accelerator tunnel. The target and blanket assembly would be housed inside a steel and concrete shield within a multistory building that would contain appropriate service equipment. At the target, the small-diameter proton beam transported magnetically from the accelerator would be converted to a much larger cross section by a beam expander to reduce the power density to acceptable levels for the target cooling systems. A source of neutrons produced by an accelerator can be used to produce plutonium-238 from neptunium-237 feedstock through the capture-and-decay nuclear processes. A 1,000-million-electron-volt proton beam produced by a radio frequency linear accelerator would bombard a heavy metal (uranium-238) target, with each proton producing about 40 neutrons. A very preliminary target and blanket design has been developed for scoping purposes, based on the architecture employed in the accelerator production of tritium target and blanket design. It would use uranium-238 (cooled by heavy water) as the neutron-production target. The target would be surrounded by a blanket of neptunium-237 in a dilute mixture of aluminum and water coolant. Enclosing the blanket would be a beryllium reflector.

To meet the plutonium-238 production goal of up to 5 kilograms (11 pounds) per year, the high-energy accelerator facility would conduct three 4-month production campaigns. Each campaign would be divided into 100 days of production and 21 days for recycling the production blanket. A 90 percent plant availability during the scheduled operating periods is assumed. Based on operating experience at the Los Alamos Neutron Science Center Linear Accelerator, 90 percent plant availability would be achievable. See Appendix F for additional details.

The preconceptual design of the high-energy accelerator presented in Appendix F focused on supporting the plutonium-238 production mission. While not evaluated in this NI PEIS, the design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce "tuneable" neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and materials interactions. The accelerator designs for Alternative 3 were developed to a level of detail that was adequate to assess the environmental impacts associated with the construction and operation of the proposed facilities and the technical feasibility of meeting the mission objectives. In the event that the NI PEIS Record of Decision selects Alternative 3, DOE would prepare conceptual, preliminary, and detailed designs and optimize the facility designs to accomplish the stated missions. Additional NEPA review would be required for site selection and to evaluate the environmental impacts of integrating the more refined accelerator designs with the existing site infrastructure(s).

2.3.1.6 New Research Reactor

A new research reactor would be constructed and operated in Alternative 4 (Construct New Research Reactor). A preconceptual design for a new research reactor was developed to meet the following DOE missions: (1) producing medical and industrial isotopes, (2) producing plutonium-238 (annual production of up to 5 kilograms [11 pounds]), and (3) supporting nuclear energy research and development. In accordance with U.S. nuclear nonproliferation policy, a design limitation of this new research reactor would be that it could only use low-enriched uranium with an enrichment of less than 20 percent uranium-235. This preconceptual design includes the basic elements of the research reactor facility, which are sufficient to support this NI PEIS, but does not include the design details (e.g., system and layout drawings, bill of materials, electrical and piping routing) commensurate with a complete preliminary reactor design.

The reactor design was developed to a level of detail that was adequate to assess the environmental impacts associated with the construction and operation of the proposed facilities and the technical feasibility of meeting the mission objectives. The design of the new research reactor is based on current research reactor designs that have been approved by both NRC and the International Atomic Energy Agency, as well as the nuclear regulatory authorities of many nations. Reactor core physics calculations were performed to evaluate three different nuclear fuel designs (described in Appendix E). Based on this analysis, the desired mission for this reactor, current nuclear fuel manufacturing capabilities, and safety considerations, a TRIGA (training, research, isotopes General Atomics) production reactor fuel design was selected for the new research reactor. The principal distinguishing features of the TRIGA fuel are its proven safety performance during power pulsing and its demonstrated long-term irradiation integrity.

To concurrently produce medical and industrial isotopes, meet the plutonium-238 production goal of up to 5 kilograms (11 pounds) per year, and provide irradiation services for civilian nuclear energy research and development, it was determined that a reactor core power of 50 megawatts-thermal would be necessary. Higher power levels and alternative target designs capable of meeting production requirements were also considered in the new research reactor design analysis but were not analyzed in this NI PEIS. For example, although not analyzed in this NI PEIS, operating at 100 megawatts-thermal could reduce the amount of neptunium-237 required to meet the plutonium-238 production requirements.

At the 50-megawatts-thermal power level, the core would require an active cooling system with forced coolant flow to maintain the fuel below its material thermal limits. The new research reactor cooling system would use a tank within a pool that is connected to primary coolant circulating pumps, heat exchangers, and an ultimate heat sink consisting of two cooling towers. The pool would be housed in a reactor building that also would enclose the pumps, heat exchangers, secondary systems, and spent nuclear fuel storage pool. The spent nuclear fuel storage pool, sized to store the reactor core's discharged spent nuclear fuel for its entire 35-year production period, could be hydraulically connected to the reactor core pool for refueling and emergency reflooding. The ultimate heat sink cooling towers, air exhaust stack, and emergency diesel generators would be located outside the reactor building.

The fuel for the new research reactor would be based on an extension of currently licensed low-enriched uranium TRIGA fuel designs for 10- to 16-megawatts-thermal reactors. The new research reactor fuel design would be identical to current low-enriched uranium TRIGA fuel for higher power cores, except the new reactor fuel would have a larger assembly configuration array (i.e., 8 by 8 versus 4 by 4) and a longer active fuel length (153.7 centimeters [60.5 inches] versus 55.88 centimeters [22.0 inches]). The larger array and length were selected to meet the plutonium-238 production requirements and to maintain high safety factors with respect to fuel thermal performance.

Along with the fuel rods, the core would contain a number of medical and industrial isotope and plutonium-238 production target rods. These rods would occupy positions in a fuel assembly where a fuel rod would otherwise exist. Each of these positions would have an Incoloy-800 alloy guide tube with the same dimensions as the fuel rod cladding. The target rods would be inserted into these guide tubes for their design irradiation time period. In addition, some fuel rod positions in core fuel assemblies would be replaced with similar guide tubes to accommodate Incoloy-800-clad boron carbide control rods. Boron carbide is a widely used, proven, and accepted neutron absorber for control rods. **Figure 2–12** presents a representative illustration of the fuel rod; the neptunium-237, medical, or industrial radioisotope target rod; and the control rod. **Figure 2–13** shows a cross-sectional view of each type of fuel assembly in the core. The new research reactor core design would consist of 68 fuel assemblies, each of which would be enclosed in a square aluminum shroud for structural support and coolant flow control. Key design features of the core are provided in Appendix E.

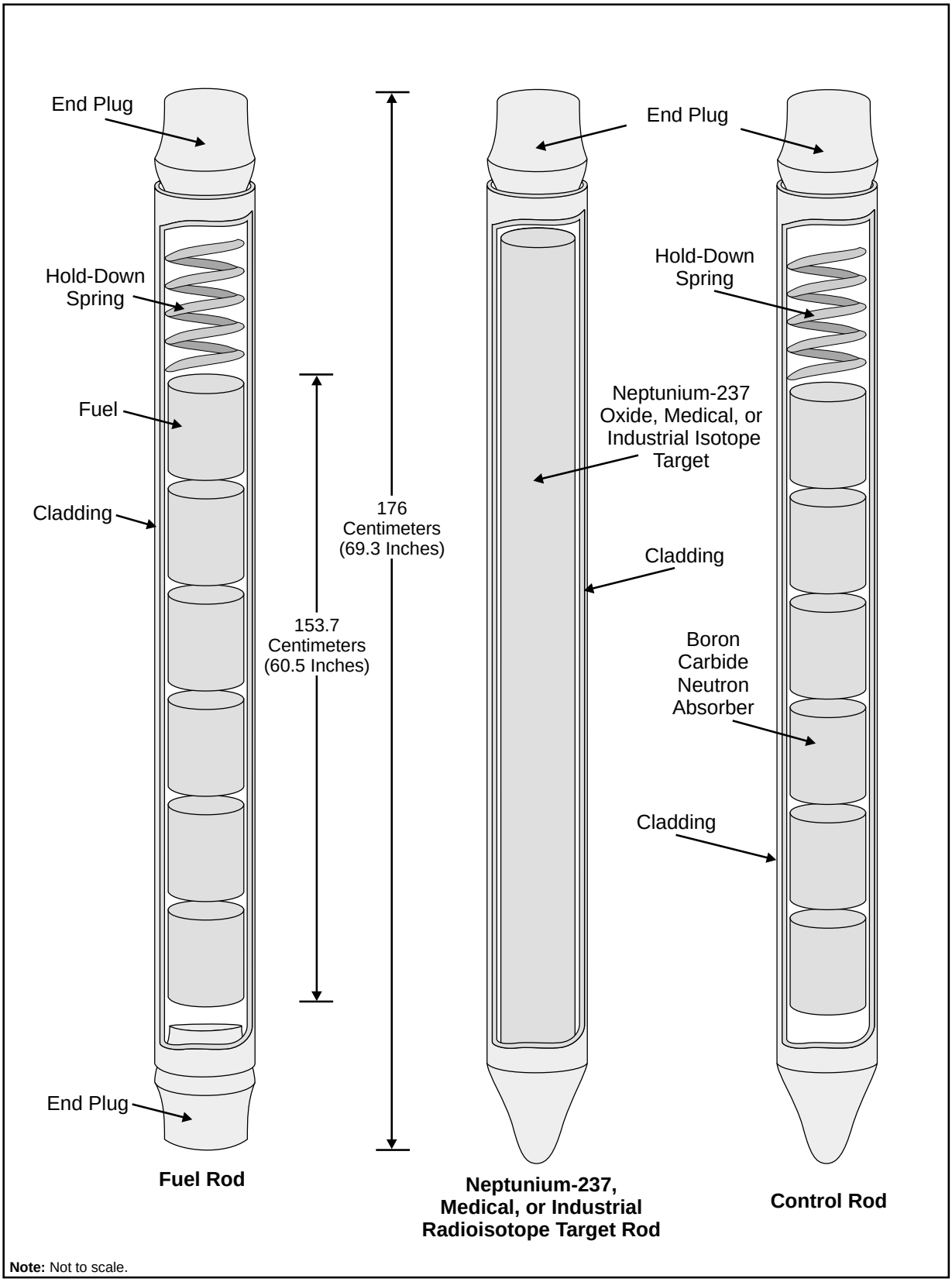


Figure 2–12 Representative Illustration of Fuel Rod; Neptunium-237, Medical, or Industrial Radioisotope Target Rod; and Control Rod (New Research Reactor)

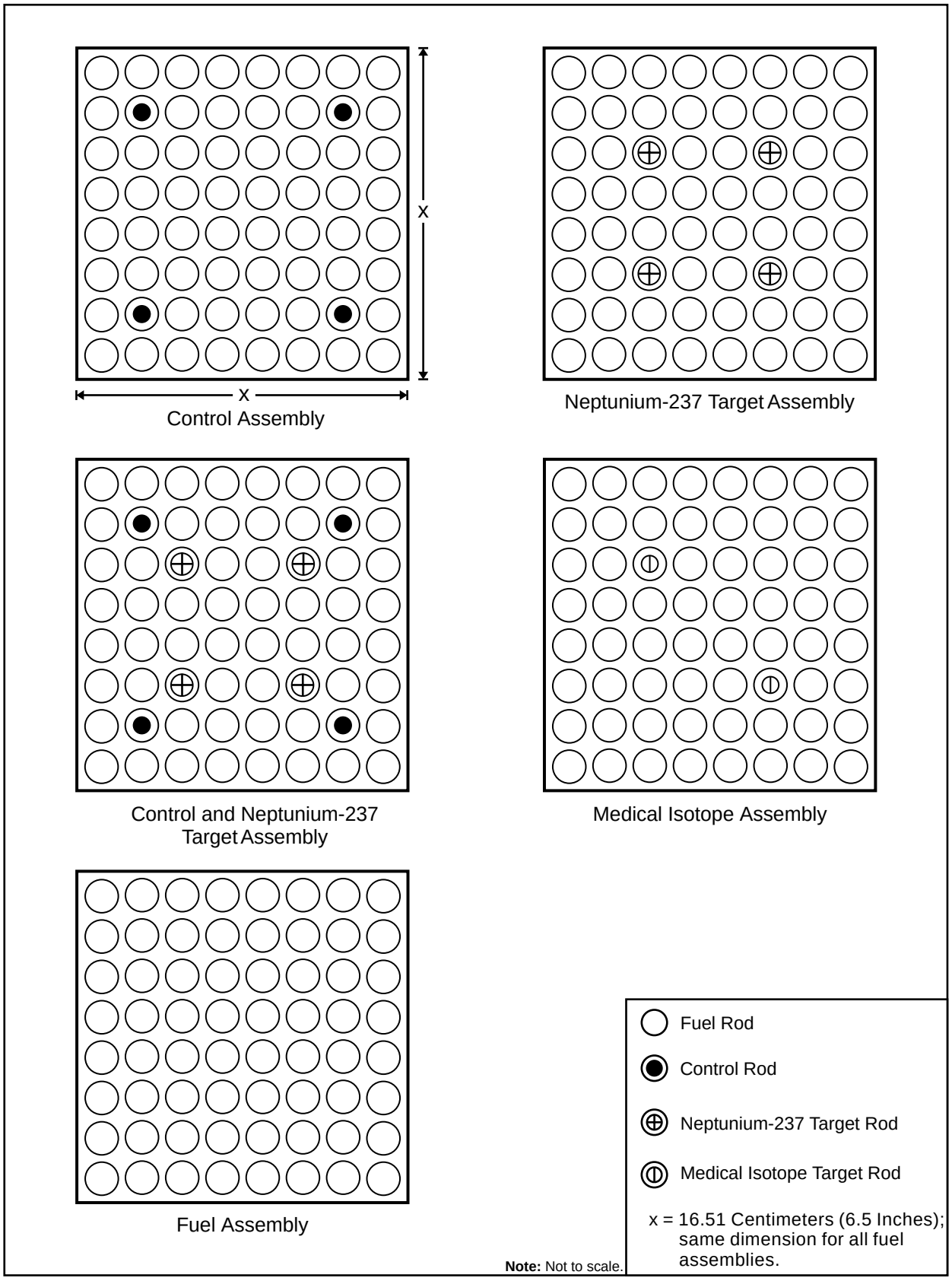


Figure 2–13 Cross-Sectional View of Fuel Assemblies in the Core (New Research Reactor)

The core would include eight rabbit tubes for short irradiation time production of medical or industrial isotopes and civilian nuclear energy research and development. These rabbit tubes would be located outside the fuel region of the core, but still within an area with a relatively high neutron flux. A cross-sectional view of the new research reactor core showing the layout of fuel assemblies, target rod assemblies, control rod assemblies, reflector, and rabbit tubes is presented in **Figure 2–14.**

The new research reactor would be constructed and operated at an existing DOE site. Since the potential site has not been selected, it is evaluated in this NI PEIS as a generic DOE site. Because Alternative 4 was evaluated at a generic DOE site, no credit was taken for any existing support infrastructure at the site, and it was postulated that a new support facility would be required to support operation of the new research reactor and its medical isotope production and civilian nuclear energy research and development missions. While this approach bounds the environmental impact assessment for the implementation of Alternative 4, it overstates the impacts because this NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 4 is selected in the Record of Decision for subsequent consideration, follow-on NEPA reviews would evaluate potential site locations. It is unlikely that DOE would consider locating the new research reactor on a DOE site that does not have an existing infrastructure capable of supporting all or most of the mission requirements. To determine the environmental impacts if Alternative 4 were implemented at a site with adequate support infrastructure, the environmental impacts for the construction of the support facility could be subtracted from the environmental impacts of Alternative 4 as presented in this NI PEIS. Section 4.6 of this NI PEIS presents the environmental impacts from construction and operation of the new support facility separately.

Reactor Operation

Operation of the new research reactor would be similar to other research reactors except that the core would be maintained at full power for a minimum 80 percent of the year. At the beginning of a cycle of operation, neptunium-237 and medical isotope target rod assemblies that require a long irradiation time would be inserted into their appropriate fuel assembly sleeve locations. The target rods would be mechanically attached to a cluster spider assembly similar to that used for the control rod assembly. The neptunium-237 target rod assemblies would remain in the core for the entire annual fuel cycle. These target rod assemblies would be removed from the host fuel assembly without removing the fuel assembly from the core, and then would be transferred to the spent fuel storage pool using the transfer canal. Medical and industrial isotope target rods that require a 100-day irradiation cycle would be removed and replaced with new target rod assemblies during brief reactor shutdown periods. These target rod assemblies would be removed and transferred in a manner similar to that of the neptunium-237 target rod assemblies. Isotopes that require only a short irradiation time would be inserted into rabbit tubes for the required 10- to 25-day time period. The eight rabbit tubes would be located outside the core, but inside the reflector region. The insertion and removal of irradiation targets in the rabbit tubes would have no significant effect on core reactivity and would not affect power operation.

After an isotope-specific cooling time in the spent fuel pool, the medical and industrial isotope and neptunium-237 target assemblies would be transferred to a shipping cask in the spent fuel storage pool. Using the overhead crane in the spent fuel pool area, shipping casks would be placed onto a truck in the reactor building bay adjacent to the fuel storage pool for shipment to the processing facility. New targets would be shipped from the target preparation facility into the reactor building bay by truck, transferred into the spent fuel storage pool, and subsequently moved to the reactor core pool or rabbit tube area for insertion into the core.

The plutonium-238 annual production goal of up to 5 kilograms (11 pounds) was calculated to be achieved with a 300-day annual irradiation time, which corresponds to a capacity factor of approximately 80 percent. Key reactor annual resource requirements are presented in Appendix E.

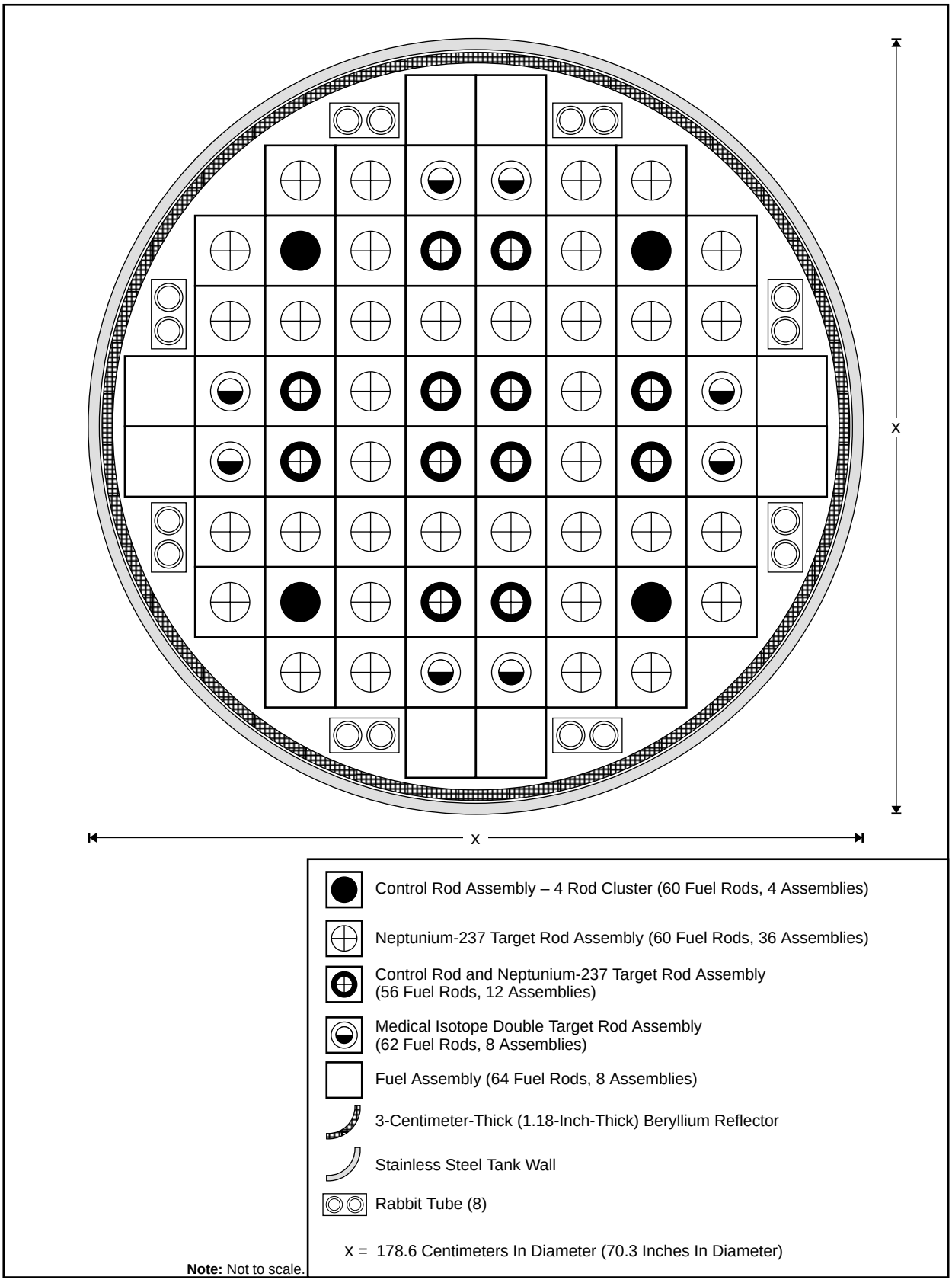


Figure 2–14 Cross-Sectional View of Research Reactor Core

Reactor Construction

Construction of the new research reactor facility was determined to require 4 years after design and licensing activities have been completed (AECL 1996; ANSTO 1999).

2.3.2 Target Fabrication and Postirradiation Processing Facilities

The proposed DOE facilities that would be used for the fabrication, storage, and postirradiation processing of the targets necessary for the program mission are (1) REDC at ORNL, (2) FDPF and/or Building CPP–651 at INEEL, (3) FMEF at Hanford, (4) RPL/Building 306–E at Hanford, or (5) a new target fabrication and processing facility at an existing DOE site that would support medical and industrial isotope production for targets irradiated in the proposed new low-energy accelerator or research reactor facilities. REDC, FDPF, and CPP–651 would support plutonium-238 production; FMEF would support both plutonium-238 and medical and industrial isotope production. The RPL/306–E facilities and the new facility would support only medical and industrial isotope production.

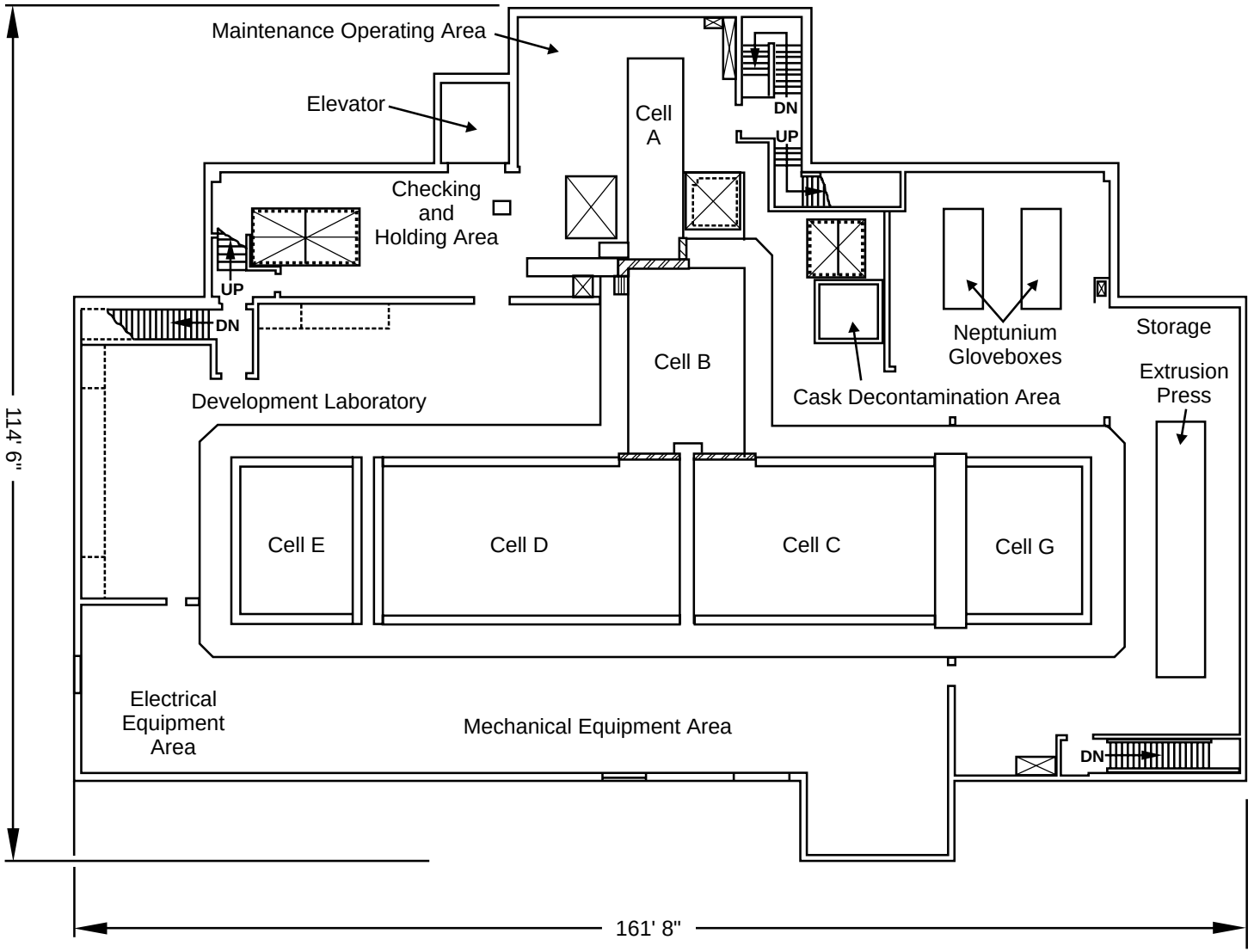
2.3.2.1 Radiochemical Engineering Development Center

REDC at ORNL is a companion facility to HFIR. The REDC's two buildings house heavily shielded hot cells and analytical laboratories that are used for remote fabrication of rods and targets (for irradiation in HFIR) and processing of irradiated rods and targets for the separation and purification of transuranic elements, process development, and product purification and packaging.

ORNL's REDC Building 7930 is proposed for storage of neptunium-237 under one option of the No Action Alternative. It also is proposed for storage of neptunium-237, fabrication of neptunium-237 targets, and processing of irradiated neptunium-237 targets under two irradiation options in Alternative 1 (FFTF Restart), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and one irradiation option each in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor). REDC's current radiochemical missions would not be impacted by the addition of the proposed storage of neptunium-237, fabrication of neptunium-237 targets, and the processing of irradiated neptunium-237 targets activities. REDC would have no role in support of Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). Figure 3–1 presents a map of ORR that depicts REDC's location.

REDC Building 7930 is divided into four major areas: (1) a cell complex with seven cells, six shielded and one unshielded; (2) maintenance and service areas surrounding the cell complex; (3) an operating control area; and (4) an office area adjacent to, but isolated from, the operating areas. Utility services, ventilating systems, crane and manipulator systems, and liquid waste systems also are included. The proposed plutonium-238 processing and storage activities would require equipment installation in three main areas of the second floor of REDC Building 7930. A plan view of these areas is shown in **Figure 2–15.** REDC hot cell facilities that would be used for the proposed action have never been used. The activities required for target fabrication would take place in shielded gloveboxes. The mechanical operations involved in the final target fabrication process may present lesser hazards that permit them to be carried out in open boxes. Cell E would contain processing equipment to purify the separated plutonium-238 product, prepare the plutonium oxide, and transfer the oxide into shipping containers. Cell E would also contain vertical storage wells for dry storage of neptunium and other actinides.

Cell D activities would include receipt of irradiated targets, as well as target dissolution, chemical separation of neptunium and plutonium from fission products, and partitioning and purification of neptunium. Cell D



Note: To convert feet to meters, multiply by 0.3048.
Source: Wham et al. 1998.

Figure 2–15 REDC Building 7930 Second Floor Plan View (Cross Section)

also contains process equipment for removing transuranic elements from the aqueous waste streams and vitrifying the waste.

The neptunium dioxide (NpO_2) containers would be stored in specially designed storage vaults to provide secure, safe storage for the materials. DOE safeguards and security guidelines would be followed whenever the material is being stored, transported, or processed. Detailed descriptions of the facility and the processes associated with storage, target fabrication, and postirradiation processing are provided in Appendix A.

2.3.2.2 Fluorinel Dissolution Process Facility

FDPF is in the Idaho Nuclear Technology and Engineering Center (INTEC) that is northeast of the Central Facilities Area at INEEL and approximately 3.2 kilometers (2 miles) southeast of ATR. FDPF is proposed for fabrication of neptunium-237 targets and processing of irradiated neptunium-237 targets under two irradiation options in Alternative 1 (FFTF Restart), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and one irradiation option each in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor). Figure 3–6 presents a map of INEEL that depicts FDPF's location.

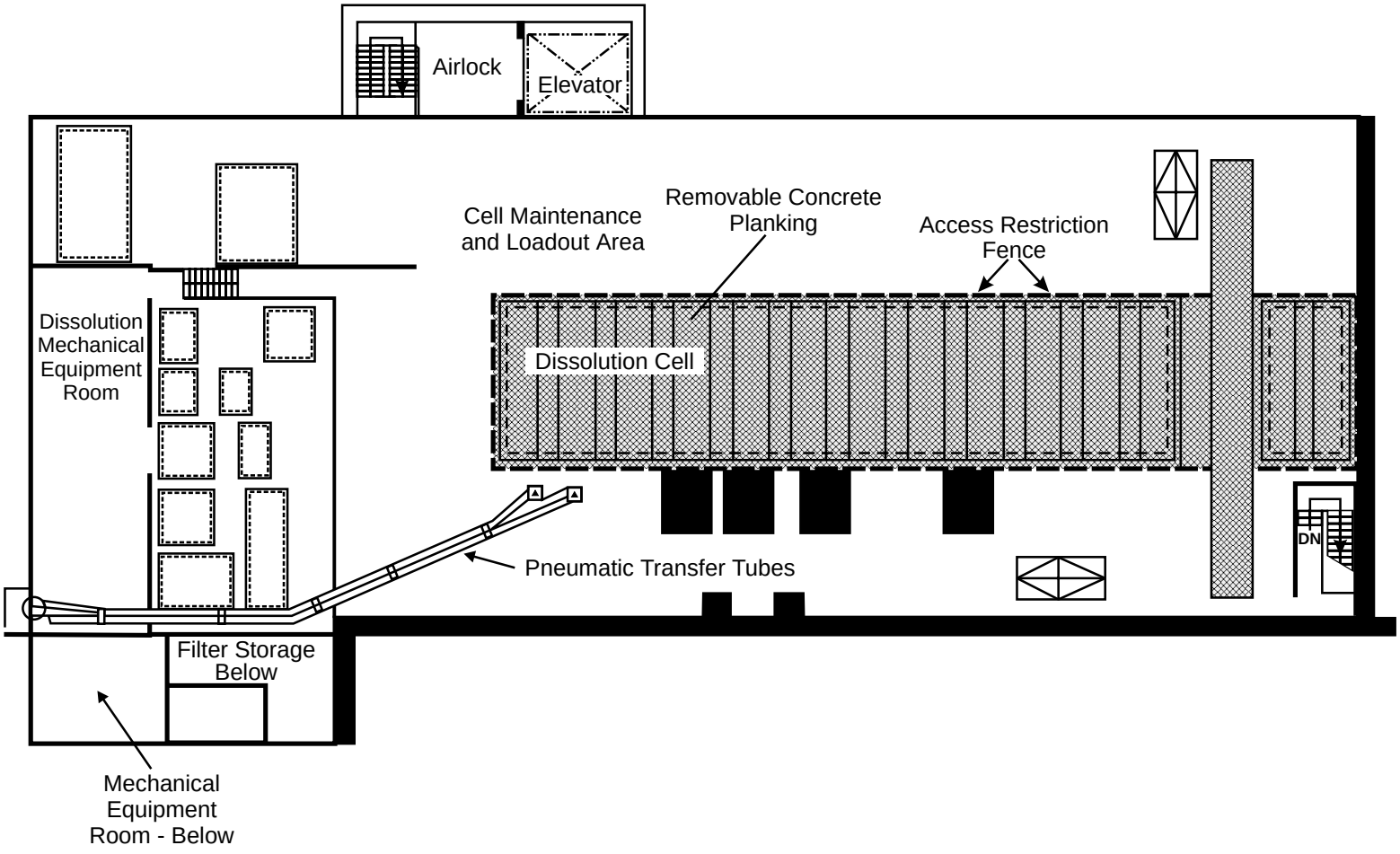
FDPF has no current mission. Historically, INTEC reprocessed spent nuclear fuel from U.S. Government reactors to recover reusable highly enriched uranium. After DOE announced in April 1992 that it would no longer reprocess spent fuel, reprocessing operations at INTEC ended. Two buildings at INTEC are candidate storage and processing sites for plutonium-238 production: Building CPP–651, the Unirradiated Fuel Storage Facility, and Building CPP–666, FDPF. Building CPP–651 was originally designed for the storage of special nuclear materials to support Defense Programs and is quite flexible in terms of the size and shape of special nuclear materials that it can receive and store. The 100 storage positions in the vault use the existing structural barriers of Building CPP–651 (earth and concrete) and provide supplemental security protection via their in-ground concrete storage silo design. Each storage position houses a rack that holds seven highly enriched uranium product cans. Racks are raised and lowered in their storage positions via an overhead 1-ton hoist.

Building CPP–666 is divided into two parts, the Fuel Storage Facility and FDPF. The Fuel Storage Facility consists of receiving and unloading areas, a fuel unloading pool, and six storage pools for storing nuclear fuel.

FDPF was designed and built to process Navy fuel via three dissolver trains. When fuel reprocessing was discontinued, uranium and hazardous materials were flushed from FDPF, and the facility is currently under consideration for new missions. FDPF consists of a large hot cell and supporting areas with a total area of approximately 3,700 square meters (40,000 square feet). The facility is divided into five levels that are identified by their elevation relative to ground level (Hochhalter 1982). A floor plan of the plus 28-foot level, the proposed location for target fabrication activities, is shown in **Figure 2–16**.

The chemical separation would take place in the FDPF cell using small centrifugal contactors installed for that purpose. Storage of neptunium-237 would be performed in Building CPP–651, which is located within 100 meters (328 feet) of FDPF. There are 100 in-ground concrete-shielded storage well positions in this vault. Each storage well contains a rack that can be modified to house cans of neptunium-237.

The neptunium dioxide containers would be stored in specially designed storage vaults to provide secure, safe storage for the materials. DOE safeguards and security guidelines would be followed whenever the material is being stored, transported, or processed.



Source: Hochhalter 1982.

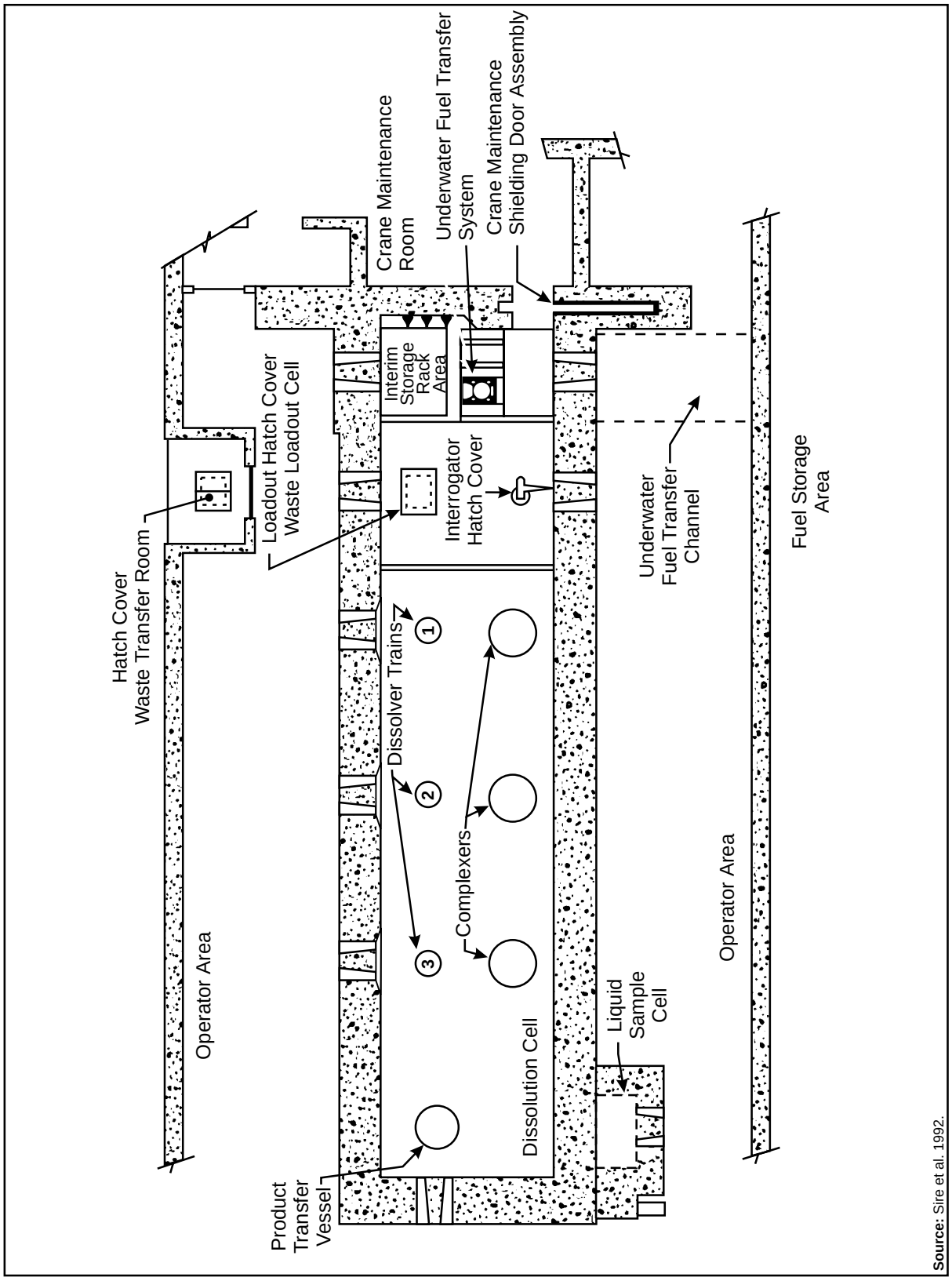
Figure 2–16 Fluorinel Dissolution Process Facility, Plus 28-Foot Level

The FDPF portion of INTEC includes a hot cell about 6.1 meters (20 feet) wide, 30.5 meters (100 feet) long, and 15.2 meters (50 feet) deep that is shielded by 1.8-meter-thick (6-foot-thick) concrete walls, as shown in **Figure 2–17**. The cell was designed to process Navy spent nuclear fuel via three dissolver trains, each of which consists of a 1,700-liter (450-gallon) Hastelloy C-4 dissolver and a 6,510-liter (1,720-gallon) Hastelloy C-4 complexer vessel in series. Each train is connected to a common 8,000-liter (2,110-gallon) stainless steel product transfer vessel that was used for accountability sampling prior to transferring the adjusted fuel dissolution product for solvent extraction separations. If the targets were dissolved in a continuous process, a small, 12.5-liter (3.3-gallon) dissolver would be skid-mounted on the grate at the level of the dissolver lids, and the dissolvers would be used for collecting the dissolution product of irradiated neptunium-237 targets. If a batch dissolution process were used, a small 200-liter (53-gallon) dissolver system designed for the small target dissolution throughput rate could be skid-mounted on the grate at the level of the dissolver lids for batch processing. Three complexer vessels in the cell could be used for waste or rework solution collection, or for the collection of condensate if a waste evaporator were employed. The head-end dissolution system is supported by remote manipulators (overhead and master/slave), as well as an underwater fuel transfer system and crane for target transfer and waste loadout. The dissolver offgas system scrubs potentially hazardous chemicals and filters radioactive particles from the offgases of the process vessels before releasing them into the heating, ventilating, and air conditioning exhaust system. The chemical separation would take place in the FDPF cell using small centrifugal contactors installed for that purpose. The storage of neptunium-237 would be performed in either FDPF or in a secure vault facility, Building CPP–651, located within 100 meters (328 feet) of FDPF. There are 100 in-ground concrete-shielded storage well positions in this vault. Each storage well contains a rack that can be modified to house cans of neptunium-237. Detailed descriptions of the facility and the processes associated with storage, target fabrication, and postirradiation processing are provided in Appendix A.

2.3.2.3 Fuels and Materials Examination Facility

Use of Hanford's FMEF is proposed for storage of neptunium-237 under one option of the No Action Alternative. It also is proposed for storage of neptunium-237, fabrication of neptunium-237 targets, and processing of irradiated neptunium-237 targets under two irradiation options in Alternative 1 (FFTF Restart), three irradiation options in Alternative 2 (Use Only Existing Operational Facilities), and one irradiation option each in Alternative 3 (Construct New Accelerator[s]) and Alternative 4 (Construct New Research Reactor). In addition to the support of the plutonium-238 production mission activities in Alternative 1, FMEF would also support medical and industrial production mission and civilian nuclear energy research and development mission activities at Hanford. FMEF would have no role in supporting Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). FMEF is adjacent to the west of FFTF in the 400 Area of Hanford. Figure 3–12 presents a map of Hanford that depicts FMEF's location.

FMEF was built during the late 1970s and early 1980s as a major addition to the breeder reactor technology development program at Hanford. Although it has never been used, the facility was constructed to perform fuel fabrication and development and postirradiation examination of breeder reactor fuels (DOE 1995b). FMEF is currently being maintained in a condition suitable for a future mission. In 1998, FMEF was placed into a partial layup condition to reduce the cost of maintaining the facility. Many systems were shut down and most hazardous materials were removed from the building. FMEF is considered clean and uncontaminated because no nuclear materials have been introduced (Hoyt et al. 1999). Some critical systems, such as the fire detection and protection systems, remain in operation. In order to avoid freezing of the fire protection water systems, limited heating and ventilating remain available. For example, the heating, ventilating, and air conditioning system has been modified to simplify its operation by blocking automatic dampers in appropriate configurations. Also, although the chillers have been laid up, including removal of the refrigerant, the chilled water system (containing an ethylene glycol-water mixture) remains available to help distribute heat within the building. Electrical power and lighting remain available, and the freight elevator remains in service to



Source: Sire et al. 1992.

Figure 2–17 Plan View of FDPF Cell

support routine facility walkdowns and any required maintenance. FFTF staff conduct surveillance and maintenance of FMEF.

FMEF consists of a 30-meter-high (98-foot-high) Process Building that has an attached Mechanical Equipment Wing on the west side and an Entry Wing on the south (front) side. The Mechanical Equipment Wing houses utility and support equipment, including water treatment equipment, air compressors, and a portion of the air conditioning equipment.

The Entry Wing contains space for reactor fuel assembly (recently used as a training facility in support of the Hanford Site cleanup mission), lunchroom and change rooms, and heating and air conditioning equipment associated with the Entry Wing. Personnel access into the Process Building is provided via a Security Guard Station and automated personnel access control portals located on the first floor of the entry wing. Office space and administrative support areas are also housed on the second floor of the Entry Wing (DOE 1995b).

The Process Building is approximately 53.3 meters (175 feet) wide by 82.3 meters (270 feet) long and extends from around 10.7 meters (35 feet) below grade to 30 meters (98 feet) above grade. Total potential operating space is approximately 17,470 square meters (188,000 square feet). The Process Building contains several large interconnected hot cells and many smaller connected hot cells. Major cranes are available, but some cranes, windows, and manipulators were not installed because construction of FMEF was halted prior to completing work on the hot cell complex (Hoyt et al. 1999). Nevertheless, the building is divided into six operating floors or levels that are identified in the following manner by their elevation relative to ground level and their primary function:

- The top floor at the 21.3-meter (70-foot) elevation is called the Secure Automated Fabrication Level. This level contains the Secure Automated Fabrication Line, automated fabrication equipment originally designed to produce reactor fuel.

- The lower Fuel Fabrication Level at the 13-meter (42.5-foot) elevation consists of two separate operating areas—one designated as the Low Gamma Test Pin Fabrication and Development Area and the other as the Unit Process Cell. This level provides approximately 470 square meters (5,100 square feet) of potential operating space around the Unit Process Cell. This cell area is highly shielded by thick concrete walls and was intended for the future development of remote fabrication and maintenance equipment or for the production of high gamma test pins. However, this cell area is not equipped at this time.

- The lower Chemistry Level at the 6.5-meter (21.25-foot) elevation surrounds the upper portions of the Nondestructive Examination Cell and the Decontamination Cell, which extend upward from the floor below. This level was designed to contain equipment to perform the chemical analyses of fuel material necessary to support fuel fabrication work. Much of the work planned in this area was to be performed in gloveboxes to reduce personnel radiation exposures. Also located on this level is an automated system that is potentially available for handling and storing the special nuclear material, such as the feed material for the fuel fabrication processes. The area encompasses approximately 790 square meters (8,500 square feet) of potential operating space.

- The Entry Level at ground level is the main operating floor of the Nondestructive Examination Cell, which also extends into the floors above and below. The Nondestructive Examination Cell was designed to contain remotely operated equipment for the nondestructive examination of irradiation fuel assemblies and pins. Maintenance and decontamination of equipment were to be performed in the adjacent Decontamination Cell. The Entry Level also contains computer and operations control rooms and inert gas systems and building air exhaust equipment. The Shipping and Receiving Area,

which is approximately 500 square meters (5,400 square feet) of operating floor space, is at the extreme east end of the Process Building on the Entry Level. This area includes a liquid waste loadout station, a solid waste storage area, a truck lock, and a large high-bay material-handling area.

- The Equipment Level at the minus 5.3-meter (17.5-foot) elevation was designed to contain a variety of support equipment, including two separate electrical switchgear rooms, emergency air compressors, heating and ventilating system air supply equipment, Nondestructive Examination Cell inert atmosphere equipment, emergency batteries, analytical chemistry cell exhaust equipment, and building air filtering system components. Also included is the vacuum equipment associated with the vacuum and air sample vacuum systems.

- The DE Cell Level at the minus 10.7-meter (35-foot) elevation (see **Figure 2–18**) contains cells originally intended for destructive examination of fuels and materials samples. These cells are arranged in two parallel rows along a horizontal transfer corridor that was to be used to transfer equipment between individual cells. The DE Cell area is heavily shielded, and work in the cells was planned to be performed using remotely operated equipment. The Entry Tunnel extends from below the Shipping and Receiving Area floor (on the Entry Level) to the DE Cell Level (10.8 meters or 35.5 feet total height). The Entry Tunnel was designed to house a 75-ton rail-mounted transporter intended to transfer casks between the Shipping and Receiving hatch and the Decontamination Cell and Nondestructive Examination Cell Floor penetrations. The transporter rails are roughly halfway up the tunnel at the 5-meter (16.2-foot) elevation.

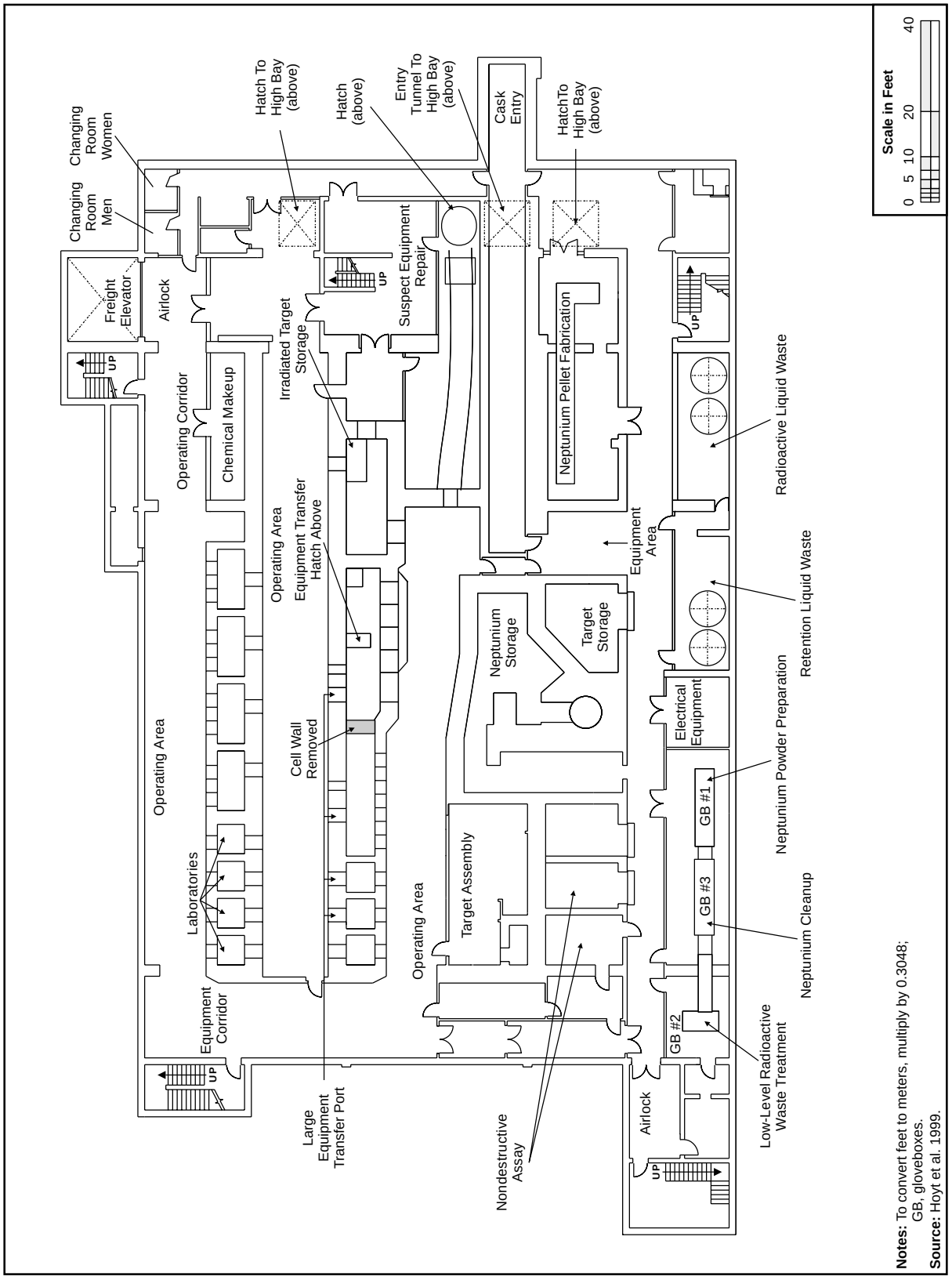
The use of FMEF for neptunium-237 target material storage, target fabrication, and postirradiation processing would require the construction of a new 76-meter (250-foot) stack. The neptunium dioxide (NpO_2) containers would be stored in specially designed storage vaults to provide secure, safe storage for the materials. DOE safeguards and security guidelines would be followed whenever the material is being stored, transported, or processed. Detailed descriptions of the facility and the processes associated with storage, neptunium-237 target fabrication, and postirradiation processing in support of plutonium-238 production are provided in Appendix A.

2.3.2.4 Radiochemical Processing Laboratory and Building 306–E

Two Hanford 300 Area facilities are proposed to support medical and industrial isotope target fabrication and postirradiation: RPL and Building 306–E (RPL/Building 306–E). The facilities support the four irradiation options of Alternative 1 (FFTF Restart) that are not supported by FMEF. RPL and Building 306–E would be used to support medical and industrial isotope production and civilian nuclear energy research and development activities. These activities would not impact current missions at the facilities. RPL and Building 306–E have no role in support of the No Action Alternative, Alternative 2 (Use Only Existing Operational Facilities), Alternative 3 (Construct New Accelerator[s]), Alternative 4 (Construct New Research Reactor), and Alternative 5 (Permanently Deactivate FFTF [with No New Missions]). Figure 3–12 presents a map of Hanford that depicts the locations of RPL and Building 306–E. The following descriptions are based on the *Hanford Data Request for FFTF Operational Support Facilities* (BWHC 1999).

2.3.2.4.1 Radiochemical Processing Laboratory

The research and development activities of the Radiochemical Processing Group are conducted at RPL in the 300 Area of Hanford. RPL consists of a central area that contains general purpose laboratories designed for low-level radioactive work, a front wing that contains office space and shops, and two annexes that provide shielded enclosures with remote manipulators for high-level radiochemical work. The facility also contains



Notes: To convert feet to meters, multiply by 0.3048; GB, gloveboxes.
Source: Hoyt et al. 1999.

Figure 2–18 FMEF Layout, Minus 35-Foot Level

laboratories and specialized facilities designed for work with nonradioactive materials, microgram-to-kilogram quantities of fissionable materials, and up to megacurie quantities of radionuclides. RPL would be the primary site for fabricating the radioactive targets (i.e., targets containing radium-226 or recycled materials from previous irradiations). Total space within RPL is 13,350 square meters (143,700 square feet), of which 4,140 square meters (44,500 square feet) are occupied by general chemistry laboratories. The floor plans for the first floor and the basement of RPL are shown in **Figures 2–19** and **2–20**. A recent space utilization survey of RPL indicated that 646 square meters (6,950 square feet), representing 15.6 percent of the laboratories' area, are presently unoccupied. All of the occupied and nearly all of the unoccupied laboratories are functional and are fully equipped with standard utilities. Several of the laboratories, especially those used for radioanalytical work, have been renovated during the past few years. Upgrading and modernization of the equipment within the chemistry laboratories has been given a high priority during the past 2 years. During the space utilization survey at RPL, an assessment was made of the number of fume hoods and shielded gloveboxes (including several small hot cells) that are available in the chemistry laboratories for additional programmatic work. Of the 79 functional fume hoods and 23 shielded gloveboxes, 50 fume hoods and 15 gloveboxes are available for additional work.

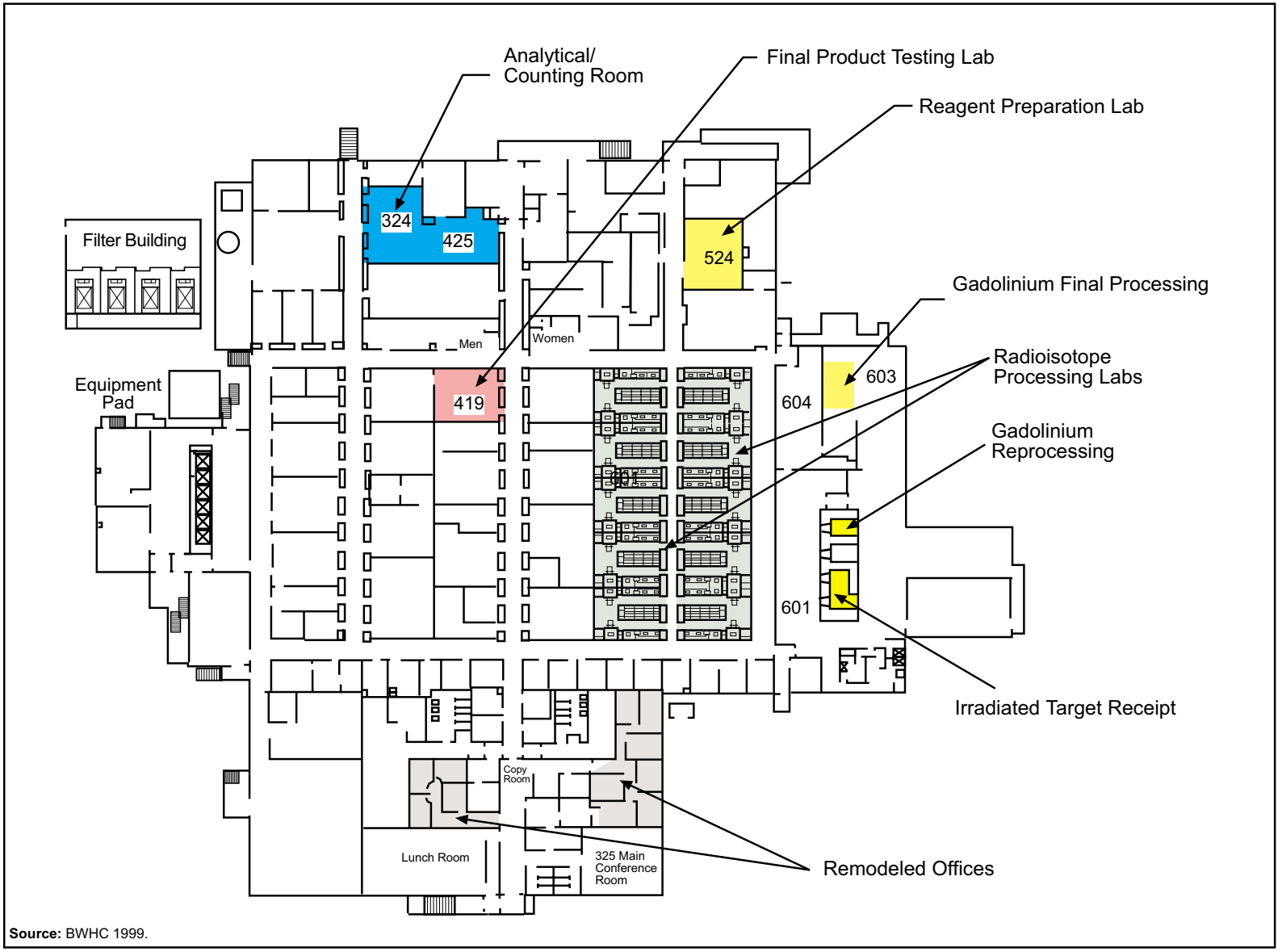
A special feature of RPL is the existence of two heavily shielded hot cell facilities located in annexes on the east and west sides of the building. These shielded facilities are the High-Level Radiochemistry Facility and the Shielded Analytical Laboratory. These two hot cell complexes are heavily used because they provide capabilities for conducting bench-scale to pilot-scale work with a wide variety of highly radioactive materials. Their capabilities include those required to conduct radiochemical separation and purification procedures, irradiated fuel or target sectioning and processing, metallography, physical properties testing of activated metals, thermal processing (including waste vitrification), and radioanalytical and preparatory chemistry operations.

The High-Level Radiochemistry Facility contains three large, interconnected hot cells designated as A-Cell, B-Cell, and C-Cell. Each of the three cells is 4.6 meters (15 feet) high and 2.1 meters (7.0 feet) deep. The A-Cell is 4.6 meters (15 feet) wide, and the B-Cell and C-Cell are each 1.8 meters (6.0 feet) wide. In-cell operations are performed using medium-duty electromechanical manipulators, and operators view their work through leaded-glass, oil-filled windows. Closed-circuit television cameras and videocassette recorders have been installed for detailed inspection work within the hot cells. The A-Cell and C-Cell also have overhead bridges that contain hoists with a 2,200-kilogram (4,840-pound) capacity. The hot cells are fully equipped with utilities and have shielded service penetrations at the front wall to allow insertion of special instruments. Each hot cell contains several process vessels located below the work deck that range in capacity from 4.0 to 320 liters (1.1 to 84.5 gallons). A large shielded door and a shielded double-door transfer port located in the rear wall of the cell provide access to each hot cell in the High-Level Radiochemistry Facility. Cask payloads weighing up to 2,200 kilograms (4,840 pounds) can be transferred into and out of the hot cells using a bridge crane located in the canyon behind the cells.

The Shielded Analytical Laboratory contains six interconnecting hot cells, each of which is 1.7 meters (5.5 feet) wide, 1.7 meters (5.5 feet) deep, and 2.9 meters (9.5 feet) high. Each hot cell is equipped with a pair of medium-duty manipulators. Turntables built into the rear walls of the hot cells provide rapid transfers of radioactive samples into and out of the cells. The Shielded Analytical Laboratory hot cells are equipped to perform a wide variety of analytical chemistry operations with highly radioactive samples.

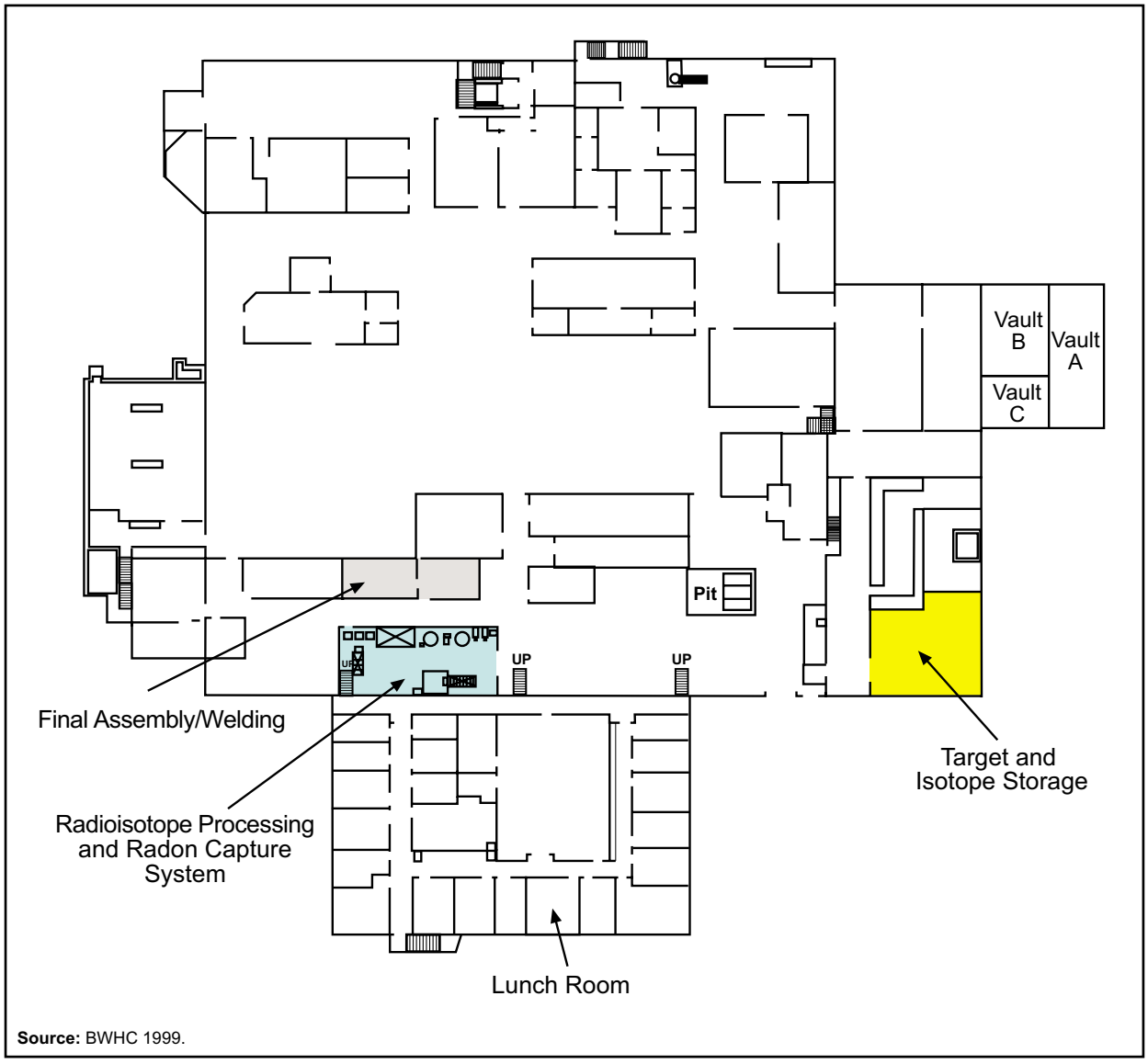
The primary features and functions of the laboratories within RPL that would be used for processing targets irradiated at FFTF are described below.

- A cluster of 10 laboratories would be available on the first floor of RPL. Each laboratory would contain a small hot cell, a shielded glovebox, and a fume hood with interconnecting transfer ports.



Source: BWHC 1999.

Figure 2–19 RPL: Proposed First Floor Locations for Hot Cell Operations and Radiochemical and Radioanalytical Laboratories for FFTF Target Processing



Source: BWHC 1999.

Figure 2–20 RPL: Proposed Basement Locations for Assembly, Processing, and Storage of FFTF Targets; Also Shown: Laboratory with Radon Gas Capture System to be Used for Processing Radium-226 Targets

- A transfer port for receiving casks containing irradiated targets into the A-Cell of the High-Level Radiochemistry Facility would be installed, and provision would be made in the C-cell for initial processing of highly radioactive targets (e.g., irradiated europium targets containing gadolinium-153 product).

- Target preparation and storage areas would be provided in the basement of RPL, in close proximity to the facilities where the radioactive and recycled targets would be assembled and welded.

- A 139.4-square-meter (1,500-square-foot) laboratory equipped with a radon gas capture system would be available in the basement of RPL to process radium-226 targets and the product isotopes generated by irradiation of these targets (all of these targets generate radon gas as intermediate products in their decay chains).

Detailed descriptions of the processes associated with medical and industrial isotope storage, target fabrication, and postirradiation processing are included in Appendix C.

2.3.2.4.2 Building 306–E

Building 306–E was constructed in 1956 as part of the nuclear material production program at Hanford and was used to develop the coextrusion process for N-Reactor fuel. Major upgrades and renovations were completed in the late 1960s and early 1970s to support the civilian reactor development program (Liquid Metal Reactor Program—FFTF). The building has 4,273 square meters (46,000 square feet) of floor space, with a 36.5-meter by 61-meter by 6.4-meter-high (120-foot by 200-foot by 21-foot-high) bay containing one 1.5-ton, one 5-ton, and three 10-ton cranes. The facility has electron beam and laser welding, certified nondestructive testing, a 3.7-meter by 3.7-meter (12-foot by 12-foot) vertical assembly and test station with a 24.4-meter (80-foot) hook height, a machine shop, and an instrument development laboratory. A description of the spaces is provided below; a view of the floor plan is provided in **Figure 2–21**.

Function	Area (square feet)
Offices	4,298
Laboratories	25,003
Shops	2,358
Conference	511
Common	14,133
Total	46,303

The building is serviced by three 1,416-cubic-meters-per-minute (50,000-cubic-feet-per-minute) supply units complete with filters, steam coils, and spray chambers. Two of the units have refrigeration coils for summer cooling. Two ceiling-mounted 1,012-cubic-meters-per-minute (35,750-cubic-feet-per-minute) recirculation fans with freon compressors provide additional cooling and air movement. Fume hoods have individual exhaust fans. Chemical and acid tanks exhaust through two 340-cubic-meters-per-minute (12,000-cubic-feet-per-minute) fume scrubbers to a 12.2-meter-high, 7.6-centimeter-diameter (40-foot-high, 3-inch-diameter) stainless steel exhaust stack. Equipment exhaust collects through a grid that leads to two 566-cubic-meters-per-minute (20,000-cubic-feet-per-minute) exhaust fans. Plastic hoods and duct work are provided for highly corrosive service.

Major equipment includes three industrial x-ray machines, a 6-kilowatt Hamilton Standard electron beam welder, five open face hoods, two inert gas welding chambers, and one electrolytic cutoff saw. Utilities include hot and cold water, deionized water, propane, helium, compressed air, argon, steam, and sanitary and process sewers as well as a special acid drain and neutralizing tank. Normal power is provided by a 1500-kilovolts ampere transformer with 150-kilovolts ampere backup power from an adjoining building and a 30-kilovolts ampere emergency transformer. The building is protected by redundant emergency alarm systems, fire gongs, and an evacuation siren.

2.3.2.5 New Support Facility

A new generic support facility would have the mission of preparing medical and industrial isotope targets for irradiation, processing irradiated targets, and housing the materials research and development activities in association with Alternatives 3 and 4. Siting of the generic support facility for medical and industrial isotope production would require the facility to be located in the same general vicinity (0.2 to 20 kilometers [0.12 to 12.4 miles]) as the new irradiation facility (accelerator or reactor). Colocation with the irradiation facility

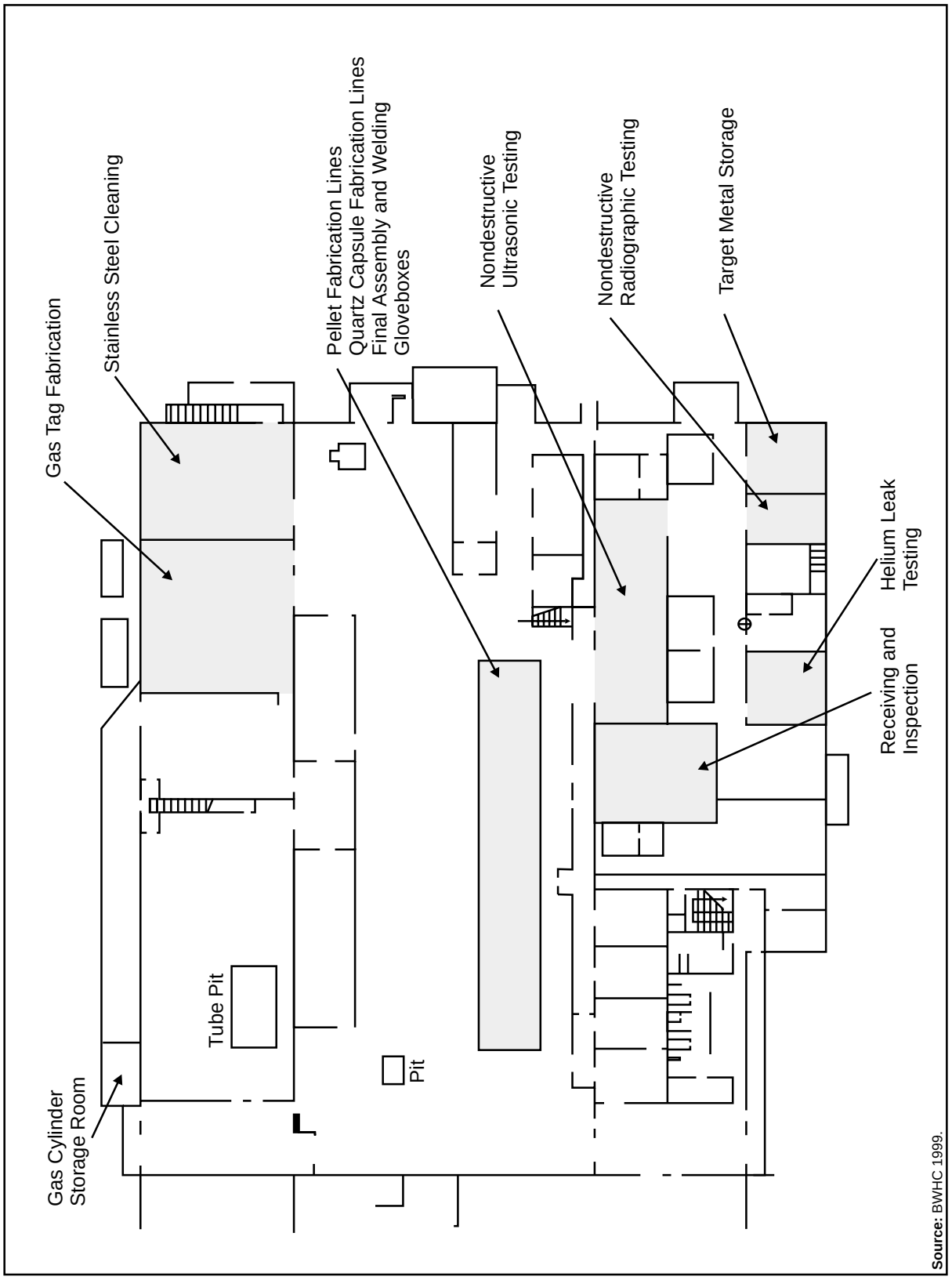


Figure 2–21 Building 306–E Floor Plan

would be needed to process some irradiated target materials promptly after removal from the reactor/accelerator because some isotopes have short half-lives. Colocation would also minimize transportation time. Although the facility could be located within the irradiation facility security protection area, the lack of a defense mission and fissile material in the generic support facility indicates that a high level of physical protection would not be warranted.

The generic support facility mission would be accommodated by a one-story, 3,345-square-meter (36,000-square-foot) above-grade building with a 1,490-square-meter (16,000-square-foot) basement area under a portion of the footprint (SAIC 2000). **Figure 2–22** provides the general layout of the building. The relative position and size of the processing, support, office, and research and development areas are shown on the diagram.

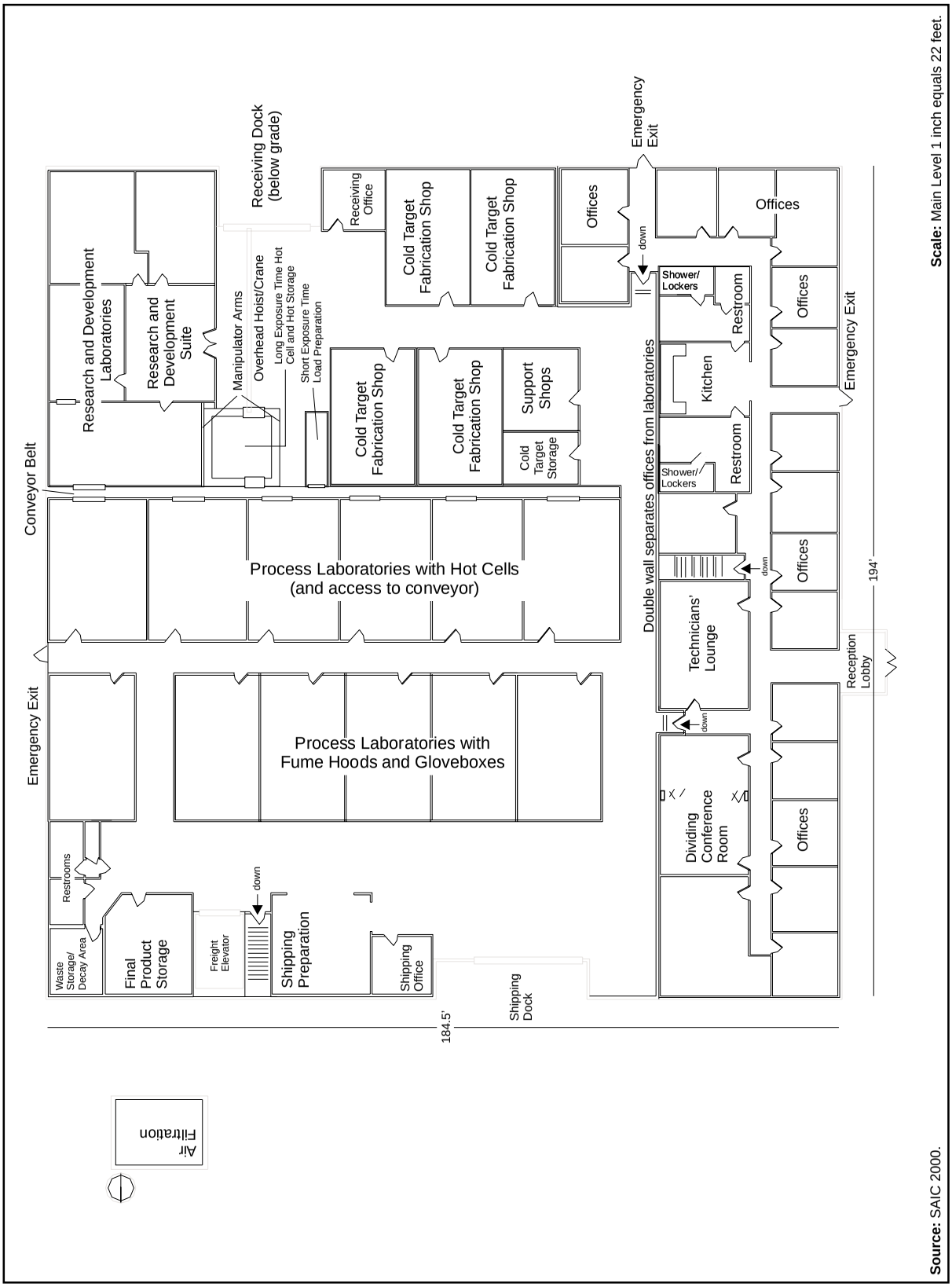
The facility is designed around a center area containing the highest-risk activities and the material inventories requiring the highest level of engineered controls. As can be seen in Figure 2–22, irradiated materials in casks or other shielded transport containers would enter a loading dock with a straight-line access to the primary facility hot cell. The hot sample entry area would be a high bay area with a high floor loading area between the loading dock and the hot cell access port. This configuration would allow transport cask access to the hot cell. In addition, an overhead hoist would be available to facilitate handling of materials and devices in the proximity of the hot cell.

The hot cell would accept high-radiation-level samples or those difficult to shield or manipulate (e.g., reactor core components containing samples). The hot cell would have access to a conveyor that can remotely transport samples to the hot process laboratories. In addition, samples from the hot cell could be transferred to the hot research and development laboratory gloveboxes for detailed analysis and testing. Hot cell manipulators would be located on both the operating gallery and the research and development sides of the hot cell. Adjacent to that would be the central receiving station for all other radioactive and short-exposure samples not contained in the reactor core components. This area, while not a hot cell, would provide personnel protection (i.e., shielding and controlled ventilation) for preliminary sample preparation and examination. It would also provide interim irradiated sample storage prior to delivery to the designated processing laboratory. When needed, samples would be transported remotely to the processing laboratories by the conveyor system. Samples requiring a lesser degree of control would be distributed for processing throughout the remaining process laboratory wing. After processing, the radiopharmaceuticals would be either stored or packaged and shipped immediately to offsite vendors.

Radioactive waste would be packaged and stored for eventual disposal. Those materials containing short-lived isotopes would be delivered to a decay/holding room so that, given appropriate decay time, they could be disposed of without a radioactive component. The process and research and development areas would be considered radiologically controlled areas, but no routinely occupied areas would require control as contaminated radiological areas. Radioactive contamination would be controlled at the hood or glovebox face. Due to this configuration, protective clothing and change rooms would be needed only for occasional maintenance activities when temporary radiological areas are established.

Cold sample (nonradioactive) preparation would be accomplished in a set of three large laboratories where radiological conditions are not anticipated. Completed samples would be stored in an adjacent room along with raw sample materials (nonradioactive).

Radioactive sample preparation and irradiated material recycling activities would be conducted in one of the laboratories adjacent to the conveyor.



Source: SAIC 2000.

Scale: Main Level 1 inch equals 22 feet.

Figure 2–22 Support Facility Layout

Irradiated research and development samples introduced into the hot cell could be processed or examined using manipulators within the hot cell. Samples could also enter the research and development suite of lab rooms through the hot cell port into a hot cell or glovebox. From there, they could be moved to additional research and development laboratory rooms within a controlled environment for detailed analysis and testing.

Support areas would include ventilation, maintenance, change rooms, quality assurance and quality control, lunch and break rooms, storage, conference rooms, basement stairwells, equipment elevators, and utility distribution. A small machine shop would accommodate light machining activities, but would not be intended to involve radioactive materials. A portion of the support functions, especially utilities, would be located in a basement portion of the building. The basement would be located under the half of the building that would not experience high weight loads (i.e., hot cell, cask receiving area).

Solid waste would be collected, packaged, and stored at a central location. Liquid waste would be processed at the point of generation (e.g., hood/glovebox) or would be collected in a retention tank for characterization and eventual transfer to the effluent treatment facility.

2.4 DESCRIPTION OF TRANSPORTATION ACTIVITIES

2.4.1 Purchase of Plutonium-238 from Russia

Under the No Action Alternative (see Section 2.5.1), DOE would continue exercising its option to purchase Russian plutonium-238 (if available) to meet the needs of future U.S. space exploration missions. In 1992, DOE signed a contract permitting the purchase from Russia of up to 40 kilograms (88.2 pounds) of plutonium-238. To date, DOE has purchased 9 kilograms (19.8 pounds). In 1997, DOE extended the contract for another 5 years, so this option remains viable. It is unclear, however, whether this option would remain reliable or viable once the existing contract expires (DOE 1997). The impacts associated with the purchase of plutonium-238 from Russia are discussed in Section 4.2.1.1.

2.4.2 Transportation of Plutonium-238 from St. Petersburg, Russia, to the Los Alamos National Laboratory

Plutonium-238 purchased from Russia would have to be transported from St. Petersburg to a U.S. port of entry, and from there to LANL where it would be used in the fabrication of radioisotope power systems and heating units. The impacts of the transportation of a total of 40 kilograms (88.2 pounds) of plutonium-238 are estimated in the *Environmental Assessment of the Import of Russian Plutonium-238* (DOE 1993) and are summarized in Section 4.2.1.1 of this NI PEIS. The impacts associated with transporting 175 kilograms (385 pounds) (5 kilograms [11 pounds] per year for the 35-year evaluation period) of plutonium-238 have been determined by extrapolation and are included in the same section.

2.4.3 Transportation of Neptunium-237 from Savannah River Site to Candidate Storage Facilities

Under the No Action Alternative (see Section 2.5.1) DOE would transport neptunium-237 oxide from SRS to a storage facility off site. Storage canisters containing the neptunium-237 oxide would be loaded into approved shielded shipping containers or casks at SRS and shipped to the designated storage facilities for long-term storage.

Truck transportation of neptunium-237 from SRS to the proposed storage facilities is assumed in this NI PEIS. The neptunium-237 would be transported in robust Type B transportation casks. Type B casks are used to transport nuclear materials with the highest radioactivity levels, and are designed to protect and retain their contents under transportation accident conditions. According to DOE policy, which requires compliance with

applicable Federal regulations regarding domestic shipments of radioactive materials, transportation of neptunium-237 in Type B casks would comply with the requirements of 10 CFR Part 71, "Packaging and Transportation of Radioactive Materials," and 49 CFR Part 173, "Shippers - General Requirements for Shipments and Packagings."

The container that would be used to transport neptunium-237 has not been proposed, but would be a Type B container similar to the Chalfont container 9975. The 9975 container includes a 132-liter (35-gallon) drum, insulation, a primary containment vessel, a secondary containment vessel, lead shielding, and aluminum honeycomb spacers. The neptunium-237 would be sealed into a can, which would be placed on a honeycomb spacer inside the stainless steel primary containment vessel. The primary containment vessel would be bolted closed and placed into a similarly constructed, but larger, secondary containment vessel. The secondary containment vessel would be bolted closed and loaded into a drum equipped with lead shielding to reduce radiation levels and fireboard insulation to protect the containment vessels in the unlikely event of a severe impact. A description of the Chalfont container 9975 is provided in Appendix J.

DOE anticipates that neptunium-237 would be transported through use of the Transportation Safeguards System and shipped using SST/SGTs. The SST/SGT, a fundamental component of the Transportation Safeguards System, is a specially designed component of an 18-wheel tractor-trailer vehicle. Although details of vehicle enhancements and some operational aspects are classified, key characteristics of the SST/SGT system include the following:

- Enhanced structural characteristics and a highly reliable tie-down system to protect cargo from impact
- Heightened thermal resistance to protect the cargo in case of fire (newer SST/SGT models)
- Established operational and emergency plans and procedures governing the shipment of nuclear materials
- Various deterrents to prevent unauthorized removal of cargo
- An armored tractor component that provides courier protection against attack and contains advanced communications equipment
- Specially designed escort vehicles containing advanced communications and additional couriers
- 24-hour-a-day real-time communications to monitor the location and status of all SST/SGT shipments via DOE's Security Communication system
- Couriers, who are armed Federal officers, receive rigorous specialized training and are closely monitored through DOE's Personnel Assurance Program
- Significantly more stringent maintenance standards than those for commercial transport equipment
- Conduct of periodic appraisals of the Transportation Safeguards System operations by the DOE Office of Defense Programs to ensure compliance with DOE orders and management directives, and continuous improvement in transportation and emergency management programs

Additional details are presented in Appendix J.

2.4.4 Transportation of Mixed Oxide Fuel from Europe to the Fast Flux Test Facility

As discussed in Section 2.3.1.1.3, a 15-year supply of mixed oxide fuel may be available from Germany to operate FFTF. Approximately 205 mixed oxide fuel assemblies were fabricated in Europe for use in Germany's SNR–300 sodium-cooled, breeder reactor before the German government suspended the reactor's operation. SNR–300 mixed oxide fuel is very similar in both composition and construction to FFTF fuel. The 205 SNR–300 mixed oxide fuel assemblies, if reconfigured for FFTF, could be used to fabricate about 150 to 160 FFTF fuel assemblies. This amount could supply two FFTF core loads for approximately 15 years of FFTF operation at the 100 megawatts thermal power level with occasional excursions to the 400 megawatts thermal power level on an as-needed basis (as proposed for this mission).

The inventory of unused SNR–300 mixed oxide fuel is now stored at Hanau, Germany, and Dounreay, Scotland. If a decision were made to use SNR–300 fuel in FFTF, security measures would be implemented to prevent unauthorized removal of the mixed oxide fuel during transportation to the United States. The requirements to ensure the safety and security of transatlantic mixed oxide fuel shipments are listed in: *The Convention on the Physical Protection of Nuclear Material*, International Atomic Energy Agency publication INFCIRC 274 (IAEA 1997); *The Physical Protection of Nuclear Material*, International Atomic Energy Agency publication INFCIRC 225 (IAEA 1999); the *Code for the Safe Carriage of Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes in Flasks on Board Ships* (IMO 1993); DOE orders; and 10 CFR Part 73. DOE estimates that as many as 11 shipments from Europe would be required. The initial shipment would transport the FFTF mixed oxide fuel lead test assembly, and the following 5 to 10 shipments would transport the SNR–300 fuel assemblies reconfigured for FFTF use.

SNR–300 mixed oxide fuel could be brought into many U.S. commercial and military ports. A port-selection process was used by DOE in its *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (Foreign Research Reactor Spent Nuclear Fuel EIS)* (DOE 1996). The criteria used for screening ports in the *Foreign Research Reactor Spent Nuclear Fuel EIS* were: (1) appropriate port experience; (2) safe port transit to open ocean; (3) appropriate port facilities for safe receipt, handling, and transshipment; (4) ready intermodal access; and (5) low human population of the ports and along transportation routes. DOE used these same criteria to identify ports for receiving mixed oxide fuel from Europe. The application of these criteria for mixed oxide fuel is discussed in Section J.3.6.1.

In the *Foreign Research Reactor Spent Nuclear Fuel EIS* Record of Decision, DOE decided to use military ports to take advantage of their capabilities to increase the safety and security of the spent fuel transportation process. DOE concluded that the use of military ports provides additional confidence in the safety of shipments due to the increased security. Since the security issues are far greater for fresh mixed oxide fuel than for spent nuclear fuel because of the potential for proliferation, DOE would use a military port to bring the SNR–300 mixed oxide fuel into the country.

Based on a review of the *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996), the following military ports were considered:

- In the eastern United States: Charleston Naval Weapons Station, South Carolina; Military Ocean Terminal Sunny Point, North Carolina; Mayport, Florida; Kings Bay, Georgia; Pensacola, Florida; Yorktown, Virginia; and Hampton Roads, Virginia

- In the western United States: Military Ocean Terminal Bay Area, California; Bremerton, Washington; Everett, Washington; Port Hueneme, California; and Port Townsend, Washington

Other military ports that could be considered in a formal port selection process include, but are not limited to: Naval Weapons Station Seal Beach, California; Naval Submarine Base, North Island, California; Naval Amphibious Base Coronado, California; Naval Station San Diego, California; Naval Station Ingleside, Texas; Naval Station Pascagola, Mississippi; Naval Air Station Pensacola, Florida; Naval Weapons Station Earle, New Jersey; and Naval Submarine Base New London, Connecticut.

The overland transportation impacts would be higher if mixed oxide fuel were accepted at an east coast port rather than a west coast port. However, accepting mixed oxide fuel at an east coast port would reduce time and eliminate the potential security risk when transiting the Panama Canal. Charleston Naval Weapons Station, an east coast port, was used for the purpose of transportation impact analysis (see Section J.6.2). If Alternative 1 (Restart FFTF) were selected for implementation, and if it were decided to import mixed oxide fuel, DOE would conduct an appropriate NEPA review prior to importation of mixed oxide fuel. The analysis would consider a variety of ports on the east and west coasts. It also would consider the potential impacts of traversing both global commons (i.e., portions of the ocean not within the territorial boundary of any nation), in accordance with Executive Order 12114 (44 FR 1957), and inland waters (such as Puget Sound), as well as shipboard fires, package handling, truck transportation, and logistical and safeguard and security concerns associated with transporting mixed oxide fuel.

Use of a military port to receive SNR–300 mixed oxide fuel would require shipment via chartered ships. The ships that would be used to transport SNR–300 mixed oxide fuel to the United States would be of the type used to transport spent nuclear fuel or mixed oxide fuel internationally. These specially equipped ships are called purpose-built vessels.

Purpose-built vessels, as used in this NI PEIS, are those vessels specifically designed to transport nuclear fuel casks. These vessels operate as dedicated vessels and, therefore, are not used to transport any other cargoes. Casks are loaded directly into the holds of the vessels because the cargo compartments contain hardware that mates to the tie-down fixtures on the casks. If a vessel has no crane, dockside cranes are used for loading and unloading. The cargo compartments are typically intended to handle a specific type of cask, and other cask types cannot be used without making modifications to the tie-down hardware.

The purpose-built vessels are equipped with double bottoms and hulls, watertight compartments, special firefighting systems, and collision-damage-resisting structures within the main hull, as well as special security features and satellite tracking systems. The crew is trained in appropriate cargo-handling techniques and in emergency response.

At present, purpose-built vessels are operated by Pacific Nuclear Transport Services of Japan, by British Nuclear Fuels, Limited, and by the Swedish Nuclear Fuel and Waste Management Company. They are used to move nuclear fuel between operating nuclear power plants and nuclear fuel reprocessing facilities operated by Cogema and British Nuclear Fuels, Limited, or in Sweden's case, the repository in Forsmark. Beginning in 1998, purpose-built vessels have transported spent nuclear fuel from foreign research reactors to the Charleston Naval Weapons Station. Additionally, in 1999 purpose-built vessels delivered mixed oxide fuel from the United Kingdom to Japan. There are no U.S.-owned purpose-built vessels for nuclear fuel transport.

DOE anticipates that the SNR–300 mixed oxide fuel would be transported overland from the seaport to FFTF using the same Transportation Safeguards System that would be required for the transportation of neptunium-237 (see Section 2.4.3).

The environmental impacts associated with importing SNR–300 mixed oxide fuel from Europe are discussed in Section 4.3 and Appendix J.

2.4.5 Transportation of Neptunium-237 from Savannah River Site to Target Fabrication Facilities

The neptunium-237 required for target fabrication in the production of plutonium-238 is currently stored at SRS. Therefore, transportation of neptunium-237 oxide from SRS to target fabrication facilities off site (REDC, FDPF, FMEF) would be required. Storage canisters containing the neptunium-237 would be loaded into approved shielded shipping containers or casks at SRS and shipped to the target fabrication facilities. Transportation of neptunium-237 from SRS to these facilities for long-time storage is discussed in Section 2.4.3.

2.4.6 Transportation of Nonirradiated and Irradiated Targets

Transportation of nonirradiated neptunium-237 targets from the target fabrication facility to the irradiation facility and irradiated targets from the irradiation facility back to the fabrication facility for processing would use Type B casks certified for the safe shipments of the neptunium-237 targets. The casks used would be similar in size and construction to a spent nuclear fuel cask. Nonirradiated neptunium-237 targets would be transported in SST/SGTs. Irradiated neptunium-237 targets could be transported in commercial trucks. The analysis in this NI PEIS assumes transportation in commercial trucks to maximize the environmental impacts. The cask selection and environmental issues are discussed in Appendix J.

2.4.7 Transportation of Plutonium-238 Product to the Los Alamos National Laboratory

After postirradiation processing at the target processing facility, the plutonium-238 product in oxide form would be packaged and shipped to LANL. The 5320 package, designed for surface transportation of americium or plutonium, would be used to carry plutonium oxide to LANL. The 5320 package is a dome-topped upright cylinder that is mounted on a baseplate supported by casters. As explained in Appendix J, Section J.3.3.3, the plutonium-238 would be loaded into an EP–60 product canister, a stainless steel shell confinement vessel used to load the product into the package safely and conveniently. The EP–60 would be seal-welded into the removable stainless steel shell primary containment vessel, the EP–61. The EP–61 would be placed into the secondary containment vessel, the EP–62. The stainless steel EP–62 has a removable bolted closure lid. The gasketed flange of the EP–62 satisfies the containment requirements for both normal transport and hypothetical accident conditions. Plutonium oxide would be transported using the Transportation Safeguards System and would be shipped using SST/SGT (see Section 2.4.3).

2.4.8 Transportation of Materials for Medical Isotope Production

The raw material for target fabrication would typically be acquired from ORNL, where enrichment processes are conducted to produce high purity target material suitable for production of medical isotopes. The raw material would be shipped from ORNL to Hanford or to the new generic support facility at an existing but undefined DOE site.

Transportation of materials for medical and industrial isotope production and research and development would take place at Hanford between FFTF and the Hanford RPL/306–E facilities or FMEF. At the existing DOE site, transportation would take place between the new generic support facility and either the new low-energy accelerator or the new research reactor.

At Hanford, two different target irradiation vehicle assemblies would be used—the Long-Term Irradiation Vehicle Assembly (up to 3.7 meters [12 feet] in length) and the Rapid Radioisotope Retrieval System Target Carrier (less than 30 centimeters [1 foot] in length). Irradiated pins or short target carriers would be shipped from the irradiation facility to the processing facility using a Type B, accident-resistant shipping cask. The elements (or pins) for the Long-Term Target Irradiation Vehicle Assemblies would be segmented in the FFTF

Interim Examination and Maintenance cell, if necessary, or could be inserted directly into the shipping cask. The Rapid Retrieval System Target Vehicle Assemblies would be inserted into a smaller "shielded pig" package, which would be inserted into the shipping cask or, ideally and as a design goal, the irradiated target carriers would be loaded directly into the shipping cask from the reactor.

At the existing DOE site, irradiated targets and research and development material would be transported in a "shielded pig" package from either the new low-energy accelerator or the new research reactor to the new generic support facility.

A variety of casks would be used to ship the separated isotopes from the processing facility to the destination (i.e., the pharmaceutical distributor). Some land and air shipments would use DOT-specified casks such as CI-20WC-2, and others would require larger Type B casks.

An existing licensed irradiated fuel shipping cask (the T-3) is available to transport material used for research and development. This cask can accommodate shipments of pins or FFTF fuel assemblies as well as nonfuel experiments and materials.

2.5 DESCRIPTION OF ALTERNATIVES

A perspective on the programmatic activities associated with the options under each alternative is presented in **Table 2–3**. Individual alternatives are described in the following sections. The environmental impacts associated with each alternative and its options are discussed in Chapter 4 and are summarized in Section 2.7.1.

Table 2–3 Alternatives and Options Matrix

Activity	Irradiation Facility and Site	Target Fabrication and Processing Facility Site	No Action Alternative Options				Alternative 1[a] Options						Alternative 2[b] Options									Alternative 3[c] Options			Alternative 4[d] Options			Alternative 5[e]
			1	2	3	4	1	2	3	4	5	6	1	2	3	4	5	6	7	8	9	1	2	3	1	2	3	
Store neptunium-237 as oxide		REDC at ORNL		●																								
		CPP–651 at INEEL			●																							
		FMEF at Hanford				●																						
Purchase plutonium-238			●	●	●	●																						
Irradiate targets for plutonium-238 production	FFTF at Hanford: mixed oxide fuel: 21 years; highly enriched uranium: 14 years						●	●	●																			
	FFTF at Hanford: mixed oxide fuel: 6 years; highly enriched uranium: 29 years									●	●	●																
	ATR at INEEL												●	●	●													
	CLWR (generic site)															●	●	●										
	ATR at INEEL plus HFIR at ORNL																		●	●	●							
	New high-energy accelerator at generic DOE site																					●	●	●				
	New research reactor (generic DOE site)																								●	●	●	
Store neptunium-237 and fabricate and process targets for plutonium-238 production		REDC at ORNL					●			●			●			●			●			●			●			
		FDPF and CPP–651[f] at INEEL						●			●			●			●			●			●			●		
		FMEF at Hanford							●			●			●			●			●			●			●	
Irradiate targets for medical and industrial isotope production and perform research and development activities	FFTF at Hanford: mixed oxide fuel: 21 years; highly enriched uranium: 14 years						●	●	●																			
	FFTF at Hanford: mixed oxide fuel: 6 years; highly enriched uranium: 29 years									●	●	●																
	New low-energy accelerator at generic DOE site																					●	●	●				
	New research reactor at generic DOE site																								●	●	●	
Fabricate and process targets for medical and industrial isotope production and perform research and development activities		FMEF at Hanford							●			●																
		RPL/306–E					●	●		●	●																	
		New facility (generic DOE site)																				●	●	●	●	●	●	
Maintain FFTF in standby status	FFTF at Hanford		●	●	●	●																						
Deactivate FFTF	FFTF at Hanford												●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●

a. Alternative1, Restart FFTF.
b. Alternative 2, Use Only Existing Operational Facilities.
c. Alternative 3, Construct New Accelerator(s).
d. Alternative 4, Construct New Research Reactor.
e. Alternative 5, Permanently Deactivate FFTF (with No New Missions).
f. CPP–651 would be used only for storage.

2.5.1 No Action Alternative

Under the No Action Alternative (maintain status quo), FFTF would be maintained in standby status for all or a portion of the 35-year evaluation period for operations covered in this NI PEIS. For the purpose of analysis in this NI PEIS, the maximum period of 35 years was assumed. Ongoing operations at existing facilities (as described in Chapter 3, Affected Environment) would continue under this alternative. DOE would not establish a domestic plutonium-238 production capability, but could instead continue to purchase Russian plutonium-238 to meet the needs of future U.S. space missions. For the purpose of analysis in this NI PEIS, DOE assumed that it would continue to purchase plutonium-238 to meet the space mission needs for the 35-year evaluation period. DOE recognizes, however, that any purchase beyond what is currently available to the United States through the existing contract will require additional NEPA review. DOE would continue its medical and industrial isotope production and civilian nuclear energy research and development activities at the current operating levels of existing facilities. A consequence of a No Action decision would be the need to determine the future of the neptunium-237 stored at SRS. Therefore, the impacts of possible future transportation and storage of neptunium-237 are evaluated as part of the No Action Alternative. Four options are identified. If DOE decides not to establish a domestic plutonium-238 production capability in the future, then neptunium-237 would have no programmatic value and Option 1 would be selected. Conversely, if DOE decides to maintain the capability to establish a domestic plutonium-238 capability in the future, the inventory of neptunium-237 must be retained. In this case, Option 2, 3, or 4 could be selected.

- **Option 1.** Under this option, DOE would follow its current stabilization strategy for the neptunium-237, currently stored in solution form at SRS. The current plan is to stabilize the material to oxide, as described in the Supplemental Record of Decision for the *Final Environmental Impact Statement, Interim Management of Nuclear Materials at SRS* (DOE 1995c; 62 FR 61099). This Record of Decision would be amended or new NEPA analysis would be performed, if necessary.

- **Options 2 through 4.** Under these options, the neptunium-237 oxide would be transported from SRS to one of three candidate DOE sites for up to 35 years of storage. For the purpose of analysis in this NI PEIS, the maximum period of 35 years was assumed. Option 2 would provide storage at ORNL's REDC facility, Option 3 at INEEL's Building CPP–651, and Option 4 at Hanford's FMEF.

2.5.2 Alternative 1—Restart FFTF

Under Alternative 1, FFTF at Hanford would be restarted and operated for the 35-year evaluation period. FFTF would be used to irradiate targets for medical and industrial isotopes production, plutonium-238 production, and civilian nuclear energy research and development irradiation requirements. Ongoing operations at existing facilities (as described in Chapter 3, Affected Environment) would continue.

Targets for medical and industrial isotope production would be fabricated in one or more facilities at Hanford. Target material would typically be acquired from ORNL, where enrichment processes are conducted to produce high-purity target material suitable for production of medical isotopes. The targets would be irradiated at FFTF and then returned to the fabrication facility for postirradiation processing. From there, the isotope products would be sent directly to commercial pharmaceutical distributors.

Targets for plutonium-238 production would be fabricated in one of three candidate facilities at ORNL, INEEL, or Hanford. The material needed for target fabrication (neptunium-237) would be transported from SRS. The nonirradiated targets would be transported and irradiated at FFTF and then transported back to the fabricating facilities for postirradiation processing. The separated plutonium-238 would be transported to LANL for fabrication into radioisotope power systems and heating units.

Under Alternative 1, raw materials, nonirradiated targets, irradiated targets, and processed materials would be transported among the locations selected for raw target material acquisition, material storage, target fabrication, target irradiation, and postirradiation processing, and eventually to the final destinations for the medical and industrial isotopes and the plutonium-238 product or various research and development test sites.

FFTF could produce high-energy neutrons and a large flux level (10^{15} neutrons per square centimeter per second) that can be tailored to nearly any desired energy level. FFTF would provide the greatest flexibility for both isotope production and nuclear-based research and development among the baseline configurations for all of the proposed alternatives. Due to its large core size, flux spectrum, demonstrated testing capability, and rated power level, it would be able to concurrently support the projected plutonium-238 production needs, production of medical and industrial isotopes, and civilian nuclear energy research and development related to a broad range of materials, advanced reactors, advanced fuels, and waste transmutation.

The six options under this alternative are associated with the type of nuclear fuel to be used for FFTF operations and the specific facilities to be used for target fabrication and processing. The first three options (Options 1 through 3) would involve operating FFTF with a mixed oxide fuel core for the first 21 years and a highly enriched uranium fuel core for the remaining 14 years. The last three options (Options 4 through 6) would involve operating FFTF with a mixed oxide fuel core for the first 6 years and a highly enriched uranium fuel core for the remaining 29 years. FFTF can provide similar irradiation services with either a mixed oxide fuel core or a highly enriched uranium fuel core. The reasons for these options in FFTF core fuel are provided in Section 2.3.1.1.3. Potential impacts from the deactivation of FFTF at the end of its operating life are not explicitly covered under this alternative, but are addressed under Alternative 5 (Section 2.5.6).

The options involving storage, fabrication, postirradiation processing, and transportation are discussed below.

- **Options 1 and 4.** REDC at ORNL would be used to fabricate and process the neptunium-237 targets required for plutonium-238 production. The neptunium-237 transported from SRS to ORNL would be stored in REDC. The plutonium-238 product would be transported from ORNL to LANL. Hanford's RPL/306–E facilities would be used to fabricate and process targets for medical and industrial isotope production and for research and development, as well as to store the materials needed to fabricate these targets.

- **Options 2 and 5.** FDPF at INEEL would be used to fabricate and process the neptunium-237 targets for plutonium-238 production. The neptunium-237 transported from SRS to INEEL would be stored in FDPF or Building CPP–651 at INEEL. The plutonium-238 product would be transported from INEEL to LANL. Hanford's RPL/306–E facilities would be used to fabricate and process targets for medical and industrial isotope production and for research and development, as well as to store the materials needed to fabricate these targets.

- **Options 3 and 6.** FMEF at Hanford would be used to fabricate and process both neptunium-237 targets for plutonium-238 production and targets for the production of medical and industrial isotopes. The neptunium-237 transported from SRS to Hanford and the other target materials transported from other offsite facilities to Hanford would be stored in FMEF. The plutonium-238 product would be transported from Hanford to LANL for fabrication into heat sources for radioisotope power systems.

2.5.3 Alternative 2—Use Only Existing Operational Facilities

Under Alternative 2, DOE would use existing operating DOE reactors or U.S. commercial nuclear power plants to produce plutonium-238 for future space missions. The production of medical and industrial isotopes

and support of civilian nuclear energy research and development in DOE reactors and accelerators would continue at the No Action Alternative levels.

The currently operating DOE reactors, HFIR and ATR, cannot fully meet the projected long-term needs for medical isotope production and civilian nuclear energy research and development with or without adding the plutonium-238 production mission. Depending on the combination of facilities used in Alternative 2, HFIR and ATR could continue their current support of the medical and industrial isotope and research and development missions, including some near-term growth, while accommodating the production of plutonium-238. Under other scenarios, some near-term growth in medical and industrial isotope production and civilian nuclear energy research and development could be limited by the addition of plutonium-238 production. In any case, non-DOE use of these facilities would be affected by the addition of the plutonium-238 mission. If a commercial reactor were used for plutonium-238 production, the DOE facilities would be unaffected and would continue operating as discussed under the No Action Alternative.

Another component of Alternative 2 is permanent deactivation of FFTF. Permanent deactivation of FFTF (Alternative 5) would occur in conjunction with any of the options under Alternatives 2, 3, or 4. Ongoing operations at existing facilities (as described in Chapter 3, Affected Environment) would continue under Alternative 2.

Targets for plutonium-238 production would be fabricated in one of three facilities at ORNL, INEEL, or Hanford. The material needed for target fabrication (neptunium-237) would be processed and transported from SRS to the fabrication facilities. The targets would be irradiated at existing reactor facilities (HFIR, ATR, a CLWR, as described in Section 2.3.1) and would be transported back to the fabricating facilities for postirradiation processing.

Under Alternative 2, nonirradiated targets, irradiated targets, and processed materials would be transported among the locations selected for storage, target fabrication, target irradiation, and postirradiation processing, and the plutonium-238 product would be transported to LANL.

Nine options are proposed under this alternative. Options 1 through 3 involve the irradiation of targets in ATR at INEEL. Options 4 through 6 involve the irradiation of targets in a generic CLWR. Options 7 through 9 involve the irradiation of targets in both INEEL's ATR and ORNL's HFIR. These options and the associated target fabrication, postirradiation processing, and transportation activities are discussed below.

- **Option 1.** REDC at ORNL would be used to store the neptunium-237 transported from SRS to ORNL and to fabricate and process the targets irradiated at ATR. Option 1 also would involve transportation of the neptunium-237 targets from ORNL to INEEL for irradiation in ATR, transportation of the irradiated targets from INEEL back to ORNL for postirradiation processing, and subsequent transportation of the plutonium-238 product from ORNL to LANL following postirradiation processing.

- **Option 2.** FDPF at INEEL would be used to store the neptunium transported from SRS to INEEL and to fabricate and process the targets (irradiated at ATR). Building CPP–651 would be used for storage. Option 2 also would involve transportation of the plutonium-238 product from INEEL to LANL following postirradiation processing.

- **Option 3.** FMEF at Hanford would be used to fabricate and process the targets (irradiated at ATR) and to store the neptunium-237 transported from SRS to Hanford. Option 3 also would involve transportation of the neptunium-237 to Hanford for target fabrication, transportation of the targets

from Hanford to INEEL for irradiation, transportation of the irradiated targets back to Hanford for postirradiation processing in FMEF, and subsequent transportation of the plutonium-238 product from Hanford to LANL.

- **Option 4.** REDC at ORNL would be used to store the neptunium-237 transported from SRS to ORNL and to fabricate and process the targets (irradiated at a generic CLWR). Option 4 also would involve transportation of the neptunium-237 targets from ORNL to the generic CLWR location for irradiation, transportation of the irradiated targets back to ORNL for postirradiation processing, and transportation of the plutonium-238 product from ORNL to LANL.

- **Option 5.** FDPF at INEEL would be used to store the neptunium transported from SRS to INEEL and to fabricate and process the targets (irradiated at a generic CLWR). Building CPP–651 would also be used for storage. In addition, Option 5 would involve transportation of the neptunium-237 targets from INEEL to the generic CLWR location for irradiation, transportation of the irradiated targets back to INEEL for postirradiation processing, and transportation of the plutonium-238 product from INEEL to LANL.

- **Option 6.** FMEF at Hanford would be used to store the neptunium-237 transported from SRS to Hanford and to fabricate and process the targets (irradiated at a generic CLWR). Option 6 also would involve transportation of neptunium-237 to Hanford for target fabrication, transportation of the targets from Hanford to the generic CLWR location for irradiation, transportation of the irradiated targets back to Hanford for postirradiation processing, and transportation of the plutonium-238 product from Hanford to LANL.

- **Option 7.** REDC at ORNL would be used to store the neptunium-237 transported from SRS to ORNL and to fabricate and process the targets (irradiated at ATR and HFIR). Option 7 also would involve transportation of the neptunium-237 targets from ORNL to the reactors for irradiation, transportation of the irradiated targets back to ORNL for processing, and transportation of the plutonium-238 product from ORNL to LANL.

- **Option 8.** FDPF at INEEL would be used to store the neptunium transported from SRS to INEEL and to fabricate and process the targets (irradiated at ATR and HFIR). Building CPP–651 would be used for storage. Option 8 also would involve transportation of the neptunium-237 targets from INEEL to the reactors for irradiation, transportation of the irradiated targets back to INEEL for postirradiation processing, and transportation of the plutonium-238 product from INEEL to LANL.

- **Option 9.** FMEF at Hanford would be used to store the neptunium-237 transported from SRS to Hanford and to fabricate and process the targets (irradiated at ATR and HFIR). Option 9 also would involve transportation of neptunium-237 to Hanford for target fabrication, transportation of the targets from Hanford to the reactors for irradiation, transportation of the irradiated targets back to Hanford for postirradiation processing, and transportation of the plutonium-238 product from Hanford to LANL.

2.5.4 Alternative 3—Construct New Accelerator(s)

Under Alternative 3, one or two new accelerators would be used for target irradiation for the evaluation period of 35 years. The new accelerator(s), which would be constructed at an existing DOE site(s), would be used to irradiate all of the targets (i.e., for production of plutonium-238, isotopes for medical and industrial uses, and materials testing for research and development). Ongoing operations at existing facilities as described in Chapter 3, Affected Environment, would continue.

The targets for plutonium-238 production would be fabricated in one of the three candidate facilities at ORNL, INEEL, or Hanford. The material needed for the target fabrication (neptunium-237) would be transported from SRS to the fabrication facilities. The targets would be irradiated at the new high-energy accelerator facility and transported back to the target fabrication facilities for postirradiation processing.

Targets for medical and industrial isotope production would be fabricated in a new support facility located at the same site as the low-energy accelerator. The targets would be irradiated in the low-energy accelerator and returned to the new support facility for postirradiation processing. Site selection for Alternative 3 is not evaluated as part of this NI PEIS. Because Alternative 3 is evaluated at a generic DOE site, no credit was taken for any support infrastructure existing at the site, and it was postulated that a new support facility would be required to support operation of the low-energy accelerator and its missions and the high-energy accelerator civilian nuclear energy research and development missions if both accelerators were located on the same site. While this approach bounds this NI PEIS for the implementation of Alternative 3, it overstates the impacts because this NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 3 or the low-energy accelerator alone were selected by the Record of Decision for subsequent consideration, follow-on NEPA reviews would evaluate potential locations for either one or both of the accelerators. It is unlikely that DOE would consider locating the new low-energy or high-energy accelerator on a DOE site that does not have existing infrastructure capable of supporting all or most of the mission requirements.

Under Alternative 3, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for storage, target fabrication, target irradiation, postirradiation processing, and the final destination of the plutonium-238. Alternative 3 also would include decontamination and decommissioning of the accelerator(s) and the support facility when the missions are completed, as well as deactivation of FFTF at Hanford.

The low-energy accelerator would serve as a dedicated isotope production facility. Due to the nature of this type of accelerator, it could only produce a limited number of isotopes (listed in Table 1–1), has no ability to satisfy the plutonium-238 needs, and has a very limited ability to support the proposed nuclear-based research and development needs. The preconceptual design of the high-energy accelerator presented in Appendix F focused on supporting the plutonium-238 production mission. The design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce “tuneable” neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and material interactions. The changes required to add additional capability to the high-energy accelerator could be provided, but they would increase the size of the facility, add complexity to the facility design and operation, increase the cost of construction and operation, and potentially require more time for design and construction.

The three options under this alternative and their associated target fabrication, postirradiation processing, and transportation activities are discussed below.

- **Option 1.** REDC at ORNL would be used to fabricate and process the neptunium-237 targets required for plutonium-238 production. The neptunium-237 transported from SRS to ORNL would be stored at REDC. The plutonium-238 product would be transported from ORNL to LANL for use in radioisotope power systems for future U.S. space missions. A new support facility at an existing

DOE site would be used to fabricate and process the targets required for the production of medical and industrial and research isotopes and to store the materials needed for target fabrication.

- **Option 2.** FDPF at INEEL would be used to fabricate and process the neptunium-237 targets associated with plutonium-238 production. The neptunium-237 transported from SRS to INEEL would be stored in FDPF or Building CPP–651 at INEEL. The plutonium-238 product would be transported from INEEL to LANL for use in radioisotope power systems for future U.S. space missions. A new support facility at an existing DOE site would be used to fabricate and process the targets required to produce medical and industrial and research isotopes and to store the materials needed for target fabrication.

- **Option 3.** FMEF at Hanford would be used to fabricate and process the neptunium-237 targets for plutonium-238 production. The neptunium-237 transported from SRS to Hanford would be stored in FMEF. The plutonium-238 product would be transported from Hanford to LANL. A new support facility at an existing DOE site would be used to fabricate and process the targets required for the production of medical and industrial and research isotopes and to store the materials needed for target fabrication.

2.5.5 Alternative 4—Construct New Research Reactor

Under Alternative 4, a new research reactor would be used for target irradiation for the evaluation period of 35 years. The new research reactor, to be constructed at an existing DOE site, would be used to irradiate all targets (i.e., for the production of plutonium-238, isotopes for medical and industrial uses, and materials testing for civilian nuclear energy research and development). Ongoing operations at existing facilities as described in Chapter 3, Affected Environment, would continue.

The targets for plutonium-238 production would be fabricated in one of the three candidate facilities at ORNL, INEEL, or Hanford. The material needed for the target fabrication (neptunium-237) would be transported from SRS to the fabrication facilities. The targets would be irradiated at the new research reactor facility and transported back to the target fabrication facilities for postirradiation processing.

Targets for medical and industrial isotope production would be fabricated in a new support facility located at the same site as the new research reactor. The targets would be irradiated in the new research reactor and returned to the new support facility for postirradiation processing.

Alternative 4 site selection is not evaluated as part of this NI PEIS. Because Alternative 4 is evaluated at a generic DOE site, no credit was taken for any existing support infrastructure existing at the site and it was postulated that a new support facility would be required to support operation of the new research reactor and its missions. While this approach bounds this NI PEIS for the implementation of Alternative 4, it overstates the impacts because this NI PEIS integrates the impacts associated with constructing new support facilities and infrastructure that may be available at the existing DOE site. In the event that Alternative 4 were selected by the Record of Decision for subsequent consideration, follow-up NEPA reviews would evaluate potential locations for the new research reactor. It is unlikely that DOE would consider locating the new research reactor on a DOE site that does not have existing infrastructure capable of supporting all or most of the proposed medical and industrial isotope production and nuclear research and development mission requirements.

Under Alternative 4, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for storage, target fabrication, target irradiation, postirradiation processing, and the final destination of the plutonium-238. Alternative 4 also would include the decontamination and

decommissioning of both the research reactor and the support facility when the missions are completed, as well as deactivation of FFTF at Hanford.

The proposed new research reactor would provide ample neutrons for the production of plutonium-238 and for many of the isotopes listed in Table 1–1. The thermal flux would limit the new research reactor's ability to produce a number of isotopes requiring fast or high-energy neutrons. Its lower flux levels (10^{13} neutrons per square centimeter per second) and predominantly thermal flux may limit its ability to support many of the projected nuclear-based research and development needs.

The three options under this alternative and their associated target fabrication, postirradiation processing, and transportation activities are discussed below.

- **Option 1.** REDC at ORNL would be used to fabricate and process the neptunium-237 targets associated with plutonium-238 production. The neptunium-237 transported from SRS to ORNL would be stored at REDC. The plutonium-238 product would be transported from ORNL to LANL. A new support facility at an existing DOE site would be used to fabricate and process the targets required to produce medical and industrial and research isotopes and to store the materials needed for target fabrication.

- **Option 2.** FDPF at INEEL would be used to fabricate and process the neptunium-237 targets associated with plutonium-238 production. The neptunium-237 transported from SRS to INEEL would be stored in FDPF or Building CPP–651. The plutonium-238 product would be transported from INEEL to LANL. A new support facility at an existing DOE site would be used to fabricate and process the targets required to produce medical and industrial and research isotopes and to store the materials needed for target fabrication.

- **Option 3.** FMEF at Hanford would be used to fabricate and process neptunium-237 targets for plutonium-238 production. The neptunium-237 transported from SRS to Hanford would be stored in FMEF. The plutonium-238 product would be transported from Hanford to LANL. A new support facility at an existing DOE site would be used to fabricate and process the targets required for the production of medical and industrial and research isotopes and to store the materials needed for target fabrication.

2.5.6 Alternative 5—Permanently Deactivate FFTF (with No New Missions)

Under Alternative 5, DOE would permanently deactivate FFTF, with no new missions. Medical and industrial isotope production and civilian nuclear energy research and development missions, at the existing facilities described in Chapter 3, would continue. DOE's nuclear facilities infrastructure would not be enhanced.

2.6 Alternatives Considered and Dismissed

In developing a range of reasonable alternatives, DOE examined the capabilities and available capacities of the existing and planned nuclear research facilities (accelerators, reactors, and processing [hot] cells) that potentially could be used to support one or all of the proposed isotope production and research missions (DOE 2000a). The following facilities were initially considered, but were subsequently dismissed as reasonable alternatives for meeting DOE's proposed nuclear infrastructure mission requirements.

2.6.1 Irradiation Facilities Dismissed

DOE evaluated the irradiation capabilities of existing government, university, and commercial irradiation facilities to determine whether they could significantly support the proposed expanded nuclear infrastructure missions. **Table 2–4** presents irradiation facilities that were initially considered but dismissed from further evaluation because they lacked technical capability or available capacity. Reasons for lacking technical capability include that the facility has been permanently shut down, it does not possess the capability to produce steady-state neutrons, or that it could not maintain sufficient power levels to adequately support steady-state neutron production. Facilities were similarly dismissed if existing capacity was fully dedicated to existing missions, or if use of existing capacity to support this NI PEIS proposed action would impact existing missions. Although a number of facilities shown in Table 2–4 have some available capacity, their combined available capacity is a very small percentage of the capacity needed to support the missions evaluated in this NI PEIS.

Two of these facilities, the Brookhaven LINAC Isotope Producer (BLIP) at the Brookhaven National Laboratory and the Isotope Production Facility (IPF) at the Los Alamos Neutron Science Center (LANSCE), were identified in the NI PEIS Notice of Intent as existing facilities that could potentially support the proposed nuclear infrastructure missions. IPF produces radioisotopes using LANSCE's half-mile accelerator that delivers medium-energy protons. IPF's three major products include germanium-68, strontium-82, and sodium-22. As a result of changing DOE missions, the production of radioisotopes at target area "A" of the LANSCE has been rendered inoperable. DOE is currently in the process of upgrading the LANSCE facility with a new 100-million-electron-volt IPF. The facility is scheduled for completion in 2001. After completion of the LANSCE upgrade, the existing capability at these two facilities will be twice the current need for accelerator-generated medical isotopes. Thus, no new accelerator capacity is needed in the short term. Should isotope demand grow consistent with the Expert Panel Report, there will be a need for expanded isotope production capacity for those isotopes generated by IPF and BLIP. IPF and BLIP were dismissed as reasonable alternatives for the production of medical isotopes because they cannot meet the projected future demand for accelerator-produced isotopes.

The Alternating Gradient System (AGS) accelerator complex at the Brookhaven National Laboratory was evaluated for meeting the mission requirements of medical and industrial isotope production, plutonium-238 production, and civilian nuclear energy research and development. AGS presently accelerates up to 7×10^{13} protons to 24 gigaelectron volts with a cycle time of approximately 2.5 seconds. This corresponds to a beam power of approximately 100 kilowatts. The complex was dismissed as a reasonable alternative because the potential neutron flux generated by the facility in the required configuration (i.e., with a spallation target) would not be adequate to meet the mission goals and, in addition, operating the complex in the required configuration would not be compatible with the present primary mission of the facility (Kovar 2000).

Two existing operating DOE facilities, ATR and HFIR, were evaluated as components of Alternative 2, Use Only Existing Operational Facilities. These two facilities currently provide isotope production capabilities and were examined for their abilities to meet the isotope production and nuclear research and development requirements of the proposed expanded missions. In addition, DOE considered whether production from ATR and HFIR could be enhanced by increasing power levels at the reactors or through other modifications to the facilities, which included the installation of rapid radioisotope retrieval systems for the production of isotopes with a short half-life. In general, the installation of rapid radioisotope retrieval systems in reactors does not increase the ability of reactors to produce larger quantities of isotopes; it enables the reactors to produce a broader spectrum of isotopes. While some growth is possible in isotope production at ATR and HFIR, such growth would be insufficient to meet the long-term growth projections discussed in Section 1.2.1. Further growth could only be enabled by increasing reactor power levels. At ATR, increases in power level are possible to the extent that priority DOE Office of Naval Reactor missions are not impacted. Raising ATR

Table 2–4 Irradiation Facilities Considered but Dismissed from Further Evaluation

Reasons for Dismissal	Facility
Facilities lacking sufficient neutron production capacity to support the NI PEIS proposed action without impacting existing missions	Neutron Radiographic Reactor Argonne National Laboratory–West
	Brookhaven Medical Research Reactor Brookhaven National Laboratory
	National Bureau of Standards Reactor National Institute of Standards and Technology
	General Atomics Training, Research, and Isotope Production Reactors
	University Small Research Reactors
	University Large Research Reactors (i.e., Massachusetts Institute of Technology and University of Missouri)
	ATLAS Heavy Ion Facility Argonne National Laboratory
	Oak Ridge Electron Linear Accelerator Oak Ridge National Laboratory
	Holifield Radioactive Ion Beam Facility Oak Ridge National Laboratory
	Heavy Ion Linear Accelerator Lawrence Berkeley National Laboratory
	Alternating Gradient Synchrotron Heavy Ion Facility Brookhaven National Laboratory
	Continuous Electron Beam Accelerator Facility Thomas Jefferson National Accelerator Facility
	Electron Linear Accelerator Lawrence Livermore National Laboratory
	University Linear Accelerators
Facilities with capacity fully dedicated to existing missions	Annular Core Research Reactor Sandia National Laboratory
	Brookhaven LINAC Isotope Producer Brookhaven National Laboratory
Facilities not capable of steady-state neutron production	Sandia Pulse Reactor II and III Sandia National Laboratory
	Transient Reactor Test Facility Argonne National Laboratory–West
	Zero Power Physics Reactor Idaho National Engineering and Environmental Laboratory
	Power Burst Facility Idaho National Engineering and Environmental Laboratory
	Intense Pulsed Neutron Source Argonne National Laboratory
	Flash X-Ray Facility Lawrence Livermore National Laboratory

Table 2–4 Irradiation Facilities Considered but Dismissed from Further Evaluation (Continued)

Reason for Dismissal	Facility
Facilities with insufficient power to sustain adequate steady-state neutron production	Brookhaven Medical Research Reactor Brookhaven National Laboratory
	Los Alamos Critical Assembly Facility Los Alamos National Laboratory
	General Atomics Training, Research, and Isotope Production Reactors
	University Small Research Reactors
	Booster Applications Facility Brookhaven National Laboratory
	Cyclotron Facility Brookhaven National Laboratory
	Low-Energy Demonstration Accelerator[a] Los Alamos National Laboratory
Facilities that jointly can meet existing accelerator-produced medical isotope demands but cannot meet projected future needs	Los Alamos Neutron Science Center Linear Accelerator Isotope Production Facility Los Alamos National Laboratory
	Brookhaven LINAC Isotope Producer Brookhaven National Laboratory
Facilities that are under construction with capacity fully dedicated to other planned missions	Dual Axis Radiographic Hydrodynamic Test Facility Los Alamos National Laboratory
	Spallation Neutron Source Oak Ridge National Laboratory
Facilities that have been permanently shut down	High Flux Beam Reactor Brookhaven National Laboratory
	Tower Shielding Facility Oak Ridge National Laboratory
	Cyclotron Facility Oak Ridge National Laboratory

a. Not listed in source document.
Key: LINAC, linear accelerator; ATLAS, Argonne Tandem - LINAC Accelerator System.
Source: DOE 2000a.

power would only delay the point in time at which capacity is reached. The power level at HFIR is already at 100 percent of its current Authorization Basis (85 megawatts), and modification of this Authorization Basis would be required to increase to full-design power (100 megawatts). Increasing the power levels at ATR and/or HFIR will enhance the isotope production capability of these reactors. However, the enhancement in production capability would not be adequate to meet the future demand for isotope production; it would only delay the point in time at which the United States' reactor isotope production capacity is reached. Therefore, increasing the power levels at ATR and/or HFIR was dismissed as a reasonable alternative for meeting the requirements of the DOE missions.

Modification of CLWRs to enable online insertion and retrieval of targets for the medical and industrial isotope production missions was evaluated and dismissed as a reasonable alternative. This decision was made because the required facility modifications would be significant and would include penetrations into the reactor vessel and, potentially, the containment vessel. Additional facility modifications would be required to enable loading of the targets into a shielded cask for transport to a processing facility. Performing these facility modifications would require an extended refueling outage (with a resulting loss of power generation revenue to the CLWR owner) and could potentially extend subsequent maintenance or refueling outages to inspect, test, and maintain the insertion and retrieval system, reactor vessel penetrations, and potential containment vessel penetrations. CLWRs were considered for the production of medical isotopes with moderate and long half-lives by

irradiating targets in the CLWR vessel but outside the reactor core region (i.e., outside of the fuel assembly region). Only one of the isotopes listed in the Expert Panel Report, *Expert Panel: Forecast Future Demand for Medical Isotopes* (Wagner et al. 1998), strontium-89, was considered a potential candidate for production in a CLWR, outside of the reactor core region. Strontium-89 has a half-life of 50.5 days. Irradiated targets containing strontium-89 could only be harvested from a CLWR every 18 to 24 months during a scheduled reactor refueling outage. Approximately 10 CLWRs with refueling outages scheduled every 2 to 3 months would be required to support a program to ensure a continuous and reliable supply of strontium-89. Due to the CLWR's ability to irradiate targets for only a very limited array of medical isotopes (only one isotope in current demand was identified), it was not considered a reasonable alternative for expanding the U.S. infrastructure to provide an overall enhancement of the medical isotope production missions. CLWRs were also considered for the proposed DOE civilian nuclear energy research and development missions. CLWRs will continue to support the commercial industry research and development activities by providing a test bed for industry-sponsored lead test assemblies and other related research. CLWRs cannot meet most of the requirements for supporting the DOE civilian nuclear energy research and development missions and, therefore, were dismissed as a reasonable alternative for supporting these missions.

CANDU reactors, operating in Canada, were considered for supplying irradiation services for the plutonium-238 production mission. (Note: Canada is currently the major supplier of medical radioisotopes used in the United States.) Since use of the CANDU reactors does not meet the programmatic issue being addressed in this NI PEIS, that is, the enhancement of the United States infrastructure to support the proposed missions, the CANDU reactors were considered but were dismissed as a reasonable alternative. However, the environmental impacts associated with transporting the nonirradiated and irradiated neptunium-237 targets between the CANDU reactors and the target fabrication and processing facilities in the United States are bound by the evaluations presented in this NI PEIS for the CLWR options of Alternative 2 (Use Only Existing Operational Facilities).

Some facilities listed in Table 2–4 do not have the capacity to support the proposed missions without impacting existing missions, but do have some existing medical or industrial isotope production or nuclear research and development missions. These facilities will continue to support their existing missions at current levels.

2.6.2 Processing Facilities Dismissed

Numerous existing U.S. processing hot cell facilities possess the capabilities and capacities to support the proposed missions. Given this general availability, only existing processing facilities that are colocated at DOE's candidate irradiation facility sites (i.e., ORNL, INEEL, and Hanford) were evaluated in this NI PEIS. Although multiple processing facilities exist at each of these sites, only the most suitable facilities in terms of capability, capacity, and availability were given further consideration. The processing facilities that were dismissed from evaluation are listed in **Table 2–5**.

Based on public comments on the scope of the proposed *Plutonium-238 Production EIS,* the H-Canyon and HB-Line facilities at SRS that previously performed the processing for the plutonium-238 production mission were reconsidered as potential processing facilities for the proposed plutonium-238 production mission even though the facilities are not colocated with a proposed irradiation facility. After reviewing the plutonium-238 production target fabrication and processing requirements, the capabilities and capacities of the facilities, and the modifications and resources required to support the plutonium-238 production mission, use of the H-Canyon and HB-Line facilities was dismissed as a reasonable alternative because:

1. DOE plans to shut down these facilities following completion of their current missions to stabilize and prepare for disposition of Cold War legacy nuclear materials and certain spent nuclear fuel, and a

Table 2–5 Processing Facilities Considered but Dismissed from Further Evaluation

Location	Facility
Argonne National Laboratory	Irradiated Materials Facility
	Alpha-Gamma Hot Cell Facility
	Building 205
Argonne National Laboratory–West	Hot Fuel Examination Facility
	Analytical Laboratory
	Fuel Conditioning Facility
Brookhaven National Laboratory	Target Processing Laboratory
	Metallurgical Evaluation Laboratory
	High Intensity Radiation Development Laboratory
Hanford Site	222-S Facility
	Postirradiation Testing Laboratory
	Shielded Material Facility
Idaho National Engineering and Environmental Laboratory	Test Area North Hot Shop and Hot Cell Facilities
	Remote Analytical Laboratory
	Fuel Processing Facility
Los Alamos National Laboratory	Chemistry and Metallurgical Research Building
	Technical Area–48
Oak Ridge National Laboratory	Radioactive Materials Analytical Laboratory
	Building 4501
	Irradiated Materials Examination and Testing Facility
	Radioisotope Development Laboratory
	Irradiated Fuels Examination Laboratory
Sandia National Laboratories	Hot Cell Facility
Savannah River Site	Defense Waste Processing Facility
	High-level cells
	Intermediate-level cells
	Californium shipping/receiving facility
	Californium processing facility

Source: DOE 2000a.

determination that a new nonchemical processing technology is capable of preparing aluminum-clad research reactor spent nuclear fuel for ultimate disposition.

2. The cost to extend the operating lives of these facilities to support plutonium-238 production for the proposed 35-year evaluation period would be approximately one order of magnitude higher than the costs associated with the processing facilities evaluated in this NI PEIS.

A commentor also proposed using the H-Canyon and HB-Line for a short campaign to produce all of the required plutonium-238. Based on prior production rates, it would take approximately 7 years to produce 175 kilograms (385 pounds) of plutonium-238, the total plutonium-238 production goal. The target fabrication and irradiation requirements to support this processing campaign to produce 25 kilograms (55 pounds) per year of plutonium-238 would be significant but feasible. The irradiation requirements could be supported by operating five CLWRs or operating FFTF at the 400-megawatt power level. However, a concern about the short campaign option is that the plutonium-238 would be stored a long time before use and because of natural decay may not meet the specification requirements when finally needed. This alternative was dismissed

because of the uncertainty that over time the plutonium-238 produced may not meet the required specification for NASA missions.

2.7 SUMMARY OF ENVIRONMENTAL IMPACTS, SCHEDULES, AND MISSION EFFECTIVENESS

The following sections summarize the environmental impacts and schedules associated with the alternatives and options and compare the impacts among the alternatives described in Sections 4.2 through 4.7. Chapter 4 shows construction impacts that would result from implementation of Alternative 3 and 4, as well as operational impacts for all of the alternatives. Section 2.7.1 compares the environmental impacts and risks among the alternatives. Section 2.7.2 summarizes the implementation schedules for each alternative. Mission effectiveness is discussed in Section 2.7.3.

As discussed in Section 1.8, tables and text in this section have been revised in response to comments about the difficulty of comparing environmental impacts among the alternatives in the Draft NI PEIS. Tables and figures in this section now focus on estimated environmental impacts that would result from implementation of the alternatives. Baseline environmental data for the sites and for the candidate facilities are now given in Chapter 3. In this NI PEIS, Option 1 of the No Action Alternative is used as a basis for the comparison of impacts at candidate sites.

Numerical values are assigned to environmental impacts that include radiological and nonradiological risks to the public and workers at the candidate sites and along representative transportation routes, potential quantities of waste generated, and potential quantities of spent nuclear fuel generated. These numerical values reflect the degree to which the proposed activities would increase the environmental impacts of current activities and operations at the candidate sites. It should be noted that most of the options being considered under the various alternatives involve the use of more than one site, so the numerical values presented are the sums of the values for all of the relevant sites or transportation routes. There are two exceptions—the health risks to the maximally exposed individual and the noninvolved worker. For these two exceptions, the numerical value presented is the maximum value among all relevant sites.

2.7.1 Summary of Environmental Impacts

2.7.1.1 Radiological and Hazardous Chemical Impacts

RADIOLOGICAL IMPACTS

Table 2–6 summarizes radiological and hazardous chemical risks that could occur under implementation of the alternatives from operations at fabrication, processing, and irradiation facilities. Radiological risks to the maximally exposed individual are listed in columns 2 and 5 for normal operations and accidents, respectively. Similarly, columns 3 and 6 display radiological risks to the public for normal operations and accidents, and columns 4 and 7 show radiological risks to workers at candidate irradiation facilities and processing and fabrication facilities. As indicated in the table, Option 1 of the No Action Alternative is the basis for comparing impacts that would result from implementation of the other alternatives and options. Impact values for Option 1 of the No Action Alternative are set to zero and provide a reference point for comparing impacts that would result from implementation of the other alternatives and options. Negative values in the table indicate a decrease in risk with respect to Option 1 of the No Action Alternative.

The risk values presented are the sum of individual risk values from operational activities in the fabrication, processing, and irradiation facilities used under each alternative and option. For Alternatives 2 through 4, where FFTF would be permanently deactivated, the values presented also include the reduction in risk from

Table 2–6 Comparison Among Alternatives: Impacts on Occupational and Public Health and Safety from Baseline Conditions

Options[a]	Radiological Risks from Normal Operations over 35 Years			Radiological Risks[b] from Accidents over 35 Years			Hazardous Chemical Risks from Normal Operations over 35 Years	
	Maximally Exposed Individual (LCF Risk)	Population (LCF)	Workforce (LCF)	Maximally Exposed Individual (LCF Risk)	Population (LCF)	Workforce (LCF)	Maximum Cancer Risk[c]	Hazard Index[d]
No Action Alternative								
1[e]	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
2	3.0×10^{-12}	1.4×10^{-7}	0.017	0.00	0.00	0.00	0.00	0.00
3	4.2×10^{-13}	6.1×10^{-9}	0.017	0.00	0.00	0.00	0.00	0.00
4	7.0×10^{-13}	7.5×10^{-8}	0.017	0.00	0.00	0.00	0.00	0.00
Alternative 1: Restart FFTF								
1 or 4	9.3×10^{-8}	0.0039	0.25	4.5×10^{-4}	0.54	3.5×10^{-4}	2.6×10^{-7}	0.0064
2 or 5	9.3×10^{-8}	0.0039	0.25	4.5×10^{-4}	0.41	3.5×10^{-4}	1.3×10^{-7}	0.0031
3 or 6	9.6×10^{-9}	0.0018	0.25	6.8×10^{-6}	0.21	4.2×10^{-4}	4.7×10^{-8}	0.0011
Alternative 2: Use Only Existing Operational Facilities[f, g]								
1	3.3×10^{-11}	-4.7×10^{-4}	0.16	5.7×10^{-5}	0.16	3.5×10^{-4}	2.6×10^{-7}	0.0064
2	4.6×10^{-12}	-4.7×10^{-4}	0.16	1.5×10^{-5}	0.03	3.5×10^{-4}	1.3×10^{-7}	0.0031
3	-2.3×10^{-9}	-4.7×10^{-4}	0.16	2.9×10^{-6}	0.11	3.5×10^{-4}	4.7×10^{-8}	0.0011
4	3.3×10^{-11}	-4.7×10^{-4}	0.16	5.7×10^{-5}	0.16	3.5×10^{-4}	2.6×10^{-7}	0.0064
5	4.6×10^{-12}	-4.7×10^{-4}	0.16	1.5×10^{-5}	0.03	3.5×10^{-4}	1.3×10^{-7}	0.0031
6	-2.3×10^{-9}	-4.7×10^{-4}	0.16	2.9×10^{-6}	0.12	3.5×10^{-4}	4.7×10^{-8}	0.0011
7	3.3×10^{-11}	-4.7×10^{-4}	0.16	5.7×10^{-5}	0.16	3.5×10^{-4}	2.6×10^{-7}	0.0064
8	4.6×10^{-12}	-4.7×10^{-4}	0.16	1.5×10^{-5}	0.03	3.5×10^{-4}	1.3×10^{-7}	0.0031
9	-2.3×10^{-9}	-4.7×10^{-4}	0.16	2.9×10^{-6}	0.11	3.5×10^{-4}	4.7×10^{-8}	0.0011
Alternative 3: Construct New Accelerator(s)[f, g]								
1	6.1×10^{-8}	0.0030	0.95	9.2×10^{-5}	0.22	5.0×10^{-4}	1.6×10^{-9}	1.1×10^{-7}
2	6.1×10^{-8}	0.0030	0.95	5.0×10^{-5}	0.09	5.0×10^{-4}	1.6×10^{-9}	1.1×10^{-7}
3	6.1×10^{-8}	0.0030	0.95	3.8×10^{-5}	0.18	5.0×10^{-4}	1.6×10^{-9}	1.1×10^{-7}
Alternative 4: Construct New Research Reactor[f, g]								
1	4.5×10^{-8}	0.002	0.49	9.0×10^{-5}	0.21	4.5×10^{-4}	6.4×10^{-10}	2.3×10^{-6}
2	4.5×10^{-8}	0.002	0.49	4.8×10^{-5}	0.08	4.5×10^{-4}	6.4×10^{-10}	2.3×10^{-6}
3	4.5×10^{-8}	0.002	0.49	3.6×10^{-5}	0.17	4.5×10^{-4}	6.4×10^{-10}	2.3×10^{-6}
Alternative 5: Permanently Deactivate FFTF (with No New Missions)								
	-2.3×10^{-9}	-4.7×10^{-4}	-0.0097	-2.2×10^{-13}	-1.6×10^{-8}	-1.3×10^{-13}	0.00	0.00

a. For descriptions of the options under each alternative, see Section 2.5.
b. Accident risks include accident likelihood over 35 years and the consequences.
c. Probability that an individual would develop cancer from exposure to hazardous (carcinogenic) chemicals.
d. A measure of hazard from exposure to multiple toxic (noncarcinogenic) chemicals. If this value is less than 1, the exposure is unlikely to produce an adverse toxic effect.
e. Baseline conditions for the comparison of impacts is Option 1 of the No Action Alternative.
f. These alternatives include FFTF deactivation impacts. The deactivation would lead to negative impacts (reduced risk); see Alternative 5.
g. The reduction in impacts from deactivating FFTF would affect the impacts to the population and workforce for Alternatives 2 through 4 and to the maximally exposed individual only for those options within Alternatives 2 through 4 that use FMEF.

Note: Refer to the text for a discussion on how the risk values in this table have been generated.
Key: LCF, latent cancer fatalities.

FFTF deactivation, where applicable. For example, the radiological risk to the population from normal operations for Option 3 of Alternative 2 (i.e., irradiation at ATR, fabrication and processing at FMEF, and deactivation of FFTF) is given as -4.7×10^{-4} latent cancer fatality. This value was calculated by adding the population risks from fabrication and processing at FMEF and irradiation at ATR, 7.7×10^{-7} latent cancer fatality (see Table 4–77), and Alternative 5 (Permanently Deactivate FFTF [with No New Missions]), -4.7×10^{-4} latent cancer fatality. The latter risk is the sum of the population risk associated with the activities during permanent deactivation of FFTF, 1.8×10^{-5} latent cancer fatality (see Section 4.4.1.2), and that resulting from not keeping FFTF in standby for 35 years, -4.9×10^{-4} latent cancer fatality (the negative value reflects the reduction in risk) (see Section 4.2.1). The radiological risks for accident conditions are the sum of accident risks evaluated for each option. For each accident, the risk value is the product of the accident consequences and its occurrence likelihood over 35 years of operation. Chapter 4, Appendix H, and Appendix I provide the details on public and occupational risk calculations.

A comparison of radiological risks estimated to result from normal operations over 35 years (columns 2 and 3 of Table 2–6) shows that implementation of the alternatives would result in a small risk of a latent cancer fatality among the general public. Radiological accident risks to the public over 35 years (columns 5 and 6 of Table 2–6) are estimated to be less than one latent cancer fatality. **Figure 2–23** shows estimated latent cancer fatalities among the population at risk from potential accidents at candidate sites. Each bar in Figure 2–23 represents the estimated latent cancer fatalities for a given option.

For example, there are six bars shown above the alternative labeled "Restart FFTF." The first of the six bars represents the estimated latent cancer fatalities for implementation of Option 1, the second bar represents the estimated latent cancer fatalities for implementation of Option 2, etc. As discussed in Section 4.2, storage containers for neptunium-237 targets would not be expected to rupture under the most severe accident evaluated in this NI PEIS. Therefore, no latent cancer fatalities would be expected under implementation of the No Action Alternative. Deactivation of FFTF (with no new missions) would result in a small reduction in radiological accident risks in comparison with the No Action Alternative. Differences in the radiological accident risks among alternatives and among options within a given alternative are driven by accident risks at the target fabrication and processing facilities. This point is illustrated in **Figure 2–24**.

Figure 2–24 shows risks to the public that would result from radiological accidents at candidate fabrication and processing facilities and candidate irradiation facilities. Latent cancer fatalities estimated for candidate fabrication and processing facilities are shown to the left of the dividing line in Figure 2–24, and the estimated latent cancer fatalities for candidate irradiation facilities appear on the right side of the dividing line. The estimated latent cancer fatalities for FMEF under Options 3 and 6 of Alternative 1 are labeled "FMEF (Hanford)." Under Options 3 and 6 of Alternative 1, FMEF would serve as the fabrication and processing facility for all targets. If FMEF were selected to fabricate and process neptunium-237 targets only, the radiological risk to the public would be reduced by approximately a factor of two, as shown by the bar labeled "FMEF (Hanford, neptunium-237 targets only)" in Figure 2–24. Among the candidate fabrication and processing facilities, accident risks to the public range from a low of 0.029 latent cancer fatality at FDPF (INEEL) to 0.377 latent cancer fatality at RPL (Hanford). Although all of the accident risks shown in Figure 2–24 are less than one latent cancer fatality, risks to the public that would be expected from radiological accidents at candidate fabrication and processing facilities are relatively large in comparison to those for candidate irradiation facilities.

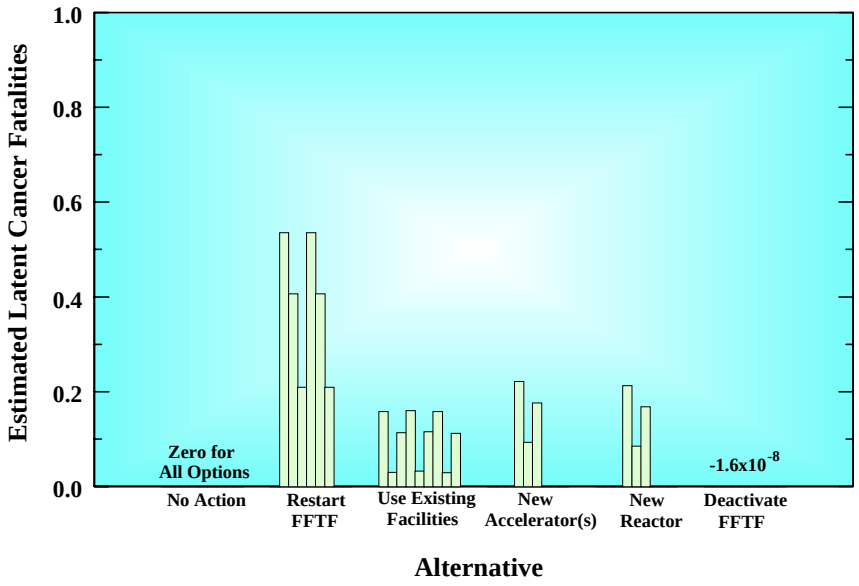


Figure 2–23 Public Risks Due to Radiological Accidents at Candidate Sites (35 Years)

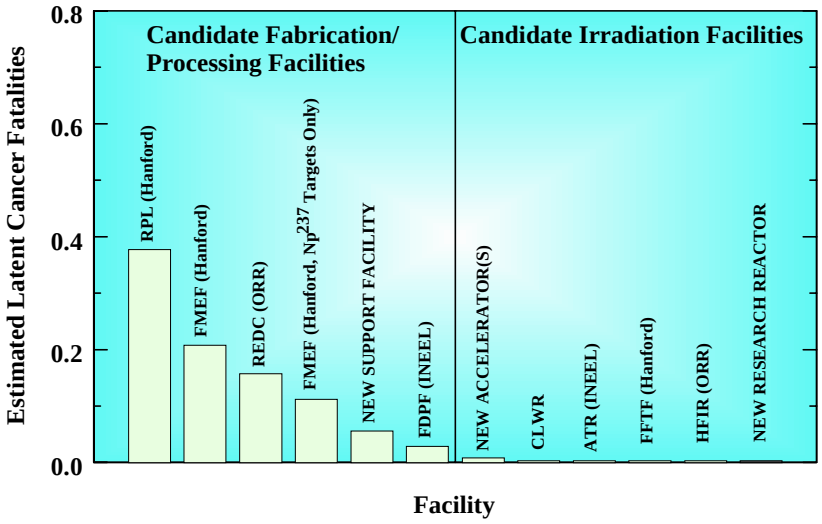


Figure 2–24 Public Risks Due to Radiological Accidents at Candidate Facilities (35 Years)

Prevailing weather conditions, the geographical distribution of the population at risk, and the type of target(s) processed (neptunium-237 only, other isotopes only, or both) all contribute to variations in the radiological risk to the public. Calculations of accident consequences and risks include populations residing within 80 kilometers (50 miles) of the accident site, although the consequences and risks decrease noticeably with increasing distance from the accident site. As shown in **Figure 2–25**, RPL (Hanford) and REDC (ORR) have the largest populations residing within 16 kilometers (10 miles) of candidate sites, while FDPF (INEEL) has the smallest. Because the total population residing within 16 kilometers (10 miles) of FDPF is relatively small, the curve representing populations residing near FDPF is nearly coincident with the horizontal axis in Figure 2–25. Comparing Figures 2–24 and 2–25, it is clear that accident risks due to fabrication and processing activities are driven by both the type of processing activities and the total population residing near the facilities. In turn, variations in accident risks among the alternatives, as well as variations among options within an alternative, are driven by the selection of fabrication and processing facilities. The choice for irradiation facility would have little effect on radiological accident risks to the public.

HAZARDOUS CHEMICAL IMPACTS

Columns 8 and 9 of Table 2–6 display cancer risks and hazard indexes that could result from airborne emissions of hazardous chemicals from candidate processing facilities. As discussed in Section H.3, cancer risk factors listed in column 8 of Table 2–6 are estimates of an upper-bound lifetime probability of an individual developing cancer due to exposure to carcinogenic chemicals. For all alternatives and options, the maximum cancer risk factor is 2.6×10^{-7} (or a likelihood of approximately 1 in 3,800,000) or less. Different carcinogens can cause or promote different forms of cancer. In general, cancer risk factors for different carcinogens are not additive because there are potential synergistic or antagonistic chemical interactions in multiple-substance exposures (EPA 1989). Therefore, column 8 of the table lists the maximum cancer risk factor for each alternative. Hazard indexes listed in column 9 of Table 2–6 estimate the potential for adverse toxic (noncancerous) health effects due to exposure to hazardous chemicals. If the hazard index is less than one, adverse (noncancerous) health effects would not be expected. For all of the alternatives and options, hazard indexes are 0.0064 or less. The results (presented in columns 8 and 9 of Table 2–6) indicate that no adverse toxic health or cancer effects would be expected from exposure to hazardous chemicals released under the implementation of any of the alternatives.

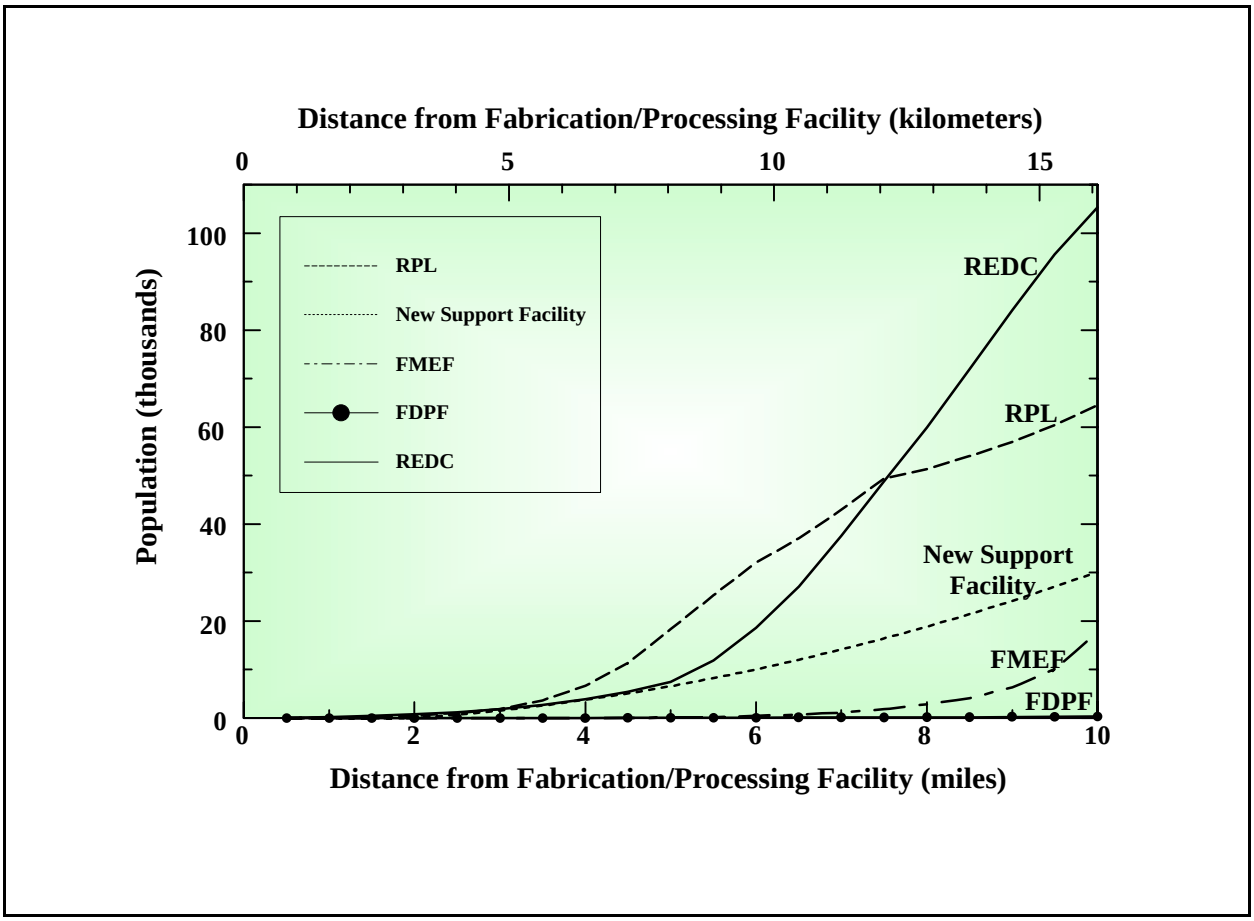


Figure 2–25 Population Residing Within 16 Kilometers (10 Miles) of Candidate Fabrication and Processing Facilities

2.7.1.2 Generation and Disposition of Waste and Spent Nuclear Fuel

Table 2–7 summarizes the estimated amount of waste and spent nuclear fuel that would be generated under implementation of the nuclear infrastructure alternatives. As discussed in Chapter 4, waste that would result from implementation of the alternatives would be relatively small in comparison to current waste generation at the candidate sites. Current waste management practices at the candidate sites would be sufficient to manage waste that would result from the nuclear infrastructure alternatives.

TRANSURANIC WASTE/HIGH-LEVEL RADIOACTIVE WASTE

The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste. Irrespective of how the waste is classified (i.e., transuranic or high-level radioactive), the waste composition and characteristics are the same, and the waste management (i.e., treatment and onsite storage) as described in this NI PEIS would be the same. In addition, either waste type would require disposal in a suitable repository. As shown in column 2 of Table 2–7, between 240 and 380 cubic meters (314 and 497 cubic yards) of transuranic waste or high-level radioactive waste would result from implementation of Alternatives 1 through 4. This waste would result from processing irradiated neptunium-237 targets to harvest plutonium-238. Approximately 380 cubic meters (497 cubic yards) of waste per year for 35 years (see Table 2–7) would be generated for all options under Alternatives 1 through 4, except those for which target fabrication and processing would be conducted at FDPF at INEEL. If FDPF were selected for neptunium target fabrication and processing, then approximately 240 cubic meters (314 cubic yards) of waste would be generated during the program.

LOW-LEVEL AND MIXED LOW-LEVEL WASTE

Columns 3 and 4 of Table 2–7 summarize the total low-level radioactive waste and mixed low-level radioactive waste generation that would be expected from implementation of the alternatives. Low-level radioactive waste would be generated at the irradiation facilities and at the fabrication and processing facilities. As shown, the low-level radioactive waste generation that would result under Alternative 2 would be less than half of that for Alternatives 1, 3, and 4, and mixed low-level radioactive waste generation would be almost half. This is because under Alternative 2 currently operational facilities would be used for target irradiation and these facilities would generate little additional low-level and mixed low-level radioactive waste. Also under Alternative 2, no waste generation would result from production of additional medical and industrial isotopes.

DOE's approach for managing low-level and mixed low-level radioactive waste is provided in the Record of Decision for its Waste Management Program (65 FR 10061). The Record of Decision states that for the management of low-level radioactive waste, minimal treatment will be performed at all sites, and disposal will continue to the extent practicable, on site at INEEL, LANL, ORR, and SRS. In addition, Hanford and the Nevada Test Site will be available to all DOE sites for low-level radioactive waste disposal. The Record of Decision does not preclude the use of commercially licensed low-level radioactive waste disposal facilities. Low-level radioactive waste generated at Hanford would be disposed of on site. However, if DOE determines that use of the Hanford waste management infrastructure or other DOE sites is not practical or cost effective, DOE may issue an exemption under DOE Order 435.1 for the use of non-DOE facilities (i.e., commercial facilities) to store, treat, and dispose of such waste generated from the restart and operation of FFTF.

Table 2–7 Comparison of Waste and Spent Nuclear Fuel Generation Among Alternatives

Options[a]	Waste Generation in Cubic Meters (35 Years)					Spent Nuclear Fuel in Metric Tons
	Transuranic/ High-Level	Low-Level	Mixed Low-Level	Hazardous	Nonhazardous	
No Action						
1	0.0	0.0	0.0	0.0	0.0	0.0
2	0.0	<10	0.0	0.0	0.0	0.0
3	0.0	<10	0.0	0.0	0.0	0.0
4	0.0	<10	0.0	0.0	0.0	0.0
Alternative 1: Restart FFTF						
1	380	5,000	320	680	943,000	16
2	240	5,200	320	680	902,000	16
3	380	5,000	320	670	1.5×10^6	16
4	380	5,000	320	680	943,000	16
5	240	5,200	320	680	902,000	16
6	380	5,000	320	670	1.5×10^6	16
Alternative 2: Use Only Existing Operational Facilities						
1	380	2,100	<180[b]	3,100[c]	105,000	0
2	240	2,300	<180[b]	3,100[c]	64,000	0
3	380	2,100	<180[b]	3,100[c]	660,000	0
4	380	2,100	<180[b]	3,100[c]	105,000	0
5	240	2,300	<180[b]	3,100[c]	64,000	0
6	380	2,100	<180[b]	3,100[c]	660,000	0
7	380	2,100	<180[b]	3,100[c]	105,000	0
8	240	2,300	<180[b]	3,100[c]	64,000	0
9	380	2,100	<180[b]	3,100[c]	660,000	0
Alternative 3: Construct New Accelerator(s)						
1	380	5,000	430[b]	3,200[c]	1.1×10^7	NA
2	240	5,200	430[b]	3,200[c]	1.1×10^7	NA
3	380	5,000	430[b]	3,200[c]	1.1×10^7	NA
Alternative 4: Construct New Research Reactor						
1	380	4,800	330[b]	3,300[c]	1.1×10^6	11
2	240	4,900	330[b]	3,300[c]	1.0×10^6	11
3	380	4,800	330[b]	3,300[c]	1.7×10^6	11
Alternative 5: Permanently Deactivate FFTF (with No New Missions)						
	0.0	0.0	(b)	2,500[d]	0.0	0

a. For descriptions of the options under each alternative, see Section 2.5.
b. The deactivation of FFTF would result in the removal of approximately 980,000 liters (260,000 gallons) of sodium. This sodium would be evaluated for alternate uses and is therefore not included in mixed low-level radioactive waste for Alternatives 2 through 5.
c. 2,500 cubic meters of these materials would be evaluated for radioactive contamination and would be reused or recycled if possible.
d. These materials would be evaluated for radioactive contamination and would be reused or recycled if possible.

Key: NA, not applicable.

Solid low-level radioactive waste generated at ORR eventually would have to be disposed of off site due to lack of low-level waste disposal capacity at ORR. Low-level radioactive waste generated at INEEL would be disposed of on site. At some future time, low-level radioactive waste would be disposed of off site.

In compliance with the Waste Management Program Record of Decision, DOE's mixed low-level radioactive waste will be treated at: Hanford, INEEL, ORR, and SRS, and disposed of at Hanford and the Nevada Test Site. Existing candidate sites analyzed in this NI PEIS all have treatment facilities for mixed low-level radioactive waste (see Section 3.2.11 [ORR], Section 3.3.11 [INEEL], and Section 3.4.11 [Hanford]). Solid mixed low-level radioactive waste generated at ORR and INEEL would have to eventually be disposed of off site due to lack of onsite mixed low-level radioactive waste disposal capacity.

HAZARDOUS WASTE

Hazardous waste that would result from implementation of the nuclear infrastructure alternatives is shown in column 5 of Table 2–7. The amount of hazardous waste generated under the alternatives is relatively small in comparison to hazardous waste currently generated at the candidate sites: ORR (Table 3–11), INEEL (Table 3–25), and Hanford (Table 3–34). Estimated amounts of hazardous waste that would be generated under Alternatives 2 through 4 include the hazardous waste that would be generated under Alternative 5 (Permanently Deactivate FFTF [with No New Missions]).

Based on the Record of Decision for hazardous waste issued on August 5, 1998 (63 FR 41810), nonwastewater hazardous waste would be treated and disposed of at offsite commercial facilities. As discussed in Chapter 4, hazardous waste generated under the nuclear infrastructure alternatives would be stored in onsite facilities permitted under the Resource Conservation and Recovery Act or generator accumulation areas prior to shipment to a commercial facility permitted to manage hazardous waste.

NONHAZARDOUS WASTE

Nonhazardous waste that would be expected from implementation of the nuclear infrastructure alternatives is listed in column 6 of Table 2–7. Nonhazardous waste that would be expected under implementation of Alternative 3 (Construct New Accelerator[s]) is at least a factor of six larger than the nonhazardous waste estimated for the other alternatives. As indicated in Sections 4.5.1.1.12 and 4.5.1.2.13, nonhazardous waste that would be produced under Alternative 3 would be driven by sanitary waste and process wastewater resulting from construction and operation of accelerators and the new support facility.

As indicated in Sections 4.3.1.1.13 and 4.3.2.1.13, nonhazardous solid waste that would be generated at ORR and INEEL would represent less than 0.5 percent of the generating site's onsite nonhazardous waste disposal capacity. Nonhazardous solid waste that would be generated at Hanford under the nuclear infrastructure alternatives would be recycled or sent off site for disposal as industrial waste. Nonhazardous process wastewater at the candidate sites would represent a small fraction of the generating sites capacity and would be treated on site. Sanitary wastewater would be treated on site as necessary prior to offsite disposition.

SPENT NUCLEAR FUEL

Changes in the generation of spent nuclear fuel would occur only under implementation of Alternatives 1 (Restart FFTF) and 4 (Construct New Research Reactor). Spent nuclear fuel that would be generated under Alternative 1 would be less than 1 percent (by weight) of the current spent nuclear fuel inventory at Hanford. As discussed in Section 4.3.1.1.14, spent nuclear fuel that would be generated at Hanford under implementation of Alternative 1 would be placed in facility storage vessels and onsite dry storage pending ultimate disposal in a geologic repository. Spent nuclear fuel generated under Alternative 4 would be stored on site in wet storage pending ultimate disposal in a geologic repository (Section 4.6.1.2.14).

2.7.1.3 Water Use

CONSTRUCTION

For construction of new facilities under Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor), water is expected to be required for such uses as mixing concrete, dust control, washing activities, and potable and sanitary needs. Water use for facility construction is estimated at 22.7 million liters (6 million gallons) for the high-energy accelerator, 14 million liters (3.7 million gallons) for the low-energy accelerator, 11.7 million liters (3.1 million gallons) for the new research reactor, and 14.6 million liters (3.85 million gallons) for the new support facility on an annualized (construction-year) basis.

OPERATIONS

Figure 2–26 shows the annual water use that would be expected to occur under the nuclear infrastructure alternatives. As discussed in Section 2.5.1 under the No Action Alternative, FFTF would remain in standby and DOE's nuclear infrastructure would not be enhanced. In standby condition, the FFTF uses approximately 197 million liters (52 million gallons) of groundwater per year. In Figure 2–26, the No Action Alternative is used as a basis for comparison of water use among the alternatives. Therefore, water use for the No Action Alternative is shown as zero. The water use shown in Figure 2–26 for Alternative 1 (Restart FFTF) is the additional groundwater use that would result from operation of the FFTF. Under Alternatives 2 through 5, FFTF would be deactivated, thus saving approximately 197 million liters (52 million gallons) per year in groundwater required for maintaining FFTF in standby. As a result, the water use is negative for Alternatives 2 (Use Only Existing Operational Facilities) and 5 (Permanently Deactivate FFTF [with No New Missions]). The negative increment in water use would be more than offset by the increase in water use estimated for Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor).

2.7.1.4 Air Quality

CONSTRUCTION

Under Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor), new irradiation and support facilities would be constructed to support DOE's nuclear missions. Facility construction would not be required under the other alternatives. **Tables 2–8** and **2–9** show the estimated concentrations of air pollutants that would be expected during construction conducted under Alternatives 3 and 4, respectively. Since no specific site has yet been selected for the new accelerator[s] or the new research reactor, Federal standards are used in column 3 of the tables. As discussed in Section 4.5.1.1.3, the effects of constructing the new high-energy accelerator were used to characterize air quality impacts under Alternative 3 (Construct New Accelerator[s]). Construction impacts of the low-energy accelerator and support facilities would add relatively small concentrations to those shown in column 4 of Table 2–8. If Alternative 3 and/or Alternative 4 were selected for implementation, site-specific environmental documentation would be prepared prior to site selection.

The negative increment in water use would be more than offset by the increase in water use estimated for Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor).

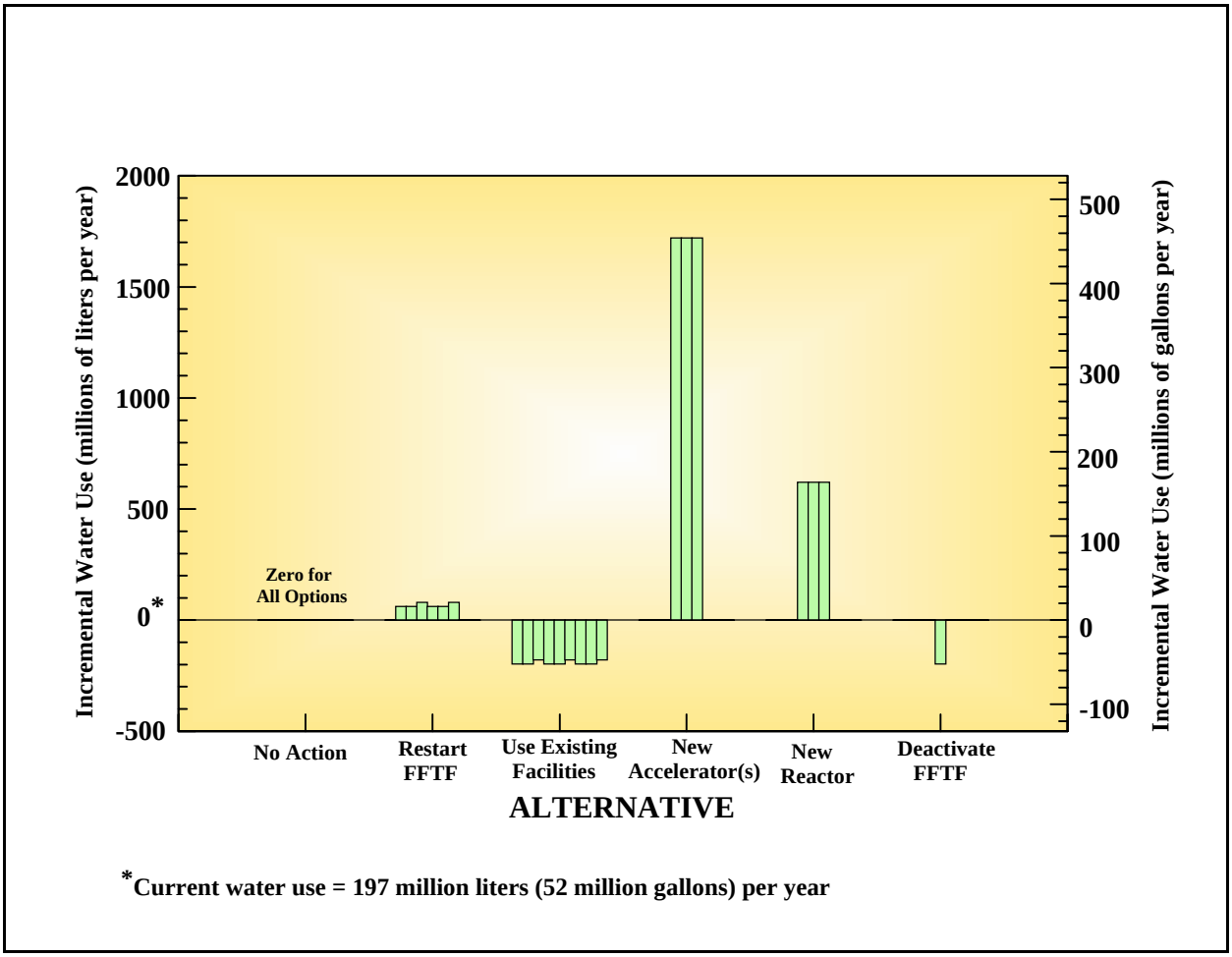


Figure 2–26 Annual Water Use Under the Nuclear Infrastructure Alternatives

Table 2–8 Air Pollutant Concentrations Resulting from Construction of a High-Energy Accelerator Under All Options of Alternative 3

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (microgram per cubic meter)
Carbon monoxide	8 hours 1 hour	10,000 40,000	436 623
Nitrogen oxide	Annual	100	42
PM_{10}	Annual 24 hours	50 150	3 69
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	3 64 143

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

Source: Modeled increments are based on SCREEN3 computer code (EPA 1995); data from TechSource 2000.

Table 2–9 Air Pollutant Concentrations Resulting from Construction of a New Research Reactor Under All Options of Alternative 4

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (microgram per cubic meter)
Carbon monoxide	8 hours 1 hour	10,000 40,000	72 103
Nitrogen oxide	Annual	100	1
PM_{10}	Annual 24 hours	50 150	3 88

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standards.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); data from Appendix E. Missions]).

As shown in Tables 2–8 and 2–9, construction of the new irradiation and support facilities would not be expected to exceed Federal standards and guidelines for ambient air quality. However, in comparison with air pollutant concentrations expected from facility operations (discussed in Section 2.7.1.5.2 below), concentrations of air pollutants that would be expected during construction are relatively large. If the new facilities were constructed in an area with existing high background concentrations, construction activities could produce enough air pollutant emissions to exceed ambient air quality standards

OPERATIONS

No Action Alternative

Under the No Action Alternative (Section 2.5.1), FFTF would remain in standby and DOE's nuclear infrastructure would not be enhanced to meet the nuclear infrastructure missions described in Section 1.2. Air quality effects that would be expected from transportation of neptunium-237 oxide to REDC (Option 2), FDPF (Option 3), or FMEF (Option 4) are summarized in Section 2.7.1.6.

Alternatives 1 through 5

Oak Ridge Reservation. Under Alternatives 1 (Options 1 and 4), 2 (Options 1, 4, and 7), 3 (Option 1), and 4 (Option 1), air quality impacts at ORR would result from the production of plutonium-238 at REDC. Concentrations of air pollutants that would be expected from facility operations conducted under these alternatives and options are shown in column 4 of **Table 2–10**. All of the expected concentrations are small in comparison with the most stringent ambient air quality standards shown in column 3 of Table 2–11. Operation of REDC in support of plutonium-238 production would not be expected to significantly affect air quality or to result in air pollutant concentrations in excess of ambient air quality standards. No air quality impacts would result from operation of HFIR under Alternative 2 (Use Only Existing Operational Facilities).

Idaho National Engineering and Environmental Laboratory. Under Alternatives 1 (Options 2 and 5), 2 (Options 2, 5, and 8), 3 (Option 2) and 4 (Option 2), air quality impacts at INEEL would result from the production of plutonium-238 at FDPF. Concentrations of air pollutants that would be expected from facility operations conducted under these alternatives and options are shown in column 4 of **Table 2–11**. All of the expected concentrations are small in comparison with the most stringent ambient air quality standards shown in column 3 of the table. Operation of FDPF in support of plutonium-238 production would not be expected to significantly affect air quality or to result in air pollutant concentrations in excess of ambient air quality

Table 2–10 Air Pollutant Concentrations Expected from Operation of the Radiochemical Engineering Development Center Under Alternatives 1 Through 4

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (micrograms per cubic meter)
Nitrogen dioxide	Annual	100	1.99×10^{-4}
Sulfur dioxide	Annual	80	0.04
	24 hours	365	0.31
	3 hours	1,300	0.70

a. For comparison with ambient air quality standards.
Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); 40 CFR Part 50.

Table 2–11 Air Pollutant Concentrations Expected from Operation of the Fluorinel Dissolution Process Facility Under Alternatives 1 Through 4

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (micrograms per cubic meter)
Criteria pollutants			
Nitrogen dioxide	Annual	100	3.66×10^{-4}
Sulfur dioxide	Annual	80	0.024
	24 hours	365	0.19
	3 hours	1,300	0.43
Toxic air pollutants			
Methanol	24 hours	13,000	0.0048
Nitric acid	24 hours	250	0.0097
Paraffin hydrocarbons	24 hours	100	0.44
Tributyl phosphate	24 hours	110	0.25

a. For comparison with ambient air quality standards.
Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); 40 CFR Part 50; WDEC 1998.

standards. No air quality impacts would result from operation of ATR under Alternative 2 (Use Only Existing Operational Facilities).

Hanford Site. If Alternative 1 were selected for implementation, impacts on air quality at Hanford would result from operation of FFTF (all options), RPL (Options 1, 2, 4, and 5), and FMEF (Options 3 and 6). FMEF could also be used for production of plutonium-238 under Alternatives 2 (Options 3, 6, and 9), 3 (Option 3), and 4 (Option 3). Concentrations of air pollutants that would be expected from facility operations conducted under these alternatives and options are shown in **Table 2–12**. Numbers in the third row of Table 2–12 are the most stringent state or Federal standard for each averaging period. FFTF would be deactivated under Alternatives 2 through 5. Deactivation would, in turn, result in the shutdown of diesel-driven fire pumps, oil-fired preheaters, and a gas turbine that currently support FFTF's standby condition. If any of Alternatives 2 through 5 were selected for implementation, emissions from this supporting equipment would cease, thereby improving the air quality near FFTF. Emissions of air pollutants from FMEF are relatively small in comparison to those associated with FFTF supporting equipment. Therefore, all the air concentrations shown in Table 2–12 for Alternatives 2 through 5 are negative to represent an overall decrease in the emission of air pollutants.

Air quality concentrations for FFTF and FMEF were calculated with the SCREEN3 model developed by EPA. The model is intended to provide conservative estimates of the concentrations of air pollutants emitted from point or extended sources. Concentrations shown under Alternatives 2 through 5 were obtained by summing estimated emissions from the diesel-driven oil pumps, the oil-fired preheaters, and the gas turbine. Because

Table 2–12 Comparison Among Alternatives: Impacts on Criteria Air Pollutants at the Hanford Site

	Carbon Monoxide (micrograms per cubic meter)		Nitrogen Dioxide (micrograms per cubic meter)	PM_{10} (micrograms per cubic meter)		Sulfur Dioxide (micrograms per cubic meter)			
Averaging Period	8 hours	1 hour	Annual	Annual	24 hours	Annual	24 hours	3 hours	1 hour
Most Stringent Standard or Guideline[a]	10,000[b]	40,000[b]	100[b]	50[c]	150[c]	50[d]	260[d]	1,300[b]	660[d]
Options[e]	**No Action Alternative**								
All	0	0	0	0	0	0	0	0	0
Alternative 1: Restart FFTF									
1 & 4	52.1	74.4	0.012	8.4×10^{-4}	9.8	0.0008	9.1	20.5	22.8
2 & 5	52.1	74.4	0.012	8.4×10^{-4}	9.8	0.0008	9.1	20.5	22.8
3 & 6	52.1	74.4	0.012	8.4×10^{-4}	9.8	0.009	9.2	20.7	23.0
Alternative 2: Use Only Existing Operational Facilities									
1, 2, 4, 5, 7, & 8	-3.5	-5.1	-0.032	-0.002	-0.898	-0.164	-29.8	-67.0	-74.4
3, 6, & 9	-3.5	-5.1	-0.032	-0.002	-0.898	-0.155	-29.7	-66.8	-74.2
Alternative 3: Construct New Accelerator(s)									
1 & 2	-3.5	-5.1	-0.032	-0.002	-0.898	-0.164	-29.8	-67.0	-74.4
3	-3.5	-5.1	-0.032	-0.002	-0.898	-0.155	-29.7	-66.8	-74.2
Alternative 4: Construct New Research Reactor									
1 & 2	-3.5	-5.1	-0.032	-0.002	-0.898	-0.164	-29.8	-67.0	-74.4
3	-3.5	-5.1	-0.032	-0.002	-0.898	-0.155	-29.7	-66.8	-74.2
Alternative 5: Permanently Deactivate FFTF (with No New Missions)									
	-3.5	-5.1	-0.032	-0.002	-0.898	-0.164	-29.8	-67.0	-74.4

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The 24-hour PM_{10} (particulate matter with an aerodynamic diameter less than or equal to 10 micrometers) standard is attained when the expected number of days with a 24-hour average concentration above the standard is equal to or less than 1. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Federal and state standard.
c. Federal standard currently under litigation.
d. State standard.
e. For descriptions of the options under each alternative, see Section 2.5.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); additional data from Nielsen 2000.

these sources operate intermittently and do not necessarily operate at the same time, estimates of the concentrations of air pollutants shown in Table 2–12 are conservative because they were obtained under the assumption that all supporting equipment for FFTF would operate simultaneously, which is considered a worst-case scenario.

Generic Site for the New Accelerator(s). Under Alternative 3 (all options), air quality impacts at the site for the new accelerator(s) would result from the operation of emergency diesel generators for the high-energy accelerator and any support facilities. The low-energy accelerator would not require emergency diesel power, and it was assumed in the analysis that air quality effects of the low-energy accelerator could be ignored. Air quality impacts of the support facilities would be assessed if Alternative 3 (Construct New Accelerator[s]) were selected for implementation. Column 4 of **Table 2–13** shows estimated air pollutant concentrations at the

generic site that would result from operation of emergency diesel generators. In comparison with the air quality concentrations that would be expected during construction (Table 2–8), air quality impacts resulting from operation of the diesel generators would be relatively small. All of the expected concentrations resulting from operation of emergency generators would be small in comparison with the most stringent ambient air quality standards shown in column 3 of Table 2–13, and would not be expected to result in air pollutant concentrations in excess of ambient air quality standards. If the new accelerator(s) were located in an area that has high background pollutant concentrations, diesel emissions could result in pollutant concentrations in excess of the ambient standards. If Alternative 3 were selected for implementation, site-specific environmental documentation would be prepared prior to site selection.

Table 2–13 Air Pollutant Concentrations Expected from Operation of the Emergency Diesel Generators for the High-Energy Accelerator

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (micrograms per cubic meter)
Carbon monoxide	8 hours 1 hour	10,000 40,000	94 135
Nitrogen oxide	Annual	100	0.47
PM_{10}	Annual 24 hours	50 150	0.03 17.7
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.03 16.5 37.2

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); TechSource 2000.

Generic Site for the New Research Reactor. Under Alternative 4 (all options), air quality impacts at the site for the new research reactor would result from the operation of emergency diesel generators for the reactor. Column 4 of **Table 2–14** shows estimated air pollutant concentrations at the generic site that would result from operation of emergency diesel generators. In comparison with the air quality concentrations that would be expected during construction (Table 2–9), air quality impacts resulting from operation of the diesel generator would be relatively small. All of the expected concentrations resulting from operation of the emergency generator would be small in comparison with the most stringent ambient air quality standards shown in column 3 of the Table 2–14, and would not be expected to result in air pollutant concentrations in excess of ambient air quality standards. If the new research reactor were located in an area that has high background pollutant concentrations, diesel emissions could result in pollutant concentrations in excess of the ambient standards. If Alternative 4 were selected for implementation, site-specific environmental documentation would be prepared prior to site selection.

Table 2–14 Air Pollutant Concentrations Expected from Operation of the Emergency Diesel Generator for the New Research Reactor

Criteria Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (micrograms per cubic meter)
Carbon monoxide	8 hours 1 hour	10,000 40,000	89.5 128
Nitrogen oxide	Annual	100	0.198
PM_{10}	Annual 24 hours	50 150	0.0035 3.46
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.062 61.2 138

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); data from Appendix E.

2.7.1.5 Socioeconomics

As discussed in Chapter 4, implementation of the nuclear infrastructure alternatives would have no significant impact on regional economic areas or community services at Hanford, INEEL, and ORR. Socioeconomic impacts at the generic sites could not be evaluated in detail because areas potentially affected under Alternatives 3 and 4 could vary widely in demographic and economic composition. If Alternatives 3 or 4 were selected for implementation, site-specific environmental analysis would be conducted prior to site selection. **Table 2–15** shows the number of direct jobs that would be generated under implementation of the nuclear infrastructure alternatives. Deactivation of FFTF under Alternatives 2 through 5 would result in the loss of 242 jobs that are required to keep the facility in standby condition. That loss would be offset under alternatives and options for which FMEF would support the production of plutonium-238 (62 direct jobs).

2.7.1.6 Transportation Impacts

As stated in Section 2.4.2 and explained in Section 4.2.1.1, the transportation impacts for Option 1 of the No Action Alternative are those resulting from transporting 175 kilograms (385 pounds) (5 kilograms [11 pounds] per year for the 35-year evaluation period) of plutonium-238 from Russia to LANL. The impacts were obtained by extrapolating the impact analysis presented in the *Environmental Assessment of the Import of Russian Plutonium-238* (DOE 1993) for the purchase of 40 kilograms (88.2 pounds) of plutonium-238. The impacts presented for the other options of the No Action Alternative include those of Option 1 plus the impact from transporting neptunium oxide from SRS to the selected facilities at ORNL, INEEL, and Hanford. Because the assumptions and data used to assess the transportation impacts in the above environmental assessment are different from those used in this NI PEIS, incremental transportation impacts compared to the baseline condition (Option 1 of the No Action Alternative) can only be presented for the options under the No Action Alternative. Therefore, the transportation impacts presented in this section are not compared to the baseline condition.

Radiological and nonradiological transportation impacts over the 35-year program duration are summarized in **Table 2–16**. Risks to the public and workers due to incident-free transportation are shown in columns 3 through 5 of the table. Columns 6 and 7 summarize radiological and nonradiological risks to the public that could result from transportation accidents. Chapter 4 and Appendix J discuss transportation impacts in more detail.

Table 2–15 Comparisons Among Alternatives: Change in Direct Jobs Under the Nuclear Infrastructure Alternatives

Options[a]	Oak Ridge Reservation	Idaho National Engineering and Environmental Laboratory	Hanford Site	Generic Accelerator(s) Site(Construction/ Operation)	Generic Research Reactor Site(Construction/ Operation)
No Action Alternative					
All	0	0	0	0	0
Alternative 1: Restart FFTF					
1 & 4	41	0	218	0	0
2 & 5	0	24	218	0	0
3 & 6	0	0	292	0	0
Alternative 2: Use Only Existing Operational Facilities					
1, 4, & 7	41	0	-242	0	0
2, 5, & 8	0	24	-242	0	0
3, 6, & 9	0	0	-180	0	0
Alternative 3: Construct New Accelerator(s)					
1	41	0	-242	410/225	0
2	0	24	-242	410/225	0
3	0	0	-180	410/225	0
Alternative 4: Construct New Research Reactor					
1	41	0	-242	0	160/120
2	0	24	-242	0	160/120
3	0	0	-180	0.00	160/120
Alternative 5: Permanently Deactivate FFTF (with No New Missions)					
	0	0	-242	0	0

a. For descriptions of the options under each alternative, see Section 2.5.

RADIOLOGICAL TRANSPORTATION RISKS

Figure 2–27 illustrates the data listed in column 6 of Table 2–16. The results indicate a large risk to the public due to transportation accidents that could occur over 35 years under implementation of Alternatives 1 (Restart of FFTF), 3 (Construct New Accelerator[s]), and 4 (Construct New Research Reactor) as compared to those from implementation of Alternative 2 (Use Only Existing Operational Facilities). This large difference is due to the more than 8,000 medical isotope shipments by air transport considered under Alternatives 1, 3, and 4, and not under Alternative 2. As explained in Appendix J and the Transportation sections in Chapter 4, nearly all of the radiological and traffic accident risk are due to those involving medical and industrial isotope shipments. No enhancement of medical and industrial isotope production is considered under Alternative 2.

Implementation of Alternative 5 (Permanently Deactivate FFTF [with No New Mission]) would not result in any new transportation activities.

Table 2–16 Comparison Among Alternatives: Impacts of Transportation on Occupational and Public Health and Safety

Options[a]	Transportation Distance (millions of kilometers)	Incident-Free Transportation over 35 Years			Transportation Accidents over 35 Years	
		Public: Radiological (LCF)	Workers: Radiological (LCF)	Public: Vehicle Emissions (fatalities)	Public: Radiological (LCF)	Public: Vehicle Collisions[b] (fatalities)
No Action Alternative						
1	0.11	0.010	0.0046	4.7×10^{-4}	4.4×10^{-4}	0.014
2	0.13	0.011	0.0047	5.9×10^{-4}	4.4×10^{-4}	0.014
3	0.20	0.014	0.0049	8.9×10^{-4}	4.4×10^{-4}	0.014
4	0.22	0.014	0.0050	9.2×10^{-4}	4.4×10^{-4}	0.014
Alternative 1: Restart FFTF						
1 and 4	8.0	0.149	0.012	0.030	0.53	0.19
2 and 5	6.2	0.044	0.008	0.024	0.53	0.13
3 and 6	5.6	0.009	0.007	0.023	0.53	0.12
Alternative 2: Use Only Existing Operational Facilities						
1	2.2	0.120	0.005	0.0064	4.4×10^{-5}	0.059
2	0.15	0.004	0.001	0.0007	2.1×10^{-5}	6.0×10^{-4}
3	0.83	0.040	0.002	0.0014	3.0×10^{-5}	0.017
4	2.6	0.150	0.006	0.0056	4.4×10^{-5}	0.074
5	3.1	0.179	0.007	0.0066	2.1×10^{-5}	0.088
6	3.6	0.205	0.008	0.0075	3.0×10^{-5}	0.100
7	1.8	0.096	0.004	0.0052	4.4×10^{-5}	0.048
8	0.99	0.052	0.002	0.0030	4.4×10^{-5}	0.024
9	1.6	0.084	0.004	0.0037	3.0×10^{-5}	0.039
Alternative 3: Construct New Accelerator(s)						
1	5.7	0.054	0.008	0.023	0.53	0.14
2	5.8	0.057	0.008	0.023	0.53	0.14
3	5.9	0.065	0.009	0.023	0.53	0.14
Alternative 4: Construct New Research Reactor						
1	7.5	0.154	0.011	0.026	0.53	0.19
2	7.5	0.157	0.012	0.026	0.53	0.19
3	7.9	0.177	0.012	0.027	0.53	0.19
Alternative 5: Permanently Deactivate FFTF (with No New Missions)						
	NA[c]	NA[c]	NA[c]	NA[c]	NA[c]	NA[c]

a. For descriptions of the options under each alternative, see Section 2.5
b. No radiological spill.
c. No new transportation activities would occur under Alternative 5.

Key: LCF, latent cancer fatalities.

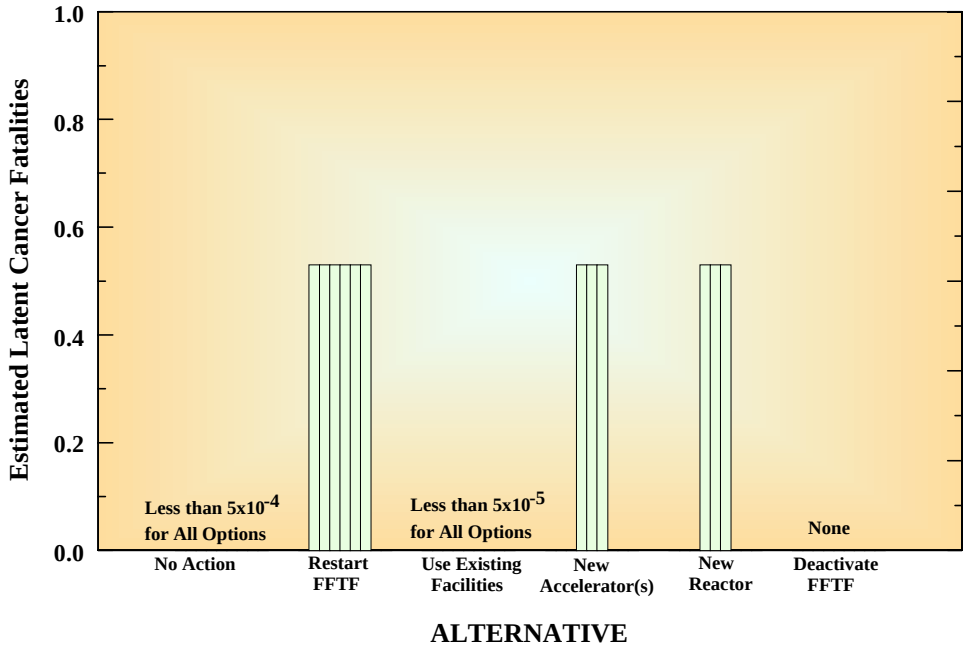


Figure 2–27 Public Risks Due to Radiological Transportation Accidents (35 Years)

Figure 2–28 shows the radiological risks to the public that could result from incident-free transportation over 35 years (column 3 of Table 2–16). For all of the alternatives and options, incident-free radiological transportation risks are approximately 0.2 latent cancer fatality over 35 years. As shown in column 4 of Table 2–16, radiological risks to workers due to incident-free transportation are less than approximately 0.012 latent cancer fatality for all alternatives and options.

NONRADIOLOGICAL TRANSPORTATION RISKS

Column 7 of Table 2–16 shows the risks of traffic fatalities that would be expected to result from vehicular collisions in which there is no radiological spill. Under all alternatives and options, the expected number of traffic fatalities would be less than approximately 0.2. Data listed in column 5 of the same table indicates that less than approximately 0.03 fatality would be expected from vehicular exhaust emissions. Fatalities that would be expected to result from both vehicular collisions and exhaust emissions are closely correlated with the estimated highway mileage that would be traveled under implementation of the alternatives (see column 2 of Table 2–16 and **Figure 2–29**). As discussed in Appendix J, traffic accident rates depend on the type of carrier. Both commercial trucks and DOE's SST/SGTSs would be used for the highway transport of isotopes. Accident rates for the safe, secure trailer system are less than those for commercial trucks by at least a factor of five. As a result, expected collision fatalities for any option would increase the total distance traveled, but the impacts would also depend on relative amounts of transportation by commercial truck and the SST/SGTs.

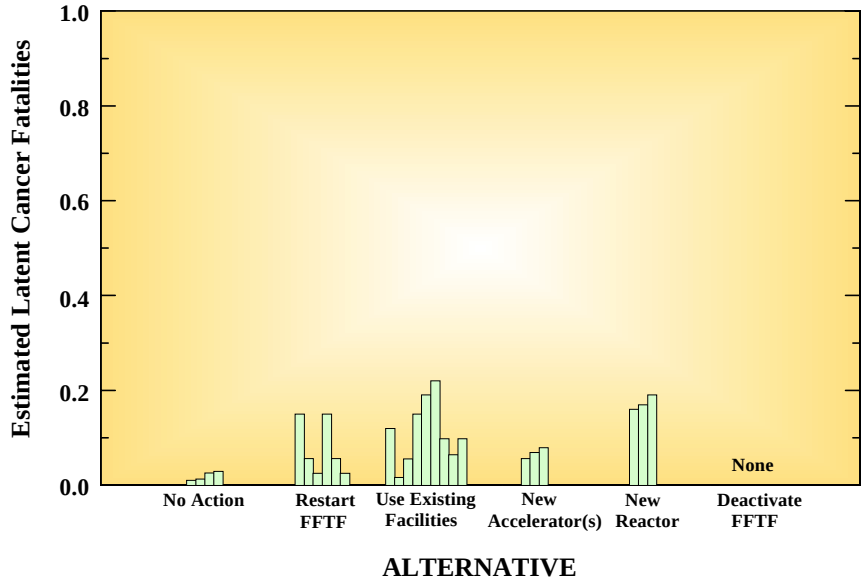


Figure 2–28 Radiological Risks to the Public Due to Incident-Free Transportation (35 Years)

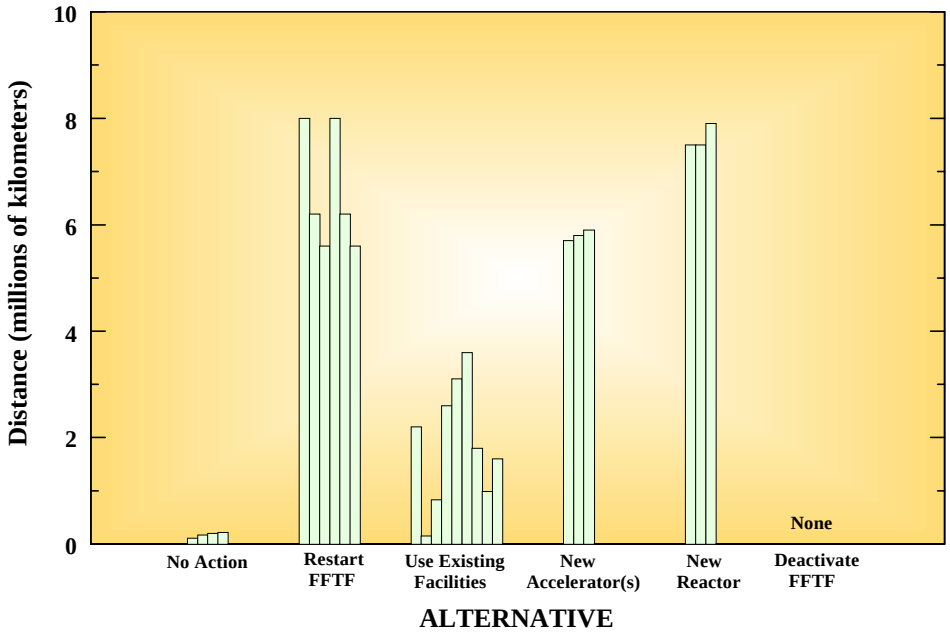


Figure 2–29 Highway Distances That Would Be Traveled Under the Alternatives (35 Years)

2.7.1.7 Resource Areas Discussed in Less Detail

As discussed in Chapter 4, implementation of the nuclear infrastructure alternatives at existing candidate sites would be expected to have little effect on land use, visual resources, noise, water quality, geology and soils, ecology, cultural resources, and environmental justice. Implementation of the alternatives at one or more generic sites could potentially result in significant impacts in one or more of these resource areas. However, these impacts are site-specific and could not be evaluated in detail in this programmatic document. If Alternative 2 (Options 4, 5, and 6), 3, or 4 were selected for implementation, site-specific environmental documentation would be prepared prior to site selection.

2.7.1.7.1 Land Use

Implementation of the nuclear infrastructure alternatives at existing operational candidate sites at Hanford, INEEL, and ORR would be consistent with ongoing activities and current land use at these sites. Irradiation of neptunium targets at an existing CLWR would also be consistent with the land use at the reactor site. If Alternatives 3 or 4 were selected for implementation, a site-specific evaluation of land use would be conducted prior to site selection. Deactivation of the FFTF under Alternatives 2 through 5 would have no effect on ongoing land use in the 400 Area of Hanford.

2.7.1.7.2 Visual Resources

Existing sites that are candidates for implementation of the nuclear infrastructure alternatives are rated Class IV under the U.S. Bureau of Land Management classification guidelines for visual resources (DOI 1986). Selection of one or more of the existing candidate sites for implementation would not affect their visual resource classification as areas in which industrial development dominates the landscape. Use of a CLWR for irradiation of neptunium targets would not alter the appearance of the reactor or the surrounding landscape. Implementation of Alternative 3 (Construct New Accelerator[s]) or 4 (Construct New Research Reactor) could result in reclassification under U.S. Bureau of Land Management guidelines. If Alternative 3 or 4 were selected for implementation, a site-specific evaluation of visual resources would be conducted prior to site selection. Deactivation of FFTF under Alternatives 2 through 5 would not significantly alter the overall landscape in the 400 Area of Hanford.

2.7.1.7.3 Noise

Noise associated with target fabrication and processing and irradiation at existing candidate sites would be similar to currently existing onsite noise and would not be audible beyond site boundaries. These activities would not produce sudden, loud noises that would startle wildlife. Noise levels that would be generated at a CLWR under Alternative 2 (options 4, 5, and 6) would be the same as those currently existing at the reactor site. Implementation of Alternative 3 (Construct New Accelerator[s]) or 4 (Construct New Research Reactor) would result in construction activities that could disturb nearby residents or wildlife. If Alternative 3 or 4 were selected for implementation, a site-specific NEPA review would be prepared, and an evaluation of potential noise impacts would be conducted prior to site selection. Deactivation of FFTF under Alternative 5 would not significantly alter the noise levels in the 400 Area of Hanford.

2.7.1.7.4 Water Quality

Under Alternative 1 (Restart FFTF), there would be no liquid radiological effluent pathways to the environment from FFTF. Process wastewater from cooling tower blow-down would be ultimately discharged to the 400 Area Pond (i.e., the 4608 B/C percolation ponds). No impact on the quality of ground or surface

water would be expected. Irradiation of neptunium targets at existing reactors and a generic CLWR would have no measurable effect on the quantity or quality of discharged effluents. Use of existing facilities for target fabrication and processing would not result in direct effluent discharge to the environment, and additional wastewater generation would be relatively small in comparison to existing wastewater treatment volumes at the sites. If Alternative 3 (Construct New Accelerator[s]) or 4 (Construct New Research Reactor) were selected for implementation, construction and operation of new facilities would not be anticipated to significantly impact water quality. While the water quality impacts are expected to be small, a site-specific environmental evaluation of potential water quality impacts and mitigation measures would be conducted prior to site selection. Sodium removal during deactivation of FFTF under Alternatives 2 through 5 would result in approximately 7,600 liters (2,000 gallons) of wastewater that would be disposed of in existing wastewater treatment facilities at Hanford. Deactivation of FFTF would not be expected to impact water quality.

2.7.1.7.5 Geology and Soils

Except for Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor), activities conducted under the nuclear infrastructure alternatives would not require construction of new facilities. No soil would be disturbed, and there would be no impacts on the geology of potentially affected sites. Construction of new accelerators and support facilities under Alternative 3 would be expected to disturb up to approximately 27 hectares (66 acres) of soil. If Alternative 4 were selected for implementation, construction of the new reactor and support facility would be expected to disturb approximately 4 hectares (10 acres) of soil. If Alternative 3 or 4 were selected for implementation, a site-specific environmental evaluation would be conducted prior to site selection. Deactivation of FFTF under Alternatives 2 through 5 would take place on previously disturbed land. Impacts of deactivation on geology and soils would be negligible.

2.7.1.7.6 Ecology

Activities that would be conducted under the nuclear infrastructure alternatives at candidate existing facilities and the generic CLWR would not involve construction of new facilities or significant changes in traffic, noise, air quality, or water quality. In addition, irradiation and processing activities would take place in established industrial areas. Impacts on terrestrial resources and wetlands would be negligible.

Under Alternatives 3 (Construct New Accelerator[s]) and 4 (Construct New Research Reactor), construction of new facilities at a yet-to-be-determined site could potentially have a significant effect on wildlife and wetlands. If Alternative 3 or 4 were selected for implementation, site-specific ecological evaluations would be conducted prior to site selection. The evaluation would include consultation with the U.S. Fish and Wildlife Service and appropriate state authorities concerning threatened and endangered species. Deactivation of FFTF under Alternatives 2 through 5 would take place on previously disturbed land in the 400 Area. No threatened or endangered species are known to reside in the 400 Area, and noise impacts on local wildlife would be temporary.

2.7.1.7.7 Cultural Resources

Existing candidate facilities that would host activities under the nuclear infrastructure alternatives are located within areas that contain National Historic Landmarks or structures that are eligible for nomination to the National Register of Historic Places. Several candidate facilities are eligible for nomination to the National Register, including the Reactor Containment Building and the Control Building for FFTF at Hanford, RPL at Hanford, and ATR at INEEL. Selection of these facilities to support the nuclear infrastructure missions would not alter their eligibility.

Under the nuclear infrastructure alternatives, activities at candidate existing sites and the generic CLWR would be conducted within existing facilities. Use of the FMEF at Hanford for target fabrication and processing would require construction of a 76-meter-high (250-feet-high) stack on previously disturbed land. Similarly, construction of a support facility for deactivation of the FFTF would take place on previously disturbed land in the 400 Area. Thus, except for Alternatives 3 (Construct New Accelerator[s] and 4 (Construct New Research Reactor), no disturbance of archeological resources would be expected under the nuclear infrastructure alternatives.

Implementation of Alternative 3 or 4 would require construction on potentially undisturbed lands. If Alternative 3 or 4 were selected for implementation, a site-specific NEPA review would be prepared, and an environmental evaluation of cultural resources would be conducted prior to site selection. The evaluation would include consultation with State Historic Preservation Offices and potentially affected Native American tribes.

2.7.1.7.8 Environmental Justice

The objective of the environmental justice analysis was to determine whether or not implementation of the nuclear infrastructure alternatives would result in significant environmental impacts that disproportionately affect low-income or minority populations. As discussed throughout Chapter 4, normal operations at the candidate sites and incident-free transportation pose no significant radiological risks to the public or to maximally exposed offsite individuals among the public.

Portions of the Fort Hall Indian Reservation and the Yakama Indian Reservation lie within potentially affected areas surrounding INEEL and Hanford, respectively. As discussed in Appendixes H and I, calculations of radiological risks considered human exposures due to inhalation and ingestion of radioactive materials. Ingestion of contaminated fish, vegetation, and/or wildlife is an environmental justice consideration due to potential patterns of subsistence consumption for minority or low-income populations (CEQ 1997:sec 4-4). Radiological health models used in the environmental evaluation assumed accidents at the irradiation facilities or the fabrication and processing facilities would contaminate all of the food produced in the area, and that all of the contaminated food would be consumed by persons residing in the potentially affected area. As discussed in Sections K.5.1 through K.5.3, the expected risk that would result from ingestion of radiologically contaminated food for persons residing near Hanford would be approximately 0.004 latent cancer fatality and essentially zero for persons residing near the INEEL or ORR. Thus, no credible pattern of food consumption would be expected to result in a significant health risk to low-income or minority populations residing within potentially affected areas surrounding the existing candidate sites. As explained in various parts of Section 2.7.1 and detailed in Chapter 4, implementation of the alternatives would not be expected to result in significant environmental impacts in any of the environmental resource areas. Thus, no disproportionately high and adverse impacts on minority and low-income populations would be expected to result from implementation of the alternatives.

As discussed in Chapter 4, accidents at candidate fabrication and processing facilities and during transportation of radioisotopes by aircraft were found to pose the largest risks to the public. Under conservative assumptions described in Appendix I, no latent cancer fatalities due to accidents would be expected at the existing sites. Accidents during air transport of radioisotopes could occur anywhere along the flight path and would not place any identifiable group within the general population at disproportionate risk.

The density and distribution of total, low-income, and minority populations varies from site to site, so that evaluations of environmental justice are necessarily site-specific. If Alternatives 3 (Construct New Accelerator[s]) or 4 (Construct New Research Reactor) were selected for implementation, a site-specific NEPA

review would be prepared, and an evaluation of environmental justice would be conducted prior to site selection. The evaluation would include patterns of food consumption that could result in disproportionately high and adverse effects on low-income or minority populations at risk.

2.7.1.8 Industrial Safety

Estimates of potential industrial impacts to workers during construction, irradiation, fabrication and processing were evaluated based on DOE and Bureau of Labor Statistics data. Impacts are classified into two groups: total recordable cases and fatalities. A recordable case includes work-related death, illness, or injury which resulted in loss of consciousness, restriction of work or motion, transfer to another job, or required medical treatment beyond first aid. The industrial safety evaluation is discussed in more detail in Section I.3.

The average occupational total recordable cases and fatality rates for construction and operation activities are presented in **Table 2–17**.

Table 2–17 Average Occupational Total Recordable Cases and Fatality Rates (per worker-year)

Labor Category	Total Recordable Cases	Fatalities
Construction	0.053	1.3×10^{-4}
Operation	0.033	1.3×10^{-5}

The expected impacts (both annual and for the duration of the activity) to workers at each facility for construction and operation are presented in **Table 2–18**.

Table 2–18 Industrial Safety Impacts from Construction and Operation

Facility	Estimated Number of Workers	Construction or Operation Duration (years)	Expected Annual Total Recordable Cases	Expected Activity Duration Total Recordable Cases	Annual Fatalities	Activity Duration Fatalities
Construction						
Low-energy accelerator	75	3	4.0	12	0.010	0.030
High-energy accelerator	410	5	22	110	0.057	0.285
New research reactor	160	7	8.5	59.5	0.022	0.154
Operation						
ATR[a]	0	35	–	–	–	–
HFIR[a]	0	35	–	–	–	–
CLWR[a]	0	35	–	–	–	–
FFTF	242	35	8.0	280	0.0031	0.109
Low-energy accelerator	13	35	0.4	14	1.7×10^{-4}	0.00595
High-energy accelerator	225	35	7.4	259	0.0029	0.102
New research reactor	120	35	4.0	140	0.0016	0.056
REDC	116	35	3.8	133	0.0015	0.0525
FDPF	75	35	2.5	87.5	9.8×10^{-4}	0.0343
FMEF	105	35	3.5	123	0.0014	0.049
RPL/306–E	30	35	1.0	35	3.9×10^{-4}	0.0137
New support facility	100	35	3.3	116	0.0013	0.0455

a. No additional workers would be required for the proposed activities evaluated in this NI PEIS.

No fatalities would be expected from either construction or operation of any facility.

2.7.2 Implementation Schedule

The implementation schedules for the alternatives in this NI PEIS are presented in **Figures 2–30** through **2–35**.

No Action Alternative

The implementation schedule for the No Action Alternative is shown in **Figure 2–30**. As indicated, the design and construction for the storage facilities would start during fiscal years 2001 and 2002 and would be completed during fiscal year 2004. Neptunium-237 shipments from SRS would take place during fiscal years 2005 to 2007. The purchase of plutonium-238 from Russia would start during fiscal year 2001.

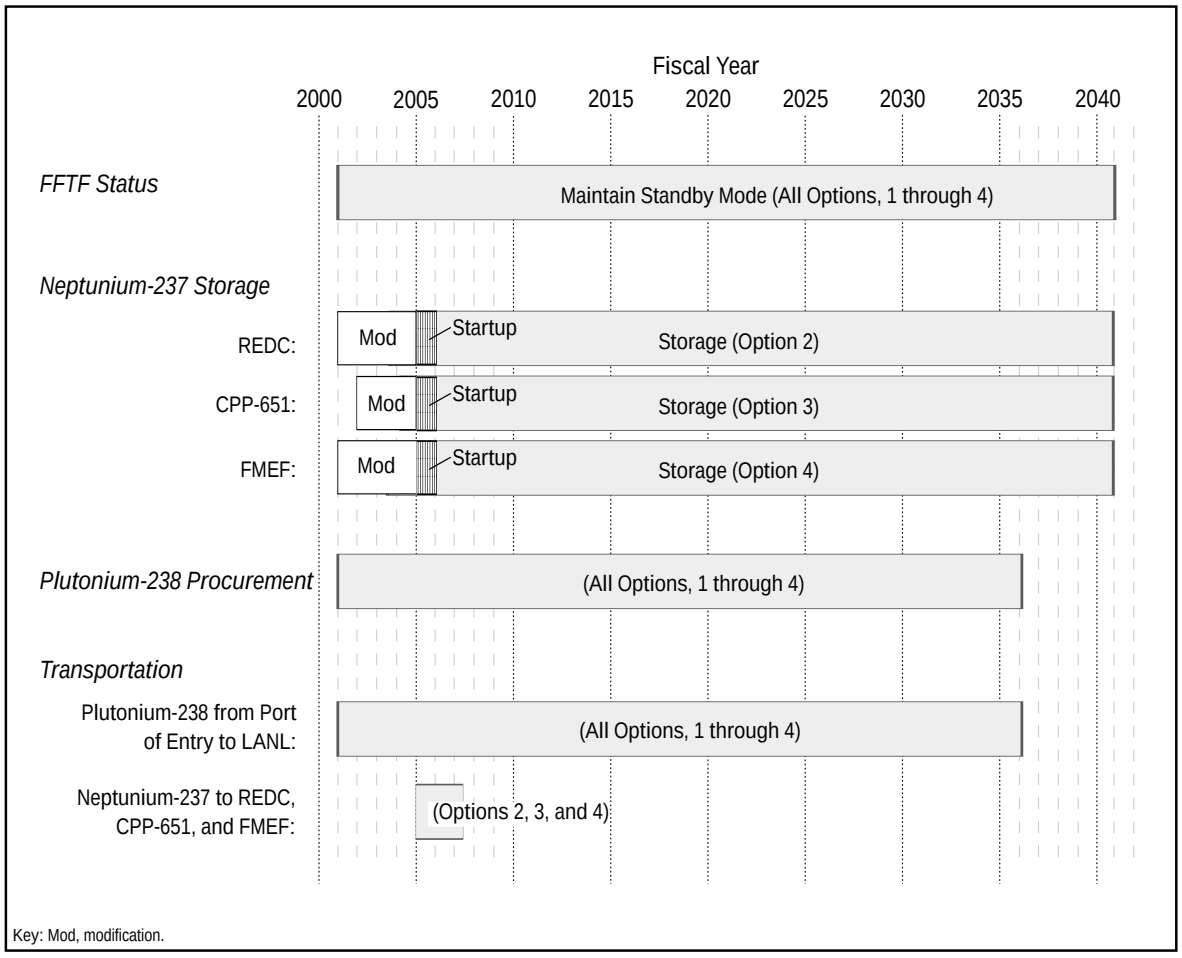


Figure 2–30 Implementation Schedule for No Action Alternative

Alternative 1—Restart FFTF

The planned implementation schedule for Alternative 1 is shown in **Figure 2–31.** As indicated, facility modification, design, and construction at target fabrication and processing facilities would take place during fiscal years 2001 to 2005.

Figure 2–31 Implementation Schedule for Alternative 1

As shown in Figure 2–31, target fabrication and processing at REDC, FDPF, and FMEF were assumed to begin in fiscal year 2006 and would continue through fiscal year 2040 in conjunction with target irradiation at FFTF. Target testing and evaluation and facility testing and startup were assumed to take place in fiscal year 2005 at FFTF.

Alternative 2—Use Only Existing Operational Facilities

The planned implementation schedule for Alternative 2 is shown in **Figure 2–32.** As indicated, facility modification, design, and construction at storage and target fabrication and processing facilities would take place during fiscal years 2001 to 2004. It was assumed that reactor facilities would not require any modifications to irradiate targets.

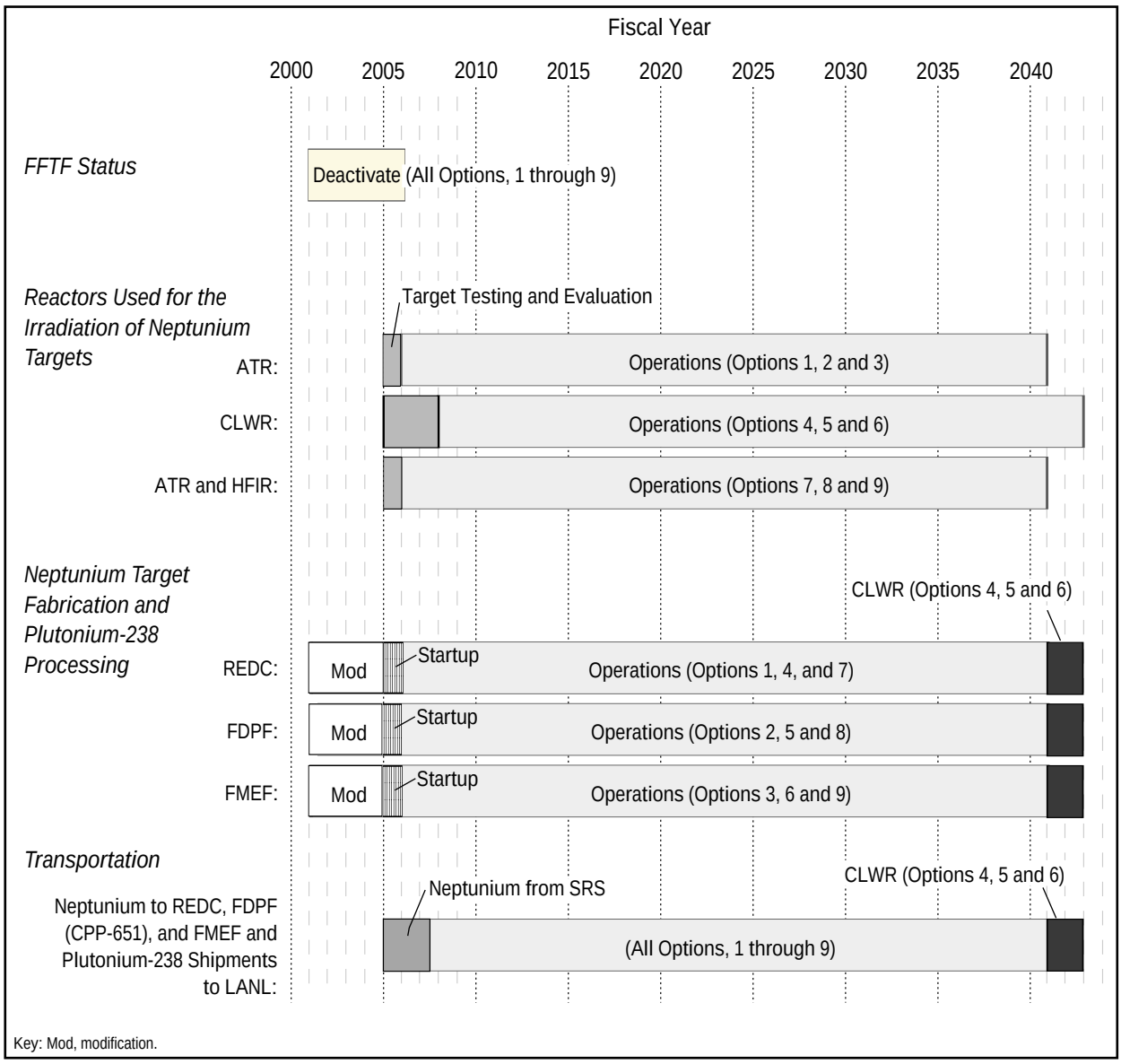


Figure 2–32 Implementation Schedule for Alternative 2

As shown in Figure 2–32, target fabrication and processing at REDC, FDPF, and FMEF were assumed to begin in fiscal year 2006 and would continue through fiscal year 2040 in conjunction with target irradiation at HFIR and/or ATR, and through fiscal year 2042 in conjunction with target irradiation at the CLWR. Target testing and evaluation and facility testing and startup were assumed to begin in fiscal year 2005 for each of the nuclear reactors and to continue to fiscal year 2006 at HFIR and/or ATR, and to fiscal year 2008 at the CLWR. Irradiation operations at nuclear reactor facilities would occur from fiscal year 2006 through fiscal year 2040 at HFIR and/or ATR, and from fiscal year 2008 through fiscal year 2042 at the CLWR.

Deactivation of FFTF would begin in fiscal year 2001 and continue through fiscal year 2006.

Alternative 3—Construct New Accelerator(s)

The planned implementation schedule for Alternative 3 is shown in **Figure 2–33**.

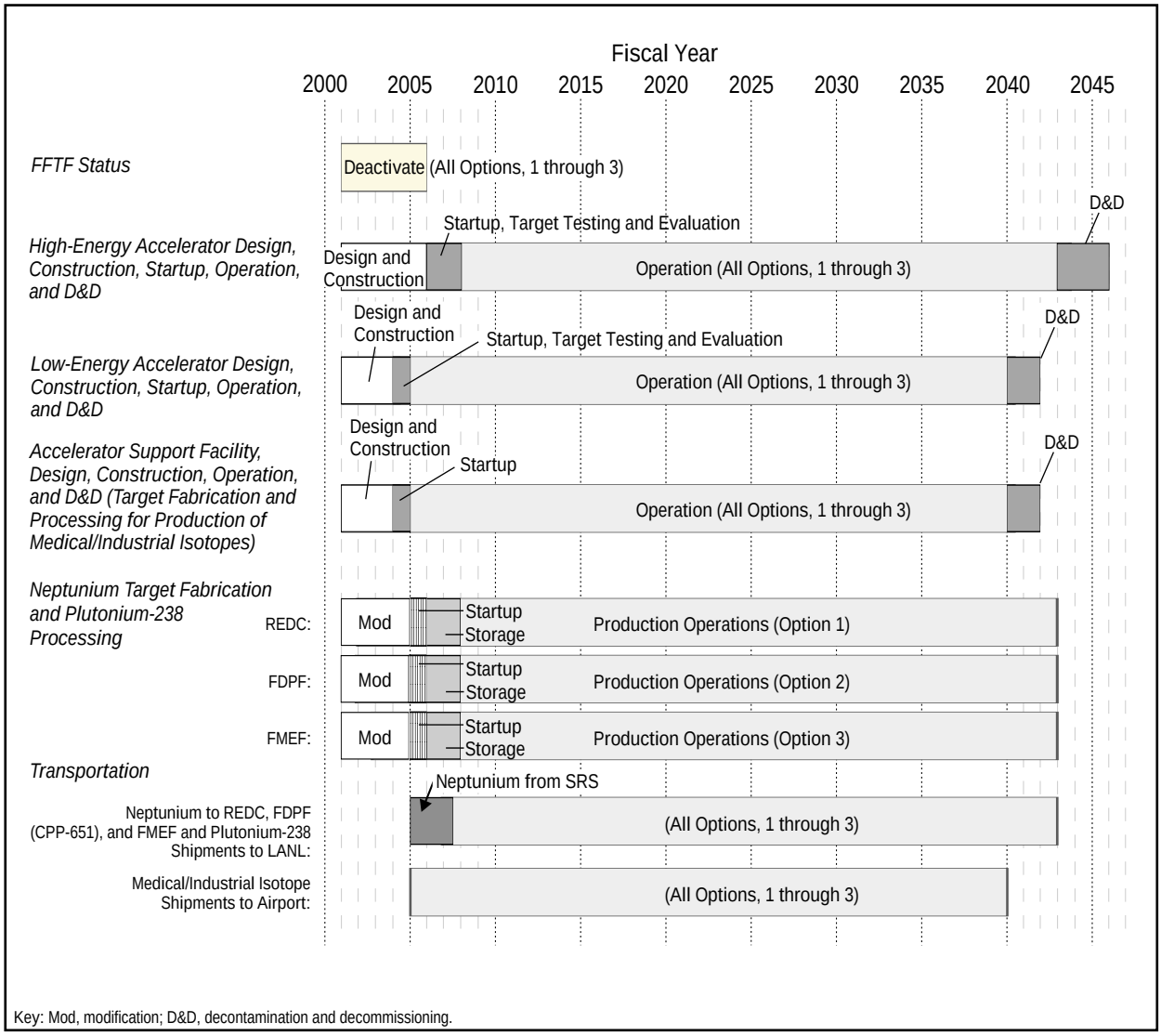


Figure 2–33 Implementation Schedule for Alternative 3

The new low-energy accelerator design and construction were assumed to begin fiscal year 2001 and continue to 2004, startup would be completed during fiscal year 2004, and full operation would commence fiscal year 2005. The new support facility schedule is driven by the reactor schedule and has similar schedule milestones. The new low-energy accelerator and support facility would be decontaminated and decommissioned within 2 years after completion of the missions.

As shown in Figure 2–33, the new high-energy accelerator design and construction were assumed to begin fiscal year 2001 and continue through 2006. The 2-year startup period would be completed during fiscal year 2007, and full operation would commence fiscal year 2008. The neptunium-238 target fabrication and processing facility (REDC, FDPF, or FMEF) modification design would take place during fiscal years 2001

to 2004 and start receiving neptunium-237 from SRS during fiscal year 2005. These facilities would be in full operation supporting the new high-energy accelerator irradiation of the neptunium-237 targets during fiscal year 2008. The new high-energy accelerator would be decontaminated and decommissioned within 3 years after completion of the mission.

Deactivation of FFTF would begin in fiscal year 2001 and continue through fiscal year 2006.

Alternative 4—Construct New Research Reactor

The planned implementation schedule for Alternative 4 is shown in **Figure 2–34**.

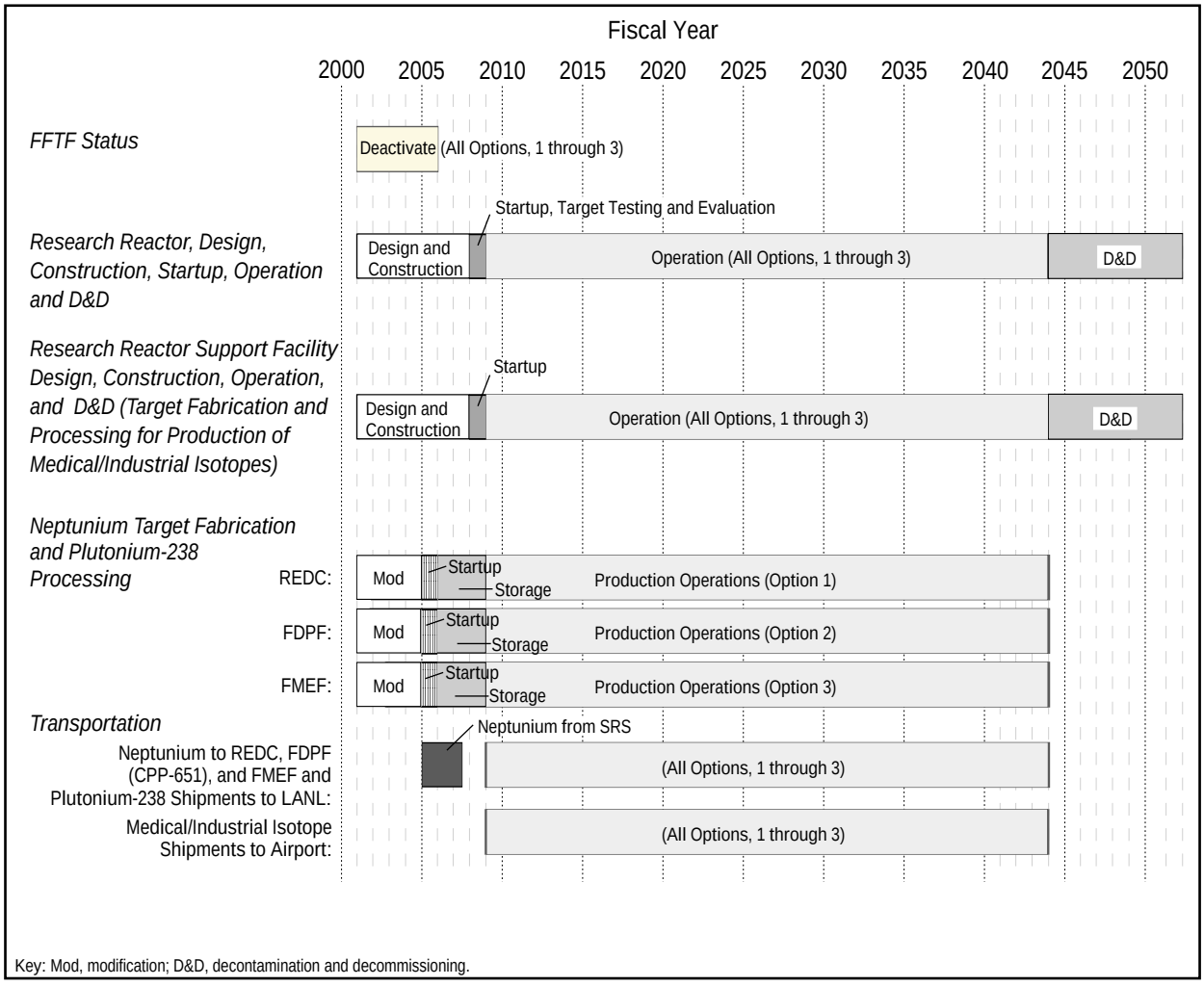


Figure 2–34 Implementation Schedule for Alternative 4

The new research reactor design and construction were assumed to begin fiscal year 2001 and continue to 2008, startup would be completed during fiscal year 2008, and full operation would commence fiscal year 2009. The new support facility schedule is driven by the reactor schedule and has similar schedule milestones.

As shown in Figure 2–34, the neptunium-238 target fabrication and processing facility (REDC, FDPF, or FMEF) modification design would take place during fiscal years 2001 to 2004 and start receiving neptunium-237 from SRS during fiscal year 2005. These facilities would be in full operation supporting the new reactor irradiation of the neptunium-237 targets during fiscal year 2009. The new research reactor and support facility would be decontaminated and decommissioned within 8 years after completion of the missions.

Deactivation of FFTF would begin in fiscal year 2001 and continue through fiscal year 2006.

Alternative 5—Permanently Deactivate FFTF (with No New Missions)

The planned implementation schedule for Alternative 5 is shown in **Figure 2–35.** As indicated, deactivation of FFTF would take place over 5 years (Battelle 1999).

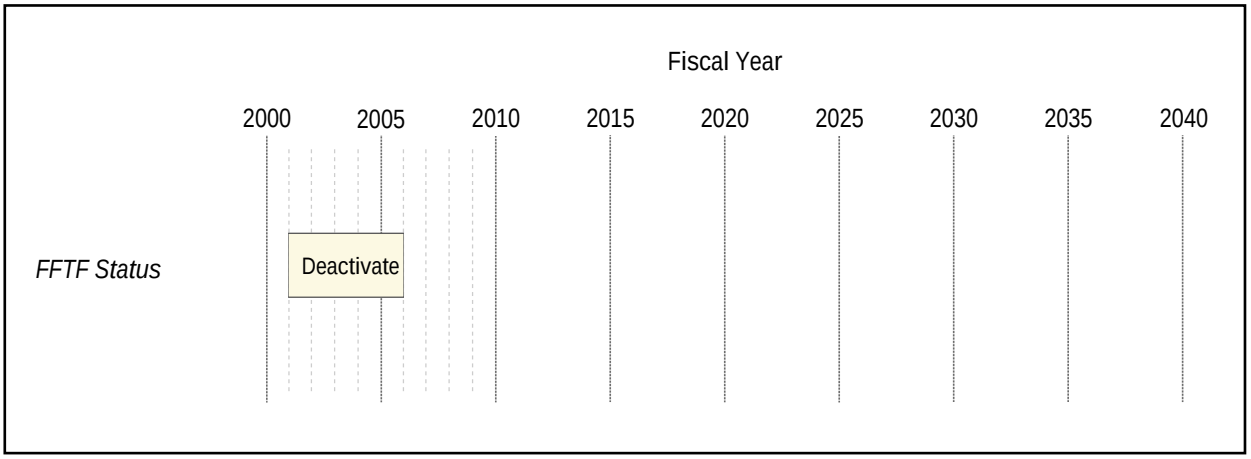


Figure 2–35 Implementation Schedule for Alternative 5

2.7.3 Comparison of Mission Effectiveness Among Alternatives

This section compares the effectiveness of Alternatives 1, 2, 3, and 4 in supporting the three missions evaluated in this NI PEIS:

- Medical and industrial isotope production
- Plutonium-238 production to support NASA space missions
- Nuclear energy research and development for civilian applications

Table 2–19 lists the medical isotopes that were included in the Expert Panel's forecast of future demands (Wagner et al. 1998), and identifies their means of production using accelerators, reactors, or separation from existing stockpiles of radioisotopes. Consistent with the panel's report, the list of isotopes is presented in three categories: proven medical isotopes currently used in clinical applications, those under development for clinical applications, and radioisotopes that have shown promise during medical research. Some are most suited for production in an accelerator, some in a nuclear reactor, and some are harvested by chemical separation from existing stockpiles of long-lived radioactive isotopes. Those isotopes that can be harvested from existing stockpiles of radioactive isotopes require only hot cells for the extraction process; neither accelerators or nuclear reactors are necessary for their production.

No single production method would satisfy all of the Expert Panel's projected requirements for medical isotopes. Isotopes produced by neutron capture are typically provided by a reactor, but could be produced by

Table 2–19 Medical Isotopes and Their Means of Production

Isotope[a]	Accelerator-Produced	Reactor-Produced	Separation from Existing Stockpiles of Radioactive Isotopes
Proven Isotopes Currently Used in Clinical Applications That Face Supply and Cost Concerns			
Yttrium-90	(b)	●	
Molybdenum-99[c]	(b)	●	
Indium-111	●		
Iodine-123	●		
Rhenium-186	(b)	●	
Developmental Isotopes for Clinical Applications That Face Availability and Cost Concerns			
Fluorine-18	●		
Phosphorus-32	(b)	●	
Krypton-81m	●		
Strontium-89	(b)	●	
Palladium-103	(b)	●	
Tin-117m	(b)	●	
Xenon-127	(b)	●	
Iodine-125	(b)	●	
Iodine-131	(b)	●	
Samarium-153	(b)	●	
Promising Research Isotopes That Are Not Being Explored Due to Lack of Availability or Cost			
Scandium-47	(b)	●	
Zinc-62	●		
Copper-64	●	●	
Copper-67	●	●	
Germanium-68	●		
Gadolinium-153	(b)	●	
Holmium-166	●	●	
Lutetium-177	(b)	●	
Rhenium-188	(b)	●	
Astatine-211	α		
Bismuth-212		●	●[d]
Bismuth-213	(b)	●	●[e]
Radium-223	(b)	●	●[f]

a. Wagner et al. 1998.
b. These isotopes are produced by neutron capture and could be produced in a high-energy accelerator. However, this capability has not been included in the design, analysis, or cost estimates of Alternative 3.
c. Sufficient supplies of this isotope are available from Canadian suppliers.
d. Bismuth-212 is a progeny of thorium-232.
e. Bismuth-213 is a progeny of uranium-233.
f. Radium-233 is a progeny of protactinium-231.

Key: α, efficient means of production with an alpha particle accelerator; ●, efficient means of production.

a high-energy accelerator with a spallation neutron source. Accelerator production of these isotopes would be relatively inefficient, and might not be practical to provide the large quantities needed to meet clinical demands. The proposed high-energy accelerator described in this NI PEIS could be modified to provide such capability, but this would add to the design, construction, and operating complexity, would require an increase in particle energy greater than 1 gigaelectron volts, and would increase the capital and operating costs.

Bismuth and radium isotopes, which were identified as promising medical isotopes by the Expert Panel, are currently harvested from existing stockpiles of long-lived radioisotopes and can also be readily produced in a reactor.

Alternative 1—Restart FFTF

FFTF would produce high-energy neutrons and a large flux level (10^{15} neutrons per square centimeter per second) that can be tailored to nearly any desired energy level. FFTF would provide the greatest flexibility for both isotope production and nuclear-based research and development among the baseline configurations for all of the proposed alternatives. Due to its large core size, flux spectrum, demonstrated testing capability, and rated power level, it would be able to concurrently support the projected plutonium-238 needs, production of medical and industrial isotopes, and civilian nuclear energy research and development related to a broad range of materials, advanced reactors, advanced fuels, and waste transmutation.

Alternative 2—Use Only Existing Operational Facilities

Due to current mission commitments at the existing DOE facilities, a large portion of the reactor irradiation space is committed to existing users. The existing reactors are able to provide for the current plutonium-238 needs. However, fulfilling this requirement with these facilities would use most, if not all, excess capacity, and may require some non-Federal missions to be terminated. The ability to expand the medical and industrial isotope production would require some current missions to be postponed or terminated. If the CLWR were used for plutonium-238 production, then the existing facilities would gain additional margin for medical and industrial isotope production and limited civilian nuclear energy research and development activities. These facilities have primary missions with sponsors who reserve the right to dictate to what degree and the times the facility could be used.

Alternative 3—Construct New Accelerator(s)

One or two accelerators, a low-energy accelerator and/or a high-energy accelerator, are proposed for Alternative 3. The low-energy accelerator would serve as a dedicated isotope production facility. Due to the nature of this type of accelerator, it could only produce a limited number of the isotopes listed in Table 2–19, it has no ability to satisfy the plutonium-238 needs, and a limited ability to support the proposed nuclear-based research and development needs. The preconceptual design of the high-energy accelerator presented in Appendix F focused on supporting the plutonium-238 production mission. The design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce "tuneable" neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and material interactions. The changes required to add additional capability to the high-energy accelerator could be provided, but they would increase the size of the facility, add complexity to the facility design and operation, increase the cost of construction and operation, and potentially require more time for design and construction.

Alternative 4—Construct New Research Reactor

The proposed new research reactor would provide ample neutrons for the production of plutonium-238 and for many of the isotopes listed in Table 2–19. The thermal flux would limit the new research reactor's ability to produce a number of isotopes requiring fast or high-energy neutrons. Its lower flux levels (10^{13} neutrons per

square centimeter per second) and predominantly thermal flux would limit its ability to support many of the projected nuclear-based research and development needs.

2.8 PREFERRED ALTERNATIVE

The Council on Environmental Quality (CEQ) regulations require an agency to identify its preferred alternative(s) in the final programmatic environmental impact statement (40 CFR 1502.14(e)). The preferred alternative is the alternative that the agency believes would fulfill its statutory mission, giving consideration to environmental, economic, technical, and other factors. Consequently, to identify a preferred alternative, DOE has developed information on potential environmental impacts, costs, policy issues, technical risks, and schedule risks for the alternatives under consideration. This NI PEIS provides information on the environmental impacts. Cost, nonproliferation policy, and various technical reports have also been prepared and are available in the appropriate DOE Reading Rooms for public review.

Based on the analysis discussed above, DOE's Preferred Alternative is to apply its existing infrastructure to the extent possible to pursue the missions outlined in this NI PEIS, that is, Alternative 2, Option 7. Under this approach, DOE proposes to consider opportunities to enhance its existing facilities to maximize the agency's ability to address future mission needs.

The Preferred Alternative also addresses the future of FFTF. While DOE recognizes that this facility has unique capabilities, the Department did not receive the commitments from the private sector or other governments that would clearly justify the restart of the facility. Lacking such commitment, DOE would permanently deactivate FFTF under the Preferred Alternative.

Finally, under the Preferred Alternative, DOE proposes to reestablish domestic production of plutonium-238, as needed, to support U.S. space exploration. ATR in Idaho and HFIR in Tennessee would be used, as appropriate, to irradiate targets for this purpose without interfering with either reactor's primary mission. The Preferred Alternative includes processing the irradiated plutonium-238 targets at REDC at ORNL.

In view of the lack of commitments that would justify the restart of FFTF or the construction of new facilities as proposed under Alternatives 3 and 4, DOE anticipates that its current infrastructure will serve the needs of the research and isotope communities for the next several years. In particular, DOE will consider opportunities to enhance its effort to provide medical and research isotopes. If significantly larger amounts of isotopes are required in the future, DOE would rely on the private sector to fulfill these needs.

As a potential option for the longer-term future, DOE proposes to work over the next 2 years to establish a conceptual design for an Advanced Accelerator Applications (AAA) facility. Such a facility, which would be used to evaluate spent fuel transmutation, conduct various nuclear research missions, and ensure a viable backup technology for the production of tritium for national security purposes, was proposed and initial work funded in the fiscal year 2001 Energy and Water Appropriation. If DOE proposes specific enhancements of existing facilities or development of the AAA facility, further NEPA review would be conducted.

2.9 REFERENCES

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Chapter 3
Affected Environment

In Chapter 3, the affected environment descriptions are presented to provide the context for understanding the environmental consequences described in Chapter 4. As such, they serve as a baseline from which any environmental changes that may be brought about by implementing the proposed action and alternatives can be identified and evaluated; the baseline conditions are the currently existing conditions. The affected environments at each site are described for the following impact areas: land resources, noise, air quality, water resources, geology and soils, ecological resources, cultural and paleontological resources, socioeconomics, existing human health risk, environmental justice, waste management, and spent nuclear fuel.

3.1 Approach to Defining the Affected Environment

For this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]), the candidate sites for neptunium-237 storage, target fabrication, and irradiated target processing facilities to recover plutonium-238 are the Oak Ridge Reservation (ORR), the Idaho National Engineering and Environmental Laboratory (INEEL), and the Hanford Site (Hanford). As described in Chapter 2, the candidate facilities for neptunium-237 storage, target fabrication, and irradiated target processing are the Radiochemical Engineering Development Center (REDC) at ORR, Building 651 and the Fluorinel Dissolution Process Facility (FDPF) at INEEL, and the Fuels and Materials Examination Facility (FMEF) at Hanford. The facilities being considered for irradiation of the neptunium-237 targets are the High Flux Isotope Reactor (HFIR) at ORR, the Advanced Test Reactor (ATR) at INEEL, the Fast Flux Test Facility (FFTF) at Hanford, a generic commercial light water reactor (CLWR), and one or two new accelerators or a nuclear research reactor that would be located at an unspecified existing U.S. Department of Energy (DOE) site.

As described in Chapter 2, the candidate sites for target fabrication and irradiated target processing facilities for industrial and medical isotope production and for research and development are Hanford, and the unspecified DOE site, where one or two new accelerators or a research reactor would be located. The candidate facilities for these target fabrication and irradiated target processing activities would be FMEF in the Hanford 400 Area, other existing processing facilities in the Hanford 300 Area, and a new processing facility located at an existing DOE site where one or two new accelerators or research reactor would be constructed. The sites being considered for research and development irradiation activities and for irradiation of targets for the production of industrial and medical isotopes are Hanford, where FFTF is located, and an unspecified existing DOE site where one or two new accelerators or research reactor would be located.

The affected environment is described for the candidate sites for the following resource areas: land use, visual resources, noise, air quality, water resources, geology and soils, ecological resources, cultural and paleontological resources, socioeconomics, existing human health risk, environmental justice, waste management, and spent nuclear fuel management. No additional spent nuclear fuel would be generated by the operation of HFIR, ATR, or a generic CLWR for neptunium-237 target irradiation; as they would be operating, even if they were not irradiating targets discussed in this NI PEIS. Additional spent nuclear fuel would be generated by the operation of FFTF or a new reactor located on an existing DOE site. Operation of the existing HFIR, ATR, or generic CLWR would continue to generate spent nuclear fuel, which is managed under current planning. New spent nuclear fuel would be generated by the restart of FFTF or a new research reactor located on an existing DOE site. Accordingly, spent nuclear fuel management is addressed in this chapter only in the sections for Hanford and an existing DOE site where a new research reactor would be located.

DOE evaluated the environmental impacts of certain research and development activities, and of industrial isotope, medical isotope, and plutonium-238 production alternatives within defined regions of influence at each

of the candidate sites and along potential transportation routes. The regions of influence are specific to the type of effect evaluated, and encompass geographic areas within which any significant impact would be expected to occur. For example, human health risks to the general public from exposure to airborne contaminant emissions were assessed for an area within an 80-kilometer (50-mile) radius of the proposed facilities. The human health risks of shipping materials between sites were evaluated for populations living along roadways linking the DOE sites. Economic effects such as job and income changes were evaluated within a socioeconomic region of influence that include the county in which the site is located, and nearby counties in which a substantial portion of the site's workforce reside. Brief descriptions of the regions of influence are given in **Table 3–1**. More detailed descriptions of the region of influence and the methods used to evaluate impacts are presented in Appendix G.

Table 3–1 General Regions of Influence for the Affected Environment

Environmental Resouces	Region of Influence
Land use and visual resources	The site and the areas immediately adjacent to the site
Noise	The site, nearby offsite areas, access routes to the sites, and the transportation corridors between the sites
Air quality	The site, nearby offsite areas within local air quality control regions, and the transportation corridors between the sites
Water resources	Onsite and adjacent surface water bodies and groundwater
Geology and soils	Geologic and soil resources within the site and nearby offsite areas
Ecological resources	The site and adjacent areas where ecological resources may be affected by construction and/or operation
Cultural and paleontological resources	The area within the site and adjacent to the site boundary
Socioeconomics	The counties where at least 90 percent of site employees reside
Existing human health risk	The site, nearby offsite areas (within 80 kilometers [50 miles] of the site, and the transportation corridors between the sites) where worker and general population radiation, radionuclide, and hazardous chemical exposures may occur
Environmental justice	The minority and low-income populations within 80 kilometers (50 miles) of the site, and along the transportation corridors between the sites
Waste management	Waste management facilities on the site
Spent fuel management	Spent fuel management facilities on the site

At each of the candidate sites, baseline conditions for each environmental resource area were determined for ongoing operations from information provided in previous environmental studies, relevant laws and regulations, and other government reports and data bases. More detailed information of the affected environment at the candidate sites can be found in annual site environmental reports and site National Environmental Policy Act (NEPA) documents.

3.2 OAK RIDGE RESERVATION

ORR, established in 1943 as one of the three original Manhattan Project sites, is located on 13,949 hectares (34,424 acres) in Oak Ridge, Tennessee, and includes the Oak Ridge National Laboratory (ORNL), the Y–12 Plant (Y–12), and the East Tennessee Technology Park. It extends over parts of Anderson and Roane counties. The primary focus of ORNL is to conduct basic and applied scientific research and technology development. Y–12 engages in national security activities and manufacturing outreach to U.S. industries. The mission of the East Tennessee Technology Park is to maintain the infrastructure until decommissioning activities have been completed.

ORNL is one of the country's largest multidisciplinary and multiprogram laboratories and research facilities. Its primary mission is to perform leading-edge nonweapons research and development in energy, health, and the environment. Other missions include production of radioactive and stable isotopes, not available from other production sources; fundamental and applied research and development in sciences and materials development; research involving hazardous and radioactive materials; environmental research; and radioactive waste disposal. These activities are primarily sponsored by various offices within DOE, including the Office of Science; Office of Environmental Management; Office of Environment, Safety and Health; and Office of Nuclear Energy, Science and Technology.

Activities at ORR that are sponsored by the DOE Office of Defense Programs are performed at Y–12, and include storage of uranium and lithium materials and weapons parts; maintenance of the capability to fabricate components for nuclear weapons; dismantlement of nuclear weapon components returned from the national stockpile; processing of special nuclear materials; and special production support to DOE design agencies and other DOE programs.

Environmental management activities are in progress at each of the major facilities within ORR. These activities consist of environmental remediation and restoration, decontamination and decommissioning of surplus facilities, and waste management.

Non-DOE activities conducted at ORR include National Oceanic and Atmospheric Administration missions and programs, which conducts meteorological and atmospheric diffusion research, sponsored by itself and DOE. This work is performed at the Atmospheric Turbulence and Diffusion Laboratory and at field sites on ORR. This laboratory also provides services to DOE contractors and operates the Weather Instrument Telemetering Monitoring System for DOE. ORR also provides support to other Federal agencies such as the U.S. Nuclear Regulatory Commission (NRC), U.S. Environmental Protection Agency (EPA), and others, and private industry in conducting basic scientific research, engineering technology development and transfer, and educational research in the areas of health, environment, and energy.

3.2.1 Land Resources

Land resources include land use and visual resources. Each of these resource areas is described for the site as a whole, as well as for the locations of the proposed activities.

3.2.1.1 Land Use

Land use may be characterized by its current use and potential for the location of human activities. Natural resource attributes and other environmental characteristics could make a site more suitable for some land uses than for others. Changes in land use may have both beneficial and adverse effects on other resources such as ecological, cultural, geological, aquatic, and atmospheric.

3.2.1.1.1 General Site Description

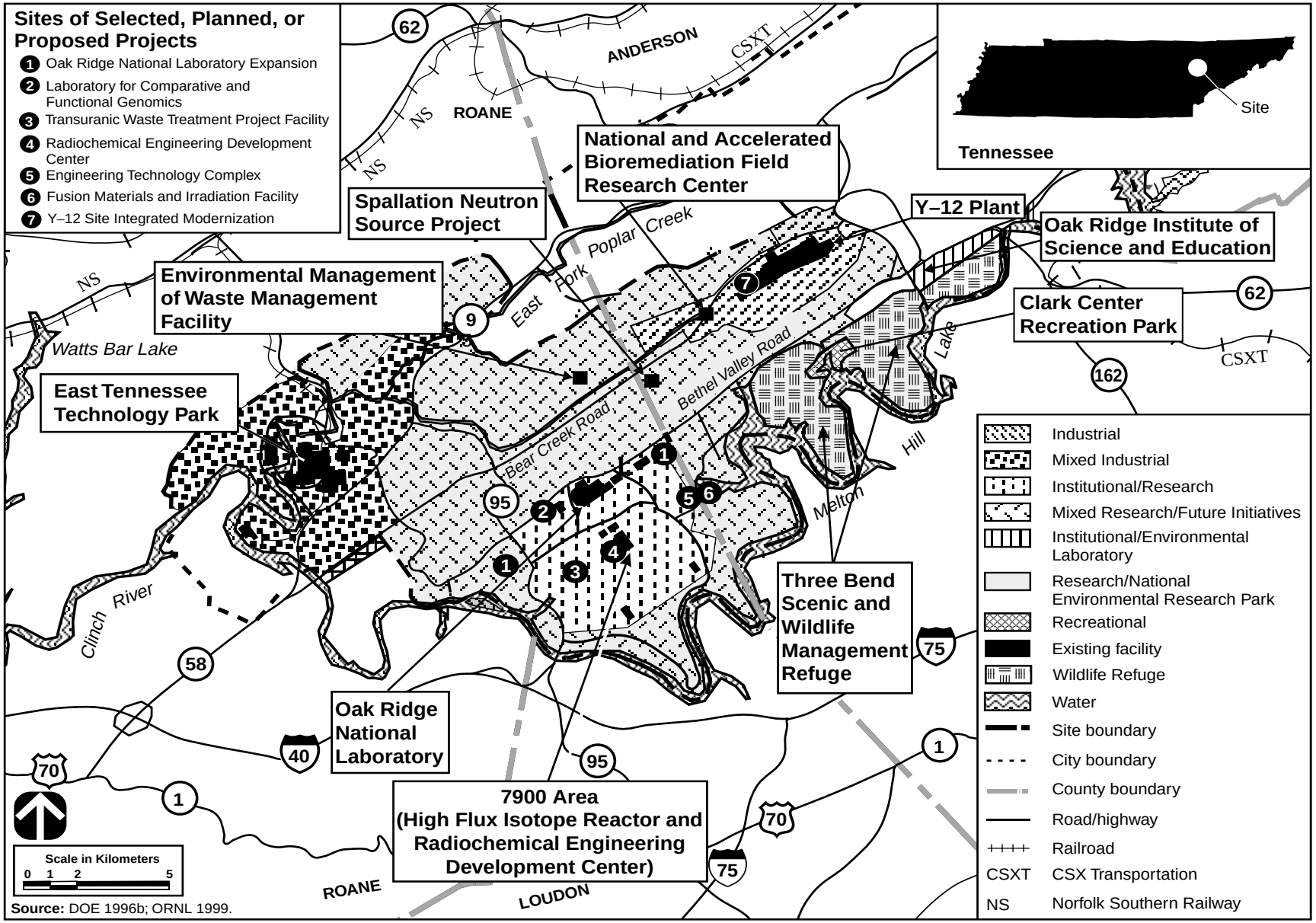
Land bordering ORR is predominantly rural and is used primarily for residences, small farms, forest land, and pasture land. The city of Oak Ridge has a typical urban mix of residential, public, commercial, and industrial land uses. It also includes almost all of ORR. There are four residential areas along the northern boundary of ORR, several of which have houses located within 30 meters (98 feet) of the site boundary.

Generalized land uses at ORR are shown in **Figure 3–1**. Land uses at the site include industrial, mixed industrial, institutional/research, institutional/environmental laboratory, and mixed research/future initiatives. Industrial and mixed industrial areas of the site include ORNL, Y–12, and the East Tennessee Technology Park. The institutional/research category applies to land occupied by central research facilities at ORNL and the Natural and Accelerated Bioremediation Field Research Center in Bear Creek Valley near Y–12. The institutional/environmental laboratory category includes the Oak Ridge Institute for Science and Education. Land within the mixed research/future initiative category includes land that is used or available for use in field research and land reserved for future DOE initiatives. Most mixed research and future initiatives areas are forested. Undeveloped forested lands on ORR are managed for multiple use and sustained yield of quality timber products. Although soils that would be identified as prime farmland occur on the site, that designation is waived because they are within the city of Oak Ridge (DOE 1999a). Only a small fraction of ORR has been disturbed by Federal activities, including the construction and operation of facilities, roadways, or other structures.

A large number of reservation-wide land uses overlay the primary land use categories and are officially designated as mixed uses. The largest mixed use is biological and ecological research in the Oak Ridge National Environmental Research Park, which is on 8,090 hectares (20,000 acres). The National Environmental Research Park, established in 1980, is used by the nation's scientific community as an outdoor laboratory for environmental science research on the impact of human activities on the eastern deciduous forest ecosystem (DOE 1996b; ORNL 1999). Recently, the Three Bend Scenic and Wildlife Management Refuge Area, on 1,215 hectares (3,000 acres), was set aside by DOE as a conservation and wildlife management area. The area is located in the ORR buffer zone, on Freels, Gallaher, and Solway Bends on the north shore of Melton Hill Lake (DOE 1999b). Additional details on land use plans at the site are provided in the *Oak Ridge National Laboratory Land and Facilities Plan* (LMER 1999).

Proposed short-range projects at ORR include the Composite Materials Laboratory; Laboratory for Comparative and Functional Genomics; Mixed Waste Treatment Facility; Transuranic Waste Treatment Project Facility; Recycle and Materials Processing Facility; Process Waste Treatment Facility; Industrial Landfill Expansion and Upgrades; and Steam Plant Waste Water Treatment Facility. The Spallation Neutron Source Project and the Environmental Management of Waste Management Facility are in early stages of development. DOE completed an environmental assessment for economic development leasing of 387 hectares (957 acres) of land located to the northeast of the East Tennessee Technology Park (DOE 1996a:S-1). The lease is for 40 years (DOE 2000e). The Community Reuse Organization of East Tennessee is currently developing the site as an industrial park. The locations of selected, planned, or proposed projects at ORR are shown in Figure 3–1.

Almost all of ORR lies within the city of Oak Ridge. A small portion of the northwest corner of the site lies outside the city in Roane County. The Oak Ridge Area Land Use Plan (city of Oak Ridge) designates ORR with the following land uses: residential, office/institutional, industrial, public, and undesignated. The city of Oak Ridge zoning ordinance classifies the entire ORR as a Forest, Agriculture, Industry, and Research District. The Roane County zoning ordinance does not classify ORR land; rather, it identifies ORR as a DOE Reservation.



Source: DOE 1996b; ORNL 1999.

Figure 3–1 Generalized Land Use at Oak Ridge Reservation and Vicinity

3.2.1.1.2 Location of Proposed Activities

ORNL is primarily located within Bethel Valley between Haw and Chestnut Ridges, and covers 1,720 hectares (4,250 acres) of land (ORNL 1999). The site is classified as an industrial area that encompasses a number of facilities dedicated to energy research. REDC and HFIR are in the 7900 Area of ORNL. The 7900 Area is situated on a low ridge in Melton Valley, just to the southwest of Haw Ridge. The nearest public access to the 7900 Area, Bethel Valley Road, is located about 1,500 meters (4,920 feet) to the north, and the nearest residential area is about 4,100 meters (13,450 feet) to the southwest. Land surrounding ORNL is largely forested and is classified as mixed research/future initiatives.

3.2.1.2 Visual Resources

Visual resources are natural and human-created features that give a particular landscape its character and aesthetic quality. Landscape character is determined by the visual elements of form, line, color, and texture. All four elements are present in every landscape. The stronger the influence exerted by these elements in a landscape, the more interesting the landscape.

3.2.1.2.1 General Site Description

The ORR landscape is characterized by a series of ridges and valleys that trend in a northeast-to-southwest direction. The vegetation is dominated by deciduous forest mixed with some coniferous forest. Most of the original open field areas on the site have been planted in shortleaf and loblolly pine, although smaller areas have been planted in a variety of deciduous and coniferous trees. The DOE facilities are brightly lit at night, making them especially visible. The developed areas of ORR are consistent with the Bureau of Land Management's Visual Resource Management Class IV rating in which management activities dominate the view and are the focus of viewer attention (DOI 1986). The remainder of ORR ranges from a Visual Resource Management Class II to Class III rating. Management activities within these classes may be seen, but should not dominate the view.

The viewshed consists mainly of rural land. The city of Oak Ridge is the only adjoining urban area. Sensitive viewpoints affected by DOE facilities are primarily associated with Interstate 40, State Highways 58, 62, and 95, and Bethel Valley and Bear Creek Roads. The Clinch River/Melton Hill Lake, and the bluffs on the opposite side of the Clinch River also have views of ORR, but views of most of the existing DOE facilities are blocked by terrain and/or vegetation. Although only a small portion of State Highway 62 crosses ORR, it is a major route for traffic to and from Knoxville and other communities. The hilly terrain, heavy vegetation, and generally hazy atmospheric conditions limit views. Partial views of the city of Oak Ridge water treatment plant can be seen from the urban areas of the city of Oak Ridge.

3.2.1.2.2 Location of Proposed Activities

ORNL is one of several highly developed areas of ORR. As noted above, such areas are consistent with the Bureau of Land Management Visual Resource Management Class IV rating. While a large part of ORNL is visible from Bethel Valley Road, it is not visible to persons in offsite locations because of the presence of the Haw and Chestnut Ridges. The 7900 Area, which is located to the south of the main ORNL complex, is not visible from any public area.

3.2.2 Noise

Noise is unwanted sound that interferes or interacts negatively with the human or natural environment. Noise may disrupt normal activities or diminish the quality of the environment.

3.2.2.1 General Site Description

Major noise emission sources within ORR include various industrial facilities, equipment, and machines (e.g., cooling systems, transformers, engines, pumps, boilers, steam vents, paging systems, construction and materials-handling equipment, and vehicles). Most ORR industrial facilities are a sufficient distance from the site boundary that noise levels at the boundary from these sources are not measurable, or are barely distinguishable from background noise levels (DOE 1996b:3-192).

Sound level measurements have been recorded at various locations within and near ORR in the process of testing sirens and preparing support documentation for the Atomic Vapor Laser Isotope Separation site. The acoustic environment along the ORR site boundary in rural areas and at nearby residences away from traffic noise is typical of a rural location, with average day-night sound levels in the range of 35 to 50 decibels A-weighted (dBA). Areas near the site within Oak Ridge are typical of a suburban area, with the average day-night sound levels in the range of 53 to 62 dBA. Traffic is the primary source of noise at the site boundary and at residences located near roads. During peak hours, the plant traffic is a major contributor to traffic noise levels in the area (DOE 1996b:3-192).

The State of Tennessee has not established specific community noise standards applicable to ORR. The city of Oak Ridge has specific acceptable sound levels at property lines (City of Oak Ridge 1999). EPA guidelines for environmental noise protection recommend a day-night average sound level of 55 dBA as sufficient to protect the public from the effects of broadband environmental noise in typically quiet outdoor and residential areas (EPA 1974:29). Land use compatibility guidelines adopted by the Federal Aviation Administration and the Federal Interagency Committee on Urban Noise indicate that yearly day-night average sound levels less than 65 dBA are compatible with residential land uses (14 CFR Part 150). These guidelines further indicate that noise levels up to 75 dBA are compatible with residential uses if suitable noise reduction features are incorporated into structures. It is expected that for most residences near ORR, the day-night average sound level is less than 65 dBA, and is compatible with the residential land use, although for some residences along major roadways noise levels may be higher.

3.2.2.2 Location of Proposed Activities

No distinguishing noise characteristics within ORNL have been identified. The ORNL 7900 Area is 2.5 kilometers (1.6 miles) from the site boundary; thus, the noise levels at the site boundary from these sources are barely distinguishable from background noise levels.

3.2.3 Air Quality

Air pollution refers to the introduction, directly or indirectly, of any substance into the air that could endanger human health, harm living sources and ecosystems and material property, and impair or interfere with the comfortable enjoyment of life or other legitimate uses of the environment. Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Air quality is affected by air pollutant emission characteristics, meteorology, and topography.

3.2.3.1 General Site Description

The climate at ORR may be classified as humid continental, but is moderated by the influence of the Cumberland and Great Smoky Mountains. Winters are mild and summers are warm, with no noticeable extremes in precipitation, temperature, or winds (DOE 1996b:3-192). The average annual temperature is 13.7 °C (56.6 °F); average monthly temperatures range from a minimum of 2.2 °C (36 °F) in January to a maximum of 24.9 °C (76.8 °F) in July. The average annual precipitation is 138.5 centimeters (54.5 inches).

Prevailing winds at ORR generally follow the valley, up the valley from the southwest daytime, or down the valley from the northeast nighttime. The wind speed is less than 11.9 kilometers per hour (7.4 miles per hour) 75 percent of the time; tornadoes and winds exceeding 30 kilometers per hour (18 miles per hour) are rare (Hamilton et al. 1999:1-4).

ORR is located in the Eastern Tennessee and Southwestern Virginia Interstate Air Quality Control Region #207. The areas within this Air Quality Control Region are in attainment with respect to the National Ambient Air Quality Standards (NAAQS) for criteria pollutants (40 CFR Section 81.343). Applicable NAAQS and Tennessee State ambient air quality standards are presented in **Table 3–2**.

Table 3–2 Comparison of Modeled Ambient Air Concentrations from Oak Ridge Reservation Sources with Most Stringent Applicable Standards or Guidelines, 1998

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meters)[a]	ORR Concentration (micrograms per cubic meters)
Criteria pollutants			
Carbon monoxide	8 hours	10,000[b]	8.05
	1 hour	40,000[b]	27.1
Nitrogen dioxide	Annual	100[b]	1.58
Ozone	1 hour	235[c]	(d)
PM_{10}	Annual	50[b]	1.6
	24 hours	150[b]	12.7
Sulfur dioxide	Annual	80[b]	4.86
	24 hours	365[b]	35.7
	3 hours	1,300[b]	112.0
Other regulated pollutants			
Total suspended particulates	24 hours	150[e]	2[f]

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean particulate matter with an aerodynamic diameter less than or equal to 10 microns (PM_{10}) standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Federal and state standard.
c. Federal 8-hour standard is currently under litigation.
d. Not directly emitted or monitored by the site.
e. State standard.
f. Based on stack emissions of particulate matter only.

Note: Emissions of hazardous air pollutants not listed here have been identified at ORR, but are not associated with any alternative evaluated. EPA revised the ambient air quality standards for particulate matter and ozone in 1997 (62 FR 38856, 62 FR 38652); however, these standards are under litigation, but could become enforceable during the life of this project.
Source: 40 CFR Part 50; Hamilton et al. 1999; TDEC 1999a; modeled site boundary concentrations using 1998 meteorological data and 1998 emissions data.

One Prevention of Significant Deterioration Class I area can be found in the vicinity of ORR. A Class I area is one in which very little increase in pollution is allowed due to the pristine nature of the area. This area, the Great Smoky Mountains, is located 48.3 kilometers (30 miles) southeast of ORR. ORR and its vicinity are classified as a Class II area in which more moderate increases in pollution are allowed. Since the creation of the Prevention of Significant Deterioration program in 1977, no Prevention of Significant Deterioration permits have been issued for any emission source at ORR (DOE 1996b:3-192).

The primary sources of criteria air pollutants at ORR are the steam plants at ORNL, Y–12, and the East Tennessee Technology Park. Other emission sources include the Toxic Substances Control Act incinerator;

various process sources; vehicles, temporary emissions from construction activities; and fugitive particulate emissions from coal piles (DOE 1996b:3-192; Hamilton et al. 1999).

The existing ambient air pollutant concentrations attributable to sources at ORR are presented in Table 3–2. These concentrations are based on dispersion modeling, using emissions for the year 1998 (Hamilton et al. 1999). Only those pollutants that would be emitted by any of the alternatives evaluated in this NI PEIS are presented. As shown in Table 3–2, modeled concentrations associated with ORR emission sources represent a small percentage of the ambient air quality standard.

The closest offsite monitors are operated by the Tennessee Department of Environment and Conservation in Anderson County and the city of Knoxville. In 1999, these monitors reported a maximum 8-hour average carbon monoxide concentration of 4,466 micrograms per cubic meter and maximum 1-hour average concentration of 12,712 micrograms per cubic meter. An annual average particulate matter with an aerodynamic diameter less than or equal to 10 microns (PM_{10}) concentration of 30.0 micrograms per cubic meter and a maximum 24-hour average concentration of 71 micrograms per cubic meter were reported. Annual, 24-hour, and 3-hour average sulfur dioxide maximum concentrations of 7.9 micrograms per cubic meter, 78.5 micrograms per cubic meter, and 293 micrograms per cubic meter, respectively, were also reported in 1999 (EPA 2000).

Because ORR sources are limited or background concentrations of criteria pollutants are well below ambient standards, ORR emissions should not result in air pollutant concentrations that violate the ambient air quality standards.

3.2.3.2 Location of Proposed Activities

HFIR and REDC are located in the 7900 Area of ORNL. The 7900 Area is situated in Melton Valley, south of the main portion of ORNL, between the Cumberland Mountains to the northwest, and the Great Smoky Mountains to the southeast. Terrain generally consists of ridges and valleys oriented southwest-northeast. The prevailing winds tend to follow this flow (LMER 1998:2.3-1, 2.3-2).

Current nonradiological emissions from the HFIR/REDC facilities are minimal, and result from wet chemistry and laboratory scale activities located at the facility. Additional nonradiological emissions result from maintenance activities inside the facility and in a small shop located adjacent to HFIR/REDC and testing of emergency diesel generators. Current Tennessee Department of Environment and Conservation air pollution control rules do not require that these emissions be permitted or quantified (Smith 2000). The existing ambient air pollutant concentrations attributable to sources at HFIR and REDC are presented in **Table 3–3**. These concentrations are estimated using SCREEN3 and are expected to over estimate the contribution to site boundary concentrations.

The primary sources of nonradiological air pollutants at ORNL include the facility steam plant, discussed in Section 3.2.3.1, and two small oil-fired boilers, which account for 98 percent of all allowable emissions. ORNL has 21 air permits covering 201 air emission sources. In 1998, the Tennessee Department of Environment and Conservation inspected all permitted sources and found them to be in compliance (Hamilton et al. 1999:2-23).

Table 3–3 Comparison of Modeled Ambient Air Concentrations from Sources at HFIR and REDC with Most Stringent Applicable Standards or Guidelines

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meters)[a]	HFIR/REDC Concentration (micrograms per cubic meters)
Criteria pollutants			
Carbon monoxide	8 hours	10,000[b]	31.5
	1 hour	40,000[b]	45.1
Nitrogen dioxide	Annual	100[b]	0.0072
Ozone	1 hour	235[c]	(d)
PM_{10}	Annual	50[b]	0.0005
	24 hours	150[b]	5.96
Sulfur dioxide	Annual	80[b]	0.0005
	24 hours	365[b]	5.51
	3 hours	1,300[b]	12.4
Other regulated pollutants			
Total suspended particulates	24 hours	150[e]	5.96

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean particulate matter with an aerodynamic diameter less than or equal to 10 microns (PM_{10}) standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Federal and state standard.
c. Federal 8-hour standard is currently under litigation.
d. Not directly emitted or monitored by the site.
e. State standard.

Source: 40 CFR Part 50; Hamilton et al.1999; TDEC 1999a; modeled concentrations using SCREEN3 and emissions estimate for periodic testing of diesel generators.

3.2.4 Water Resources

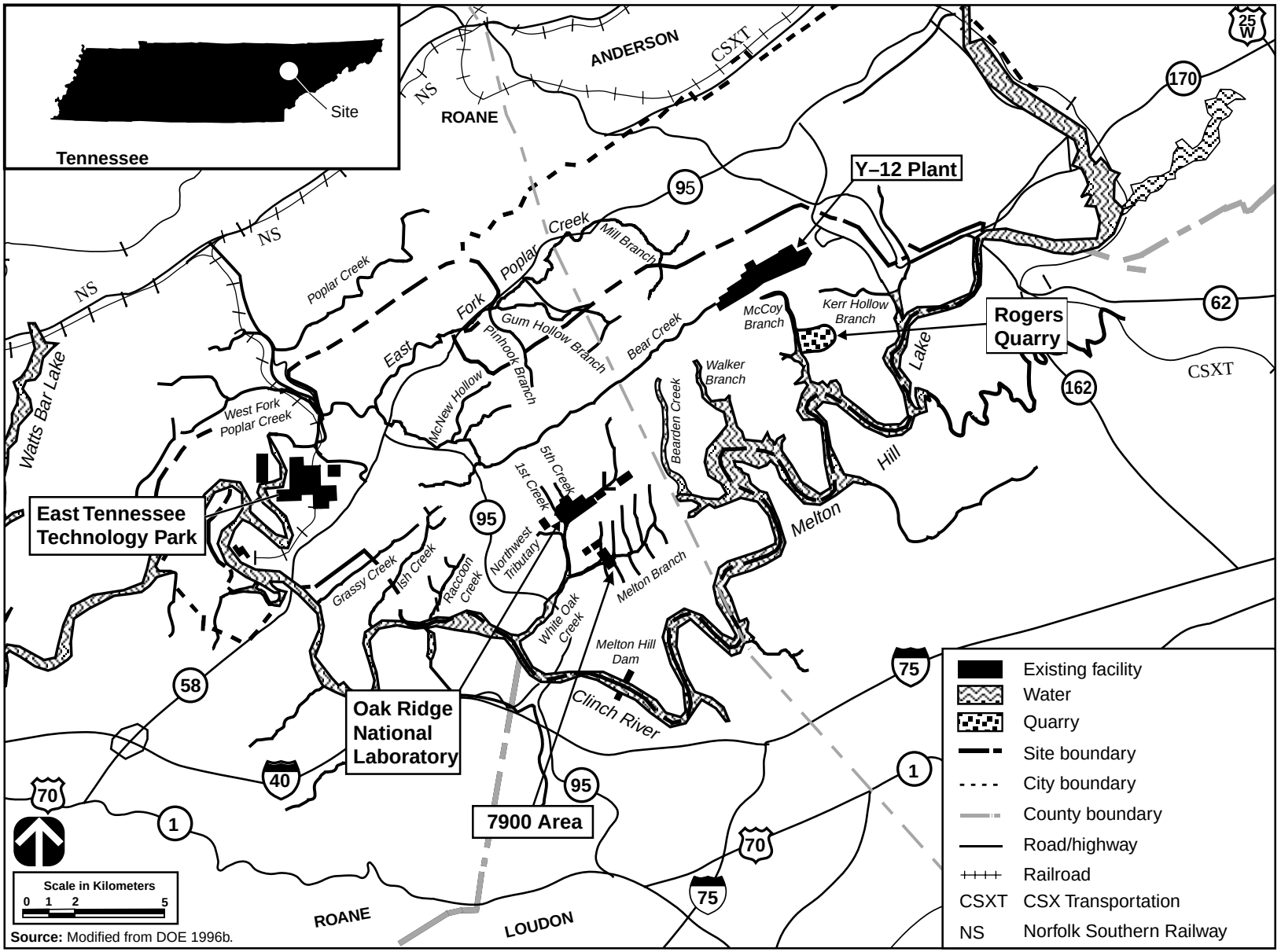
Water resources include all forms of surface water and subsurface groundwater.

3.2.4.1 Surface Water

Surface water includes marine or freshwater bodies that occur above the ground surface, including rivers, streams, lakes, ponds, rainwater catchments, embayments, and oceans.

3.2.4.1.1 General Site Description

The major surface water feature in the immediate vicinity of ORR is the Clinch River, which borders the site to the south and west. There are four major sub-drainage basins on ORR that flow into the Clinch River and are affected by site operations: Poplar Creek, East Fork Poplar Creek, Bear Creek, and White Oak Creek. Several smaller drainage basins, including Ish Creek, Grassy Creek, Bearden Creek, McCoy Branch, Kerr Hollow Branch, and Raccoon Creek, drain directly to the Clinch River (**Figure 3–2**). Each drainage basin takes the name of the major stream flowing through the area. The three major facilities at ORR each affect



Source: Modified from DOE 1996b.

Figure 3–2 Surface Water Features at Oak Ridge Reservation

different basins of the Clinch River. Drainage from Y–12 enters both Bear Creek and East Fork Poplar Creek; the East Tennessee Technology Park drains mainly into Poplar Creek; and ORNL drains into White Oak Creek (DOE 1996b:3-194).

The Clinch River and connected waterways supply the raw water for ORR. The Clinch River has an average flow rate of 132 cubic meters (4,647 cubic feet) per second, as measured at the downstream side of Melton Hill Dam. The average flow rates of Grassy, Ish, and Bear Creeks in the ORR area are 0.08 cubic meters (2.8 cubic feet) per second, 0.05 cubic meters (1.8 cubic feet) per second, and 0.11 cubic meters (3.9 cubic feet) per second, respectively. The average flow rate at East Fork Poplar Creek is 1.46 cubic meters (51.4 cubic feet) per second. Y–12 uses 7,530 million liters (1,989 million gallons) per year of water, and ORR uses 14,210 million liters (3,754 million gallons) per year (DOE 1996b:3-194). The ORR water supply system, which includes the city of Oak Ridge treatment facility (formerly the DOE treatment facility) and the East Tennessee Technology Park treatment facility, has a capacity of 90.8 to 121.5 million liters (24 to 32.1 million gallons) per day (DOE 1996b:3-194; LMER 1999:3-24).

The Clinch River water levels in the vicinity of ORR are regulated by a system of dams operated by the Tennessee Valley Authority. Melton Hill Dam controls the flow of the Clinch River along the northeast and southeast sides of ORR. Watts Bar Dam on the Tennessee River near the lower end of the Clinch River controls the flow of the Clinch River along the southwest side of ORR (DOE 1996b:3-194).

The surface streams of Tennessee are classified by the Tennessee Department of Environmental Conservation according to the Use Classifications for Surface Waters. Classifications are based on water quality, beneficial uses, and resident aquatic biota. The Clinch River is the only surface water body on or near ORR classified for domestic water supply. Unless otherwise specified in these rules, all streams in Tennessee are classified for use for fish and aquatic life, recreation, irrigation, and for livestock watering and wildlife. In addition, the Clinch River and a short segment of Poplar Creek from its confluence with the Clinch River are also classified for industrial water supply use. White Oak Creek and Melton Branch are the only streams on ORR not classified for irrigation (TDEC 1999b). East Fork Poplar Creek is posted by the State of Tennessee with warnings against fishing and contact recreation (O'Donnell 2000).

Wastewater treatment facilities are located throughout ORR, including six treatment facilities at Y–12 that discharge to East Fork Poplar Creek, and three treatment facilities at ORNL that discharge into White Oak Creek Basin. These discharge points are included in existing National Pollutant Discharge Elimination System (NPDES) permits. Y–12 also has a permit to discharge wastewater to the city of Oak Ridge Treatment Facility. The East Tennessee Technology Park operates one sanitary sewage system discharging to Poplar Creek (DOE 1996b:3-196).

There are more than 400 NPDES-permitted outfalls at ORR associated with the three major facilities (Y–12 Plant, East Tennessee Technology Park, and ORNL); many of these are storm water outfalls. ORNL is currently operating under NPDES Permit TN0002941, which was renewed by the Tennessee Department of Environmental Conservation on December 6, 1996, and went into effect on February 3, 1997. This permit lists 164 point-source discharges that require compliance monitoring. Approximately 100 of these are storm drains, roof drains, and parking lot drains. Compliance was determined by approximately 6,500 laboratory analyses and measurements in 1998, in addition to numerous field observations by ORNL field technicians. The NPDES permit compliance rate was nearly 100 percent with only three permit exceedances. The NPDES permit limit compliance rate for all discharge points for the three major facilities in 1998 was over 99 percent (Hamilton et al. 1999:2-18, 2-19).

Compared with the previous permit, the new ORNL NPDES permit includes more stringent limits based on water quality criteria at a number of outfalls. The new permit also requires ORNL to conduct detailed

characterization of numerous storm water outfalls, conduct an assessment and evaluation for the modification of the Radiological Monitoring Plan, develop and implement a Storm Water Pollution Prevention Plan, implement a revised Biological Monitoring and Abatement Program Plan, and develop and implement a Chlorine Control Strategy (Hamilton et al. 1999:2-19).

At ORR, water samples are collected and analyzed from 22 locations around the reservation to assess the impact of past and current DOE operations on the quality of local surface water. Sampling locations include streams, both upstream and downstream of ORR waste sources, and public water intakes. Samples are collected and analyzed for general water quality parameters at all locations, and are screened for radioactivity and analyzed for specific radionuclides, when appropriate. Based on 1998 sampling data, radionuclides were detected at all but two surface water locations, which were dry when sampling was attempted. High levels of radioactivity (gross alpha, gross beta, and total radioactive strontium) relative to applicable standards or criteria detected at First Creek within ORNL are attributed to leakage to backfill and soil from underground waste storage Tank W-1A at ORNL. Uranium isotopes were determined to be the primary alpha emitters. Excluding the First Creek site, the highest levels of gross beta, total radioactive strontium and tritium were detected at White Oak Creek at White Oak Dam and at White Oak Creek and Melton Branch, both downstream from ORNL. These data are consistent with historical data and with process or legacy activities nearby or upstream from these sites. Elevated levels of gross beta and total radioactive strontium have also been detected at Raccoon Creek and Northwest Tributary. Locations that were checked for volatile organic compounds showed either low or undetectable levels. Polychlorinated biphenyls were not detected at either of the two sites sampled. Except for lead in 1 sample, and zinc in 3 samples out of 12 respectively, at the Clinch River upstream from all DOE inputs, no metals of human health concern were detected (Hamilton et al. 1999:7-6, 7-10, 7-11, D-5, D-8, D-11, D-13, D-14).

In Tennessee, the state's water right laws are established under the Water Quality Control Act. In effect, water rights are similar to riparian rights, in that designated usages of a water body cannot be impaired. Before withdrawing water from available supplies, a U.S. Army Corps of Engineers permit to construct intake structures would need to be obtained (DOE 1996b:3-196). In addition, projects and activities with the potential to affect aquatic resources could require permits from the Tennessee Department of Environment and Conservation and the Tennessee Valley Authority (Hamilton et al. 1999:2-20).

The Tennessee Valley Authority has conducted flood studies along the Clinch River, Bear Creek, and East Fork Poplar Creek. Portions of Y–12 lie within the 100- and 500-year floodplain boundaries of East Fork Poplar Creek. Studies have not been performed to delineate the 100- or 500-year floodplain boundaries of Grassy, Ish, and Bear Creeks in the western half of the site (DOE 1996b:3-194).

The Tennessee Valley Authority has performed probable maximum flood studies along the Clinch River. The probable maximum flood is the flood that can be expected from the most severe combination of critical hydrometeorological conditions that are reasonably possible over the entire watershed. The probable maximum flood level along the Clinch River at the mouth of Bearden Creek occurred at elevation 248.3 meters (814.7 feet), while the probable maximum flood level at the mouth of White Oak Creek occurred at elevation 237.5 meters (779.3 feet). Based on the studies, most of ORR is above the probable maximum flood elevation along the Clinch River (LMER 1999:2-8).

3.2.4.1.2 Location of Proposed Activities

HFIR and REDC are in the 7900 Area on a low ridge in Melton Valley. HFIR overlooks Melton Branch, a tributary of White Oak Creek, with the REDC complex (Buildings 7920 and 7930) just north and upslope of HFIR (LMER 1998:2.4-1). Two reservoir systems supply water to ORNL facilities in Melton Valley and the 7900 Area, in particular, and also to facilities in Bethel Valley. The first is to the north of the 7900 Area on

Chestnut Ridge and consists of a concrete storage reservoir with a capacity of 11.4 million liters (3 million gallons). A project to construct a second 3.8-million-liter (1-million-gallon) storage reservoir adjacent to the existing one is planned. The second system is on Haw Ridge and consists of two steel reservoir tanks, each with a storage capacity of 5.7 million liters (1.5 million gallons). These tanks are designated to provide reserve capacity for HFIR, REDC, and other facilities in Melton Valley. Water usage by ORNL Melton Valley facilities ranges from 9.5 million liters (2.5 million gallons) per day in the winter to a maximum of 18.9 million liters (5 million gallons) per day in the summer. The reservoir distribution system can supply 26.5 million liters (7 million gallons) per day (LMER 1999:3-25). These reservoirs are supplied by the city of Oak Ridge water treatment plant, which receives its water from the Melton Hill Reservoir via a pumping station upstream from HFIR. Either of the two reservoirs is capable of supplying the normal 3,785 liters (1,000 gallons) per minute cooling water requirements of HFIR (Building 7900) (LMER 1998:2.4-1). Based on the most recent water use survey, the HFIR complex (i.e., Buildings 7900–7903, 7910, and 7916) uses a total of approximately 6.1 million liters (1.6 million gallons) of water per day or about 2.23 billion liters (589 million gallons) annually. REDC (Buildings 7920 and 7930) uses approximately 294,000 liters (77,800 gallons) of water per day or 107 million liters (28.4 million gallons) per year (Wham 2000c). The major water demand by both HFIR and REDC is for cooling purposes.

Sanitary wastewater from the 7900 Area is conveyed to the ORNL Sewage Treatment Plant, which provides primary, secondary, and tertiary sewage treatment. The Sewage Treatment Plant has a treatment capacity of 1.1 million liters (300,000 gallons) per day. Since 1997, treated flows have ranged from about 685,000 to 821,000 liters (181,000 to 217,000 gallons) per day (LMER 1999:3-62, 3-63). Specifically, the HFIR complex is estimated to generate about 7.3 million liters (1.93 million gallons) of sanitary wastewater per year with REDC generating an additional 3.1 million liters (828,000 gallons) annually (Wham 2000a).

Process wastewater from HFIR and REDC is collected and conveyed to storage tanks prior to processing in the Process Waste Treatment Complex. Continuous monitoring of the wastewater in the collection system is used to route the wastewater to the appropriate treatment process. The Process Waste Treatment Complex consists of two facilities, Buildings 3544 and 3608, which provide both nonradiological and radiological effluent treatment. Treatment in Building 3608 consists of precipitation, filtration, air stripping, and neutralization to remove particulates, heavy metals, and organics, and to control pH before discharge. A clarifier is also used to perform process wastewater softening prior to transfer to Building 3544 for further treatment. Treatment capacity is 4.2 million liters (1.1 million gallons) per day. Treatment in Building 3544 consists of precipitation, filtration, and ion exchange. The maximum treatment capacity is about 1.9 million liters (504,000 gallons) per day (LMER 1999:3-64, 3-65).

All wastewater treated at Buildings 3544 and 3608 is ultimately discharged to White Oak Creek through a single NPDES-permitted outfall (Outfall X12). The flow rate from this outfall averages about 2.08 million liters (550,000 gallons) per day, of which approximately 66,245 liters (17,500 gallons) per day are attributable to process wastewater from HFIR and REDC. The treated effluent from Outfall X12 meets NPDES water quality–based limits for metals and organics and DOE Derived Concentration Guides (DOE Order 5400.5), and is not toxic to aquatic species based on NPDES-required toxicity testing. HFIR and REDC also discharge dechlorinated cooling water and cooling tower blowdown to Melton Branch through NPDES-permitted outfalls 081 and 281. Discharge from Outfall 281, which is predominantly HFIR cooling tower blowdown, averages about 378,500 liters (100,000 gallons) per day in the warm months. The discharge rate from Outfall 081 averages approximately 265,000 liters (70,000 gallons) per day during the warm months and consists primarily of REDC cooling water (Valentine 2000). Waste management activities and facilities are discussed in greater detail under Section 3.2.11.

Melton Branch, the primary stream in the immediate vicinity of HFIR and REDC, was analyzed to assess the potential for flooding from a locally intense storm, based on probable maximum precipitation events. The

analysis determined that the relatively high elevation of the terrain and slope of the 7900 Area ensures that locally intense precipitation would not cause the Melton Branch to flood equipment at the HFIR site and vicinity. Likewise, the occurrence of a probable maximum flood at the mouth of White Oak Creek or along Melton Branch due to probable maximum precipitation events would not inundate the HFIR and vicinity. Surface runoff and facility drainage flows to either of two headwater tributaries of Melton Branch on the east and west sides, respectively, of the 7900 Area (LMER 1998:2.4-6, 2.4-7).

3.2.4.2 Groundwater

Aquifers are classified by Federal and state authorities according to use and quality. The Federal classifications include Classes I, II, and III groundwater. Class I groundwater is either the sole source of drinking water, or is ecologically vital. Classes IIA and IIB are current or potential sources of drinking water (or other beneficial use), respectively. Class III is not considered a potential source of drinking water and is of limited beneficial use.

3.2.4.2.1 General Site Description

ORR is in an area of sedimentary rocks of widely varying hydrologic character. Groundwater flow occurs at shallow depths with discharge to nearby surface waters. Depth to groundwater is generally 5 to 9 meters (16 to 30 feet), but may be as little as 1.5 meters (5 feet). All aquifers are considered Class II (DOE 1996b:3-196).

Two broad hydrologic regimes have been characterized at ORR, each having fundamentally different hydrologic characteristics. The Knox Group and the Maynardville Limestone of the Conasauga Group constitute the Knox aquifer, in which flow is dominated by solution conduits formed along fractures and bedding planes. The less permeable ORR aquitard units constitute the second regime, in which flow is dominated by fractures (DOE 1999a:4-12; Hamilton et al. 1999:1-5, 1-6). These hydrologic groupings and the geologic units comprising them are illustrated in **Figure 3–3**. The combination of fractures and solution conduits in the dolostones and limestones of the Knox aquifer control flow over substantial areas, and rather large quantities of water may move relatively long distances. The Knox aquifer is the primary source of groundwater to many streams (base-flow), and most large springs on ORR receive discharge from the Knox aquifer. Yields of some wells penetrating larger solution conduits are reported to exceed 3,785 liters (1,000 gallons) per minute. Units at ORR constituting the ORR aquitards include the Rome Formation, the Conasauga Group below the Maynardville Limestone, and the Chickamauga Group, and consist mainly of siltstone, shale, sandstone, and thinly bedded limestone of low to very low permeability. The typical yield of a well in the aquitards is less than 3.8 liters (1 gallon) per minute, and the base flows of streams draining areas underlain by the aquitards are poorly sustained because of such low flow rates (Hamilton et al. 1999:1-5).

Subsurface flow in both the Knox aquifer and in the aquitards is recharged mainly on ridges and is discharged into lakes, streams, springs, and seeps (DOE 1999a:4-12). Within ORR, the Knox aquifer underlies some of the major ridges (e.g., Chestnut and Copper Ridge) and aquitard units predominate under the valleys (e.g., Bear Creek, Bethel, and Melton valley) (Hamilton et al. 1999:1-7, 1-8) (**Figure 3–4**). Because of the abundance of surface water and its proximity to the points of use, very little groundwater is used at ORR. Only one water supply well exists on ORR; it provides a supplemental water supply to an ORNL aquatic biology laboratory during extended droughts (DOE 1996b:3-196).

Groundwater samples are collected quarterly from representative wells selected from more than 1,000 monitoring wells throughout ORR. Samples collected from monitoring wells are analyzed for a standard set of parameters, including trace metals, volatile organic compounds, radioactive materials, and acidity/basicity. Background groundwater quality at ORR is generally good in the near-surface aquifer zones, and poor in the bedrock aquifer at depths greater than 305 meters (1,000 feet), due to high total dissolved

Age		Group	Formation		Thickness, meters	Hydrologic Unit
Ordovician	Upper	Chickamauga Group	Moccasin Formation		100–170	Aquitard
			Witten Formation		105–110	
			Bowen Formation		5–10	
	Middle		Benbolt/Wardell Formation		110–115	Aquifer
			Rockdell Formation		80–85	
			Hogskin Member Fleanor Shale Member	Lincolnshire Formation	75–80	Aquitard
			Eidson Member		70–80	
			Blackford Formation			
	Lower	Knox Group	Mascot Dolomite		75–150	Knox Aquifer
			Kingsport Formation		90–150	
			Longview Dolomite		40–60	
			Chepultepec Dolomite		152–213	
Cambrian	Upper		Copper Ridge Dolomite		244–335	
		Conasauga Group	Maynardville Limestone		100–110	Aquitard
	Middle		Nolichucky Shale		150–180	
			Dismal Gap Formation (Formerly Maryville Limestone)		98–125	
			Rogersville Shale		25–34	
			Friendship Formation (Formerly Rutledge Limestone)		31–37	
			Pumpkin Valley Shale		56–70	
	Lower		Rome Formation		122–183	

Source: Modified from DOE 1999a.

Note: To convert meters to feet multiply by 3.281.

Figure 3–3 Stratigraphic Column for the Oak Ridge Reservation

solids (DOE 1996b:3-197). Information on more recent groundwater monitoring and chemical analysis is presented in the annual site environmental report (Hamilton et al. 1999).

Groundwater in the Bear Creek Valley near Y–12 and in the ORNL and East Tennessee Technology Park areas has been locally contaminated by hazardous chemicals and radionuclides from past process activities. The contaminated sites include past waste disposal sites, waste storage tanks, spill sites, and contaminated inactive facilities (DOE 1996b:3-197).

Industrial and drinking water supplies are primarily taken from surface water sources. However, single-family wells are common in adjacent rural areas not served by the public water supply system. Most of the residential wells in the immediate vicinity of ORR are south of the Clinch River (DOE 1996b:3-197).

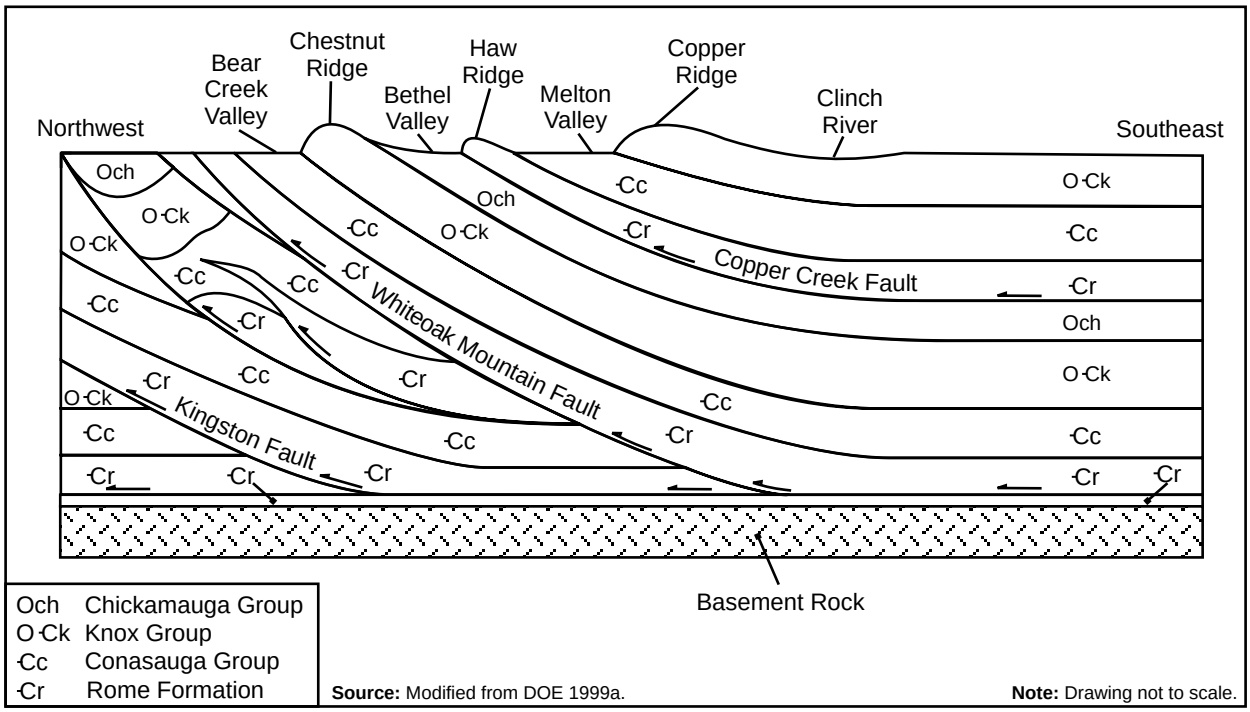


Figure 3–4 Geologic Cross Section of the Oak Ridge Reservation

Groundwater rights in the State of Tennessee are traditionally associated with the Reasonable Use Doctrine. Under this doctrine, landowners can withdraw groundwater as long as they exercise their rights reasonably in relation to the rights of others (DOE 1996b:3-199).

3.2.4.2.2 Location of Proposed Activities

Groundwater sampling performed in 1998 at 11 monitoring wells located in waste area groupings 8 and 9 in Melton Valley, encompassing HFIR and REDC, showed evidence of radioactivity attributable to former effluent-handling practices in the 7900 Area (Smith 2000). Two of the sampled wells exceeded Federal drinking water standards: one well for tritium contamination, and a second well for gross beta activity and total radioactive strontium contamination. Gross alpha activity ranged from undetectable to 6.7 picocuries per liter; the drinking water standard is 15 picocuries per liter. Gross beta activity ranged from undetectable to 1,400 picocuries per liter; the drinking water standard is 50 picocuries per liter. Total radioactive strontium ranged from undetectable to 630 picocuries per liter; the drinking water standard is 8 picocuries per liter. Tritium ranged from undetectable to 53,000 picocuries per liter; the drinking water standard is 20,000 picocuries per liter (Hamilton et al. 1999:5-27, 5-33). Note that groundwater is not used for drinking water at ORNL (Hamilton et al. 1999:5-28). In general, contaminant plumes in groundwater at ORNL and elsewhere at ORR are relatively small in areal extent as contaminant sources are discretely located, and flow paths to surface water outlets are short (Hamilton et al. 1999:1-9, 5-28, 5-30).

3.2.5 Geology and Soils

Geologic resources are consolidated or unconsolidated earth materials, including ore and aggregate materials, fossil fuels, and significant landforms. Soil resources are the loose surface materials of the earth in which plants grow, usually consisting of disintegrated rock, organic matter, and soluble salts.

3.2.5.1 General Site Description

ORR is in the southwestern portion of the Valley and Ridge physiographic province in east-central Tennessee. The topography consists of alternating valleys and ridges that have a southwest-northeast trend, with most ORR facilities occupying the valleys (DOE 1996b:3-200). The topography reflects the underlying geology, which consists of a sequence of sedimentary rocks deformed by a series of major southeast-dipping thrust faults (Figures 3–3 and 3–4). The ridges are underlain by relatively erosion-resistant rocks, while weaker rock strata underlie the valleys (DOE 1999a:4-1, 4-3). Y–12 is in Bear Creek Valley between Pine and Chestnut Ridges, East Tennessee Technology Park is located along Poplar Creek between McKinney and Pine Ridges, and the ORNL main site is in Bethel Valley between Haw and Chestnut Ridges. The 7900 Area of ORNL is on a low ridge in Melton Valley, south of Haw Ridge. ORNL and the East Tennessee Technology Park are underlain primarily by calcareous siltstones and silty to clean limestone of the Chickamauga Group of upper and middle Ordovician age (about 440 to 480 million years old). The Conasauga Group underlies Y–12 and the 7900 Area; it is comprised of shales, calcareous siltstones, and silty-to-clean limestones of upper and middle Cambrian age (about 505 to less than 550 million years old). Pine Ridge and Haw Ridge are underlain by the Rome Formation, which consists of sandstone with thin shale interbeds. Chestnut Ridge is underlain by the cherty dolomite of the Knox Group. The Knox Group typically has well-developed karst features such as sinkholes, large solution cavities, and caves (DOE 1996b:3-200; 1999a:4-3–4-5). Structurally, two major thrust faults factor into the subsurface geology of ORR. Chestnut Ridge and Bethel Valley are underlain by the Whiteoak Mountain thrust fault. Haw Ridge and Melton Valley are underlain by the Copper Creek thrust fault (Figure 3–4). These faults formed during the Permian-Pennsylvanian periods (occurring about 245 to 320 million years ago) but have not been historically active (DOE 1999a:4-3–4-5). The present topography of the valleys is a result of stream action preferentially eroding the softer shales and limestones; the ridges are composed of relatively more resistant sandstones and dolomites. With the exception of strata suited to hard-rock quarrying for stone and aggregate (e.g., limestone, shale), no economically viable geologic resources have been identified at ORR (DOE 1996b:3-200).

There is no evidence of capable faults in the Valley and Ridge physiographic province, or within the rocks comprising the Appalachian Basin structural feature, where ORR is located. A capable fault is one that has had movement at or near the ground surface at least once within the past 35,000 years, or recurrent movement within the past 500,000 years (10 CFR Part 100, Appendix A). The nearest capable faults are approximately 480 kilometers (298 miles) northwest in the New Madrid (Reelfoot rift) fault zone. Historical earthquakes occurring in the Valley and Ridge are not attributable to fault structures in underlying sedimentary rocks, but rather occur at depth in basement rock (DOE 1999a:4-9; LMER 1998:2.5-19, 2.5-20).

The historical seismicity of the southeastern United States relative to ORR has been extensively reviewed in recent years. Since the New Madrid earthquakes of 1811-1812, at least 27 other earthquakes with a Modified Mercalli Intensity of III to VI (**Table 3–4**) have been felt in the Oak Ridge area (DOE 1996b:3-200; LMER 1998:2.5-16, 2.5-17, 2.5-29, 2.5-30). Second to the New Madrid earthquakes in intensity, the Charleston, South Carolina, earthquake of 1886, located about 515 kilometers (320 miles) from ORR, is estimated to have produced effects at ORR equivalent to Modified Mercalli Intensity VI (LMER 1998:2.5-29). One of closest and most intense seismic events occurred in 1930, approximately 8 kilometers (5 miles) from ORR, and had a Modified Mercalli Intensity of V at the site. The largest most recent earthquake in eastern Tennessee registered 4.6 on the Richter scale and occurred on November 30, 1973, in Maryville, Tennessee, about 32 kilometers (20 miles) southeast of ORR. This earthquake produced a Modified Mercalli Intensity of V to VI at ORR (as estimated at HFIR) (DOE 1996b:3-200; LMER 1998: 2.5-17, 2.5-30). The region has continued to be seismically active, with 42 earthquakes recorded within a radius of 90 kilometers (56 miles) of ORNL since 1973. In 1987, a magnitude 4.2 earthquake occurred about 38 kilometers (24 miles) from ORR producing a Modified Mercalli Intensity of VI at its epicenter. Since 1995, two earthquakes with a reported

Table 3–4 The Modified Mercalli Intensity Scale of 1931, with Approximate Correlations to Richter Scale and Maximum Ground Acceleration[a]

Modified Mercalli Intensity[b]	Observed Effects of Earthquake	Approximate Richter Magnitude[c]	Maximum Ground Acceleration[d](g)
I	Usually not felt	Less than 2	Negligible
II	Felt by persons at rest on upper floors or favorably placed	2 to 3	Less than 0.003
III	Felt indoors; hanging objects swing; vibration like passing of light truck occurs; might not be recognized as earthquake	3	0.003 to 0.007
IV	Felt noticeably by persons indoors, especially in upper floors; vibration occurs like passing of heavy truck; jolting sensation; standing automobiles rock; windows, dishes, and doors rattle; wooden walls and frames may creak	4	0.007 to 0.015
V	Felt by nearly everyone; sleepers awaken; liquids disturbed and may spill; some dishes break; small unstable objects are displaced or upset; doors swing; shutters and pictures move; pendulum clocks stop or start	Between 4 and 5	0.015 to 0.03
VI	Felt by all; many are frightened; persons walk unsteadily; windows and dishes break; objects fall off shelves, pictures fall off shelves and walls; furniture moves or overturns; weak masonry cracks; small bells ring; trees and bushes shake	5	0.03 to 0.09
VII	Difficult to stand; noticed by car drivers; furniture breaks; damage moderate in well built ordinary structures; poor quality masonry cracks and breaks; chimneys break at roof line; loose bricks, stones, and tiles fall; waves appear on ponds and water is turbid with mud; small earthslides; large bells ring	6	0.07 to 0.22
VIII	Automobile's steering affected; some walls fall; twisting and falling of chimneys, stacks, and towers; frame houses shift if on unsecured foundations; damage slight in specially designed structures, considerable in ordinary substantial buildings; changes in flow of wells or springs; cracks appear in wet ground and steep slopes	Between 6 and 7	0.15 to 0.3
IX	General panic; masonry heavily damaged or destroyed; foundations damaged; serious damage to frame structures, dams and reservoirs; underground pipes break; conspicuous ground cracks	7	0.03 to 0.7
X	Most masonry and frame structures destroyed; some well built wooden structures and bridges destroyed; serious damage to dams and dikes; large landslides; rails bent	8	0.45 to 1.5
XI	Rails bent greatly; underground pipelines completely out of service	Between 8 and 9	0.5 to 3
XII	Damage nearly total; large rock masses displaced; objects thrown into air; lines of sight distorted	9	0.5 to 7

a. This table illustrates the approximate correlation between the Modified Mercalli Intensity scale, the Richter scale, and maximum ground acceleration.
b. Intensity is a unitless expression of observed effects.
c. Magnitude is an exponential function of seismic wave amplitude, related to the energy released.
d. Acceleration is expressed in relation to the earth's gravitational acceleration (g).

Source: DOE 1996b:3-39.

Modified Mercalli Intensity of at least III and two with a Modified Mercalli Intensity of V have occurred within approximately 90 kilometers (56 miles) of ORR. The most recent of those events occurred on June 17, 1998, with an epicenter within ORR near the East Tennessee Technology Park, registering a magnitude 3.6 (USGS 2000a, 2000c). Based on historical observations, the maximum earthquake for ORR would be a Modified Mercalli Intensity VIII event, having an epicenter at ORR (DOE 1999a:4-9, 4-10). Numerous studies have been conducted as part of establishing the design-basis earthquake for evaluating and designing new ORR facilities. For this purpose, an earthquake producing an effective peak-ground acceleration of 0.15g has been established and calculated to have an annual probability of occurrence of about 1 in 1,000

(LMER 1998:2.5-18, 2.5-59). For comparison, an earthquake with a peak acceleration of 0.32g has an annual probability of occurrence of 1 in 5,000 (Barghusen and Feit 1995:2.8-14).

Measures of peak (ground) acceleration are indicative of what an object on the ground would experience during an earthquake. This motion is customarily expressed in units of g (percent of gravity). While peak acceleration is generally adequate to approximate what a short structure would experience in terms of horizontal force during an earthquake, it does not account for the range of energies experienced by a building during an earthquake, particularly for taller buildings. Thus, building design based on peak acceleration alone does not provide a uniform margin against collapse. However, the U.S. Geological Survey (USGS) has developed new seismic hazard maps as part of the National Seismic Hazard Mapping Project that are based on response spectral acceleration.

Spectral acceleration maps account for the natural period of vibration of structures (i.e., short buildings have short natural periods [up to 0.6 second] and taller buildings longer periods [less than or equal to 0.7 second]) (USGS 2000b). These maps have been adapted for use in the new *International Building Code* (ICC 2000) (Figures 1615(1) and 1615(2) in the code) and depict maximum considered earthquake ground motion of 0.2- and 1.0-second spectral response acceleration, respectively, based on a 2 percent probability of exceedance in 50 years. This corresponds to an annual recurrence interval of about 1 in 2,500. ORR lies within the 0.50g to 0.60g mapping contours for a 0.2-second spectral response acceleration and the 0.10g to 0.15g contours for a 1.0-second spectral response acceleration.

There is no volcanic hazard at ORR. The area has not experienced volcanic activity within the last 230 million years (DOE 1996b:3-200).

Four general soil map units occur on ORR. These are described based on the Anderson County soil survey (Moneymaker 1981:5-7). The soil survey for Roane County has not been updated since 1942 (Swann et al. 1942) and does not specifically identify general soil map units. The four soil map units of ORR are Fullerton-Claiborne-Bodine; Collegedale-Gladeville-Rock outcrop; Lehew-Armuchee-Muskingum; and Armuchee-Montevallo-Hamblen units. Soils of the Fullerton-Claiborne-Bodine unit may be described as deep, rolling-to-steep, well-drained cherty and noncherty soils underlain by dolomite. They occur on rolling ridgetops and on all aspects of steep side slopes. The Collegedale-Gladeville-Rock outcrop soil unit consists of deep and shallow, rolling and hilly well-drained soils that are underlain by limestone and have many outcrops of limestone. Soils of this group occur on uplands. Soils of the Lehew-Armuchee-Muskingum unit are moderately deep, steep, well-drained soils underlain by multicolored shale, siltstone, and sandstone. This unit is found on high winding ridges. The Armuchee-Montevallo-Hamblen soil unit is made up of shallow-to-deep, steep to nearly level, well-drained and moderately well-drained soils underlain by shale. This unit occurs on uplands and bottomlands.

Prime farmland is land with the best combination of physical and chemical characteristics for producing food, feed, forage, fiber, and oilseed crops, and is available for these uses. While there are soils that would be classified as prime farmland on ORR, that designation is waived within the city limits of Oak Ridge and ORR (DOE 1999a:4-7).

3.2.5.2 Location of Proposed Activities

There are no capable faults on or near ORR. As noted above, the ORNL main site is underlain primarily by calcareous siltstones and silty-to-clean limestone of the Chickamauga Group. Melton Valley, and the 7900 Area in particular, are underlain by the interbedded limestones and shales of the Conasauga Group. Most of the 7900 Area is mapped as underlain by the Maryville Limestone with the southern limits of the site bordering the Nolichucky Shale (DOE 1999a:4-4, 4-5; LMER 1998:2.5-48) (Figures 3–3 and 3–4). In

particular, the bedrock beneath the HFIR complex is described as a dark-gray, calcareous clay shale overlain by up to 6 meters (20 feet) of saprolite (weathered bedrock) with only a thin topsoil (LMER 1998:2.5-16). Karst features are less developed in the Chickamauga Group than in the Knox Group. Cavities encountered are smaller and often clay-filled, and caves are sparse and typically small, with the same observation expected for the Conasauga Group (LMER 1999:2-8). Soils of ORNL, including the 7900 Area, are highly disturbed and would be classified as Urban Land. Urban Land includes areas where more than 80 percent of the surface is covered with industrial plants, paved parking lots, and other impervious surfaces (Moneymaker 1981:44).

3.2.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species.

3.2.6.1 Terrestrial Resources

This section addresses the plant and animal communities of ORR and includes a plant community map of the site. Terrestrial resources are described for the site as a whole, as well as the proposed facility locations.

3.2.6.1.1 General Site Description

Plant communities at ORR are characteristic of the intermountain regions of central and southern Appalachia. Only a small fraction of ORR has been disturbed by Federal activities; the remainder of the site has reverted to or been planted with natural vegetation. The vegetation of ORR has been categorized into seven plant communities (**Figure 3–5**). Pine and pine-hardwood forest is the most extensive plant community on the site. Another abundant community is the oak-hickory forest, which is commonly found on ridges throughout ORR. Northern hardwood forest and hemlock-white pine-hardwood forest are the least common forest community types on the site. Forest resources on ORR are managed for multiple use and sustained yield of quality timber products (DOE 1996b). Over 1,100 vascular plants species are found on ORR (LMER 1999).

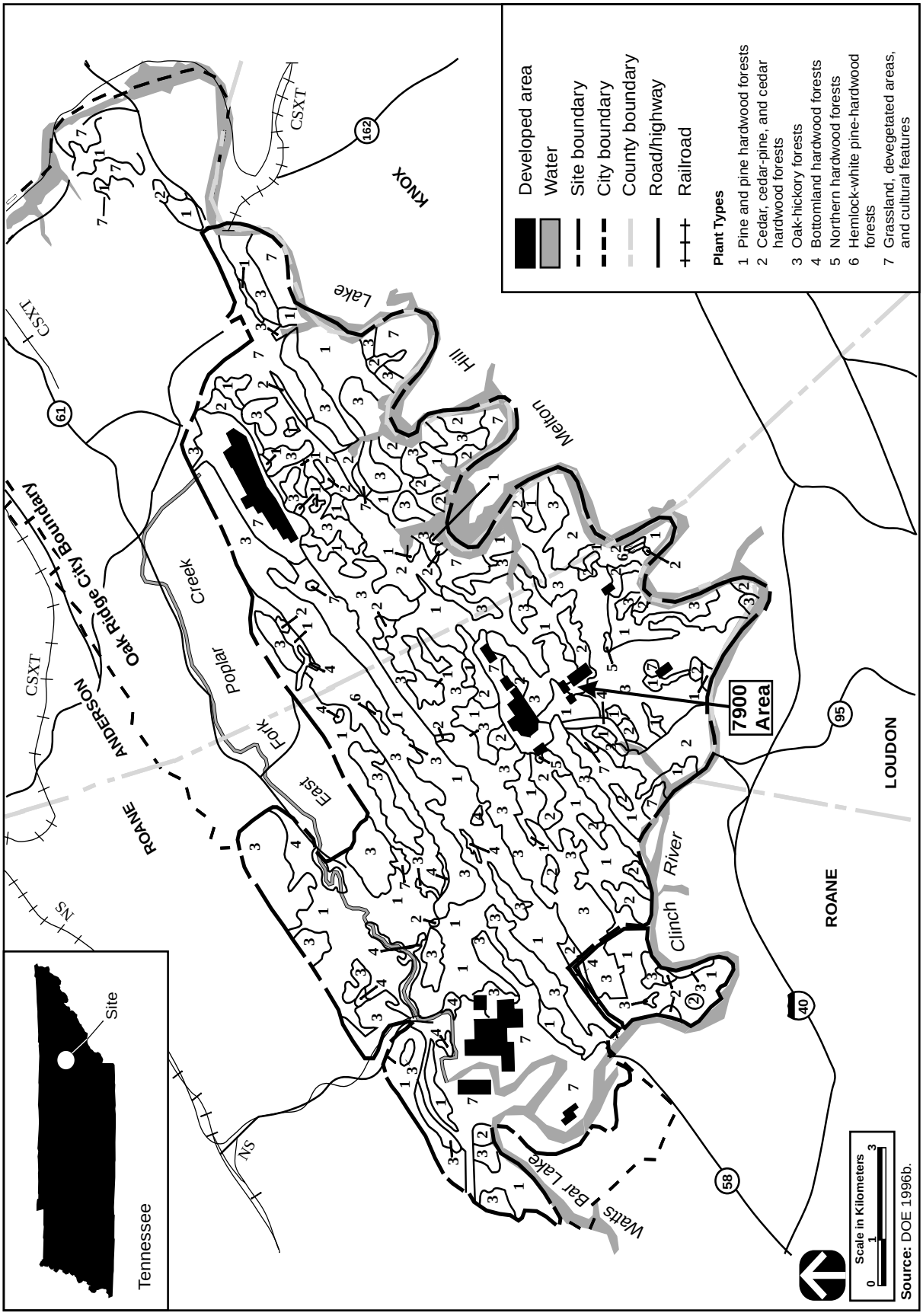
Animal species found on ORR include 59 amphibians and reptiles, 260 birds, and 38 mammals (LMER 1999). Animals commonly found on the site include the American toad, eastern garter snake, Carolina chickadee, northern cardinal, white-footed mouse, and raccoon. Most of ORR is within the Oak Ridge Wildlife Management Area. Wildlife management is carried out by the Tennessee Wildlife Resources Agency in cooperation with ORNL's Environmental Sciences Division. The whitetail deer and wild turkey are the only species hunted on site; however, other game animals are also present (LMER 1999). Raptors, such as the northern harrier and great horned owl, and carnivores, such as the gray fox and mink, are ecologically important groups on ORR. A variety of migratory birds have been found at ORR.

3.2.6.1.2 Location of Proposed Activities

Vegetative communities in the vicinity of the 7900 Area are typical of ORR as a whole, with pine, pine-hardwood forests, cedar, cedar-pine, cedar hardwood, and oak-hickory forests being the predominant community types (Figure 3–5). Fauna of the area are similar to that found throughout ORR. The 7900 Area itself is highly developed and provides minimal wildlife habitat.

3.2.6.2 Wetlands

Wetlands include "those areas that are inundated or saturated by surface or groundwater at a frequency and duration sufficient to support, and that under normal circumstances do support a prevalence of vegetation



Source: DOE 1996b.

Figure 3–5 Distribution of Plant Communities at the Oak Ridge Reservation

typically adapted for life in saturated soil conditions" (33 CFR Section 328.3). Wetlands are described for ORR as a whole, as well as for the proposed facility locations.

3.2.6.2.1 General Site Description

Approximately 235 hectares (580 acres) of wetlands occur on ORR (LMER 1999). These include emergent, scrub and shrub, and forested wetlands associated with bays (embayments) of the Melton Hill and Watts Bar Lake, areas bordering major streams and their tributaries (riparian), old farm ponds, and groundwater seeps. Well-developed communities of emergent wetland plants in the shallow embayments of the two reservoirs typically intergrade into forested wetland plant communities, which extend upstream through riparian areas associated with streams and their tributaries. Old farm ponds on ORR vary in size and support diverse plant communities and fauna. Although most riparian wetlands on ORR are forested, areas within utility rights-of way, such as those in Bear Creek and Melton Valley, support emergent wetland vegetation.

3.2.6.2.2 Location of Proposed Activities

There are six wetlands in the vicinity of the 7900 Area, including one small, unclassified wetland (Rosensteel 1996:25, 42); however, none are within the developed area. These wetlands, which were identified using the criteria and methods set forth in the *Corps of Engineers Wetland Delineation Manual* (Environmental Laboratory 1987), are generally classified as palustrine forested broad-leaved deciduous wetlands, although one also includes areas of emergent vegetation. Not including the unclassified wetland, the size of these areas range from 0.14 hectare (0.3 acre) to 1.23 hectares (3.0 acres). Mowing routinely disturbs two of the six wetlands.

3.2.6.3 Aquatic Resources

Aquatic resources at ORR are described for the site as a whole, as well as for the proposed facility locations.

3.2.6.3.1 General Site Description

Aquatic habitat on or adjacent to ORR ranges from small, free flowing streams in undisturbed watersheds to larger streams with altered flow patterns due to dam construction. These aquatic habitats include tailwaters, impoundments, reservoir embayments, and large and small perennial streams. Aquatic areas in ORR also include seasonal and intermittent streams and old farm ponds.

Sixty-three fish species have been collected on ORR (LMER 1999). The minnow family has the largest number of species and is numerically dominant in most streams. Fish species representative of the Clinch River in the vicinity of ORR are shad, herring, common carp, catfish, bluegill, crappie, and freshwater drum. The most important fish species taken commercially in the ORR area are common carp and catfish. Commercial fishing is permitted on the Clinch River downstream from Melton Hill Dam. Area recreational species consist of crappie, largemouth bass, sauger, sunfish, and catfish. Sport fishing is not permitted within ORR.

3.2.6.3.2 Location of Proposed Activities

ORNL is drained by White Oak Creek. The upper portion of the creek is similar to the upper reaches of other streams originating on Chestnut Ridge. These streams typically have alternating riffle and pool habitats. The stoneroller and blacknose dace are the fish species most commonly collected; 24 taxa of macroinvertebrates are present. Historically, operations at ORNL have had an adverse ecological effect on White Oak Creek. For

example, the influence of ORNL is reflected in the fact that benthic macroinvertebrate populations are less diverse downstream of the site than upstream (DOE 1999a).

There are three Aquatic Reference Areas and one Reference Area in the ORNL area: Aquatic Reference Areas 3, 4, and 5, and Reference Area 28 (Pounds, Parr, and Ryon 1993:5,15-17). Reference Areas are areas that are representative of the communities of the southern Appalachian region or that possess unique biotic features. Aquatic Reference Area 3, Northwest Tributary, is a second-order, frequently intermittent stream that flows along the wooded base of Haw Ridge, but with mowed fields, parking lots, and experimental ponds on the opposite bank. Aquatic Reference Area 4, First Creek, and Aquatic Reference Area 5, Fifth Creek, are first-order, spring-fed streams that flow out of Chestnut Ridge. Each area has rich benthic fauna, but is somewhat more limited with regard to the number of fish species present. Reference Area 28, Spring Pond, is a small spring-fed pond with unusually clear water for ponds on ORR; it is dominated by Nutall waterweed.

3.2.6.4 Threatened and Endangered Species

Endangered species are those plants and animals in danger of extinction throughout all or a large portion of their range. Threatened species are those species likely to become endangered within the foreseeable future. Threatened and endangered species are described for ORR as a whole, as well as for the proposed facility locations.

3.2.6.4.1 General Site Description

Forty Federal and state-listed threatened, endangered, and other special status species have been found on ORR; additional species that occur near the site may also be present (ORNL 2000). The gray bat (endangered) and bald eagle (threatened, but proposed to be delisted) are the only federally listed threatened or endangered species observed on or near ORR. The bald eagle has been seen on Melton Hill and Watts Bar Lakes (DOE 1996b). A dead gray bat was found several years ago at Y–12 (Barclay 1999). The Indiana bat (endangered) has not been reported from the site (Mitchell et al. 1996a; ORNL 2000). State-listed threatened or endangered species observed on ORR include the gray bat, bald eagle, peregrine falcon, osprey, and 10 plant species. No critical habitat for threatened or endangered species, as defined in the Endangered Species Act, exists on ORR or adjacent lakes. Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service. Consultation was also conducted with the state. The results of these consultations are presented in Chapter 4.

3.2.6.4.2 Location of Proposed Activities

No threatened, endangered, or sensitive plant or animal species have been recorded at or in the vicinity of the 7900 Area. Further, there is no potential habitat for such species confirmed in close proximity to the area (Parr 1999).

3.2.7 Cultural and Paleontological Resources

Cultural resources are human imprints on the landscape and are defined and protected by a series of Federal laws, regulations, and guidelines. The three general categories of cultural resources addressed in this section are prehistoric, historic, and Native American. Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age, and may be sources of information on paleoenvironments and the evolutionary development of plants and animals.

3.2.7.1 Prehistoric Resources

Prehistoric resources are physical properties that remain from human activities that predate written records.

3.2.7.1.1 General Site Description

More than 20 cultural resources surveys have been conducted at ORR. About 90 percent of ORR has received at least some preliminary walkover or archival-level study, but less than 5 percent has been intensively surveyed. Most cultural resources studies have occurred along the Clinch River and adjacent tributaries. Prehistoric sites recorded at ORR include villages, potential burial mounds, camps, quarries, a chipping station, limited activity locations, and shell scatters. More than 45 prehistoric sites have been recorded at ORR to date. At least 13 prehistoric sites are considered potentially eligible for the National Registry of Historic Places, but most of these sites have not yet been evaluated. Additional prehistoric sites may be anticipated in the unsurveyed portions of ORR. In 1994, a Programmatic Agreement concerning the management of historic and cultural properties at ORR was executed among the DOE Oak Ridge Operation Office, the Tennessee State Historic Preservation Officer, and the Advisory Council on Historic Preservation. This agreement was executed to satisfy DOE's responsibilities regarding Sections 106 and 110 of the National Historic Preservation Act, and resulted in DOE preparing a cultural resources management plan for ORR (Souza, DuVall, and Tinker 1997).

3.2.7.1.2 Location of Proposed Activities

No prehistoric properties have been located within or immediately adjacent to the 7900 Area (Souza, DuVall, and Tinker 1997:F-5).

3.2.7.2 Historic Resources

Historic resources consist of physical properties that postdate the existence of written records. In the United States, historic resources are generally considered to be those that date no earlier than 1492.

3.2.7.2.1 General Site Description

Several historic resources surveys have been conducted at ORR. Historic resources identified at ORR include both archaeological remains and standing structures. Documented log, wood frame, or fieldstone structures include cabins, barns, churches, gravehouses, springhouses, storage sheds, smokehouses, log cribs, privies, henhouses, and garages. Archaeological remains consist primarily of foundations, roads, and trash scatters. A total of 32 cemeteries are located within the present boundaries of ORR (Souza, DuVall, and Tinker 1997). More than 240 historic resources have been recorded at ORR, and 38 of those sites may be considered potentially eligible for listing on the National Registry of Historic Places. Freel's Cabin and two church structures, George Jones Memorial Baptist Church and the New Bethel Baptist Church, are listed on the National Registry. These structures date from before the establishment of the Manhattan Project. National Registry sites associated with the Manhattan Project include the Graphite Reactor at ORNL, listed on the National Registry of Historic Places as a National Historic Landmark, and three traffic checkpoints, Bear Creek Road, Bethel Valley Road, and Oak Ridge Turnpike Checking Stations (DOE 1999a). Many other buildings and facilities at ORR are associated with the Manhattan Project and are eligible for the National Registry. Historic building surveys have been completed for the Oak Ridge Townsite, ORNL, Y–12, the East Tennessee Technology Park, and the Oak Ridge Institute for Science and Education (Souza, DuVall, and Tinker 1997). Additional historic sites may be anticipated in the unsurveyed portions of ORR. Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office. The results of this consultation are presented in Chapter 4.

3.2.7.2.2 Location of Proposed Activities

A survey was conducted in 1993 to identify properties at ORNL that are included or are eligible for inclusion in the National Register of Historic Places. Eligible properties include the ORNL Historic District; Buildings 7001 and 7002 in the ORNL East Support Area; the Molten Salt Reactor Experiment Facility (Building 7503, previously known as the Aircraft Reactor Experiment Building); the Tower Shielding Facility; and White Oak Lake and Dam. Of these structures, the Molten Salt Reactor Experiment Facility is the closest eligible property to the 7900 Area. It is located about 0.4 kilometer (0.25 mile) to the north of REDC and HFIR (Souza, DuVall, and Tinker 1997:3-70).

3.2.7.3 Native American Resources

Native American resources are sites, areas, and materials important to Native Americans for religious or heritage reasons. In addition, cultural values are placed on natural resources such as plants, which have multiple purposes within various Native American groups. Concepts of sacred space that create the potential for land use conflicts are of primary concern.

3.2.7.3.1 General Site Description

The Overhill Cherokee Tribe occupied portions of the Tennessee, Hiwassee, Clinch, and Little Tennessee River Valleys in the 1700s. Overhill Cherokee villages consisted of a large townhouse, a summer pavilion, and a plaza, and residences had both summer and winter structures. Subsistence was based on hunting, gathering, and horticulture. The Cherokee were relocated to the Oklahoma territory in 1838, although some individuals refused to be moved and some Cherokee later returned to the area from Oklahoma. Resources that may be sensitive to Native American groups include remains of prehistoric and historic villages, ceremonial lodges, cemeteries, burials, and traditional plant gathering areas. Apart from prehistoric archaeological sites, to date no Native American resources have been identified at ORR.

3.2.7.3.2 Location of Proposed Activities

No Native American sacred sites or cultural items have been located within or immediately adjacent to the 7900 Area (Souza, DuVall, and Tinker 1997:3-66, 3-69, F-5).

3.2.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age. Paleontological remains consist of fossils and their associated geological information.

3.2.7.4.1 General Site Description

The majority of geological units with surface exposures at ORR contain paleontological materials. Paleontological materials consist primarily of invertebrate remains, and these have relatively low research potential.

3.2.7.4.2 Location of Proposed Activities

Paleontological resources at ORNL would not be expected to differ from those found elsewhere on ORR.

3.2.8 Socioeconomics

Statistics for employment and regional economy are presented for the regional economic area, as defined in Appendix G.8, which encompasses 15 counties around ORR in Tennessee. Statistics for population, housing, community services, and local transportation are presented for the region of influence, a four-county area in which 89.9 percent of all ORR employees reside (**Table 3–5**). In 1998, ORR employed 14,215 persons (about 3.4 percent of the regional economic area civilian labor force) (DOE 1999c).

Table 3–5 Distribution of Employees by Place of Residence in the ORR Region of Influence, 1998

County	Number of Employees	Total Site Employment (percent)
Anderson	4,061	28.6
Knox	5,615	39.5
Loudon	828	5.8
Roane	2,275	16.0
Region of influence total	12,779	89.9

Source: DOE 1999c.

3.2.8.1 Regional Economic Characteristics

Between 1990 and 1998, the civilian labor force in the ORR regional economic area increased 17.6 percent to the 1998 level of 484,774. In 1998, the unemployment rate in the regional economic area was 4.1 percent, which was slightly less than the unemployment rate for Tennessee (4.2 percent) (DOL 2000).

In 1993, services represented the largest sector of employment in the regional economic area (26 percent), followed by retail (19 percent), and manufacturing (18 percent). In Tennessee, the services sector comprised 26 percent of total employment, followed by manufacturing (19 percent), and retail (17 percent) (DOE 1996b).

3.2.8.2 Population and Housing

In 1998, the ORR region of influence population totaled 528,017. From 1990 to 1998, the region of influence population grew by 9.4 percent, compared to 10.9 percent growth in Tennessee (Forstall 1995; DOC 1999). Between 1980 and 1990, the number of housing units in the region of influence increased by about 13.8 percent, nearly 2 percent less than the increase for the entire State of Tennessee. In 1998, the total number of owner and rental housing units within the region of influence was 225,636. In 1990, the homeowner and rental vacancy rates for the region of influence were 1.7 percent, compared to the state's rate of 8.5 percent (DOE 1996b; State of Tennessee 2000; DOC 1992).

3.2.8.3 Community Services

3.2.8.3.1 Education

School districts providing public education in the ORR region of influence operated at capacities ranging from 74.7 to 100 percent. Total student enrollment in the region of influence in 2000 was 70,493. The average student-to-teacher ratio was 16.7:1 (Davis 2000; Garza 2000; Groover 2000; McKinney 2000; Pierce 2000).

3.2.8.3.2 Public Safety

In 1999, a total of 1,501 sworn police officers served the four-county region of influence. The average officer-to-population ratio was 2.8 officers per 1,000 persons (HPI 1999). In 1998, 1,293 paid and volunteer

firefighters provided fire protection services in the ORR region of influence. The average firefighter-to-population ratio was 2.4 firefighters per 1,000 persons (State of Tennessee 1998).

3.2.8.3.3 Health Care

In 1995, a total of 1,525 physicians served the ORR region of influence, with the majority practicing in Knox County (Randolph, Seidman, and Pasko, 1995). The average physician-to-population ratio was 3.0 physicians per 1,000 persons. In 1994, there were 13 hospitals serving the region of influence with a total of 2,833 beds (AHA 1995).

3.2.8.4 Local Transportation

Vehicles access ORR via three state routes. State Route 95 forms an interchange with Interstate 40, and enters the reservation from the south. State Route 58 enters the reservation from the west, and passes just south of the East Tennessee Technology Park. State Route 162 extends from Interstate 75 and Interstate 40 just west of Knoxville, and provides eastern access to ORR (Figure 3–1).

Within ORR, several routes are used to transfer traffic from the state routes to the main plant areas. Bear Creek Road, north of Y–12, flows in an east-west direction and connects Scarboro Road on the east end of the plant with State Road 95 and State Road 58. Bear Creek Road has restricted access around Y–12, and is not a public thoroughfare. Bethel Valley Road, a public roadway, provides access to ORNL, and extends from the east end of ORR at State Road 62 to the west end at State Route 95. Access to the 7900 Area is provided by secondary roads with controlled access: First Street, which runs north-south from Bethel Valley Road, and Melton Valley Road, which runs east-west and passes the 7900 Area entry road (McGee 2000).

Two main branches provide rail service for ORR. The CSX Transportation line at Elza (just east of Oak Ridge) serves Y–12 and the Office of Science and Technological Information in east Oak Ridge. The Norfolk and Southern main line from Blair provides easy access to the East Tennessee Technology Park. The Clinch River has a barge facility located on the west end of ORR near the East Tennessee Technology Park and is occasionally used to receive shipments that are too large or too heavy to be transported by rail or truck. McGhee Tyson Airport, 37 kilometers (23 miles) from ORR, is the nearest airport serving the region, with major carriers providing passenger and cargo service. A private airport, Atomic Airport, Inc., is the closest air transportation facility to Oak Ridge. Oak Ridge has a part-time public transportation system (DOE 1996b).

3.2.9 Existing Human Health Risk

Existing human health risk issues include the determination of potentially adverse effects on human health that result from acute and chronic exposures to ionizing radiation and hazardous chemicals.

3.2.9.1 Radiation Exposure and Risk

3.2.9.1.1 General Site Description

Major sources and levels of background radiation exposure to individuals in the vicinity of ORR are shown in **Table 3–6**. Annual background radiation doses to individuals are expected to remain constant over time. The total dose to the population, in terms of person-rem, changes as the population size changes. Background radiation doses are unrelated to ORR operations.

Table 3–6 Sources of Radiation Exposure to Individuals in the ORR Vicinity Unrelated to ORR Operations

Source	Effective Dose Equivalent (millirem per year)
Natural background radiation[a]	
Cosmic radiation	36
External terrestrial radiation	51
Internal terrestrial radiation	39
Radon in homes (inhaled)	200
Other background radiation[b]	
Diagnostic x-rays and nuclear medicine	53
Weapons test fallout	Less than 1
Air travel	1
Consumer and industrial products	10
Total	390

a. Hamilton et al. 1999.
b. NCRP 1987.
Note: Value of radon is an average for the United States.

Releases of radionuclides to the environment from ORR operations provide another source of radiation exposure to individuals in the vicinity of ORR. Types and quantities of radionuclides released from ORR normal operations in 1998 are listed in the *Oak Ridge Reservation Annual Site Environmental Report for 1998* (Hamilton et al. 1999). The doses to the public resulting from these releases are presented in **Table 3–7**. These doses fall within radiological limits per DOE Order 5400.5 and are much lower than those of background radiation.

Table 3–7 Radiation Doses to the Public from ORR Normal Operations in 1998 (Total Effective Dose Equivalent)

Members of the Public	Atmospheric Releases		Liquid Releases		Total	
	Standard[a]	Actual	Standard[a]	Actual	Standard[a]	Actual
Maximally exposed individual (millirem)	10	0.73	4	2.6[b]	100	4.4[c]
Population within 80 kilometers (person-rem)[d]	None	12.3	None	48	100	60.3
Average individual within 80 kilometers (millirem)[e]	None	0.014	None	0.055	None	0.069

a. The standards for individuals are given in DOE Order 5400.5. As discussed in that order, the 10-millirem-per-year limit from airborne emissions is required by the Clean Air Act, and the 4-millirem-per-year limit is required by the Safe Drinking Water Act; for this NI PEIS, the 4-millirem-per-year value is conservatively assumed to be the limit for the sum of doses from all liquid pathways. The total dose of 100 millirem per year is the limit from all pathways combined. The 100-person-rem value for the population is given in proposed 10 CFR Part 834, as published in 58 FR 16268. If the potential total dose exceeds the 100-person-rem value, it is required that the contractor operating the facility notify DOE.
b. These doses are mainly from drinking water (approximately 0.35 millirem) and eating fish from the Clinch River section of Poplar Creek.
c. This total dose includes a conservative value of 1 millirem per year from direct radiation exposure to a cesium field near the Clinch River.
d. Based on a population of about 880,000 in 1998.
e. Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site.
Source: Hamilton et al. 1999.

Using a risk estimator of 500 cancer deaths per 1 million person-rem to the public (Appendix H), the risk of a latent cancer fatality to the maximally exposed member of the public due to radiological releases from ORR operations in 1998 is estimated to be 2.2×10^{-6}. That is, the estimated probability of this person dying of cancer

at some point in the future from radiation exposure associated with 1 year of ORR operations is approximately 2 in 1 million, as it takes several to many years from the time of radiation exposure for a cancer to manifest itself.

According to the same risk estimator, 0.030 excess latent cancer fatality is projected in the population living within 80 kilometers (50 miles) of ORR from normal operations in 1998. To place this number in perspective, it may be compared with the number of cancer fatalities expected in the same population from all causes. The 1997 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year (Famighetti 1998:964). Based on this mortality rate, the number of cancer fatalities expected during 1998 from all causes in the population living within 80 kilometers (50 miles) of ORR was 1,760, which was much higher than the 0.030 latent cancer fatality estimated from ORR operations in 1998.

ORR workers receive the same doses as the general public from background radiation, but they also receive an additional dose from working in facilities with nuclear materials. The average dose to the individual worker and the cumulative dose to all workers at ORR from operations in 1998 are presented in **Table 3–8**. These doses fall within the radiological regulatory limits of 10 CFR Part 835. According to a risk estimator of 400 cancer fatalities per 1 million person-rem among workers (Appendix H), the number of projected latent cancer fatalities among ORR workers from normal operations in 1998 is 0.041.

Table 3–8 Radiation Doses to Workers from ORR Normal Operations in 1998 (Total Effective Dose Equivalent)

Occupational Personnel	Onsite Releases and Direct Radiation	
	Standard[a]	Actual
Average radiation worker (millirem)	None[b]	47
Total workers (person-rem)[c]	None	103

a. The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. It has therefore established the Administrative Control Level of 2,000 millirem per year; the site must make reasonable attempts to maintain individual worker doses below this level.
b. No standard is specified for an "average radiation worker"; however, the maximum dose that this worker may receive is limited to that given in footnote "a."
c. Based on a worker population of 2,187 with measurable doses in 1998.

Source: 10 CFR Section 835.202; DOE 1999p.

A more detailed presentation on the radiation environment, including background exposures and radiological releases and doses, is presented in the *Oak Ridge Reservation Annual Site Environmental Report for 1998* (Hamilton et al. 1999). The concentrations of radioactivity in various environmental media (including air, water, and soil) in the site region (on and off site) are also presented in the report.

3.2.9.1.2 Location of Proposed Activities

Radiological health effects resulting from the release of radionuclides from the stack that serves HFIR and REDC are shown in **Table 3–9**. Estimates shown in the table are based on the 1997 through 1999 release data discussed in Appendix H. Doses listed in Table 3–9 show that the risk of a latent cancer fatality to the maximally exposed member of the public due to emissions from HFIR and REDC would be 2.3×10^{-7}. The risk of an excess cancer fatality among the public residing within 80 kilometers (50 miles) of ATR would be 0.0042.

Table 3–9 Radiation Doses to the Public from Normal Operations at HFIR and REDC in 1999 (Total Effective Dose Equivalent)

Members of the Public	Atmospheric Releases		Liquid Releases		Total	
	Standard	Actual	Standard	Actual	Standard	Actual
Maximally exposed individual (millirem)	10	0.46	4	0	100	0.46
Population within 80 kilometers (person-rem)	None	8.4	None	0	100	8.4
Average individual within 80 kilometers (millirem)[a]	None	7.4×10^{-3}	None	0.00	None	7.4×10^{-3}

a. For an affected population of 1,134,000.

Table 3–10 lists average annual radiation doses to HFIR and REDC workers for the years 1998 and 1999. The average risk of an excess cancer fatality among workers at HFIR and REDC due to onsite releases and direct radiation for each of the two years would be 1.5×10^{-5} and 6.6×10^{-5}, respectively.

Table 3–10 Radiation Doses to Workers from HFIR and REDC Normal Operations (Total Effective Dose Equivalent)

Occupational Personnel	Onsite Releases and Direct Radiation	
	Standard[a]	Actual (1998 through 1999 average)
Average HFIR worker (millirem)	None[b]	38
Average REDC worker (millirem)	None[b]	165

a. The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. It has therefore established the Administrative Control Level of 2,000 millirem per year; the site must make reasonable attempts to maintain individual worker doses below this level.

b. No standard is specified for an "average radiation worker"; however, the maximum dose that this worker may receive is limited to that given in footnote "a."

Source: Boyd 2000a; Wham 2000d.

3.2.9.2 Chemical Environment

The background chemical environment important to human health consists of the atmosphere, which may contain hazardous chemicals that can be inhaled; drinking water, which may contain hazardous chemicals that can be ingested; and other environmental media through which people may come in contact with hazardous chemicals (e.g., surface water during swimming, soil through direct contact, or food). Hazardous chemicals can cause cancer and other adverse health effects.

Carcinogenic Effects. Health effects in this case are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogen. This could be incremental or excess individual lifetime cancer risk.

Noncarcinogenic Effects. Health effects in this case are determined by the ratio between the calculated, or measured concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the Hazard Quotient. Hazard Quotients for noncarcinogens are summed to obtain the Hazard Index. If the Hazard Index is less than 1, no adverse health effects would be expected.

Effective administrative and design controls that decrease hazardous chemical releases to the environment and help achieve compliance with permit requirements (e.g., air emissions and NPDES permit requirements) contribute to minimizing health impacts on the public. The effectiveness of these controls is verified through the use of monitoring information and inspection of mitigation measures. Health impacts on the public may

occur by inhaling air containing hazardous chemicals released to the atmosphere during normal ORR operations. Risks to public health from other possible pathways, such as ingestion of contaminated drinking water or direct exposure, are lower than those via the inhalation pathway.

Baseline concentrations are estimates of the highest existing offsite concentrations and represent the highest concentrations to which members of the public could be exposed from normal operations at ORR. These concentrations are in compliance with applicable guidelines and regulations. Information on estimating the health impacts of hazardous chemicals is presented in Appendix H.

Exposure pathways to ORR workers during normal operation may include inhaling contaminants in the workplace atmosphere and through direct contact with hazardous materials. The potential for health impacts varies among facilities and workers, and available information is insufficient for a meaningful estimate of impacts. However, workers are protected from workplace hazards through appropriate training, protective equipment, monitoring, substitution, and engineering and management controls. ORR workers are also protected by adherence to Occupational Safety and Health Administration (OSHA) and EPA standards that limit the workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Appropriate monitoring that reflects the frequency and amounts of chemicals used in the operational processes ensure that these standards are not exceeded. Additionally, DOE requires that conditions in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm.

3.2.9.3 Health Effects Studies

Two epidemiologic studies were conducted to determine whether ORR contributed to any excess cancers in communities surrounding the facility. One study found no excess cancer mortality in the population living in counties surrounding ORR, when compared to the control populations in other nearby counties and elsewhere in the United States. The other study found slight excess cancer incidences of several types in the counties near ORR, but less than the number of expected cancers incidences for other types of cancers.

A pilot study on mercury contamination conducted by the Tennessee Department of Health and Environment showed no difference in urine or hair mercury levels between individuals with potentially high mercury exposures compared to those with little potential for exposure. However, soil analysis showed that the mercury in soil is inorganic, which decreases the likelihood of a toxic accumlation in living tissue (bioaccumulation) and adverse health effects. Studies are continuing on the long-term effects of exposure to mercury and other hazardous chemicals.

More epidemiologic studies have been conducted to assess health effects on the population working at ORR than any other site reviewed for this document. Excess cancer mortalities have been reported and linked to specific job categories, age, and length of employment, as well as to the levels of exposure to radiation.

For a more detailed description of the epidemiologic studies, refer to Appendix M.4.6 of the *Storage and Disposition PEIS* (DOE 1996b:M-235 to M-242).

3.2.9.4 Accident History

There have been no safety-related accidents causing significant injury or harm to workers, or posing any sort of harm to the offsite public, at HFIR or REDC during their operational lifetimes (DOE 1999e).

In addition, there have been no accidents with a measurable impact on offsite population during nearly 50 years of Y–12 operations at ORR. The most noteworthy accident in Y–12's history was a 1958 criticality accident, which resulted in temporary radiation sickness for a few ORR employees. In 1989, there was a one-time

accidental release of xylene into the ORR sewer system with no offsite impacts. Accidental releases of anhydrous hydrogen fluoride occurred in 1986, 1988, and 1992, with little onsite and negligible offsite impacts. The hydrogen fluoride system where these accidents occurred is being modified to reduce the probability of future releases, and to minimize the potential consequences if a release does occur.

3.2.9.5 Emergency Preparedness

Each DOE site has established an emergency management program that would be activated in the event of an incident that threatens the health and safety of workers and the public. This program has been developed and maintained to ensure adequate response to most incident conditions and to provide response efforts for incidents not specifically considered. The emergency management program includes emergency planning, preparedness, and response.

DOE has overall responsibility for emergency planning and operations at ORR. However, DOE has delegated primary authority for event response to the operating contractor. Although the contractor's primary response responsibility is on site, the contractor does provide offsite assistance, if requested, under the terms of existing mutual aid agreements. If a hazardous materials event with offsite impacts occurs at a DOE facility, elected officials and local governments are responsible for the state's response efforts. The Tennessee Emergency Management Agency is the established agency responsible for coordinating state emergency services. When a hazardous materials event occurring at DOE facilities is beyond the capability of local government and assistance is requested, the Tennessee Emergency Management Agency Director may direct state agencies to provide assistance to the local governments. To accomplish this task and ensure prompt initiation of emergency response actions, the Director may cause the state Emergency Operations Center and Field Coordination Center to be activated. City or county officials may activate local Emergency Operations Centers in accordance with existing emergency plans.

DOE has specified actions to be taken at all DOE sites to implement lessons learned from the emergency response to an accidental explosion at Hanford in May 1997.

3.2.10 Environmental Justice

Under Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, Federal agencies are responsible for identifying and addressing the possibility of disproportionately high and adverse health, economic, and environmental impacts of programs and activities on minority and low-income populations in potentially affected areas. Minority populations refer to persons of any race self-designated as Asian, Black, Native American, or Hispanic. Low-income populations refer to households with incomes below the Federal poverty thresholds. In the case of ORR, the potentially affected area includes parts of Tennessee, North Carolina, and Kentucky.

The potentially affected area surrounding ORNL is defined by a circle with an 80-kilometer (50-mile) radius centered at HFIR/REDC (latitude 35° 55'8" N, longitude 84° 18'14" W). The total population residing within that area in 1990 was 881,987, while the minority population was 6.1 percent of that (DOC 1992). In 1990, approximately one-fourth of the total national population was comprised of persons self-designated as members of a minority group. Percentage minority populations residing in the States of Tennessee, North Carolina, and Kentucky were 17.4 percent, 25.0 percent, and 8.3 percent, respectively.

At the time of the 1990 census, Blacks were the largest minority group within the potentially affected area, constituting 4.7 percent of the total population. Asians made up 0.5 percent, Hispanics, 0.5 percent, and Native Americans made up 0.4 percent of the population (DOC 1992).

In 1990, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 137,708 persons (16 percent of the total population) residing within the potentially affected area around ORNL reported incomes below that threshold (DOC 1992). Data obtained during the 1990 census show that of the total population of the contiguous United States, 13.1 percent reported incomes below the poverty threshold. Percentages for those below the poverty threshold in Tennessee, North Carolina, and Kentucky were 19.0 percent, 13.1 percent, and 15.7 percent, respectively.

A more detailed description of the environmental justice analysis is given in Appendix K.

3.2.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing DOE activities. The waste is managed using appropriate treatment, storage and disposal technologies and in compliance with all Federal and state statutes and DOE orders. Disposal and management of previously generated ORR waste, known as legacy waste, is the responsibility of DOE's environmental management contractor, which is working to repackage, remove, and dispose of the existing legacy waste and newly generated wastes. The strategy is to dispose of current inventories of all waste types and close many of the existing storage facilities. The long-range strategy is to rely on a combination of onsite and offsite facilities to dispose of newly generated waste.

3.2.11.1 Waste Inventories and Activities

ORR manages the following types of waste: transuranic, mixed transuranic, low-level radioactive, mixed low-level radioactive, hazardous, and nonhazardous. Waste generation rates and the inventory of stored waste from activities at ORR are provided in **Table 3–11**. Waste generation rates specifically for HFIR and REDC activities are provided in **Table 3–12.** ORR waste management capabilities are summarized in **Table 3–13**. More detailed descriptions of the waste management system capabilities at ORR are included in the *Storage and Disposition PEIS* (DOE 1996b:3-219, E-63).

DOE is working with Federal and state regulatory authorities to address compliance and cleanup obligations arising from its past operations at ORR. DOE is engaged in several activities to bring its operations into full regulatory compliance. These activities are set forth in negotiated agreements that contain schedules for achieving compliance with applicable requirements and financial penalties for nonachievement of agreed-upon milestones.

EPA placed ORR on the National Priorities List, which identifies sites for possible long-term remedial action under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), on November 21, 1989. DOE, EPA Region IV, and the Tennessee Department of Environment and Conservation completed a Federal Facility Agreement, effective January 1, 1992. This agreement coordinates ORR inactive site assessment and remedial actions. Portions of the Federal Facility Agreement are applicable to operating waste management systems. Existing actions are conducted under the Resource Conservation and Recovery Act (RCRA) and applicable state laws that minimize duplication, expedite response actions, and achieve a comprehensive remediation of the site. More information on regulatory requirements for waste disposal is provided in Chapter 5.

Table 3–11 Waste Generation Rates and Inventories at ORR and ORNL

Waste Type	Generation Rates (cubic meters per year)		Inventory (cubic meters)	
	ORR[a]	ORNL	ORR[a]	ORNL
Transuranic				
Contact handled	12	12	1,000	1,000
Remotely handled	10	10	550	550
Remotely handled sludge (tank waste)	1.5	1.5	900	900
Low-level radioactive				
Liquid[b]	12,500 (total)	1,200	20,000[c] (total)	1,600
Solid		2,400		3,614
Process waste	283,900	283,900	0[d]	0[d]
Mixed low-level radioactive				
Liquid	(e)	(e)	(e)	(e)
Solid	1,600	475	26,000	3,000
Hazardous	36,000 kg/yr	–	1,689	–
Nonhazardous				
Liquid	269,000	60,600	NA[f]	NA[f]
Solid	29,500	5,700	NA[f]	NA[f]

a. Represents entire waste generated or managed at ORR, including ORNL.
b. Liquid low-level radioactive waste is processed through an evaporator for volume reduction, and the evaporator bottoms are stored as a concentrated solution.
c. Excludes waste from DOE environmental restoration activities.
d. This inventory is zero because the process waste is treated and discharged.
e. Mixed liquid low-level radioactive waste is reported as low-level radioactive waste. Certain contents are mixed-permit-by-rule.
f. Generally, this waste is not held in long-term storage.

Note: To convert from cubic meters to cubic yards, multiply by 1.308. To convert from kilograms to pounds, multiply by 2.2.
Key: kg/yr, kilograms per year; NA, not applicable.
Source: Brunson 1999; DOE 1997a; Wham 1999.

Table 3–12 Waste Generation Rates at HFIR and REDC

Waste Type	HFIR (cubic meters per year)	REDC (cubic meters per year)
Transuranic		
Contact handled	0	16
Remotely handled	0	9
Low-level radioactive		
Liquid	0	52
Solid	48	65
Process waste	19,700	0
Mixed low-level radioactive	0	<1
Hazardous	0	13,200 kg
Nonhazardous		
Liquid	138,200	96,700
Sanitary wastewater	7,310	3,130
Solid	0	294

Note: To convert from cubic meters to cubic yards, multiply by 1.308. To convert from kilograms to pounds, multiply by 2.2.
Key: kg, kilograms.
Source: Boyd 2000b; Valentine 2000; Wham 2000a, 2000b.

Table 3–13 Waste Management Capabilities at ORR

Facility Name/ Description	Capacity	Status	Applicable Waste Type					
			TRU	Mixed TRU	LLW	Mixed LLW	Haz	Non-Haz
Y–12: Treatment Facility (cubic meters per year except as otherwise specified)								
West End Treatment Facility, Building 9616-7	10,221	Online			X	X	X	X
Central Pollution Control Facility	10,200	Online			X	X	X	
Acid Neutralization and Recovery Facility, Building 9818	2,100	Online				X		
Uranium Chip Oxidizer Facility	Classified	Online			X			
Cyanide Treatment Facility	185	Online				X	X	
Plating Rinsewater Treatment Facility (Building 9623)	30,283	Online					X	X
Steam Plant Wastewater Facility	177,914	Online					X	X
Oak Ridge Sewage Treatment Plant (offsite) cubic meters per day	5,300	Online						X
Building 9720-25 Baler Facility	41,700	Online						X
Waste Coolant Processing Facility, Building 9983-78	1,363	Online			X	X		
Organic Handling Unit, Building 9815 (gallons per day)	500	Online			X	X		
Uranium Recovery Operations, Building 9212	2,100	Online				X		
Y–12: Storage Facility (cubic meters)								
Aboveground Storage Pads (Building 9830-2 through 7)	7,130	Online			X			
Buildings 9206 and 9212, Container Storage Areas	30	Online			X	X		
Building 9720-12, Container Storage Facility	123	Online			X	X		
Contaminated Scrap Metal Storage Yard	4,740	Online			X			X
Cyanide Treatment Facility (Building 9201-5N)	8	Online				X	X	
Liquid Organic Waste Storage Facility (Building 9720-45, OD-10)	198	Online				X	X	
Liquid Storage Facility (Building 9416-35)	416	Online				X	X	
PCB and RCRA Hazardous Drum Storage Facility (Building 9720-9)	1,404	Online				X	X	
RCRA and PCB Container Storage Area (Building 9720-58)	1,130	Online				X	X	
RCRA Staging and Storage Facility (Building 9720-31)	170	Online				X	X	
RCRA Storage Facility (Building 9811-1, OD-8)	723	Online			X	X	X	
Waste Oil/Solvent Storage Facility (Building 9811-8, OD-9)	790	Online			X	X	X	
Tank Farm, Building 9212	151	Planned				X		

Facility Name/ Description	Capacity	Status	Applicable Waste Type					
			TRU	Mixed TRU	LLW	Mixed LLW	Haz	Non-Haz
Container Storage Area/Production Waste Storage Facility, Building 9720-32	2,335	Online					X	
Low Level Waste Storage Pad, Building 9720-44	Not specified	Online			X			
Classified Waste (Container) Storage Area, Building 9720-59	1,090	Online			X	X		
Organic Handling Unit, Building 9815	8	Online					X	
Depleted Uranium Storage Vaults I and II (Building 9825-1 and 2 oxide vaults) and Building 9809	1,020	Online			X			
West Tank Farm	10,600	Online			X	X		
Y–12: Disposal Facility (cubic meters)								
Industrial and Sanitary Landfill V[a]	1,100,000[a]	Online						X
Construction Demolition Landfill VI[a]	119,000[a]	Online						X
Oak Ridge National Laboratory: Treatment Facility (cubic meters per year)								
Process Waste Treatment Plant	280,000	Online			X			
Melton Valley Low-Level Waste Immobilization Facility and Liquid Low-Level Waste Evaporation Facility	110,000	Online			X			
Waste Compaction Facility (Building 7831)	11,300	Online			X			
Sanitary Waste Water Treatment Facility (design capacity)	414,000	Online						X
Nonradiological Wastewater Treatment Facility	1,510,000	Online					X	
Oak Ridge National Laboratory: Storage Facility (cubic meters)								
Buildings 7826, 7834, 7842, 7878, 7879, and 7934	1,760	Online	X	X				
Bunker and Earthen Trenches (SWSA 5N Building 7855 and SWSA7 Building 7883)	1085	Online	X		X			
Liquid Low-Level Radioactive Waste Systems	3,230	Online			X			
Onsite tanks	7,850	Online			X			
Buildings 7507W, 7654, 7823, and Tank 7830a	393	Online Tank 7830a (standby)				X		
Hazardous Waste Storage Facility (Buildings 7507 and 7652) and Buildings 7651 and 7653	130	Online					X	
Interim Waste Management Facility	5,365 (1,730)[b]	Online			X			

Facility Name/ Description	Capacity	Status	Applicable Waste Type					
			TRU	Mixed TRU	LLW	Mixed LLW	Haz	Non-Haz
Oak Ridge National Laboratory: Disposal Facility (cubic meters)								
Shared Landfills V and VI	(Refer to footnote a)	Online						X
TRU Waste Treatment Facility (low-temperature drying) (five year capacity)	4,050	Planned (2002)	X	X	X	X		
East Tennessee Technology Park: Treatment Facility (cubic meters per year)								
TSCA Incinerator (Building K-1435)	15,700	Online			X	X		
Central Neutralization Facility (permitted operating capacity)	221,000	Online				X		
Sewage Treatment Plant (Building K-1203)	829,000	Online						X
East Tennessee Technology Park: Storage Facility (cubic meters)								
Building K-25, outside areas, K-1313 A and K-33	44,000	Online			X			
Current permitted container (solids/sludges/liquid wastes) and tank (liquids) storage capacity	97,000	Online				X		
Total current permitted waste pile unit storage capacity	120,000	Online				X		
Stockpiled at scrap yard	Not specified	Online						X
East Tennessee Technology Park: Disposal Facility (cubic meters)								
Shared Landfills V and VI	(Refer to footnote a)	Online						X

a. Industrial and Sanitary Landfill V and Construction Demolition Landfill VI serve all three sites for disposal of solid nonhazardous waste. Their disposal capacities are 1,100,000 cubic meters and 119,000 cubic meters, respectively.
b. Available as of June 1999.
Note: To convert from cubic meters to cubic yards, multiply by 1.308.
Key: Haz, hazardous; LLW, low-level waste; PCB, polychlorinated biphenyl; RCRA, Resource Conservation and Recovery Act; TRU, transuranic; TSCA, Toxic Substances Control Act.
Source: DOE 1996b:3-200–3-225; 2000c:2-14; PAI Corporation 1996; Rathke 2000; Wham 1999.

3.2.11.2 High-Level Radioactive Waste

ORR does not manage high-level radioactive waste at the site.

3.2.11.3 Transuranic Waste

Although ORNL is the only current generator of transuranic wastes on ORR, other sites at ORR have produced small quantities of transuranic wastes in the past and are likely to do so again during decontamination and decommissioning activities. Transuranic waste includes contact-handled transuranic and remotely handled transuranic. Normally, contact-handled transuranic waste consists primarily of miscellaneous waste from glovebox operations (e.g., paper, glassware, plastic, shoe covers, and wipes), discarded high-efficiency particulate air filters, and discarded equipment (e.g., gloveboxes and processing equipment). Contact-handled transuranic waste has a surface dose rate that does not exceed 200 millirem per hour. Generally, contact-handled transuranic waste is contained within polyethylene bags inside 208-liter (55-gallon) stainless steel drums. Metal paint cans, plastic buckets, and other similar containers are also used to package waste inside the drums.

Remotely handled transuranic waste consists primarily of miscellaneous hot cell waste (e.g., paper, glass, plastic tubing, and wipes), high efficiency particulate air filters, and discarded equipment (e.g., processing racks, vacuum pumps, and furnaces). Unshielded remotely handled transuranic waste packages typically have radiation levels that measure between 10 and 2,000 rem per hour; however, most are below 100 rem per hour. Shielding generally reduces the levels at the surface of the container to approximately 1 rem per hour. Remotely handled transuranic waste contains activation and fission products that decay and emit neutron and gamma radiation on the surface of the packaging that exceeds 200 millirem per hour. The activation materials are transuranium radionuclides ranging from plutonium-238 to californium-252, but are usually dominated by curium-244 which contributes to the neutron dose from spontaneous fission and alpha-n reactions. The alpha-n reactions contribute to the external dose rate measured at the surface of a container for both the contact-handled transuranic and remotely handled transuranic solid waste.

Remotely handled transuranic wastes are usually contained in concrete casks (1.4 meters [4.5 feet] in diameter by 2.3 meters [7.5 feet] high). The wall thicknesses of the casks are currently either 15 centimeters (6 inches) or 30.5 centimeters (12 inches) thick, depending on the radiation level of the contents. A large polyethylene bag is placed inside the cask for additional contamination control prior to use. Most remotely handled transuranic wastes inside the concrete casks are also contained inside polyethylene bags. Smaller waste packages such as 11-liter (2.9-gallon) plastic buckets, 3.7-liter (0.98-gallon) paint cans, and 18.9-liter (5.0-gallon) metal cans are packaged within the polyethylene bags. Fiber drums and carbon and steel drums have also been used to package waste inside the concrete cask. Intermediate-sized items that will not fit in the previously mentioned packages are generally placed in vinyl bags, then placed inside the lined waste cask. Large cask items may be placed directly in the cask.

As of January 1999, approximately 1,000 cubic meters (1,310 cubic yards) of contact-handled transuranic waste was in retrievable drum storage in the Bunker and Earthen Trenches. The amount of remotely handled transuranic waste was about 550 cubic meters (719 cubic yards) (64 FR 4079). Current activities center around certification of contact-handled waste, designing of a repackaging and certification facility for remote-handled wastes, and planning for shipment of waste to the Waste Isolation Pilot Plant (WIPP) or another suitable geologic repository for disposal.

3.2.11.4 Low-Level Radioactive Waste

Solid low-level radioactive waste is compactible radioactive waste such as paper, plastic, cloth, glass, cardboard, filters, floor sweepings, styrofoam, clothing, ceiling tile, and miscellaneous radioactively contaminated trash. The waste may include up to 20 percent lightweight or non-smeltable metal items. The solid low-level radioactive waste normally generated at ORNL consists primarily of radioactively contaminated personnel protection equipment, paper debris, trapping media, and process equipment. The Interim Waste Management Facility at ORNL only accepts low-level radioactive waste generated at ORNL. However, the Interim Waste Management Facility is at two-thirds of capacity, and access to this facility for the proposed plutonium-238 production, new medical and industrial isotope production, or new nuclear reasearch and development activities is not expected. Solid low-level radioactive waste is being stored at the East Tennessee Technology Park and Y–12 for future disposal. Contaminated scrap metal is stored above ground at the K-770 scrap metal facility, the old salvage yard at Y–12, and at ORNL which is being managed by the DOE scrap metal program until further disposal methods are evaluated.

The basic low-level radioactive waste strategy is to:

1. Use the Interim Waste Management Facility for legacy waste until it is filled to capacity.

2. Stage low-level radioactive waste at all sites, with emphasis on storage at the East Tennessee Technology Park until a disposal site is available.

3. Ship waste to the Nevada Test Site, Hanford, or a commercial disposal site as access is approved, and according to site-specific waste acceptance criteria.

3.2.11.5 Mixed Low-Level Radioactive Waste

RCRA mixed low-level radioactive waste is in storage at Y–12, East Tennessee Technology Park, and ORNL. Because prolonged storage of these wastes exceeded the one-year limit imposed by RCRA, ORR entered into a Federal Facility Compliance Agreement for RCRA Land Disposal Restriction wastes with EPA on June 12, 1992. This agreement was terminated with the issuance of the Tennessee Department of Environmental Conservation Commissioner's Order, effective October 1, 1995, which requires DOE to comply with the Site Treatment Plan prepared by ORR. The plan contains milestones and target dates for DOE to characterize and treat its inventory of mixed wastes at ORR. Sludges contaminated with low-level radioactivity are generated by settling and scrubbing operations, and in the past were stored in K-1407-C ponds at the East Tennessee Technology Park.

Sludges have been removed from these ponds and a portion has been fixed in concrete at the K–1419 Sludge Treatment Facility, and stored at the K–33 building. The concreted sludges are being shipped off site for disposal. The raw sludges are stored in the K–1065 building, pending further treatment. Mixed waste sludges are also generated at Y–12 in the treatment of nitrate waste from purification and recycling of uranium and in the treatment of plating shop waste.

The primary facility generator of liquid mixed waste is the K–1435 Toxic Substances Control Act Incinerator from the wet scrubber blowdown. This waste is currently being treated at the Central Neutralization Facility, which provides pH adjustment and chemical precipitation. Treated effluents are discharged through a NPDES outfall. The contaminated sludges are stored as mixed waste at the East Tennessee Technology Park.

The East Tennessee Technology Park Toxic Substances Control Act Incinerator has a design capacity to incinerate 909 kilograms (2,000 pounds) per hour of mixed liquid waste and up to 455 kilograms (1,000 pounds) per hour of solids and sludge (91 kilograms [200 pounds] per hour maximum sludge content). The Toxic Substances Control Act Incinerator is capable of incineration of both Toxic Substances Control Act- and RCRA-mixed wastes. The Toxic Substances Control Act Incinerator capacity utilization for incinerable solids is limited to ORR wastes to support the completion of enforceable milestones required by the ORR Site Treatment Plan. Because of permit limits (Toxic Substances Control Act, RCRA, State of Tennessee), the incinerator is not running at full capacity. In 1994, approximately 2,590 cubic meters (683,000 gallons) of mixed liquid waste was incinerated (DOE 1996b:3-226).

The major type of mixed waste generated at ORNL is mixed waste oils. Mixed waste oils are generated when oils are removed from systems that have operated in radiation environments. Radiation levels in these oils are typically low (less than or equal to 10 millirem per hour). Generally, these wastes consist of vacuum pump oil, axle oil, refrigeration oil, mineral oil, or oil/water mixtures. The principal components of scintillation fluids are toluene and/or xylene, culture medium, and miscellaneous organics. Other mixed wastes generated at ORNL include organic wastes, carcinogenic wastes, mercury-contaminated solid waste, waste solvents, corrosives, poisons, and other process waste. Because of the diversity of the mixed waste generated at ORNL, quantities are usually small.

Radioactive wastes contaminated with polychlorinated biphenyl are being stored because of lack of treatment and disposal capacities. DOE and EPA signed a Federal Facility Compliance Agreement, effective

December 16, 1996, to bring East Tennessee Technology Park into compliance with Toxic Substances Control Act regulations for use, storage, and disposal of polychlorinated biphenyls. It also addressed the approximately 10,000 pieces of nonradioactive polychlorinated biphenyls-containing dielectric equipment used in the shutdown of diffusion plant operations.

3.2.11.6 Hazardous Waste

RCRA-regulated wastes are generated by ORR in laboratory research, electroplating operations, painting operations, descaling, demineralizer regeneration, and photographic processes. Certain other wastes (e.g., spent photographic processing solutions) are processed on site into a nonhazardous state. Those wastes that are safe to transport, and have been certified as having no radioactivity added, are shipped off site to RCRA-permitted commercial treatment and disposal facilities. Small amounts of reactive chemical explosives that would be dangerous to transport off site, such as aged picric acid, are processed on site in the Chemical Detonation Facility at ORNL.

3.2.11.7 Nonhazardous Waste

Nonhazardous wastes are generated from numerous ORR activities. For example, the steam plant produces nonhazardous sludge. Scrap metals are discarded from maintenance and renovation activities and are recycled when appropriate. Construction and demolition projects produce nonhazardous industrial wastes. Other nonhazardous wastes include paper, plastic, glass, can, cafeteria wastes, and general trash. All nonradioactive medical wastes are autoclaved to render them noninfectious and are sent to the Y–12 Sanitary Landfill. Remedial action projects also produce wastes requiring proper management. The State of Tennessee permitted landfill (Construction Demolition Landfill VI) receives nonhazardous industrial materials such as fly ash and construction debris. Asbestos and general refuse are managed in Industrial and Sanitary Landfill V located at Y–12.

3.2.11.8 Waste Minimization

The DOE Oak Ridge Operations Office has an active waste minimization and pollution prevention program to reduce the total amount of waste generated and disposed of at ORR. This is accomplished by eliminating waste through source reduction or material substitution; recycling potential waste materials that cannot be minimized or eliminated; and treating waste generated to reduce its volume, toxicity, or mobility prior to storage or disposal. Implementing pollution prevention projects reduced the amount of waste generated at ORR in 1998 by approximately 64,900 cubic meters (84,000 cubic yards). Examples of pollution prevention projects completed in 1998 at the Oak Ridge Operations Office include: reducing cleanup/stabilization of low-level radioactive waste by approximately 395 cubic meters (517 cubic yards), mixed low-level radioactive waste by approximately 119 cubic meters (156 cubic yards), and hazardous waste by approximately 83 metric tons (91 tons) by providing incentives in contracts for projects to turn over vacant and decontaminated buildings to the DOE Oak Ridge Operation Office; reducing routine operations mixed low-level radioactive waste by approximately 693 cubic meters (906 cubic yards) by selling various scrap metals (including clean and contaminated carbon steel and copper) to an outside vendor for cleaning and recycling; and reducing transuranic waste generation by less than 1 cubic meter (1.3 cubic yards) per year by replacing three oil-lubricated vacuum pumps with dry pumps, which eliminated the transuranic-contaminated waste oil stream and associated waste (DOE 1999f:56).

3.2.11.9 Waste Management PEIS Records of Decision

The *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (*Waste Management PEIS*) Records of Decision affecting ORR are shown in **Table 3–14** for the waste types analyzed in this NI PEIS. Decisions on the various waste types are being announced in a series of Records of Decision that have been issued on the *Waste Management PEIS*. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629); the hazardous waste Record of Decision was issued on August 5, 1998 (63 FR 41810); the high-level radioactive waste Record of Decision was issued on August 12, 1999 (64 FR 46661); and the low-level radioactive waste and mixed low-level radioactive waste Record of Decision was issued on February 18, 2000 (65 FR 10061). The transuranic waste Record of Decision states that DOE will develop and operate mobile and fixed facilities to characterize and prepare transuranic waste for disposal at WIPP. Each DOE site that has or will generate transuranic waste will, as needed, prepare and store its transuranic waste on site. The hazardous waste Record of Decision states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of the nonwastewater hazardous waste, with ORR and the Savannah River Site (SRS) continuing to treat some of their own nonwastewater hazardous waste on site in existing facilities, where this is economically favorable. The high-level radioactive waste Record of Decision states that immobilized high-level radioactive waste will be stored at the site of generation until transfer to a geologic repository. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision states that for the management of low-level radioactive waste, minimal treatment will be performed at all sites and disposal will continue, to the extent practicable, on site at INEEL, Los Alamos National Laboratory (LANL), ORR, and SRS. In addition, Hanford and the Nevada Test Site will be available to all DOE sites for low-level radioactive waste disposal. Mixed low-level radioactive waste will be treated at Hanford, INEEL, ORR, and SRS and disposed of at Hanford and the Nevada Test Site. More detailed information concerning DOE's preferred alternatives for the future configuration of waste management facilities at ORR is presented in the *Waste Management PEIS* and the high-level radioactive waste, transuranic waste, hazardous waste, and low-level radioactive and mixed low-level radioactive waste Records of Decision.

Table 3–14 Waste Management PEIS Records of Decision Affecting ORR

Waste Type	Preferred Action
High-level radioactive	ORR does not currently manage high-level radioactive waste.[a]
Transuranic and mixed transuranic	DOE has decided that ORR should prepare and store its transuranic waste on site pending disposal at WIPP[b] or another suitable geologic repository.
Low-level radioactive	DOE has decided to treat ORR's liquid low-level radioactive waste on site.[c] Separate from the *Waste Management PEIS*, DOE prefers offsite management of ORR's solid low-level radioactive waste after temporary onsite storage.
Mixed low-level radioactive	DOE has decided to regionalize treatment of mixed low-level radioactive waste at ORR.[c] This includes the onsite treatment of ORR's waste and could include treatment of some mixed low-level radioactive waste generated at other sites.
Hazardous	DOE has decided to use commercial and onsite ORR facilities for treatment of ORR nonwastewater hazardous waste. DOE will also continue to use onsite facilities for wastewater hazardous waste.[d]

a. From the Record of Decision for high-level radioactive waste (64 FR 46661).
b. From the Record of Decision for transuranic waste (63 FR 3629).
c. From the Record of Decision for low-level radioactive and mixed low-level radioactive waste (65 FR 10061).
d. From the Record of Decision for hazardous waste (63 FR 41810).

Source: 63 FR 3629; 63 FR 41810; 64 FR 46661; 65 FR 10061.

3.3 IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL LABORATORY

INEEL is on 230,700 hectares (570,000 acres) in southeastern Idaho and is 55 kilometers (34 miles) west of Idaho Falls, 61 kilometers (38 miles) northwest of Blackfoot, and 35 kilometers (22 miles) east of Arco. INEEL is owned by the Federal Government and administered, managed, and controlled by DOE. It is primarily within Butte County, but portions of the site are also in Bingham, Jefferson, Bonneville, and Clark counties. The site is roughly equidistant from Salt Lake City, Utah, and Boise, Idaho.

There are approximately 450 buildings and 2,000 support structures at INEEL, with more than 279,000 square meters (3,000,000 square feet) of floor space in varying conditions of utility. INEEL has approximately 25,100 square meters (270,000 square feet) of covered warehouse space and an additional 18,600 square meters (200,000 square feet) of fenced yard space. The total area of the various machine shops is 3,035 square meters (32,665 square feet).

Fifty-two research and test reactors have been designed and deployed at INEEL over the years to test reactor systems, develop fuel and target designs, and test the overall safety of reactor systems. In addition to nuclear reactor research, other INEEL facilities are operated to support reactor operations. These facilities include high-level radioactive and low-level radioactive waste processing and storage sites; hot cells; nuclear materials storage vaults; analytical laboratories; machine shops; laundry, railroad, and administrative facilities. Other activities include management of one of DOE's largest storage sites for low-level radioactive waste, transuranic waste, and spent nuclear fuel (both highly enriched and low-enriched uranium).

3.3.1 Land Resources

Land resources include land use and visual resources. Each of these resource areas is described for the site as a whole, as well as for the locations of the proposed activities.

3.3.1.1 Land Use

Land may be characterized by its potential for the location of human activities (land use). Natural resource attributes and other environmental characteristics could make a site more suitable for some land uses than for others. Changes in land use may have both beneficial and adverse effects on other resources such as ecological, cultural, geological, aquatic, and atmospheric.

3.3.1.1.1 General Site Description

The Federal Government, the State of Idaho, and private parties own lands surrounding INEEL. Regional land uses include grazing, wildlife management, mineral and energy production, recreation, and crop production. Approximately 60 percent of the surrounding area is used by sheep and cattle for grazing. Small communities and towns near the INEEL boundaries include Mud Lake to the east; Arco, Butte City, and Howe to the west; and Atomic City to the south. Two National Natural Landmarks border INEEL: Big Southern Butte (2.4 kilometers [1.5 miles] south) and Hell's Half Acre (2.6 kilometers [1.6 miles] southeast). A portion of Hell's Half Acre National Natural Landmark is designated as a Wilderness Study Area. The Black Canyon Wilderness Study Area is also adjacent to INEEL.

Land use categories at INEEL include facility operations, grazing, general open space, and infrastructure such as roads. Generalized land uses at INEEL and vicinity are shown in **Figure 3–6**. Facility operations include industrial and support operations associated with energy research and waste management activities. Land is also used for recreation and environmental research associated with the designation of INEEL as a National

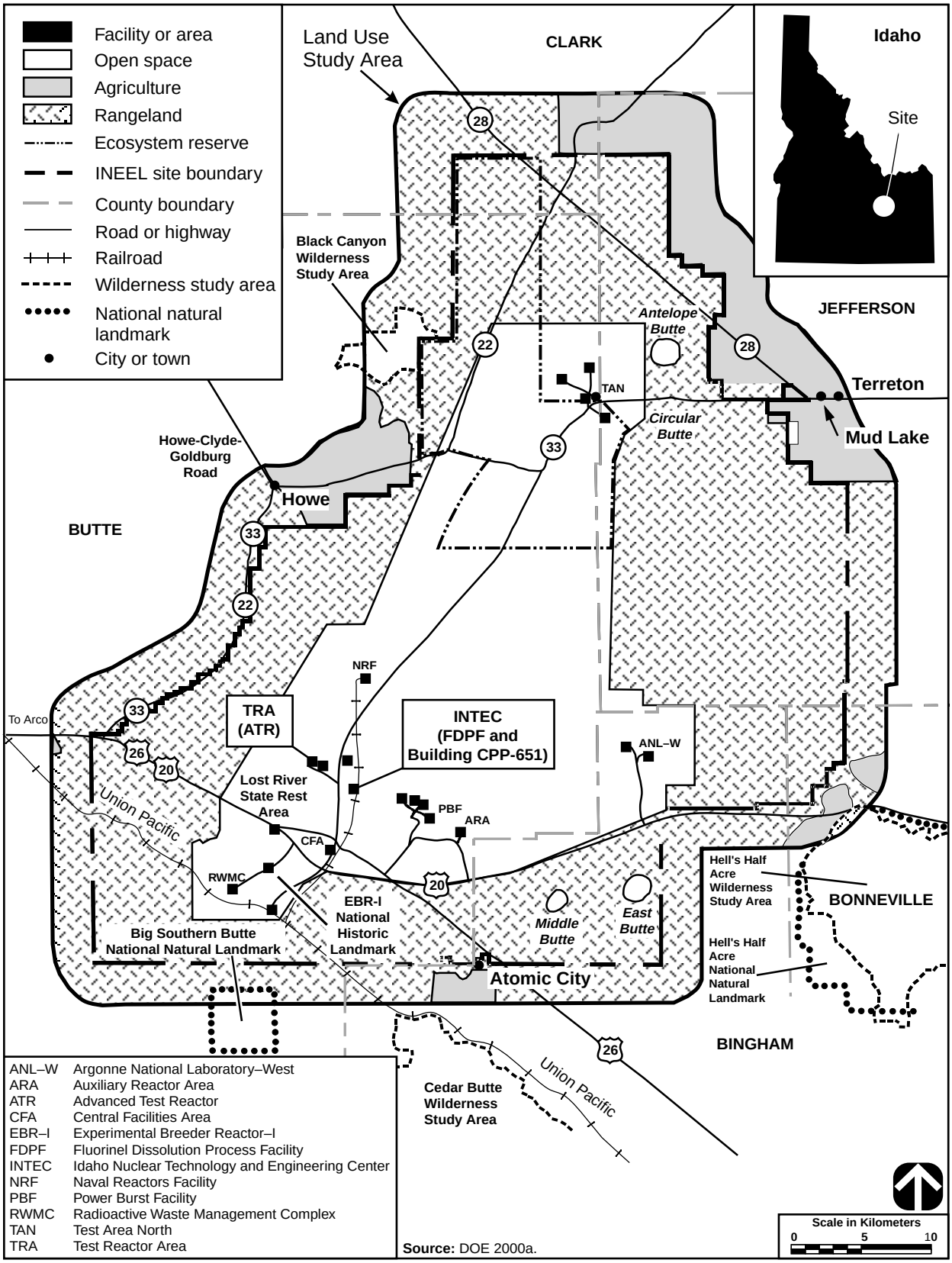


Figure 3–6 Generalized Land Use at Idaho National Engineering and Environmental Laboratory and Vicinity

Environmental Research Park. Much of INEEL is open space that has not been designated for specific use. Some of this space serves as a buffer zone between INEEL facilities and other land uses. Recently, 29,950 hectares (74,000 acres) of open space in the north central portion of the site has been designated as the INEEL Sagebrush Steppe Ecosystem Reserve (DOE 1999g). This area represents one of the last sagebrush steppe ecosystems in the United States and provides a home for a number of rare and sensitive species of plants and animals. Approximately 2 percent of the total INEEL site area (4,600 hectares [11,400 acres]) is used for facilities and operations. Facilities are sited within a central core area of about 93,100 hectares (230,000 acres) (Figure 3–6). Public access to most facilities is restricted. DOE land use plans and policies applicable to INEEL are discussed in the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995a:vol. 2, part A, 4.2-1–4.2-4).

All county plans and policies encourage development adjacent to previously developed areas to minimize the need for infrastructure improvements and to avoid urban sprawl. Because INEEL is remote from most developed areas, its lands and adjacent areas are not likely to experience residential and commercial development, and no new development is planned near the site. Recreational and agricultural uses, however, are expected to increase in the surrounding area in response to greater demand for recreational areas and the conversion of rangeland to cropland (DOE 1999e:3-82).

The Fort Bridger Treaty of July 3, 1868, secured the Fort Hall Reservation as the permanent homeland of the Shoshone-Bannock Peoples. According to the treaty, tribal members reserved rights to hunting, fishing, and gathering on surrounding unoccupied lands of the United States. While INEEL is considered occupied land, it was recognized that certain areas on the INEEL site have significant cultural and religious significance to the tribes. A *1994 Memorandum of Agreement with the Shoshone-Bannock Tribes* (DOE 1994) provides tribal members access to the Middle Butte to perform sacred or religious ceremonies or other educational or cultural activities.

3.3.1.1.2 Locations of Proposed Activities

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Land within Idaho Nuclear Technology and Engineering Center (INTEC) is highly disturbed, and is used to store spent nuclear fuel and radioactive wastes, treat radioactive wastes, and develop waste management technologies. The area includes about 85 hectares (210 acres) within the perimeter fence and an additional 22 hectares (54 acres) outside the fence (DOE 1997b:31, 95–111). A number of wastewater and percolation ponds are also present on the site. INTEC is 12 kilometers (7.5 miles) north of the site boundary, and 0.8 kilometer (0.5 mile) southeast of the Big Lost River. Facilities at INTEC include spent fuel storage and processing areas, a waste solidification facility and related high-level waste storage facilities, remote analytical laboratories, and a coal-fired steam-generating plant that is in standby.

TEST REACTOR AREA

The Test Reactor Area is in the southwestern portion of INEEL (Figure 3–6). Land in the Test Reactor Area is currently disturbed, and is designated for reactor operations. The area includes about 15 hectares (37 acres) within the security fence, plus several sewage and waste ponds outside of the fence. The Test Reactor Area is about 11 kilometers (6.8 miles) southeast of the nearest site boundary and about 2.6 kilometers (1.6 miles) northwest of the Big Lost River. The Materials Test Reactor and Engineering Test Reactor (both shut down), the Test Reactor Area Hot Cells, and ATR, which achieved initial criticality in 1967, are in the Test Reactor Area. In addition, numerous support facilities (i.e., storage tanks, maintenance buildings, warehouses), laboratories, and sanitary and radioactive waste treatment facilities are in the area (DOE 1997b:32, 189–201).

3.3.1.2 Visual Resources

Visual resources are natural and human-created features that give a particular landscape its character and aesthetic quality. Landscape character is determined by the visual elements of form, line, color, and texture. All four elements are present in every landscape. The stronger the influence exerted by these elements in a landscape, the more interesting the landscape.

3.3.1.2.1 General Site Description

The Bitterroot, Lemhi, and Lost River mountain ranges border INEEL on the north and west. Volcanic buttes near the southern boundary of INEEL can be seen from most locations on the site. INEEL generally consists of open desert land predominantly covered by big sagebrush and grasslands. Pasture and farmland border much of the site.

Ten facility areas are on the INEEL site. Although INEEL has a comprehensive facility and land use plan (DOE 1997b), no specific visual resource standards have been established. INEEL facilities have the appearance of low-density commercial/industrial complexes widely dispersed throughout the site. Structure heights generally range from 3 to 30 meters (10 to 100 feet); a few stacks and towers reach 76 meters (250 feet). Although many INEEL facilities are visible from highways, most facilities are more than 0.8 kilometer (0.5 mile) from public roads. The operational areas are well defined at night by the security lights.

Lands adjacent to INEEL, under Bureau of Land Management jurisdiction, are designated as Visual Resource Management Class II areas. Lands within the INEEL site are designated as Visual Resource Management Class II and III. Management activities within these classes may be seen but should not dominate the review (DOI 1986). The Black Canyon Wilderness Study Area, adjacent to INEEL, is under consideration by Bureau of Land Management for Wilderness Area designation, approval of which would result in an upgrade of its Visual Resource Management rating from Class II to Class I. The Hell's Half Acre Wilderness Study Area is 2.6 kilometers (1.6 miles) southeast of INEEL's eastern boundary. This area, famous for its lava flow and hiking trails, is managed by the Bureau of Land Management. The Craters of the Moon Wilderness Area is about 20 kilometers (12 miles) southwest of INEEL's western boundary.

3.3.1.2.2 Locations of Proposed Activities

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While the Fuel Processing Facility is the largest building at INTEC, the tallest structure is the main stack, which is 76 meters (250 feet) tall. The Visual Resource Management rating of INTEC is Class IV, which means management activities dominate the view and are the focus of the viewers attention. INTEC is visible in the middle ground from State Highways 20 and 26, with Saddle Mountain in the background. Natural features of visual interest within a 40-kilometer (25-mile) radius include Big Lost River at 0.8 kilometer (0.5 mile), Middle Butte at 18 kilometers (11 miles), Big Southern Butte National Natural Landmark at 20 kilometers (11 miles), East Butte at 23 kilometers (14 miles), Hell's Half Acre Wilderness Study Area at 33 kilometers (21 miles), and Saddle Mountain at 40 kilometers (25 miles).

TEST REACTOR AREA

The tallest structure at ATR within the Test Reactor Area is the main stack, which can be seen from Highways 20, 26, and 22. Developed areas within the Test Reactor Area are consistent with a Visual Resource Management Class IV rating. Natural features of visual interest within a 40-kilometer (25-mile) radius include

Big Lost River at 2.6 kilometers (1.6 miles), Middle Butte at 20 kilometers (12 miles), Big Southern Butte National Natural Landmark at 18 kilometers (11 miles), East Butte at 23 kilometers (14 miles), Hell's Half Acre Wilderness Study area at 35 kilometers (22 miles), and Saddle Mountain at 40 kilometers (25 miles).

3.3.2 Noise

Noise is unwanted sound that interferes or interacts negatively with the human or natural environment. Noise may disrupt normal activities or diminish the quality of the environment.

3.3.2.1 General Site Description

Major noise emission sources within INEEL include various industrial facilities, equipment, and machines (e.g., cooling systems, transformers, engines, pumps, boilers, steam vents, paging systems, construction and materials-handling equipment, and vehicles). Most INEEL industrial facilities are far enough from the site boundary that noise levels at the boundary from these sources are not measurable, or are barely distinguishable from background levels (DOE 1996b:3-112).

Existing INEEL-related noises of public significance are from the transportation of people and materials to and from the site and in-town facilities via buses, trucks, private vehicles, and freight trains. Noise measurements along U.S. Route 20, about 15 meters (50 feet) from the roadway, indicate that the sound levels from traffic range from 64 to 86 dBA, and that the primary source is buses (71 to 80 dBA). While few people reside within 15 meters (50 feet) of the roadway, the results indicate that INEEL traffic noise might be objectionable to members of the public residing near principal highways or busy bus routes. Noise levels along these routes may have decreased somewhat due to reductions in employment and bus service at INEEL in the last few years. The acoustic environment along the INEEL site boundary in rural areas and at nearby areas away from traffic noise is typical of a rural location; the average day-night sound level is in the range of 35 to 50 dBA. Except for the prohibition of nuisance noise, neither the State of Idaho nor local governments have established any regulations that specify acceptable community noise levels applicable to INEEL (DOE 1996b:3-114). The EPA guidelines for environmental noise protection recommend an average day-night sound level of 55 dBA as sufficient to protect the public from the effects of broadband environmental noise in typically quiet outdoor and residential areas (EPA 1974:29). Land use compatibility guidelines adopted by the Federal Aviation Administration and the Federal Interagency Committee on Urban Noise indicate that yearly day-night average sound levels less than 65 dBA are compatible with residential land uses (14 CFR Part 150). These guidelines further indicate that levels up to 75 dBA are compatible with residential uses if suitable noise reduction features are incorporated into structures. It is expected that for most residences near INEEL, the day-night average sound levels are compatible with the residential land use, although for some residences along major roadways, noise levels may be higher than 65 dBA.

3.3.2.2 Locations of Proposed Activities

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No distinguishing noise characteristics at INTEC have been identified. INTEC is far enough from the site boundary (12 kilometers [7.5 miles]) that noise levels at the boundary from these sources are not measurable or are barely distinguishable from background levels.

TEST REACTOR AREA

No distinguishing noise characteristics at the Test Reactor Area have been identified. The Test Reactor Area is far enough from the site boundary (11 kilometers [6.8 miles]) that noise levels at the site boundary from these sources are not measurable or are barely distinguishable from background levels.

3.3.3 Air Quality

Air pollution refers to the introduction, directly or indirectly, of any substance into the air that could endanger human health, harm living resources and ecosystems as well as material property, and impair or interfere with the comfortable enjoyment of life or other legitimate uses of the environment. Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Air quality is affected by air pollutant emission characteristics, meteorology, and topography.

3.3.3.1 General Site Description

The climate at INEEL and the surrounding region is characterized as that of a semiarid steppe. The average annual temperature at INEEL is 5.6 °C (42 °F); average monthly temperatures range from a minimum of -8.8 °C (16.1 °F) in January to a maximum of 20 °C (68 °F) in July. The average annual precipitation is 22 centimeters (8.7 inches) (Clawson, Start, and Ricks 1989:55, 77). Prevailing winds at INEEL are southwest or northeast (DOE 1999h:4.7-1). The annual average wind speed is 3.4 meters per second (7.5 miles per hour) (DOE 1996b:3-112).

INEEL is within the Eastern Idaho Intrastate Air Quality Control Region #61. None of the areas within INEEL and its surrounding counties are designated as nonattainment areas with respect to the NAAQS for criteria air pollutants (40 CFR Section 81.313). The nearest nonattainment area for particulate matter is in Pocatello, about 80 kilometers (50 miles) to the south. Applicable NAAQS and Idaho State ambient air quality standards are presented in **Table 3–15**.

The primary sources of air pollutants at INEEL include calcination of sodium-bearing waste, combustion of coal for steam, and combustion of fuel oil for heating. Other emission sources include waste burning, coal piles, industrial processes, stationary diesel engines, vehicles, and fugitive dust from waste burial and construction activities. The existing ambient air concentrations attributable to sources at INEEL are presented in Table 3–15. These concentrations are based on dispersion modeling at the INEEL site boundary centered at the INTEC facility, performed for the *High-Level Waste and Facilities Disposition Draft EIS* using 1997 actual emissions and excluding Argonne National Laboratory–West; dispersion modeling at the INEEL site boundary centered on Argonne National Laboratory–West using 1997 actual emissions for Argonne National Laboratory–West; and meteorological data from 1991–1992 (DOE 1999i, 2000a). The estimated concentrations are conservative and bound the actual INEEL contribution to ambient levels, as some of the modeled sources are currently in standby. Concentrations shown in Table 3–15 represent a small percentage of the ambient air quality standards. Concentrations of any hazardous and toxic compounds would be well below regulatory levels.

Because INEEL sources are limited and background concentrations of criteria pollutants are well below ambient standards, INEEL emissions should not result in air pollutant concentrations that violate the ambient air quality standards.

Table 3–15 Comparison of Modeled Ambient Air Concentrations from INEEL Sources with Most Stringent Applicable Standards or Guidelines

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meters)[a]	INEEL Concentration without ANL–W (micrograms per cubic meters)	ANL–W Concentration (micrograms per cubic meters)
Criteria pollutants				
Carbon monoxide	8 hours	10,000[b]	78	41
	1 hour	40,000[b]	206	59
Nitrogen dioxide	Annual	100[b]	0.46	13
Ozone	1 hour	235[c]	(d)	(d)
PM_{10}	Annual	50[b]	0.49	0.14
	24 hours	150[b]	12	1.1
Sulfur dioxide	Annual	80[b]	0.14	3.3
	24 hours	365[b]	5.3	27
	3 hours	1,300[b]	24	60

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The NAAQS (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic PM_{10} mean standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Federal and state standard.
c. Federal 8-hour standard is currently under litigation.
d. Not directly emitted or monitored by the site.

Note: NAAQS also include standards for lead. No sources of lead emissions have been identified for any alternative evaluated. Emissions of hazardous air pollutants not listed here have been identified at INEEL, but are not associated with any of the alternatives evaluated. EPA revised the ambient air quality standards for particulate matter and ozone in 1997 (62 FR 38856, 62 FR 38652); however, these standards are currently under litigation, but could become enforceable during the life of this project.
Source: 40 CFR Part 50; DOE 1999i, 2000a; ID DHW 1998.

The nearest Prevention of Significant Deterioration Class I area to INEEL is Craters of the Moon Wilderness Area, Idaho, 53 kilometers (33 miles) west-southwest from the center of the site. A Class I area is one in which very little increase in pollution is allowed due to the pristine nature of the area. There are no other Class I areas within 100 kilometers (62 miles) of INEEL. INEEL and its vicinity are classified as a Class II area in which more moderate increases in pollution are allowed (DOE 1996b:3-112).

EPA has established Prevention of Significant Deterioration increments for certain pollutants: sulfur dioxide, nitrogen dioxide and particulate matter less than or equal to 10 microns in diameter. The increments specify a maximum allowable increase above a certain baseline concentration for a given averaging period, and apply only to sources constructed or modified after a specified baseline date. These sources are known as increment-consuming sources. The baseline date is the date of submittal of the first application for a Prevention of Significant Deterioration permit in a given area.

Prevention of Significant Deterioration permits have been obtained for the coal-fired steam-generating facility next to the Idaho Nuclear Technology and Engineering Center and Fuel Processing Facility, which is not expected to be operated (DOE 1996b). In addition to this facility, INEEL has other increment consuming sources on site. The current amounts of Prevention of Significant Deterioration increment consumption in Class I and Class II areas by INEEL's increment-consuming sources based on dispersion modeling analyses are specified in **Tables 3–16** and **3–17**, respectively.

Table 3–16 Prevention of Significant Deterioration Increment Consumption at Craters of the Moon Wilderness (Class I) Area by Existing (1996) and Projected Sources Subject to Prevention of Significant Deterioration Regulation

Pollutant	Averaging Period	Allowable Prevention of Significant Deterioration Increment[a] (micrograms per cubic meter)	Amount of Prevention of Significant Deterioration Increment Consumed (micrograms per cubic meter)
Nitrogen dioxide	Annual	2.5	0.06
Respirable particulates[b]	Annual	4	0.008
	24 hours	8	0.7
Sulfur dioxide	Annual	2	0.09
	24 hours	5	1.9
	3 hours	25	6.2

a. All increments specified are State of Idaho standards (ID DHW 1998).
b. Data on particulate size are not available for most sources. For purposes of comparison to the respirable particulate increments, it is conservatively assumed that all particulates emitted are of respirable size (i.e., 10 microns or less in diameter).

Note: Estimated increment consumption includes existing sources, projected increases from planned projects, including the Advanced Mixed Waste Treatment Project, and excludes the New Waste Calcining Facility.
Source: DOE 1999i.

Table 3–17 Prevention of Significant Deterioration Increment Consumption at Class II Areas by Existing (1996) and Projected Sources Subject to Prevention of Significant Deterioration Regulation at INEEL

Pollutant	Averaging Period	Allowable Prevention of Significant Deterioration Increment[a] (micrograms per cubic meter)	Amount of Prevention of Significant Deterioration Increment Consumed (micrograms per cubic meter)
Nitrogen dioxide	Annual	25	1.6
Respirable particulates[b]	Annual	17	0.92
	24 hours	30	17
Sulfur dioxide	Annual	20	2.4
	24 hours	91	31
	3 hours	512	140

a. All increments specified are State of Idaho standards (ID DHW 1998).
b. Data on particulate size are not available for most sources. For purposes of comparison to the respirable particulate increments, it is conservatively assumed that all particulates emitted are of respirable size (i.e., 10 microns or less in diameter).

Note: Estimated increment consumption includes existing sources, projected increases from planned projects, including the Advanced Mixed Waste Treatment Project, and excludes the New Waste Calcining Facility.
Source: DOE 1999i.

Routine offsite monitoring for nonradiological air pollutants is generally only performed for particulates. Monitoring for PM_{10} is performed by the Environmental Science and Research Foundation at the site boundary and at communities beyond the boundary. In 1998, 55 samples were collected at Rexburg (about 60 kilometers [19.3 miles] east of the site) by the Foundation. The mean PM_{10} concentration at Rexburg for 1998 was 27 micrograms per cubic meter. Forty-eight samples were collected at the Mountain View Middle School in Blackfoot, with a mean concentration of 23 micrograms per cubic meter. Forty-four samples were collected at Atomic City in 1998, with a mean concentration of 21 micrograms per cubic meter (Saffle et al. 2000).

Some monitoring data has also been collected by the National Park Service at the Craters of the Moon Wilderness Area. The monitoring program has shown no exceedances of the 1-hour ozone standard, low levels of sulfur dioxide (except for one exceedance of the 24-hour standard in 1985), and total suspended particulates within applicable standards (DOE 1999h). Note that the total suspended particulate standards have been replaced with PM_{10} standards.

3.3.3.2 Locations of Proposed Activities

The meteorological conditions for INEEL are considered to be representative of the INTEC and Test Reactor Area sites.

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Primary sources of nonradiological air pollutants include the New Waste Calcining Facility and coal-fired steam-generating facility. Both of these facilities are in standby. These facilities are sources of carbon monoxide, nitrogen dioxide, sulfur dioxide, and PM_{10}. The New Waste Calcining Facility is a large source of nitrogen dioxide at INEEL.

TEST REACTOR AREA

The ATR facility operates a diesel generator as a source of backup electrical power. This generator is a source of nonradioactive air emissions at ATR. Other diesel engines are also operated periodically and contribute to air emissions (LMIT 1997:11–23). The existing ambient air pollutant concentrations attributable to sources at ATR are presented in **Table 3–18**. These concentrations are estimated using SCREEN3 and are expected to overestimate the contribution to site boundary concentrations.

Table 3–18 Comparison of Modeled Ambient Air Concentrations from ATR Sources with Most Stringent Applicable Standards or Guidelines

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meters)[a]	ATR Concentration (micrograms per cubic meters)
Criteria pollutants			
Carbon monoxide	8 hours	10,000[b]	33.6
	1 hour	40,000[b]	48
Nitrogen dioxide	Annual	100[b]	9.19
Ozone	1 hour	235[c]	(d)
PM_{10}	Annual	50[b]	4.72
	24 hours	150[b]	37.7
Sulfur dioxide	Annual	80[b]	1.50
	24 hours	365[b]	12
	3 hours	1,300[b]	26.9

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The NAAQS (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Federal and state standard.
c. Federal 8-hour standard is currently under litigation.
d. Not directly emitted or monitored by the site.

Source: Modeled concentrations using SCREEN3 and emissions estimates for diesel generators.

3.3.4 Water Resources

Water resources include all forms of surface water and subsurface groundwater.

3.3.4.1 Surface Water

Surface water includes marine or freshwater bodies that occur above the ground surface, including rivers, streams, lakes, ponds, rainwater catchments, embayments, and oceans.

3.3.4.1.1 General Site Description

INEEL is in the Mud Lake-Lost River Basin (also known as the Pioneer Basin). This closed drainage basin includes three main streams—the Big and Little Lost Rivers and Birch Creek (**Figure 3–7**). These three streams are essentially intermittent and drain the mountain areas to the north and west of INEEL, although most flow is diverted for irrigation in the summer months before it reaches the site boundaries. Flow that reaches INEEL infiltrates the ground surface along the length of the stream beds, in the spreading areas at the southern end of INEEL and, if the stream flow is sufficient, in the ponding areas (playas or sinks) in the northern portion of INEEL. During dry years, there is little or no surface water flow on INEEL. Because the Mud Lake-Lost River Basin is a closed drainage basin, water does not flow off INEEL but rather infiltrates the ground surface to recharge the aquifer or is consumed by evapotranspiration. The Big Lost River flows southeast from Mackay Dam, past Arco and onto the Snake River Plain. On INEEL, near the southwestern boundary, a diversion dam prevents flooding of downstream areas during periods of heavy runoff by diverting water to a series of natural depressions or spreading areas. During periods of high flow or low irrigation demand, the Big Lost River continues northeastward past the diversion dam, passes within about 60 meters (200 feet) of INTEC and ends in a series of playas 24 to 32 kilometers (15 to 20 miles) northeast of INTEC and the Test Reactor Area, where the water infiltrates the ground surface.

Flow from Birch Creek and the Little Lost River infrequently reaches INEEL. The water in Birch Creek and Little Lost River is diverted in summer months for irrigation prior to reaching INEEL. During periods of unusually high precipitation or rapid snow melt, water from Birch Creek and Little Lost River may enter INEEL from the northwest and infiltrate the ground, recharging the underlying aquifer (DOE 1999i:4-50, 4-51). Other than the three intermittent streams, the only other surface water bodies on the site include natural wetland-like ponds and manmade percolation and evaporation ponds (DOE 1999h:4.8-1).

Big Lost River, Little Lost River, and Birch Creek have been classified by the State of Idaho for irrigation for agriculture, cold water biota development, salmon spawning, and primary and secondary recreation (DOE 1999h:4.8-9). Surface waters, however, are not used for drinking water on the site, nor is effluent discharged directly to them; thus, there are no surface water rights issues at INEEL (DOE 1996b:3-115). Although there are no routine wastewater discharges to surface waters, an NPDES permit application has been filed with EPA Region 10 for minor discharges from INTEC production wells to the Big Lost River. However, these discharges are subject to Idaho water quality standards and criteria. INEEL facilities are also covered by EPA's multisector general stormwater permit issued in 1998 (63 FR 52430). Stormwater is managed via the INEEL Storm Water Pollution Prevention Plan (first implemented in 1993). Annual stormwater evaluations are conducted as part of the plan, and stormwater is monitored in accordance with the permit and DOE Orders. In 1998, INEEL also submitted a Notice of Intent to EPA for renewal of the site's General Permit for Storm Water Discharges from Construction Sites. As for industrial activities, a pollution prevention plan covering construction activities is maintained. Application has been made to the State of Idaho for Wastewater Land Application Permits for all existing wastewater treatment facilities on the site (e.g., percolation ponds and sewage treatment irrigation systems); four permits have been issued (Saffle et al. 2000:2-6, 2-7).

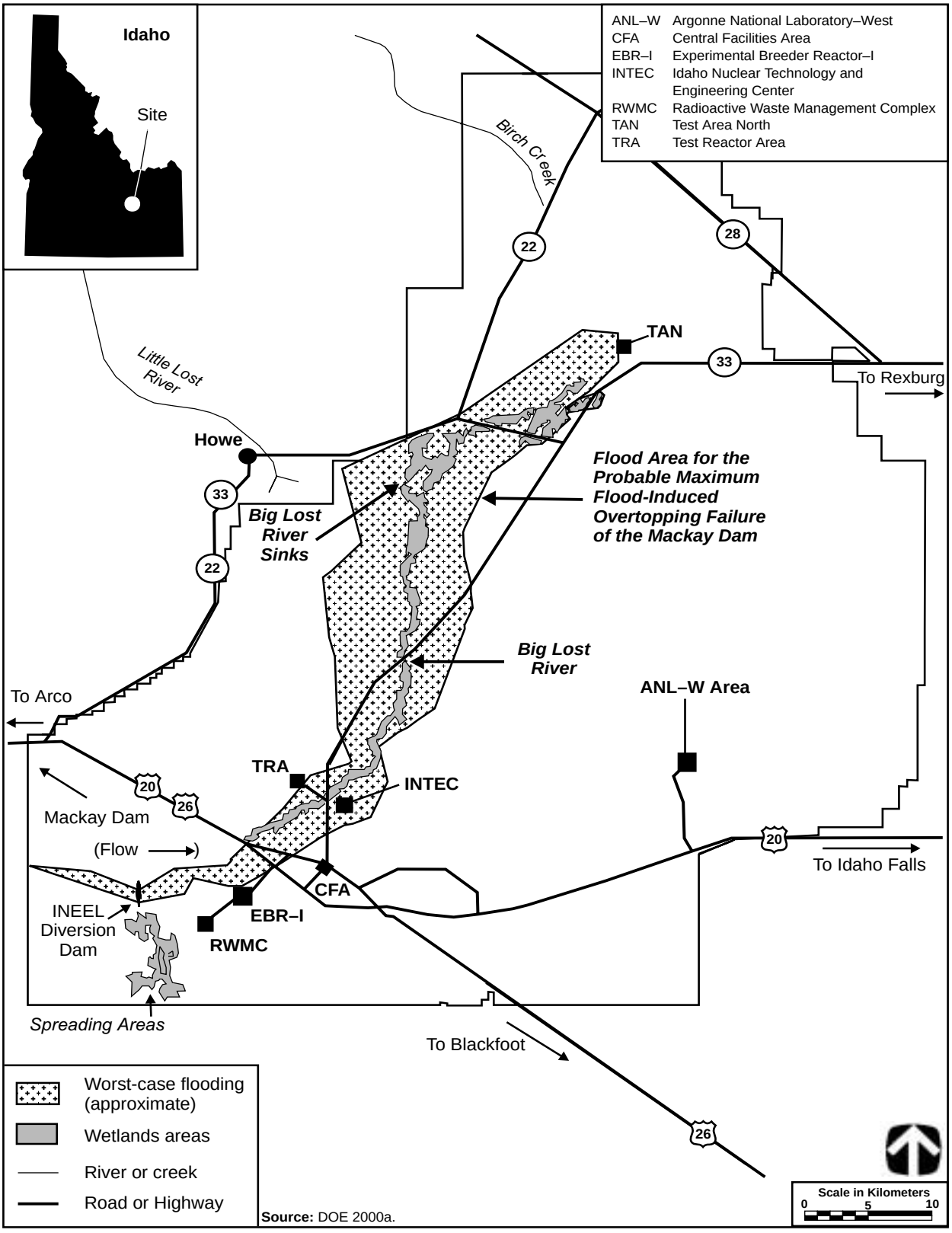

Figure 3–7 Surface Water Features at the Idaho National Engineering and Environmental Laboratory

None of the rivers on or near INEEL have been classified as a Wild and Scenic River. The INEEL diversion dam constructed in 1958 and enlarged in 1984 secured INEEL from the 300-year flood of the Big Lost River by directing flow through a diversion channel into four spreading areas (DOE 1995a:4.8-3, 4.8-4, 4.8-13; 1996b:3-115).

3.3.4.1.2 Locations of Proposed Activities

There are no named streams within INTEC and the Test Reactor Area; there are only unnamed drainage ditches that carry storm flows away from buildings and facilities at the site. Outside INTEC and the Test Reactor Area, the only surface water is a stretch of Big Lost River. As described above, this is an intermittent stream that flows past the diversion dam and across INEEL near INTEC and the Test Reactor Area mainly during wet periods such as when it carries snowmelt from the nearby mountains, and/or when upstream irrigation demand is low (Abbott, Crockett, and Moor 1997:5; DOE 1999i:4-50). The stream channel is immediately adjacent to the northwest corner of INTEC and is 1,365 meters (4,480 feet) from the southeast corner of the Test Reactor Area fenced boundary (LMIT 1997:2-47). During the period September 1995 to July 1996, flow of the Big Lost River on INEEL averaged 1.51 cubic meters (53.5 cubic feet) per second with the highest one-day flow of 10.36 cubic meters (366 cubic feet) per second (DOE 1999h:4.8-1). A summary of water quality data for Big Lost River in the vicinity of INEEL is provided in the *Storage and Disposition PEIS* and shows no unusual concentrations of the parameters analyzed (DOE 1996b:3-115–3-117). In general, the water quality of Big Lost River, Little Lost River, and Birch Creek is similar with the chemical quality reflecting the carbonate mineral composition of the mountain ranges drained by them, along with the quality of irrigation water return flows (DOE 1995a:4.8-4; 1999i:4-54).

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Sanitary waste with no potential for radioactive contamination is treated in the INTEC Sewage Treatment Facility (CPP–615) and associated treatment lagoons. This facility has a Wastewater Land Application Permit from the State of Idaho for discharge to infiltration trenches, located on the northeast corner of INTEC, after treatment and does not discharge to surface waters. The only effluent criteria associated with flows to the sewage treatment ponds are for total suspended solids and nitrogen. All compliance points for the ponds are in wells downgradient from the ponds, and the maximum allowable concentrations are similar to those in the National Primary and Secondary Drinking Water Standards (Abbott, Crockett, and Moor 1997:9, 10, 13; DOE 1999i:4-52, 4-58). In 1998, INTEC generated and disposed of about 60.4 million liters (15.95 million gallons) of sanitary wastewater (French, Tallman, and Taylor 1999a:INTEC-12).

Drainage from corridors, roof and floor drains, and condensate from process heating, and heating, ventilation, and air conditioning systems, with very low potential for radiological contamination are routed to the INTEC service waste system. This system discharges under a Wastewater Land Application Permit to two percolation ponds located on the south side of the INTEC complex. Service Waste Pond 1 has a surface area of about 18,400 square meters (198,000 square feet) and is 4.9 meters (16 feet) deep. It has a disposal capacity of 5.7 million liters (1.5 million gallons) per day. Service Waste Pond 2, immediately west of Service Waste Pond 1, has a surface area of approximately 23,100 square meters (248,700 square feet). It has a disposal capacity of 11 million liters (2.9 million gallons) per day. Both ponds are fenced to keep out wildlife (Abbott, Crockett, and Moor 1997:9). Approximately 1.96 billion liters (517 million gallons) of process wastewater was discharged to the service waste percolation ponds in 1998 (French, Tallman, and Taylor 1999a:INTEC-7, 12). Based on 1998 monitoring results from the INTEC service waste system, none of the parameter concentrations exceeded applicable standards that would define the effluent as hazardous (Saffle et al. 2000:7-5).

Consideration is being given to relocating the percolation pond to reduce the potential impacts on a contaminated perched water zone. Consideration is also being given to obtaining an NPDES permit to allow direct discharge into Big Lost River. These actions are independent of the proposed action analyzed in this NI PEIS and would be preceded by appropriate NEPA documentation (Abbott, Crockett, and Moor 1997:10). Waste management activities and facilities are discussed in greater detail under Section 3.3.11.

Flooding scenarios that involve the failure of MacKay Dam and high flows in the Big Lost River have been evaluated. The results indicate that in the event of a failure of this dam, flooding would occur at INTEC. The flood area calculated for this worse-case event is shown on Figure 3–7. The low velocity and shallow depth of the water, however, would not pose a threat of structural damage to most facilities (Barghusen and Feit 1995:2.3-21; DOE 1999i:4-51, 4-53, 4-54). Localized flooding can occur due to rapid snowmelt and frozen ground conditions, but none has been reported at INTEC (Barghusen and Feit 1995:2.3-23).

A separate flood study conducted by USGS and published in 1998 calculated that the 100-year flood would produce a flow at the Arco gauging station of about 205.6 cubic meters (7,260 cubic feet) per second, resulting in failure of the INEEL Diversion Dam and inundating the northern third of INTEC (DOE 1999i:4-54, 4-55). A 1999 Bureau of Reclamation paleoflood study confirms that while INTEC is potentially subject to flooding by the Big Lost River, it is predominantly sited on geomorphic surfaces that are well in excess of 10,000 years of age, indicating that the hazard of significant flooding is low under natural channel conditions. However, extensive modification of the Big Lost River channel throughout much of INEEL indicates that the characterization of flood stage due to Big Lost River flows will require a detailed assessment of channel stability and behavior for different flows (DOI 1999). Nevertheless, the results of the Bureau of Reclamation study indicate that neither the 100- or 500-year flood would inundate any more than the northern-most portion of INTEC. The study did not, however, consider dam failure (DOE 1999i:4-54, 4-56, 4-57). No flood maps are available from the Federal Emergency Management Agency.

TEST REACTOR AREA

Sanitary wastewater from Test Reactor Area facilities is collected by the sanitary sewer system and discharged to two sewage evaporation lagoons located just to the east of the Test Reactor Area (i.e., Test Reactor Area Sanitary Waste Ponds) (Saffle et al. 2000:7-6). In 1998, the Test Reactor Area generated and disposed of about 42.4 million liters (11.2 million gallons) of sanitary wastewater (French, Tallman, and Taylor 1999a:TRA-12).

Radiological liquid effluents at the Test Reactor Area result from canal wastewater, primary coolant leakage, and activities associated with ATR power monitoring. This process wastewater is treated by the ATR Warm Waste Treatment Facility system. The resultant wastewater, containing tritium, limited concentrations of activation, and fission products below the volatile and nonvolatile release limits established by the State of Idaho, is released to the Test Reactor Area Warm Waste Evaporation Ponds, Test Reactor Area–715. As a result, there is no direct discharge to groundwater. Nevertheless, this released wastewater is also below applicable requirements for nonradiological hazardous constituents specified in the pond operating permit (LMIT 1997:11-10, 11-11, 11-42; Moor and Peterson 1999:7). The ATR Warm Waste Treatment Facility has a design flow rate of 567.8 liters (150 gallons) per minute or about 817,646 liters (216,000 gallons) per day (LMIT 1997:11-41, 11-43). Effluent discharges to the Warm Waste Evaporation Ponds totaled approximately 15.8 million liters (4.17 million gallons) in 1998 (French, Tallman, and Taylor 1999a:TRA-8).

Nonradiological process waste effluents (primary ATR secondary cooling water) collect at the cold well sump (Test Reactor Area–703) and sampling station (Test Reactor Area–764) where they are collected continuously, sampled daily, and pumped out to the Cold Waste Pond (Test Reactor Area–702) located outside the Test Reactor Area fence. Sampling data indicate that during routine operation, the Test Reactor Area cold waste

effluent is characterized as nonhazardous industrial wastewater (LMIT 1997:11-10–11-12; Moor and Peterson 1999:7; Saffle et al. 2000:7-6, 7-7). Approximately 793.7 million liters (209.67 million gallons) of process wastewater was discharged to the Cold Waste Pond in 1998 (French, Tallman, and Taylor 1999a:TRA-5). Waste management activities and facilities are discussed in greater detail under Section 3.3.11.

Flooding scenarios that involve the failure of MacKay Dam have been evaluated and the results indicate that flood waters would not reach ATR, even if the failure was concurrent with the probable maximum flood (Figure 3–7). The effects of intense local precipitation and snowmelt runoff have also been evaluated and are not expected to result in flood damage to ATR because the reactor building main floor is at a higher elevation than its surroundings (LMIT 1997:2-47-2-51).

The 1998 USGS and 1999 Bureau of Reclamation flood studies described earlier also evaluated the potential for flooding at Test Reactor Area with the results indicating that none of the scenarios evaluated would result in inundation of the Test Reactor Area.

3.3.4.2 Groundwater

Aquifers are classified by Federal and state authorities according to use and quality. The Federal classifications include Classes I, IIA, IIB, and III groundwater. Class I groundwater is either the sole source of drinking water or is ecologically vital. Classes IIA and IIB are current or potential sources of drinking water or other beneficial use, respectively. Class III is not considered a potential source of drinking water and is of limited beneficial use.

3.3.4.2.1 General Site Description

The Snake River Plain aquifer is classified by EPA as a Class I sole source aquifer. It lies below the INEEL site and covers about 2,486,000 hectares (6,143,000 acres) in southeastern Idaho. This aquifer serves as the primary drinking water source in the Snake River Basin and is believed to contain 1.2 quadrillion to 2.5 quadrillion liters (317 trillion to 660 trillion gallons) of water. The aquifer consists of 610 to 3,048 meters (2,000 to 10,000 feet) of interbedded sediments, lava flows, and rhyolite. Recharge of the groundwater comes from Henry's Fork of the Snake River, Big Lost River, Little Lost River, and Birch Creek. Rainfall and snowmelt also contribute to the aquifer's recharge (DOE 1996b:3-115–3-117). Groundwater generally flows laterally at a rate of 1.5 to 6.1 meters (5 to 20 feet) per day. Groundwater flow is toward the south-southwest. It emerges in springs along the Snake River from Milner to Bliss, Idaho. Depth to the groundwater table ranges from about 60 meters (200 feet) below ground in the northeast corner of the site, to about 300 meters (1,000 feet) in the southeast corner (DOE 1995a:4.8-5; 1996b:3-117). Perched water tables also occur below the site. These perched water tables tend to slow the migration of pollutants that might otherwise reach the Snake River Plain aquifer (DOE 1996b:3-117). Perched water tables have been detected beneath INTEC and the Test Reactor Area mainly attributable to disposal ponds (DOE 1995a:4.8-8).

INEEL has a large network of monitoring wells that are maintained and monitored by USGS. This network includes 125 observation wells in the Snake River Plain aquifer and 45 drilled to monitor perched aquifers. An additional 120 auger holes have been drilled for monitoring shallow perched groundwater (Saffle et al. 2000:3-34, 3-35). INEEL's management and operations contractor also routinely monitors drinking water quality via 17 production wells and 10 distribution systems (Saffle et al. 2000:3-27).

Historical waste disposal practices have produced localized plumes of radiochemical and chemical constituents in the Snake River Plain Aquifer at INEEL. Of principal concern over the years has been the movements of the tritium and strontium-90 plumes. The general extent of these plumes beneath INEEL are shown in **Figure 3–8**.

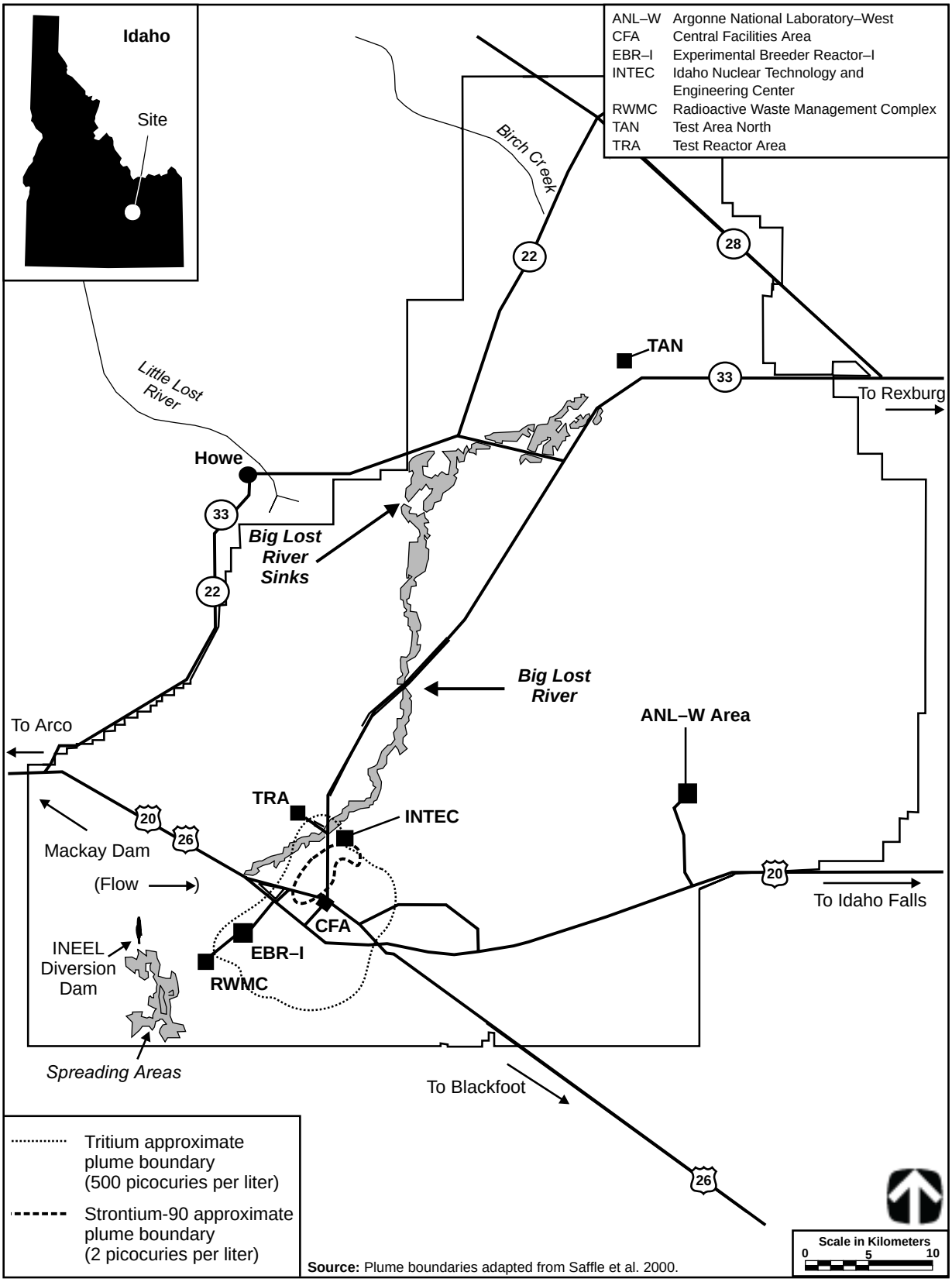

Figure 3–8 Extent of Tritium and Strontium-90 Plumes within the Snake River Plain Aquifer on the Idaho National Engineering and Environmental Laboratory (1995)

The main sources of tritium contamination of groundwater have been the injection of wastewater through the INTEC disposal well and the discharge of wastewater to the infiltration/percolation ponds at INTEC and Test Reactor Area. Since 1984, wastewater has been discharged only to the infiltration ponds, and since 1993 principally to lined evaporation ponds at the Test Reactor Area. The extent of the tritium contamination plume has remained about the same since 1991; however, concentrations in well water within the plume have decreased significantly. This is attributed to radioactive decay and a decrease in tritium disposal rates (Saffle et al. 2000:6-10, 6-12, 6-13).

The extent of the strontium-90 contaminant plume, also originating from INTEC, as well as the concentrations of strontium-90 have remained essentially constant since 1991. This is attributed to a lack of groundwater recharge from the Big Lost River that would otherwise dilute concentrations, and to the disposal of other chemicals in the INTEC infiltration ponds which may have decreased strontium-90 adsorption to soil and rock causing more to remain in the liquid phase (Saffle et al. 2000:6-13). Other known contaminants include cesium-137, iodine-129, strontium-90, and nonradioactive compounds such as trichloroethylene. Components of nonradioactive waste entered the aquifer as a result of past waste disposal practices. Elimination of groundwater injection exemplifies a change in disposal practices that has reduced the amount of these constituents in the groundwater (DOE 1996b:3-117, 3-119). Information on more recent groundwater monitoring and chemical analysis is presented in the annual site environmental report (Saffle et al. 2000).

From 1982 to 1985, INEEL used about 7.9 billion liters (2.1 billion gallons) per year from the Snake River Plain aquifer, the only source of water at INEEL. This represents less than 0.3 percent of the groundwater withdrawn from that aquifer. Since 1950, DOE has held a Federal Reserved Water Right for the INEEL site that permits a pumping capacity of approximately 2.3 cubic meters (80 cubic feet) per second, with a maximum water consumption of 43 billion liters (11.4 billion gallons) per year. Total groundwater withdrawal at INEEL historically averages between 15–20 percent of that permitted amount (DOE 1996b:3-119; Moor and Peterson 1999:6). In 1998, INEEL's production well system withdrew a total of about 4.83 billion liters (1.276 billion gallons) of water. Most of the groundwater withdrawn for use by INEEL facilities is returned to the subsurface via percolation ponds (French, Tallman, and Taylor 1999a:v).

3.3.4.2.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

Water for INTEC is supplied by two deep wells (CPP–01 and CPP–02) in the northwest corner of the area. The wells are about 180 meters (590 feet) deep and about 36 centimeters (14 inches) in diameter (Abbott, Crockett, and Moor 1997:9, 13). These wells can each supply up to approximately 11,400 liters per minute (3,000 gallons per minute) of water for use in the INTEC fire water, potable water, treated water, and demineralized water systems (Werner 1997). INTEC withdrew approximately 2.20 billion liters (581 million gallons) of groundwater in 1998 (French, Tallman, and Taylor 1999a:INTEC-12). Water use by individual facilities within INTEC (e.g., FDPF or CPP–651) is not generally metered (Folk 2000). Pumping has little effect on the level of the groundwater, because the withdrawals are small relative to the volume of water in the aquifer and the amount of recharge available. The production wells at INTEC have historically contained measurable quantities of strontium-90 (Barghusen and Feit 1995:2.3-23–2.3-29).

Water from the potable production well system at INTEC was sampled and analyzed in 1998 for lead, copper, and nitrogen as nitrate, with maximum levels measuring 0.004, 0.3, and 2.0 milligrams per liter, respectively. None of these constituents were above the EPA maximum contaminant levels or the State of Idaho drinking water limits of 0.015, 1.3, and 10 milligrams per liter, respectively. Also, routine sampling and analysis in 1998 of the potable water distribution system serving INTEC for purgeable organic compounds revealed the presence of total trihalomethanes (i.e., trichloromethane [chloroform], dibromochloromethane,

bromodichloromethane, and tribromomethane [bromoform]) and total xylenes at maximum levels of 11.3 and 0.3 micrograms per liter, respectively. However, these concentrations are well below the corresponding maximum contaminant levels for these contaminants of 100 and 10,000 micrograms per liter. Monitoring was also conducted in 1998 for radiochemical contaminants in INEEL production well and distribution systems. Of the 59 samples analyzed in 1998, seven revealed detectable gross alpha activity with the highest level (7 picocuries per liter) at INEEL from the INTEC distribution system. This is below the maximum contaminant level of 15 picocuries per liter. Tritium was also detected in several wells and distribution systems sampled in 1998. The maximum tritium concentration was 15,700 picocuries per liter (maximum contaminant level of 20,000 picocuries per liter) in the Central Facilities Area; the maximum concentration measured from an INTEC production well was 500 picocuries per liter (Saffle et al. 2000:6-10–15).

Purgeable (volatile) organics such as 1,1-dichloroethylene, toluene, and 1,1,1-trichloroethane have also historically been detected in monitoring wells within and near INTEC but at levels below maximum contaminant levels. Maximum values for tritium in samples from INTEC wells previously averaged 23,700 picocuries per liter, and maximum strontium-90 values averaged 53 picocuries per liter (Abbott, Crockett, and Moor 1997:11, 12). These values exceed the drinking water standards for tritium and strontium-90 of 20,000 picocuries per liter and 8 picocuries per liter, respectively. Selected USGS monitoring wells were sampled in 1998 for volatile organics with measurable quantities found in four wells downgradient (southwest) of INTEC. Contaminants found included 1,1,1-trichloroethane (maximum contaminant level of 200 micrograms per liter) in two wells at a maximum concentration of 0.4 micrograms per liter and dichlorodiflouromethane in four wells at a maximum concentration of 0.2 micrograms per liter (no established maximum contaminant level) (Saffle et al. 2000:6-8, 6-9). Based on the most recently published data for USGS monitoring wells, concentrations in the tritium plume originating at INTEC have continued to decrease with the concentration in well 77 south of INTEC decreasing from about 41,700 picocuries per liter in 1991 to 25,100 picocuries per liter in 1995. In contrast, strontium-90 concentrations have remained relatively constant since 1991 with concentrations between 1992 and 1995 ranging from 2.6 to 76 picocuries per liter. Nevertheless, while sampling has historically found detectable levels of strontium-90 in INTEC production wells, no strontium-90 was detected in INTEC production wells based on 1998 sampling results (Saffle et al. 2000:6-12–6-14). The general extent of the tritium and strontium-90 plumes is depicted in Figure 3–8.

TEST REACTOR AREA

All water used at Test Reactor Area is groundwater from the Snake River Plain aquifer tapped by three deep wells (TRA-01, TRA-02, and TRA-03). The depth to the groundwater at the Test Reactor Area is approximately 140 meters (460 feet). In general, Test Reactor Area, encompassing the ATR complex, uses approximately 190 million liters (50 million gallons) per month of water (Moor and Peterson 1999:6; LMIT 1997:2-59). In 1998, groundwater withdrawals from these three wells for Test Reactor Area uses totaled approximately 1.80 billion liters (475.5 million gallons) (French, Tallman, and Taylor 1999a:TRA-11). For 1999, total groundwater production was similar at about 1.78 billion liters (471 million gallons) (Perry 2000). Water use by individual facilities within the Test Reactor Area is not generally metered (Folk 2000).

As part of routine potable production well system monitoring, water from the Test Reactor Area distribution system was sampled and analyzed in 1998 for copper and nitrogen as nitrate, with concentrations measuring 1.2 and 1.1 milligrams per liter, respectively; results were below the established maximum contaminant levels (Saffle et al. 2000:6-11, 6-12). In 1998, the Test Reactor Area distribution system was also monitored for purgeable organics such as total trihalomethanes with a maximum detected concentration of 0.3 micrograms per liter, below the maximum contaminant level of 100 micrograms per liter. The tritium concentration measured in the Test Reactor Area potable water distribution system during 1998 was much lower than at INTEC and other sites with a maximum concentration of 30 picocuries per liter (maximum contaminant level of 20,000 picocuries per liter). USGS monitoring well data for tritium indicate that tritium concentrations

continue to decrease, as observed near INTEC, with the concentration in well 65 south of the Test Reactor Area decreasing from about 37,800 picocuries per liter in 1991 to 21,200 picocuries per liter in 1995 (Saffle et al. 2000:6-12, 6-15).

3.3.5 Geology and Soils

Geologic resources are consolidated or unconsolidated earth materials, including ore and aggregate materials, fossil fuels, and significant landforms. Soil resources are the loose surface materials of the earth in which plants grow, usually consisting of disintegrated rock, organic matter, and soluble salts.

3.3.5.1 General Site Description

INEEL is on the northwestern edge of the eastern Snake River Plain that is bounded on the north and south by north to northwest trending mountains and valleys of the Basin and Range physiographic province (DOE 1999h:4.6-1; LMIT 1997:2A-3). The upper 1 to 2 kilometers (0.6 to 1.2 miles) of the crust beneath INEEL is composed of a sequence of Quaternary age (recent to 2 million years old) basalt lava flows and poorly consolidated sedimentary interbeds collectively called the Snake River Group. The sediments are composed of fine-grained silts that were deposited by wind; silts, sands, and gravels deposited by streams; and clays, silts, and sands deposited in lakes. Rhyolitic (granite-like) volcanic rocks of unknown thickness lie beneath the basalt sediment sequence. The rhyolitic volcanic rocks were erupted between 4.3 and 6.5 million years ago during the upper Tertiary Period (Barghusen and Feit 1995:2.3-17; LMIT 1997:2A-15, 2A-16). The variability of the volcanic-sedimentary sequences underlying INEEL is illustrated by logs of deep drill holes completed at INEEL (**Figure 3–9**). Lava tubes, which could have similar adverse effects as karst, occur in the INEEL area (Abbott, Crockett, and Moor 1997:10).

Within INEEL, economically viable sand, gravel, pumice, silt, clay, and aggregrate resources exist. Several quarries supply these materials to various onsite construction and maintenance projects (DOE 1999h:4.6-4). Geothermal resources are potentially available in parts of the Eastern Snake River Plain, but neither of two boreholes drilled near INTEC encountered rocks with significant geothermal potential (Abbott, Crockett, and Moor 1997:12).

The Arco Segment of the Lost River Fault is thought to terminate about 7 kilometers (4.3 miles) from the INEEL boundary. The Howe Segment of the Lemhi Fault terminates near the northwest boundary of the site (LMIT 1997:2A-44, 2A-45, 2A-77) **(Figure 3–10)**. Both segments are considered capable. A capable fault is one that has had movement at or near the ground surface at least once within the past 35,000 years, or recurrent movement within the past 500,000 years (10 CFR Part 100, Appendix A).

The seismic characteristics of the Eastern Snake River Plain and the adjacent Basin and Range Province are different; the Snake River Plain has historically experienced few and small earthquakes (DOE 1999h:4.6-1). Monitoring by the INEEL seismic network has detected relatively few microearthquakes (magnitude less than 1.5) as having occurred on or near the site (Barghusen and Feit 1995:2.3-17; Jackson et al. 1993:680–695). Since 1973, there have been a total of nine small earthquakes (ranging in magnitude from 3.0 to 3.6) recorded within a radius of 90 kilometers (56 miles) of central INEEL (INTEC and Test Reactor Area), with none closer than 75 kilometers (47 miles) (USGS 2000c).

The largest historic earthquake near INEEL took place in October 1983, about 90 kilometers (56 miles) to the northwest, near Borah Peak in the Lost River Range. It occurred on the middle portion of the Lost River Fault. The earthquake had a surface-wave magnitude of 7.3 (moment magnitude of 6.9) producing peak horizontal accelerations of 0.022g to 0.078g at INEEL (DOE 1999h:E-2-1; Jackson 1985:385; USGS 2000c). The reported Modified Mercalli Intensity was VII at the event's epicenter (USGS 2000c). The Test Reactor Area

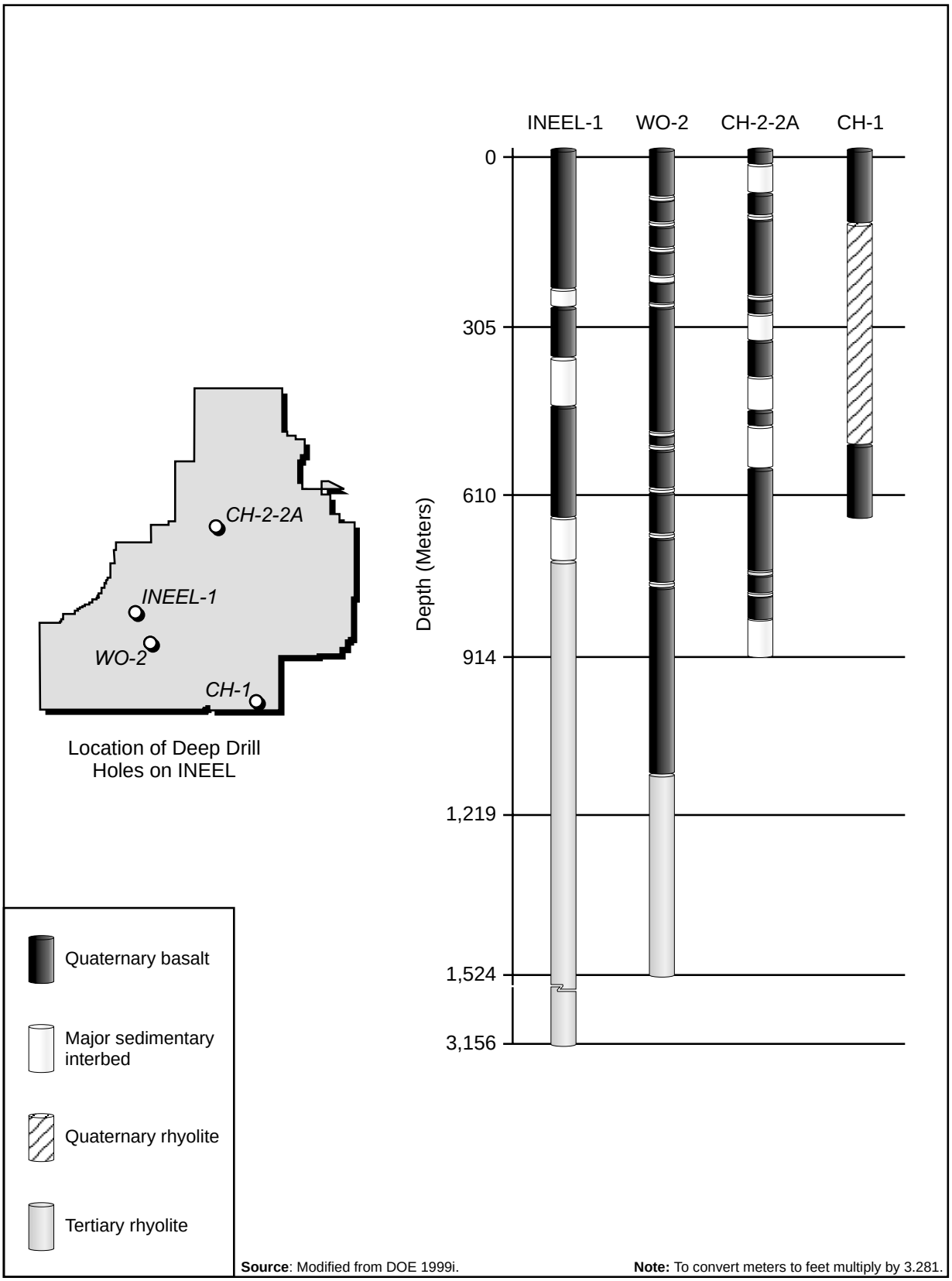

Figure 3–9 Lithologic Logs of Deep Drill Holes on Idaho National Engineering and Environmental Laboratory

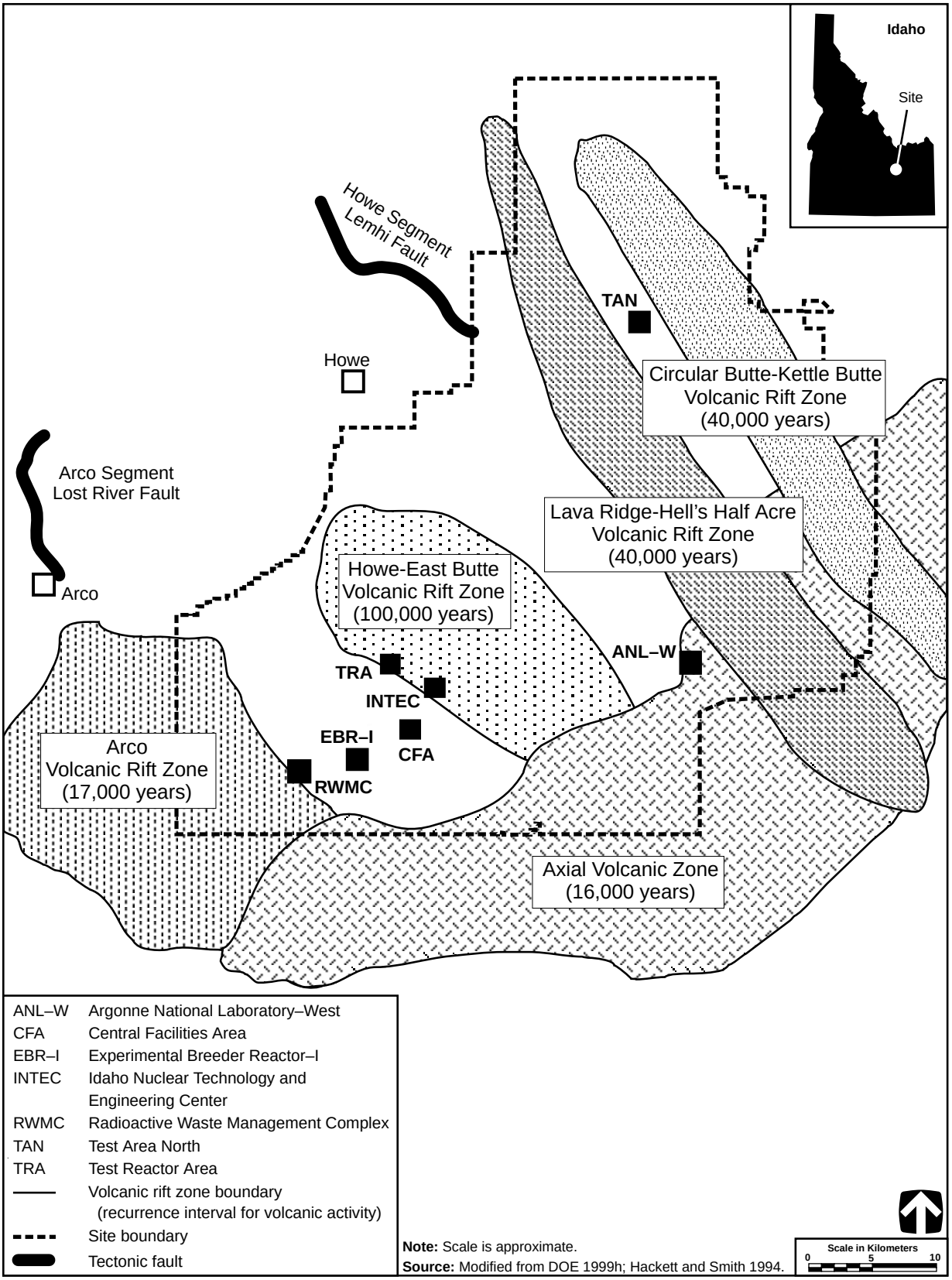

Figure 3–10 Major Geologic Features of the Idaho National Engineering and Environmental Laboratory

(i.e., ATR) experienced a Modified Mercalli Intensity of VI during this event with no damage to ATR found upon inspection (LMIT 1997:2A-29). An earthquake with a maximum horizontal acceleration of 0.15g is calculated to have an annual probability of occurrence of 1 in 5,000 at a central INEEL location (Barghusen and Feit 1995:2.3-17).

As discussed in more detail in Section 3.2.5.1, USGS has developed new seismic hazard maps as part of the National Seismic Hazard Mapping Project that are based on response spectral acceleration. These maps have been adapted for use in the new *International Building Code* (ICC 2000) (Figures 1615 (1) and 1615(2) in the code) and depict maximum considered earthquake ground motion of 0.2- and 1.0-second spectral response acceleration, respectively, based on a 2 percent probability of exceedance in 50 years. INEEL lies within the 0.35g to 0.40g mapping contours for a 0.2-second spectral response acceleration and the 0.10g to 0.15g contours for a 1.0-second spectral response acceleration.

Basaltic volcanic activity occurred from about 2,100 to 4 million years ago in the INEEL site area. Although no eruptions have occurred on the Eastern Snake River Plain during recorded history, lava flows of the Hell's Half Acre lava field erupted near the southern INEEL boundary as recently as 5,400 years ago. The most recent eruptions within the site area occurred about 2,100 years ago 30 kilometers (19 miles) southwest of the site at the Craters of the Moon Wilderness Area. Five volcanic zones have been identified on INEEL. The estimated recurrence interval for volcanism in these zones ranges from 16,000 to 100,000 years (DOE 1999h:4.6-3, 4.6-4; Hackett and Smith 1994:9, 12, 14, 20). These zones are depicted in Figure 3–10.

Four basic soilscapes exist at INEEL: river-transported sediments deposited on alluvial plains, fine-grained sediments deposited into lake or playa basins, colluvial sediments originating from bordering mountains, and wind-blown sediments over lava flows. The alluvial deposits follow the courses of the modern Big Lost River and Birch Creek. The playa soils are in the north-central part of the site. The colluvial sediments are along the western edge of INEEL. Wind-blown sediments (silt and sand) covering lava plains occupy the rest of the landscape of the site (DOE 1997b:52–54). The thickness of surfical sediments ranges from less than 0.3 meters (1 foot) at basalt outcrops east of INTEC to 95 meters (312 feet) near the Big Lost River sinks (DOE 1999h:4.6-1). No prime farmland lies within INEEL boundaries (DOE 1999e:3-71).

3.3.5.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

The nearest capable fault to INTEC is the Howe Segment of the Lemhi Fault, located about 19 kilometers (12 miles) north of the site (Abbott, Crockett, and Moor 1997:10; LMIT 1997:2A-82). Surficial geologic materials at INTEC include alluvial materials deposited by the Big Lost River. These alluvial deposits are a mixture of gravel, sand, and silt ranging in thickness from approximately 7.6 to 19.8 meters (25 to 65 feet) and locally interbedded with silt and clay deposits up to 2.9 meters (9.5 feet) thick (DOE 1999i:4-31). These surficial materials overlie the interbedded basaltic lavas of the Snake River Group. While lava tubes do occur in the INEEL area, extensive drilling in the INTEC area has not revealed any lava tubes below the site. All soil near INTEC was originally fine loam over a sand or sand-cobble mix deposited in the floodplain of the Big Lost River. However, all natural soils within INTEC fences have been disturbed. The soils beneath INTEC area are not subject to liquefaction because of the high content of gravel mixed with the alluvial sands and silts. In addition, the sediments are not saturated (Abbott, Crockett, and Moor 1997:10, 12; LMIT 1997:2A-83).

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The nearest capable fault to the Test Reactor Area is the Howe Segment of the Lemhi Fault, which is about 19 kilometers (12 miles) north-northeast of ATR (LMIT 1997:2A-82). Surficial materials within the site area, like INTEC, consist of Big Lost River alluvium comprised mostly of gravel, gravelly sands, and sands ranging from 9 to 15 meters (30 to 50 feet) in depth. A relatively thin layer of silt and clay underlies the alluvium in some locations creating a low-permeability layer at the basalt bedrock interface. These sediments overlie the interbedded basalts of the Snake River Group, with basaltic rock exposed at the surface to the north and west of the Test Reactor Area. The sedimentary interbeds of the Snake River Group consist mainly of silts, clayey silts, and sandy silts (LMIT 1997:2A-17, 2A-18). There is no potential for unstable conditions due to lava tubes at the site. Soils on the site, although highly disturbed by existing facilities, are derived from the Big Lost River alluvium. The soils and sediments are not subject to liquefaction (LMIT 1997:2A-17, 2A-83; Moor and Peterson 1999:7).

3.3.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species. Material presented in this section, unless otherwise noted, is from the *Storage and Disposition PEIS* (DOE 1996b).

3.3.6.1 Terrestrial Resources

This section addresses the plant and animal communities of INEEL and includes a plant community map of the site. Terrestrial resources are described for the site as a whole, as well as for the proposed facility locations.

3.3.6.1.1 General Site Description

INEEL lies in a cool desert ecosystem dominated by shrub-steppe communities. Most land within the site is relatively undisturbed and provides important habitat for species native to the region. Facilities and operating areas occupy 2 percent of INEEL; approximately 60 percent of the area around the periphery of the site is grazed by sheep and cattle. Although sagebrush communities occupy about 80 percent of INEEL, a total of 20 plant communities have been identified (**Figure 3–11**). In total, 398 plant taxa have been documented at INEEL.

The interspersion of low and big sagebrush communities in the northern portion of INEEL, and juniper communities in the northwestern and southeastern portions of the site are considered sensitive habitats. The former provides critical winter and spring range for sage grouse and pronghorn, while the latter is important to nesting raptors and songbirds. Riparian vegetation, primarily cottonwood and willow along the Big Lost River and Birch Creek provides nesting habitat for hawks, owls, and songbirds. Recently, approximately 29,950 hectares (74,000 acres) of open space in the north central portion of the site have been designated as the INEEL Sagebrush Steppe Ecosystem Reserve (DOE 1999g). The area represents some of the last sagebrush steppe habitat in the United States and provides habitat for numerous rare and sensitive plants and animals.

INEEL supports numerous animal species, including two amphibian, 11 reptile, 225 bird, and 44 mammal species (Reynolds 1999). Common animals on INEEL include the short-horned lizard, gopher snake, sage sparrow, Townsend's ground squirrel, and black-tailed jackrabbit. Important game animals include the sage grouse, mule deer, elk, and pronghorn. During some winters, 4,500 to 6,000 pronghorn, or about 30 percent of Idaho's total pronghorn population, may be found on INEEL. Pronghorn wintering areas are located in the northeastern portion of the site, in the area of the Big Lost River sinks, in the west-central portion of the site

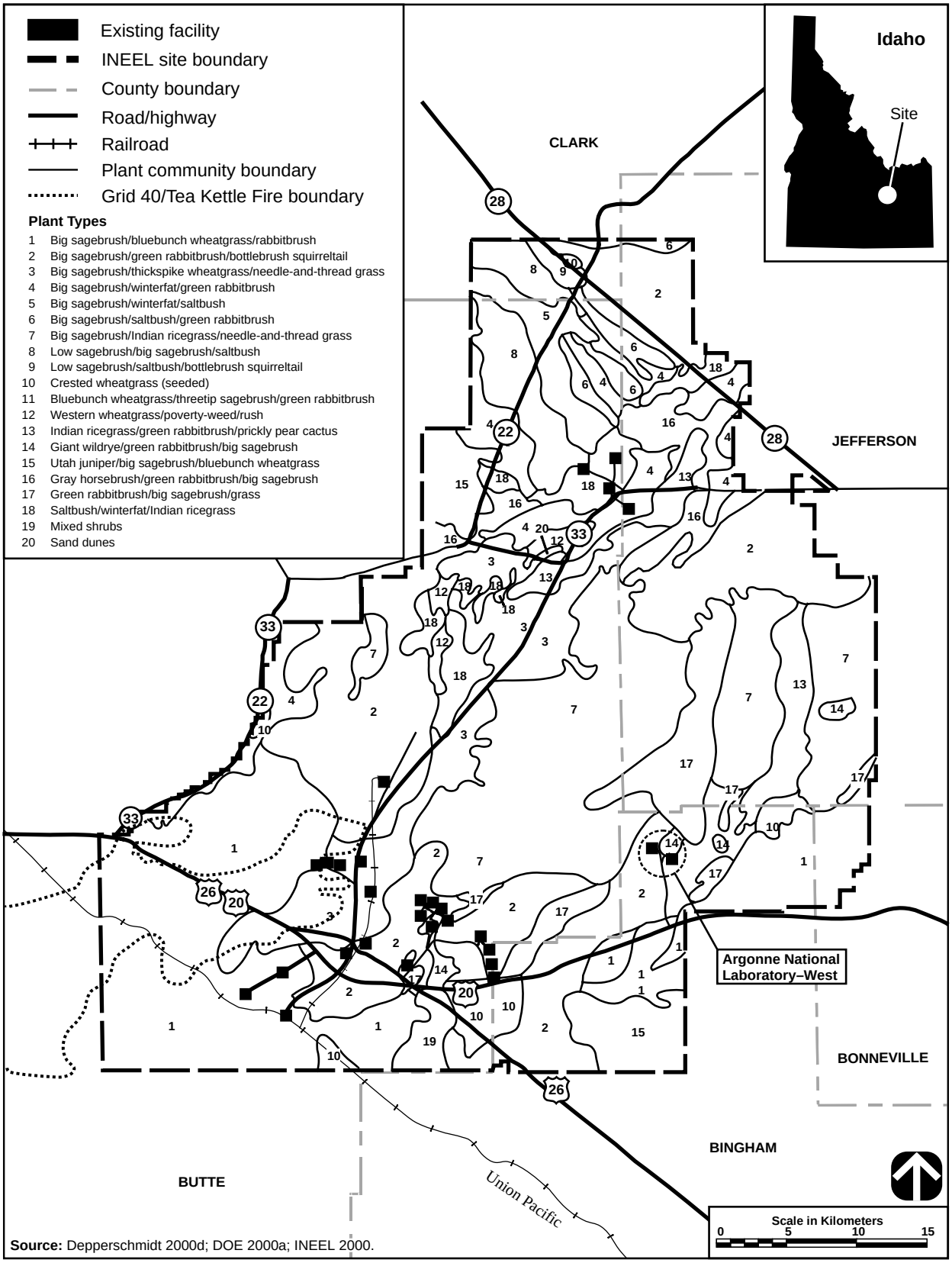

Figure 3–11 Distribution of Plant Communities at Idaho National Engineering and Environmental Laboratory

along the Big Lost River, and in the south-central portion of the site (DOE 1996b:3-125). Hunting elk and pronghorn is permitted only within 0.8 kilometers (0.5 miles) of the site boundary on INEEL lands adjacent to agricultural lands (DOE 1997b). Numerous raptors, such as the golden eagle and prairie falcon, and carnivores, such as the coyote and mountain lion, are also found on INEEL. A variety of migratory birds have been found at INEEL.

On July 27 and 28, 2000, a wildfire, known as the Grid 40/Tea Kettle Fire, burned across the southwestern portion of INEEL (Figure 3–11). The total burn area encompassed an estimated 19,830 hectares (49,000 acres) (Depperschmidt 2000a). The immediate effect of the fire on ecological resources on INEEL, aside from plants and animals that perished as a direct result of the fire, was the displacement of animals from their habitat. A longer-term concern for plant communities affected by fire and the animals that depend on them is that nonnative, invasive plant species may have a better competitive advantage at the expense of the native grasses and shrubs.

3.3.6.1.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

INTEC is within an area dominated by big sagebrush communities. The site itself is developed with little vegetation, other than that associated with landscaped areas. In fact, bare ground comprises 85 percent of the site, while facilities and pavement make up 13 percent of the area. Animal species present at INTEC are primarily limited to those adapted to disturbed industrial areas, such as mice, rabbits, sparrows, finches, and lizards (DOE 1999e:3.3.8.1.2). Wastewater ponds associated with INTEC attract a variety of wildlife (Cieminski and Flake 1995:105).

TEST REACTOR AREA

Vegetative communities in which big sagebrush is the dominant plant occur in the vicinity of the Test Reactor Area (Figure 3–11). Grasslands comprised primarily of wheat grasses also occur in the area. The Test Reactor Area itself is a developed area with little or no vegetation. Lawns and ornamental vegetation are used by a number of species such as songbirds, raptors, rabbits, and mule deer. Ponds in and around the Test Reactor Area are known to be frequented by waterfowl, shorebirds, swallow, passerines, and to a limited extent, by raptors such as the American kestrel, ferruginous hawk, and northern harrier. Mammals have been observed at the disposal ponds despite perimeter fences, and amphibians have been reported at Test Reactor Area industrial waste and sewage disposal ponds (Moor and Peterson 1999:9).

3.3.6.2 Wetlands

Wetlands include "those areas that are inundated or saturated by surface or groundwater at a frequency and duration sufficient to support, and that under normal circumstances do support, a prevalence of vegetation typically adapted for life in saturated soil conditions" (33 CFR Section 328.3). Wetlands are described for INEEL as a whole, as well as for the proposed facility locations.

3.3.6.2.1 General Site Description

National Wetland Inventory maps prepared by the U.S. Fish and Wildlife Service have been completed for most of INEEL. These maps indicate that the primary wetland areas are associated with the Big Lost River, the Big Lost River spreading areas, and the Big Lost River sinks, although smaller (less than about 0.4 hectares [1 acre]) isolated wetlands also occur. Wetlands associated with the Big Lost River are classified as riverine/intermittent, indicating a defined stream channel with flowing water during only part of the year. The

only area of jurisdictional wetland is the Big Lost River sinks (Evans et al. 1998:2-7). Wetland areas on INEEL are shown in Figure 3–11.

3.3.6.2.2 Locations of Proposed Activities

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The Big Lost River spreading areas and Big Lost River sinks are seasonal wetlands and are approximately 14 kilometers (8.7 miles) southwest, and 25 kilometers (15.5 miles) north of INTEC. These areas can provide more than 809 hectares (2,000 acres) of wetland habitat during wet years. Riparian wetland vegetation exists along the Big Lost River and along Birch Creek. Plants found along the Big Lost River, which is about 0.8 kilometer (0.5 mile) northwest of the site, are in poor condition due to recent years of only intermittent flows. There are no wetlands within the immediate INTEC area (Abbott, Crockett, and Moor 1997:15).

TEST REACTOR AREA

The Big Lost River, Big Lost River spreading areas, and the Big Lost River sinks are about 2 kilometers (1.2 miles) southeast, 13 kilometers (8 miles) southwest, and 21 kilometers (13 miles) north-northeast of the Test Reactor Area. Natural wetlands do not occur in the immediate vicinity of the Test Reactor Area.

3.3.6.3 Aquatic Resources

Aquatic resources at INEEL are described for the site as a whole, as well as for the proposed facility location.

3.3.6.3.1 General Site Description

Aquatic habitat on INEEL is limited to the Big Lost River, Little Lost River, Birch Creek, and a number of liquid waste disposal ponds. All three streams are intermittent and drain into four sinks in the north-central part of the site. Six species of fish have been observed within water bodies located on site (Reynolds 1999). Species observed in the Big Lost River include: brook trout, rainbow trout, mountain whitefish, speckled dace, shorthead sculpin, and kokanee salmon. The Little Lost River and Birch Creek, northwest and northeast of the Test Reactor Area, respectively, enter INEEL only during periods of high flow. Surveys of fish in these surface water bodies have not been conducted. The liquid waste disposal ponds on INEEL, while considered aquatic habitat, do not support fish.

3.3.6.3.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

There is no natural aquatic habitat on the INTEC site. The nearest such habitat is the Big Lost River which is about 0.8 kilometer (0.5 mile) to the northwest. Disposal ponds in the vicinity of INTEC do not support populations of fish. However, these ponds do support a variety of aquatic invertebrates (Cieminski and Flake 1995).

TEST REACTOR AREA

Although a number of disposal ponds occur in the vicinity of the Test Reactor Area, they do not support populations of fish. Aquatic invertebrates, however, are supported by habitat provided by the ponds (Moor and Peterson 1999:9). The Big Lost River is 2 kilometers (1.2 miles) southeast of the Test Reactor Area.

3.3.6.4 Threatened and Endangered Species

Endangered species are those plants and animals in danger of extinction throughout all or a large portion of their range. Threatened species are those species likely to become endangered within the foreseeable future. Threatened and endangered species are described for INEEL as a whole, as well as for the proposed facility locations.

3.3.6.4.1 General Site Description

Fifteen Federal and state-listed threatened, endangered, and other special status species occur, or possibly occur, on INEEL (see Table 4–18 of the *Idaho High-Level Waste Facilities Disposition Draft Environmental Impact Statement* [DOE 1999i). The bald eagle is listed by the U.S. Fish and Wildlife Service as threatened (but has been proposed to be delisted) and by the State of Idaho as endangered. The bald eagle has rarely been seen in the western and northern portions of INEEL. The gray wolf (listed endangered, experimental population) has been sighted several times on INEEL since 1993. On July 27 and 28, 2000, a wildland fire called the Grid 40/Tea Kettle fire burned across 19,830 hectares (49,000 acres) of the southwestern portion of INEEL. DOE is currently assessing the impacts of that fire on threatened and endangered species and species of concern (e.g., sage grouse) (Depperschmidt 2000a, 2000c). No critical habitat for threatened or endangered species, as defined in the Endangered Species Act, exists on INEEL. Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service. Consultation was also conducted with the state. The results of these consultations are presented in Chapter 4.

3.3.6.4.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

No threatened, endangered or other special status plant or wildlife species have been recorded at or within 0.5 kilometer (0.3 mile) of INTEC (Abbott, Crockett, and Moor 1997:15). The common loon, listed by Idaho as a species of special concern, was observed once at a percolation pond during a 3-year study in the early 1990s (Werner 1997:7). Other state species of special concern potentially occurring in the vicinity of the site include the black tern, loggerhead shrike, northern goshawk, trumpeter swan, pygmy rabbit, and Townsend's western big-eared bat. A complete list of threatened, endangered, or other special status species potentially occurring in areas surrounding INTEC is provided in the *Surplus Plutonium Disposition EIS* (DOE 1999e).

TEST REACTOR AREA

No threatened, endangered, or other special status plant or wildlife species have been recorded at or near the Test Reactor Area. However, one federally listed species, the bald eagle, and a number of state-listed species of special concern potentially occur in the area. State species of special concern include the northern goshawk, loggerhead shrike, black tern, trumpeter swan, pygmy rabbit, and Townsend's western big-eared bat. Of these species, only the loggerhead shrike is commonly seen in areas surrounding the Test Reactor Area (Moor and Peterson 1999:10-11).

3.3.7 Cultural and Paleontological Resources

Cultural resources are human imprints on the landscape and are defined and protected by a series of Federal laws, regulations, and guidelines. INEEL has a well-documented record of cultural and paleontological resources. Guidance for the identification, evaluation, recordation, curation, and management of these resources is included in the *Idaho National Engineering Laboratory Management Plan for Cultural Resources (Final Draft)* (Miller 1995). Past studies, which covered 4 percent of the site, identified 1,506 cultural

resource sites and isolated finds including 688 prehistoric sites, 38 historic sites, 753 prehistoric isolates, and 27 historic isolates (DOE 1996b). As of January 1998, approximately 7 percent of INEEL had been surveyed, raising the number of potential archeological sites to 1,839 (DOE 1999h). Most surveys have been conducted near major facility areas in conjunction with major modification, demolition, or abandonment of site facilities.

Cultural sites are often occupied continuously or intermittently over substantial time spans. For this reason, a single location may contain evidence of use during both historic and prehistoric periods. In the discussions that follow, the numbers of prehistoric and historic resources are presented. However, the sum of these resources may be greater than the total number of sites reported due to this dual-use history at sites. Therefore, where the total number of sites reported is less than the sum of prehistoric and historic sites, certain locations were used during both periods. DOE is currently evaluating the impacts to cultural resources from fire suppression activities during the Grid 40/Tea Kettle fire that burned across 19,830 hectares (49,000 acres) of the southwestern portion of INEEL on July 27 and 28, 2000 (Depperschmidt 2000a, 2000c).

3.3.7.1 Prehistoric Resources

Prehistoric resources are physical properties that remain from human activities that predate written records.

3.3.7.1.1 General Site Description

Prehistoric resources identified at INEEL are generally reflective of Native American hunting and gathering activities. Resources appear to be concentrated along the Big Lost River and Birch Creek, atop buttes, and within craters or caves. They include residential bases, campsites, caves, hunting blinds, rock alignments, and limited-activity locations such as lithic and ceramic scatters, hearths, and concentrations of fire-affected rock. Most sites have not been formally evaluated for nomination to the National Register of Historic Places, but are considered to be potentially eligible. Given the rather high density of prehistoric sites at INEEL, additional sites are likely to be identified as surveys continue.

3.3.7.1.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

The INTEC area has been subjected to a number of archaeological survey projects over the past two decades. Most of these investigations have been concentrated around the perimeter of the site and along existing roadways or power line corridors. Survey coverage within 1 kilometer (0.6 mile) of Building 691, located roughly in the center of INTEC, is complete. Archaeological resources identified within the surveyed area include prehistoric isolates such as camp sites and isolated artifacts reflecting Native American hunting and gathering activities. These resources are not likely to yield additional information and are, therefore, not likely to be potentially eligible for National Register nomination (Abbott, Crockett, and Moor 1997:16).

TEST REACTOR AREA

A variety of archaeological survey projects have been completed in the Test Reactor Area. During a 1984 examination of a 100-meter-wide (328-foot-wide) corridor surrounding the fenced perimeter of the Test Reactor Area, no prehistoric resources were identified. It is also unlikely that undisturbed prehistoric resources are present within the fenced perimeter of the facility, although no specific archaeological surveys have been conducted inside the fence. Although no prehistoric sites are known to occur around the periphery of the Test Reactor Area, significant sites have been documented in the vicinity, including a multi-component archaeological site, and smaller Native American campsites (Moor and Peterson 1999:12).

3.3.7.2 Historic Resources

Historic resources consist of physical properties that postdate the existence of written records. In the United States, historic resources are generally considered to be those that date no earlier than 1492.

3.3.7.2.1 General Site Description

Thirty-eight historic sites and 27 historic isolates have been identified at INEEL. These resources are representative of European-American activities, including fur trapping and trading, immigration, transportation, mining, agriculture, and homesteading, as well as more recent military and scientific/ engineering research and development activities. Examples of historic resources include Goodale's Cutoff (a spur of the Oregon Trail), remnants of homesteads and ranches, irrigation canals, and a variety of structures from the World War II era. The Experimental Breeder Reactor–I, the first reactor to achieve a self-sustaining chain reaction using plutonium instead of uranium as the principal fuel component, is listed on the National Register of Historic Places and is designated as a National Historic Landmark. Many other INEEL structures built between 1949 and 1974 are considered eligible for the National Register because of their exceptional scientific and engineering significance, and their major role in the development of nuclear science and engineering since World War II. Additional historic sites are likely to exist in unsurveyed portions of INEEL. Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office. The results of this consultation are presented in Chapter 4.

3.3.7.2.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

Two historic sites that may be eligible for nomination to the National Register of Historic Places, a homestead and nearby trash dump, have been identified near INTEC. These sites are potential sources of information on Carey Act–sponsored agricultural activities in the region. This act, which was passed in 1894, was designed to aid in the reclamation (through irrigation) and settlement of desert lands. In addition, six historic structures associated with INTEC have been identified. An historic resource inventory of all buildings within INTEC is being conducted and will likely identify additional historic structures built between 1949 and 1974 (Abbott, Crockett, and Moor 1997:16).

TEST REACTOR AREA

All three of the major reactors within the Test Reactor Area (the Materials Test Reactor, the Engineering Test Reactor, and ATR), along with numerous support facilities, are considered eligible for listing on the National Register of Historic Places. As a result of an historic building inventory conducted in 1997, 59 Test Reactor Area buildings are considered to be eligible for the National Register (Moor and Peterson 1999:13).

3.3.7.3 Native American Resources

Native American resources are sites, areas, and materials important to Native Americans for religious or heritage reasons. In addition, cultural values are placed on natural resources such as plants, which have multiple purposes within various Native American groups. Of primary concern are concepts of sacred space that create the potential for land use conflicts.

3.3.7.3.1 General Site Description

Native American resources at INEEL are associated with the two groups of nomadic hunters and gatherers that used the region at the time of European-American contact: the Shoshone and Bannock. Both of these groups used the area that now encompasses INEEL, as they harvested plant and animal resources and obsidian from Big Southern Butte and Howe Point. Because INEEL is considered part of the Shoshone-Bannock Tribes' ancestral homeland, it contains many localities that are important for traditional, cultural, educational, and religious reasons. This includes not only prehistoric archaeological sites which are important in a religious or cultural heritage context, but also includes features of the natural landscape and air, plant, water, and animal resources that have special significance. Consultation was conducted with the Shoshone and Bannock Tribes. The results of this consultation are presented in Chapter 4.

3.3.7.3.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

Although INTEC and the surrounding area may contain Native American resources (Abbott, Crockett, and Moor 1997:16), it is unlikely that undisturbed Native American resources exist within the fenced perimeter of the site.

TEST REACTOR AREA

Over the past two decades, efforts have been underway to assemble complete inventories of cultural resources in the vicinity of major operating facilities at INEEL. A variety of survey projects have been completed near the Test Reactor Area, including a 1984 examination of a 100-meters-wide (328-foot-wide) corridor surrounding the fenced perimeter of the site. No Native American resources were identified within the surveyed area, and it is unlikely that undisturbed Native American resources are present within the fenced perimeter of the Test Reactor Area, although no specific surveys have been conducted. Cultural resource surveys in the vicinity of the Test Reactor Area have identified small Native American campsites, and an area that may be of traditional and cultural importance to the Shoshone-Bannock Tribes (Moor and Peterson 1999:12).

3.3.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age. Paleontological remains consist of fossils and their associated geologic information.

3.3.7.4.1 General Site Description

The region encompassing INEEL has abundant and varied paleontological resources, including plant, vertebrate, and invertebrate remains in soils, lake and river sediments, and organic materials found in caves and archaeological sites. Vertebrate fossils recovered from the Big Lost River floodplain consist of isolated bones and teeth from large mammals of the Pleistocene or Ice Age. These fossils were discovered during excavations and well drilling operations. Fossils have been recorded in the vicinity of the Naval Reactors Facility. Occasional skeletal elements of fossil mammoth, horse, and camel have been retrieved from the Big Lost River diversion dam and Radioactive Waste Management Complex on the southwestern side of INEEL, and from river and alluvial fan gravels and Lake Terreton sediments near Test Area North (DOE 1999e). In total, 24 paleontological localities have been identified in INEEL (Miller 1995).

3.3.7.4.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

Vertebrate fossils recovered from the Big Lost River floodplain consist of isolated bones or teeth from large mammals of the Pleistocene or Ice Age. These fossils were discovered during excavations and well-drilling operations. A single mammoth tooth was salvaged during the excavation of a percolation pond located to the south of INTEC (Abbott, Crockett, and Moor 1997:16).

TEST REACTOR AREA

A mammoth tooth dating from the late Pleistocene has been recovered from the Test Reactor Area (Miller 1995:J-15).

3.3.8 Socioeconomics

Statistics for employment and regional economy are presented for the regional economic area, as defined in Appendix G.8, which encompasses 13 counties around INEEL located in Idaho and Wyoming. Statistics for population, housing, community services, and local transportation are presented for the region of influence, a four-county area in Idaho in which 94.4 percent of all INEEL employees reside, as shown in **Table 3–19**. In 1997, INEEL employed 8,291 persons (about 5.5 percent of the regional economic area civilian labor force) (DOE 1999e).

Table 3–19 Distribution of Employees by Place of Residence in the INEEL Region of Influence, 1997

County	Number of Employees	Total Site Employment (percent)
Bonneville	5,553	67.0
Bingham	1,077	13.0
Bannock	615	7.4
Jefferson	583	7.0
Region of influence total	7,828	94.4

Source: DOE 1999e.

3.3.8.1 Regional Economic Characteristics

Between 1990 and 1996, the civilian labor force in the regional economic area increased 26 percent, to the 1996 level of 150,835. In 1996, the annual unemployment average in the regional economic area was 4.8 percent, which was slightly less than the annual unemployment average for Idaho (5.2 percent) and Wyoming (5.0 percent) (DOE 1999e).

In 1995, service activities represented the largest sector of employment in the regional economic area (27.1 percent). This was followed by retail trade (20.4 percent), and government (19.5 percent). The totals for these employment sectors in Idaho were 21.5 percent, 19.6 percent, and 18.7 percent, respectively. The totals for these employment sectors in Wyoming were 21.1 percent, 20.8 percent, and 25 percent, respectively (DOE 1999e).

3.3.8.2 Population and Housing

In 1996, the region of influence population totaled 213,547. Between 1990 and 1996, the region of influence population increased by 10.6 percent, compared to a 17.5 percent increase in Idaho's population. Between

1980 and 1990, the number of housing units in the region of influence increased by 6.7 percent, compared to a 10.2 percent increase in Idaho. The total number of housing units in the region of influence in 1990 was 69,760. The 1990 region of influence homeowner vacancy rate was 2.1 percent, compared to Idaho's rate of 2 percent. The region of influence rental vacancy rate was 8.3 percent compared to Idaho's rate of 7.3 percent (DOE 1999e).

3.3.8.3 Community Services

3.3.8.3.1 Education

In 1997, thirteen school districts providing public education in the INEEL region of influence were operating at capacities between 50 percent and 100 percent. Total student enrollment in the region of influence in 1997 was 50,168, and the student-to-teacher ratio in the region of influence averaged 18.8:1. In 1990, Idaho's average student-to-teacher ratio was 12.8:1 (DOE 1999e).

3.3.8.3.2 Public Safety

In 1997, a total of 475 sworn police officers served the four-county region of influence. In 1997, the average region of influence officer-to-population ratio was 2.2 officers per 1,000 persons, compared to the 1990 state average of 1.5 officers per 1,000 persons. In 1997, 560 paid and volunteer firefighters provided fire protection services in the INEEL region of influence. The region of influence firefighter-to-population ratio in 1997 was 2.6 firefighters per 1,000 persons, compared to the 1990 state average of 1.2 firefighters per 1,000 persons (DOE 1999e).

3.3.8.3.3 Health Care

In 1996, a total of 329 physicians served the region of influence. The average region of influence physician-to-population ratio was 1.5 physicians per 1,000 persons, compared to the 1996 state average of 1.7 physicians per 1,000 persons. In 1997, there were five hospitals serving the four-county region of influence. The hospital bed-to-population ratio was 4.6 hospital beds per 1,000 persons, compared to the 1990 state average of 3.3 hospital beds per 1,000 persons (DOE 1999e).

3.3.8.4 Local Transportation

Vehicular access to INEEL is provided by U.S. Routes 20 and 26 to the south, and State Routes 22 and 33 to the north. U.S. Routes 20 and 26, and State Routes 22 and 33 all share rights-of-way west of INEEL (Figure 3–6).

There are two road segments that could be affected by the alternatives considered in this NI PEIS; U.S. Route 20 from U.S. Routes 26 and 91 at Idaho Falls to U.S. Route 26 East, and U.S. Routes 20 and 26 from U.S. Route 26 East to State Routes 22 and 33.

There are at least 10 pending projects to be completed by the Idaho Transportation Department that could impact access into INEEL as well as within the INEEL site. The type of work includes laying new base, widening and rehabilitation, and pavement rehabilitation, which are all very extensive. Some projects only include a new seal coat, which can be completed in 2 weeks. The projects include: (1) a base and resurfacing and minor widening project scheduled for late in the summer of 2000 for U.S. Route 20 from Brunt Road to Cinder Butte Road; (2) a minor widening and pavement rehabilitation for State Highway 22 from the Butte County line to the Clark County line scheduled for October 2000; (3) a minor widening and rehabilitation and base resurfacing project for State Highway 33 on the INEEL site at the Test Reactor Area NE in Jefferson

County scheduled for October 2000; (4) a minor widening and restoration and paving rehabilitation project for State Highway 22 on the INEEL site from the junction of State Highway 33 to the Clark County line scheduled for July 2001; (5) a new seal coat will be placed on State Highway 33 at Terreton East and West scheduled after July 2000; (6) a new seal coat for U.S. Route 20 leading into the INEEL site at Arco East scheduled for October 2000; (7) a paving rehabilitation and restoration and new base and resurfacing for State Highway 22 located off site from the Jefferson County line to Mile Post 52.24 scheduled in July 2002; (8) a new seal coat for U.S. Route 20 leading into the site just above the Bonneville County line scheduled for October 2002; (9) a new seal coat for State Highway 22 leading into the site from the junction of State Highway 28 to Medicine Lodge scheduled for July 2003; (10) a new seal coat for State Highway 33 on the INEEL site from Mile Post 38.5 to the junction of State Highway 28 in Jefferson County scheduled for 2003 (Cole 2000).

DOE buses provide transportation between INEEL facilities and Idaho Falls for DOE and contractor personnel. The major railroad in the region of influence is the Union Pacific Railroad. The railroad's Blackfoot-to-Arco Branch provides rail service to the southern portion of INEEL. A DOE-owned spur connects the Union Pacific Railroad to INEEL by a junction at Scoville Siding. There are no navigable waterways within the region of influence capable of accommodating waterborne transportation of material shipments to INEEL. Fanning Field in Idaho Falls and Pocatello Municipal Airport in Pocatello provide jet air passenger and cargo service for both national and local carriers. Numerous smaller private airports are located throughout the region of influence (DOE 1999e).

3.3.9 Existing Human Health Risk

Existing human health risk issues include the determination of potentially adverse effects on human health that result from acute and chronic exposures to ionizing radiation and hazardous chemicals.

3.3.9.1 Radiation Exposure and Risk

3.3.9.1.1 General Site Description

Major sources and levels of background radiation exposure to individuals in the vicinity of INEEL are shown in **Table 3–20**. Annual background radiation doses to individuals are expected to remain constant over time. The total dose to the population, in terms of person-rem, changes as the population size changes. Background radiation doses are unrelated to INEEL operations.

Releases of radionuclides to the environment from INEEL operations provide another source of radiation exposure to individuals in the vicinity of INEEL. Types and quantities of radionuclides released from INEEL operations in 1998 are listed in the *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 1998* (Saffle et al. 2000:8-4, 8-8). The doses to the public resulting from these releases are presented in **Table 3–21**. These doses fall within radiological limits per DOE Order 5400.5, and are much lower than those of background radiation.

Using a risk estimator of 500 cancer deaths per 1 million person-rem to the public (Appendix H), the risk of a latent cancer fatality to the maximally exposed member of the public due to radiological releases from INEEL operations in 1998 is estimated to be 4×10^{-9}. That is, the estimated probability of this person dying of cancer at some point in the future from radiation exposure associated with 1 year of INEEL operations is less than 1 in 200 million. It takes several to many years from the time of radiation exposure for a cancer to manifest itself.

Table 3–20 Sources of Radiation Exposure to Individuals in the INEEL Vicinity Unrelated to INEEL Operations

Source	Effective Dose Equivalent (millirem per year)
Natural background radiation[a]	
Cosmic radiation	48
External terrestrial radiation	71
Internal radiation	40
Radon in homes (inhaled)	200
Other background radiation[b]	
Diagnostic x-rays and nuclear medicine	53
Weapons test fallout	Less than 1
Air travel	1
Consumer and industrial products	10
Total	424

a. Saffle et al. 2000.
b. NCRP 1987:11, 40, 53.

Note: Value of radon is an average for the United States.

Table 3–21 Radiation Doses to the Public from INEEL Normal Operations in 1998 (Total Effective Dose Equivalent)

Members of the Public	Atmospheric Releases		Liquid Releases		Total	
	Standard[a]	Actual	Standard[a]	Actual	Standard[a]	Actual
Maximally exposed individual (millirem)	10	0.008	4	0	100	0.008
Population within 80 kilometers (person-rem)[b]	None	0.075	None	0	100	0.075
Average individual within 80 kilometers (millirem)[c]	None	6.2×10^{-4}	None	0	None	6.2×10^{-4}

a. The standards for individuals are given in DOE Order 5400.5. As discussed in that order, the 10-millirem-per-year limit from airborne emissions is required by the Clean Air Act, and the 4-millirem-per-year limit is required by the Safe Drinking Water Act; for this NI PEIS, the 4-millirem-per year value is conservatively assumed to be the limit for the sum of doses from all liquid pathways. The total dose of 100 millirem per year is the limit from all pathways combined. The 100-person-rem value for the population is given in proposed 10 CFR Part 834, as published in 58 FR 16268. If the potential total dose exceeds the 100-person-rem value, it is required that the contractor operating the facility notify DOE.
b. Based on a population of about 121,500 in 1998.
c. Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site.

Source: Saffle et al. 2000.

According to the same risk estimator, 3.8×10^{-5} excess cancer fatality is projected in the population living within 80 kilometers (50 miles) of INEEL from normal operations in 1998. To place this number in perspective, it may be compared with the number of cancer fatalities expected in the same population from all causes. The 1997 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year (Famighetti 1998:964). Based on this mortality rate, the number of cancer fatalities expected during 1998 from all causes in the population living within 80 kilometers (50 miles) of INEEL was 243. This expected number of cancer fatalities is much higher than the 3.8×10^{-5} cancer fatality estimated from INEEL operations in 1998.

INEEL workers receive the same doses as the general public from background radiation, but they also receive an additional dose from working in facilities with nuclear materials. The average dose to the individual worker and the cumulative dose to all workers at INEEL from operations in 1998 are presented in **Table 3–22**. These doses fall within the radiological regulatory limits of 10 CFR Section 835.202. According to a risk estimator

Table 3–22 Radiation Doses to Workers from INEEL Normal Operations in 1998 (Total Effective Dose Equivalent)

Occupational Personnel	Onsite Releases and Direct Radiation	
	Standard[a]	Actual
Average radiation worker (millirem)	None[b]	87[c]
Total workers (person-rem)[d]	None	65[c]

a. The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. It has therefore established the Administrative Control Level of 2,000 millirem per year; the site must make reasonable attempts to maintain individual worker doses below this level.
b. No standard is specified for an "average radiation worker"; however, the maximum dose that this worker may receive is limited to that given in footnote "a."
c. Does not include doses received at the Naval Reactors Facility. The impacts associated with this facility fall under the jurisdiction of the Navy as part of the Nuclear Propulsion Program.
d. Based on a worker population of 743 with measurable doses in 1998.

Source: 10 CFR Section 835.202; DOE 1999p.

of 400 cancer fatalities per 1 million person-rem among workers (Appendix H), the number of projected latent cancer fatalities among INEEL workers from normal operations in 1998 is 0.026.

A more detailed presentation on the radiation environment, including background exposures and radiological releases and doses, is presented in the *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 1998* (Saffle et al. 2000). The concentrations of radioactivity in various environmental media (including air, water, and soil) in the site region (on and off site) are also presented in that report.

3.3.9.1.2 Locations of Proposed Activities

IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER

External radiation doses and concentrations of gross alpha and beta in air have been measured in the vicinity of INTEC. In 1998, the annual average dose within INTEC was about 171 millirem. This is about 40 millirem higher than the average dose measured at the offsite control locations. This onsite dose would affect workers only and is well below annual worker dose limits as identified in Table 3–22. Concentrations in air of gross alpha and beta were 1.1×10^{-15} microcuries per milliliter and 2.4×10^{-14} microcuries per millilter, respectively (Saffle et al. 2000:4-4, 4-5, 4-15, 4-20).

TEST REACTOR AREA

Radiological health effects resulting from the release of radionuclides from the stack that serves ATR and the ATR Critical Facility are shown in **Table 3–23**. Estimates shown in the table are based on 1999 release data discussed in Appendix H. Doses listed in Table 3–23 show that the risk of a latent cancer fatality to the maximally exposed member of the public due to emissions from ATR and the ATR Critical Facility in 1999 would be 6.5×10^{-10}. The risk of an excess cancer fatality among the public residing within 80 kilometers (50 miles) of ATR would be 6.5×10^{-6}.

Table 3–24 lists radiation doses to average Test Reactor Area and ATR workers in 1998. The average risk of an excess cancer fatality among workers at the Test Reactor Area and ATR due to onsite releases and direct radiation in 1999 would be 4.2×10^{-5} and 5.0×10^{-5}, respectively.

Table 3–23 Radiation Doses to the Public from Normal Operations at ATR and ATR Critical Facility in 1999 (Total Effective Dose Equivalent)

Members of the Public	Atmospheric Releases		Liquid Releases		Total	
	Standard	Actual	Standard	Actual	Standard	Actual
Maximally exposed individual (millirem)	10	1.3×10^{-3}	4	0	100	1.3×10^{-3}
Population within 80 kilometers (person-rem)	None	0.013	None	0	100	0.013
Average individual within 80 kilometers (millirem)	None	7.9×10^{-4}	None	0.00	None	7.9×10^{-4}

Table 3–24 Radiation Doses to Workers from the Test Reactor Area and ATR Normal Operations in 1998 (Total Effective Dose Equivalent)

Occupational Personnel	Onsite Releases and Direct Radiation	
	Standard[a]	Actual
Average Test Reactor Area worker (millirem)	None[b]	105
Average ATR worker (millirem)	None[b]	126

a. The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. It has therefore established the Administrative Control Level of 2,000 millirem per year; the site must make reasonable attempts to maintain individual worker doses below this level.

b. No standard is specified for an "average radiation worker"; however, the maximum dose that this worker may receive is limited to that given in footnote "a."

Source: DOE 1999p.

3.3.9.2 Chemical Environment

The background chemical environment important to human health consists of the atmosphere, which may contain hazardous chemicals that can be inhaled; drinking water, which may contain hazardous chemicals that can be ingested; and other environmental media through which people may come in contact with hazardous chemicals (e.g., surface water during swimming, soil through direct contact, or food). Hazardous chemicals can cause cancer and other adverse health effects.

Carcinogenic Effects. Health effects in this case are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogen. This could be incremental or excess individual lifetime cancer risk.

Noncarginogenic Effects. Health effects in this case are determined by the ratio between the calculated, or measured concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the Hazard Quotient. Hazard Quotients for noncarcinogens are summed to obtain the Hazard Index. If the Hazard Index is less than 1, no adverse health effects would be expected.

Effective administrative and design controls that decrease hazardous chemical releases to the environment and help achieve compliance with permit requirements (e.g., air emissions and NPDES permit requirements) contribute to minimizing health impacts on the public. The effectiveness of these controls is verified through the use of monitoring information and inspection of mitigation measures. Health impacts on the public may occur through inhaling air containing hazardous chemicals released to the atmosphere during normal INEEL operations. Risks to public health from other possible pathways, such as ingestion of contaminated drinking water or direct exposure, are lower than those via the inhalation pathway. At INEEL, the risk to public health from water ingestion and direct exposure pathways is low because surface water is not used for drinking or as a receptor for wastewater discharges.

Baseline air emission concentrations and applicable standards for hazardous chemicals are addressed in Section 3.3.3.

The baseline concentrations are estimates of the highest existing offsite concentrations and represent the highest concentrations to which members of the public could be exposed. These concentrations are in compliance with applicable guidelines and regulations. Information on estimating the health impacts of hazardous chemicals is presented in Appendix H.

Exposure pathways to INEEL workers during normal operations may include inhaling contaminants in the workplace atmosphere and direct contact with hazardous materials. The potential for health impacts varies among facilities and workers, and available information is insufficient for a meaningful estimate of impacts. However, workers are protected from workplace hazards through appropriate training, protective equipment, monitoring, substitution, and engineering and management controls. INEEL workers are also protected by adherence to OSHA and EPA standards that limit workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Appropriate monitoring that reflects the frequency and amounts of chemicals used in the operational processes ensures that these standards are not exceeded. Additionally, DOE requires that conditions in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm.

3.3.9.3 Health Effects Studies

Epidemiological studies were conducted on communities surrounding INEEL to determine whether there are excess cancers in the general population. Two of these are described in more detail in Appendix M.4.4 of the *Storage and Disposition PEIS* (DOE 1996b:M-233, M-234). No excess cancer mortality was reported, and although excess cancer incidence was observed, no association thereof with INEEL was established. A study by the State of Idaho completed in June 1996, found excess brain cancer incidence in the six counties surrounding INEEL, but a follow-up survey concluded that "No common factors were identified that clearly linked the cases, and individuals expressed varying concerns about possible exposures or causes for brain cancer" (ID DHW 1997).

No occupational epidemiological studies have been completed at INEEL to date, but several worker health studies have been initiated recently at INEEL. Researchers from the Boston University School of Public Health, in cooperation with the National Institute of Occupational Safety and Health, are investigating the effects of workforce restructuring (downsizing) in the nuclear weapons industry. The health of displaced workers will be studied. Under a National Institute of Occupational Safety and Health cooperative agreement, the epidemiologic evaluation of childhood leukemia and paternal exposure to ionizing radiation now includes INEEL as well as other DOE sites. Another study, begun in October 1997, *Medical Surveillance for Former Workers at INEEL*, is being carried out by a group of investigators from the Oil, Chemical, and Atomic Workers International Union, Mt. Sinai School of Medicine, the University of Massachusetts at Lowell, and the Alice Hamilton College. DOE has implemented an epidemiologic surveillance program to monitor the health of current INEEL workers. A discussion of this program is given in Appendix M.4.4 of the *Storage and Disposition PEIS* (DOE 1996b:M-233, M-234).

3.3.9.4 Accident History

DOE conducted a study, the Idaho National Engineering Laboratory Historical Dose Evaluation, to estimate the potential offsite radiation doses for the entire operating history of INEEL (DOE 1996b:3-139). Releases resulted from a variety of tests and experiments as well as a few accidents at INEEL. The study concluded that these releases contributed to the total radiation dose during test programs of the 1950s and early 1960s. The frequency and size of releases have declined since that time. There have been no serious unplanned or

accidental releases of radioactive or other hazardous substances at INEEL facilities in the 10 years of operation prior to 1998. However, in July 1998, an incident occurred at INEEL's Test Reactor Area. One fatality and several injuries resulted from an accidental release of fire retardant carbon dioxide during routine maintenance operation. No nuclear materials were involved and there was no threat to public safety (DOE 1998a, 1998b, 1998c, 1998d).

3.3.9.5 Emergency Preparedness

Each DOE site has established an emergency management program that would be activated in the event of an accident. This program has been developed and maintained to ensure adequate response to most accident conditions and to provide response efforts for accidents not specifically considered. The emergency management program includes emergency planning, preparedness, and response.

Government agencies whose plans are interrelated with the INEEL emergency plan for action include the State of Idaho, Bingham County, Bonneville County, Butte County, Clark County, Jefferson County, the Bureau of Indian Affairs, and the Fort Hall Indian Reservation. INEEL contractors are responsible for responding to emergencies at their facilities. Specifically, the emergency action director is responsible for recognition, classification, notification, and protective action recommendations. At INEEL, emergency preparedness resources include fire protection from onsite and offsite locations and radiological and hazardous chemical material response. Emergency response facilities include an emergency control center at each facility, the INEEL Warning Communication Center, and the INEEL site Emergency Operations Center. Seven INEEL medical facilities are also available to provide routine and emergency service.

DOE has specified actions to be taken at all DOE sites, implementing lessons learned from the emergency response to an accidental explosion at Hanford in May 1997.

3.3.10 Environmental Justice

Under Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, Federal agencies are responsible for identifying and addressing the possibility of disproportionately high and adverse health, economic, and environmental impacts of programs and activities on minority and low-income populations in potentially affected areas. Minority populations refer to persons of any race self-designated as Asian, Black, Native American, or Hispanic. Low-income populations refer to households with incomes below the Federal poverty thresholds. In the case of INEEL, the potentially affected area includes only parts of central Idaho.

The potentially affected area surrounding ATR is defined by a circle with an 80-kilometer (50-mile) radius centered at the Test Reactor Area (latitude 43° 35'8" N, longitude 112° 57'47" W). The total population residing within that area in 1990 was 105,939. Minorities made up 10.1 percent of the total population. In 1990, approximately one-fourth of the total population was comprised of persons self-designated as members of a minority group; minorities made up 7.8 percent of the State of Idaho's total population.

At the time of the 1990 census, Hispanics and Native Americans were the largest minority groups within the potentially affected area, accounting for 6.2 percent and 2.7 percent of the total population, respectively. Asians constituted about 1.1 percent, and Blacks, about 0.3 percent (DOC 1992).

In 1990, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 13,188 persons (12.6 percent of the total population) residing within the potentially affected area around ATR reported incomes below that threshold (DOC 1992). Data obtained during the 1990 census also

show that of the total population of the contiguous United States, 13.1 percent reported incomes below the poverty threshold. The corresponding percentage for Idaho was 13.3 percent.

The potentially affected area surrounding FDPF is defined by an 80-kilometer (50-mile) circle centered at INTEC (latitude 43° 34′12.5″ N, longitude 112° 55′55.4″ W). The total population residing within the potentially affected area in 1990 was 121,472. Approximately 10 percent of the total population was comprised of persons self-designated as members of a minority. Data from the1990 census show that minorities represented approximately 24 percent of the national population, and approximately 8 percent of the population of the State of Idaho. Hispanics (6.2 percent) and Native Americans (2.7 percent) were the largest minority groups within the population at risk. Asians (1.1 percent) and Blacks (0.3 percent) made up the remainder of the minority population at risk. Approximately 12 percent of the population at risk reported incomes below the poverty threshold of 1990.

3.3.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing DOE activities. The waste is managed using appropriate treatment, storage, and disposal technologies, and in compliance with all applicable Federal and state statutes and DOE orders.

3.3.11.1 Waste Inventories and Activities

INEEL manages the following types of waste: high-level radioactive, transuranic, mixed transuranic, low-level radioactive, mixed low-level radioactive, hazardous, and nonhazardous. Waste generation rates and the inventory of stored waste from activities at INEEL are provided in **Table 3–25**.

Table 3–25 Waste Generation Rates and Inventories at INEEL

Waste Type	Generation Rate (cubic meters per year)	Inventory (cubic meters)
High-level radioactive	0[a]	4,200[b]
Transuranic and mixed transuranic	0[a, c]	65,000[d, e]
Sodium-bearing waste	0[a]	5,300[b]
Low-level radioactive	6,400[f]	6,000[g]
Mixed low-level radioactive[h]	230	1,700
Hazardous	835[c, i]	NA[j]
Nonhazardous		
Liquid	2,000,000[c, k]	NA[j]
Solid	62,000[c]	NA[j]

a. Refer to the text.
b. DOE 1999i. The sodium-bearing waste is managed by the high-level radioactive waste program.
c. Moor and Peterson 1999:chap. 3.
d. Includes both alpha low-level radioactive waste and transuranic waste.
e. DOE 1995a:4.14-2.
f. Willson 1998:2-9 and 2-10.
g. Bright 1999.
h. DOE 1998e:4-5.
i. Includes 760 cubic meters that is recyclable.
j. Generally, hazardous and nonhazardous wastes are not held in long-term storage.
k. Projected annual average generation amounts for 1997–2006.

Note: To convert from cubic meters to cubic yards, multiply by 1.308.
Key: NA, not applicable.

Waste generation rates specifically for ATR and FDPF activities are provided in **Table 3–26**. The INEEL waste management capabilities are summarized in **Table 3–27**. More detailed descriptions of the waste management system capabilities at INEEL are included in the *Storage and Disposition PEIS* (DOE 1996b:3-141–145, E-33–E-48) and the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995a:2.2-30).

Table 3–26 Waste Generation Rates at ATR and FDPF

Waste Type	ATR[a] (cubic meters per year)	FDPF (cubic meters per year)
High-level radioactive	0	0
Transuranic	0	0
Low-level radioactive	404	0
Mixed low-level radioactive	<1	0
Hazardous	190	0
Nonhazardous		
Process wastewater	794,000	0
Sanitary wastewater	42,000	0
Solid	4,208	0

a. The data includes all facilities within the Test Reactor Area, which includes ATR.
Note: To convert from cubic meters to cubic yards, multiply by 1.308.
Source: Depperschmidt 2000b; French, Tallman, and Taylor 1999a, 1999b.

Table 3–27 Waste Management Capabilities at INEEL

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed LLW	Haz	Non-Haz
Treatment Facility (cubic meters per year except as otherwise specified)									
INTEC HEPA Filter Leach, cubic meters per day	0.21	Online			X		X		
INTEC Debris Treatment and Containment, cubic meters per day	88	Waiting on Part B Permit			X		X		
Advanced Mixed Waste Treatment Project	6,500	Planned for 2003			X		X		
INTEC NWCF	248	Standby[a]	X		X				
ANL–W Remote Treatment Facility	42	Planned for 2000		X	X	X	X		
ANL–W HFEF Waste Characterization Area	37	Online		X	X				
INTEC Waste Immobilization Facility	48	Planned for 2008	X						
INTEC Liquid Effluent Treatment and Disposal Facility	11,365	Online					X		
INTEC High-Level Radioactive Waste Evaporator	6,138	Online	X		X		X		
INTEC Process Equipment Waste Evaporator	13,000	Online			X	X	X		
ANL–W Sodium Processing Facility	698	Online					X		

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed LLW	Haz	Non-Haz
Test Area North Cask Dismantlement	11	Online					X		
Test Reactor Area Evaporation Pond, cubic meters per day	820	Online				X			
WROC - Debris Sizing, kilograms per hour	1,149	Planned for 2000				X	X		
WROC - Macroencapsulation, kilograms per hour	2,257	Planned for 2001					X		
WROC - Stabilization, cubic meters per day	7.6	Online					X		
WERF	49,610	Shutdown[b]				X	X	X	
INTEC Sewage Treatment Plant	3,200,000	Online							X
Storage Facility (cubic meters)									
INTEC Calcine Bin Sets	6,950	Online	X						
ANL–W Radioactive Sodium Storage	75	Online			X		X		
ANL–W Sodium Components Maintenance Shop	200	Online					X		
ANL–W Radioactive Scrap and Waste Storage	193	Online		X	X	X	X		
ANL–W EBR II Sodium Boiler Drain Tank	64	Online					X		
ANL–W HFEF Waste Characterization Area	37	Online		X	X				
INTEC Tank Farm	12,533	Online	X[c]		X				
INTEC FDPF HEPA Storage	25	Online			X		X		
INTEC NWCF HEPA Storage	56	Online			X		X		
INTEC CPP-1619 Storage	45	Online					X	X	
INTEC CPP-1617 Staging	8,523	Online					X	X	
RWMC Transuranic Storage Area-RE[c]	64,900	Online		X	X	X	X		
RWMC Waste Storage[d]	112,400	Online		X	X	X	X		
RWMC Intermediate-Level Storage	100	Online		X					
WROC PBF Mixed Low-level Radioactive Waste Storage	129	Online					X	X	
Portable Storage at SPERT IV	237	Online					X	X	
PBF WERF Waste Storage Building	685	Online					X	X	
Test Area North 647 Waste Storage	104	Online					X		
Test Area North 628 SMC Container Storage	125	Online					X		

Facility Name/Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed LLW	Haz	Non-Haz
Disposal Facility (cubic meters per year)									
RWMC Disposal Facility	37,700	Online				X			
CFA Landfill Complex	48,000	Online							X
Percolation Ponds	2,000,000	Online							X
FPF Sanitary Sewer	166,000	Online							X
TRA Warm Waste Evaporation Ponds	31,830	Online				X			
TRA Sanitary Waste Ponds	51,720	Online							X
TRA Cold Waste Pond	795,800	Online							X

a. NWCF was shut down on June 1, 2000, and is in standby pending facility upgrades and issuance of a new air permit.
b. WERF was denied its RCRA permit and required to shutdown by November 2, 2000.
c. Sodium-bearing waste is managed by the high-level radioactive waste program.
d. For these facilities, the low-level radioactive and mixed low-level radioactive wastes are considered alpha-contaminated low-level radioactive waste and alpha-contaminated mixed low-level radioactive waste (waste containing between 10 and 100 nanocuries of alpha activity per gram).

Note: To convert from cubic meters to cubic yards, multiply by 1.308.
Key: ANL–W, Argonne National Laboratory–West; CFA, Central Facilities Area; CPP, Chemical Processing Plant; EBR, Experimental Breeder Reactor; FDPF, Fluorinel Dissolution Process Facility; FPF, Fuel Processing Facility; Haz, hazardous; HEPA, high-efficiency particulate air; HFEF, Hot Fuel Examination Facility; HLW, high-level radioactive waste; INTEC, Idaho Nuclear Technology and Engineering Center; LLW, low-level radioactive waste; NWCF, New Waste Calcining Facility; PBF, Power Burst Facility; RWMC, Radioactive Waste Management Complex; SMC, Specific Manufacturing Complex; SPERT, Special Power Excursion Reactor Test; TRA, Test Reactor Area; TRU, transuranic waste, WERF, Waste Experimental Reduction Facility; WROC, Waste Reduction Operations Complex.
Source: Depperschmidt 2000b; DOE 1999e:3-54, 3-55; Perry 2000.

EPA placed INEEL on the National Priorities List on December 21, 1989. In accordance with CERCLA, DOE entered into a consent order with EPA and the State of Idaho to coordinate cleanup activities at INEEL under one comprehensive strategy. This agreement integrates DOE's CERCLA response obligations with RCRA corrective action obligations. Aggressive plans are in place to achieve early remediation of sites that represent the greatest risk to workers and the public. The goal is to complete remediation of contaminated sites at INEEL to support delisting from the National Priorities List by 2019 (DOE 1996b:3-141). More information on regulatory requirements for waste disposal is provided in Chapter 5.

3.3.11.2 High-Level Radioactive Waste

High-level radioactive waste at INEEL was generated in the process of extracting useful isotopes from spent nuclear fuel at INTEC. Most of this fuel was from the Naval Reactors Program. Most aqueous solutions from spent nuclear fuel processing and isotope extraction were concentrated by evaporation and separated into low-level and high-level radioactive waste streams in the Process Equipment Waste Evaporator. The liquid high-level radioactive waste was stored in subsurface tanks and then transformed by calcination into solid metallic oxides in a granular form. This calcination was completed in February 1998. The calcine is stored in stainless steel bins in near-surface concrete vaults where it awaits further processing into a form suitable for emplacement in a Federal repository. INEEL met the requirements of a December 1991 consent order with the State of Idaho and EPA to calcine all the high-level radioactive waste by June 30, 1998. Subsequently, the calcined waste will be treated to meet RCRA provisions on a schedule to be negotiated with the State of Idaho under the Federal Facility Compliance Act.

Although sodium-bearing waste is not high-level radioactive waste as specified in the Nuclear Waste Policy Act of 1982, it has been historically managed as high-level radioactive waste at INEEL. This is because some

of the physical and chemical properties of these two waste types are similar (e.g., both are acidic and both contain similar radionuclides, including transuranics) (DOE 1999i:1-11). About 5,300 cubic meters (1.4×10^6 gallons) of liquid-sodium-bearing waste remain in the INTEC Tank Farm. New treatment processes for the remaining liquid-sodium-bearing wastes are being analyzed in the *Idaho High-Level Waste and Facilities Disposition Environmental Impact Statement*.

3.3.11.3 Transuranic and Mixed Transuranic Waste

Transuranic waste generated since 1972 is segregated into contact-handled and remotely handled categories and stored at the Radioactive Waste Management Complex in a form designed for eventual retrieval (DOE 1996b:3-144). Some transuranic waste is also stored at the Radioactive Scrap and Waste Facility at Argonne National Laboratory–West (DOE 1995a:2.2-36). There is virtually no transuranic waste generated at INEEL. Most of the transuranic waste in storage was received from the Rocky Flats Environmental Technology Site (DOE 1996b:3-144). Transuranic waste is currently being stored, pending shipment to WIPP. Transuranic waste will be treated to meet WIPP waste acceptance criteria, packaged in accordance with DOE and Department of Transportation (DOT) requirements, and transported to WIPP for disposal (DOE 1996b:3-144) or a suitable geologic repository. The first shipment of transuranic waste from INEEL was received at WIPP on April 28, 1999 (DOE 1999j).

The existing treatment facilities for transuranic waste at INEEL are limited to testing, characterization, and repackaging. The Advanced Mixed Waste Treatment Project will be operated as a private sector treatment facility after its construction is completed (Moor and Peterson 1999). This facility will (1) treat waste to meet the most current requirements; (2) reduce waste volume and life-cycle cost to DOE; and (3) perform tasks in a safe and environmentally compliant manner (Saffle et al. 2000:3-11). The construction of the incinerator component of this facility has been deferred, pending the recommendation of a blue ribbon panel of experts. This panel of experts will assess and recommend new technology alternatives to incineration. The panel's recommendation is expected in December 2000 (DOE 2000b).

Waste containing between 10 and 100 nanocuries of alpha activity per gram of transuranic radionuclides is called alpha low-level radioactive waste. Although this waste is technically considered low-level radioactive waste rather than transuranic waste, it cannot be disposed of at INEEL because it does not meet all of the INEEL low-level radioactive waste disposal facility acceptance criteria. Alpha low-level radioactive waste and alpha mixed low-level radioactive waste are managed together as part of the transuranic waste program. It is expected that these wastes will be treated by the Advanced Mixed Waste Treatment Project and then be disposed of at WIPP (DOE 1995a:2.2-34, 2.2-35).

3.3.11.4 Low-Level Radioactive Waste

Liquid low-level radioactive waste is solidified before disposal (DOE 1996b:E-35). INTEC has the capability to treat aqueous low-level radioactive waste. Liquid low-level radioactive waste is concentrated at the INTEC Process Equipment Waste Evaporator, with the condensed vapor processed by the Liquid Effluent Treatment and Disposal Facility. The concentrated materials remaining after evaporation are stored in the Chemical Processing Plant–604 storage tanks. The Waste Experimental Reduction Facility was required to shutdown by November 2, 2000. In addition, aqueous low-level radioactive wastes are discharged to the two double-lined ponds at the Test Reactor Area for evaporation (DOE 1995a:2.2-39). The two Test Reactor Evaporation Ponds have a capacity of 36,790 cubic meters (48,100 cubic yards) each with a flow rate of 30 liters (8 gallons) per minute.

Low-level radioactive waste disposal occurs in pits and concrete-lined soil vaults in the subsurface disposal area of the Radioactive Waste Management Complex (DOE 1995a:2.2-39). Approximately 60 percent of the

low-level radioactive waste generated at INEEL is treated for volume reduction prior to disposal at the Radioactive Waste Management Complex. Additionally, some low-level radioactive waste is shipped off site to be incinerated, and the residual ash is returned to INEEL for disposal. The Radioactive Waste Management Complex is expected to be filled to capacity by the year 2030 (Mitchell et al. 1996b:3-26), although some proposals would close the low-level radioactive waste disposal facility by 2006 (DOE 1998f:B-4).

3.3.11.5 Mixed Low-Level Radioactive Waste

Mixed low-level radioactive waste is divided into two categories for management purposes: alpha mixed low-level radioactive waste and beta-gamma mixed low-level radioactive waste. Most of the alpha mixed low-level radioactive waste stored at INEEL is waste that has been reclassified from mixed transuranic waste and is managed as part of the transuranic waste program. Therefore, this section deals only with beta-gamma mixed low-level radioactive waste (DOE 1995a:2.2-39, 2.2-40).

Mixed low-level radioactive waste, including polychlorinated biphenyl–contaminated low-level radioactive waste, is stored at several onsite areas awaiting the development of treatment methods (DOE 1996b:3-144). Mixed low-level radioactive waste is stored at the Mixed Waste Storage Facility (or Waste Experimental Reduction Facility Waste Storage Building) and in portable storage units at the Power Burst Facility area. In addition, smaller quantities of mixed low-level radioactive waste are stored in various facilities at INEEL, including the Hazardous Chemical/Radioactive Waste Facility at INTEC, and the Radioactive Sodium Storage Facility and Radioactive Scrap and Waste Storage Facility at Argonne National Laboratory–West (DOE 1995a:2.2-41). Although mixed wastes are stored in many locations at INEEL, the bulk of that volume is solid waste stored at the Radioactive Waste Management Complex (DOE 1996b:E-38).

As part of the INEEL Site Treatment Plan and Consent Order required by the Federal Facility Compliance Act, preferred treatment options have been identified to eliminate the hazardous waste component for many types of mixed low-level radioactive waste (DOE 1995a:2.2-42). Mixed low-level radioactive waste is or will be processed to RCRA Land Disposal Restrictions treatment standards through several treatment facilities. Those treatment facilities and their operational status are: (1) Waste Experimental Reduction Facility Incinerator (shutdown), (2) Waste Experimental Reduction Facility Stabilization (operational), (3) Test Area North cask dismantlement (operational), (4) Sodium Process Facility (operational), (5) High-Efficiency Particulate Air Filter Leach (operational), (6) Waste Reductions Operations Complex Macroencapsulation (March 2001), (7) Debris Treatment (operational), and (8) Advanced Mixed Waste Treatment Project (March 2003). Commercial treatment facilities are also being considered, as appropriate. Currently, limited amounts of mixed low-level radioactive waste are disposed of at Envirocare of Utah (DOE 1999e:3-57).

3.3.11.6 Hazardous Waste

Approximately 1 percent of the total waste generated at INEEL (not including liquid nonhazardous waste) is hazardous waste. Most of the hazardous waste generated annually at INEEL is transported off site for treatment and disposal (DOE 1995a:2.2-45). Offsite shipments are surveyed to determine that the wastes have no radioactive content, and therefore are not mixed waste (DOE 1996b:3-145).

Highly reactive or unstable materials, such as waste explosives, are addressed on a case-by-case basis, and are either stored, burned, or detonated, as appropriate. The Waste Handling Facility Project at Argonne National Laboratory–West will be implemented to handle Argonne National Laboratory–West hazardous waste (DOE 1996b).

3.3.11.7 Nonhazardous Waste

Approximately 90 percent of the solid waste generated at INEEL is classified as industrial waste and is disposed of on site in a landfill complex in the Central Facilities Area and off site at the Bonneville County landfill (DOE 1995a:2.2-47). The onsite landfill complex contains separate areas for petroleum-contaminated media, industrial waste, and asbestos waste (Werner 1997). The onsite landfill is 4.8 hectares (12 acres), and is being expanded by 91 hectares (225 acres) to provide capacity for at least 30 years (DOE 1996b:3-145).

Sewage is disposed of in surface impoundments in accordance with terms of the October 7, 1992 Consent Order. Waste in the impoundments is allowed to evaporate, and the resulting sludge is placed in the landfill. Solids are separated and reclaimed where possible (DOE 1996b:3-145). Nonhazardous service wastewater generated at INTEC is disposed of in percolation ponds at a flow rate of 3.8 million to 7.6 million liters (1 million to 2 million gallons) per day (Werner 1997). The INTEC sanitary sewer system collects and transfers sanitary waste to the sewage treatment lagoons east of INTEC for treatment and disposal. This system has a capacity of 3,200,000 cubic meters (4,190,000 cubic yards) per year (DOE 1999e:3-58). The Test Reactor Area Cold Waste Pond disposes of about 1,500 liters (400 gallons) per minute of nonhazardous wastewater with a capacity of 33,960 cubic meters (44,400 cubic yards). The TRA sanitary sewer system collects and transfers about 98 liters (26 gallons) per minute of sanitary wastewaters to the sewage treatment lagoons east of the Test Reactor Area for treatment and disposal. This system has a capacity of 20,600 cubic meters (26,900 cubic yards).

3.3.11.8 Waste Minimization

The DOE Idaho Operations Office has an active waste minimization and pollution prevention program to reduce the total amount of waste generated and disposed of at INEEL. This is accomplished by eliminating waste through source reduction or material substitution; by recycling potential waste materials that cannot be minimized or eliminated; and by treating all waste that is generated to reduce its volume, toxicity, or mobility prior to storage or disposal. The Idaho Operations Office published its first Waste Minimization Plan in 1990, which defined specific goals, methodology, responsibility, and achievements of programs and organizations. The achievements and progress have been updated at least annually. Implementing pollution prevention projects reduced the total amount of waste generated at INEEL in 1998 by approximately 1,100 cubic meters (1,400 cubic yards). Examples of pollution prevention projects completed in 1998 at INEEL include: reduction of routine operations hazardous waste by approximately 55 metric tons (61 tons) by collecting engine oil by a recycling vendor for energy recovery; reducing cleanup/stabilization of hazardous waste by approximately 20 metric tons (22 tons) by dismantling the Mobile Test Assembly Cask and sending the clean lead to the clean lead storage area for recycling; and reducing both routine operations and cleanup/stabilization low-level radioactive waste by approximately 19 cubic meters (25 cubic yards) by recycling depleted uranium scrap metal from both normal facility operations and deactivation of a facility (DOE 1999f:44).

3.3.11.9 Waste Management PEIS Records of Decision

The *Waste Management PEIS* Records of Decision affecting INEEL are shown in **Table 3–28** for the waste types analyzed in this NI PEIS. Decisions on the various waste types were announced in a series of Records of Decisions that have been issued on the *Waste Management PEIS*. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629); the hazardous waste Record of Decision was issued on August 5, 1998 (63 FR 41810); the high-level radioactive waste Record of Decision was issued on August 12, 1999 (64 FR 46661); and the low-level radioactive and mixed low-level radioactive waste Record of Decision was issued on February 18, 2000 (65 FR 10061). The transuranic waste Record of Decision states that DOE will develop and operate mobile and fixed facilities to characterize and prepare transuranic waste

Table 3–28 Waste Management PEIS Records of Decision Affecting INEEL

Waste Type	Preferred Action
High-level radioactive	DOE has decided that INEEL should store its immobilized high-level radioactive waste on site until transfer to a geologic repository.[a]
Transuranic and mixed transuranic	DOE has decided that INEEL should prepare and store its transuranic waste on site pending disposal at WIPP[b] or another suitable geologic repository.
Low-level radioactive	DOE has decided to treat INEEL's low-level radioactive waste on site.[c]
Mixed low-level radioactive	DOE has decided to regionalize treatment of mixed low-level radioactive waste at INEEL.[c] This includes the onsite treatment of INEEL's wastes and could include treatment of some mixed low-level radioactive waste generated at other sites.
Hazardous	DOE has decided to continue to use commercial facilities for treatment of INEEL nonwastewater hazardous waste. DOE will also continue to use onsite facilities for wastewater hazardous waste.[d]

a. From the Record of Decision for high-level radioactive waste (64 FR 46661).
b. From the Record of Decision for transuranic waste (63 FR 3629).
c. From the Record of Decision for low-level radioactive and mixed low-level radioactive waste (65 FR 10061).
d. From the Record of Decision for hazardous waste (63 FR 41810).

Source: 63 FR 3629; 63 FR 41810; 64 FR 46661; 65 FR 10061.

for disposal at WIPP. Each DOE site that has or will generate transuranic waste will, as needed, prepare and store its transuranic waste on site. The hazardous waste Record of Decision states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of the nonwastewater hazardous waste with ORR and SRS continuing to treat some of their own nonwastewater hazardous waste, on site in existing facilities, where this is economically feasible. The high-level radioactive waste Record of Decision states that immobilized high-level radioactive waste will be stored at the site of generation until transfer to a geologic repository. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision states that for the management of low-level radioactive waste, minimal treatment will be performed at all sites, and disposal will continue, to the extent practicable, on site at INEEL, LANL, ORR, and SRS. In addition, Hanford and the Nevada Test Site will be available to all DOE sites for low-level radioactive waste disposal. Mixed low-level radioactive waste will be treated at Hanford, INEEL, ORR, and SRS, and disposed of at Hanford and the Nevada Test Site. More detailed information concerning DOE's decisions for the future configuration of waste management facilities at INEEL is presented in the high-level radioactive waste, transuranic waste, hazardous waste, and low-level radioactive waste and mixed low-level radioactive waste Records of Decision.

3.4 HANFORD SITE

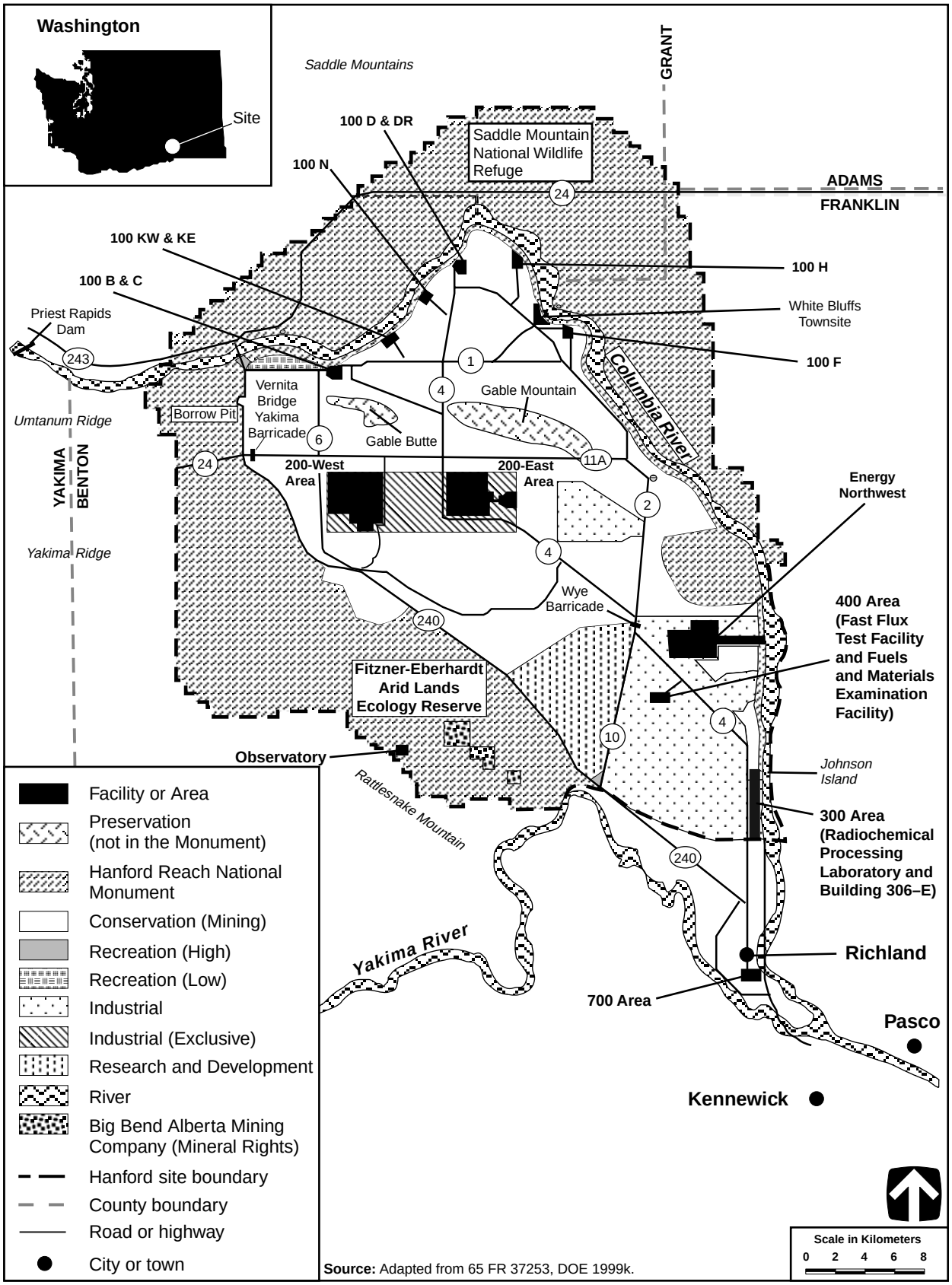
Hanford, established in 1943 as one of the three original Manhattan Project sites, is located on approximately 148,000 hectares (365,000 acres) in Washington State, just north of Richland. It extends over parts of Adams, Benton, Grant, and Franklin counties. Hanford was a U.S. Government defense materials production site that included nuclear reactor operation, uranium and plutonium processing, storage and processing of spent nuclear fuel, and management of radioactive and hazardous and state dangerous wastes. Present Hanford programs are diversified and include management of radioactive wastes, cleanup of waste sites, soil, and groundwater related to past releases, stabilization and storage of spent nuclear fuel, renewable energy technologies, waste disposal technologies, contamination cleanup, and plutonium stabilization and storage (DOE 1999k). The primary emphasis at the site is on cleanup activities.

Hanford is owned and used primarily by DOE, but portions of it are owned, leased, or administered by other Government agencies. Public access is limited to travel on the Route 4 and Route 10 access roads as far as the Wye Barricade, State Routes 24 and 240, and the Columbia River. By restricting access to the site, the public is buffered from areas formerly used for production of nuclear materials and currently used for waste storage and disposal. Only about 6 percent of the land area has been disturbed and is actively used, leaving mostly vacant land with widely scattered facilities. On June 9, 2000, the President issued a proclamation that established the 78,900-hectare (195,000-acre) Hanford Reach National Monument (65 FR 37253). This proclamation recognizes the unique character and biological diversity of the area, as well as its geological, paleontological, historic, and archaeological significance. The monument includes not only land adjacent to the Columbia River, but also other areas on the Hanford Site as depicted on **Figure 3–12**. The U.S. Fish and Wildlife Service will manage the monument under existing agreements with DOE. Land within the monument that is not subject to existing agreements will be managed by DOE; however, DOE will consult with the Secretary of the Interior when developing any management plans affecting these lands.

Hanford includes extensive production, service, research, and development areas. Onsite programmatic and general purpose facilities, many of which are inactive, total approximately 799,000 square meters (8.6 million square feet) of space. Fifty-one percent (408,000 square meters [4.4 million square feet]) is general purpose space, including offices, laboratories, shops, warehouses, and other support facilities. The remaining 392,000 square meters (4.2 million square feet) of space are programmatic facilities including processing, evaporation, filtration, waste recovery, waste treatment, waste storage facilities, and research and development laboratories. More than half of the general purpose and programmatic facilities are more than 30 years old. Facilities designed to perform previous missions are being evaluated for reuse in the cleanup mission. The existing facilities are grouped into the following numbered operational areas (DOE 1996b:3-20, 3-21).

The 100 Areas, in the northern part of the site on the southern shore of the Columbia River, are the site of eight retired plutonium production reactors and the dual-purpose N Reactor, all of which have been permanently shut down since 1991. Waste sites throughout the 100 Areas are currently undergoing remediation, consisting of excavating contaminated soils and structural materials. Contaminated groundwater in the 100 Areas is being treated via both ex situ and in situ methods. Approximately 2,000 metric tons (2,200 tons) of spent nuclear fuel are currently stored in indoor basins in the 100 Areas pending approval and storage in the 200 Areas. The 100 Areas cover about 1,100 hectares (2,720 acres).

The 200-West and 200-East Areas are in the center of the site and are about 8 and 11 kilometers (5 and 6.8 miles), respectively, south of the Columbia River. The 200-West and 200-East Areas are also about 20 and 12 kilometers (12.2 and 7.3 miles), respectively, west of the Columbia River. Historically, these areas have been used for fuel processing; plutonium processing, fabrication, and storage; and waste management and disposal activities. DOE has constructed the Environmental Restoration Disposal Facility in the 200 Area to provide disposal capacity for environmental remediation waste (e.g., low-level, mixed low-level, and



Source: Adapted from 65 FR 37253, DOE 1999k.

Figure 3–12 Generalized Land Use at the Hanford Site and Vicinity

dangerous wastes) generated during remediation of the 100, 200, and 300 Areas of the Hanford Site. The facility currently covers about 130 hectares (320 acres) and can be expanded up to 414 hectares (1,020 acres) as additional waste disposal capacity is required (DOE 1999k). The 200 Areas cover about 1,600 hectares (3,950 acres).

The 300 Area is in the southern part of the site, just north of the city of Richland. A few of the facilities continue to support nuclear and nonnuclear research and development of the Pacific Northwest National Laboratory. Many of the facilities in the 300 Area are in the process of being deactivated. The Environmental Molecular Sciences Laboratory and associated research programs provide research capability to advance technologies in support of DOE's environmental remediation and waste management programs (DOE 1999k). Waste sites in the 300 Area are currently undergoing remediation, consisting of excavating contaminated soils and structural materials. The 300 Area has also been proposed for accelerated remediation of waste sites and inactive buildings to support future non-DOE uses. The 300 Area covers 150 hectares (370 acres). The Radiochemical Processing Laboratory (RPL) (Building 325) and the Development Fabrication Test Laboratory (Building 306–E) are located in this area and would be used under certain alternatives under this NI PEIS.

The 400 Area, 8 kilometers (5 miles) northwest of the 300 Area, is the location of FFTF and FMEF. FFTF was designed and built as a liquid-metal (sodium) cooled reactor to be the nation's leading test reactor for development and testing of materials and equipment for the Liquid Metal Fast Breeder Reactor Programs. The reactor was neither designed nor operated as a breeder reactor itself. FFTF operated for about 10 years (1982–1992) as a national research facility testing advanced nuclear fuels, materials, components, active and passive reactor safety technologies, and gaining operating experience for the next generation of nuclear reactors. FFTF also produced a wide variety of medical isotopes and made tritium for the U.S. fusion research program. In 1995, FFTF was in the process of being shutdown, but was directed in 1997 to maintain a standby condition. The final decision on this reactor is to be determined in the Record of Decision for this NI PEIS.

FMEF, located in the 400 Area adjacent to the west of FFTF, was constructed in the late 1970s and early 1980s to perform fuel fabrication and development and postirradiation examination of breeder reactor fuels. FMEF was never operated and is currently in a lay-up condition suitable for a future mission. The building is clean and uncontaminated, as no nuclear materials were ever introduced into the building. The six-level process building (Building 427) is the main structure of FMEF and encloses about 17,400 square meters (188,000 square feet) of operating area. FMEF also consists of several connected buildings. The exterior walls are reinforced concrete and the cell walls are constructed of high-density concrete. The facility was designed and constructed for spent fuel examination and was equipped for mixed oxide fuel fabrication.

Other areas at Hanford include Energy Northwest facilities and a section of land currently owned by Washington State for the disposal of hazardous substances. Energy Northwest currently operates Washington Nuclear Plant Number 2 on leased land approximately 4 kilometers (2.5 miles) northeast of the 400 Area. Originally leased for the operation of three nuclear power plants, construction of two of the plants was stopped and now other industrial options are being considered. Other facilities at Hanford include a specialized training center, the Hazardous Materials Management and Emergency Response (HAMMER) Volpentest Training and Education Center, which is used to train hazardous materials response personnel. It is located in the southeastern portion of the site and covers about 32 hectares (80 acres). The Hanford Patrol Training Academy, a regional law-enforcement training facility, provides classrooms, library resources, practice shoot houses, an exercise gym, and an obstacle course. The Laser Interferometer Gravitational Wave Observatory, a national research facility, built by the National Science Foundation for scientific research, is designed to detect cosmic gravitational waves. The facility consists of two optical tube arms, each 4 kilometers (2.5 miles) long, arrayed in an "L" shape, and extremely sensitive to vibrations (DOE 1999k). The 700 Area is the administrative center in downtown Richland and consists of Government-owned buildings (e.g., the Federal Building).

In addition, there are DOE-leased facilities and DOE contractor-owned facilities that support Hanford operations. These facilities are on private land south of the 300 Area and outside of the 1100 and 3000 Areas (DOE 1996b:3-21).

DOE has transferred the 1100 Area (which served as a procurement, central warehousing, vehicle maintenance, transportation, and distribution center for the Hanford Site) and the smaller 3000 Area to the Port of Benton for use in economic development and diversification (DOE 1998g, 1998h, 1998i).

3.4.1 Land Resources

Land resources include land use and visual resources. Each of these resource areas is described for the site as a whole, as well as for the locations of the proposed activities.

3.4.1.1 Land Use

Land may be characterized by its potential for the location of human activities (land use). Natural resource attributes and other environmental characteristics could make a site more suitable for some land uses than for others. Changes in land use may have both beneficial and adverse effects on other resources such as ecological, cultural, geological, aquatic, and atmospheric.

3.4.1.1.1 General Site Description

The Tri-Cities area southeast of Hanford includes residential, commercial, and industrial land use. This area, encompassing the cities of Richland, Kennewick, and Pasco, is the population center closest to Hanford. Additional cities near the southern boundary of Hanford include Benton City, Prosser, and West Richland. Agriculture is a major land use in the remaining areas surrounding Hanford. In 1996, wheat was the largest crop in terms of area planted in Benton, Franklin, and Grant counties. Alfalfa, apples, asparagus, cherries, corn, grapes, and potatoes are some of the other major crops in Benton, Franklin, and Grant counties.

DOE has designated the entire Hanford Site as a National Environmental Research Park, an outdoor laboratory for ecological research to study the environmental effects of energy development. The Hanford National Environmental Research Park is a shrub-steppe habitat that contains a wide range of semiarid land ecosystems and offers the opportunity to examine linkages between terrestrial, subsurface, and aquatic environments.

Land use designations at Hanford include preservation, conservation, recreation, industrial, and research and development (Figure 3–12). Approximately 6 percent of the site has been disturbed and is occupied by DOE facilities (Neitzel 1999). Hanford contains a variety of widely dispersed facilities, including retired reactors, research and development facilities, and various deactivated production and processing plants. Preservation and conservation are the largest land use categories at Hanford. Industrial areas include the 200 Areas, an area to the east of the 200 Areas, and most of the southeast corner of the site.

Important areas within the preservation land use category include the Hanford Reach National Monument, that incorporates a portion of the Columbia River corridor, as well as the Fitzner-Eberhardt Arid Lands Ecology Reserve to the south and west, and portions of the Hanford Site north of the Columbia River (65 FR 37253). Other special status land in the vicinity include McNary National Wildlife Refuge, administered by the U.S. Fish and Wildlife Service, and the Columbia River Islands Area of Critical Environmental Concern and McCoy Canyon, both administered by the Bureau of Land Management. The Columbia River, which is adjacent to and runs through the Hanford Site, is used for numerous purposes including public boating, water skiing, fishing, hunting, transportation, irrigation, and municipal water supply. Public access is allowed on certain islands, while other areas are considered sensitive because of unique habitats and the presence of

cultural resources. The area known as the Hanford Reach includes the quarter-mile strip of public land on either side of the last free-flowing, nontidal segment of the Columbia River. On June 9, 2000, the President issued a proclamation that established the Hanford Reach National Monument (65 FR 37253) covering 78,900 hectares (195,000 acres). This proclamation recognizes the unique character and biological diversity of the area, as well as its geological, paleontological, historic, and archaeological significance. The U.S. Fish and Wildlife Service will manage the monument under existing agreements with DOE. Land within the monument that is not subject to existing agreements will be managed by DOE; however, DOE will consult with the Secretary of the Interior when developing any management plans affecting these lands.

On June 27, 2000, a fire known as the 24 Command Fire, was started by a fatal motor vehicle accident on State Route 24, about 2 miles west of the State Route 240 intersection. As a result of high winds and temperatures and low humidity, the fire spread rapidly and eventually consumed 66,322 hectares (163,884 acres) of Federal, state, and private lands. A total of 24,384 hectares (60,254 acres) within Hanford burned, including lands within the Hanford Reach National Monument, most of the Arid Lands Ecology Reserve, and areas near former production sites (Figures 3–12 and 3–21). The fire was declared controlled on July 2, 2000. Fire suppression impacts included construction of 66 kilometers (41 miles) of bulldozed fire lines, widened dirt roads, and cut fences (DOI 2000). Wind and sheet and rill erosion are likely due to the loss of vegetation and fire fighting activities. Impacts to the land should not be permanent because rehabilitation measures, including revegetation and fence repair, are being implemented.

The Hanford Site has developed a comprehensive land use plan to define how to best use the land at the site for the next 50 years (DOE 1999k). The plan provides the framework within which future use of the site's lands and resources will occur. This framework consists of four basic elements including: a land use map depicting land uses for the site; land use definitions describing the purpose, intent, and principal uses of each land use designation; a set of policies directing land use actions; and implementing procedures. Figure 3–12 reflects the land use categories developed in the *Hanford Comprehensive Land-Use Plan EIS* (DOE 1999k) as modified by the designation of the Hanford Reach National Monument.

Under separate treaties signed in 1855, lands occupied by the present Hanford Site were ceded to the United States by the Confederated Tribes and Bands of the Yakama Indian Nation, the Confederated Tribes of the Umatilla Indian Reservation, and the Nez Perce Tribe of Western Idaho. Under these treaties, the tribes retained the right to fish in their usual and accustomed places, hunt, gather roots and berries, and pasture horses and cattle on open, unclaimed lands. Tribal fishing rights have been recognized as effective within the Hanford Reach. Tribal governments and DOE, however, disagree over the applicability of tribal member's treaty-reserved rights to hunt, gather plants, and pasture livestock on the Hanford Site. The tribes and DOE have proceeded with the land use planning process, while reserving all rights to assert their respective positions regarding treaty rights (DOE 1999k).

3.4.1.1.2 Locations of Proposed Activities

300 AREA

The *Hanford Comprehensive Land-Use Plan EIS* (DOE 1999k) and Record of Decision (64 FR 61615) have designated the 300 Area as an industrial area for the foreseeable future. An industrial area is defined in that EIS as an area that is suitable and desirable for activities such as reactor operations, transport facilities, mining, manufacturing, warehousing, and distribution operations. The 300 Area, which is just north of the city of Richland and west of the Columbia River, covers 150 hectares (371 acres). It is the site of former reactor fuel fabrication facilities and is also the principal location of nuclear research and development facilities serving the Hanford Site. The RPL/306–E buildings are in the 300 Area.

400 AREA

Under the *Hanford Comprehensive Land-Use Plan EIS* (DOE 1999k) and Record of Decision (64 FR 61615) land in the 400 Area is designated for industrial use, including reactor operations, for the foreseeable future. The 400 Area occupies 60 hectares (150 acres) and is 7 kilometers (4.3 miles) to the west of the nearest site boundary. It is the site of FFTF and FMEF. FFTF is a test reactor that was used for the development and testing of materials and equipment for the liquid metal breeder reactor program. FMEF is an unused building designed and constructed for spent fuel examination and equipped for mixed oxide fuel fabrication.

3.4.1.2 Visual Resources

Visual resources are natural and human-created features that give a particular landscape its character and aesthetic quality. Landscape character is determined by the visual elements of form, line, color, and texture. All four elements are present in every landscape. The stronger the influence exerted by these elements in a landscape, the more interesting the landscape.

3.4.1.2.1 General Site Description

Hanford is in the Pasco Basin of the Columbia Plateau north of the city of Richland, where the Yakima and Columbia Rivers join. The topography of land in the vicinity of Hanford ranges from generally flat to gently rolling. Rattlesnake Mountain, rising to 1,060 meters (3,480 feet) above mean sea level, forms the southwestern boundary of the site. Gable Mountain and Gable Butte are the highest land forms within the site, rising approximately 60 meters (200 feet) and 180 meters (590 feet), respectively. The Columbia River flows through the northern part of the site and, turning south, forms part of the eastern site boundary. White Bluffs, steep whitish-brown bluffs adjacent to the Columbia River and above the northern boundary of the river in this region, are a striking feature of the landscape.

Typical of the regional shrub-steppe desert, the site is dominated by widely spaced, low-brush grasslands. A large area of unvegetated, stabilized sand dunes extends along the east boundary, and unvegetated blowouts are scattered throughout the site. Hanford is characterized by mostly undeveloped land, with widely spaced clusters of industrial buildings along the southern and western banks of the Columbia River and at several interior locations.

Between June 27 and July 2, 2000, a fire known as the 24 Command Fire burned 66,322 hectares (163,884 acres) of Federal, state, and private lands, including 24,384 hectares (60,254 acres) within Hanford (DOI 2000). Areas burned included land within the Hanford Reach National Monument, most of the Arid Lands Ecology Reserve, and areas near former production sites (see Figure 3–12). Firefighting activities resulted in the construction of 66 kilometers (41 miles) of bulldozed fire lines, widened dirt roads, and cut fences. Thus, both the fire and the activities required to control it resulted in dramatic changes to the visual character of affected portions of the site. Visual resources would likely also be affected by dust storms resulting from exposed soil. These alterations to the visual character of Hanford will change over time as rains promote the growth of vegetation, fire lines are rehabilitated, and fences are repaired. Because of the slow regeneration of sagebrush, however, it will be years before the visual character of the landscape will mirror pre-fire conditions.

The adjacent visual landscape consists primarily of rural rangeland and farms. The city of Richland, part of the Tri-Cities area, is the only adjoining urban area. Viewpoints affected by DOE facilities are primarily associated with the public access roadways (including State Routes 24 and 240, Hanford Road, Horn Rapids Road, Route 4 South, and Steven Drive), the bluffs, and the northern edge of the city of Richland. The Energy Northwest (formerly known as the Washington Public Power Supply System) nuclear reactors and DOE

facilities are brightly lit at night and are highly visible from many areas. Developed areas are consistent with a Bureau of Land Management Visual Resource Management Class IV rating, while the remainder of the Hanford Site ranges in Visual Resource Management rating from Class II to Class III (DOI 1986). Management activities within Class II and III areas may be seen, but should not dominate the view, while management activities in Class IV areas dominate the view and are the focus of viewer attention.

3.4.1.2.2 Locations of Proposed Activities

300 AREA

The tallest structures within the 300 Area vicinity are the water towers, with a height of 40 meters (130 feet), and the meteorological tower with a height of 61 meters (200 feet) in height. The 300 Area is visible from Route 4, which runs in a north-south direction along the western boundary of the site (Nielsen 2000). Because the 300 Area is a highly developed industrial area, it has a Visual Resource Management Class IV rating. Natural features of visual interest within a 40-kilometer (25-mile) radius include the Columbia River immediately to the east, Rattlesnake Mountain at 24 kilometers (15 miles), Gable Mountain at 27 kilometers (17 miles), and Gable Butte at 35 kilometers (22 miles).

400 AREA

FMEF, the tallest building in the 400 Area, is 30 meters (100 feet) tall and can be seen from State Route 240. Developed areas within the 400 Area are consistent with a Visual Resource Management Class IV rating. Natural features of visual interest within a 40-kilometer (25-mile) radius include the Columbia River at 6.8 kilometers (4.2 miles), Rattlesnake Mountain at 17 kilometers (11 miles), Gable Mountain at 19 kilometers (12 miles), and Gable Butte at 27 kilometers (17 miles).

3.4.2 Noise

Noise is unwanted sound that interferes or interacts negatively with the human or natural environment. Noise may disrupt normal activities or diminish the quality of the environment.

3.4.2.1 General Site Description

Major noise sources within the Hanford Site include various facilities, equipment, and machines (e.g., cooling systems, transformers, engines, pumps, boilers, steam vents, paging systems, construction and materials-handling equipment, and vehicles). Wind has been identified as a major source of background sound levels at Hanford. Data from two noise surveys indicate that background noise levels (measured as the 24-hour equivalent sound level) at Hanford range from 30 to 60.5 dBA. The 24-hour background sound level in undeveloped areas at Hanford ranges from 24 to 36 dBA, except when high winds elevate sound levels. The primary source of noise at the site and nearby residences is traffic. Most Hanford industrial facilities are far enough from the site boundary that noise levels at the boundary from these sources are not measurable, or are barely distinguishable from background noise levels. Hanford is currently in compliance with state noise regulations. Noise sources, existing noise levels at Hanford, and noise standards are described in the *Storage and Disposition PEIS* (DOE 1996b:3-29–3-31, F-31, F-32) and in the 1999 *Hanford Site National Environmental Policy Act (NEPA) Characterization* (Neitzel 1999:4.137-138).

The potential impact of traffic noise resulting from activities at Hanford was evaluated for a draft EIS addressing the siting of the proposed New Production Reactor (Neitzel 1999:4.138). Estimates were made of baseline traffic noise along two major access routes: State Route 24, leading from the Hanford Site west to Yakima, and State Route 240, south of the site and west of Richland, where it handles maximum traffic

volume. About 9 percent of the employees at Hanford commute by vanpool or bus. Modeled traffic noise levels (equivalent sound level [1-hour]) at 15 meters (50 feet) from State Route 24 for both peak and offpeak periods were 62 dBA. Traffic noise levels from State Route 240 for both peak and offpeak periods were 70 dBA. These traffic noise levels were projections based on employment levels about 30 percent higher than actual levels at Hanford in 1997. Existing traffic noise levels may be different as a result of changes in site employment and ride-sharing activities (DOE 1999e:3-8; Neitzel 1999:4.138-4.141).

Washington State has established noise standards for different source and receiving areas. Hanford belongs to source area Class C (industrial). The maximum allowable noise level for residential, commercial, and industrial areas is 50 to 70 dBA (DOE 1996b:3-29 and 3-31, Neitzel 1999:4.138). The EPA guidelines for environmental noise protection recommend a day-night average sound level of 55 dBA as sufficient to protect the public from the effects of broadband environmental noise in typically quiet outdoor and residential areas (EPA 1974:29). Land use compatibility guidelines adopted by the Federal Aviation Administration and the Federal Interagency Committee on Urban Noise indicate that yearly day-night average sound levels less than 65 dBA are compatible with residential land uses (14 CFR Part 150). These guidelines further indicate that noise levels up to 75 dBA are compatible with residential uses if suitable noise reduction features are incorporated into structures. It is expected that for most residences near Hanford, the day-night average sound level is less than 65 dBA and is compatible with the residential land use, although for some residences along major roadways noise levels may be higher.

3.4.2.2 Locations of Proposed Activities

300 AREA

No distinguishing noise characteristics in the 300 Area have been identified. The 300 Area is just north of Richland and adjacent to the site boundary along the Columbia River. No sound level data have been collected in this area except for measurements that reflect traffic noise levels.

400 AREA

No distinguishing noise characteristics in the 400 Area have been identified. The 400 Area is far enough away from the site boundary, 7 kilometers (4.3 miles), that noise levels at the site boundary from these sources are not measurable or are barely distinguishable from background levels.

3.4.3 Air Quality

Air pollution refers to the introduction, directly or indirectly of any substance into the air that could endanger human health, harm living resources and ecosystems as well as material property, and impair or interfere with the comfortable enjoyment of life or other legitimate uses of the environment. Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Air quality is affected by air pollutant emission characteristics, meteorology, and topography.

3.4.3.1 General Site Description

The climate at Hanford and the surrounding region is characterized as that of a semiarid steppe. The humidity is low and winters are mild. The average annual temperature is 11.8 °C (53.3 °F); average monthly temperatures range from a minimum of -0.4 °C (31.3 °F) in January to a maximum of 24.6 °C (76.2 °F) in July. The average annual precipitation is 16 centimeters (6.3 inches). Prevailing winds at the Hanford Meteorological Station are from the west-northwest. The average annual wind speed is 3.4 meters per second (7.6 miles per hour) (Dirkes, Hanf, and Poston 1999:7.5; DOE 1999e:3-5).

Most of Hanford is within the South-Central Washington Intrastate Air Quality Control Region #230, but a small portion of the site is in the Eastern Washington-Northern Idaho Interstate Air Quality Control Region #62. None of the areas within Hanford and its surrounding counties are designated as nonattainment areas, with respect to NAAQS for criteria air pollutants (40 CFR Section 81.348). However, particulate matter concentrations can reach relatively high levels in eastern Washington State because of extreme natural events, such as dust storms, volcanic eruptions, and large brush fires. Washington State ambient air quality standards have not considered "rural fugitive dust" from exceptional natural events when estimating the maximum background concentrations of particulates in the area east of the Cascade Mountain crest. In June 1996, EPA adopted the policy that allows dust storms to be treated as uncontrollable natural events. The air quality impact of dust storms can therefore be excluded during the determination of whether this area is in nonattainment for atmospheric particulates (Neitzel 1999). Applicable NAAQS and Washington State ambient air quality standards are presented in **Table 3–29**.

The primary sources of air pollutants at Hanford include emissions from power generation and chemical processing (Neitzel 1999:4.30). Other sources include vehicles, construction, environmental remediation, and waste management activities (Wisness 2000). The existing ambient air pollutant concentrations at the site boundary attributable to sources at Hanford are presented in Table 3–29. These concentrations are based on dispersion modeling using emissions for Hanford, excluding the 400 Area for 1999 (Wisness 2000). The 400 Area emissions during FFTF standby are estimated using the EPA Standard AP-42 guideline. The concentrations at the site boundary for the 400 Area were calculated using EPA's SCREEN3 dispersion model; however, the concentrations from the other sources at the site were calculated using the ISCST3 dispersion model. SCREEN3 estimates of maximum concentrations are conservative when compared to the ISCST3 estimates. The ISCST3 modeling was performed using the 1999 meteorological data for Hanford, whereas the SCREEN3 modeling was performed using a set of worst-case meteorological conditions. Although the location for maximum concentrations may be different, for the purpose of this NI PEIS, it was assumed to be occurring at the same location.

Only those pollutants that would be emitted by any of the alternatives evaluated in this NI PEIS are presented. Hanford sources are limited and background concentrations of criteria pollutants are well below ambient standards. As shown in Table 3–29, these modeled concentrations from Hanford sources represent a small percentage of the ambient air quality standards. Hanford emissions should not result in air pollutant concentrations that violate the ambient air quality standards for criteria pollutants. Detailed information on emissions of other pollutants at Hanford is discussed in the *Hanford Site NEPA Characterization* (Neitzel 1999:4.27–4.32).

There are no Prevention of Significant Deterioration Class I areas within 100 kilometers (62 miles) of Hanford. A Class I area is one in which very little increase in pollution is allowed due to the pristine nature of the area. Hanford and its vicinity are classified as a Class II area in which more moderate increases in pollution are allowed. Hanford operates under a Prevention of Significant Deterioration permit issued in 1980. New emission sources require a Prevention of Significant Deterioration increment consumption analysis. The recent designation of the Hanford Reach as a national monument (65 FR 37253) might lead to a proposal to redesignate this area, that includes part of Hanford and adjoining areas, as Prevention of Significant Deterioration Class I, although that appears unlikely at this time due to a variety of political and technical issues.

Table 3–29 Comparison of Modeled Ambient Air Concentrations from Hanford Sources with Most Stringent Applicable Standards or Guidelines

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Maximum Hanford Concentration Excluding 400 Area (micrograms per cubic meter)[b]	Maximum 400 Area Concentration (micrograms per cubic meter)
Criteria pollutants				
Carbon monoxide	8 hours	10,000[c]	23.8	3.5
	1 hour	40,000[c]	58.2	5.1
Nitrogen dioxide	Annual	100[c]	0.634	0.032
Ozone	1 hour	235[d]	(e)	(e)
PM_{10}	Annual	50[c]	0.0162	0.002
	24 hours	150[c]	0.112	0.898
Sulfur dioxide	Annual	50[f]	0.0114	0.164
	24 hours	260[f]	0.365	29.8
	3 hours	1,300[c]	2.41	67.0
	1 hour	1,000[f]	5.02	74.4
	1 hour	660[f, g]	5.02	74.4
Other regulated pollutants				
Total suspended particulates	Annual	60	0.0162	0.002
	24 hours	150	0.112	0.898

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Site contributions based on a 1999 emissions inventory, excluding the 400 Area.
c. Federal and state standard.
d. Federal 8-hour standard is currently under litigation.
e. Not directly emitted or monitored by the site.
f. State standard.
g. Not to be exceeded more than twice in any 7 consecutive days.

Note: NAAQS also includes standards for lead. No sources of lead emissions have been identified at the site. Emissions of other air pollutants not listed here have been identified at Hanford, but are not associated with any alternative evaluated. EPA revised the ambient air quality standards for particulate matter and ozone in 1997; however, these standards were under litigation. In 1999, new standards, effective on September 16, 1997, could not be enforced. The ozone standard is a 1-hour concentration of 235 micrograms per cubic meter (0.12 parts per million) (62 FR 38856). The 8-hour standard could not be enforced. For particulate matter, the current PM_{10} annual standard is retained (62 FR 38652).
Source: 40 CFR Part 50; WDEC 1998; Wisness 2000.

A sitewide air operating permit is being developed for Hanford, scheduled to be issued as a draft by the end of 2000, in accordance with Title V of the Clean Air Act and Amendments of 1990 and the Federal and state programs under 40 CFR Part 70 and WAC 173-401, respectively (WDEC 1997). The Hanford air operating permit will include a compilation of requirements for both radioactive emissions now covered by the existing state license and nonradioactive emissions. The primary effects of the air operating permit will be to consolidate approval orders and applicable requirements into one permit, require the permitted party to conduct periodic monitoring to show continuous compliance with permit conditions and applicable requirements, require biannual reporting and annual certification of continuous compliance, and increase the state's and EPA's enforcement position.

Based on 1996 monitoring conducted off site by the Washington State Department of Ecology, the annual and 24-hour PM_{10} standards were not exceeded (Neitzel 1999:4.29). Ambient air quality at Hanford is discussed in more detail in the *Hanford Site Environmental Report for Calendar Year 1998* (Dirkes, Hanf, and Poston 1999).

Routine monitoring of most nonradiological pollutants is not conducted at the site. Monitoring of nitrogen oxides and total suspended particulates at Hanford has been discontinued as a result of phasing out programs for which the monitoring was required. Carbon monoxide, sulfur dioxide, and nitrogen dioxide have been monitored periodically in communities and commercial areas southeast of Hanford. In 1995, air samples of semivolatile organic compounds were collected on the site and at an offsite location, and the results are discussed in the site's annual environmental report. All concentrations of these compounds were below the applicable risk-based concentrations.

3.4.3.2 Locations of Proposed Activities

300 AREA

Prevailing winds in the 300 Area are from the southwest. The 300 Area emits various nonradiological air pollutants from power generation and process sources (Neitzel 1999:4.30, 4.31).

400 AREA

Prevailing winds in the 400 Area are from the south-southwest, with a secondary maximum from the northwest. The 400 Area emits various nonradiological air pollutants (see Sections 3.4.3.1 and 4.4.1.2.3) (Neitzel 1999:4.30).

3.4.4 Water Resources

Water resources include all forms of surface water and subsurface groundwater.

3.4.4.1 Surface Water

Surface water includes marine or freshwater bodies that occur above the ground surface, including rivers, streams, lakes, ponds, rainwater catchments, embayments, and oceans.

3.4.4.1.1 General Site Description

Major surface water features at Hanford include the Columbia River, Columbia riverbank seepage, springs, and ponds (**Figure 3–13**). In addition, the Yakima River flows along a short section of the southern boundary of the site. The Columbia River is the second largest river in the contiguous United States in terms of total flow and is the dominant surface water feature on the site. Flow of the Columbia River is regulated by several dams, seven upstream and four downstream from the site. The nearest dam upstream from Hanford is the Priest Rapids Dam, and the closest downstream dam is the McNary Dam. The Hanford Reach is the portion of the Columbia River that extends from Priest Rapids Dam to the upstream edge of Lake Wallula behind McNary Dam. Because the flows are regulated, flow rates in the Hanford Reach can vary considerably; it is the last remaining free-flowing, nontidal section of the river. The average daily flow rate at Priest Rapids Dam is 3,360 cubic meters (118,700 cubic feet) per second. Peak flows generally occur from April through June corresponding to runoff from snowmelt. Due to larger than normal snowpacks, the peak flow rate in 1997 was nearly 11,750 cubic meters (415,000 cubic feet) per second. The width of the river varies from approximately 300 to 1,000 meters (1,000 to 3,300 feet) within the Hanford Site (Neitzel 1999:4.55, 4.56).

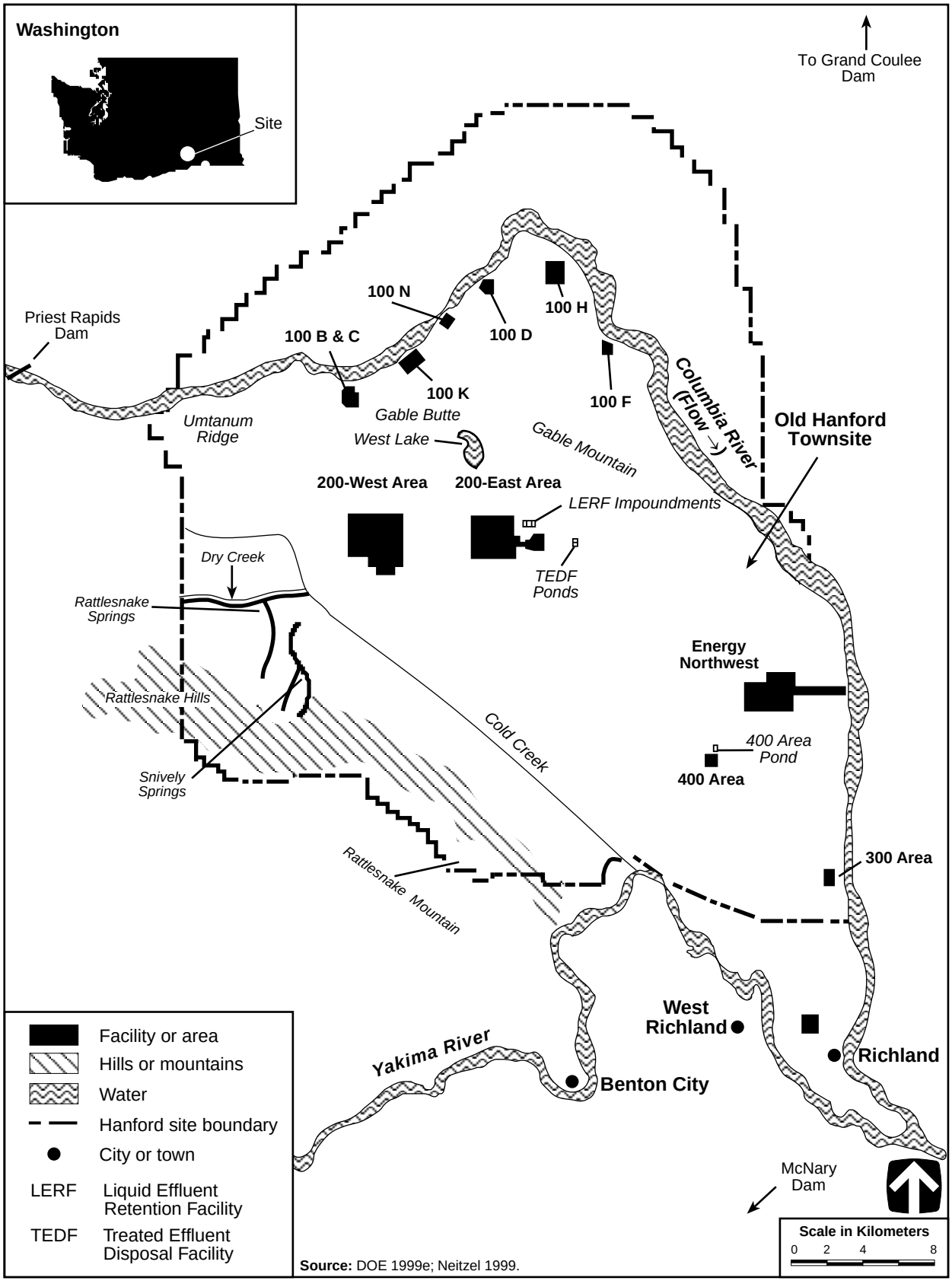


Figure 3–13 Surface Water Features at the Hanford Site

Primary uses of the Columbia River include hydroelectric power generation, irrigation of crops in the Columbia Basin, and barge transportation. The Hanford Reach is the upstream navigable limit of barge traffic. The Columbia River is also used extensively for recreation, including fishing, hunting, boating, sailboarding, water-skiing, diving, and swimming. In addition to a water supply source for the Hanford Site, several communities use the Columbia River as their source of drinking water (Neitzel 1999:4.56). Nine of the 12 DOE-owned, contractor-operated water plants on the Hanford Site use water from the Columbia River (Dirkes, Hanf, and Poston 1999:4.47–4.49).

The Washington State Department of Ecology classifies the Columbia River, from Grand Coulee to the Washington-Oregon border and encompassing the Hanford Reach, as Class A (excellent). Class A waters are suitable for essentially all uses, including raw drinking water, recreation, and wildlife habitat. Federal and state drinking water standards, and DOE Order 5400.5, apply to the Columbia River and are currently being met (DOE 1999k:4-35). Although no federally designated Wild and Scenic Rivers exist in the Hanford Site vicinity, the Hanford Reach is being considered for listing under the National Wild and Scenic Rivers Act as part of broader resource conservation initiatives (DOE 1999k:4-5; Neitzel 1999:6.10). The Hanford Reach was recently proclaimed a National Monument (refer to Section 3.4.1.1.1).

DOE continues to assert a federally reserved water withdrawal right for the Columbia River. Hanford withdraws approximately 13.5 billion liters (3.6 billion gallons) per year from the Columbia River (DOE 1999e:3-30).

About one-third of the Hanford Site drains into the Yakima River. The average daily flow rate for the Yakima River is 104 cubic meters (3,670 cubic feet) per second. The peak average daily flow rate in 1997 was nearly 1,300 cubic meters (45,900 cubic feet) per second (Neitzel 1999:4.58).

Rattlesnake Springs and Snively Springs are in the western portion of the site and flow into intermittent streams that infiltrate rapidly into the surface sediments (Figure 3–13). Water discharged from Rattlesnake Springs flows down Dry Creek, a tributary to Cold Creek, for about 3 kilometers (1.9 miles) before infiltrating into the ground. An alkaline spring has also been documented at the east end of Umtanum Ridge. Several springs are also found on the slopes of Rattlesnake Mountain along the western and southwestern edges of the site (DOE 1999k:4-30; Neitzel 1999:4.58). The seepage of groundwater into the Columbia River was documented along the Hanford Reach long before Hanford Site operations began. This seepage occurs both below the river surface and on the exposed riverbank. These relatively small seeps flow intermittently, influenced primarily by changes in river level. Hanford-origin contaminants have been documented in these discharges along the Hanford Reach (DOE 1999k:4-30; Neitzel 1999:4.65).

Other naturally occurring surface water features include West Lake and three previously undocumented clusters of approximately 20 vernal ponds or pools. The clusters are located on the eastern end of Umtanum Ridge, in the central part of Gable Butte, and at the eastern end of Gable Mountain. The ponds appear to form during the relatively wetter winter ponds in shallow depressions underlain by a layer of basalt (DOE 1999k:4-31; Neitzel 1999:4.67).

Artificial ponds also exist on the site primarily associated with waste management activities. These include: water storage ponds in the 100 K-Area, the two Treated Effluent Disposal Facility (TEDF) disposal ponds and the three Liquid Effluent Retention Facility impoundments adjacent to the 200-East Area, and the 400 Area Pond (FFTF Pond or 4608 B/C ponds) used by FFTF and other facilities (Figure 3–13) (DOE 1999k:4-31; Neitzel 1999:4.57, 4.67). While West Lake, a natural pond located north of the 200 Areas that predates Hanford operations, has not received effluents, it was sustained by the artificially elevated water table beneath much of Hanford, attributable to historic waste management activities and current wastewater disposal in the 200 Areas. Although not accessible to the public, these ponds are accessible by waterfowl (DOE 1999k:4-32;

Neitzel 1999:4.67, 4.88). In addition to these features, there are irrigation ponds and wetlands located in the northwest portion of the site and north of the Columbia River (Neitzel 1999:4.57, 4.67).

In 1998, the Hanford Site had two NPDES permits: Permit #WA-000374-3 and Permit #WA-002591-7. Permit #WA-000374-3 included four inactive outfalls in the 100-N Area and three active outfalls (two in the 100-K Area and one in the 300 Area). There were two instances of noncompliance for these outfalls in 1998. Permit #WA-002591-7 covered one outfall located at the 300 Area TEDF. The 300 Area TEDF had 14 exceedances in 1998. This disposal facility was in normal operation and meeting design specifications at the time of these events. All indications suggest that the facility is unable to consistently meet the restrictions of the facility's NPDES permit, despite the use of the best available technology. An application for a permit modification was submitted to EPA in November 1997 (Dirkes, Hanf, and Poston 1999:2.24, 2.25). The modification requested transfer of the two active 100-K Area outfalls from Permit #WA-000374-3 to Permit #WA-002591-7, among other items. A revised permit was issued April 2, 1999, and became effective May 5, 1999 (Chapin 1999). Revised effluent limits for the 300 Area TEDF were established under the modified Permit #WA-002591-7 (Dirkes, Hanf, and Poston 1999:2.25). Permit #WA-000374-3 has lapsed.

Hanford was covered by two industrial stormwater permits (WAR-00-000F, WAR-10-000F) in 1998. An annual comprehensive site compliance evaluation was performed and documented in 1998. In accordance with the September 30, 1998, Federal Register (63 FR 52430), the stormwater general permit for industrial activity (WAR-00-000F) was terminated and replaced by the multisector general stormwater permit (WAR-10-000F). On December 28, 1998, a Notice of Intent was submitted to EPA for coverage under the NPDES multisector permit (WAR-10-000F) (Dirkes, Hanf, and Poston 1999:2.25).

DOE Richland Operations Office has a pretreatment permit (CR-IU005) from the city of Richland for the discharge of wastewater from the Environmental Molecular Sciences Laboratory in the 300 Area. Also, there are numerous sanitary waste discharges to the ground through sanitary systems permitted by the Washington State Department of Health, as well as 400 Area sanitary waste discharges to the Energy Northwest treatment facility. Sanitary waste from the 300 Area and other facilities north of and in Richland discharge to the city of Richland treatment facility (Dirkes, Hanf, and Poston 1999:2.25).

Hanford is subject to a Washington State Department of Ecology liquid effluent consent order that regulates liquid effluent discharges to the ground. All state waste discharge permit applications for discharges covered under the consent order have been submitted. One new state waste discharge permit was issued on May 1, 1998, by the Washington State Department of Ecology (Permit ST-4509 for Hanford cooling water and condensate discharges). In 1998, there were eight noncompliances in three of the seven state waste discharge permits currently in place at Hanford. One of these was for exceeding the permit limit for manganese in the cooling water discharge to the 400 Area Pond. The exceedance was attributed to the naturally high levels of the metal in the source water (Dirkes, Hanf, and Poston 1999:2.25, 2.26).

All radiological contaminant concentrations measured in the Columbia River in 1998 were lower than the DOE Derived Concentration Guides and Washington State ambient surface water quality criteria. For nonradiological parameters, applicable standards for Class A–designated surface water were met, with results comparable to those over the past 5 years. During 1998, there was no evidence of deterioration in water quality attributable to Hanford operations along the Hanford Reach (Dirkes, Hanf, and Poston 1999:4.22, 4.27–4.29).

The Columbia River is also the primary discharge area for the unconfined aquifer underlying Hanford. The site conducts sampling of these groundwater seeps during low flow and refers to them as riverbank springs. Water samples were collected from eight Columbia River shoreline spring areas in 1998. All samples were analyzed for gamma-emitting radionuclides, gross alpha, gross beta, and tritium. Samples from selected

springs were analyzed for strontium-90; technetium-99; iodine-129; and uranium-234, 235, and 238. Samples were also analyzed for metals and anions (Dirkes, Hanf, and Poston 1999:4.20, 4.34–4.36).

All radiological contaminant concentrations measured in 1998 were less than the DOE Derived Concentration Guides. Tritium in riverbank springs water at the Old Hanford Townsite (refer to Figure 3–13 for locations) and the 100-N Area exceeded the state ambient surface water quality criterion (20,000 picocuries per liter), with the maximum of 120,000 picocuries per liter observed at the Old Hanford Townsite. Gross beta activities in riverbank springs water at the 100-H Area exceeded the ambient criterion (50 picocuries per liter), with a maximum observed value of 72 picocuries per liter. While there are no ambient surface water quality criteria directly applicable to uranium, total uranium levels exceeded the site-specific proposed EPA drinking water standard in the 300 Area (equivalent to 13.4 picocuries per liter), with a maximum total uranium activity of 58 picocuries per liter. Gross alpha activity exceeded the ambient surface water quality criterion of 15 picocuries per liter in riverbank springs water at the 300 Area, with a maximum observed value of 56 picocuries per liter. This is consistent with the elevated uranium levels. All other radionuclide activities in 300 Area springs water were less than ambient surface water quality criteria (Dirkes, Hanf, and Poston 1999:4.36, A.10, A.11, C.3). Elevated uranium activities exist in the unconfined aquifer beneath the 300 Area in the vicinity of uranium fuel fabrication facilities and inactive waste sites. Elevated tritium activities have also been measured in the 300 Area riverbank springs and are indicators of the contaminated groundwater plume emanating from the 200 Areas. However, in 1998, the maximum observed activity level was 9,600 picocuries per liter and below the ambient surface water quality criterion (Dirkes, Hanf, and Poston 1999:4.38, C.4).

Nonradiological contaminants measured in riverbank springs located on the Hanford shoreline in 1998 were below the applicable Washington State ambient surface water criteria except for chromium concentrations in 100-B, 100-K, 100-D, and 100-H Area springs exceeding the acute toxicity level of 16 micrograms per liter (Dirkes, Hanf, and Poston 1999:4.38, C.4).

Flooding of the site has occurred along the Columbia River, but chances of recurrence have been greatly reduced by the construction of dams to regulate river flow. Major floods are typically due to the melting of the winter snowpack combined with above normal precipitation (Neitzel 1999:4.60). No maps of flood-prone areas have been produced by the Federal Emergency Management Agency. The Federal Emergency Management Agency produces these maps for areas capable of being developed, and the Hanford Site is not designated for commercial or residential development (DOE 1999k:4-34). However, analyses have been completed to determine the potential for the probable maximum flood. This is determined through hydrologic factors, including the amount of precipitation within the drainage basin, snow melt, and tributary conditions. The probable maximum flood for the Columbia River below the Priest Rapids Dam has been calculated at 40,000 cubic meters (1.4 million cubic feet) per second, which is greater than the 500-year flood (DOE 1999k:4-34; Neitzel 1999:4.60). The extent of the 1948 flood, and the extent of the probable maximum flood, are shown in **Figure 3–14**. Potential flooding due to dam failure has been evaluated by the U.S. Army Corps of Engineers. Upstream failures could have any number of causes, the magnitude of the resultant flooding depending on the size of the breach in the dam. The U.S. Army Corps of Engineers evaluated various scenarios for failure of the Grand Coulee Dam, located approximately 130 kilometers (80 miles) from Hanford, and assumed flow conditions of about 11,300 cubic meters (400,000 cubic feet) per second. The worst-case scenario assumed a 50 percent breach in the dam (**Figure 3–15**). The flood wave from an instantaneous 50 percent breach was calculated to be 600,000 cubic meters (21 million cubic feet) per second. In addition to the areas affected by the probable maximum flood, the remainder of the 100 Area, the 300 Area, and nearly all of Richland, Washington, would be flooded. Determinations were not made for larger instantaneous breaches in the Grand Coulee Dam, because the 50 percent scenario was believed to be the largest conceivable flow from a natural or manmade breach. It was not considered credible that a structure as large as the Grand

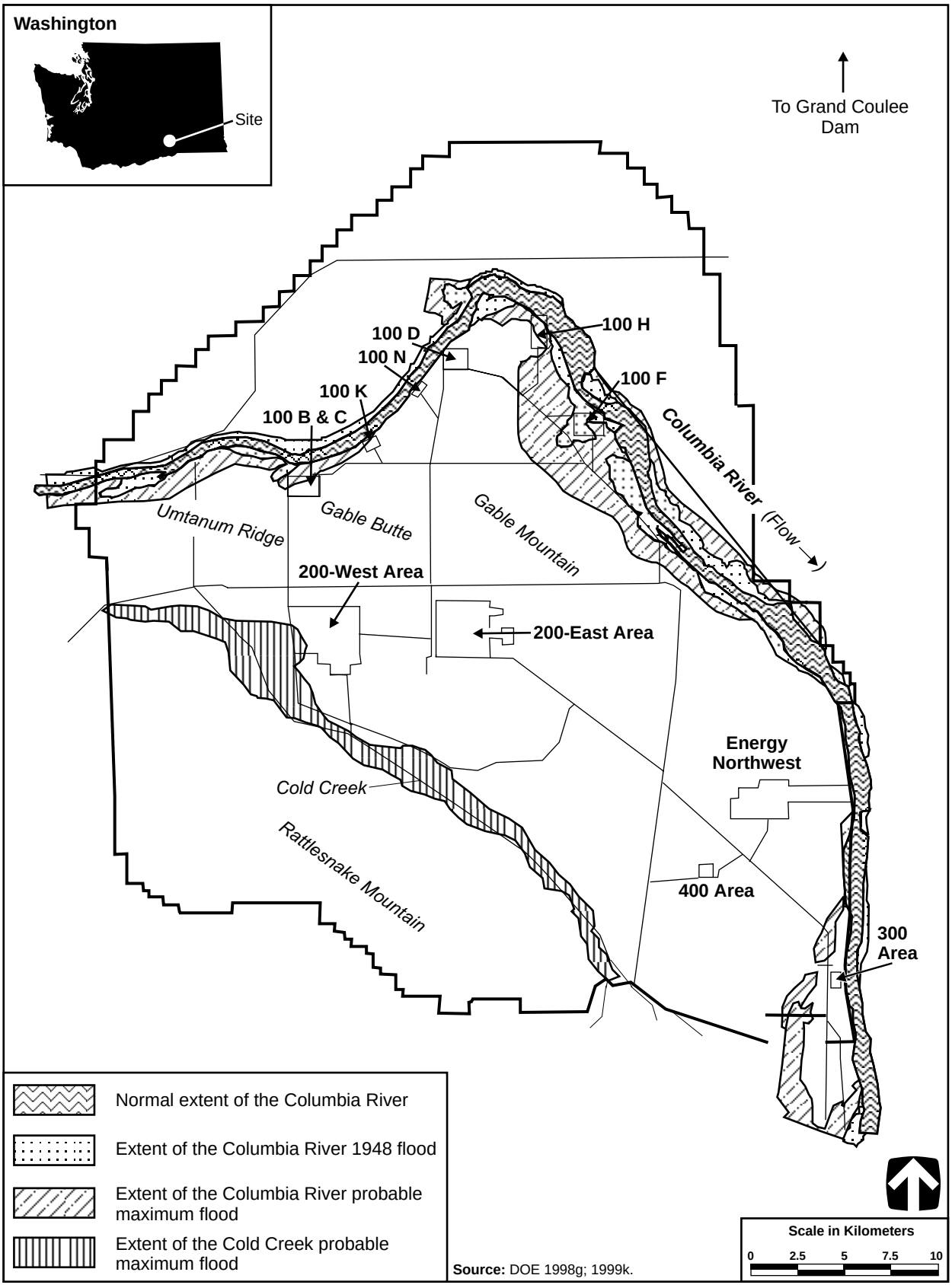


Figure 3–14 Flood Area for the Probable Maximum Flood and Columbia River 1948 Flood

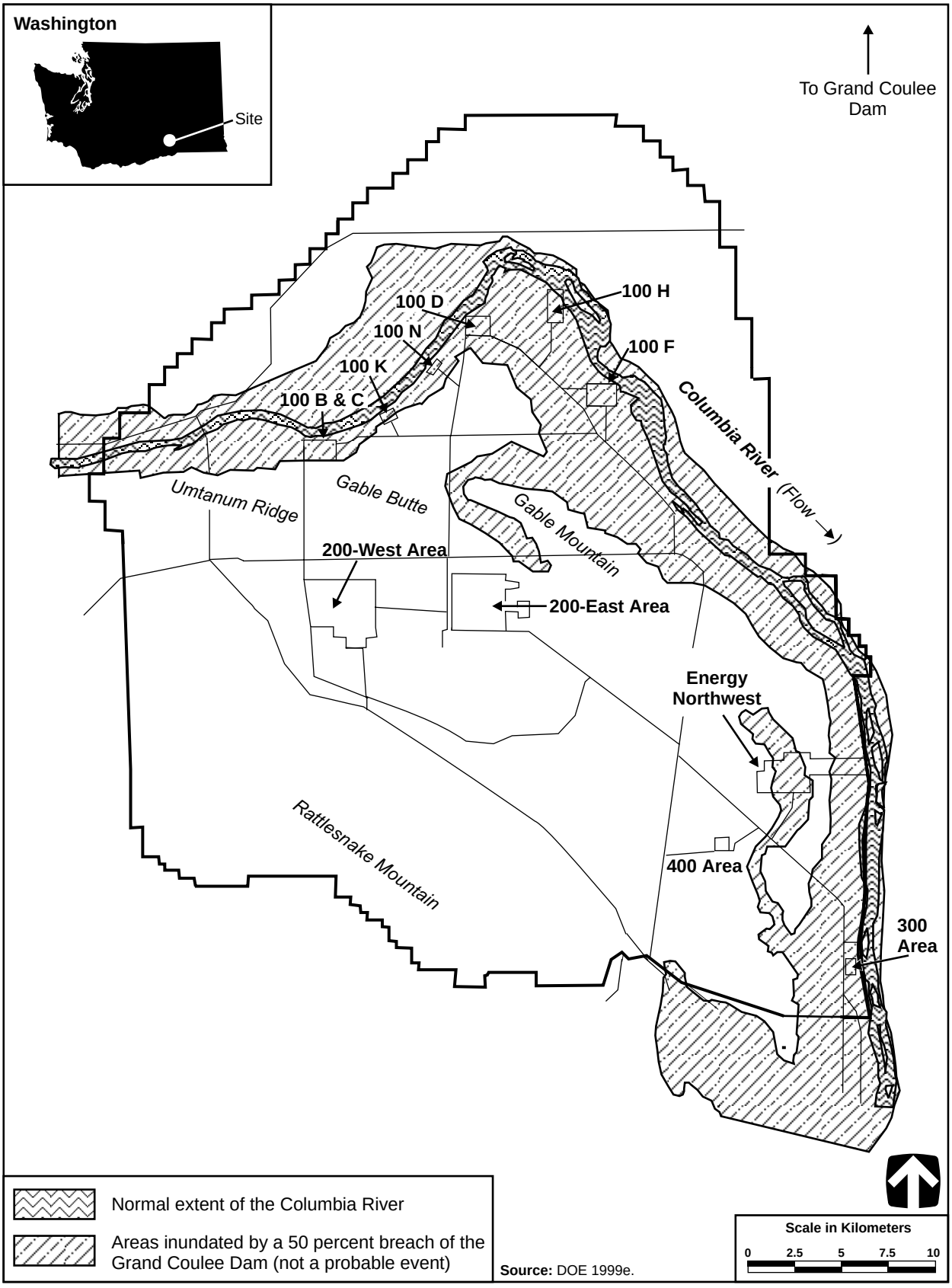


Figure 3–15 Flood Area of a 50 Percent Breach of the Grand Coulee Dam

Coulee Dam could be 100 percent destroyed instantaneously. The analysis also assumed that the 50 percent breach would occur only as the result of direct explosive detonation, and not because of some natural event such as an earthquake (DOE 1999k:4.34; Neitzel 1999:4.60, 4.65).

3.4.4.1.2 Locations of Proposed Activities

300 AREA

The 300 Area is located in the southeast corner of the site adjacent to the Columbia River. Although no site-specific flood analysis is available for the 300 Area, analyses have been completed for the site as a whole, as previously discussed. The 300 Area does not lie within the area postulated to be affected by the probable maximum flood, but locations just to the west of the area would be affected (Figure 3–14). However, the 300 Area would be inundated from a 50 percent breach of the Grand Coulee Dam (Figure 3–15). Water for the 300 Area, including for RPL/306–E, is provided by the city of Richland, which obtains about two-thirds of its water from the Columbia River (FDH 1999:3; Neitzel 1999:4.133). Water consumption in the 300 Area is approximately 594 million liters (157 million gallons) per year (FDH 1999:3). Sanitary wastewater from the 300 Area is discharged to the city of Richland treatment facility (Dirkes, Hanf, and Poston 1999:2.25).

RPL is connected to the 300 Area sanitary sewer system and to a separate retention process sewer system. This system collects equipment cooling water, laboratory waste liquids, and other liquids that have a slight potential for radioactive contamination. The retention process sewer system routes process wastewater to the 307 basins at the 340 Complex and ultimately to the 300 Area TEDF, which operates under NPDES Permit WA002591-7. The system is monitored for radioactivity and, if an alarm is triggered, the effluent is diverted to a dedicated basin at the 340 Complex. Otherwise, the effluent is screened at the 307 basins before being conveyed to the 300 Area TEDF. Direct sampling and analysis of the system is also conducted as needed (DOE 1997c: 4-58; 2000c:C-2, C-3). Historically, RPL has generated an average of 1.13 million liters (300,000 gallons) of sanitary wastewater annually and 2.27 million liters (600,000 gallons) of process wastewater per year (DOE 1997c: 4-58). RPL currently generates an average of 3.98 million liters (1.05 million gallons) of sanitary wastewater annually and 3.6 million liters (950,000 gallons) per year of process wastewater (DOE 2000c:C-3; Tenforde 2000). Liquid, low-level radioactive waste generation has averaged less than 3,800 liters (1,000 gallons) per year (DOE 1997c:4-58, 4-59). Building 306–E is also served by the sanitary sewer and process sewer systems. For Building 306–E, sanitary wastewater generation averages 995,000 liters (262,000 gallons) on an annual basis and process wastewater generation averages 24.9 million liters (6.57 million gallons) per year (Tenforde 2000). Process wastewater with the potential for radioactive contamination is not routinely generated at the facility (DOE 1997c:4-60, B.2-2). Waste management activities and facilities are discussed in greater detail under Section 3.4.11.

400 AREA

The 400 Area is located 6.3 kilometers (3.9 miles) from the west bank of the Columbia River. No specific flooding analyses have been completed for the 400 Area, but analyses have been completed for the site as a whole. According to the sitewide data, the elevation of the ground surface in the 400 Area is about 30 meters (100 feet) above that of the maximum calculated flood from a 50 percent breach of the Grand Coulee Dam (Mecca 1997a:4) (Figure 3–15). Also, the 400 Area is above the elevation of the maximum historical floods of 1894 (Neitzel 1999:4.61) and 1948 (Figure 3–14).

The only surface water body in the vicinity of the 400 Area is the 400 Area Pond (i.e., FFTF Pond or 4608 B/C ponds) located just north of the 400 Area (DOE 1999k:4-31; Neitzel 1999:4.67). It is designed and used to dispose of nonradioactive process wastewater collected by the process sewer system from four 400 Area facilities including FFTF, FMEF, the Maintenance and Storage Facility, and the water pumphouse. The

400 Area Pond consists of two cells measuring 15 by 30 meters (50 by 100 feet) with 1.2-meter (4-foot) walls. The majority of the wastewater discharged to the pond system is cooling tower blowdown from FFTF's eight auxiliary cooling towers and FMEF's three cooling towers (currently inactive). Individual effluent streams are collected at a central drain line that runs to the ponds, with the effluent monitored before discharge. Wastewater rapidly percolates into the ground, leaving the ponds dry under normal conditions. The discharges are regulated under State Waste Discharge Permit No. ST-4501, and the effluent is periodically sampled and analyzed for permit compliance. Approximately 76 million liters (20 million gallons) per year of process wastewater is discharged to the ponds. Also, about 3.8 million liters (1 million gallons) of sanitary wastewater is discharged annually from 400 Area facilities to the Energy Northwest system for treatment (DOE 2000c:11; Nielsen 1999:38, 39, 41). There are no radiological liquid effluent pathways to the environment from either FFTF or FMEF under normal operations (DOE 1997c:4-6, 4-29). Liquid, low-level radioactive waste from equipment washing is generated during standby operations at a maximum rate of about 3,785 liters (1,000 gallons) per year. It is collected in tanks and transported to the 200 Area Effluent Treatment Facility for treatment and disposal (DOE 1997c:4-54; Nielsen 1999:39).

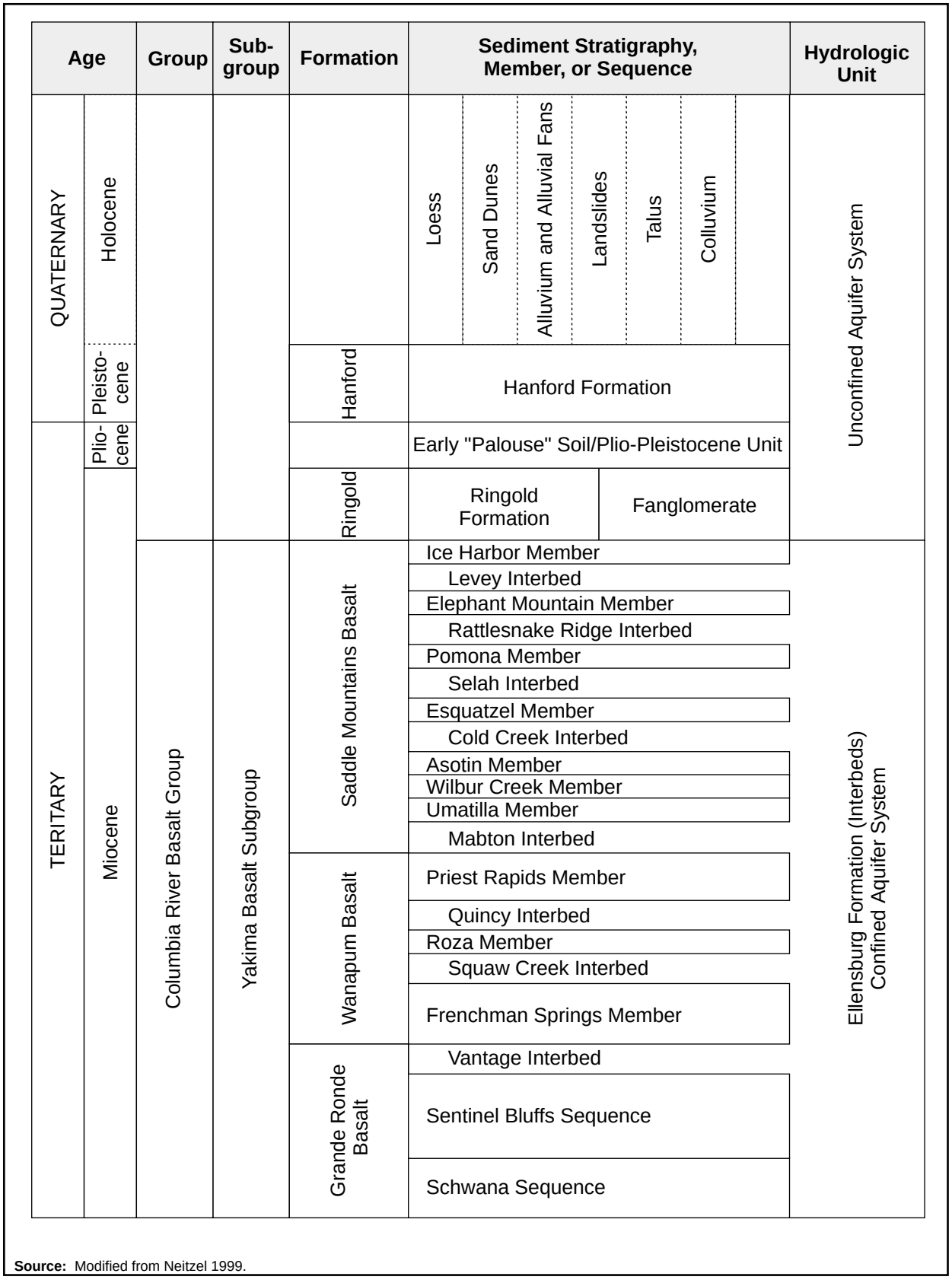
FMEF is also equipped with a separate retention/radioactive liquid waste system for handling wastewater not conveyed to the sanitary system due to the slight potential for radioactive contamination of some wastewater streams. Wastewater first flows to two 22,700-liter (6,000-gallon) collection tanks, where the wastewater can be sampled and either discharged by operator command to the process sewer system or, if contaminated, can be trucked to the 200 Area Effluent Treatment Facility, or other suitable facility, for processing (DOE 1997c:B.1-11; 2000c:7). Waste management activities and facilities are discussed in greater detail under Section 3.4.11.

3.4.4.2 Groundwater

Aquifers are classified by Federal and state authorities according to use and quality. The Federal classifications include Classes I, II, and III groundwater. Class I groundwater is either the sole source of drinking water or is ecologically vital. Classes IIA and IIB are current or potential sources of drinking water (or other beneficial use), respectively. Class III is not considered a potential source of drinking water and is of limited beneficial use.

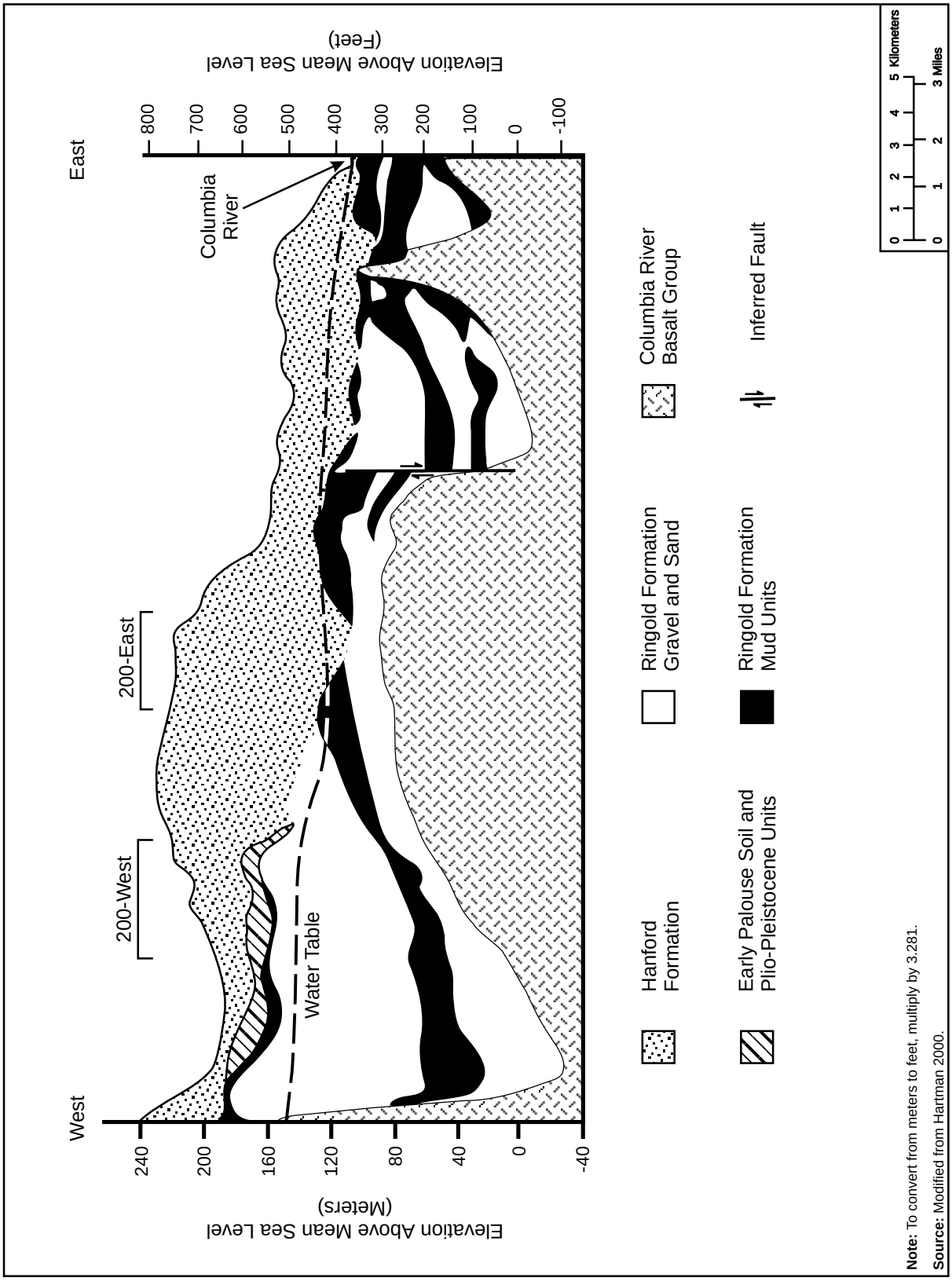
3.4.4.2.1 General Site Description

Groundwater under Hanford occurs in confined and unconfined aquifer systems. The hydrostratigraphic (water bearing) units comprising these systems are illustrated in **Figure 3–16**. The unconfined aquifer system, referred to as the suprabasalt aquifer system, lies within the glacioalluvial sands and gravels of the Hanford Formation and, to a greater degree, the fluvial and lacustrine sediments of the Ringold Formation. Groundwater generally flows eastward across the site from recharge areas in the higher elevations on the western site boundary, with discharge primarily to the Columbia River (**Figure 3–17**) (DOE 1999e:3-31; Neitzel 1999:4.68). The Yakima River is also considered a source of recharge (Neitzel 1999:4.68). Because of site wastewater disposal practices, however, the water table has risen as much as 27 meters (89 feet) in the 200 West Area. This has caused groundwater mounding with radial and northward flow components in the 200 Area, although groundwater elevations have declined since 1984 due to decreased wastewater disposal (DOE 1999e:3-31; Neitzel 1999:4.70). Depth to groundwater across the site ranges from 0.3 meters (1 foot) along the Columbia River to more than 106 meters (348 feet) near the center of the site (Dirkes, Hanf, and Poston 1999:6.10). Daily river level fluctuations may result in water table fluctuations of up to 3 meters (10 feet) near the Columbia River (Neitzel 1999:4.68).



Source: Modified from Neitzel 1999.

Figure 3–16 Stratigraphic Column for the Pasco Basin and Hanford Site



Note: To convert from meters to feet, multiply by 3.281.

Source: Modified from Hartman 2000.

Figure 3–17 Geologic Cross Section of the Hanford Site

The confined aquifer system at Hanford consists of sedimentary interbeds and interflow zones that occur between basalt flows in the Columbia River Basalt Group. Aquifer thickness varies from several centimeters to at least 52 meters (171 feet). Recharge of the confined aquifer occurs where the basalt formations are near ground level, and thus surface water is allowed to infiltrate them. Groundwater in the confined aquifer system discharges to the Columbia River, but in some places, moves toward areas of vertical interconnection with the overlying unconfined aquifer system. One such area is near the Gable Mountain anticline (DOE 1999e:3-32; Neitzel 1999:4-68).

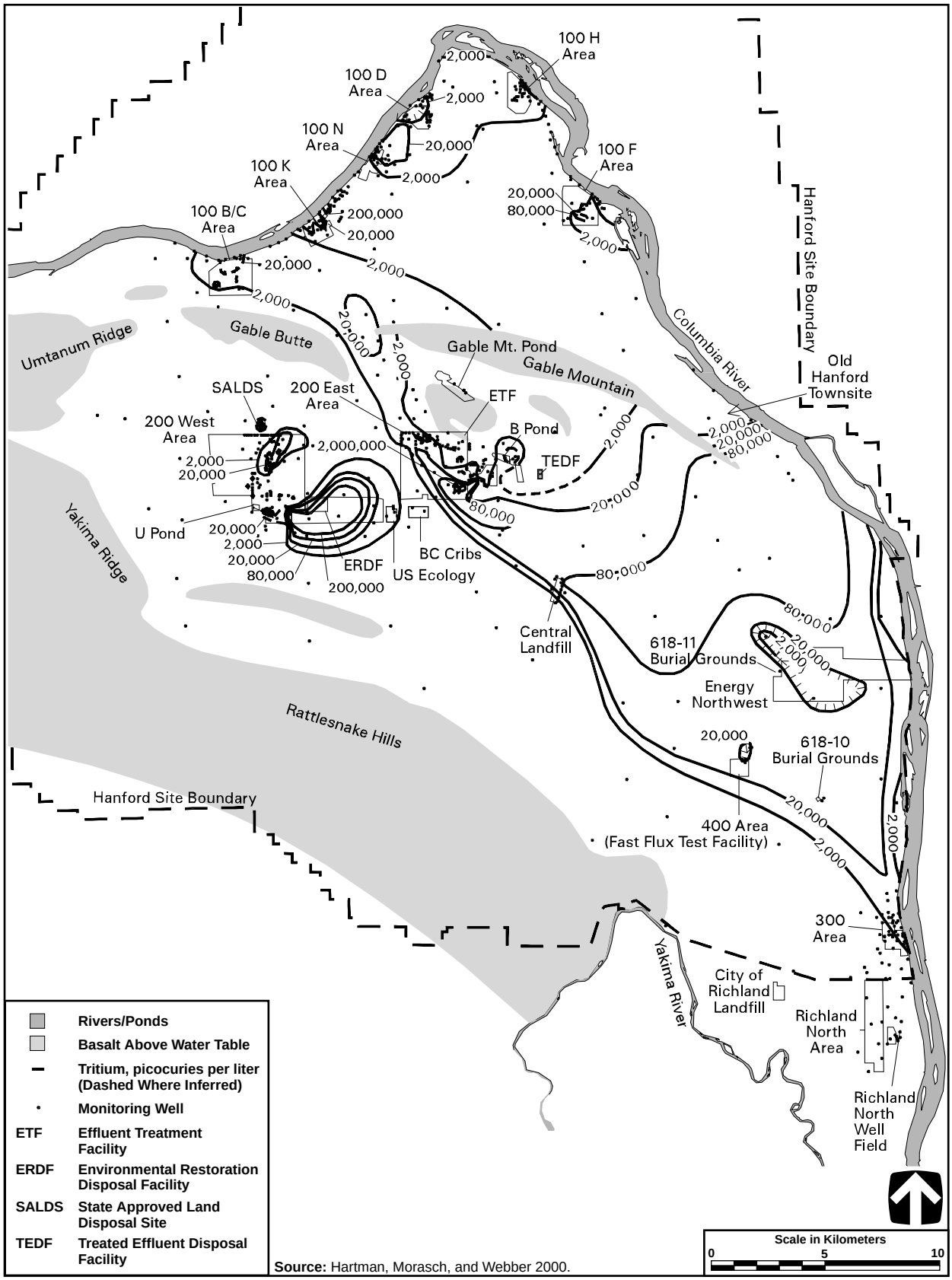
Water use in the Pasco Basin, which includes Hanford, is primarily via surface water diversion; groundwater accounts for less than 10 percent of water use (DOE 1999k:4-49). While most of the water used by Hanford is surface water withdrawn from the Columbia River, some groundwater is used. One of the principal users of groundwater was FFTF, which used about 697,000 liters (184,000 gallons) per day when it operated. In addition to the 400 Area, other facilities that use groundwater are the Yakima Barricade and the Patrol Training Academy (Barghusen and Feit 1995:2.2-22). DOE currently asserts an unlimited federally reserved groundwater withdrawal right with respect to existing Hanford operations, and withdraws about 197 million liters (52 million gallons) per year (DOE 1999e:3-32).

Groundwater quality beneath large portions of the Hanford Site has been affected by past liquid waste discharges, primarily to ditches, trenches, and ponds and from spills, injection wells, and leaks from waste storage tanks (Neitzel 1999:4.72). The unconfined aquifer system contains radiological and nonradiological contaminants at levels exceeding water quality criteria and standards. During fiscal year 1999 (October 1998 to September 1999), 623 wells were sampled for radiological and chemical constituents. Tritium and iodine-129 are the most widespread radiological contaminants in the unconfined aquifer system, with tritium exceeding the drinking water standard in the 100, 200, 400, and 600 Areas in 1998 and fiscal year 1999. Tritium levels are expected to decrease over time because of dispersion and radioactive decay. Nitrate, chromium, and carbon tetrachloride are the most widely distributed nonradiological contaminants (Dirkes, Hanf, and Poston 1999:6.27, 6.49; Hartman, Morasch, and Webber 2000:2.5–2.8). Tritium, iodine-129, and nitrate are the most widespread groundwater contaminants associated with Hanford legacy activities. Their distribution in the unconfined aquifer system are illustrated in **Figures 3–18, 3–19, and 3–20,** respectively. The figures also depict the locations of former waste management sites (e.g., Gable Mt. Pond, U Pond, B Pond, effluent disposal cribs) and burial grounds. Also shown are locations of active waste management and treatment facilities such as the State Approved Land Disposal Site, the Effluent Treatment Facility, the 200 Areas TEDF, and the Environmental Restoration Disposal Facility. Information on groundwater monitoring and chemical analysis is summarized in the annual site environmental report with detailed results in the site groundwater monitoring report (Dirkes, Hanf, and Poston 1999; Hartman, Morasch, and Webber 2000). Contamination in the confined aquifer system is typically limited to areas of exchange with the unconfined aquifer system (Dirkes, Hanf, and Poston 1999:6.65). No aquifers have been designated sole-source aquifers (Barghusen and Feit 1995:2.2-22).

3.4.4.2.2 Locations of Proposed Activities

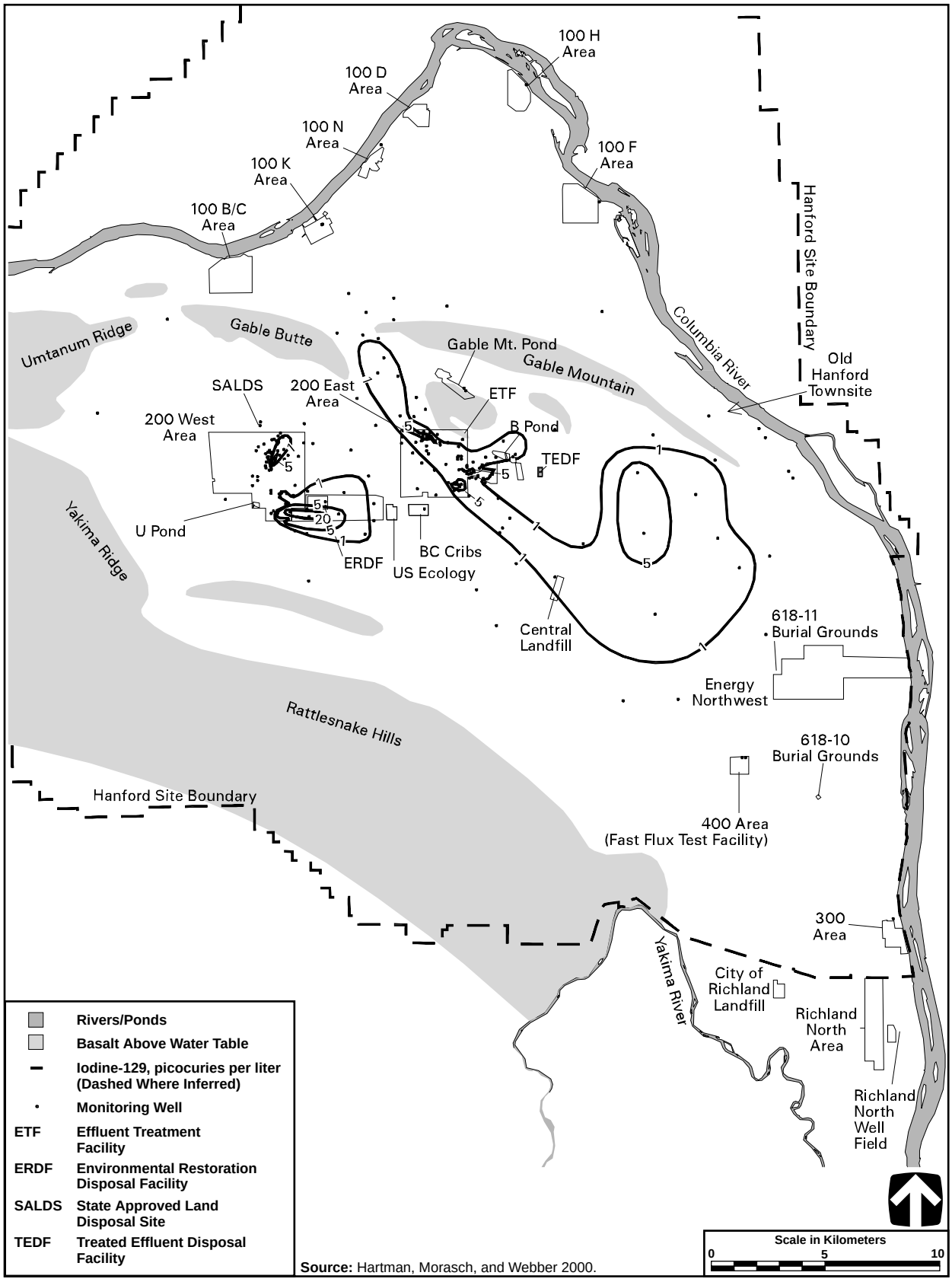
300 AREA

Groundwater flow direction and the water table in the unconfined aquifer system beneath the 300 Area are greatly affected by fluctuations in the level of the Columbia River. During low to average river level conditions, groundwater in the unconfined aquifer system converges beneath the 300 Area from the northwest and southwest and flows in a west to east or northwest to southeast direction, with discharge to the river. High river flows cause the water table to rise above the Hanford-Ringold formation contact and groundwater temporarily flows in a generally southwest to south direction. The unconfined aquifer system consists mainly of Hanford Formation gravels and sands, and Ringold Formation gravels and sands with varying amounts of



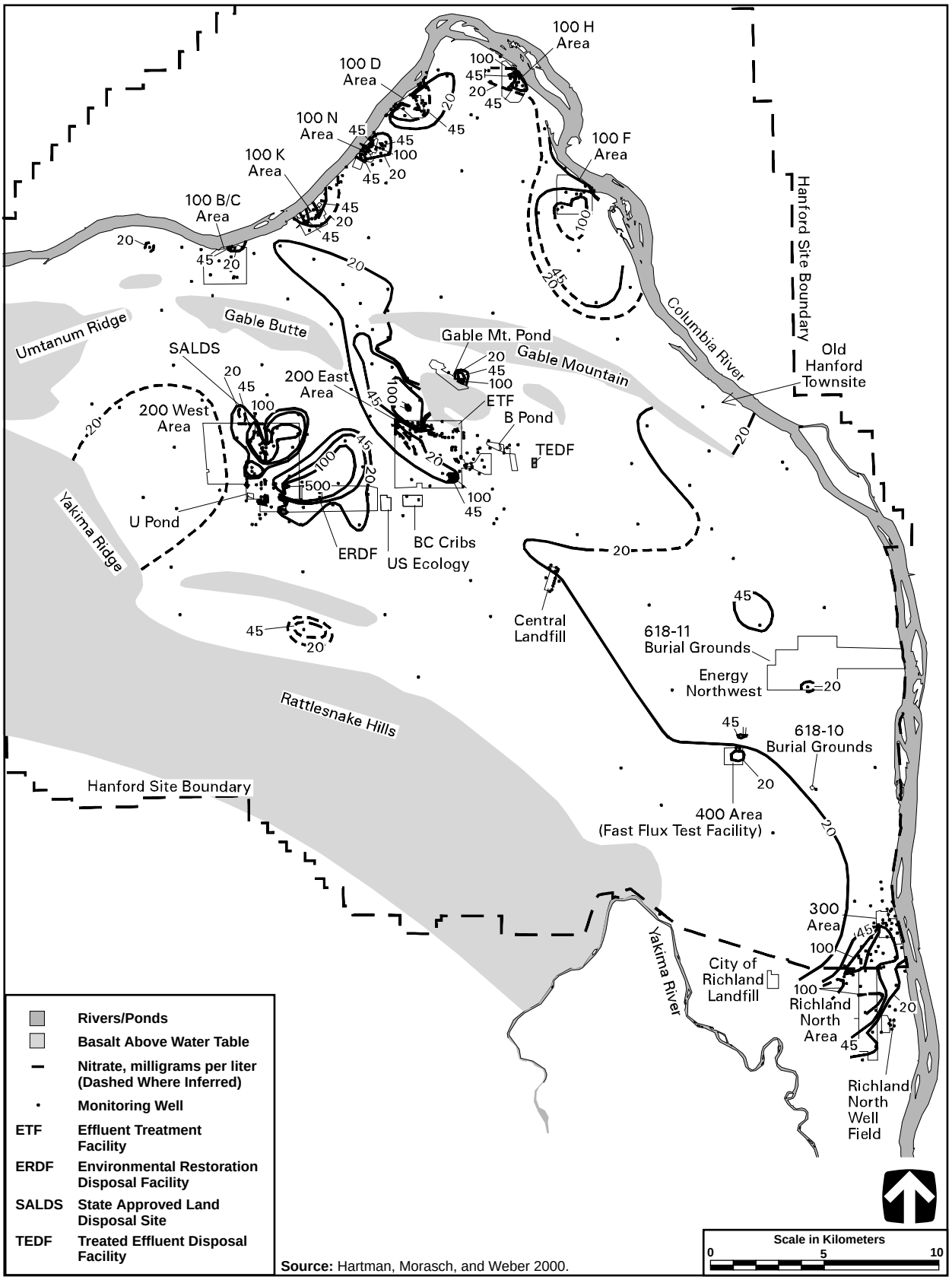
Source: Hartman, Morasch, and Webber 2000.

Figure 3–18 Average Tritium Concentrations on the Hanford Site, Top of Unconfined Aquifer



Source: Hartman, Morasch, and Webber 2000.

Figure 3–19 Average Iodine-129 Concentrations on the Hanford Site, Top of Unconfined Aquifer



Source: Hartman, Morasch, and Weber 2000.

Figure 3–20 Average Nitrate Concentrations on the Hanford Site, Top of Unconfined Aquifer

silt and clay. The water table lies within the Hanford Formation in most of the 300 Area. The depth to the water table beneath the 300 Area ranges from less than 1 meter (3 feet) near the Columbia River to 18 meters (59 feet) further inland (Hartman 2000:4.27, 4.28).

Groundwater quality in the 300 Area has primarily been affected by the uranium fuel fabrication facility and related cooling and sanitary wastewater discharges to the former 316-1 and 316-2 process ponds and subsequently to the 316-5 process trenches (Hartman 2000:4.28, 4.29). Uranium is the major contaminant of concern in the 300 Area with a plume in the upper unconfined aquifer system extending from the northeast and north-central portions of the 300 Area and south and east across the area to the Columbia River. In fiscal year 1999, uranium was detected at levels above the proposed drinking water maximum contaminant level (20 micrograms per liter) over much of the northeastern and eastern parts of the 300 Area, with a high of 322 micrograms per liter detected in one well (Hartman, Morasch, and Webber 2000:2.256, 2.260). Other groundwater contaminants detected at levels above their maximum contaminant levels (5 and 70 micrograms per liter, respectively) in the bottom of the unconfined aquifer system in the 300 Area during 1999 include trichloroethylene and cis-1,2-dichloroethylene in one well, with concentrations of 6 and 180 micrograms per liter, respectively. Tetrachloroethylene was detected above the maximum contaminant level (5 micrograms per liter) in one well in the upper part of the unconfined aquifer east and southeast of the 316-5 process trenches at a concentration of 7 micrograms per liter. Nitrate was above the maximum contaminant level (45 milligrams per liter) in two wells in the southern and southwestern portions of the 300 Area, with a maximum concentration of 110 milligrams per liter. This contaminant has been attributed to offsite industry and agriculture. The southward migrating tritium plume originating in the 200-East Area has also impacted the unconfined aquifer in the 300 Area, but with levels below the interim drinking water standard of 20,000 picocuries per liter (Hartman, Morasch, and Webber 2000:2.255, 2.257, 2.258, 2.265, 2.267, A-78).

400 AREA

Groundwater flow across the 400 Area is generally from west to east. The Hanford Formation immediately underlying the area consists mainly of the sand-dominated sediments. The water table is located near the contact between the Hanford and Ringold Formations, with the depth to the water table in the 400 Area ranging from about 45 to 50 meters (148 to 164 feet). Hanford Formation sediments dominate groundwater flow in the 400 Area because of their relatively high permeability, compared to that of the Ringold Formation sediments. The Ringold Formation consists of gravelly sands, sandy gravel, silty sands and fluvial gravels and overbank and lacustrine silt and clay. The saturated thickness of this aquifer system is about 140 meters (460 feet) (Hartman 2000:4.25).

The 400 Area receives its water from three supply wells, each with a pumping capacity of 833 liters (220 gallons) per minute (FDH 1999:3-4). One well (499-S1-8J) serves as the primary supply well for all 400 Area needs, including potable, process, and fire protection uses. The second and third wells (499-S0-8 and 499-S0-7) provide backup and emergency supply, respectively. Chlorination is the only water treatment provided to these wells (FDH 1999:4; Dirkes, Hanf, and Poston 1999:4.48, 4.49, 6.8). All of the wells are completed in the unconfined (Hanford/Ringold) aquifer system. The primary production well was installed in 1985 in the lower unconfined aquifer system after tritium contamination was detected in the original two wells, completed near the top of the aquifer (Hartman 2000:4.25). Water usage in the 400 Area ranges from about 284 to 681 liters (75 to 180 gallons) per minute on a seasonal basis. Water is stored in three aboveground storage tanks with a total capacity of about 3 million liters (800,000 gallons) (FDH 1999:4). Average annual groundwater use in the 400 Area is currently about 197 million liters (52 million gallons) (Nielsen 1999:41).

Nitrate is the only significant contaminant attributable to 400 Area operations. Elevated nitrate concentrations in excess of the drinking water maximum contaminant level have been attributed to the former sanitary sewage

lagoon located west and upgradient of the 400 Area process ponds. The maximum concentration in fiscal year 1999 was 92 milligrams per liter; the maximum contaminant level is 45 milligrams per liter. As disposal to the lagoon has been discontinued and the lagoon backfilled, nitrate contamination from this source should diminish with time. Elevated levels of tritium in 400 Area wells continued in 1999 and are associated with the groundwater plume from the vicinity of the Plutonium Uranium Extraction Plant in the 200-East Area. Tritium was found at levels at or above the interim drinking water standard (20,000 picocuries per liter) in samples from the 400 Area backup supply wells (wells 499-S0-7 and 499-S0-8). The maximum in the backup water supply in fiscal year 1999 was 68,400 picocuries per liter (Hartman, Morasch, and Webber 2000:2.8, 2.235, 2.236). All samples collected from the primary supply well (499-S1-8J) were below the drinking water standard for tritium. Tritium activities were also below the drinking water standard, and the 4-millirem-per-year dose equivalent in the drinking water supply (sampled at the tap), for all sampling events in fiscal year 1998. Nitrate levels also remained below the maximum contaminant level in fiscal year 1999 for the water-supply wells. Fiscal year 1999 and past data from 400 Area and surrounding wells indicates no other constituents are present at levels above drinking water maximum contaminant levels (Hartman, Morasch, and Webber 2000:2.236).

One recent finding based on groundwater monitoring for nearby areas is particularly noteworthy with regard to tritium concentrations near the 400 Area. In January 1999, a sample from well 699-13-3A, located along the eastern (downgradient) fence line of the 618-11 burial ground and about 4 kilometers (2.5 miles) southeast of the 400 Area, contained 1.86 million picocuries per liter of tritium. The result was confirmed via reanalysis and represents the first time that tritium has been detected in this well. This value is also much higher than data from the surrounding wells. A January 2000 sample contained 8.1 million picocuries per liter of tritium. This is the highest concentration of tritium detected onsite in recent years. A special investigation of the groundwater at the 618-11 burial ground was to be undertaken in fiscal year 2000 to determine the source of the high tritium results (Hartman, Morasch, and Webber 2000:2.246). The results should be available in time to be published in the fiscal year 2000 groundwater monitoring report.

3.4.5 Geology and Soils

Geologic resources are consolidated or unconsolidated earth materials, including ore and aggregate materials, fossil fuels, and significant landforms. Soil resources are the loose surface materials of the earth in which plants grow, usually consisting of disintegrated rock, organic matter, and soluble salts.

3.4.5.1 General Site Description

Hanford lies within the Pasco Basin of the Columbia Plateau that is encompassed by the Columbia Intermontane physiographic province (Barghusen and Feit 1995:2.2-12). The rocks beneath Hanford consist of Miocene age (5 to 24 million years old) and younger rocks that overlay older Cenozoic sedimentary and volcanic basement rocks. The major geologic units underlying Hanford are, in ascending order: subbasalt (basement) rocks; the Columbia River Basalt Group; and the Ringold Formation, the Plio-Pleistocene unit, early "Palouse" soil, and the Hanford Formation, collectively known as the Suprabasalt Sediments (Figures 3–16 and 3–17).

The Columbia River Basalt Group consists of sequences of continental flood basalts of Miocene age that cover an extensive area across Washington, Oregon, and Idaho. Nearly all of the flood basalts were erupted in a span dating from approximately 14.5 to 17 million years ago. Volcaniclastic (volcanic-sedimentary) and fluvial (stream deposited) sedimentary materials of the Ellensburg Formation are interbedded within the group. Airfall tuff (consolidated volcanic ash) is the dominant volcaniclastic material at the Hanford Site. The Ringold Formation is exposed in the White Bluffs east of the Columbia River on the site and consists of sedimentary deposits dominated by fluvial gravel and sand deposits along with lake-deposited sand, silt, and

clay. The Plio-Pleistocene unit is locally derived, consisting of alluvium, colluvium, and/or calcium-cemented soil material (caliche). Wind-deposited sand and silt characterizes the early "Palouse" soil. This unit occurs in the western portion of the site. Because it is hard to distinguish from overlying and underlying units, it is generally grouped together with the Plio-Pleistocene unit. Gravel, sand, and silt deposits, comprising the unit informally designated as the Hanford Formation, are the products of cataclysmic floods that inundated the Pasco Basin and the Hanford Site during the Pleistocene between about 13,000 and 1 million years ago. Younger surficial materials also include alluvium deposited by streams and rivers, as well as active sand dune fields (i.e., north of Energy Northwest) (DOE 1999k:4.12–4-22; Hartman 2000:3.1–3.4; Neitzel 1999:4.35–4.44).

Basalt outcrops are exposed on ridges at Gable Mountain, Gable Butte, and the Saddle Mountains in the northern part of Hanford, and on Rattlesnake Hills and Yakima Ridge, overlapping the western and southwestern edges of the site. Other than crushed rock, sand, and gravel, no economically viable geologic resources have been identified at Hanford (DOE 1999e:3-24).

Known faults in the Hanford area include those on Gable Mountain and the Rattlesnake-Wallula alignment. The faults in Central Gable Mountain are considered capable, although there is no observed seismicity on or near Gable Mountain. The Rattlesnake-Wallula alignment is interpreted as possibly being capable because there appear to be active portions of the fault system 56 kilometers (35 miles) southwest of the central part of Hanford (Barghusen and Feit 1995:2.2-13, 2.2-14). A capable fault is one that has had movement at or near the ground surface at least once within the past 35,000 years, or recurrent movement within the past 500,000 years (10 CFR Part 100, Appendix A).

Seismicity of the Columbia Plateau, as determined by the rate of earthquakes per area and the historical magnitude of these events, is lower than that of other regions in the Pacific Northwest. The two largest earthquakes near Hanford occurred in 1918 and 1973; each had an approximate Richter magnitude of 4.4 and a Modified Mercalli Intensity of V (Table 3–3). They occurred in the central portion of the Columbia Plateau north of Hanford, with the December 20, 1973, event epicentered approximately 22 kilometers (14 miles) northwest of the Hanford Site boundary (Neitzel 1999:4.52; USGS 2000c). There have been 45 small earthquakes (ranging in magnitude from 2.5 to 3.9) recorded within a radius of 90 kilometers (56 miles) of the Hanford Site 400 Area since the 1973 earthquake. The closest of these was a magnitude 3.3 event that occurred on June 12, 1995, and had an epicenter about 8 kilometers (5 miles) southeast of the 400 Area (Chapin 2000; USGS 2000c). Based on the most recent seismic analyses, an earthquake with a maximum horizontal acceleration of 0.2g is calculated to have an annual probability of occurrence of 1 in 2,500 at Hanford (Neitzel 1999:4.55). While evidence has been mounting since at least the early 1990s that great earthquakes, with a magnitude of 8 to 9, have occurred in the past in association with the Cascadia Subduction Zone off the coast of the Pacific Northwest, the increased risk is primarily to Western Washington (USGS 1995).

As discussed in more detail in Section 3.2.5.1, USGS has developed new seismic hazard maps as part of the National Seismic Hazard Mapping Project that are based on response spectral acceleration. These maps have been adapted for use in the new *International Building Code* (ICC 2000) (Figures 1615 (1) and 1615(2) in the code) and depict maximum considered earthquake ground motion of 0.2- and 1.0-second spectral response acceleration, respectively, based on a 2 percent probability of exceedance in 50 years. Hanford lies within the 0.40g to 0.50g mapping contours for a 0.2-second spectral response acceleration and the 0.10g to 0.15g contours for a 1.0-second spectral response acceleration.

There is some potential for slope failure at Hanford, although only the slopes of Gable Mountain and White Bluffs are steep enough to warrant landslide concern. White Bluffs, east of the Columbia River, poses the greatest concern. This risk is in part attributable to the largely unconsolidated and uncemented nature of the

Ringold sediments comprising much of the bluffs, the discharge of irrigation water atop the bluffs and subsequent percolation through the sediments, and the general dip of the sediments toward the Columbia River (DOE 1999k:4-18, 4-21; Neitzel 1999:4.43). A large landslide along White Bluffs could fill the Columbia River channel and divert water onto Hanford. Calculations of the potential impacts of such a landslide indicate a flood area similar to the probable maximum flood (Neitzel 1999:4.64, 4.65).

Several major volcanoes are in the Cascade Range west of Hanford, including Mount Adams and Mount St. Helens, located 165 kilometers (102 miles) and 220 kilometers (137 miles) from the site, respectively. Ashfalls from at least three Cascade volcanoes have blanketed the central Columbia Plateau since the late Pleistocene epoch. Generally, ashfall layers have not exceeded more than a few centimeters (less than 1.5 inches) in thickness, with the exception of the Mount Mazama (Crater Lake, Oregon) eruption, when as much as 10 centimeters (3.9 inches) of ash fell over western Washington (Barghusen and Feit 1995:2.2-14).

Fifteen different soil types occur at Hanford. These soils vary from sand to silty and sandy loam. The dominant soil types are the Quincy (Rupert) sand, Burbank loamy sand, Ephrata sandy loam, and the Warden silt loam. No soils at Hanford are currently classified as prime farmlands because there are no current soil surveys, and the only prime farmland soils in the region are irrigated. The Quincy (Rupert) sand is the most widespread soil type at Hanford, but particularly encompasses much of the southeast and east-central portions of the site (south and east of the 200 Areas). It developed from sandy alluvial deposits mantled by wind-blown sand. Burbank loamy sand soils mainly occur north of the 200 Areas and south of the Columbia River along with Ephrata sandy loams. Both soils are underlain by gravelly material. The Warden silt loam occurs in a broad band in the south and southwestern portions of the site, running from the south boundary of the site and downslope of Rattlesnake Mountain (DOE 1999k:4.23–4.27; Neitzel 1999:4.48–4.51). More detailed descriptions of the geology and the soil conditions at Hanford are included in the *Hanford Site NEPA Characterization* (Neitzel 1999) and the *Hanford Comprehensive Land-Use Plan Environmental Impact Statement* (DOE 1999k).

3.4.5.2 Locations of Proposed Activities

300 AREA

The Central Gable Mountain fault is the nearest capable fault to the 300 Area and is located 28 kilometers (17 miles) away (DOE 1999k:4-19; Mecca 1997a:6, 78, 79). The surficial stratigraphy of the 300 Area is dominated by the gravel and sands of the Hanford Formation that overlie the sediments of the Ringold Formation. Total thickness of these units is approximately 52 meters (171 feet) (Hartman 2000:4.27, 4.28; Neitzel 1999:4.45). The predominant soil type is the Quincy (Rupert) sand, and the soils and surface sediments are not subject to liquefaction or other instabilities (Mecca 1997a:6; Neitzel 1999:4.49).

400 AREA

The nearest capable fault to the 400 Area (Central Gable Mountain fault) is 19 kilometers (12 miles) away (Mecca 1997a:6, 78, 79). 400 Area stratigraphy consists of sand-dominated sediments of the Hanford Formation which attain a thickness of about 50 meters (164 feet) beneath the site. Locally, surface sediments also consist of stabilized sands deposited in dune fields (Hartman 2000:4.25; Neitzel 1999:4.44). The predominant soil type in the 400 Area is the Quincy (Rupert) sand, and the soils and surface sediments are not subject to liquefaction or other instabilities (Mecca 1997a:6; Neitzel 1999:4.49).

3.4.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species. Material presented in this section, unless otherwise noted, is from the *Storage and Disposition PEIS* (DOE 1996b:3-83-3-87).

3.4.6.1 Terrestrial Resources

This section addresses the plant and animal communities of Hanford and includes a plant community map of the site. Terrestrial resources are described for the site as a whole, as well as for the proposed facility locations.

3.4.6.1.1 General Site Description

Vegetation at Hanford has been characterized as shrub-steppe. Shrub-steppe ecosystems are typically dominated by a shrub overstory with a grass understory. Present site development consists of clusters of large buildings that are found at widely spaced locations. Developed areas encompass about 6 percent of the site. The remaining areas of the site can be divided into 10 major plant communities (**Figure 3–21**). Hanford is dominated by plant communities in which big sagebrush is a major component. Other plant communities contain a variety of grasses and herbaceous plants. Areas previously disturbed by agricultural activities are dominated by nonnative species, such as cheatgrass. Trees are uncommon on the site, but those that are present include cottonwood and willow, which are both found near water bodies, and a few other deciduous species which were originally planted near farmsteads as windbreaks. Five hundred ninety species of plants have been identified at Hanford (Neitzel 1999).

Unique habitats on the Hanford Site include bluffs, dunes, and islands within the Columbia River. The White Bluffs, Umtanum Ridge, and Gable Mountain include rock outcrops that occur infrequently on the site. Vegetation on basalt outcrops includes snow buckwheat and Sandberg's bluegrass. The terrain of the dune habitat rises and falls between 3 and 5 meters (10 and 16 feet). The dune are vegetated by bitterbrush, dune scurfpea, and thickspike wheatgrass. Riparian vegetation that characterizes the islands of the Columbia River includes willow, white mulberry, snow buckwheat, lupine, yarrow, and thickspike wheatgrass among others (Neitzel 1999).

Hanford provides suitable habitat for numerous animal species, including over 1,500 species of insects, 4 species of amphibians, 9 species of reptiles, 246 species of birds, and 40 species of mammals. Grasshoppers and darkling beetles are among the more conspicuous groups of insects, and along with other insects, are an important food source for local birds and mammals (Neitzel 1999:4.87). Common animal species at Hanford include the side-blotched lizard, gopher snake, western meadowlark, horned lark, Great Basin pocket mouse, black-tailed jackrabbit, and mule deer. Trees planted around former farmsteads serve as nesting platforms for several species of birds, including hawks, owls, ravens, magpies, and great blue herons; these trees also serve as night roosts for bald eagles. The Hanford Reach of the Columbia River, including several sparsely vegetated islands, provides nesting habitat for the Canada goose, ring-billed gull, Forester's tern, and great blue heron. Several game animals are found at Hanford. Hunting is permitted on site north of the Columbia River. Numerous raptors, such as the Swainson's hawk and red-tailed hawk, and carnivores, such as the coyote and bobcat, are found on Hanford. A variety of migratory birds have been found at Hanford.

Unique habitats on the Hanford Site provide habitat for a number of species of wildlife. Bluff areas provide nesting habitat for prairie falcons, red-tailed hawks, and several species of swallows and roosting habitat for bald eagles. Mule deer, burrowing owls, and coyotes, as well as many transient species, may be found in dune habitat. Islands in the Columbia River provide nesting habitat for Canada geese, California gulls, ring-billed gulls and Forster's tern (Neitzel 1999).

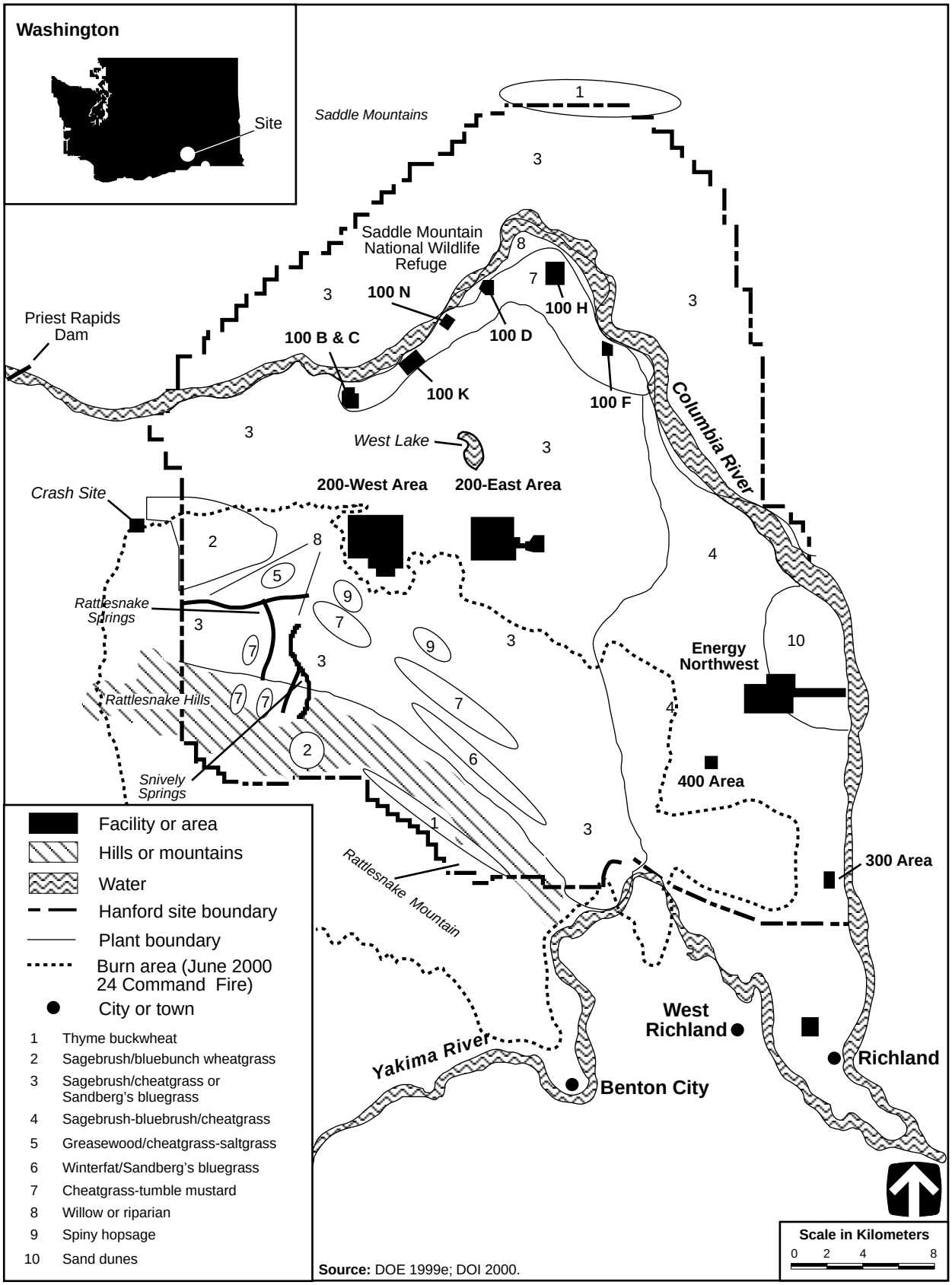


Figure 3–21 Distribution of Plant Communities at the Hanford Site

On June 27, 2000, a fire known as the 24 Command Fire was started by a fatal motor vehicle accident on State Route 24, about 2 miles west of the State Route 240 intersection. As a result of high winds and temperatures and low humidity, the fire spread rapidly and eventually consumed 66,322 hectares (163,884 acres) of Federal, state, and private lands. A total of 24,384 hectares (60,254 acres) within Hanford burned, including lands within the Hanford Reach National Monument, most of the Arid Lands Ecology Reserve, and areas near former production sites (Figures 3–12 and 3–21). The fire was declared controlled on July 2, 2000 (DOI 2000).

The USFWS has prepared a *Burned Area Emergency Rehabilitation Plan* in which resource issues and impacts were assessed and recommendations outlined (DOI 2000). Due to the extremely dry conditions and high winds, vegetation resources were significantly reduced on about 85 percent of the fire area. However, because of the relatively fast passage of the fire over any one area, soils showed little damage and seed bank sources in the soil were not adversely impacted. While this will aid natural vegetation, recovery of some plant associations (e.g., sagebrush) may require planting and could take years. Plant associations extensively affected by the fire include those containing big sagebrush, bluebunch wheatgrass, and three-tipped sage. Some riparian vegetation was also impacted. Fire suppression tactics, including construction of firebreaks and backfire operations, also adversely affected plant communities. Potential long-term impacts from the fire include the establishment of noxious weeds and changes in natural plant communities.

The 24 Command Fire had immediate direct impacts on wildlife, including loss of individual animals, especially smaller less mobile species and young of the year, as well as displacement of more mobile animals to unaffected areas. However, displacement itself can lead to an increase in mortality due to road kills, and in the case of elk, this has already occurred. Additionally, long-term impacts to wildlife due to loss of food, cover, and breeding habitat are expected as a result of the fire (DOI 2000).

3.4.6.1.2 Locations of Proposed Activities

300 AREA

While the 300 Area is located within the big sagebrush/bunchgrasses–cheatgrass vegetation community, it is heavily developed (DOE 1999k). Vegetation within the 300 Area is characteristic of disturbed areas consisting of sparse amounts of cheatgrass and Russian thistle (Nielsen 2000). Due to the disturbed nature of most of the 300 Area, wildlife use of developed portions of the areas is limited.

400 AREA

The 400 Area is located within postfire shrub-steppe habitat dominated by cheatgrass and small shrubs, including gray and green rabbitbrush. Due to past disturbances and human occupancy in the 400 Area, wildlife is limited. Several animal species may be found in the area, including the gopher snake, northern Pacific rattlesnake, burrowing owl, Swainson's hawk, western meadowlark, black-tailed jackrabbit, and Great Basin pocket mouse (DOE 1999e:3-35).

3.4.6.2 Wetlands

Wetlands include "those areas that are inundated or saturated by surface or groundwater at a frequency and duration sufficient to support, and that under normal circumstances do support, a prevalence of vegetation typically adapted for life in saturated soil conditions" (33 CFR Section 328.3). Wetlands are described for Hanford as a whole, as well as for the proposed facility locations.

3.4.6.2.1 General Site Description

Primary wetland areas at Hanford are found in the riparian zone along the Columbia River. The extent of this zone varies, but includes large stands of willows, grasses, and other plants. This area has been extensively affected by hydropower operations at Priest Rapids Dam (Neitzel 1999).

Other large areas of wetlands at Hanford can be found north of the Columbia River within the Saddle Mountain National Wildlife Refuge and the Wahluke Wildlife Unit Columbia Basin Area. These two areas encompass all the lands extending from the north bank of the Columbia River northward to the site boundary and east of the Columbia River down to Ringold Springs. Wetland habitat in these areas consists of fairly large ponds resulting from irrigation runoff. These ponds have extensive stands of cattails and other emergent aquatic vegetation surrounding the open water regions. They are extensively used as nesting sites by waterfowl (Neitzel 1999).

On the western side of Hanford, Rattlesnake Springs supports a riparian zone of 3.0 kilometers (1.9 miles) in length, featuring watercress, bulrush, spike rush, cattail and peachleaf willow. Snively Springs also contains a diverse biotic community similar to Rattlesnake Springs (Neitzel 1999). The 24 Command Fire affected approximately 18 hectares (44 acres) of willow riparian habitat, including areas around Rattlesnake Spring, Snively Canyon, Benson Springs, and the Yakima River (DOI 2000). Several semi-permanent artificial ponds and ditches that receive cooling water or irrigation wastewater are also present on Hanford. These waterbodies provide a source of water for terrestrial animals (Neitzel 1999).

3.4.6.2.2 Locations of Proposed Activities

300 AREA

The 300 Area is bounded on the eastern side by the Columbia River. The riparian zone bordering the river is the only wetland area associated with the site (Nielsen 2000). The general nature of this zone is discussed in Section 3.4.6.2.1.

400 AREA

There are no natural wetlands in the 400 Area, although a small cooling and wastewater pond does contain some wetland vegetation. Wildlife species observed using the site include a variety of mammals and waterfowl (DOE 1999e:3-36).

3.4.6.3 Aquatic Resources

Aquatic resources at Hanford are described for the site as a whole, as well as for the proposed facility locations.

3.4.6.3.1 General Site Description

Aquatic resources on Hanford include the Columbia River, ephemeral streams, springs, surface ponds, and ditches. The Columbia River flows along the northern and eastern edges of the site. The Hanford Reach supports 44 anadromous and resident species of fish. Many of the fish species present in the Hanford Reach are dependent upon flowing water and rocky substrate for at least part of their life cycles. Fall chinook salmon, steelhead trout, mountain whitefish, and smallmouth bass spawn in this area. The destruction of other mainstream Columbia River spawning areas by dams has increased the relative importance of the Hanford Reach for spawning (Neitzel 1999).

The Hanford Reach provides a migration route to upstream spawning areas for spring, summer, and fall adult chinook salmon, coho salmon, sockeye salmon, and steelhead trout. It also provides rearing habitat for the salmonid juveniles in their downstream migration to the sea. Principal resident fish species sought by anglers in the Hanford Reach include mountain whitefish, white sturgeon, smallmouth bass, crappie, catfish, walleye, and yellow perch (Neitzel 1999).

The Yakima River borders the southern portion of Hanford. Game fish found in the river in the vicinity of the site are smallmouth bass, steelhead trout, and channel catfish. Cold Creek and its tributary, Dry Creek, are ephemeral streams within the Yakima River drainage system along the southern boundary of Hanford. These streams do not support any fish populations.

There are several springs at Hanford. Rattlesnake Springs and Snively Springs, which are in the western portion of the site, form short streams that seep into the ground. None of the springs support any fish populations.

3.4.6.3.2 Locations of Proposed Activities

300 AREA

The 300 Area is immediately to the west of the Columbia River. There are no aquatic resources on the site itself (Nielsen 2000).

400 AREA

Although no natural aquatic habitat occurs in the 400 Area, a small cooling and wastewater pond is present (DOE 1999e:3-36). The 400 Area is 6.8 kilometers (4.2 miles) west of the Columbia River.

3.4.6.4 Threatened and Endangered Species

Endangered species are those plants and animals in danger of extinction throughout all or a large portion of their range. Threatened species are those species likely to become endangered within the foreseeable future. Threatened and endangered species are described for Hanford as a whole, as well as for the proposed facility locations.

3.4.6.4.1 General Site Description

Eighty-one Federal- and state-listed threatened, endangered, and other special status species may be found on Hanford. These are listed in Tables 4–6 and 4–7 of the *Hanford Comprehensive Land-Use Plan Environmental Impact Statement* (DOE 1999k). Nineteen of these are Federal- or state-listed as threatened or endangered, while the remainder are listed by the state within one of several special status classifications.

The threatened bald eagle, which has been proposed to be delisted, is the only federally listed species known to be found regularly at Hanford, although there are occasional sitings of the threatened Aleutian Canada goose. The bald eagle, which is also listed by the state as endangered, is a regular winter resident along the Hanford Reach where it forages for salmon and waterfowl. Trees in the historic Hanford Townsite area are used by eagles for perching. Recently, eagles have attempted to nest on the site. The peregrine falcon, listed as endangered by the state, is a migrant in the Hanford area (Dirkes and Hanf 1997:F.1; DOE 1996b:3-44; Neitzel 1999). The Upper Columbia River run of steelhead and Upper Columbia River spring run of Chinook salmon are listed as endangered and the Middle Columbia River run of steelhead are listed as threatened by the Federal government. Spring-run chinook salmon do not spawn in the Hanford Reach but use it as a

migration corridor. Little is known about the quality and quantity of steelhead spawning, rearing, and adult holding habitat in the Hanford Reach and Upper Columbia River (DOE 1999k). Recently, the Hanford Reach has been designated as critical habitat for Upper Columbia River spring-run chinook salmon and Upper Columbia River steelhead (65 FR 7764). Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service and National Marine Fisheries Service. Consultation was also conducted with the state. The results of these consultations are presented in Chapter 4.

The 24 Command Fire burned 24,384 hectares (60,254 acres) of Hanford resulting in potential impacts to a number of threatened, endangered, or other special status species. A total of 9 plant and 12 animal special status species could potentially be found in the burn area (DOI 2000). A post-fire survey determined that suitable habitat for the threatened Ute ladies'-tresses, the only federally listed plant species, did not exist in the burn area. The fire could have directly or indirectly affected seven state-listed plants. Direct effects could include loss of plants and seed stock. Indirect effects could include adverse impacts such as competition from invasive plant species, potential loss of soil productivity due to wind erosion, and loss of seed viability. Indirect effects could also be of a beneficial nature and include release of nutrients back into the soil and reduction in competition for soil nutrients, sun, and soil moisture. With respect to wildlife, the 24 Command Fire was determined to have had no effect on any federally listed species. Potential direct impacts to state-listed species include direct loss of adults and young, while indirect effects include loss of habitat used for feeding, cover, and raising young. Monitoring special status species will be necessary to determine the exact nature and extent to which plants and animals were impacted by the fire.

3.4.6.4.2 Locations of Proposed Activities

300 AREA

A survey of the 300 Area made in conjunction with an environmental assessment of RPL did not locate any Federal- or state-listed threatened or endangered plant or animal species within the site (DOE 1997d). However, more recently, the peregrine falcon and bald eagle have been observed in the area (Nielsen 2000).

400 AREA

No Federal- or state-listed threatened or endangered plants or animals reside in the vicinity of the 400 Area (DOE 1999e), although potential exists for the incidental occurrence of some migratory species, such as the peregrine falcon. State sensitive plant species have not been found in the 400 Area, although Piper's daisy does occur in the vicinity. A fire also burned the area in the mid 1980s, leaving it dominated by cheatgrass and some small shrubs (Mecca 1997b; Schinner 1999).

3.4.7 Cultural and Paleontological Resources

Cultural resources are human imprints on the landscape and are defined and protected by a series of Federal laws, regulations, and guidelines. The three general categories of cultural resources addressed in this section are prehistoric, historic, and Native American. Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age, and may be sources of information on paleoenvironments and the evolutionary development of plants and animals.

Hanford has a well-documented record of cultural and paleontological resources. The *Hanford Cultural Resources Management Plan* (Battelle 1989), establishes guidance for identifying, evaluating, recording, curating, and managing these resources. There are approximately 930 cultural resource sites and isolated finds recorded (Neitzel 1999:4.104). Forty-eight archaeological sites and one building are included on the National Register of Historic Places. Nominations have been prepared for several archaeological districts and sites

considered to be eligible for listing on the National Register. While many significant cultural resources have been identified, only about 6 percent of the Hanford Site has been surveyed, and few of the known sites have been evaluated for their eligibility for listing on the National Register. Cultural resource reviews are conducted whenever projects are proposed in previously unsurveyed areas. In recent years, reviews have exceeded 500 per year.

Cultural sites are often occupied continuously or intermittently over substantial time spans. For this reason, a single location may contain evidence of use during both historic and prehistoric periods. In the discussions that follow, the numbers of prehistoric and historic resources are presented. The sum of these resources may be greater than the total number of sites reported due to this dual-use history at sites. Therefore, where the total number of sites reported is less than the sum of prehistoric and historic sites, certain locations were used during both periods.

The 24 Command Fire burned 24,384 hectares (60,254 acres) of Hanford, resulting in potential impacts to cultural resources. A preliminary assessment of possible effects to cultural resources determined that a minimum of 190 previously recorded historic and prehistoric archaeological sites may have been affected (DOI 2000). These sites range from lithic scatters to can scatters, Indian hunting sites to ranch buildings, and spirit quest monuments to gas production wells. The preliminary assessment found that wooden structures (e.g., a corral) were destroyed, but that other surface and subsurface artifacts such as glass and lithic debris were not severely altered by the fire. Post-fire surface visibility has been greatly enhanced, which presents opportunities for archaeologists and historians to refine the boundaries of known sites and locate new sites, but also increases the potential for looting and vandalism.

3.4.7.1 Prehistoric Resources

Prehistoric resources are physical properties that remain from human activities that predate written records.

3.4.7.1.1 General Site Description

About 365 prehistoric archaeological sites and isolated finds have been recorded on Hanford. Of 48 sites included on the National Register of Historic Places, two are individual sites (Hanford Island Site and Paris Site), and the remainder are located in seven archaeological districts. In addition, four other archaeological districts have been nominated or are planned to be nominated for the National Register. A number of sites have been identified along the Middle Columbia River and in inland areas away from the river, but near other water sources. Some evidence of human occupation has been found in the arid lowlands. Sites include remains of numerous pithouse villages, various types of open campsites, graves along the riverbanks, spirit quest monuments (rock cairns), hunting camps, game drive complexes, quarries in mountains and rocky bluffs, hunting and kill sites in lowland stabilized dunes, and small temporary camps near perennial sources of water away from the river (Neitzel 1999).

More than 10,000 years of prehistoric human activity in the largely arid environment of the Middle Columbia River region have left extensive archaeological deposits. Archaeological surveys have been conducted at Hanford since 1926; however, little excavation has been conducted at any of the sites. Surveys have included studies of Gable Mountain, Gable Butte, Snively Canyon, Rattlesnake Mountain, Rattlesnake Springs, and a portion of the Basalt Waste Isolation Project Reference Repository location. Most of the surveys have focused on islands and on a 400-meter (1,312-foot) wide area on either side of the river. From 1991 through 1993, the 100 Areas were surveyed, and new sites were identified. Excavations have been conducted at several sites on the riverbanks and islands and at two unnamed sites. Test excavations have been conducted at the Wahluke, Vernita Bridge, and Tsulim sites and at other sites in Benton County (Neitzel 1999).

3.4.7.1.2 Locations of Proposed Activities

300 AREA

Much of the 300 Area has been highly disturbed by industrial activities and is unlikely to contain intact prehistoric sites (Neitzel 1999). The *Hanford Cultural Resources Management Plan* (Battelle 1989) provides for survey work before construction, and has contingency guidelines for handling the discovery of previously unknown archaeological resources encountered during construction.

400 AREA

Most of the 400 Area has been highly disturbed and is unlikely to contain intact prehistoric sites. A cultural resources survey found only 12 hectares (30 acres) that were undisturbed, and no sites were identified either within the 400 Area or within 2 kilometers (1.2 miles) of the 400 Area. The *Hanford Cultural Resources Management Plan* (Battelle 1989) provides for survey work before construction, and has contingency guidelines for handling the discovery of previously unknown archaeological resources encountered during construction.

3.4.7.2 Historic Resources

Historic resources consist of physical properties that postdate the existence of written records. In the United States, historic resources are generally considered to be those that date no earlier than 1492.

3.4.7.2.1 General Site Description

Five hundred seventeen historic archaeologic sites associated with the pre-Hanford Site and the Cold War eras, including an assortment of farmstead, corrals, dumps, and military sites, have been recorded since 1977 (Neitzel 1999). Of these sites, one is included on the National Register of Historic Places as a historic site, and 56 are listed as archaeological sites. Sites and localities that predate the Hanford era include homesteads, ranches, trash scatters, dumps, gold mine tailings, roads, and townsites, including the Hanford townsite and the East White Bluffs townsite and ferry landing. More recent historic structures include the defense reactors and associated materials-processing facilities that played an important role in the Manhattan Project and the Cold War era. A Programmatic Agreement for the maintenance, deactivation, alteration, and demolition of the built environment on Hanford has been reached between DOE, the Advisory Council on Historic Preservation, and the Washington State Historic Preservation Office (DOE 1996d).

Lewis and Clark were some of the first European Americans to visit the Hanford region during their 1804 to 1806 expedition. They were followed by fur trappers, military units, and miners. It was not until the 1860s that merchants set up stores, a freight depot, and the White Bluffs Ferry on the Hanford Reach, and Chinese gold miners began to work the gravel bars. Cattle ranches opened in the 1880s, and farmers soon followed. Several small thriving towns, including Hanford, White Bluffs, and Ringold, grew up along the riverbanks in the early 20th century. Other ferries were established at Wahluke and Richland. These towns, and nearly all other structures, were razed after the U.S. Government acquired the land for the original Hanford Engineer Works in the early 1940s (part of the Manhattan Project). Plutonium produced at the 100 B-Reactor was used in the first nuclear explosion at the White Sands Missile Range in New Mexico, and later in the bomb that destroyed Nagasaki, Japan, to help end World War II. The Hanford 100 B-Reactor is listed on the National Register and is designated a National Mechanical Engineering Landmark, a National Historic Civil Engineering Landmark, and a National Nuclear Engineering Landmark. Consultation to comply with Section 106 of the National Historic Preservation Act was conducted with the State Historic Preservation Office. The results of this consultation are presented in Chapter 4.

3.4.7.2.2 Locations of Proposed Activities

300 AREA

The 300 Area has been highly disturbed by industrial activities. Five recorded archaeological sites, including campsites, housepits, and historic trash scatter, are located at least partially within the 300 Area; many more may by located in subsurface deposits. Twenty-seven archaeological sites and 13 isolated artifacts have been recorded within 2 kilometers (1.2 miles) of the site. The historic archaeological sites contain debris scatters and roadbeds associated with farmsteads. One site has been tested and is recognized as eligible for listing on the National Register of Historic Places. One hundred fifty-eight buildings or structures in the 300 Area have been inventoried and of that number, 47 have been determined eligible for the National Register as contributing properties within the Historic District recommended for mitigation, including RPL/306–E (DOE 1996d; Neitzel 1999).

400 AREA

Most of the 400 Area has been so disrupted by construction activities, that a 1978 archaeological survey found only 12 hectares (30 acres) that were undisturbed. No cultural resources were located in those 12 hectares (30 acres). No archaeological sites are known to be located within 2 kilometers (1.2 miles) of the 400 Area.

All of the building and structures in the 400 Area were constructed during the Cold War era. Six buildings/structures have been determined eligible for the National Register of Historic Places as contributing properties within the Historic District recommended for mitigation. These include the 405 Reactor Containment Building, 436 Training Facility, 4621-W Auxiliary Equipment Facility, 4703 Fast Flux Test Facility Control Building, 4710 Operation Support Building, and the 4790 Patrol Headquarters (DOE 1996d).

3.4.7.3 Native American Resources

Native American resources are sites, areas, and materials important to Native Americans for religious or heritage reasons. In addition, cultural values are placed on natural resources such as plants, which have multiple purposes within various Native American groups. Of primary concern are concepts of sacred space that create the potential for land use conflicts.

3.4.7.3.1 General Site Description

In prehistoric and early historic times, Native Americans of various tribal affiliations heavily populated the Hanford Reach. The Wanapum and the Chamnapum lived along the Columbia River at what is now Hanford. Some of their descendants still live nearby at Priest Rapids, northwest of Hanford. Palus People, who lived on the lower Snake River, joined the Wanapum and Chamnapum to fish the Hanford Reach, and some inhabited the east bank of the river. Walla Walla and Umatilla People also make periodic visits to fish in the area. These people retain traditional secular and religious ties to the region, and many have knowledge of the ceremonies and lifeways of their culture. The Washani, or Seven Drums religion, which has ancient roots and originated among the Wanapum, is still practiced by many people on the Yakama, Umatilla, Warm Springs, and Nez Perce Reservations. Native plant and animal foods, some of which can be found at Hanford, are used in the ceremonies performed by tribal members.

Consultation is required, and was conducted, to identify the traditional cultural properties that are important in maintaining the cultural heritage of Native American tribes. The results of this consultation are presented in Chapter 4. Under separate treaties signed in 1855, the Confederated Tribes and Bands of the Yakama Nation and the Confederated Tribes of the Umatilla Indian Reservation ceded lands to the United States that

include the present Hanford Site. Under the treaties, the tribes reserved the right to fish at usual and accustomed places in common with the citizens of the territory, and retained the privilege of hunting, gathering roots and berries, and pasturing horses and cattle upon open, unclaimed land. The Treaty of 1855 with the Nez Perce Tribe includes similar reservations of rights, and the Nez Perce have identified the Hanford Reach as the location of usual and accustomed places for fishing. The Wanapum People are not signatory to any treaty with the United States and are not a federally recognized tribe; however, they live about 8 kilometers (5 miles) west of the Hanford boundary, they were historical residents of Hanford, and their interests in the area have been acknowledged.

All of these tribes are active participants in decisions regarding Hanford and have expressed concerns about hunting, fishing, pasture rights, and access to plant and animal communities and important sites. Sites sacred to Native Americans at Hanford include remains of prehistoric villages, burial grounds, ceremonial longhouses or lodges, rock art, fishing stations, and vision quest sites. Culturally important localities and geographic features include Rattlesnake Mountain, Gable Mountain, Gable Butte, Goose Egg Hill, Coyote Rapids, and the White Bluffs portion of the Columbia River.

3.4.7.3.2 Locations of Proposed Activities

300 AREA

One documented locality with great importance to the historic Wanapum is near the 300 Area. Certain areas near the 300 Area have been found to be of great importance to Native Americans and are fenced (Neitzel 1999).

400 AREA

The 400 Area is not known to contain any Native American resources.

3.4.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age.

3.4.7.4.1 General Site Description

Remains from the Pliocene and Pleistocene Ages have been identified at Hanford. The Upper Ringold Formation dates to the Late Pliocene Age and contains fish, reptile, amphibian, and mammal fossil remains. Late Pleistocene Touchet beds have yielded mammoth bones. These beds are composed of fluvial sediments deposited along ridge slopes that surround Hanford.

3.4.7.4.2 Locations of Proposed Activities

300 AREA

Paleontological resources are limited in the vicinity of the 300 Area, and no such resources have been located within the site itself (Nielsen 2000).

400 AREA

No paleontological resources have been reported in the 400 Area. Late Pleistocene Touchet beds, which have yielded mammoth bones, are located at distances greater than 5 kilometers (3.1 miles) from the 400 Area.

3.4.8 Socioeconomics

Statistics for employment and regional economy are presented for the regional economic area, as defined in Appendix G.8, which encompasses nine counties surrounding Hanford in Washington. Statistics for population, housing, community services, and local transportation are presented for the region of influence, a two-county area in which 91 percent of all Hanford employees reside (**Table 3–30)**. In 1997, Hanford employed 12,882 persons, 3.8 percent of the regional economic area civilian labor force (DOE 1999e).

Table 3–30 Distribution of Employees by Place of Residence in the Hanford Region of Influence, 1997

County	Number of Employees	Total Site Employment (percent)
Benton	10,563	82.0
Franklin	1,159	9.0
Region of influence total	11,722	91.0

Source: DOE 1999e.

3.4.8.1 Regional Economic Characteristics

Between 1990 and 1996, the civilian labor force in the regional economic area increased 34.6 percent, to the 1996 level of 342,941. In 1996, the annual unemployment average in the regional economic area was 11.1 percent, significantly higher than the annual unemployment average of 6.5 percent in Washington State (DOE 1999e).

In 1995, service activities represented the largest sector of employment in the regional economic area (22.3 percent). This was followed by agriculture (19.6 percent) and government (17.4 percent). Overall, the state total for these employment sectors was 25.0 percent, 3.7 percent, and 18.0 percent, respectively (DOE 1999e).

3.4.8.2 Population and Housing

In 1996, the region of influence population totaled 179,949. Between 1990 and 1996, the region of influence population increased 18.9 percent, compared with the 12.9 percent increase experienced in Washington. Between 1980 and 1990, the number of housing units in the region of influence increased by 4.6 percent, compared with a 20.3 percent increase in Washington. The 1990 homeowner vacancy rates for the region of influence was 1.4 percent, compared with the state's rate of 1.3 percent. The region of influence renter vacancy rate was 5.5 percent, compared with 5.8 percent for the state (DOE 1999e).

3.4.8.3 Community Services

3.4.8.3.1 Education

In 1997, ten school districts providing public education in the Hanford region of influence were operating at capacities ranging from 65 to 100 percent. Total student enrollment in the region of influence in 1997 was 38,206 and the student-to-teacher ratio in the region of influence averaged 16:1. In 1990, the average student-to-teacher ratio for Washington was 11.4:1 (DOE 1999e).

3.4.8.3.2 Public Safety

In 1997, a total of 281 sworn police officers were serving the region of influence. The region of influence average officer-to-population ratio was 1.6 officers per 1,000 persons. This compares with the 1990 state average of 1.7 police officers per 1,000 persons. In 1997, 616 paid and volunteer firefighters provided fire protection services in the Hanford region of influence. The average firefighter-to-population ratio in 1997 in the region of influence was 3.4 firefighters per 1,000 persons. This compares with the 1990 state average of 1 firefighter per 1,000 persons (DOE 1999e).

3.4.8.3.3 Health Care

In 1996, a total of 257 physicians served the region of influence. The average physician-to-population ratio in the region of influence was 1.4 physicians per 1,000 persons compared with the 1996 state average of 3.7 per 1,000 persons. In 1997, there were four hospitals serving the region of influence. The hospital bed-to-population ratio averaged 2.1 beds per 1,000 persons. This compares with a state 1991 average of 2.4 beds per 1,000 persons (DOE 1999e).

3.4.8.4 Local Transportation

Vehicle access to Hanford is provided by State Routes 240, 243, and 24. State Route 240 connects to the Richland bypass highway, which interconnects with I–182. State Route 243 exits the site's northwestern boundary and serves as a primary link between the site and I–90. State Route 24 enters the site from the west and continues eastward across the northernmost portion of the site and intersects State Route 26 about 16 kilometers (10 miles) east of the site boundary (Figure 3–12) (DOE 1999e). Only routine preservation projects are planned by the Washington State Department of Transportation for the state routes listed above and are not considered to impact access into the site (Trepanier 2000).

The local intercity transit system, Ben Franklin Transit, supplies bus service between the Tri-Cities and Hanford, although bus service is provided only to the 300 Area and Energy Northwest. Both private interests and Ben Franklin Transit provide van pooling opportunities in the region of influence.

There is presently no rail service at Hanford, except for a spur to Energy Northwest. Onsite rail transport was formerly provided by a short-line railroad that connected with the Union Pacific line just south of the Yakima River. The Union Pacific line interchanges with the Burlington Northern-Santa Fe at the city of Kennewick. The Hanford railroad is still intact and service could be restored if needed.

In the region of influence, the Columbia River is used as an inland waterway for barge transportation from the Pacific Ocean. The Port of Benton provides a barge slip where shipments arriving at Hanford may be off-loaded.

Tri-Cities Airport, near the city of Pasco, provides jet air passenger and cargo service by both national and local carriers. Numerous smaller private airports are located throughout the region of influence (DOE 1999e).

3.4.9 Existing Human Health Risk

Existing human health risk issues include the determination of potentially adverse effects on human health that result from acute and chronic exposures to ionizing radiation and hazardous chemicals.

3.4.9.1 Radiation Exposure and Risk

3.4.9.1.1 General Site Description

Major sources and levels of background radiation exposure to individuals in the vicinity of Hanford are shown in **Table 3–31**. Annual background radiation doses to individuals are expected to remain constant over time.

Table 3–31 Sources of Radiation Exposure to Individuals in the Hanford Vicinity Unrelated to Hanford Operations

Source	Effective Dose Equivalent (millirem per year)
Natural background radiation[a]	
Cosmic radiation	30
External terrestrial radiation	30
Internal radiation	40
Radon in homes (inhaled)	200
Other background radiation[b]	
Diagnostic x-rays and nuclear medicine	53
Weapons test fallout	Less than 1
Air travel	1
Consumer and industrial products	10
Total	365

a. Dirkes, Hanf, and Poston 1999.
b. NCRP 1987:11, 40, 53.
Note: Value of radon is an average for the United States.

The total dose to the population, in terms of person-rem, changes as the population size changes. Background radiation doses, as identified in Table 3–31, are unrelated to Hanford operations.

Releases of radionuclides to the environment from Hanford operations provide another source of radiation exposure to individuals in the vicinity of Hanford. Types and quantities of radionuclides released from Hanford operations in 1998 are listed in the *Hanford Site Environmental Report for Calendar Year 1998* (Dirkes, Hanf, and Poston 1999:5.5–5.10). Doses to the public resulting from these releases are presented in **Table 3–32**. These doses fall within radiological limits per DOE Order 5400.5 and are much lower than those of background radiation.

Using a risk estimator of 500 cancer deaths per 1 million person-rem to the public (Appendix H), the risk of a latent cancer fatality to the maximally exposed member of the public due to radiological releases from Hanford operations in 1998 is estimated to be 1.1×10^{-8}. That is, the estimated probability of this person dying of cancer at some point in the future from radiation exposure associated with 1 year of Hanford operations is approximately 1 in 100 million. It takes several to many years from the time of radiation exposure for a cancer to manifest itself.

According to the same risk estimator, 1×10^{-4} excess latent cancer fatality are projected in the population of 370,000 living within 80 kilometers (50 miles) of Hanford from normal operations in 1998. To place this number in perspective, it may be compared with the number of cancer fatalities expected in the same population from all causes. The 1997 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year. Based on this mortality rate, the number of cancer fatalities expected during 1998 from all causes in the population living within 80 kilometers (50 miles) of Hanford was 740. This expected number of cancer fatalities (which excludes any radiation dose contribution from Hanford) is much higher than the 1×10^{-4} latent cancer fatality estimated from Hanford operations in 1998.

Table 3–32 Radiation Doses to the Public from Hanford Normal Operations in 1998 (Total Effective Dose Equivalent)

Members of the Public	Atmospheric Releases[a]		Liquid Releases		Total	
	Standard[b]	Actual	Standard[b]	Actual	Standard[b]	Actual
Maximally exposed individual (millirem)	10	0.015	4	0.0077[c]	100	0.022
Population within 80 kilometers (person-rem)[d]	None	0.084	None	0.11	100	0.19
Average individual within 80 kilometers (millirem)[e]	None	3.4×10^{-4}	None	2.9×10^{-4}	None	5.0×10^{-4}

a. Includes direct radiation dose from surface deposits of radioactive material.
b. The standards for individuals are given in DOE Order 5400.5. As discussed in that order, the 10-millirem-per-year limit from airborne emissions is required by the Clean Air Act, and the 4-millirem-per-year limit is required by the Safe Drinking Water Act; for this NI PEIS, the 4-millirem-per-year value is conservatively assumed to be the limit for the sum of doses from all liquid pathways. The total dose of 100-millirem-per-year is the limit from all pathways combined. The 100-person-rem value for the population is given in proposed 10 CFR Part 834, as published in 58 FR 16268. If the potential total dose exceeds the 100-person-rem value, it is required that the contractor operating the facility notify DOE.
c. Includes the drinking water dose.
d. Based on a population of about 380,000 in 1998.
e. Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site.

Source: Dirkes, Hanf, and Poston 1999:5.9, 5.10.

Hanford workers receive the same dose as the general public from background radiation, but they also receive an additional dose from working in facilities with nuclear materials. The average dose to the individual worker and the cumulative dose to all workers at Hanford from operations in 1998 are presented in **Table 3–33**. These doses fall within the radiological regulatory limits of 10 CFR Part 835. According to a risk estimator of 400 cancer fatalities per 1 million person-rem among workers[1] (Appendix H), the number of projected latent cancer fatalities among Hanford workers from normal operations in 1998 is 0.072.

Table 3–33 Radiation Doses to Workers from Hanford Normal Operations in 1998 (Total Effective Dose Equivalent)

Occupational Personnel	Onsite Releases and Direct Radiation	
	Standard[a]	Actual
Average radiation worker (millirem)	None[b]	102
Total workers[c] (person-rem)	None	181

a. The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. It has therefore established the Administrative Control Level of 2,000 millirem per year; the site must make reasonable attempts to maintain individual worker doses below this level.
b. No standard is specified for an "average radiation worker"; however, the maximum dose that this worker may receive is limited to that given in footnote "a."
c. 1,772 with measurable doses in 1998.

Source: 10 CFR Section 835.202; DOE 1999p.

A more detailed presentation of the radiation environment, including background exposures and radiological releases and doses, is presented in the *Hanford Site Environmental Report for Calendar Year 1998* (Dirkes, Hanf, and Poston 1999). The concentrations of radioactivity in various environmental media (including air, water, and soil) in the site region (on and off site) are also presented in that report.

[1] The risk estimator for workers is lower than the estimator for the public because of the absence from the workforce of the more radiosensitive infant and child age groups.

3.4.9.1.2 Locations of Proposed Activities

300 AREA

External radiation doses have been measured in the 300 Area. In 1998, the annual dose in the 300 Area was about 83 millirem. This is about 5 to 12 millirem higher than the value measured at the off site control locations. This onsite dose affects workers only, and is well below limits identified in Table 3–33. No measurements of plutonium concentrations in air were reported for the 300 Area (Dirkes, Hanf, and Poston 1999:4.84, 4.85).

400 AREA

External radiation doses have been measured in the 400 Area. In 1998, the annual dose in the 400 Area was about 83 millirem. This is about 5 to 12 millirem higher than the value measured at the offsite control locations. This onsite dose affects workers only, and is well below limits identified in Table 3–33. No measurements of plutonium concentrations in air were reported for the 400 Area (Dirkes, Hanf, and Poston 1999:4.84, 4.85).

3.4.9.2 Chemical Environment

The background chemical environment important to human health consists of the atmosphere which may contain hazardous chemicals that can be inhaled; drinking water, which may contain hazardous chemicals that can be ingested; and other environmental media through which people may come in contact with hazardous chemicals (e.g., surface water during swimming, soil through direct contact, or food). Hazardous chemicals can cause cancer and noncancer health effects.

Carcinogenic Effects. Health effects in this case are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogenic. This could be incremental or excess individual lifetime cancer risk.

Noncarginogenic Effects. Health effects in this case are determined by the ratio between the calculated or measured concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the Hazard Quotient. Hazard Quotients for noncarcinogens are summed to obtain the Hazard Index. If the Hazard Index is less than 1, no adverse health effects would be expected.

Effective administrative and design controls that decrease hazardous chemical releases to the environment and help achieve compliance with permit requirements (e.g., air emissions and NPDES permit requirements) contribute to minimizing health impacts on the public. The effectiveness of these controls is verified through the use of monitoring information and inspection of mitigation measures. Health impacts on the public may occur by inhaling air containing hazardous chemicals released to the atmosphere during normal Hanford operations. Risks to public health from other possible pathways, such as ingestion of contaminated drinking water or direct exposure, are lower than those via the inhalation pathway.

Baseline air emission concentrations and applicable standards for hazardous chemicals are addressed in Section 3.4.3. The baseline concentrations are estimates of the highest existing offsite concentrations and represent the highest concentrations to which members of the public could be exposed. These concentrations are in compliance with applicable guidelines and regulations.

Exposure pathways to Hanford workers during normal operations may include inhaling contaminants in the workplace atmosphere and direct contact with hazardous materials. The potential for health impacts varies

among facilities and workers, and available information is insufficient for a meaningful estimate of impacts. However, workers are protected from workplace hazards through appropriate training, protective equipment, monitoring, substitution, and engineering and management controls. They are also protected by adherence to OSHA and EPA standards that limit workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Appropriate monitoring that reflects the frequency and amounts of chemicals used in the operational processes ensures that these standards are not exceeded. Additionally, DOE requires that conditions in the workplace be as free as possible from recognized hazards that cause, or are likely to cause, illness or physical harm.

3.4.9.3 Health Effects Studies

The question of whether or not the population surrounding Hanford is subject to elevated rates of cancer incidence or cancer mortality is unresolved. Existing studies and data suggest that cancer mortality rates among populations residing near Hanford are not elevated. A survey sponsored by the National Cancer Institute and published in the Journal of the American Medical Association in 1991 (Jablon, Hrubec, and Boice 1991:1403–1408) detected no general increase in the risk of cancer death for people living in 107 counties adjacent to or containing 62 nuclear facilities. Hanford, INEEL, and ORR were included in the survey. The study used cancer mortality data from Benton, Franklin, and Grant counties in the survey for Hanford. The methodology used in the survey did not attempt to estimate actual exposures to ionizing radiation or hazardous chemicals and did not allow identification of areas within a given county that might have increased or decreased cancer rates relative to the country as a whole. The authors of the study concluded that if any excess cancer mortality risk were present in U.S. counties with nuclear facilities, it was too small to be detected with the methods employed.

Sixteen counties are within 80 kilometers (50 miles) of the Hanford boundary—13 counties in Washington and 3 in Oregon. Prevailing winds at Hanford blow toward Grant County, Washington, from the south (14 percent of the time) and south-southwest (11.5 percent of the time). Therefore, Grant County would be expected to bear the major burden of wind-borne contamination from Hanford. Cancer mortality data published by the National Cancer Institute (www.nci.nih.gov/atlas) for white female and white male residents for all U.S. counties from 1970 to 1994 show no elevated cancer rates for white residents of Grant County. Cancer mortality rates among white females in the 16 counties ranged from a low of 80.1 per 100,000 person-years in Gilliam County, Oregon, to a high of 149.5 per 100,000 person-years in Lincoln County, Washington. Adams, Klickitat, and Lincoln counties were found to have cancer mortality rates among white females above the National cancer mortality rate for white females of 135.9 per 100,000 person-years. The remaining 13 counties have cancer mortality rate for white females below the National cancer mortality rate for white females. Cancer mortality rates among white males in the 16 counties range from a low of 161.9 per 100,000 person-years in Gilliam County, Oregon, to a high of 211.8 per 100,000 person-years in Morrow County, Oregon. Morrow County was found to have a cancer mortality rate among white males above the National cancer mortality rate for white males of 209.5 per 100,000 person-years. The remaining 15 counties were found to have cancer mortality rates below the National cancer mortality rate for white males. The data does not include estimates of human exposures to ionizing radiation or hazardous chemicals.

Two studies of birth defects in Benton and Franklin counties were published in 1988 (Sever et al. 1988a:226–241; 1988b:243–254). The studies focused on congenital malformations among infants born from 1968 to 1980. The studies showed a statistically significant association between parental preconception exposure to ionizing radiation and neural tube defects in their infants. Other defects in the infant studies showed no statistically significant association with parental radiation exposure.

Many epidemiological studies have been carried out on the Hanford workers over the years. The studies have consistently shown a statistically significant elevated risk of death from multiple myeloma associated with

radiation exposure among Hanford male workers. The elevated risk was observed only among workers exposed to 10 rads (approximately10 rem) or more. Other studies have also identified an elevated risk of death from pancreatic cancers, but a recent reanalysis did not conclude there was an elevated risk. Studies of female Hanford workers have shown an elevated risk of deaths from musculoskeletal system and connective tissue conditions. For a more detailed description of the studies reviewed and their findings, and for a discussion of the epidemiologic surveillance program implemented by DOE to monitor the health of current workers, refer to Appendix M.4.2 of the *Storage and Disposition PEIS* (DOE 1996b:M-224–M-230).

3.4.9.4 Accident History

DOE maintains a safe and healthy workplace in accordance with DOE P 450.4, *Safety Management Systems Policy*; DOE Order 420.1, *Facility Safety*; DOE Order 151.1A, *Comprehensive Emergency Management System*; 29 CFR 1910.119, *Process Safety Management of Highly Hazardous Chemicals*; 29 CFR 1910.120, *Hazardous Waste Operations and Emergency Response*; and the Washington Administrative Code 267-247. There are three tiers of safety organizations at Hanford: the DOE Operations Office, Fluor Hanford, and project-specific organizations. Each safety organization is responsible for protection of the public, workers, and the environment. Information concerning safety-related events at Hanford and other sites is available from the DOE occurrence reporting system on the Internet at tis.eh.doe.gov/oeaf/orps.html.

Hanford implements corrective actions for all safety-related incidents. For example, although unrelated to candidate facilities discussed in this NI PEIS, a chemical explosion occurred at the Hanford Plutonium Reclamation Plant in a room where nonradioactive bulk chemicals were mixed for the now-discontinued plutonium recovery process. The direct cause of the accident was the concentration by evaporation of the dilute solution in a tank to the point where a spontaneous reaction occurred, creating a rapid gas evolution that over-pressurized the tank beyond its physical design limitations. No one was injured and no radioactive materials were released to the environment (DOE 1997h). Eight workers outside the plant at the time of the explosion complained of symptoms that included headaches, dizziness, and an unidentified metallic taste. All eight workers were transferred to a nearby medical center where they were examined and released. The explosion caused significant localized damage to the facility. Corrective actions focused on improving shutdown planning to maintain the facility in a safe condition, consistent with the approved safety authorization documentation, and improving emergency preparedness and response. As discussed in Section 3.4.9.5, lessons learned from this event were implemented across the DOE complex to improve emergency preparedness and response.

There have been no nuclear-related accidents or accidental releases of hazardous or radioactive materials causing injury or harm to workers, or posing any threat to the offsite public at FFTF or at the candidate Hanford support facilities evaluated in this NI PEIS. Examples of the most severe safety incidents that have occurred at these facilities are discussed below. In all cases, corrective actions were completed to address the cause of each event, and there were no long-term programmatic consequences:

- A loss of contamination control event occurred in February 1990 at a maintenance facility adjacent to FFTF during a filter replacement on the bottom-loading transfer cask (an FFTF fuel-handling machine) resulting in contamination spread to the adjacent area within the facility. The contamination was successfully cleaned up and the facility was restored to normal access and work control. Corrective actions to prevent similar occurrences included (1) training program changes and additional training of plant personnel on job controls and planning for these types of hazards, (2) strengthened requirements for pre-job briefings and Person-in-Charge responsibilities, and (3) more detailed requirements for the various types of radiological control areas.

- A sodium pump developed a leak in 1984. As a result of the leak, 75 gallons of sodium spilled in a closed room filled with inert gas. It was determined that the hole was created by two conditions: tube wall thinning due to cavitation and rapid heatup during a previous meltout. Enhanced leak detection was installed on all normally inaccessible pumps, and a visual inspection was conducted on the remaining pumps. Procedural flow restrictions were placed on the pumps to preclude any additional cavitation conditions, and changes were made to the meltout procedures to reduce the allowable heatup rate. The sodium was removed and the pump was replaced with a spare.

- Two unplanned tritium releases occurred at RPL in 1998 because of equipment failure or operator error. These releases were within the levels specified by the facility's air operating permit and did not result in the exposure of site personnel or members of the public in excess of regulatory standards. Corrective actions included the redesign of a sampling system to permit more effective leak testing, and implementation of administrative controls to eliminate operator error.

3.4.9.5 Emergency Preparedness

Each DOE site has established an emergency management program that would be activated in the event of an accident. This program has been developed and maintained to ensure adequate response to most accident conditions and to provide response efforts for accidents not specifically considered. The emergency management program includes emergency planning, preparedness, and response.

Accordingly, the DOE Richland Operations Office has developed and maintains a comprehensive set of emergency preparedness plans and procedures for Hanford to support onsite and offsite emergency management actions in the event of an accident. The DOE Richland Operations Office also provides technical assistance to other Federal agencies and to state and local governments. Hanford contractors are responsible for ensuring that emergency plans and procedures are prepared and maintained for all facilities, operations, and activities under their jurisdiction, and for directing implementation of those plans and procedures during emergency conditions. The DOE Richland Operations Office, contractor, and state and local government plans are fully coordinated and integrated. Emergency control centers have been established by the DOE Richland Operations Office and its contractors for the principal work areas to provide oversight and support to emergency response actions within those areas.

Following the May 1997 explosion at Hanford discussed in Section 3.4.9.4, a review of the emergency management response indicated that multiple programs and systems failed in the hours following the accident. In a letter to Secretarial Offices, Secretary of Energy Federico Peña identified actions to be taken at all DOE sites for implementing lessons learned from the emergency response. The actions involve the following elements:

- Improve training for facility and site emergency personnel
- Ensure that equipment and qualified personnel are ready for the wide variety of potential radiological and chemical hazards
- Improve coordination with local medical communities
- Have in place comprehensive procedures to attend to personnel who are potentially affected by an accident

3.4.10 Environmental Justice

Under Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, Federal agencies are responsible for identifying and addressing the possibility of disproportionately high and adverse health, economic, and environmental impacts of programs and activities

on minority and low-income populations in potentially affected areas. Minority populations refer to persons of any race self-designated as Asian, Black, Native American, or Hispanic. Low-income populations refer to households with incomes below the Federal poverty thresholds. In the case of Hanford, the potentially affected area includes parts of Washington and Oregon.

The potentially affected area surrounding the 400 Area is defined by a circle with an 80-kilometer (50-mile) radius centered at FMEF (latitude 46° 26'7" N, longitude 119° 21'55" W). The total population residing within that area in 1990 was 277,515; minorities made up 25.4 percent of the total population (DOE 1999e). In 1990, approximately one-fourth of the total national population was comprised of persons self-designated as members of a minority group, and minorities made up 13.2 percent of Washington State's total population and 9.2 percent of Oregon's total population.

According to the 1990 census, the racial and ethnic composition of the minority population in the potentially affected area around FMEF are as follows: Hispanics were the largest minority group, constituting 21.5 percent of the total population; Asians comprised 1.4 percent of the total population, Native Americans 1.4 percent, and Blacks 1.0 percent (DOE 1999e).

In 1990, the poverty threshold was $9,981 for a family of three with one related child under 18 years of age. A total of 45,820 persons, 17.3 percent of the total population, residing within the potentially affected area around the 400 Area reported incomes below the poverty threshold. Data obtained during the 1990 census also show that of the total population of the contiguous United States, 13.1 percent reported incomes below the poverty threshold, corresponding percentages for Washington and Oregon were 10.9 and 12.4 percent, respectively.

3.4.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing DOE activities. The waste is managed using appropriate treatment, storage, and disposal technologies and in compliance with all applicable Federal and state statutes and DOE orders.

3.4.11.1 Waste Inventories and Activities

Hanford manages the following types of waste: high-level, transuranic, mixed transuranic, low-level radioactive, mixed low-level radioactive, hazardous, and nonhazardous. Waste generation rates and the inventory of stored waste from activities at Hanford are provided in **Table 3–34**. Waste generation rates specifically for FFTF in standby and RPL/306–E activities are provided in **Table 3–35**. The Hanford waste management capabilities are summarized in **Table 3–36**. More detailed descriptions of the waste management system capabilities at Hanford are included in the *Storage and Disposition PEIS* (DOE 1996b:3-61, E-12).

EPA placed Hanford on the National Priorities List on November 3, 1989. In accordance with CERCLA, DOE entered into a Tri-Party Agreement with EPA and the Washington State Department of Ecology to govern the environmental compliance and cleanup of Hanford. This agreement meets the legal requirements specified under the Federal Facility Compliance Act. An aggressive environmental restoration program is under way using priorities established in the Tri-Party Agreement (DOE 1996b:3-61). More information on regulatory requirements for waste disposal is provided in Chapter 5.

Table 3–34 Waste Generation Rates and Inventories at Hanford

Waste Type	Generation Rate (cubic meters per year)	Inventory (cubic meters)
High-level radioactive	0	213,000
Transuranic and mixed transuranic		
Contact handled	450	11,450
Remotely handled	72	273
Low-level radioactive	3,902[a]	0
Mixed low-level radioactive		
RCRA	840	8,170
TSCA	7	103
Hazardous	560	NA[b]
Nonhazardous		
Liquid	200,000	NA[b]
Solid	43,000	NA[b]

a. Excludes waste from DOE environmental restoration activities.
b. Generally, hazardous and nonhazardous wastes are not held in long-term storage.
Note: To convert from cubic meters to cubic yards, multiply by 1.308.
Key: NA, not applicable; RCRA, Resource Conservation and Recovery Act; TSCA, Toxic Substances Control Act.
Source: LMER 1996e:15, 16, except high-level radioactive waste (DOE 1997a), hazardous and nonhazardous solid wastes (DOE 1996b:3-62, E-19), and nonhazardous liquid wastes (Teal 1997).

Table 3–35 Waste Generation Rates at FFTF and RPL/306–E

Waste Type	FFTF (cubic meters per year)	RPL/306–E (cubic meters per year)
High-level radioactive	0	0
Transuranic	0	8
Low-level radioactive		
Liquid	<6	104[a]
Solid	17	-
Mixed low-level radioactive	<0.5	15
Hazardous	4[b]	6
Nonhazardous		
Process wastewater	76,000	28,400
Sanitary wastewater	3,800	4,970
Solid	120	4

a. Represents both liquid and solid low-level radioactive waste.
b. Represents both liquid and solid hazardous waste.
Note: To convert from cubic meters to cubic yards, multiply by 1.308.
Source: DOE 2000c, Tenforde 2000.

Table 3–36 Waste Management Capabilities at Hanford

Facility Name/ Description	Capacity	Status	Applicable Waste Type						
			HLW	TRU	Mixed TRU	LLW	Mixed LLW	Haz	Non-Haz
Treatment Facility (cubic meters per year except as otherwise specified)									
242-A Evaporator, cubic meters per day	265	Online	X	X	X	X	X		
Waste Receiving and Processing Facility[a]	1,820	Online		X	X	X	X		
M91 Facility	Will be negotiated	Will be negotiated		X	X	X	X		
Shielded Anlytical Lab Waste Treatment Unit, kilograms per hour	4	Online					X		
Maintenance & Storage Facility, batch per year	26	Online					X		
200 Area Liquid Effluent Treatment Facility, cubic meters per minute	0.57	Online				X	X		
200 East Area Sanitary Wastewater Treatment Facility	120,000	Online							X
Storage Facility (cubic meters)									
Tank Farms	146,000	Online	X	X	X	X			
Central Waste Complex	16,800	Online		X	X	X	X		
Transuranic Waste Storage and Assay Facility	416	Standby		X	X	X	X		
305-B Storage Facility	20	Online				X	X	X	
B-Plant Canyon Waste Pile	5	Online				X			
B-Plant Container Storage	51	Online					X		
PUREX Tunnel 1	4,141	Online				X	X		
PUREX Tunnel 2	19,528	Online				X	X		
PUREX Canyon Waste Pile	432	Online					X		
200 Area Liquid Effluent Rentention Facility	59,000	Online				X	X		
4843 Alkali Metal Storage Facility	95	Standby					X	X	
Disposal Facility (cubic meters except as otherwise specified)									
Grout Vault	230,000	Online				X			
Low-Level Radioactive Waste Burial Ground	1,740,000	Online				X			
Radioactive Mixed Waste Disposal Facility	14,200	Standby				X	X		
200 Area Treated Effluent Disposal Facility, cubic meters per minute	8.7	Online							X
Energy Northwest Sewage Treatment Facility, cubic meters per year	235,000	Online							X

a. The facility is used primarily for certification and repackaging transuranic wastes for shipment to WIPP and is also used to verify small quantities of low-level radioactive and mixed low-level radioactive wastes.

Note: To convert from cubic meters to cubic yards, multiply by 1.308.

Key: Haz, hazardous; HLW, high-level radioactive waste; LLW, low-level radioactive waste; PUREX, Plutonium-Uranium Extraction (Plant); TRU, transuranic radioactive waste; WIPP, Waste Isolation Pilot Plant.

Source: DOE 1996b:E-15; 1999e:3-10; Teal 1997.

3.4.11.2 High-Level Radioactive Waste

High-level radioactive waste was generated in the recovery of uranium and plutonium from spent fuel generated in the production reactors. All of this radioactive waste is considered mixed waste because of its toxic and hazardous constituents as defined by RCRA. It must be remotely handled because of its high radiation levels. The waste was generated as liquids and sludges and stored in underground tanks where the sludges and salts in the liquid have precipitated out of solution as porous solids(called salt cake) and settled to the bottom of the tanks. The liquid above the solids has been pumped from the older, single-shelled tanks into newer, double-shelled tanks. The liquids that remain in the porous salt cake will be removed by boring holes through the salt cake and extracting liquids from near the tank bottoms. The wastes are segregated and handled according to their hazardous nature (corrosivity, chemical stability, heat generation rates), and require special monitoring and venting. Cooling is needed for some of these wastes. The wastes are concentrated by evaporation and returned to the tanks for storage until final processing to a form suitable for disposal in a geologic repository. It is planned to vitrify high-level radioactive waste water-soluble sludges and selected radionuclides separated from liquids retrieved from the tanks. In addition to this liquid and solid high-level radioactive waste, an inventory of encapsulated cesium and strontium is stored in the Waste Encapsulation and Storage Facility in a water-cooled pool. Some of this material was used as irradiation sources in, for example, radiography and food irradiation (DOE 1996:3-65).

3.4.11.3 Transuranic and Mixed Transuranic Waste

All generated contact-handled transuranic waste is being placed in above-grade storage buildings at the Hanford Central Waste Complex (DOE 1996b:3-64). Transuranic waste will be maintained in storage until it is shipped to WIPP in Carlsbad, New Mexico, for disposal, or to a suitable geologic repository. The new Waste Receiving and Processing Facility has the capability to certify drummed or small container transuranic waste for shipment to WIPP (Dirkes and Hanf 1996:10). Transuranic wastes to be transported to WIPP will be packaged and shipped to WIPP for disposal in accordance with DOE and DOT requirements and WIPP waste acceptance criteria. Mixed transuranic wastes are included in the transuranic waste category because these wastes are expected to go to WIPP for ultimate disposal (DOE 1996b:3-64). The first shipment of transuranic waste from Hanford was received at WIPP on July 14, 2000 (DOE 2000d:2).

3.4.11.4 Low-Level Radioactive Waste

Solid low-level radioactive waste is compacted and sent to the Low-level Radioactive Waste Burial Ground in the 200 West Area for disposal in trenches. Additional low-level radioactive waste is received from offsite generators and disposed of at the Low-level Radioactive Waste Burial Ground. Low-level radioactive waste resulting from the River Protection Project-tank waste treatment will be vitrified. The vitrified low-level radioactive waste will be disposed of on site in the 200 Area as part of the tank waste remediation system program (DOE 1996b:3-64). Low-level radioactive waste resulting from CERCLA cleanup activities are disposed of on site at the Environmental Restoration Disposal Facility.

U.S. Ecology operates a licensed commercial low-level radioactive waste Burial Ground on a site southwest of the 200-East Area that is leased to the State of Washington. The facility is not a DOE facility and is not considered part of DOE's Hanford operations (DOE 1996b:E-17).

3.4.11.5 Mixed Low-Level Radioactive Waste

Miscellaneous dilute aqueous low-level radioactive and liquid mixed low-level radioactive wastes are temporarily stored in the Liquid Effluent Retention Facility until treated in the Liquid Effluent Treatment Facility. The Liquid Effluent Retention Facility consists of three RCRA-compliant surface impoundments for

storing process condensate from the 242-A Evaporator. This facility provides equalization of the flow and pH to the Liquid Effluent Treatment Facility. The Liquid Effluent Treatment Facility provides ultraviolet light/peroxide destruction of organic compounds, reverse osmosis to remove dissolved solids, and ion exchange to remove the last traces of contaminants. Discharge of the treated effluent is via a dedicated pipeline to an underground drain field. The effluent treatment process produces a mixed low-level radioactive waste sludge that is concentrated, dried, packaged in 208-liter (55-gallon) drums, and transferred to the Central Waste Complex. This secondary waste is stored prior to treatment, if necessary, and disposed in the Mixed Waste Trench (Dirkes and Hanf 1996:10, 45, 46).

The Waste Receiving and Processing Facility, near the Central Waste Complex in the 200 West Area, provides analyses, characterization, and preparation of drums and boxes for disposal of Hanford's mixed waste. The Waste Receiving and Processing Facility is designed to process 6,800 drums of waste annually and to prepare retrieved and stored transuranic waste for disposal (Dirkes and Hanf 1996:40).

The Radioactive Mixed Waste Disposal Facilities are in the Hanford Low-level Radioactive Waste Burial Ground and are designated as 218-W-5, Trench 31, and Trench 34. The facilities consist of rectangular trenches with approximate dimensions of 76 by 30 meters (250 by 100 feet). These facilities are RCRA compliant, with double liners and leachate collection and removal systems (Dirkes and Hanf 1996:40).

3.4.11.6 Hazardous Waste

There are no treatment facilities for hazardous waste at Hanford; therefore, the wastes are accumulated in satellite storage areas for less than 90 days, or at interim RCRA-permitted facilities, such as the 305-B Waste Storage Facility. The common practice for newly generated hazardous waste is to ship it off site by truck using DOT-approved transporters for treatment, recycling, recovery, and disposal at RCRA-permitted commercial facilities (DOE 1999e:3-11, 12).

3.4.11.7 Nonhazardous Waste

Sanitary wastewater is discharged to onsite treatment facilities such as septic tanks, subsurface soil adsorption systems, and wastewater treatment plants. These facilities treat an average of 600,000 liters (158,000 gallons) per day of sewage (DOE 1996b:E-19).

The 200 Area TEDF industrial sewer collects the treated wastewater streams from various plants in the 200 Areas and disposes of the clean effluent at two 2-hectare (5-acre) ponds permitted by the State of Washington (DOE 1996b:E-19). The design capacity of the facility is approximately 8,700 liters (2,300 gallons) per minute, although the discharge permit presently limits the average monthly flow to about 2,400 liters (640 gallons) per minute (Dirkes and Hanf 1996:46).

Nonhazardous solid wastes include construction debris, office trash, cafeteria wastes, furniture and appliances, nonradioactive friable asbestos, powerhouse ash, and nonradioactive/nonhazardous demolition debris. Until 1997, nonhazardous solid wastes were disposed of in the 600 Area central landfill. Under an agreement between DOE and the city of Richland, most of the site's nonregulated and nonradioactive solid wastes are now sent to the Richland Sanitary Landfill for disposal (DOE 1996b:3-65, E-19). The Richland Sanitary Landfill is at the southern edge of the Hanford Site boundary. Nonradioactive friable asbestos and medical waste are shipped off site for disposal to a commerical facility (DOE 1999e:3-12).

3.4.11.8 Waste Minimization

The Hanford Site Pollution Prevention Program is a comprehensive and continual effort to systematically reduce the quantity and toxicity of hazardous, radioactive, mixed, and sanitary wastes; conserve resources and energy; reduce hazardous substance use; and prevent or minimize pollutant releases to all environmental media from all operations and site cleanup activities. In accordance with sound environmental management, preventing pollution through source reduction is the first priority in the Hanford Site Pollution Prevention Program, and the second priority is environmentally safe recycling. Implementation of pollution prevention projects reduced the total amount of waste generated at Hanford in 1998 by approximately 17,500 cubic meters (23,000 cubic yards). Examples of pollution prevention projects completed in 1998 at Hanford include: the reduction of cleanup and stabilization of mixed low-level radioactive waste by approximately 170 cubic meters (220 cubic yards) by decontaminating numerous items (including process tanks, machinery, floors, and associated equipment and piping) to low-level radioactive waste status, avoiding a mixed low-level radioactive wastestream and associated disposal costs; the reduction of hazardous waste by 22 metric tons (24 tons) by removing CFC-12 refrigerant from four of eight chillers and selling it to a vendor for reuse; and the reduction of cleanup and stabilization of mixed low-level radioactive waste by approximately 11 cubic meters (14 cubic yards) by recycling scrap metal from an underground tank system for use as radiation shielding blocks (DOE 1999f:64).

DOE has developed a draft *Waste Minimization and Management Plan for FFTF* to incorporate pollution prevention and waste minimization practices in its consideration of the future of FFTF (DOE 2000c). If a decision were made to restart FFTF, this plan would be used to ensure that optimum opportunities are provided for characterizing potential waste streams, identifying source reduction and recycling strategies, evaluating disposition options, developing sustainable designs, and implementing effective management strategies. This plan identifies DOE's preferred options for management, treatment, and/or disposition of all waste streams related to the restart and operation of FFTF. These preferred options primarily use commercial waste handling and disposal facilities.

3.4.11.9 Waste Management PEIS Records of Decision

The *Waste Management PEIS* Records of Decision affecting Hanford are shown in **Table 3–37** for the waste types analyzed in this NI PEIS. Decisions on the various waste types were announced in a series of Records of Decision that have been issued on the *Waste Management PEIS*. The transuranic waste Record of Decision was issued on January 20, 1998 (63 FR 3629); the hazardous waste Record of Decision was issued on August 5, 1998 (63 FR 41810); the high-level readioactive waste Record of Decision was issued on August 12, 1999 (64 FR 46661); and the low-level radioactive waste Record of Decision was issued on February 18, 2000 (65 FR 10061). The transuranic waste Record of Decision states that DOE will develop and operate mobile and fixed facilities to characterize and prepare transuranic waste for disposal at WIPP.

Each DOE site that has or will generate transuranic waste will, as needed, prepare and store its transuranic waste on site. The hazardous waste Record of Decision states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of the nonwastewater hazardous waste, with ORR and SRS continuing to treat some of their own nonwastewater hazardous waste on site in existing facilities where this is economically favorable. The high-level radioactive waste Record of Decision states that immobilized high-level radioactive waste will be stored at the site of generation until transfer to a geologic repository. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision states that for the management of low-level radioactive waste, minimal treatment will be performed at all sites, and disposal will continue, to the extent practicable, on site at INEEL, LANL, ORR, and SRS. In addition,

Table 3–37 Waste Management PEIS Records of Decision Affecting Hanford

Waste Type	Preferred Action
High-level radioactive	DOE has decided that Hanford should store its high-level radioactive waste on site until transfer to a geologic repository[a]
Transuranic and mixed transuranic	DOE has decided that Hanford should prepare and store its transuranic waste on site pending disposal at WIPP or another suitable geologic repository.[b]
Low-level radioactive	DOE has decided to treat Hanford's low-level radioactive waste on site. Hanford has been selected as one of the regional disposal sites for low-level radioactive waste.[c]
Mixed low-level radioactive	DOE has decided to regionalize treatment at Hanford. This includes the onsite treatment of Hanford's wastes and could include treatment of some mixed low-level radioactive waste generated at other sites. Hanford has been selected as one of the regional disposal sites for mixed low-level radioactive waste.[c]
Hazardous	DOE has decided to continue to use commercial facilities for treatment of Hanford nonwastewater hazardous waste. DOE will also continue to use onsite facilities for wastewater hazardous waste.[d]

a. From the Record of Decision for high-level radioactive waste (65 FR 46661).
b. From the Record of Decision for transuranic waste (63 FR 3629).
c. From the Record of Decision for low-level radioactive and mixed low-level radioactive waste (65 FR 10061).
d. From the Record of Decision for hazardous waste (63 FR 41810).

Source: 63 FR 3629; 63 FR 44810; 64 FR 46661; 65 FR 10061.

Hanford and the Nevada Test Site will be available to all DOE sites for low-level radioactive waste disposal. Mixed low-level radioactive waste will be treated at Hanford, INEEL, ORR, and SRS, and disposed of at Hanford and the Nevada Test Site. More detailed information concerning DOE's alternatives for the future configuration of waste management facilities at Hanford is presented in the *Waste Management PEIS* and the transuranic waste, hazardous waste, and low-level radioactive and mixed low-level radioactive waste Records of Decision.

3.4.12 Spent Nuclear Fuel

When nuclear assemblies can no longer be used in the nuclear reactor, they are designated as "spent nuclear fuel," which is removed from the reactor and stored in the spent fuel storage pool, vessel, or basin. The Nuclear Waste Policy Act of 1982, as amended, assigned the Secretary of Energy the responsibility for developing of a repository for the disposal of high-level radioactive waste and spent nuclear fuel. When such a repository is available, spent nuclear fuel would be transferred for disposal from nuclear reactor site to the repository. Until a repository is available, spent nuclear fuel is stored in the reactor vessel, or in another acceptable method, such as in a dry cask storage system.

The current inventory of spent nuclear fuel at FFTF is approximately 11 metric tons of heavy metal, predominantly mixed plutonium-uranium oxide encapsulated in stainless steel. About 3 percent, (i.e., 0.3 metric tons of heavy metal) is of sodium-bonded spent nuclear fuel. In addition, there is 0.02 metric tons of heavy metal of training, research, isotopes General Atomics (TRIGA) spent nuclear fuel. This constitutes less than 1 percent of the cumulative spent nuclear fuel (about 2,133 metric tons of heavy metal), including defense and nondefense fuel at Hanford. DOE is managing this spent fuel in accordance with the Environmental Assessment, *Management of Hanford Site Non-Defense Production Reactor Spent Nuclear Fuel* and the associated Finding of No Significant Impact (DOE 1997f, 1997g).

3.5 GENERIC COMMERCIAL LIGHT WATER REACTOR SITE

Existing CLWRs use both pressurized water and boiling water technologies. Previous studies for the tritium supply program showed that, of the two types of commercial reactors, pressurized-water reactors are more readily adaptable than boiling-water reactors to the production of isotopes by target irradiation (DOE 1995b). DOE published a request for Expressions of Interest in the January 4, 1999, *Commerce Business Daily* for the production of plutonium-238 for space missions (DOE 1999n). No responses by the commercial nuclear industry to DOE's request for Expressions of Interest were provided by, or on behalf of, boiling-water reactor owners. The evaluation of CLWRs in this NI PEIS, therefore, will only be based on the use of pressurized-water reactors.

The use of CLWRs is not appropriate for the production of medical and industrial isotopes or to support civilian nuclear research and development because CLWRs operate on a 9- to 18-month cycle between refueling outages. Many medical and industrial isotopes have short half-lives and would decay before they could be removed from the reactors. In addition, CLWRs are not good irradiation sources for many civilian nuclear research tests because the range of neutron fluxes present in CLWR is limited, and the flux is optimized for power production rather than research. Accordingly, CLWRs are not appropriate irradiation sources for either medical and industrial isotope production or civilian nuclear research and development.

Because it is unreasonable for this NI PEIS to analyze all CLWRs, the environmental baseline was developed for a generic CLWR site description that is representative of existing reactor sites in the contiguous United States. The generic CLWR analysis in this NI PEIS is not site specific. Any one of the commercial, operating pressurized-water reactors is a potential candidate for the plutonium-238 production mission. Currently, 72 pressurized-water reactors are located at 42 sites in 27 states. The commercial, pressurized-water reactors operating in the United States that would be representative of the CLWR described and analyzed in this NI PEIS are shown in **Figure 3–22**. If an alternative were selected that involves the use of an existing CLWR site, site-specific environmental conditions would be identified in tiered NEPA documentation.

3.5.1 Land Resources

Land resources include land use and visual resources. Each of these resources is described below.

3.5.1.1 Land Use

Land may be characterized by its potential for the location of human activities (land use). Natural resource attributes and other environmental characteristics could make a site more suitable for some land uses than for others. Changes in land use may have both beneficial and adverse effects on other resources such as ecological, cultural, geological, aquatic, and atmospheric.

CLWR site areas within the United States range from 34 to 12,000 hectares (84 to 29,700 acres). Almost 60 percent of the plant sites encompass from 200 to 800 hectares (490 to 1,980 acres). Approximately half of the sites contain two or three nuclear units per site. Larger land use areas are associated with plant cooling systems that include reservoirs, artificial lakes, and buffer areas. Plant facilities are typically sited on 3 to 9 percent of the total site area. For sites that use cooling ponds instead of cooling towers, facilities could occupy a larger percentage, 67 to 76 percent, of the total site area (DOE 1996b). Typically, nuclear power plant sites are on and near flat-to-rolling countryside in wooded or agricultural areas. More than 50 percent of the sites have 80-kilometer (50-mile) population densities of fewer than 77 persons per square kilometer (200 persons per square mile) and more than 80 percent have 80-kilometer (50-mile) densities of fewer than 193 persons per square kilometer (500 persons per square mile) (DOE 1996b).

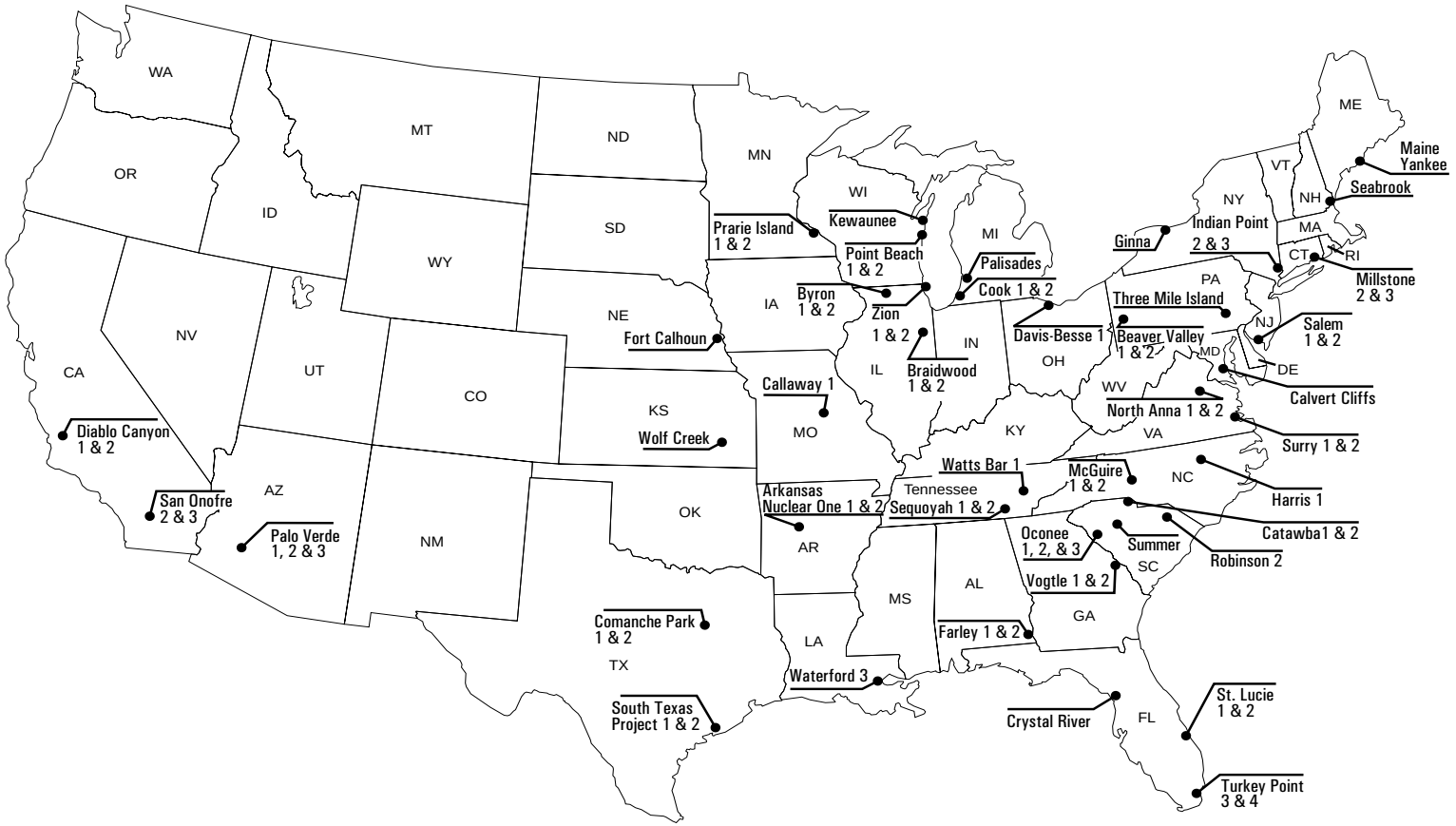


Figure 3–22 Commercial Pressurized-Water Reactors Operating in the United States

The location of a generic CLWR site would range between 3 and 55 kilometers (2 and 34 miles) from the nearest city, and most likely be further from the closest metropolitan area than 80 kilometers (50 miles). The site would likely be located adjacent to a large water body, such as a lake, river, or bay.

3.5.1.2 Visual Resources

Visual resources are natural and human-created features that give a particular landscape its character and aesthetic quality. Landscape character is determined by the visual elements of form, line, color, and texture. All four elements are present in every landscape. The stronger the influence exerted by these elements in a landscape, the more interesting the landscape.

The visual environment of a generic CLWR site would likely be characterized by flat to gently rolling topography adjacent to a large water body. The site would be a developed area that contains facilities and activities, surrounded by an undeveloped buffer area. The viewshed would likely include a small-to-medium sized urbanized area with surrounding forest and agricultural use. Depending on topography, atmospheric conditions, vegetation, and distance, the facilities of a generic existing CLWR site could be visible from adjacent viewpoints. Stack plumes from cooling towers could be visible under most meteorological conditions. Median visible plume lengths would usually range from less than 500 meters (1,640 feet) in summer to 1,000 meters (3,280 feet) in winter. The facilities would be brightly lit at night. The range of public viewpoints could include public access roadways, urbanized areas, and recreation and scenic areas with high user volumes. Since the site would be adjacent to a large water body, it would be likely that distance zones would range from foreground to middleground. The developed areas of a generic existing CLWR site would likely be consistent with a Bureau of Land Management Visual Resource Management Class IV rating indicating that the level of change to the characteristic landscape is high and that management activities dominate the view and are the major focus of view attention (DOE 1996b).

3.5.2 Noise

Principal noise sources at nuclear power plants include cooling towers and transformers. The impacts of these and other sources at the plants have been found to be small and generally not noticed by the public (NRC 1996:17). In most cases, noise sources are sufficiently distant from sensitive receptors that noise is attenuated to nearly ambient levels, although at some sites, sensitive receptors were identified during licensing at which noise levels would be greater than 10 decibels above ambient (NRC 1996:139). An area near a CLWR site would be essentially rural in character and would have typically low background sound levels. Typical day-night average sound levels in the range of 35 to 50 dBA can be expected for such a rural location where noise sources may include wind, insect activity, aircraft, and agricultural activity. Existing industrial noise sources and traffic noise at the site would result in higher background noise levels near the site and along site access routes (DOE 1996b:3-387).

3.5.3 Air Quality

Air pollution refers to the introduction, directly or indirectly, of any substance into the air that could endanger human health, harm living resources and ecosystems as well as material property, and impair or interfere with the comfortable enjoyment of life or other legitimate uses of the environment.

Ambient air quality conditions at CLWR sites in the United States could include a wide range of pollutants and conditions. The baseline air concentrations for criteria pollutants at a generic CLWR site are presented in **Table 3–38**. These concentrations are based on ambient monitoring data collected near a representative CLWR site. Some potential CLWR sites are near or within nonattainment areas for PM_{10}, ozone, and carbon monoxide. The maximum ground-level pollutant concentrations that would result from CLWR emissions are

Table 3–38 Comparison of Baseline Air Concentrations with Most Stringent Applicable Regulations or Guidelines at the Generic CLWR Site

Criteria Pollutant	Averaging Period	Most Stringent Regulation or Guideline[a] (micrograms per cubic meter)	Baseline Concentration (micrograms per cubic meter)
Carbon monoxide	8 hours	10,000	1,250
	1 hour	40,000	1,250
Lead	Calendar quarter	1.5	0.03
Nitrogen dioxide	Annual	100	26.3
Ozone	1 hour	235	(b)
PM_{10}	Annual	50	20.3
	24 hours	150	39.000
Sulfur dioxide	Annual	80	10.5
	24 hours	365	65.5
	3 hours	1300	204

a. The Federal standards are presented.
b. Ozone, as a criteria pollutant, is not directly emitted or monitored by the sites.
Key: PM_{10}, particulate matter with an aerodynamic diameter less than or equal to 10 microns.
Source: DOE 1999o.

low when compared to NAAQS. However, if the CLWR is in an area that may already have high background pollutant concentrations, resultant pollutant concentrations could approach or exceed NAAQS. As a result, regulatory compliance will need to be assesed on a case-by-case basis.

3.5.4 Water Resources

Major surface water features near a generic CLWR site could range from a large navigable river to a large lake. These surface waters would be classified and protected by regulation for specified uses, such as water supply. CLWRs would also have NPDES permits that specify the concentrations of pollutants and temperature permissible for liquid effluents and stormwater runoff discharged to surface waters. Other surface water bodies could include ponds and/or site-bordering ephemeral or perennial streams (DOE 1996b:3-388).

CLWRs withdraw large amounts of mainly surface water to meet a variety of plant needs. Water withdrawal rates from adjacent bodies of water for plants with once-through cooling systems are large. Flow through the condenser for a 1,000 megawatt plant may be 2.6 million to 3.8 million liters (700,000 to 1 million gallons) per minute. Water lost by evaporation from the heated discharge is about 60 percent of that which is lost through cooling towers. Additional water needs for service water, auxiliary systems, and radioactive waste systems account for 1 to 15 percent of that needed for condenser cooling (DOE 1995b:4-510).

Water withdrawal from adjacent bodies of water for plants with closed-cycle cooling systems is 5 to 10 percent of that with once-through cooling systems, with much of this water being used for makeup of water by evaporation. With once-through cooling systems, evaporative losses are about 40 percent less but occur externally in the adjacent body of water instead of in the closed-cycle system. The average makeup water withdrawals for several of the more recently constructed plants having closed-cycle cooling, normalized to 1,000 megawatts, are about 53,000 to 68,000 liters (14,000 to 18,000 gallons) per minute. Variation is due to cooling tower design, concentration factor of recirculated water, climate at the site, plant operating conditions, and other plant-specific factors. Consumptive loss normalized to 1,000 megawatts is about 42,400 liters (11,200 gallons) per minute, which is about 80 percent of the water volume taken in (DOE 1995b:4-510).

These consumptive water losses remove surface water from other uses downstream. In those areas experiencing water availability problems, nuclear power plant consumption may conflict with other existing or potential closed-cycle uses (e.g., municipal and agricultural water withdrawals) and in-stream uses (e.g., adequate in-stream flows to protect aquatic biota, recreation, and riparian communities) (DOE 1995b:4-510).

Some CLWRs use groundwater as an additional source of water. The rate of usage varies greatly among users. Many plants use groundwater only for the potable water system and require less than 380 liters (100 gallons) per minute; however, withdrawals at other sites can range from 1,500 to 11,000 liters (400 to 3,000 gallons) per minute (DOE 1995b:4-510).

3.5.5 Geology and Soils

The physiography of a CLWR site could range from a flat nearly featureless plain to a highly dissected plain of arid to humid environments. The geology could range from alluvium to thick sequences of unconsolidated marine sediments, glaciofluvial material, and crystalline and sedimentary bedrock. These materials could range in age from Cenozoic to Precambrian (recent to over 600 million years) (DOE 1996b:3-389).

The generic CLWR site could be located in regions that may have a low to moderate seismic risk as a result of an earthquake based on historical seismic activity. The location of the nearest capable fault could range from within the site boundaries to 350 kilometers (217 miles) away from CLWR sites. The nearest known epicenter of a damaging earthquake could be approximately 350 kilometers (217 miles) from existing CLWR sites (DOE 1996b:3-389).

The CLWR sites are not within a region of active volcanism; however, a generic CLWR site could be within 164 kilometers (102 miles) of a volcano (DOE 1996b:3-389).

The CLWR sites could be located where the predominant soil types are loamy clays to gravel silty loams. These soils range from moderate to well drained soils. The erosion potential could range from minor to severe in those areas with slopes greater than 25 percent and which have been eroded in the past. Shrink-swell potential could range from low to severe, which is acceptable for standard construction techniques, depending upon the engineering controls employed. Wind erosion potential ranges from minor to severe (DOE 1996b:3-389).

3.5.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species. The nature of these resources in the vicinity of a CLWR is highly dependent upon the specific location of an existing reactor. All CLWR sites were developed within the requirements of applicable Federal and state natural resource laws and regulations.

3.5.6.1 Terrestrial Resources

Terrestrial resources in the vicinity of a generic CLWR site would include those plant and animal communities typical of the ecoregion within which the facility is located. Ecoregions are characterized by distinctive flora, fauna, climate, landform, soil, vegetation, and ecological climax (Bailey 1976). Within such a region, ecological relationships between plant species, and soil and climate are essentially similar. Provinces are subdivisions that are a broad vegetation region with the same type or types of zonal soils. Provinces within which a CLWR could be located may include, but are not limited to the eastern deciduous forest; southeastern mixed forest; outer coastal plain forest; prairie parkland; Great Plains short-grass prairie; tall-grass prairie;

American desert; and California chaparral. These provinces are further subdivided by Bailey (1976) based on specific climax vegetation.

3.5.6.2 Wetlands

Since the need for cooling water is an important operational requirement, most CLWRs are constructed near rivers, lakes, reservoirs, or oceans. In each case, the presence of wetlands in the vicinity of the faciltiy may be expected. Major types of wetlands, which could occur near a generic CLWR, include tidal salt marshes, freshwater marshes, northern peatlands, shrub swamps, and forested wetlands. Wetlands serve a variety of important functions including maintaining water quality, controlling floodwaters, stabilizing shorelines, and providing recreational uses such as hunting and fishing. Wetlands are also important in providing habitat for terrestrial and aquatic organisms including migratory birds and threatened and endangered plants and animals.

3.5.6.3 Aquatic Resources

Nearly all CLWRs are constructed near a source of cooling water such as a river, lake, reservoir, or ocean. The abiotic and biotic characteristics of each type of water body vary with its geographic location.

3.5.6.4 Threatened and Endangered Species

Threatened and endangered species could be present in each of the ecoregions within which a generic CLWR could be located. At present, there are 1,233 federally listed threatened and endangered species in the United States (FWS 2000). States also typically identify threatened and endangered, as well as other special status species, found within their borders. Endangered plants and animals often rely on sensitive environments, such as wetlands, for habitat. Critical habitats, areas that are considered essential to the conservation of a species and that could require special management considerations or protection, can be designated and protected under the Endangered Species Act. Protection of threatened and endangered species and their habitat is important for maintaining biodiversity.

3.5.7 Cultural and Paleontological Resources

Cultural and paleontological resources include prehistoric resources, historic resources, Native American resources, and paleontological resources. The presence or absence of such resources is highly dependent upon the location of a specific existing CLWR. In accordance with applicable Federal and state laws and regulations, all existing sites would have been surveyed for such resources prior to site construction. Further, consultation with the State Historic Preservation Officer and tribal governments would have been required.

3.5.7.1 Prehistoric Resources

Prehistoric resources in the vicinity of the generic CLWR may include sites, districts, or isolated artifacts. Archaeological sites may represent occupation during the Archaic through later prehistoric periods and can include hunting and butchering sites, cemeteries, campsites, and tool manufacturing areas. They may yield artifacts such as stone tools and associated manufacturing debris, and ceramic potsherds. Some sites may be included on the National Register of Historic Places, while others may be eligible for listing.

3.5.7.2 Historic Resources

Historic resources may include cemeteries, remains of commercial or residential structures, or standing structures. While some sites may already be listed on the National Register of Historic Places, others may be eligible for listing.

3.5.7.3 Native American Resources

Native American resources can include cemeteries, geological or geographic elements such as mountains or creeks, certain species of animals or plants, architectural structures, such as pueblos; battlefields, or trails. Such resources are important to Native American groups for religious or historical reasons.

3.5.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age. Paleontological remains consist of fossils and their associated geological information. The presence of such resources at a generic CLWR site is dependent upon the past geologic history of the site.

3.5.8 Socioeconomics

The CLWR site could potentially affect the socioeconomic environment of a given regional economic area or region of influence. The characteristics of the regional economic area, region of influence, and community are dependent upon geographic location. For employment and income, the economic area would be based upon industry interaction and linkages in the region. The anticipated residential distribution of project-related employees and their families would determine the region of influence. This region of influence would contain all principal jurisdictions and school districts likely to be affected by the proposed activity.

Socioeconomic characteristics described for the generic CLWR site include employment and local economy, population and housing, and local transportation. Four hypothetical sites (A, B, C, and D) have been developed for the generic CLWR for purposes of making these characterizations. Site A, which had a nearby 1992 population of 2,604, was located about 160 kilometers (100 miles) from a large metropolitan area. Site B, with a nearby 1992 population of 5,236, was located approximately 8 kilometers (5 miles) from a small community and approximately 64 kilometers (40 miles) from a large metropolitan area. Site C, with a nearby 1992 population of 44,384, was located approximately 16 kilometers (10 miles) from a medium-size community and approximately 48 kilometers (30 miles) from a large metropolitan area. Site D, with a nearby 1992 population of 34,201 and a total urban population of more than 100,000. Statistics for employment and local economy were based on the regional economic area for each site. Statistics for the remaining socioeconomic characteristics were based on the sites' regions of influence (DOE 1996b).

3.5.8.1 Regional Economic Characteristics

Employment and regional economy statistics for each representative site's regional economic area are discussed in this section. Between 1980 and 1990, the civilian labor force in the region of economic area encompassing Site A increased 7.7 percent to the 1990 level of 4,811,800, and for Site B increased 49.6 percent to the 1990 level of 1,162,300. The civilian labor force for Site C, located near a large metropolitan area, increased 21.9 percent to the 1990 level of 862,500. The civilian labor force for Site D, located in an urbanized area, increased 9.9 percent to 254,800 persons. The 1994 unemployment rates in the two small hypothetical communities' (A and B) regional economic areas were 5.6 percent and 5.2 percent, respectively. Sites C and D had unemployment of 4.3 percent and 9.1 percent, respectively.

For the two small representative communities, the portions of total employment involving farming in the regional economic areas were about 1 percent. Governmental activities for Sites A and B represented about 12 percent and 14 percent, respectively. Manufacturing was 16 percent of the total employment for site A and 10 percent for site B. Retail trade accounted for 16 percent and 18 percent of the total sector employment for Sites A and B, respectively. Service activities represented a 30 percent share of the total employment for Sites A and B.

For Sites C and D, the portion of total employment was about 1 percent and 12 percent for farming and 11 and 15 percent for governmental activities, respectively. The nonfarm private sector activities of retail trade and services were 16 and 22 percent of total employment, respectively, for Site C and 16 and 26 percent, respectively, for Site D. Employments for manufacturing were 23 and 8 percent of total employment for Sites C and D, respectively (DOE 1996b).

3.5.8.2 Population and Housing

Between 1980 and 1994 the region of influence population increase for the two small hypothetical communities, A and B, was 6.4 percent (average annual increase of 0.5 percent) and 54.6 percent (average annual increase of 3.9 percent), respectively. The number of housing units in the region of influence increased 8.9 percent for Site A and 55.8 percent for Site B between 1980 and 1990. The 1990 region of influence homeowner vacancy rates were 1.1 and 3.9 percent, while the renter vacancy rates were 5.9 and 16.4 percent for Sites A and B, respectively.

The regions of influence surrounding Sites C and D experienced a 31.8 percent (average annual increase of 2.3 percent) and 19.8 percent (average annual increase of 1.4 percent) increase in population, between 1980 and 1994, and a 32.7 and 5.4 percent increase, respectively, in the number of housing units between 1980 and 1990.

The 1990 homeowner and renter vacancy rates were 2.0 and 8.9 percent for Site C and 1.3 and 5.6 percent for Site D (DOE 1996b).

3.5.8.3 Community Services and Local Transportation

These characteristics are dependent upon geographic location. The region of influence would determine all principal jurisdictions and school districts likely to be affected by the proposed activity. Local transportation would be the existing principal road, air, and rail networks required to support the project activities (DOE 1996b).

3.5.9 Existing Human Health Risk

3.5.9.1 Radiation Exposure and Risk

Major sources and levels of background radiation exposure to individuals in the vicinity of the CLWR site are shown in **Table 3–39.** Annual background radiation doses to individuals are expected to remain constant over time. The total dose to the population size changes as the population size changes. Background radiation doses are unrelated to CLWR site operations.

Releases of radionuclides to the environment from CLWR site operations provide another source of radiation exposure to individuals in the vicinity of CLWR sites. Types and quantities of radionuclides released from CLWR site operations are listed in the annual radiological effluent release reports for the reference sites. The doses to the public resulting from these releases are presented in **Table 3–40.** These doses fall within radiological guidelines and limits (10 CFR Part 50, Appendix I, and 40 CFR Part 190) and are small (less than 0.01 percent) in comparison to background radiation.

Based on a risk estimator of 500 cancer deaths per 1 million person-rem to the public, the latent cancer fatality risk to the maximally exposed member of the public due to radiological releases from operations at the CLWR site is estimated to range from 3.9×10^{-9} to 7.0×10^{-7} per year. That is, the estimated probability of this person

Table 3–39 Sources of Radiation Exposure to Individuals in the Vicinity Unrelated to Operation at the CLWR Site

Source	Effective Dose Equivalent (millirem per year)
Natural background radiation	
Cosmic radiation	27 to 29
Cosmogenic radiation	1
External terrestrial radiation	29 to 30
Radon in homes (inhaled)	200
Internal terrestrial radiation	39
Other background radiation	
Diagnostic x rays and nuclear medicine	53
Weapons test fallout	Less than 1
Air travel	1
Consumer and industrial products	10
Total	361 to 364

Note: Value of radon is an average for the United States.
Source: DOE 1996b.

Table 3–40 Radiation Doses to the Public from Normal Operation in 1994 at the Generic Existing CLWR Site (Committed Effective Dose Equivalent)

	Atmospheric Releases		Liquid Releases		Total	
Members of the Public	**Standard[a]**	**Actual**	**Standard[a]**	**Actual**	**Standard[a]**	**Actual[b]**
Maximally exposed individual (millirem)	5	0.0013 to 1.10	3 per reactor	0 to 0.29	25	0.0078 to 1.39
Population within 80 kilometers[c] (person-rem)	None	0.016 to 13.3	None	0 to 1.28	None	0.020 to 13.3
Average individual within 80 kilometers [d] (millirem)	None	6.3×10^{-5} to 6.8×10^{-3}	None	0 to 8×10^{-4}	None	7.9×10^{-5} to 6.8×10^{-3}

a. The standards for individuals are given in 10 CFR Part 50, Appendix I, and 40 CFR Part 190. As discussed in Appendix I of 10 CFR Part 50, the 5-millirem-per-year value is an airborne emission guideline, and the 3-millirem-per-year per reactor value is a liquid release guideline. Meeting these guideline values serves as a numerical demonstration that doses are as low as is reasonably achievable. The total dose of 25 millirem per year is the limit from all pathways combined as given in 40 CFR Part 190.
b. Totals cannot be obtained by summing the atmospheric and liquid release components since these component entries can be for different reactor sites.
c. This population ranges from 252,000 to 1,960,000.
d. Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site.

Source: DOE 1996b:3-398.

dying of cancer at some point in the future from radiation exposure associated with 1 year of CLWR site operation ranges from about 4 in 1 billion to 7 in 10 million. Note that it takes several to many years from the time of exposure to radiation for a cancer to manifest itself.

Based on the same risk estimator, a range of 1.0×10^{-5} to 6.7×10^{-3} excess fatal cancers is projected in the population living within 80 kilometers (50 miles) of the CLWR site from normal operations. To place these numbers into perspective, they can be compared with the numbers of fatal cancers expected in these populations from all causes. The 1990 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year. Based on this mortality rate, the number of fatal cancers expected from all causes in the population living within 80 kilometers (50 miles) of the CLWR site ranged from 505 to 3,920. These numbers of expected fatal cancers are much higher than the estimated range of 1.0×10^{-5} to 6.7×10^{-3} fatal cancers that could result from operations at the CLWR site.

At the CLWR site, workers receive the same dose as the general public from background radiation but also receive an additional dose from working at the site. The range of the average worker and total worker dose from operations at the generic existing CLWR site are presented in **Table 3–41**. These doses fall within radiological regulatory limits (10 CFR Part 20). Based on a risk estimator of 400 fatal cancers per 1 million person-rem among workers, the number of excess fatal cancers to CLWR site workers from operations is estimated to range from 0.16 to 0.34 per year (DOE 1996b).

Table 3–41 Annual Doses to Workers from Normal Operation at the Generic CLWR Site (Committed Effective Dose Equivalent)

Occupational Personnel	Onsite Releases and Direct Radiation	
	Standard[a]	Actual
Average worker (millirem)	ALARA[b]	114 to 322
Total workers[c] (person-rem)	ALARA	396 to 854

a. NRC's goal is to maintain radiological exposures as low as is reasonably achievable.
b. As low as is reasonably achievable.
c. The number of badged workers ranges from 2,650 to 4,370.
Source: DOE 1996b:3-399.

3.5.9.2 Chemical Environment

The background chemical environment important to human health consists of the atmosphere, which may contain hazardous chemicals that can be inhaled; drinking water, which may contain hazardous chemicals that can be ingested; and other environmental media with which people may come in contact (e.g., surface waters during swimming and soil through direct contact or via the food pathway).

Carcinogenic Effects. Health effects in this case are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogen. This could be incremental or excess individual lifetime cancer risks.

Noncarcinogenic Effects. Health effects in this case are determined by the ratio between the calculated or measured concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the Hazard Quotient. Hazard Quotients for noncarcingoens are summed to obtain the Hazard Index. If the Hazard Index is less than 1, no adverse health effects would be expected.

Effective administrative and design controls that decrease hazardous chemical releases to the environment and help achieve compliance with permit requirements, for example, air emissions and NPDES permit requirements contribute toward minimizing potential health impacts to the public. The effectiveness of these controls is verified through the use of monitoring information and inspection of mitigation measures. Health impacts to the public may occur during normal operations at CLWR sites via inhalation of air containing hazardous chemicals released to the atmosphere by site operations. Risks to public health from other possible pathways, such as ingestion of contaminated drinking water or direct exposure, are low relative to the inhalation pathway.

Exposure pathways for CLWR site workers during normal operation may include inhaling the workplace atmosphere and direct contact with hazardous material associated with work assignments. Occupational exposure varies from facility to facility and from worker to worker, and available information is not sufficient to allow a meaningful estimation and summation of these impacts. However, workers are protected from hazards specific to the workplace through appropriate training, protective equipment, monitoring, and management controls. At the CLWR site, workers are also protected by adherence to OSHA and EPA standards that limit workplace atmospheric and drinking water concentrations of potentially hazardous

chemicals. Appropriate monitoring that reflects the frequency and amounts of chemicals used in the operational processes ensures that these standards are not exceeded (DOE 1996b).

3.5.9.3 Health Effects Studies

CLWRs have been operating for many years. Site-specific epidemiological studies may be available, and these studies would be reviewed for specific CLWR locations. Epidemiologic studies will be considered in the future.

3.5.9.4 Accident History

CLWRs have been operating in the United States for many years. Accident information for these reactors, where applicable, can be found in documentation available from NRC.

3.5.9.5 Emergency Preparedness

The CLWR site would have an NRC-approved emergency management program that would be activated in the event of an accident. The programs are compatible with other Federal, state, and local plans and are thoroughly coordinated with all interested groups.

3.5.10 Environmental Justice

As discussed in Appendix K, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority and low-income populations. The Executive order does not alter prevailing statutory interpretations under NEPA or existing case law. Regulations prepared by the Council on Environmental Quality remain the foundation for preparing environmental documentation in compliance with NEPA (40 CFR Parts 1500 through 1508) and the Council's guidelines for inclusion of environmental justice under NEPA (CEQ 1997). As the present document is a programmatic EIS, environmental justice issues would be addressed in a site-specific EIS if an option using a CLWR were to be selected.

3.5.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing CLWR activities. The waste is managed using appropriate treatment, storage, and disposal technologies, and in compliance with all NRC and other applicable governmental regulations.

3.5.11.1 Waste Inventories and Activities

The amounts of waste generated are reported on a quarterly basis by each nuclear utility. The waste volumes of CLWRs are given in **Table 3–42**. These volumes are based on site-specific data (DOE 1996b and DOE 1999o). Because high-level radioactive waste as defined by DOE Order 435.1 would not be generated by neptunium-237 target irradiation activities at the generic CLWR site, it is not included in this table or discussed any further in this section.

Waste management and activities specific to each category of waste are discussed in the following sections.

Table 3–42 Existing Pressurized-Light Water Reactor Site Waste Management Characteristics

Characteristic	Range	Average
Low-level radioactive waste shipped (cubic meters per year)	57.04 to 636.85	178.22
Number of low-level radioactive waste shipments per year	6.00 to 31.00	16.17
Stored mixed low-level radioactive waste per 1,000 megawatt (cubic meters per year)	Not reported	101.90[a]
Hazardous waste generation (cubic meters per year)	11.4 to 29	23
Nonhazardous waste generation (cubic meters per year)		
Liquids	682 to 60,794	37,072
Solids	909 to 10,400	4,148

a. This is the average of both pressurized-water reactors and boiling-water reactors. A value was not specifically reported for the pressurized-water reactor category.

Note: To convert from cubic meters to cubic yards, multiply by 1.308.

Source: DOE 1996b:3-401 for low-level radioactive waste and mixed low-level radioactive waste; DOE 1999o:chap. 3 for hazardous and nonhazardous waste.

3.5.11.2 Transuranic Waste

Transuranic elements are contained within spent nuclear fuel. Transuranic waste is not generated or managed at CLWR sites.

3.5.11.3 Low-Level Radioactive Waste

Liquid low-level radioactive waste generated in CLWRs could be classified as either clean waste, dirty waste, turbine building floor drain water, or steam generator blowdown. Clean wastes come from equipment leaks and drains, certain valve and pump seal leakoffs not collected in the reactor coolant drain tank, and other aerated leakage sources. Primary coolant is also considered a clean waste. Liquid wastes collected in the containment building sump, auxiliary building sumps and drains, laboratory drains, sample station drains, and other miscellaneous floor drains are termed dirty wastes because of their moderate conductivity. Clean and dirty wastes will have variable radioactivity content. Detergent wastes, which consist of laundry wastes and personnel and equipment decontamination wastes, normally have a low radioactivity content. Turbine building floor drain water usually exhibits high conductivity with low radionuclide content. Depending on the amount of primary-to-secondary leakage, steam generator blowdown could have relatively high concentrations of radionuclides. The chemical and radionuclide content of the waste would determine the type and degree of treatment before storage for reuse or discharge to the environment. Operating plants have steadily increased the degree of processing, storing, and recycling of liquid radioactive waste (DOE 1996b:3-402).

Solid low-level radioactive waste is generated by removal of radionuclides from liquid waste streams, filtration of airborne gaseous emissions, and removal of contaminated material from various reactor areas. Liquid contaminated with radionuclides comes from primary and secondary coolant systems, spent-fuel pools, decontaminated wastewater, and laboratory operations. Concentrated liquids, filter sludges, waste oils, and other liquid sources are segregated by type, flushed to storage tanks, stabilized for packaging in a solid form by dewatering, and slurried into 208-liter (55-gallon) steel drums prior to disposal. High efficiency particulate air filters are used to remove radioactive material from gaseous plant effluents. These filters are compacted and disposed of as solid low-level radioactive waste. Other solid low-level radioactive waste includes contaminated protective clothing, paper, rags, glassware, compactible and noncompactible trash, and nonfuel irradiated reactor components and equipment. Tools and other material exposed to the reactor environment would also be considered solid low-level radioactive waste. Compactible solid low-level radioactive waste

is taken to an offsite or onsite volume reduction facility before disposal. Solid low-level radioactive waste is stored in shielded prefabricated steel buildings or other facilities until suitable for disposal at an approved low-level radioactive waste disposal facility (DOE 1996b:3-402).

3.5.11.4 Mixed Low-Level Radioactive Waste

Mixed low-level radioactive waste generated by a nuclear power plant covers a broad spectrum of waste types. The vast majority of mixed waste in storage at nuclear plants is chlorinated fluorocarbons and waste oil. Mixed low-level radioactive waste is stored on site until treatment and disposal is available at an offsite RCRA-permitted facility. Because of the occupational exposure from testing radioactive wastes to determine if they are chemically hazardous, the utilities have been looking at ways to eliminate, or at least minimize, the generation of mixed wastes. These efforts include removing and separating hazardous constituents from radioactive streams by remote methods; minimizing the use of solvents exposed to the reactor environment; relying on substitute processes; and recycling and reusing cleaning materials, resins, and waste oils (DOE 1996b). Stored mixed low-level radioactive waste per 1,000 megawatt averages about 100 cubic meters (130 cubic yards) per year for the existing plants studied.

3.5.11.5 Hazardous Waste

Hazardous wastes are generated from nonradioactive materials such as wipes contaminated with oils, lubricants, and cleaning solvents that are used outside the reactor environment. Hazardous wastes are packaged and shipped to offsite RCRA-permitted treatment and disposal facilities.

3.5.11.6 Nonhazardous Waste

Nonhazardous wastes include boiler blowdown, water treatment wastes, boiler metal cleaning wastes, floor and yard drain wastes, storm water runoff, and sewage wastes. Depending on the design of the individual reactor, other small volumes of wastewater are released from other plant systems or combined with the cooling water discharges. Sanitary wastes that cannot be processed by onsite waste treatment systems are collected by independent contractors and trucked to offsite treatment facilities (DOE 1996b:3-402).

3.5.11.7 Waste Minimization

Because of the increased disposal costs for low-level radioactive waste, utility companies have undertaken major volume reduction and waste minimization efforts. These efforts include segregation, decontamination, minimizing the exposure of materials and tools to the contaminated environment, and sorting. Compacting, consolidating, and monitoring waste streams to reduce the volume of low-level radioactive waste requiring storage, and lessening the exposure of routine equipment to the reactor environment, have been the most effective volume reduction strategies. Current industry-wide volume reduction practices include ultra-high pressure compaction of waste drums, incineration of waste oils and resins, mobile thin-film evaporation, waste crystallization, and asphalt solidification of resins and sludges (DOE 1996b:3-400).

Nuclear power plants typically have waste minimization programs in place to minimize both the volume and cost impact of waste generation. In existing operating plants, a number of the design considerations that affect the plant waste streams are already in place, and improvements in waste management are continually being implemented. Waste minimization steps include more economical use of disposables or elimination of disposables in favor of recyclables. Process improvements aimed at more efficient use of ion exchange resins and reductions of waste streams from the waste processes are being implemented. In general, wastes generated by operating plants have been decreasing in recent years.

3.6 DOE Site for New Accelerator(s) or a New Research Reactor

Under Alternative 3, DOE would construct one or two new accelerators for medical and industrial production, plutonium-238 production, and civilian nuclear energy research and development. In addition, DOE would construct a support facility for the processing of medical and industrial isotopes and for processing associated with research and development activities. Under Alternative 4, DOE would construct a new research reactor and support facility for this same purpose. Processing activities associated with plutonium-238 production would be performed at an existing DOE facility. The new accelerator(s), research reactor, and support facility would be located at an existing DOE site. No DOE site has been identified as the location of these facilities. If DOE were to select these alternatives, a follow-on EIS would be required to select the specific DOE site where the new accelerator(s), research reactor, and support facility would be located. In that document, DOE would identify site-specific environmental conditions, as well as evaluate the environmental impacts of facility construction and operation on the DOE sites being considered.

Because it is unreasonable for this NI PEIS to analyze all DOE sites, the environmental baseline was developed for a generic DOE site description that is representative of existing DOE sites. The generic DOE analysis does not include a specific DOE site for analysis in this NI PEIS. Any existing DOE site is a potential candidate for the new accelerator(s) or research reactor to support DOE's civilian missions for nuclear research and development and isotope production. One factor that would be considered in identifying candidate DOE sites would be the availability of existing facilities and infrastructure at the sites for support of the accelerator(s), research reactor, and support facility.

3.6.1 Land Resources

Land resources include land use and visual resources. Each of these resource areas is described for the site as a whole, as well as for the proposed facility locations.

3.6.1.1 Land Use

Land may be characterized by its potential for the location of human activities (land use). Natural resource attributes and other environmental characteristics could make a site more suitable for some land uses than for others. Changes in land use may have both beneficial and adverse effects on other resources such as ecological, cultural, geological, aquatic, and atmospheric.

DOE sites range from 165 hectares (408 acres) to 350,000 hectares (865,000 acres) in size. While these sites were established for a variety of reasons including nuclear weapons research, development, production, and testing, and energy research and development, the extent of development within each site varies greatly. Facilities located within smaller sites typically occupy a greater percentage of the site than on the larger sites, where from 1 to 25 percent of the site is developed. Undeveloped portions of the sites are used as buffers and in many cases represent land that has remained largely undisturbed since it first came under the jurisdiction of the Federal government. Depending upon the site, undeveloped land may be used for forestry, grazing, wildlife management, or for ecological research. For example, a number of the sites have areas designated as National Environmental Research Parks within their borders. These areas are devoted to research by the nation's scientific community on the impact of human activities on the natural environment. Land uses bordering DOE sites varies from developed urban areas to open spaces in which forestry, wildlife management, farming, grazing, and other rural land uses predominate. Many sites have developed land use plans and recently some have released land for redevelopment by the private sector.

3.6.1.2 Visual Resources

Visual resources are natural and human-created features that give a particular landscape its character and aesthetic quality. Landscape character is determined by the visual elements of form, line, color, and texture. All four elements are present in every landscape. The stronger the influence exerted by these elements in a landscape, the more interesting the landscape.

The visual environment of DOE sites is extremely varied. Certain sites are more highly developed and located near urban areas, while others are only sparsely developed and located many miles from human settlement. Smaller sites and developed portions of larger sites would have a Bureau of Management Visual Resource Management Class IV rating, indicating that the level of change to the characteristic landscape is high and that management activities dominate the view and are the major focus of viewer attention. The Visual Resource Management rating of undeveloped portions of larger DOE sites may range from Class II to Class III. In general, these ratings are characteristic of a less developed landscape and, although management activities may be seen, they should not attract the attention of the casual observer or dominate the view. Views of developed portions of sites located within the eastern United States are often limited due to screening by vegetation and terrain. In the western Unites States vegetation is generally more sparse and in many case the landscape is relatively flat. Thus, developed portions of these sites are typically visible from greater distances. Sites located near urban areas are viewed by more people than are the more isolated sites.

3.6.2 Noise

Existing noise sources and characteristics at a DOE site where the new accelerator(s), research reactor, and support facility might be sited can be expected to be similar to existing DOE sites and are generally described as follows. Major noise emission sources include various industrial facilities, equipment, and machines (e.g., cooling systems, transformers, engines, pumps, boilers, steam vents, construction and materials-handling equipment, and vehicles). Most industrial facilities are a sufficient distance from the site boundary that noise levels at the boundary from these sources are not measurable, or are barely distinguishable from background noise levels.

Existing site-related noises of public significance would be from the transportation of people and materials to and from the site. Noise measurements taken near the site would likely indicate that noise levels are consistent with nearby land uses which are primarily rural. Noise levels along roads and access routes to the site would be higher and may result in some annoyance at residences and other noise sensitive land uses near the roads, especially during peak traffic hours.

3.6.3 Air Quality

Ambient air quality at a generic DOE site would be expected to be in compliance with the NAAQS and with the state ambient air quality regulations. A range of ambient air concentrations for criteria pollutants representative of existing DOE sites are presented in **Table 3–43**.

The primary sources of criteria air pollutants could include steam and power generation facilities, incinerators, waste processing sources, various other process sources, vehicles, temporary emissions from construction activities, and fugitive dust from coal piles, construction activities, and waste disposal operations.

Table 3–43 Comparison of Baseline Ambient Air Concentrations with NAAQS at a Generic DOE Site

Pollutant	Averaging Period	NAAQS (micrograms per cubic meter)[a]	Baseline Concentration Range (micrograms per cubic meter)
Criteria pollutants			
Carbon monoxide	8 hours 1 hour	10,000 40,000	8 to 119 27 to 265
Lead	Calendar quarter	1.5	0.05
Nitrogen dioxide	Annual	100	1 to 14
Ozone	1 hour	235	(b)
PM_{10}	Annual 24 hours	50 150	1 to 2 1 to 13
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	1 to 5 1 to 36 2 to 112

a. NAAQS (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic PM_{10} mean standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

b. Not directly emitted or monitored by the site.

Note: EPA revised the ambient air quality standards for particulate matter and ozone in 1997 (62 FR 38856, 62 FR 38652); however, these standards are currently under litigation, but could become enforceable during the life of this project.

Source: 40 CFR Part 50.

3.6.4 Water Resources

Major surface water features in the vicinity of a generic DOE site could range from seasonally ephemeral (intermittent) streams to perennial streams and rivers more characteristic of an eastern site. These surface waters would be classified and protected by regulation for specified uses (e.g., water supply, agriculture, fish and wildlife uses, recreation). Existing facilities would have NPDES permits that specify the concentrations of pollutants for liquid effluents and stormwater runoff discharged to surface waters. However, some surface waters could have been impacted from historic waste management activities. Process and sanitary effluents from existing facilities would be managed by wastewater treatment plants and/or by seepage or evaporation ponds. Routine and compliance monitoring of discharges would be conducted with results reported in annual site environmental reports. Some generic DOE site locations could potentially be affected by flooding (DOE 1996b:3-115, 3-194–3-196).

Groundwater could occur in aquifers comprised of strata ranging from interbedded volcanic rocks and sediments to sedimentary rocks consisting of limestone, sandstone, siltstone, and shale. Classifications of major aquifers range from Class I to Class II. Groundwater could also occur as perched groundwater. Depth to groundwater could average from about 60 to 300 meters (200 to 1,000 feet) at a western generic DOE site to about 5 to 9 meters (16 to 30 feet) beneath eastern sites. Portions of some aquifer systems, and perched groundwater tables, could have been impacted by radiological and nonradiological contaminants. Like surface waters, routine monitoring of groundwater would be conducted with results reported in annual site environmental reports (DOE 1996b:3-115–3-117, 3-196–3-199).

Water supply for a generic DOE site could be obtained from either surface water or groundwater sources (DOE 1996b:3-115, 3-194).

3.6.5 Geology and Soils

The physiography of a generic DOE site could range from the high, flat to rolling plateaus and plains underlain by nearly horizontal rock strata of the western physiographic provinces to the alternating valleys and ridges

comprised of weakly to strongly folded strata of the eastern Valley and Ridge physiographic province. Surficial geology could range from relatively young (Miocene to Holocene) strata consisting of interlayered volcanic rocks (basalt, rhyolite) and unconsolidated sediments to the older (Cambrian to Ordovician), consolidated sedimentary rocks (limestone, sandstone, shale) of the eastern valleys and ridges (DOE 1996b:3-121–3-123, 3-200; 1999e:3-69, 3-70).

The generic DOE site could be located in regions that may have a low to moderate seismic risk as a result of an earthquake based on historical seismic activity. The location of the nearest capable fault could range from about 19 kilometers (12 miles) to more than 480 kilometers (298 miles) (DOE 1996b:3-200; 1999e:3-70, 3-71). The nearest known center of a potentially damaging earthquake to an accelerator(s) or research reactor at a generic DOE site could range from less than 10 kilometers (6 miles) to more than 100 kilometers (62 miles) away (DOE 1996b:3-200; 1999e:3-70, 3-71). New seismic hazard maps have been developed as part of the National Seismic Hazard Mapping Project which have been adapted for use in the new *International Building Code* (ICC 2000) (Figures 1615(1) and 1615(2) in the code) (see Section 3.2.5.1). These maps depict maximum considered earthquake ground motion of 0.2- and 1.0-second spectral response acceleration, respectively, based on a 2 percent probability of exceedance in 50 years. Based on these maps, a generic DOE accelerator(s) or reactor site could be located anywhere within the 0.35g to 0.60g mapping contours for a 0.2-second spectral response acceleration and the 0.10g to 0.15g contours for a 1.0-second spectral response acceleration.

Future risks of volcanic activity affecting a generic site range from a low risk in the west to no risk in the east, with the closest volcanic features occurring 20 kilometers (12 miles) away from a DOE site (DOE 1996b:3-200; 1999e:3-71).

Soil types could range from sands to loams and clays with depths ranging from shallow to deep. The soils developed from materials ranging from volcanic to sedimentary rocks including limestone, sandstone, shale, and siltstone. The soils are largely well drained. Shrink-swell potential generally ranges from low to moderate. In general, most soils are acceptable for standard construction techniques (Barghusen and Feit 1995:2.3-20, 2.8-14, 2.8-15; DOE 1996b:3-123, 3-200).

3.6.6 Ecological Resources

Ecological resources include terrestrial resources, wetlands, aquatic resources, and threatened and endangered species. The nature of these resources in the vicinity of a generic DOE site is highly dependent upon site location. Therefore, the following discussion addresses only the broad ecological characteristics of the regions within which potential DOE sites fall. If the new accelerator(s) or research reactor alternative were selected, site-specific details would be addressed in NEPA documentation tiered to this NI PEIS.

3.6.6.1 Terrestrial Resources

Terrestrial resources in the vicinity of a generic DOE site would include those plant and animal communities typical of the ecoregion within which the facility is located. Ecoregions are characterized by distinctive flora, fauna, climate, landform, soil, vegetation, and ecological climax (Bailey 1976). Within such a region, ecological relationships between plant species, and soil and climate are essentially similar. Provinces are subdivisions that are a broad vegetation region with the same type or types of zonal soils. DOE sites are located in a broad range of provinces, including, but not limited to: Eastern deciduous forest, Southeastern mixed forest, Great plains short-grass prairie, Rocky Mountain forest, Colorado plateau, and Intermountain sagebrush. These provinces are further subdivided by Bailey (1976), based on specific climax vegetation.

3.6.6.2 Wetlands

The presence of wetlands on DOE sites vary, depending upon whether the site is located in the eastern United States where rainfall is plentiful or the western part of the country where it is sparse. Wetlands are common at eastern sites and generally uncommon at western locations. Major types of wetlands which could occur at a generic DOE site include freshwater marshes, shrub swamps, and wooded swamps. Wetlands may be either permanent or intermittent depending upon local rainfall and soil conditions. The existence of manmade wetlands associated with some sites is dependent on continued site operations. Wetlands serve a variety of important functions including maintaining water quality, controlling floodwaters, stabilizing shorelines, and providing recreational uses such as hunting and fishing. They are also important in providing habitat for terrestrial and aquatic organisms including migratory birds and threatened and endangered plants and animals.

3.6.6.3 Aquatic Resources

Aquatic resources vary greatly between potential DOE sites located in the eastern United States and those located in the western part of the country. In the eastern United States, ample rainfall results in the presence of permanent water bodies varying from small streams to major rivers. Natural and manmade ponds and reservoirs are more prominent on or in the vicinity of eastern sites. Numerous species of aquatic flora and fauna occur at these sites. DOE sites located in the western United States typically experience limited rainfall and therefore, have few aquatic resources. In many cases the only water bodies present are evaporation and waste ponds, although major rivers do occur in the vicinity of some sites. Western sites typically have fewer species of aquatic organisms than eastern sites.

3.6.6.4 Threatened and Endangered Species

Threatened and endangered species could be present at a generic DOE site; however, the species involved would be highly dependent on site location. At present, there are 1,233 federally listed threatened and endangered species in the United States (FWS 2000). States also typically identify threatened and endangered, as well as other special status species, found within their borders. Endangered plants and animals often rely on sensitive environments, such as wetlands, for habitat. Critical habitats, areas that are considered essential to the conservation of a species and that could require special management consideration or protection, can be designated and protected under the Endangered Species Act. Protection of threatened and endangered species and their habitat is important for maintaining biodiversity, which is essential for full ecological function.

3.6.7 Cultural and Paleontological Resources

Cultural and paleontological resources include prehistoric resources, historic resources, Native American resources, and paleontological resources. The presence or absence of such resources at a generic accelerator(s) or research reactor site is highly dependent upon the specific location of the DOE site involved. In accordance with applicable Federal and state laws and regulations, any site selected for the accelerator(s) or research reactor would have to be surveyed before construction could begin. Also, consultation with State Historic Preservation Officers and tribal representatives would be required.

3.6.7.1 Prehistoric Resources

Prehistoric resources in the vicinity of a generic DOE site may include sites, districts, or isolated artifacts. Archaeological sites may represent occupation during the Archaic through later prehistoric periods and can include hunting and butchering sites, cemeteries, campsites, and tool manufacturing areas. They may yield

artifacts such as stone tools and associated manufacturing debris, and ceramic potsherds. Some prehistoric sites may be included on the National Register of Historic Places, while others may be eligible for listing.

3.6.7.2 Historic Resources

Historic resources potentially present on a generic DOE site include cemeteries, remains of commercial or residential structures, standing structures, or routes used by settlers during westward expansion. While some of these sites may already be on the National Register of Historic Places, others may be eligible for listing. DOE sites may also contain more recent structures of historic significance including those associated with the Manhattan Project and the Cold War era.

3.6.7.3 Native American Resources

Native American resources can include cemeteries, geological or geographic elements (such as mountains or creeks), certain species of animals or plants, architectural structures (such as pueblos), battlefields, or trails. Such resources are important to Native American groups for religious or historical reasons. Many DOE sites contain Native American resources and some sites have signed agreements with local tribes that designate certain rights to those tribes.

3.6.7.4 Paleontological Resources

Paleontological resources are the physical remains, impressions, or traces of plant or animals from a former geological age. Paleontological remains consist of fossils and their associated geological information. The presence of such resources at a generic DOE site is dependent upon the past geologic history of the site.

3.6.8 Socioeconomics

The socioeconomic characteristics of a generic DOE site will vary widely depending on whether the site is located near a large urbanized area or in a remote rural area. Statistics for employment and regional economy are defined for the regional economic area. Statistics for population, housing and community services are defined for the region of influence, which include the counties where nearly 90 percent of the DOE site's employees reside. Since the region of influence population for a generic DOE site could range from nearly 2,000,000 people for a site located in a large metropolitan area, to less than 200,000 for a site located in a small rural community, the socioeconomic impacts of the proposed action will vary immensely. The construction and operation of one or two new accelerators or a new research reactor at a generic DOE site are more likely to have an impact on housing and community services in a remote rural community than one located near a large metropolitan area. Likewise, the impacts on the regional economy and employment could also vary widely. Taking the unemployment rate into account, siting the new accelerator(s) or research reactor at a generic DOE site located in a rural area would have more of an impact on the economy than one located near a large metropolitan area.

If DOE were to select an alternative to build one or two new accelerators or a new research reactor, another EIS would be required to select the specific DOE site to locate the facility. In that document, DOE would perform a thorough evaluation of the socioeconomic impacts of the sites under consideration.

3.6.9 Existing Human Health Risk

3.6.9.1 Radiation Exposure and Risk

Major sources and levels of background radiation exposure to individuals in the vicinity of the generic site where the new accelerator(s) and research reactor could be located are shown in **Table 3–44**. Annual background radiation doses to individuals are expected to remain constant over time. The total dose to the population changes as the population size changes. Background radiation doses are unrelated to accelerator(s) or reactor site operations.

Table 3–44 Sources of Radiation Exposure to Individuals in the Vicinity Unrelated to Operation at the Accelerator(s) or Reactor Site

Source	Effective Dose Equivalent (millirem per year)
Natural background radiation	
Cosmic radiation	27 to 48
External terrestrial radiation	28 to 74
Radon in homes (inhaled)	200
Internal terrestrial radiation	40
Other background radiation	
Diagnostic x-rays and nuclear medicine	53
Weapons test fallout	Less than 1
Air travel	1
Consumer and industrial products	10
Total	360 to 427

Note: Value of radon is an average for the United States.
Source: Evans et al. 1998:4-19; Hamilton et al. 1999.

Releases of radionuclides to the environment from accelerator(s) or reactor site operations provide another source of radiation exposure to individuals in the vicinity of the site. Types and quantities of radionuclides released from accelerator(s) or reactor site operations are listed in the annual radiological effluent release reports for the reference sites. The doses to the public resulting from these releases are presented in **Table 3–45.** (The data provides a range of consequences to the public based on those associated with DOE sites whose offsite consequences are expected to bound those from the accelerator(s) or reactor site.) These doses fall within radiological guidelines and limits (DOE Order 5400.5) and are small in comparison to background radiation.

Based on a risk estimator of 500 cancer deaths per 1 million person-rem to the public, the latent cancer fatality risk to the maximally exposed member of the public due to radiological releases from operations at the accelerator(s) or reactor site is estimated to range from 1.1×10^{-8} to 2.2×10^{-6} per year. That is, the estimated probability of this person dying of cancer at some point in the future from radiation exposure associated with 1 year of accelerator(s) or reactor site operations ranges from about 1 in 90 million to 1 in 450,000. Note that it takes several to many years from the time of exposure to radiation for a cancer to manifest itself.

Based on the same risk estimator, a range of 1.2×10^{-4} to 3.0×10^{-2} excess fatal cancers is projected in the population living within 80 kilometers (50 miles) of the accelerator(s) or reactor site from normal operations. To place these numbers into perspective, they can be compared with the numbers of fatal cancers expected in these populations from all causes. The 1990 mortality rate associated with cancer for the entire U.S. population was 0.2 percent per year. Based on this mortality rate, the number of fatal cancers expected from all causes in the population living within 80 kilometers (50 miles) of the accelerator(s) or reactor site

Table 3–45 Radiation Doses to the Public from Normal Operation at the Accelerator(s) or Reactor Site (Committed Effective Dose Equivalent)

Members of the Public	Atmospheric Releases		Liquid Releases		Total	
	Standard[a]	Actual	Standard[a]	Actual[b]	Standard[a]	Actual
Maximally exposed individual (millirem)	10	0.021 to 0.73	4	0 to 2.7	100	0.021 to 4.4[c]
Population within 80 kilometers (person-rem)	None	0.23 to 12.3	None	0 to 48	None	0.23 to 60.3
Average individual within 80 kilometers (millirem)[d]	None	0.0019 to 0.014	None	0 to 0.055	None	0.0019 to 0.069

a. The standards for individuals are given in DOE Order 5400.5. As discussed in that order, the 10-millirem-per-year limit from airborne emissions is required by the Clean Air Act, and the 4-millirem-per-year limit is required by the Safe Drinking Water Act; for this NI PEIS, the 4-millirem-per-year value is conservatively assumed to be the limit for the sum of doses from all liquid pathways. The total dose of 100 millirem per year is the limit from all pathways combined. The 100-person-rem value for the population is given in proposed 10 CFR Part 834, as published in 58 FR 16268. If the potential total dose exceeds the 100-person-rem value, it is required that the contractor operating the facility notify DOE.
b. These doses are mainly from drinking water and eating fish.
c. This total dose includes a conservative value of 1 millirem per year from direct radiation exposure.
d. Obtained by dividing the population dose by the number of people living within 80 kilometers (50 miles) of the site.

Source: Evans et al. 1998:4-19; Hamilton et al. 1999.

ranged from 243 to 1,760. These numbers of expected fatal cancers are much higher than the estimated range of 1.2×10^{-4} to 0.030 fatal cancers that could result from operations at the accelerator(s) or reactor site.

At the accelerator(s) or reactor site, workers receive the same dose as the general public from background radiation but also receive an additional dose from working at the site. The range of the average worker and total worker dose from operations at the accelerator(s) or reactor site are presented in **Table 3–46**. These doses fall within radiological regulatory limits (10 CFR Part 20). Based on a risk estimator of 400 fatal cancers per 1 million person-rem among workers, the number of excess fatal cancers to accelerator(s) or reactor site workers from operations is estimated to range from 0.031 to 0.046 per year (DOE 1999d).

Table 3–46 Annual Doses to Workers from Normal Operation at the Accelerator(s) or Reactor Site (Committed Effective Dose Equivalent)

Occupational Personnel	Onsite Releases and Direct Radiation	
	Standard[a]	Actual
Average worker (millirem)	ALARA[b]	48 to 101
Total workers[c] (person-rem)	ALARA	78 to 115

a. The radiological limit for an individual worker is 5,000 millirem per year. However, DOE's goal is to maintain radiological exposure as low as is reasonably achievable. It has therefore established the Administrative Control Level of 2,000 millirem per year; the site must make reasonable attempts to maintain individual worker doses below this level.
b. As low as is reasonably achievable.
c. The number of badged workers ranges from 1,141 to 1,614.

Source: 10 CFR Section 835.202; DOE 1999d.

3.6.9.2 Chemical Environment

The background chemical environment important to human health consists of the atmosphere, which may contain hazardous chemicals that can be inhaled; drinking water, which may contain hazardous chemicals that can be ingested; and other environmental media with which people may come in contact, for example, surface waters during swimming and soil through direct contact or via the food pathway.

Carcinogenic Effects. Health effects in this case are estimated as the incremental probability of an individual developing cancer over a lifetime as a result of exposure to the potential carcinogen. This could be incremental or excess individual lifetime cancer risks.

Noncarcinogenic Effects. Health effects in this case are determined by the ratio between the calculated or measured concentration of the chemical in the air and the reference concentration or dose. This ratio is known as the Hazard Quotient. Hazard Quotients for noncarcinogens are summed to obtain the Hazard Index. If the Hazard Index is less than 1, no adverse health effects would be expected.

Effective administrative and design controls that decrease hazardous chemical releases to the environment and help achieve compliance with permit requirements, such as air emissions and NPDES permit requirements, contribute toward minimizing potential health impacts to the public. The effectiveness of these controls is verified through the use of monitoring information and inspection of mitigation measures. Health impacts to the public may occur during normal operations at the accelerator(s) or reactor site via inhalation of air containing hazardous chemicals released to the atmosphere by site operations. Risks to public health from other possible pathways, such as ingestion of contaminated drinking water or direct exposure are low, relative to the inhalation pathway.

Exposure pathways for accelerator(s) or reactor site workers during normal operations may include inhaling the workplace atmosphere and direct contact with hazardous material associated with work assignments. Occupational exposure varies from facility to facility and from worker to worker, and available information is not sufficient to allow a meaningful estimation and summation of these impacts. However, workers are protected from hazards specific to the workplace through appropriate training, protective equipment, monitoring, and management controls. At the accelerator(s) or reactor site, workers are also protected by adherence to OSHA and EPA standards that limit workplace atmospheric and drinking water concentrations of potentially hazardous chemicals. Appropriate monitoring that reflects the frequency and amounts of chemicals used in the operational processes, ensures that these standards are not exceeded. Worker health conditions at the generic existing accelerator(s) or reactor site are expected to be substantially better than required by the standards.

3.6.9.3 Health Effects Studies

Under Alternatives 3 and 4 of this NI PEIS, DOE would construct one or two accelerators or a research reactor at a generic DOE site for irradiation of targets to produce isotopes or for research. Once the specific sites are identified, DOE would review epidemiologic studies for the specific sites under consideration.

3.6.9.4 Accident History

Accelerators and research reactors have been operating in the United States for many years. Accident information for these accelerators and research reactors, where applicable, can be found in documentation available from DOE and NRC. Estimates of potential accidents and their consequences can also be found in safety analysis reports and probabilistic risk assessments prepared by the accelerator or reactor owners and filed with NRC.

3.6.9.5 Emergency Preparedness

The generic DOE accelerator(s) or reactor site would have a DOE-approved emergency management program that would be activated in the event of an accident. The programs are compatible with other Federal, state, and local plans and are thoroughly coordinated with all interested groups.

3.6.10 Environmental Justice

As discussed in Appendix K, Executive Order 12898 directs Federal agencies to address disproportionately high and adverse health or environmental effects of alternatives on minority and low-income populations. The Executive order does not alter prevailing statutory interpretations under NEPA or existing case law. Regulations prepared by the Council on Environmental Quality remain the foundation for preparing environmental documentation in compliance with NEPA (40 CFR Parts 1500 through 1508) and the Council's guidelines for inclusion of environmental justice under NEPA (CEQ 1997). Specific locations must be designated before detailed reviews of Environmental Justice can be conducted.

3.6.11 Waste Management

Waste management includes minimization, characterization, treatment, storage, transportation, and disposal of waste generated from ongoing DOE activities. The waste is managed using appropriate treatment, storage, and disposal technologies and in compliance with all applicable Federal and state statutes and DOE orders.

3.6.11.1 Waste Inventories and Activities

DOE facilities manage the following types of waste: high-level, transuranic, mixed transuranic, low-level radioactive, mixed low-level radioactive, hazardous, and nonhazardous. The volume of high-level radioactive waste currently stored in addition to expected generation for individual DOE sites ranges from 0 to about 213,000 cubic meters (280,000 cubic yards) (DOE 1997a:summary, 71). The volume of transuranic and mixed transuranic waste currently stored and projected through the year 2033 for individual DOE sites ranges from 0 to about 80,000 cubic meters (100,000 cubic yards). These volumes include estimates from environmental restoration, decontamination and decommisssioning, and future Departmental missions, such as the disposition of weapons-usable plutonium at SRS (DOE 1997e:3).

Based on current inventories and 20 year projections, the disposal volume for low-level radioactive waste for individual DOE sites ranges from 0 to about 3.5 million cubic meters (4.6 million cubic yards) and the disposal volume for mixed low-level radioactive waste for individual DOE sites ranges from 0 to about 320,000 cubic meters (420,000 cubic yards). These volumes include waste resulting from waste management operations and environmental restoration activities (DOE 1998k:app. A). Hazardous waste is generated or exists at most DOE facilities. The annual volume for individual DOE sites ranges from 0 to about 640,000 metric tons per year. These volumes include both wastewater and nonwastewater, RCRA-defined waste only (it does not include Toxic Substance Control Act regulated hazardous waste, state-regulated hazardous waste, and environmental restoration-generated hazardous waste) (DOE 1997a:summary, 80).

Waste management and activities specific to each category of waste are discussed in the following sections.

3.6.11.2 High-Level Radioactive Waste

High-level radioactive waste is the highly radioactive waste resulting from the reprocessing of spent nuclear, fuel including liquid waste produced directly in reprocessing and any solid material derived from the liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation (DOE Order 435.1). High-level waste is also a mixed waste because it contains hazardous constituents that are regulated under RCRA (DOE 1997a:1-27). Although the proposed plutonium-238 production, new medical and industrial isotope production, or new nuclear energy research and development activities would not generate high-level radioactive waste, some DOE facilities manage its transuranic waste as high-level waste. The high-level

radioactive waste Record of Decision issued on August 12, 1999 (64 FR 46661) states that immobilized high-level radioactive waste will be stored at the DOE site of generation until transfer to a geologic repository.

3.6.11.3 Transuranic and Mixed Transuranic Waste

Transuranic waste is waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years and atomic number greater than 92, except for (a) high-level radioactive waste, (b) waste that the Secretary of Energy has determined, with the concurrence of the Administrator, does not need the degree of isolation required by the disposal regulations, or (c) waste that NRC has approved for disposal on a case-by-case basis in accordance with 10 CFR Part 61. Transuranic waste is produced during reactor fuel assembly, nuclear weapons production, research and development, and spent nuclear fuel processing. The transuranic waste Record of Decision, issued on January 20, 1998 (63 FR 3629), states that DOE will develop and operate mobile and fixed facilities to characterize and prepare transuranic waste for disposal at WIPP. Each DOE site that has or will generate transuranic waste will, as needed, prepare and store its transuranic waste on site.

3.6.11.4 Low-Level Radioactive Waste

Low-level radioactive waste includes all radioactive wastes that is not classified as high-level radioactive waste, spent nuclear fuel, transuranic waste, uranium and thorium mill tailing, or waste from processed ore. Most low-level radioactive waste consists of relatively large amounts of waste materials contaminated with small amounts of radionuclides, such as contaminated equipment (e.g., gloveboxes, ventilation ducts, shielding, and laboratory equipment), protective clothing, paper, rags, packing material, and solidified sludges. Test specimens of fissionable material irradiated for research and development, only, and not for the production of power or plutonium, may be classified as low-level radioactive waste, provided the concentration of transuranics is less than 100 nanocuries per gram of waste. Low-level radioactive waste is subject to the Atomic Energy Act and is categorized as contact handled or remote handled, and as alpha or nonalpha on the basis of the types and levels of radioactivity present. However, most low-level radioactive waste contains short-lived radionuclides and generally can be handled without additional shielding or remote handling equipment (DOE 1997a:1-24). Currently, DOE sites which manage low-level radioactive waste treat and/or dispose of the waste on site or off site either at another DOE facility or commercial facility. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision issued on February 18, 2000 (65 FR 10061), states that for the management of low-level radioactive waste, minimal treatment will be performed at all sites, and disposal will continue, to the extent practicable, on site at INEEL, LANL, ORR, and SRS. In addition, Hanford and the Nevada Test Site will be available to all DOE sites for low-level radioactive waste disposal.

3.6.11.5 Mixed Low-Level Radioactive Waste

Mixed low-level radioactive waste contains both hazardous and low-level radioactive components. The hazardous component in mixed low-level radioactive waste is subject to RCRA, whereas the radioactive components are subject to the Atomic Energy Act (42 U.S.C. 2011 et seq.). Mixed low-level radioactive waste is characterized as either contact handled or remote handled and as alpha or nonalpha. Mixed low-level radioactive waste results from a variety of activities, including the processing of nuclear materials used in nuclear weapons production, and energy research and development activities. Although there are some commercial and DOE treatment facilities available, commercial and DOE facilities are insufficient to treat DOE's inventory of mixed low-level radioactive waste (DOE 1997a:1-24). Most of DOE's mixed low-level radioactive waste is being stored on site awaiting the development of treatment methods. DOE is subject to the requirements mandated by the Federal Facility Compliance Act of 1992, and most DOE facilities that currently store or generate mixed low-level radioactive waste have either a state-approved or EPA

region-approved Site Treatment Plan or another type of agreement. Each Site Treatment Plan or agreement requires treatment of mixed waste, including mixed low-level radioactive waste, in accordance with its provisions.

The low-level radioactive waste and mixed low-level radioactive waste Record of Decision, issued on February 18, 2000 (65 FR 10061), states that mixed low-level radioactive waste will be treated at Hanford, INEEL, ORR, and SRS and disposed of at Hanford and the Nevada Test Site.

3.6.11.6 Hazardous Waste

The quantities and types of hazardous waste generated as a result of DOE activities vary considerably and include acids, metals, industrial solvents, paints, oils, rags contaminated with hazardous cleaning compounds, and other hazardous materials that are byproducts of routine maintenance and operations. About 99 percent of DOE's hazardous waste is wastewater and is treated at DOE sites. Treatment residues and the remaining 1 percent, predominantly solvents and cleaning agents, are treated at commercial facilities. The hazardous waste Record of Decision, issued on August 5, 1998 (63 FR 41810), states that most DOE sites will continue to use offsite facilities for treatment and disposal of major portions of the nonwastewater hazardous waste, with ORR and SRS continuing to treat some of their own nonwastewater hazardous waste on site in existing facilities, where this is economically favorable.

3.6.11.7 Nonhazardous Waste

Nonhazardous and nonradioactive sanitary and industrial waste requires limited handling and can be treated or disposed of in properly designed facilities or used in energy production. DOE currently manages sanitary and industrial waste on a site-by-site basis. Some DOE sites dispose of this waste in onsite landfills that have permits issued by appropriate State agencies, while other sites use commercial landfills (DOE 1997a:1-29).

3.6.11.8 Waste Minimization

The DOE complex-wide waste reduction goals for achievement by December 31, 1999 (compared to the 1993 Baseline) are to reduce low-level radioactive waste, mixed low-level radioactive waste, and hazardous waste generation by 50 percent, and nonhazardous waste by 33 percent (DOE 1999f:1).

3.6.12 Spent Nuclear Fuel

The operation of the new reactor will generate nuclear spent fuel at a rate of about 0.31 metric tons of heavy metal per year (Appendix E of this NI PEIS). The Nuclear Waste Policy Act of 1982, as amended, assigned the Secretary of Energy the responsibility for the development of a repository for the disposal of high-level radioactive waste and spent nuclear fuel. When such a repository is available, spent nuclear fuel would be transferred for disposal from the nuclear reactor site to the repository. Until a repository is available, spent nuclear fuel generated from the operation of the new reactor is expected to be stored on site in the reactor spent fuel pool, which provides the capacity for spent fuel generated from 35 years of operation.

3.7 REFERENCES

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10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," U.S. Nuclear Regulatory Commission.

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Chapter 4
Environmental Consequences

The impact analyses in Chapter 4 focus on those areas where the potential exists for effects on the environment. Each of the options, including the four options under the No Action Alternative, the six options under the Restart Fast Flux Test Facility Alternative, the nine options under the Use Only Existing Operational Facilities Alternative, the three options under the Construct New Accelerator(s) Alternative, the three options under the Construct New Research Reactor Alternative, and the one option under the Permanently Deactivate Fast Flux Test Facility (with No New Missions) Alternative, is discussed separately in Sections 4.2 through 4.7. The cumulative impacts associated with the alternatives are presented in Section 4.8. A detailed discussion of each alternative is given in Chapter 2; a comparison of the environmental effects among alternatives and among options within alternatives is presented in Section 2.7.1.

4.1 INTRODUCTION

In this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]), the impact analyses assess all disciplines where the potential exists for effects on the environment, as follows:

- Land resources
- Noise
- Air quality
- Water resources
- Geology and soils
- Ecological resources
- Cultural and paleontological resources
- Socioeconomics
- Public and occupational health and safety (associated with normal operations, facility accidents, and transportation)
- Environmental justice
- Waste management
- Spent nuclear fuel management

These disciplines are analyzed in a manner commensurate with their importance under a specific option—the sliding-scale assessment approach. For example, under all options of Alternative 2 (Use Only Existing Operational Facilities), the U.S. Department of Energy (DOE) has determined that minimal or no impacts would be associated with land resources, noise, water resources, geology and soils, ecological resources, and cultural and paleontological resources. This is because existing facilities in developed areas would be used, no new land disturbance would take place, proposed activities would be consistent with current operations, and wastewater discharges would continue through permitted outfalls. Therefore, impacts associated with these resources are assessed in less detail. Where construction and decommissioning are integral parts of an option (all options under both Alternative 3 [Construct New Accelerator(s)] and Alternative 4 [Construct New Research Reactor]), the impacts associated with such construction and decommissioning are included in the assessments, and disciplines such as land resources and noise are assessed in more detail. The sliding-scale assessment approach has been applied in the evaluation of all the options addressed in this NI PEIS.

The environmental consequences associated with the alternatives assessed in this NI PEIS were generally calculated using appropriate computer models and by applying facility operational characteristics from

Appendixes A through F. The analyses were performed in accordance with the impact assessment methods described in Appendix G. More detailed descriptions of the development of the impacts for some resource areas are presented in Appendixes H through L, as follows:

- Appendix H, Evaluation of Human Health Effects from Normal Facility Operations
- Appendix I, Evaluation of Human Health Effects from Facility Accidents
- Appendix J, Evaluation of Human Health Effects of Transportation
- Appendix K, Environmental Justice Analysis
- Appendix L, Socioeconomic Analysis

The results of the assessments of environmental consequences associated with the various alternatives and their options are presented in this chapter. For brevity, numerical results are often rounded. Portions of some alternative options are equivalent. For example, for Alternative 2 (Use Only Existing Operational Facilities), the Radiochemical Engineering Development Center (REDC) at the Oak Ridge Reservation (ORR) would be used to fabricate and process the neptunium-237 targets under Options 1, 4, and 7. Therefore, the activities at REDC would be virtually the same for these three alternative options. The organization of Chapter 4 takes advantage of these equivalencies. When the impacts have already been described for a previous alternative or alternative option, the later impacts discussion provides a reference to the earlier section, rather than repeating the information.

4.2 NO ACTION ALTERNATIVE

Under the No Action Alternative (maintain status quo), Fast Flux Test Facility (FFTF) would be maintained in standby status for all or a portion of the 35-year evaluation period for operations covered in this NI PEIS. For the purpose of analysis in this NI PEIS, the maximum period of 35 years was assumed. Ongoing operations at existing facilities as described in Chapter 3, Affected Environment, would continue under this alternative. DOE would not establish a domestic plutonium-238 production capability, but could, instead, continue to purchase Russian plutonium-238 to meet the needs of future U.S. space missions. For the purpose of analysis in this NI PEIS, DOE assumed that it would continue to purchase plutonium-238 to meet the space mission needs for the 35-year evaluation period. DOE would continue its medical and industrial isotope production and civilian nuclear energy research and development activities at the current operating levels of existing facilities. A consequence of a No Action decision would be the need to determine the future of the neptunium-237 stored at the Savannah River Site (SRS). Therefore, the impacts of possible future transportation and storage of neptunium-237 are evaluated as part of the No Action Alternative. Four options are identified. If DOE decides not to establish a domestic plutonium-238 production capability in the future, the neptunium-237 would have no programmatic value and Option 1 would be selected. Conversely, if DOE decides to maintain the capability to establish a domestic plutonium-238 capability in the future, the inventory of neptunium-237 must be retained. In this case, Option 2, 3, or 4 could be selected.

- **Option 1.** Under this option, DOE would reconsider its stabilization strategy for the neptunium-237, currently stored in solution form at SRS, possibly leading to final disposition. The current plan is to stabilize the material to oxide, as described in the Supplemental Record of Decision for the *Final Environmental Impact Statement, Interim Management of Nuclear Materials at SRS* (62 FR 61099). This Record of Decision would be amended or new NEPA analysis performed, if necessary.

- **Options 2 through 4.** Under these options, the neptunium-237 oxide would be transported from SRS to one of three candidate DOE sites for up to 35 years of storage. For the purpose of analysis in this NI PEIS, the maximum period of 35 years was assumed. Option 2 would provide storage at the Oak Ridge National Laboratory's (ORNL) REDC facility, Option 3 at Idaho National Engineering and Environmental Laboratory's (INEEL) Building CPP–651, and Option 4 at the Hanford Site's (Hanford) Fuels and Materials Examination Facility (FMEF).

Each of the four options under the No Action Alternative includes importing plutonium–238 from Russia and maintaining FFTF in standby. Option 1 includes no other activities, whereas the other three options include the transportation of neptunium-237 from SRS to, and storage at, another DOE site. Under Option 2, this neptunium would be stored at ORR in REDC, under Option 3 at INEEL in Building CPP–651, and under Option 4 at Hanford in FMEF.

4.2.1 No Action Alternative—Option 1

Under Option 1, the United States would continue to purchase the plutonium-238 from Russia that is needed to fabricate radioisotope power systems for future U.S. space missions. As part of this option, FFTF at Hanford would be maintained in standby. This option does not include the transportation of neptunium-237 from SRS and its storage at another DOE site, as do the other three options under the No Action Alternative.

4.2.1.1 Importation of Plutonium-238 from Russia

Activities and impacts associated with transporting plutonium-238 to the United States from Russia are evaluated in two other NEPA documents: *Environmental Assessment of the Import of Russian Plutonium-238* (*Russian Plutonium-238 EA*) (DOE 1993a), and *Finding of No Significant Impact for Import of Russian*

Plutonium-238 Fuel (DOE 1993b). The proposed action in the *Russian Plutonium-238 EA* is to import up to 40 kilograms (88 pounds) of plutonium-238 fuel (isotopic mass) in dioxide form from Russia to supplement the current U.S. inventory. The action includes the transportation by ship of Russian plutonium-238 in 5-kilogram (11-pound) increments from St. Petersburg, Russia, to a U.S. port of entry. From the U.S. port of entry, the plutonium-238 would be ground transported by DOE safe, secure trailer/SafeGuards Transport (SST/SGT) to Los Alamos National Laboratory (LANL) in New Mexico and would be added to LANL's portion of the existing U.S. plutonium-238 inventory. As of November 2000, two shipments have been safely and securely transported to LANL.

The dose to transportation workers associated with importing 40 kilograms (88 pounds) of plutonium-238 to LANL was reported to be 2.6 person-rem; the dose to the public would be 4.5 person-rem. Accordingly, incident-free transportation of plutonium-238 would result in 0.0011 latent cancer fatality among transportation workers and 0.0023 latent cancer fatality in the total affected population over the duration of the transportation activities. The number of nonradiological fatalities from vehicular emissions was not reported (DOE 1993a).

The reported transportation accident risks under this option are as follows: a radiological dose to the population of 0.2 person-rem, resulting in 1.0×10^{-4} latent cancer fatality; and traffic accidents resulting in 0.0032 traffic fatality. These estimates include the risk to the crew, handlers, and the public during both ocean and highway transportation. DOE considered the environmental consequences on global commons (i.e., portions of the ocean not within the territorial boundary of any nation) in accordance with Executive Order 12114 (44 FR 1957) (DOE 1993a).

The risk estimated for importing 40 kilograms (88 pounds of plutonium-238) can be scaled to estimate the risk of importing 175 kilograms (5 kilograms per year times 35 years) (385 pounds) of plutonium-238 over the 35-year period covered by this NI PEIS. Approximately 35 shipments of plutonium-238 would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 114,000 kilometers (71,000 miles); and at sea by ships carrying plutonium-238, 298,000 kilometers (161,000 nautical miles).

The transportation impacts analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 12 person-rem; the dose to the public would be 20 person-rem. Accordingly, incident-free transportation of radioactive material would result in 0.0046 latent cancer fatality among transportation workers and 0.0099 latent cancer fatality in the total affected population over the duration of the transportation activities. Latent cancer fatalities associated with radiological releases were estimated by multiplying the occupational (worker) dose by 4.0×10^{-4} latent cancer fatality per person-rem of exposure, and the public accident and accident-free doses by 5.0×10^{-4} latent cancer fatality per person-rem of exposure (ICRP 1991). The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 4.7×10^{-4}.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.88 person-rem, resulting in 4.4×10^{-4} latent cancer fatality; and traffic accidents resulting in 0.014 traffic fatality.

4.2.1.2 Maintenance of FFTF in Standby

The environmental impacts associated with maintaining FFTF in standby for 35 years are discussed in the following sections.

4.2.1.2.1 Land Resources

LAND USE. Maintaining FFTF in standby would not change land use in the 400 Area for 35 years because maintenance activities would not require the development of additional land areas. Further, maintenance activities are consistent with the site's industrial nature.

VISUAL RESOURCES. Impacts on visual resources would not change for 35 years because no new construction or modification of existing structures would be required. Since there would be no change in the appearance of FFTF, the current Visual Resource Management Class IV rating for the 400 Area would continue for 35 years.

4.2.1.2.2 Noise

Maintaining FFTF in standby would not involve any new construction, major change in activities, or change in employment. Thus, there would be no change in noise impacts on wildlife around the 400 Area or on people near Hanford and this would be expected to continue for the next 35 years to the extent it is dependent on activities at FFTF.

4.2.1.2.3 Air Quality

Maintaining FFTF in standby for 35 years would not involve any new construction, change in activities, or change in employment. Thus, there would be no change in nonradiological air quality at Hanford. Emissions from maintaining FFTF in standby would be expected to continue for the next 35 years.

4.2.1.2.4 Water Resources

Impact on water resources associated with maintaining FFTF in standby for 35 years would include the continuation of groundwater withdrawals and process and sanitary wastewater discharges associated with Hanford 400 Area facilities (**Table 4–1**). Specifically, groundwater withdrawals by 400 Area facilities (mainly FFTF) would continue to average about 197 million liters (52 million gallons) per year. The discharge of approximately 76 million liters (20 million gallons) per year of FFTF cooling water to the 400 Area process sewer system and the 400 Area Pond (i.e., 4608 B/C percolation ponds) would continue. Also, it is expected that 400 Area sanitary wastewater flows of about 3.8 million liters (1 million gallons) annually could continue to be discharged to Energy Northwest for treatment. However, as groundwater use during standby would not be expected to affect regional groundwater levels and effluents would continue to be discharged to appropriate treatment facilities, the overall impact on water resources at Hanford should be negligible (DOE 2000a:11; Nielsen 1999:38, 39, 41). Further information on current water usage, effluent discharge, and water quality at Hanford is presented in Section 3.4.4.

Table 4–1 Water Use and Wastewater Generation Associated with Maintaining FFTF in Standby Under All Options of the No Action Alternative

Indicator (million liters per year)	Hanford 400 Area
	FFTF
Water use	197
Process wastewater generation	76
Sanitary wastewater generation	3.8

Note: To convert from liters per year to gallons per year, multiply by 0.264.
Source: DOE 2000a:11; Nielsen 1999:38, 39, 41.

4.2.1.2.5 Geology and Soils

Maintaining FFTF in standby for 35 years would not involve new construction. Therefore, geologic and soil resources in the 400 Area would not be disturbed. In the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (*Storage and Disposition PEIS*) (DOE 1996a:4-45), hazards from large-scale geologic conditions at Hanford, such as earthquakes and volcanoes, were evaluated. That analysis was reviewed in the *Surplus Plutonium Disposition Final Environmental Impact Statement* (*Surplus Plutonium Disposition EIS*) (DOE 1999a:4-260). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions continue to present a low risk to FFTF. Ground shaking of Modified Mercalli Intensity V to VII (refer to Table 3–4) associated with postulated earthquakes would be expected to primarily affect the integrity of inadequately designed or nonreinforced structures. Damage to properly or specially designed or upgraded facilities would not be expected. Also, only minimal effects (e.g., ashfall) would be expected from postulated volcanic events in the Cascade Region. The potential for other nontectonic events to affect the facility is also low.

As stated in DOE Order 420.1, DOE requires that nuclear or nonnuclear facilities be designed, constructed, and operated so that the public, the workers, and the environment are protected from the adverse impacts of natural phenomena hazards, including earthquakes. DOE Order 420.1, Section 4.4, as supplemented by DOE Guide 420.1-2, stipulates the natural phenomena hazards mitigation requirements for DOE facilities and specifically provides for the reevaluation and upgrade of existing DOE facilities when there is a significant degradation in the safety basis for the facility. DOE uses the requirements of the latest model building codes and national standards to mitigate the consequences of natural phenomena hazards. Further, the natural phenomena hazards mitigation requirements of DOE Order 420.1 are consistent with the guidance for seismic design and construction contained in the National Earthquake Hazards Reduction Program 1997 provisions (BSSC 1997). In addition, DOE Guide 420.1-2 was recently issued to recognize the consolidation of the three previous U.S. model building codes, including the Uniform Building Code, into the *International Building Code* (ICC 2000). The DOE requirements for seismic engineering have followed the Uniform Building Code, unless the importance of achieving a high level of protection warrants the use of more demanding methods and criteria (DOE Guide 420.1-2). As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1.

4.2.1.2.6 Ecological Resources

Maintaining FFTF in standby for 35 years would not involve new construction or other disturbance to the natural environment. As noted in Section 4.2.1.2.2, there would be no change in noise impacts on wildlife. Impacts on wetlands and aquatic resources (associated with manmade ponds on the site) would not change because water usage and wastewater discharge would not change. Due to the developed nature of the area and because no construction would take place, impacts on threatened and endangered species would not occur.

Consultation letters concerning threatened and endangered species were sent to the U.S. Fish and Wildlife Service, the National Marine Fisheries Service, the Washington State Department of Natural Resources, and the State of Washington Department of Fish and Wildlife (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. Both the Washington State Department of Natural Resources and the State of Washington Department of Fish and Wildlife provided lists of state species of concern that occur in the vicinity of the project area. As noted above, no impacts to any threatened or endangered species are expected, including those of concern to these agencies. While DOE has made additional contacts with the U.S. Fish and Wildlife Service and the National Marine Fisheries Service, responses are pending from these agencies. Although no federally listed species are

expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at Hanford prior to the receipt of input from these Federal agencies.

4.2.1.2.7 Cultural and Paleontological Resources

Maintaining FFTF in standby for 35 years would not involve new construction and, thus, would not disturb cultural and paleontological resources in the 400 Area. No prehistoric, historic, or paleontological sites have been identified either in the 400 Area or within 2 kilometers (1.2 miles) of the 400 Area. Six buildings in the 400 Area, including two FFTF structures (the Reactor Containment Building and the FFTF Control Building), are eligible for the National Register of Historic Places as contributing properties in the Historic District recommended for mitigation. Maintaining FFTF in standby for 35 years would not affect the status of these structures. No Native American resources are known to occur in the 400 Area.

Consultation to comply with Section 106 of the National Historic Preservation Act was conducted with the State Historic Preservation Office (see Table 5–3) and resulted in concurrence by the State Historic Preservation Office that the proposed action would have no effect on historic properties at Hanford. Consultation was also conducted with interested Native American tribes that resulted in comments at public hearings by members representing the Nez Perce and Confederated Tribes of the Umatilla Indian Reservation. Responses to their specific comments are addressed in Volume 3.

4.2.1.2.8 Socioeconomics

Under the No Action Alternative, FFTF would continue to be maintained in standby for 35 years. Current employment of approximately 242 workers would be continued for the next 35 years (Nielsen 1999). No new employment or in-migration of workers would be required. Thus, there would be no additional impact on the socioeconomic conditions around Hanford.

4.2.1.2.9 Public and Occupational Health and Safety—Normal Standby Activities

Assessments of radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

RADIOLOGICAL IMPACTS. Potential radiological doses to three receptor groups are given in **Table 4–2**: the population within 80 kilometers (50 miles) of FFTF in the year 2020 (approximate midlife of the nuclear infrastructure activities assessed in this NI PEIS), the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–2 Radiological Impacts on the Public Around Hanford from Maintaining FFTF in Standby Under All Options of the No Action Alternative

Receptor	Standby
Population within 80 kilometers (50 miles) in the year 2020	
Dose (person-rem)	0.028
35-year latent cancer fatalities	4.9×10^{-4}
Maximally exposed individual	
Annual dose (millirem)	1.4×10^{-4}
35-year latent cancer fatality risk	2.4×10^{-9}
Average exposed individual within 80 kilometers (50 miles)	
Annual dose[a] (millirem)	5.7×10^{-5}
35-year latent cancer fatality risk	9.9×10^{-10}

a. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FFTF in the year 2020 (503,300).

Source: Model results, using the GENII computer code (Napier et al. 1988).

Probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991). A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers. The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

A collective dose of 0.028 person-rem would be incurred by the surrounding population in the year 2020. The corresponding number of latent cancer fatalities in this population from 35 years of maintaining FFTF in standby would be 4.9×10^{-4}. Here, and throughout this document, a latent cancer fatality value of less than one can be related to the statistical probability of a latent cancer fatality. The most probable outcome of a population dose of 0.028 person-rem would be no latent cancer fatalities. However, in a small number of cases, this dose would result in a latent cancer fatality. The lower the number of cases, the less likely is this outcome. This issue is addressed in more detail in Appendix H.

An annual dose of 1.4×10^{-4} millirem is shown for the maximally exposed individual. From 35 years of standby activities, the corresponding risk of a latent cancer fatality to this individual would be 2.4×10^{-9}.

The expected average dose to a worker involved with storage activities while FFTF is maintained in standby and the associated expected dose to the total storage workforce would be 3.5 millirem and 0.69 person-rem, respectively (refer to **Table 4–3**). The associated risk of a latent cancer fatality to the average worker from 35 years of standby activities would be 4.9×10^{-5}, and the estimated number of latent cancer fatalities in the total workforce from 35 years of operations would be 0.0097.

Table 4–3 Radiological Impacts on FFTF Workers from Maintaining FFTF in Standby Under All Options of the No Action Alternative

Receptor—No Action Workers[a]	Standby
Total dose (person-rem per year)	0.69[b]
35-year latent cancer fatalities	0.0097
Average worker dose (millirem per year)	3.5
35-year latent cancer fatality risk	4.9×10^{-5}

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with storage operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.

b. Based on an estimated 200 workers.

Source: Nielsen 1999.

HAZARDOUS CHEMICAL IMPACTS. No new chemicals would be introduced by maintaining FFTF in standby. Thus, there would be no change in impacts from emissions of hazardous chemicals. Emissions of hazardous air pollutants from maintaining FFTF in standby, would be expected to continue for the next 35 years.

4.2.1.2.10 Public and Occupational Health and Safety—Standby Accidents

In its current standby condition, FFTF is defueled with slightly radioactive sodium circulating through the primary heat transport system. A primary heat transport system sodium spill would be the accident with the highest consequences. A detailed description of the accident analysis is provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of FFTF, and a noninvolved worker at a distance of 640 meters (0.4 miles) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per unit of time (i.e., 1 year or 35 years) for an individual (the maximally exposed offsite individual or a noninvolved worker), and as the increased number of latent cancer fatalities per unit of time (i.e., 1 year or 35 years) in the offsite population.

Consequences to involved workers are addressed Appendix I, Section I.1.7.

Probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991). For low doses or dose rates, a probability coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers, and 5×10^{-4} latent cancer fatality per rem for the public. For high doses received at a high rate, probability coefficients of 8×10^{-4} and 0.001 latent cancer fatality per rem are applied for workers and the public, respectively. These higher probability coefficients apply for doses above 20 rads and dose rates above 10 rads per hour.

Potential consequences and associated risks are presented in **Tables 4–4** and **4–5**, respectively.

Table 4–4 FFTF Standby Accident Consequences Under All Options of the No Action Alternative

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
Primary heat transport system sodium spill	1.34×10^{-7}	6.70×10^{-11}	9.99×10^{-3}	4.99×10^{-6}	1.62×10^{-8}	6.48×10^{-12}

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table 4–5 FFTF Standby Accident Risks Under All Options of the No Action Alternative

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual primary heat transport system sodium spill risk (1×10^{-4})	6.70×10^{-15}	4.99×10^{-10}	6.48×10^{-16}
35-year risk	2.35×10^{-13}	1.75×10^{-8}	2.27×10^{-14}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

With FFTF in standby for 35 years, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.35×10^{-13} and 2.27×10^{-14}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 1.75×10^{-8}.

The 35-year risks are conservative because they are based on current primary sodium radioactivity levels. The radioisotopes contained in the primary sodium are sodium-22, cesium-137, plutonium-239, and tritium. Examining the current inventories, half-lives, and dose conversion factors of these isotopes, it was determined that currently 99 percent of the dose is attributable to plutonium-239 and sodium-22, with sodium-22 accounting for 78 percent of the total dose. Plutonium-239 has a 24,400-year half-life and would have only decayed 0.1 percent after 35 years. Sodium-22, however, has a 2.6-year half-life and would have decayed over 99.99 percent after 35 years. In year 35, only 21 percent of the original total dose level would remain. The annual risks would decrease each year due mainly to the radioactive decay of sodium-22. Therefore, if the annual risks were recalculated for each subsequent year based on lower activity levels and then summed for the 35-year period, the resulting risks would be lower than those presented.

Maintaining FFTF in standby would not introduce any additional operations that require the use of hazardous chemicals. Thus, there would be no postulated hazardous chemical accidents attributable to maintaining FFTF in standby.

4.2.1.2.11 Public and Occupational Health and Safety—Transportation

There would be no transportation impacts associated with maintaining FFTF in standby.

4.2.1.2.12 Environmental Justice

As discussed in other parts of Section 4.2.1.2, normal and incident-free operations required to maintain FFTF in standby pose no significant risks to the public. For 35 years of normal standby operations, FFTF would be a small contributor to baseline emissions from the Hanford Site. Chemical emissions would not be altered and no transportation impacts are associated with maintenance of FFTF in standby. As discussed in Appendix K,

under the conservative assumption that all food consumed in the potentially affected area during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to ingestion of contaminated food supplies. Radiological risks to the public due to accidents occurring at FFTF while in standby would be essentially zero. Thus, maintaining FFTF in standby would pose no disproportionately high and adverse risks to minority or low-income populations.

4.2.1.2.13 Waste Management

The expected generation rates of waste at Hanford that would be associated with maintaining FFTF in standby for 35 years are compared with Hanford's treatment, storage, and disposal capacities in **Table 4–6**. The impacts on the Hanford waste management systems, in terms of managing the waste, are discussed in this section. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.2.1.2.9 through 4.2.1.2.11.

Table 4–6 Waste Management Impacts of Maintaining FFTF in Standby Under All Options of the No Action Alternative

Waste Type[a]	Estimated Waste Generation for FFTF in Standby (cubic meters per year)	Estimated Waste Generation as a Percent of[b]		
		Onsite Treatment Capacity	Onsite Storage Capacity	Onsite Disposal Capacity
Low-level radioactive waste				
Liquid	<6	(c)	(c)	(c)
Solid	17	NA	NA	0.03
Mixed low-level radioactive waste	<0.5	NA	0.11	0.13
Hazardous	4[d]	NA	NA	NA
Nonhazardous				
Process wastewater	76,000	(c)	(c)	(c)
Sanitary wastewater	3,800	1.6[e]	NA	NA
Solid	120	NA	NA	NA

a. See definitions in Section G.9.
b. The estimated amounts of waste generated annually are compared with the annual site treatment capacities. The estimated total amounts of waste generated over the assumed 35-year operational period are compared with the site storage and disposal capacities.
c. Refer to the text.
d. Represents both liquid and solid hazardous waste.
e. Percent of capacity of the Energy Northwest Sewage Treatment Facility.

Note: To convert from cubic meters per year to cubic yards per year, multiply by 1.308; < means "less than."
Key: NA, not applicable (i.e., the majority of this waste is not routinely treated, stored, or disposed of on site; refer to the text).
Source: DOE 2000a; Nielsen 1999.

In accordance with the Records of Decision for the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste (Waste Management PEIS)* (DOE 1997a), waste could be treated and disposed of on site at Hanford or at other DOE sites or commercial facilities. Based on the Record of Decision for hazardous waste issued on August 5, 1998 (63 FR 41810), nonwastewater hazardous waste would continue to be treated and disposed of at offsite commercial facilities. Based on the Record of Decision for low-level radioactive waste and mixed low-level radioactive waste issued on February 18, 2000 (65 FR 10061), minimal treatment of low-level radioactive waste will be performed at all sites and, to the extent practicable, onsite disposal of low-level radioactive waste will continue. Hanford and the Nevada Test Site will be made available to all DOE sites for disposal of low-level radioactive waste. Mixed low-level radioactive waste analyzed in the *Waste Management*

PEIS will be treated at Hanford, INEEL, ORR, and SRS and will be disposed of at Hanford and the Nevada Test Site.

It is also assumed in this NI PEIS that low-level radioactive waste, mixed low-level radioactive waste, hazardous waste, and nonhazardous waste would be treated, stored, and disposed of in accordance with current and developing site practices. No high-level radioactive waste or transuranic waste would be associated with maintaining FFTF in standby.

Solid low-level radioactive waste associated with maintaining FFTF in standby would continue to be compacted, if possible, and packaged in appropriate containers or burial casks, certified, and transferred for disposal in the existing onsite low-level radioactive Burial Grounds.

Six hundred cubic meters (780 cubic yards) of solid low-level radioactive waste would be generated over the 35-year period as a result of maintaining FFTF in standby. This solid low-level radioactive waste represents approximately 0.03 percent of the 1.74-million-cubic-meter (2.28-million-cubic-yard) capacity of the low-level radioactive Burial Grounds. Using the 3,480-cubic-meter-per-hectare (1,842-cubic-yard-per-acre) disposal land usage factor for Hanford published in the *Storage and Disposition PEIS* (DOE 1996a:E-9), 600 cubic meters (780 cubic yards) of waste would require 0.17 hectare (0.42 acre) of disposal space at Hanford. The impacts of managing this low-level radioactive waste at Hanford would be minimal.

Maintaining FFTF in standby would result in 210 cubic meters (275 cubic yards) of liquid low-level radioactive waste over the 35-year period. Liquid low-level radioactive waste associated with maintaining FFTF in standby would continue to be stored in FFTF or the Maintenance and Storage Facility and transported, as necessary, to the 200 Area Effluent Treatment Facility for processing and disposal. This liquid low-level radioactive waste resulting from maintaining FFTF in standby represents a very small amount of waste that can be managed by the 200 Area Liquid Effluent Treatment Facility, which has a capacity of 0.57 cubic meter per minute (0.75 cubic yard per minute).

Mixed low-level radioactive waste would be stabilized, packaged, and stored on site for treatment and disposal in a manner consistent with the Tri-Party Agreement (EPA et al. 1989) for Hanford. Over the 35-year period, it is estimated that less than 18 cubic meters (24 cubic yards) of mixed low-level radioactive waste would be generated as a result of maintaining FFTF in standby. This mixed low-level radioactive waste is expected to be treated at a nearby commercial facility. This waste is also estimated to be less than 0.11 percent of the 16,800-cubic-meter (22,000-cubic-yard) storage capacity of the Central Waste Complex and less than 0.13 percent of the 14,200-cubic-meter (18,600-cubic-yard) planned disposal capacity of the Radioactive Mixed Waste Disposal Facility. Therefore, this waste would only have a minimal impact on the management of mixed low-level radioactive waste at Hanford.

Hazardous waste generated during maintaining FFTF in standby would be packaged in U.S. Department of Transportation (DOT)-approved containers and shipped off site to permitted commercial recycling, treatment, and disposal facilities. The waste load generated during the 35-year period would have only a minimal impact on the Hanford hazardous waste management system.

Nonhazardous solid waste would be packaged and transported in conformance with standard industrial practice. Solid waste such as office paper, metal cans, and plastic and glass bottles that can be recycled would be sent off site for that purpose. The remaining solid sanitary waste would be sent for offsite disposal in a municipal landfill. This waste load would have only a minimal impact on the nonhazardous solid waste management system at Hanford.

Nonhazardous process wastewater, which is composed mainly of blowdown water from the eight FFTF cooling towers and would continue to be discharged into the 400 Area Ponds. This discharge is regulated by State Waste Discharge Permit ST-4501.

Nonhazardous sanitary wastewater would continue to be discharged from the 400 Area, which is connected to the Energy Northwest Treatment System. Nonhazardous sanitary wastewater generated from maintaining FFTF in standby would represent 1.6 percent of the 235,000-cubic-meter-per-year (307,000-cubic-yard-per-year) capacity of the Energy Northwest Sewage Treatment Facility.

The generation rates of waste at Hanford that are associated with maintaining FFTF in standby (refer to Table 4–6) can be compared with the current total waste generation rates at the site, given in Table 3–34 (Section 3.4.11). The waste generation rates associated with maintaining FFTF in standby is a small fraction of the current total waste generation rates at the site.

4.2.1.2.14 Spent Nuclear Fuel Management

Ongoing surveillance and minimum maintenance would continue while FFTF is in standby, and no irradiated nuclear fuel would be transferred to dry storage (WHC 1994).

The current inventory of spent nuclear fuel at FFTF is approximately 11 metric tons (24,200 pounds) of heavy metal, composed predominantly of mixed plutonium-uranium oxide encapsulated in stainless steel cladding (DOE 1995a). The spent nuclear fuel is stored in the sodium-filled vessels and in the dry cask storage system. Spent nuclear fuel stored at FFTF during standby, would continue to be stored there under existing conditions. There is no radiological liquid released to the environment from spent nuclear fuel storage. During operation, the airborne releases from FFTF, including spent nuclear fuel storage, resulted in an annual total effective dose equivalent to the public of less than 1.0×10^{-4} millirem (Nielsen 1999). This dose is negligible compared with the U.S. Environmental Protection Agency's (EPA) Clean Air Act standard of 10 millirem per year.

4.2.2 No Action Alternative—Option 2

Under Option 2 of the No Action Alternative, the United States would continue to purchase the plutonium-238 from Russia that is needed to fabricate radioisotope power systems for future U.S. space missions. However, to allow for potential future production of plutonium-238, neptunium-237 that could be used in targets would be transported from SRS to a new storage facility. This option evaluates REDC at ORR as that storage facility.

FFTF at Hanford would be maintained in standby as part of this option.

4.2.2.1 Importation of Plutonium-238 from Russia

The environmental impacts associated with importing the plutonium-238 from Russia are given in Section 4.2.1.1.

4.2.2.2 Transportation and Storage

The environmental impacts associated with transporting neptunium-237 oxide from SRS to ORR and storing it in REDC are addressed in the following sections.

4.2.2.2.1 Land Resources

LAND USE. REDC is an existing facility in the 7900 Area of ORNL. The use of this facility for storing neptunium-237 for 35 years would require internal modifications, but no new facilities would be built. Since no additional land would be disturbed and the use of REDC for neptunium-237 storage would be compatible with its present mission, there would be no change in land use at ORR.

VISUAL RESOURCES. All activities associated with storing neptunium-237 would take place over a 35-year period in REDC. Because REDC is an existing facility that would require no external modifications, there would be no change in its appearance. Thus, the current Visual Resource Management Class IV rating for the 7900 Area would continue for 35 years. Since there would be no change in the appearance of REDC or of the 7900 Area, there would be no impact on visual resources.

4.2.2.2.2 Noise

Neptunium-237 storage would generate noise levels similar to those presently associated with REDC operations, as well as other operations in the 7900 Area. Onsite noise impacts would be expected to be minimal, and changes in offsite noise levels would not be noticeable since the nearest site boundary is 2.5 kilometers (1.6 miles) to the southeast. Changes in traffic volume going to and from REDC would be small, and would result in only minor changes to existing onsite and offsite noise levels. There would be no loud noises associated with neptunium-237 storage that would adversely impact wildlife. Noise impacts from this option would be expected to be the same over the next 35 years.

4.2.2.2.3 Air Quality

There would be no additional nonradiological air pollutant emissions associated with the storage of neptunium-237 at REDC over the next 35 years; thus, there would be no change in nonradiological air quality impacts at ORR (Wham 1999a).

The air quality impacts of transportation are presented in Section 4.2.2.2.11.

4.2.2.2.4 Water Resources

There would be no additional impact on water resources associated with the storage of neptunium-237 at REDC over 35 years because there would be no incremental use of surface water or groundwater, and there would be no change in the quantity or quality of effluents discharged to surface water or groundwater (Wham 1999a). Information on current water usage, effluent discharge, and water quality at ORR is presented in Section 3.2.4.

4.2.2.2.5 Geology and Soils

Using REDC for storing neptunium-237 would not involve new construction. Therefore, geologic and soil resources in the 7900 Area of ORNL would not be disturbed. Hazards from large-scale geologic conditions at ORR, such as earthquakes and volcanoes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-260). The analysis determined that these hazards present a low risk to long-term storage facilities. Further review of the data and analyses presented in the referenced document and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to REDC. This is based on the fact that there is no evidence of capable faults on or near ORR and no volcanic hazard exists. Ground shaking of Modified Mercalli Intensity VI (refer to Table 3–4) associated with postulated earthquakes would be expected to primarily affect the integrity of inadequately designed or

nonreinforced structures. Damage to properly or specially designed or upgraded facilities would not be expected. While sinkholes are present in the Knox Group, the 7900 Area is underlain by the Conasauga Group, in which karst features are less well-developed. Thus, sinkholes do not present a geologic hazard to REDC. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.2.2.2.6 Ecological Resources

Because no new construction is planned, direct disturbance to ecological resources, including wetlands, would not occur. As noted in Section 4.2.2.2.2, wildlife would not be adversely affected by noise associated with neptunium-237 storage. There would be no change in impacts on aquatic resources for 35 years because water usage and wastewater discharge would not change from current values (Section 4.2.2.2.4). Due to the developed nature of the area and because no new construction would take place, impacts on threatened and endangered species would not occur.

Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service (see Table 5–3) and resulted in the Service concluding that it does not anticipate adverse effects to federally listed endangered species that occur near the project area. DOE has also consulted with the Tennessee Department of Environment and Conservation; a response concerning state-listed species is pending from this agency. Although no state-listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the state.

4.2.2.2.7 Cultural and Paleontological Resources

Because no new construction is planned, impacts on cultural and paleontological resources would not occur. One structure on ORNL, the Graphite Reactor, is listed on the National Register of Historic Places as a National Historic Landmark. Additionally, several other structures proposed for listing on the National Register of Historic Places are found on or near ORNL. However, neither the Graphite Reactor nor any of the other structures is in the 7900 Area; thus, the status of cultural resources would not change for 35 years as a result of using REDC for neptunium-237 storage.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). While DOE has made additional contact with the State Historic Preservation Office, a response is pending from this office. Although impacts to cultural resources are not expected as a result of the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the State Historic Preservation Office.

4.2.2.2.8 Socioeconomics

The existing storage facilities at ORR would remain operational. The effort associated with this option can be filled from within the currently projected site employment of 16,276. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ORR.

4.2.2.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

RADIOLOGICAL IMPACTS. Upper-bounding radiological doses to three receptor groups over a 35-year period are given in **Table 4–7**: the population within 80 kilometers (50 miles) of REDC in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. For purposes of this evaluation, it is conservatively assumed that the doses from neptunium-237 storage would be 10 percent of the doses from neptunium-237 target fabrication and processing (refer to Appendix H). The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–7 Radiological Impacts on the Public Around ORR from Storage in REDC Under Option 2 of the No Action Alternative

Receptor	Storage in REDC[a]
Population within 80 kilometers (50 miles) in the year 2020	
Dose (person-rem)	8.0×10^{-6}
35-year latent cancer fatalities	1.4×10^{-7}
Maximally exposed individual	
Annual dose (millirem)	1.7×10^{-7}
35-year latent cancer fatality risk	3.0×10^{-12}
Average exposed individual within 80 kilometers (50 miles)	
Annual dose[b] (millirem)	7.1×10^{-9}
35-year latent cancer fatality risk	1.2×10^{-13}

a. Because exposure data are not available for neptunium-237 storage exclusively, values are conservatively estimated to be 10 percent of the fabrication and processing component of the total neptunium-237 target fabrication, processing, and storage doses (see Table H–12). These values serve as an upper-bounding representation of the potential impacts that could be incurred from neptunium-237 storage (refer to Appendix H). Realistically, these values would be expected to be virtually zero.

b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of REDC in the year 2020 (1,134,200).

Source: Model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

A collective dose of 8.0×10^{-6} person-rem would be incurred by the surrounding population in the year 2020. The corresponding number of latent cancer fatalities in this population from 35 years of storage would be 1.4×10^{-7}. A bounding annual dose of 1.7×10^{-7} millirem is shown for the maximally exposed individual. From 35 years of storage, the corresponding risk of a latent cancer fatality to this individual would be 3.0×10^{-12}.

The upper-bound estimate of the average dose to a worker involved with neptunium-237 storage operations and the corresponding upper-bound dose to the total storage workforce would be 17 millirem and 1.2 person-rem, respectively (refer to **Table 4–8**). The associated risk of a latent cancer fatality to the average worker from 35 years of storage operations would be 2.3×10^{-4}, and the estimated number of latent cancer fatalities in the total storage workforce from 35 years of operations would be 0.017. The total workforce dose presented in Table 4–8 was assumed to be 10 percent of the annual average worker doses reported at REDC for the years 1998 and 1999 (Wham 2000). This reduction factor was applied because the values given in that document include dose components associated with all REDC activities required for neptunium-237 processing, and not just the storage of neptunium-237 (refer to Appendix H). The resulting dose still serves as a conservative representation of potential worker impacts associated with neptunium-237 storage.

Table 4–8 Radiological Impacts on ORR Workers from Operational Facilities Under Option 2 of the No Action Alternative

Receptor—No Action Workers[a]	Storage in REDC[b]
Total dose (person-rem per year)	1.2[c]
35-year latent cancer fatalities	0.017
Average worker dose (millirem per year)	17
35-year latent cancer fatality risk	2.3×10^{-4}

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with storage operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.

b. Because exposure data are not available for neptunium-237 storage exclusively, values are conservatively estimated to be 10 percent of the total dose from neptunium-237 target fabrication/processing and neptunium-237 storage, given in Section 4.4.1.1.9 (Table 4–57), and serve as an upper-bounding representation of the potential impacts that could be incurred from neptunium-237 storage (refer to Appendix H).

c. Based on an estimated 75 badged workers.

Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts would be unchanged from baseline site operations because no new chemicals would be used at REDC (Wham 1999a). Ongoing emissions of hazardous chemicals would be expected to continue for the next 35 years.

4.2.2.2.10 Public and Occupational Health and Safety—Facility Accidents

There would be no consequences from postulated accidents for neptunium-237 storage in REDC. The most severe accident evaluated in this NI PEIS is the beyond-design-basis catastrophic earthquake. Although the building would be expected to collapse, the hot cells would be expected to remain intact, but with cracked walls. In addition, one or more of the shielded viewing windows could be cracked or broken. The neptunium-237 is stored in double steel cans, with both the inner and outer cans sealed. The double cans are stacked in an array of seismically supported steel storage tubes inside the hot cell. The storage tube array would maintain geometry and not be damaged by equipment dislodged in the hot cell during the earthquake. The storage cans would not be stressed to a level that would breach the double containment of the can design. Therefore, no neptunium would be released from the storage cans.

Storage of neptunium-237 at REDC would not require the introduction of hazardous chemicals. Thus, there would be no hazardous chemical accidents associated with the storage of neptunium-237 at REDC.

4.2.2.2.11 Public and Occupational Health and Safety—Transportation

Transportation impacts may be divided into two parts: the impacts of incident-free or routine transportation, and the impacts of transportation accidents. Incident-free transportation and transportation accident impacts are divided into two components: nonradiological and radiological. Incident-free transportation impacts include radiological impacts on the public and the crew from the radiation field that surrounds the package; nonradiological impacts are from vehicular emissions. Nonradiological impacts of potential transportation accidents include traffic accident fatalities. Only as a result of a severe fire and/or a powerful collision, which are of extremely low probability, could a transportation package of the type used to transport radioactive material be so damaged that there could be a release of radioactivity to the environment.

The impact of a specific accident is expressed in terms of probabilistic risk, which is defined as the accident probability (i.e., accident frequency) multiplied by the accident consequences. The overall risk is obtained by summing the individual risks from all reasonably conceivable accidents. The risks for radiological accidents

are expressed as additional latent cancer fatalities, and for nonradiological accidents as additional immediate fatalities. The risks of incident-free effects are expressed in additional latent cancer fatalities.

The first step in the analysis was to determine the incident-free and accident risk factors, on a per-shipment basis, for ground transportation of the various materials. Calculation of risk factors was accomplished by using the HIGHWAY (Johnson et al. 1993) computer code to choose representative routes in accordance with DOT regulations. This code provides population estimates so that the RADTRAN 5 (Neuhauser and Kanipe 2000) code could be used to determine the radiological risk factors. This analysis is described in Appendix J.

Neptunium-237 would be transported from storage at SRS to REDC at ORR. The neptunium-237 would be shipped in Type B packages. Plutonium-238 would be imported from Russia and shipped to LANL. No other shipments of neptunium-237 or waste are anticipated.

Approximately 59 shipments of radioactive materials would be made by DOE under this option. The total distance traveled on public roads by trucks carrying radioactive materials would be 128,000 kilometers (80,000 miles), and at sea by ships carrying plutonium-238 would be 298,000 kilometers (161,000 nautical miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 12 person-rem; the dose to the public would be 21 person-rem. Accordingly, incident-free transportation of radioactive material would result in 0.005 latent cancer fatality among transportation workers and 0.011 latent cancer fatality in the total affected population over the duration of the transportation activities. Latent cancer fatalities associated with radiological releases were estimated by multiplying the occupational (worker) dose by 4.0×10^{-4} latent cancer fatality per person-rem of exposure, and the public accident and accident-free doses by 5.0×10^{-4} latent cancer fatality per person-rem of exposure (ICRP 1991). The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 5.9×10^{-4}.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: more than 1 in 10 million per year) would not breach the transportation package. The consequences of more severe accidents that could breach the transportation package and release radioactive material were evaluated and estimated to have probabilities of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.88 person-rem, resulting in 4.4×10^{-4} latent cancer fatality; and traffic accidents resulting in 0.014 traffic fatality.

4.2.2.2.12 Environmental Justice

As discussed in other parts of Section 4.2.2.2, neptunium-237 storage operations at REDC would pose no significant health or other environmental risks to the public.

NORMAL OPERATIONS. For 35 years of normal operations, the likelihood of a radiological latent cancer fatality among the population residing within 80 kilometers (50 miles) of REDC would be essentially zero (derived from information in Table 4–7). There would be no significant incremental impact associated with emissions of hazardous chemicals at REDC (Section 4.2.2.2.9). As discussed in Section 4.2.2.2.11, no radiological or nonradiological fatalities would be expected to result from incident-free transportation.

ACCIDENTS. Postulated accidents that would affect neptunium-237 storage were found to have no radiological consequences because the storage containers would not be breached (Section 4.2.2.2.10). Accidents during ground transportation were found to have essentially no radiological consequences because credible transportation accidents would not breach the transportation packages for neptunium-237. As discussed in Section 4.2.2.2.11, a fatal vehicle collision would be unlikely.

The implementation of this option would no pose significant radiological or other environmental risks to the public. Under the conservative assumption that all food consumed in the potentially affected area during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to ingestion of contaminated food supplies (see Appendix K). The transportation of neptunium-237 to ORR and storage at REDC would pose no disproportionately high and adverse risks for minority or low-income populations.

4.2.2.2.13 Waste Management

The only anticipated waste generated would be from the decontamination of the shipping containers used to transport neptunium-237 from SRS to ORR for storage at REDC. The minor amounts of low-level radioactive waste that would be generated—less than 10 cubic meters (13.1 cubic yards) over a 35-year period (Brunson 1999a)—could be managed under the existing waste management practices discussed in Section 3.2.11. Incremental impacts on the environment would be negligible.

4.2.2.3 Maintenance of FFTF in Standby

The environmental impacts associated with maintaining FFTF in standby for 35 years are addressed in Section 4.2.1.2.

4.2.3 No Action Alternative—Option 3

Under Option 3 of the No Action Alternative, the United States would continue to purchase the plutonium-238 from Russia that is needed to fabricate radioisotope power systems for future U.S. space missions. However, to allow for potential future production of plutonium-238, neptunium-237 that could be used in targets would be transported from SRS to a new storage facility. This option evaluates the Building CPP-651 vault at INEEL as that storage facility. The CPP–651 vault is within 91 meters (100 yards) of the Fluorinel Dissolution Process Facility (FDPF). This vault has 100 in-ground concrete storage silo positions sealed with 5.1-centimeter (2-inch) stainless steel shielding plugs. The neptunium-237 storage cans would be placed in a rack inside the silo.

FFTF at Hanford would be maintained in standby as part of this option.

4.2.3.1 Importation of Plutonium-238 from Russia

The environmental impacts associated with importing the plutonium-238 from Russia are given in Section 4.2.1.1.

4.2.3.2 Transportation and Storage

The environmental impacts associated with transporting neptunium-237 oxide from SRS to INEEL and storing it in the Building CPP–651 vault are addressed in this section.

4.2.3.2.1 Land Resources

LAND USE. Building CPP–651 is in the Idaho Nuclear Technology and Engineering Center (INTEC) area of INEEL. The use of this facility for storing neptunium-237 for 35 years would require internal modifications of the facility, but no new facilities would be built. Since no additional land would be disturbed and the use of Building CPP–651 for neptunium-237 storage would be compatible with the missions for which it was designed, there would be no change in land use at INEEL.

VISUAL RESOURCES. All activities associated with storing neptunium-237 would take place in Building CPP–651. Because this facility would not require external modifications, there would be no change in its appearance. Thus, the current Visual Resource Management Class IV rating for INTEC would continue for 35 years. Since there would be no change in the appearance of Building CPP–651 or INTEC, there would be no impact on visual resources.

4.2.3.2.2 Noise

Neptunium-237 storage in Building CPP–651 would generate noise levels similar to those presently associated with operations conducted in INTEC. Onsite noise impacts would be expected to be minimal, and changes in offsite noise levels should not be noticeable since the nearest site boundary is 12 kilometers (7.5 miles) to the south. Changes in traffic volume going to and from INTEC would be small and would result in only minor changes to onsite and offsite noise levels. There would be no loud noises associated with neptunium-237 storage that would adversely impact wildlife. Noise impacts from this option would be expected to be the same over the next 35 years.

4.2.3.2.3 Air Quality

There would be no additional nonradiological air pollutant emissions associated with the storage of neptunium-237 at INEEL over the next 35 years; thus, there would be no change in nonradiological air quality impacts.

The air quality impacts of transportation are presented in Section 4.2.3.2.11.

4.2.3.2.4 Water Resources

There would be no additional impact on water resources associated with the storage of neptunium-237 in Building CPP–651 for 35 years because there would be no incremental use of surface water or groundwater, and there would be no change in the quantity or quality of effluents discharged to surface water or groundwater. Information on current water usage, effluent discharge, and water quality at INEEL is presented in Section 3.3.4

4.2.3.2.5 Geology and Soils

Building CPP–651 would be used to store neptunium-237. Because this is an existing facility, there would be no disturbance to either geologic or soil resources at INTEC. Hazards from large-scale geologic conditions at INEEL, such as earthquakes and volcanoes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-148). The analysis determined that these hazards present a low risk to long-term storage facilities. That analysis was reviewed in the *Surplus Plutonium Disposition EIS* (DOE 1999a:4-267-268). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to the proposed INTEC facilities. Ground shaking of Modified Mercalli Intensity VI to VII (refer to Table 3–4)

associated with postulated earthquakes would be expected to primarily affect the integrity of inadequately designed or nonreinforced structures. Damage to properly or specially designed or upgraded facilities would not be expected. Also, the likelihood of future volcanic activity during the 35-year storage period is considered low. The potential for other nontectonic events to affect INEEL facilities is also low. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.2.3.2.6 Ecological Resources

Because no new construction is planned, direct disturbance to ecological resources would not occur. As noted in Section 4.2.3.2.2, wildlife would not be affected by noise associated with neptunium-237 storage. There would be no impact on aquatic resources for 35 years because water usage and wastewater discharge would not change from current values (Section 4.2.3.2.4). Due to the developed nature of the area and the fact that no new construction would take place, impacts on threatened and endangered species would not occur.

Consultation letters to comply with Section 7 of the Endangered Species Act were sent to the U.S. Fish and Wildlife Service and the Idaho Department of Fish and Game (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. The Idaho Department of Fish and Game indicated that their database contained no known occurrences of special status plants or animals near the project area. While DOE has made additional contact with the U.S. Fish and Wildlife Service, a response is pending from this agency. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at INEEL prior to the receipt of input from the Service.

4.2.3.2.7 Cultural and Paleontological Resources

Because no new construction is planned, impacts on cultural and paleontological resources at INTEC would not occur. The use of Building CPP–651 to store neptunium-237 for 35 years would not change the status of six historic structures located at INTEC. Also, Native American resources occurring in the vicinity of INTEC would not be impacted by the storage of neptunium-237.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). The State Historic Preservation Office indicated that Building CPP-651 is likely to be eligible for the National Register of Historic Places as a contributory property in a potential historic district of exceptional significance. However, at this time, the State Historic Preservation Office has determined that more information is needed prior to assisting DOE in evaluating this property. The State Historic Preservation Office also indicated that since there would be no new construction, there is little potential for effects on archaeological properties. DOE would provide additional information as required to the Idaho State Historic Preservation Office prior to the use of any facility at INEEL for the proposed project. Consultation was conducted with interested Native American tribes; however, responses are pending.

4.2.3.2.8 Socioeconomics

The existing storage facilities at INEEL would remain operational. The effort associated with this option can be filled from within the currently projected site employment of 7,993. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL.

4.2.3.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

RADIOLOGICAL IMPACTS. Under this option, INEEL would store neptunium-237 in CPP–651 in the INTEC area. Upper-bounding radiological doses to three receptor groups are given in **Table 4–9**: the population within 80 kilometers (50 miles) of INTEC in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. For purposes of this evaluation, it is conservatively assumed that the doses from neptunium-237 storage would be 10 percent of the doses from neptunium-237 target fabrication and processing (refer to Appendix H). The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–9 Radiological Impacts on the Public Around INEEL from Operational Facilities Under Option 3 of the No Action Alternative

Receptor	Storage in CPP–651[a]
Population within 80 kilometers (50 miles) in the year 2020	
Dose (person-rem)	3.5×10^{-7}
35-year latent cancer fatalities	6.1×10^{-9}
Maximally exposed individual	
Annual dose (millirem)	2.4×10^{-8}
35-year latent cancer fatality risk	4.2×10^{-13}
Average exposed individual within 80 kilometers (50 miles)	
Annual dose[b] (millirem)	1.9×10^{-9}
35-year latent cancer fatality risk	3.3×10^{-14}

a. Because exposure data are not available for neptunium-237 storage exclusively, values are conservatively estimated to be 10 percent of the fabrication and processing component of the total neptunium-237 target fabrication, processing, and storage doses (see Table H–12). These values serve as an upper-bounding representation of the potential impacts that could be incurred from neptunium-237 storage (refer to Appendix H). Realistically, these values would be expected to be virtually zero.

b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of INTEC in the year 2020 (188,400).

Source: Model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

A collective dose of 3.5×10^{-7} person-rem would be incurred in the surrounding population in the year 2020. The corresponding number of latent cancer fatalities in this population from 35 years of storage would be 6.1×10^{-9}. A bounding annual dose of 2.4×10^{-8} millirem is shown for the maximally exposed individual. From 35 years of storage, the corresponding risk of a latent cancer fatality to this individual would be 4.2×10^{-13}.

The upper-bound estimate of the average dose to a worker involved with neptunium-237 storage operations and the corresponding upper-bound dose to the total storage workforce would be 17 millirem and 1.2 person-rem, respectively (refer to **Table 4–10**). The associated risk of a latent cancer fatality to the average worker from 35 years of storage operations would be 2.3×10^{-4}, and the estimated number of latent cancer fatalities in the total storage workforce from 35 years of operations would be 0.017. The total workforce dose presented in Table 4–10 was assumed to be 10 percent of the annual average worker doses reported at REDC for the years 1998 and 1999 (Wham 2000). This reduction factor was applied because the values given in that document include dose components associated with all REDC activities, and not just the storage of

neptunium-237 (refer to Appendix H). The resulting dose still serves as a conservative representation of potential worker impacts associated with neptunium-237 storage.

Table 4–10 Radiological Impacts on INEEL Workers from Operational Facilities Under Option 3 of the No Action Alternative

Receptor—No Action workers[a]	Storage in CPP–651[b]
Total dose (person-rem per year)	1.2[c]
35-year latent cancer fatalities	0.017
Average worker dose (millirem per year)	17
35-year latent cancer fatality risk	2.3×10^{-4}

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with storage operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Because exposure data are not available for neptunium-237 storage exclusively, values are conservatively estimated to be 10 percent of the total dose from neptunium-237 target fabrication/processing and neptunium-237 storage, given in Section 4.4.2.1.9 (Table 4–69), and serve as an upper-bounding representation of the potential impacts that could be incurred from neptunium-237 storage (refer to Appendix H).
c. Based on an estimated 75 badged workers.

Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts at INEEL would be unchanged from baseline site operations because no new chemicals would be emitted to the air at INEEL. Ongoing emissions would be expected to continue for the next 35 years.

4.2.3.2.10 Public and Occupational Health and Safety—Facility Accidents

At INEEL, neptunium-237 would be stored in the Building CPP–651 vault, which is within 91 meters (100 yards) of FDPF. The Building CPP–651 vault has 100 in-ground concrete storage silo positions sealed with 5.1-centimeter (2-inch) stainless steel shielding plugs. The neptunium-237 storage cans would be placed in a rack inside the silo. While the postulated beyond-design-basis earthquake may cause portions of the facility to collapse, none of the storage cans in the in-ground storage silos would be breached. The storage cans would not be stressed to a level that would breach the double containment of the can design.

Storage of neptunium-237 in Building CPP–651 would not require the introduction of hazardous chemicals. Thus, there would be no hazardous chemical accidents associated with the storage of neptunium-237 in Building CPP–651.

4.2.3.2.11 Public and Occupational Health and Safety—Transportation

Neptunium-237 would be transported from storage at SRS to the Building CPP–651 vault at INEEL. The neptunium-237 would be shipped in Type B packages. Plutonium-238 would be imported from Russia and shipped to LANL. No other shipments of neptunium-237 and no shipments of waste are anticipated. The analysis is described in Appendix J.

Approximately 59 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 203,000 kilometers (127,000 miles), and at sea by ships carrying plutonium-238 would be 298,000 kilometers (161,000 nautical miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 12 person-rem; the dose to the public would be 28 person-rem. Accordingly, incident-free transportation of radioactive material would result in 0.005 latent cancer fatality among transportation workers and 0.014 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0009.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: more than 1 in 10 million per year) would not breach the transportation package. The consequences of more severe accidents that could breach the transportation package and release radioactive material were evaluated and estimated to have probabilities of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.88 person-rem, resulting in 4.4×10^{-4} latent cancer fatality; and traffic accidents resulting in 0.014 traffic fatality.

4.2.3.2.12 Environmental Justice

As discussed in other parts of Section 4.2.3.2, neptunium-237 storage operations would pose no significant health or other environmental risks to the public.

NORMAL OPERATIONS. For 35 years of normal storage operations, the likelihood of a radiological latent cancer fatality among the population residing within 80 kilometers (50 miles) of neptunium storage facilities at INEEL would be essentially zero (derived from information in Table 4–9). There would be no significant incremental impact associated with emissions of hazardous chemicals (Section 4.2.3.2.9). As discussed in Section 4.2.3.2.11, incident-free transportation activities conducted under this option would not be expected to result in fatalities.

ACCIDENTS. Postulated accidents that would affect neptunium-237 storage were found to have no radiological consequences because the storage containers would not be breached (Section 4.2.3.2.10). Accidents during ground transportation were found to have essentially no radiological consequences because credible transportation accidents would not breach the transportation packages for neptunium-237. No fatalities due to vehicle collisions would be expected.

The implementation of this option would pose no significant radiological or other environmental risks to the public. Under the conservative assumption that all food consumed in the potentially affected area during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to ingestion of contaminated food supplies (see Appendix K). The transportation of neptunium-237 to INEEL and storage in Building CPP–651 would pose no disproportionately high and adverse risks for minority or low-income populations.

4.2.3.2.13 Waste Management

The only anticipated waste associated with this option would be from decontamination of the shipping containers used to transport neptunium-237 from SRS to INEEL for storage. The minor amounts of low-level radioactive waste that would be generated—less than 10 cubic meters (13.1 cubic yards) over a 35-year period (Brunson 1999a)—could be managed under the existing waste management practices discussed in Section 3.3.11. Incremental impacts on the environment would be negligible.

4.2.3.3 Maintenance of FFTF in Standby

The environmental impacts associated with maintaining FFTF in standby for 35 years are addressed in Section 4.2.1.2.

4.2.4 No Action Alternative—Option 4

Under Option 4 of the No Action Alternative, the United States would continue to purchase the plutonium-238 from Russia that is needed to fabricate radioisotope power systems for future U.S. space missions. However, to allow for potential future production of plutonium-238, neptunium-237 that could be used in targets would be transported from SRS to a new storage facility. This option evaluates FMEF at Hanford as that storage facility.

FFTF at Hanford would be maintained in standby as part of this option.

4.2.4.1 Importation of Plutonium-238 from Russia

The environmental impacts associated with importing the plutonium-238 from Russia are given in Section 4.2.1.1.

4.2.4.2 Transportation and Storage

The environmental impacts associated with transporting neptunium-237 oxide from SRS to Hanford and storing it at FMEF are addressed in this section.

4.2.4.2.1 Land Resources

LAND USE. FMEF is in the 400 Area of Hanford. The use of this facility for storing neptunium-237 for 35 years would require internal modifications, but no new facilities would be built. Since no additional land would be disturbed and the use of FMEF for neptunium-237 storage would be compatible with the mission for which it was designed, there would be no change in land use at Hanford.

VISUAL RESOURCES. All activities associated with storing neptunium-237 would take place over 35 years in FMEF. Because FMEF would require no external modifications, there would be no change in its appearance. Therefore, the current Visual Resource Management Class IV rating for the 400 Area would continue for 35 years. Since there would be no change in the appearance of FMEF or that of the 400 Area, there would be no impact on visual resources.

4.2.4.2.2 Noise

Neptunium-237 storage would generate noise levels similar to those presently associated with operations in the 400 Area. Onsite noise impacts would be expected to be minimal, and changes in offsite noise levels should not be noticeable since the nearest site boundary is 7 kilometers (4.3 miles) to the east. Changes in traffic volume going to and from FMEF would be small and would result in only minor changes to onsite and offsite noise levels. There would be no loud noises associated with neptunium-237 storage that would adversely impact wildlife. Noise impacts from this option would be expected to be the same over the next 35 years.

4.2.4.2.3 Air Quality

There would be no additional nonradiological air pollutant emissions associated with the storage of neptunium-237 at Hanford over the next 35 years; thus, there would be no change in nonradiological air quality impacts.

The air quality impacts of transportation from SRS to Hanford are presented in Section 4.2.4.2.11.

4.2.4.2.4 Water Resources

There would be no additional impact on water resources associated with the storage of neptunium-237 in FMEF for 35 years because there would be no incremental use of surface water or groundwater, and there would be no change in the quantity or quality of effluents discharged to surface water or groundwater. Information on current water usage, effluent discharge, and water quality at Hanford is presented in Section 3.4.4.

4.2.4.2.5 Geology and Soils

Because the neptunium-237 would be stored in FMEF, an existing facility, there would be no disturbance to either geologic or soil resources in the 400 Area. Hazards from large-scale geologic conditions at Hanford, such as earthquakes and volcanoes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-45). The analysis determined that these hazards present a low risk to long-term storage facilities. That analysis was reviewed in the *Surplus Plutonium Disposition EIS* (DOE 1999a:4-260). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to FMEF. Ground shaking of Modified Mercalli Intensity V to VII (refer to Table 3–4) associated with postulated earthquakes would be expected to primarily affect the integrity of inadequately designed or nonreinforced structures. Damage to properly or specially designed or upgraded facilities would not be expected. Also, only minimal effects (e.g., ashfall) would be expected from postulated volcanic events in the Cascade Region. The potential for other nontectonic events to affect the facility is also low. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.2.4.2.6 Ecological Resources

Because no new construction is planned in the 400 Area, direct disturbance to ecological resources would not occur. As noted in Section 4.2.4.2.2, wildlife would not be affected by noise associated with neptunium-237 storage. Because water usage and wastewater discharge would not change from current values, there would be no change in impacts on aquatic habitat or wetlands associated with the Columbia River for 35 years (Section 4.2.4.2.4). Due to the developed nature of the area and the fact that no new construction would take place, impacts on threatened and endangered species would not occur.

Consultation letters concerning threatened and endangered species were sent to the U.S. Fish and Wildlife Service, the National Marine Fisheries Service, the Washington State Department of Natural Resources, and the State of Washington Department of Fish and Wildlife (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. Both the Washington State Department of Natural Resources and the State of Washington Department of Fish and Wildlife provided lists of state species of concern that occur in the vicinity of the project area. As noted above, no impacts to any threatened or endangered species are expected, including those of concern to these agencies. While DOE has made additional contacts with the U.S. Fish and Wildlife Service and the National Marine

Fisheries Service, responses are pending from these agencies. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at Hanford prior to the receipt of input from these Federal agencies.

4.2.4.2.7 Cultural and Paleontological Resources

Because FMEF is an existing facility in the highly disturbed 400 Area and new construction would not be required, there would be no change in the status of cultural and paleontological resources. No prehistoric, historic, or paleontological sites have been identified either in the 400 Area or within 2 kilometers (1.2 miles) of the 400 Area. Six buildings in the 400 Area have been determined to be eligible for listing on the National Register of Historic Places as contributing properties in the Historic District recommended for mitigation. The use of FMEF to store neptunium-237 for 35 years would not affect the eligibility of these structures for the National Register of Historic Places. No Native American resources are known to occur in the 400 Area.

Consultation to comply with Section 106 of the National Historic Preservation Act was conducted with the State Historic Preservation Office (see Table 5–3) and resulted in concurrence by the State Historic Preservation Office that the proposed action would have no effect on historic properties at Hanford. Consultation was also conducted with interested Native American tribes that resulted in comments at public hearings by members representing the Nez Perce and Confederated Tribes of the Umatilla Indian Reservation. Responses to their specific comments are addressed in Volume 3.

4.2.4.2.8 Socioeconomics

The existing storage facilities at Hanford would remain operational. The effort associated with this option can be filled from within the currently projected site employment of 16,005. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around Hanford.

4.2.4.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

RADIOLOGICAL IMPACTS. This option involves the storage of neptunium-237 at FMEF. Upper-bounding radiological doses to three receptor groups are given in **Table 4–11**: the population within 80 kilometers (50 miles) of FMEF in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. For purposes of this evaluation, it is conservatively assumed that the doses from neptunium-237 storage would be 10 percent of the doses from neptunium-237 target fabrication and processing (refer to Appendix H). The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

A collective dose of 4.0×10^{-6} person-rem would be incurred in the surrounding population in the year 2020. The corresponding number of latent cancer fatalities in this population from 35 years of storage would be 7.0×10^{-8}. A bounding annual dose of 4.3×10^{-8} millirem is shown for the maximally exposed individual. From 35 years of storage, the corresponding risk of a latent cancer fatality to this individual would be 7.5×10^{-13}.

Table 4–11 Radiological Impacts on the Public Around Hanford from Operational Facilities Under Option 4 of the No Action Alternative

Receptor	Storage in FMEF[a]
Population within 80 kilometers (50 miles) in the year 2020	
Dose (person-rem)	4.0×10^{-6}
35-year latent cancer fatalities	7.0×10^{-8}
Maximally exposed individual	
Annual dose (millirem)	4.3×10^{-8}
35-year latent cancer fatality risk	7.5×10^{-13}
Average exposed individual within 80 kilometers (50 miles)	
Annual dose[b] (millirem)	8.1×10^{-9}
35-year latent cancer fatality risk	1.4×10^{-13}

a. Because exposure data are not available for neptunium-237 storage exclusively, values are conservatively estimated to be 10 percent of the fabrication and processing component of the total neptunium-237 target fabrication, processing, and storage doses (see Table H–12). These values serve as an upper-bounding representation of the potential impacts that could be incurred from neptunium-237 storage (refer to Appendix H). Realistically, these values would be expected to be virtually zero.

b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FMEF in the year 2020 (494,400).

Source: Model results, using the GENII computer code (Napier et al. 1988).

The upper-bound estimate of the average dose to a worker involved with neptunium-237 storage operations and the corresponding upper-bound dose to the total storage workforce would be 17 millirem and 1.2 person-rem, respectively (refer to **Table 4–12**). The associated risk of a latent cancer fatality to the average worker from 35 years of storage operations would be 2.3×10^{-4}, and the estimated number of latent cancer fatalities in the total storage workforce from 35 years of operations would be 0.017. The total workforce dose presented in Table 4–12 was assumed to be 10 percent of the average annual workforce doses reported at REDC for the years 1998 and 1999 (Wham 2000). This reduction factor was applied because the values given in that document include dose components associated with all REDC activities, and not just the storage of neptunium-237 (refer to Appendix H). The resulting dose still serves as a conservative representation of potential worker impacts associated with neptunium-237 storage.

Table 4–12 Radiological Impacts on Hanford Workers from Operational Facilities Under Option 4 of the No Action Alternative

Receptor—No Action Workers[a]	Storage in FMEF[b]
Total dose (person-rem per year)	1.2[c]
35-year latent cancer fatalities	0.017
Average worker dose (millirem per year)	17
35-year latent cancer fatality risk	2.3×10^{-4}

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with storage operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.

b. Because exposure data are not available for neptunium-237 storage exclusively, values are conservatively estimated to be 10 percent of the total dose from neptunium-237 target fabrication/processing and neptunium-237 storage, given in Section 4.4.3.1.9 (Table 4–78), and serve as an upper-bounding representation of the potential impacts that could be incurred from neptunium-237 storage (refer to Appendix H).

c. Based on an estimated 75 badged workers.

Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts would be unchanged from baseline site operations because no new chemicals would be emitted to the air at Hanford. Ongoing emissions associated with storage at FMEF would be expected to continue for the next 35 years.

4.2.4.2.10 Public and Occupational Health and Safety—Facility Accidents

There would be no consequences from postulated accidents for neptunium-237 storage in FMEF. The most severe accident evaluated in this NI PEIS is the beyond-design-basis catastrophic earthquake. Although the building would be expected to collapse, the hot cells would be expected to remain intact, but with cracked walls. In addition, one or more of the shielded viewing windows could be cracked or broken. The neptunium-237 is stored in double steel cans, with both the inner and outer cans sealed. The double cans are stacked in an array of seismically supported steel storage tubes inside the hot cell. The storage tube array would maintain geometry and not be damaged by equipment dislodged in the hot cell during the earthquake. The storage cans would not be stressed to a level that would breach the double containment of the can design. Therefore, no neptunium would be released from the storage cans.

Storage of neptunium-237 at FMEF would not require the introduction of hazardous chemicals. Thus, there are no hazardous chemical accidents associated with the storage of neptunium-237 at FMEF.

4.2.4.2.11 Public and Occupational Health and Safety—Transportation

Neptunium-237 would be transported from storage at SRS to FMEF at Hanford. The neptunium-237 would be shipped in Type B packages. Plutonium-238 would be imported from Russia and shipped to LANL. No other shipments of neptunium-237 and no shipments of waste are anticipated.

Approximately 59 shipments of radioactive materials would be made by DOE under this option. The total distance traveled on public roads by trucks carrying radioactive materials would be 220,000 kilometers (137,000 miles), and at sea by ships carrying plutonium-238 would be 298,000 kilometers (161,000 nautical miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 12 person-rem; the dose to the public, 29 person-rem. Accordingly, incident-free transportation of radioactive material would result in 0.005 latent cancer fatality among transportation workers and 0.014 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this alternative is 0.0009.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million) would not breach the transportation package. The consequences of more severe accidents that could breach the transportation package and release radioactive material were evaluated and estimated to have probabilities of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.88 person-rem, resulting in 4.4×10^{-4} latent cancer fatality; and traffic accidents resulting in 0.014 traffic fatality.

4.2.4.2.12 Environmental Justice

As discussed in other parts of Section 4.2.4.2, neptunium-237 storage operations at FMEF would pose no significant health or other environmental risks to the public.

NORMAL OPERATIONS. For 35 years of normal storage operations, the likelihood of a radiological latent cancer fatality among the population residing within 80 kilometers (50 miles) of neptunium storage facilities at Hanford would be essentially zero (derived from information in Table 4–11). There would be no significant incremental impact associated with emissions of hazardous chemicals at Hanford (Section 4.2.4.2.9). No fatalities would be expected from incident-free transportation (Section 4.2.4.2.11).

ACCIDENTS. Postulated accidents that would affect neptunium-237 storage were found to have no radiological consequences because the storage containers would not be breached (Section 4.2.4.2.10). Accidents during ground transportation were found to have essentially no radiological consequences because credible transportation accidents would not breach the transportation packages for neptunium-237. No fatal vehicle collisions would be expected.

The implementation of this option would pose no significant radiological or other environmental risks to the public. Under the conservative assumption that all food consumed in the potentially affected area during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to ingestion of contaminated food supplies (see Appendix K). The transportation of neptunium-237 to Hanford and storage in FMEF would pose no disproportionately high and adverse risks to minority or low-income populations.

4.2.4.2.13 Waste Management

The only anticipated waste associated with this option would be from decontamination of the shipping containers used to transport neptunium-237 from SRS to Hanford for storage at FMEF. The minor amounts of low-level radioactive waste that would be generated—less than 10 cubic meters (13.1 cubic yards) over a 35-year period (Brunson 1999a)—could be managed under the existing waste management practices discussed in Section 3.4.11. Incremental impacts on the environment would be negligible.

4.2.4.3 Maintenance of FFTF in Standby

The environmental impacts associated with maintaining FFTF in standby for 35 years are addressed in Section 4.2.1.2.

4.3 ALTERNATIVE 1—RESTART FFTF

Under Alternative 1, FFTF at Hanford would be restarted and operated for the 35-year evaluation period. FFTF would be used to irradiate targets for medical and industrial isotopes production, plutonium-238 production, and civilian nuclear energy research and development irradiation requirements. Ongoing operations at existing facilities as described in Chapter 3, Affected Environment, would continue.

Targets for medical and industrial isotope production would be fabricated in one or more facilities at Hanford. Target material would typically be acquired from ORNL, where enrichment processes are conducted to produce high purity target material suitable for production of medical isotopes. The targets would be irradiated at FFTF and then returned to the fabrication facility for postirradiation processing. From there, the isotope products would be sent directly to commercial pharmaceutical distributors.

Targets for plutonium-238 production would be fabricated in one of three candidate facilities at ORNL, INEEL, or Hanford. The material needed for target fabrication (neptunium-237) would be transported from SRS. The nonirradiated targets would be transported and irradiated at FFTF and transported back to the fabricating facilities for postirradiation processing. The separated plutonium-238 would be transported to LANL for fabrication into heat sources for radioisotope power systems.

Under Alternative 1, raw materials, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for raw target material acquisition, material storage, target fabrication, target irradiation, and postirradiation processing and the final destination for the isotopes and the plutonium-238 product or various research and development test sites.

FFTF could produce high-energy neutrons and a large flux level (10^{15} neutron per square centimeters per second) that can be tailored to nearly any desired energy level. FFTF would provide the greatest flexibility for both isotope production and nuclear-based research and development among the baseline configurations for all of the proposed alternatives. Due to its large core size, flux spectrum, demonstrated testing capability, and rated power level, it would be able to concurrently support the projected plutonium-238 needs, production of medical and industrial isotopes (including those isotopes normally produced in particle accelerators), and civilian nuclear energy research and development related to a broad range of materials, advanced reactors, advanced fuels and waste transmutation.

The six options under this alternative are associated with the type of nuclear fuel to be used for FFTF operations and the specific facilities to be used for target fabrication and processing. The first three options (Options 1 through 3) would involve operating FFTF with a mixed oxide fuel core for the first 21 years and a highly enriched uranium fuel core for the remaining 14 years. The last three options (Options 4 through 6) would involve operating FFTF with a mixed oxide fuel core for the first 6 years and a highly enriched uranium fuel core for the remaining 29 years. FFTF can provide similar irradiation services with either a mixed oxide core or a highly enriched uranium core. The reasons for these options in FFTF core fuel are provided in Section 2.3.1.1.3.

The options involving storage, fabrication, postirradiation processing, and transportation are discussed below.

- **Options 1 and 4.** REDC at ORNL would be used to fabricate and process the neptunium-237 targets required for plutonium-238 production. The neptunium-237 transported from SRS to ORNL would be stored in REDC. The plutonium-238 product would be transported from ORNL to LANL. Hanford's Radiochemical Processing Laboratory (RPL)/306–E facilities would be used to fabricate and process targets for medical and industrial isotope production and for research and development, as well as to store the materials needed to fabricate these targets.

- **Options 2 and 5.** FDPF at INEEL would be used to fabricate and process the neptunium-237 targets for plutonium-238 production. The neptunium-237 transported from SRS to INEEL would be stored in FDPF or Building CPP–651 at INEEL. The plutonium-238 product would be transported from INEEL to LANL. Hanford's RPL/306–E facilities would be used to fabricate and process targets for medical and industrial isotope production and for research and development, as well as to store the materials needed to fabricate these targets.

- **Options 3 and 6.** FMEF at Hanford would be used to fabricate and process neptunium-237 targets for plutonium-238 production, targets for the production of medical and industrial isotopes, and targets for research and development. The neptunium-237 transported from SRS to Hanford and the other target materials transported from other offsite facilities to Hanford would be stored in FMEF. The plutonium-238 product would be transported from Hanford to LANL for fabrication into heat sources for radioisotope power systems.

As described in Section 1.2.3, the civilian nuclear energy research and development initiatives requiring an enhanced DOE nuclear infrastructure fall into three basic categories: materials research, nuclear fuels research, and advanced reactor development.

- Materials research involves irradiating materials in a high-flux field to determine the radiation effect during reactor normal operating conditions or to perform accelerated life-cycle testing. This form of testing would not introduce material into FFTF that would result in additional releases during normal operation or accident conditions.

- Nuclear fuels research involves irradiating test fuel pellets, fuel pins, and fuel assemblies in high-temperature environments expected in future reactor designs. When the test specimens are inserted into FFTF, there would be no significant increase of fissile material in the FFTF core inventory that would result in additional releases during normal operation or accident conditions.

- Advanced reactor development involves test loop experiments under prototypical reactor conditions. When the test loop is operating in the FFTF core, there would be no significant increase of fissile material in the core inventory that would result in additional releases during normal operation or accident conditions.

The environmental impacts associated with implementation of the proposed civilian nuclear energy research and development missions cannot be distinguished from the impacts of operating FFTF without the civilian nuclear energy research and development mission.

4.3.1 Alternative 1 (Restart FFTF)—Option 1

Option 1 involves operating FFTF at Hanford to irradiate all targets and materials associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating REDC at ORR to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and operating facilities in the Hanford 300 Area to fabricate and process the other targets and materials and to process the associated products. This option includes storage in REDC of the neptunium-237 transported to ORR from SRS and storage in the Hanford 300 Area facilities of the other target materials transported to Hanford from other offsite facilities.

The transportation of the mixed oxide and highly enriched uranium fuel to Hanford for use in FFTF, the transportation of the neptunium-237 to ORR and then to Hanford, the transportation of the other target material

to Hanford, and the transportation of the product materials following irradiation and postirradiation processing are also part of this option.

Under Option 1, FFTF would operate with a mixed oxide fuel core for the first 21 years and with a highly enriched uranium fuel core for the next 14 years.

4.3.1.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations and with all transportation activities are assessed in this section.

4.3.1.1.1 Land Resources

LAND USE. FFTF is in the 400 Area of Hanford. For the foreseeable future, land use in the 400 Area is anticipated to be industrial, which is defined to include FFTF operations. The use of the facility for the irradiation services assessed in this NI PEIS would be compatible with the mission for which the facility was originally built. Although internal modifications could be required, no new facilities would be built and thus, there would be no change in land use in the 400 Area.

REDC at ORR would be used for neptunium-237 storage, target fabrication, and processing. REDC is an existing operating facility in the 7900 Area of ORNL, and use of this facility would require internal modifications, but no new facilities would be built. Because no additional land would be disturbed and the use of REDC for neptunium-237 storage, target fabrication, and processing would be compatible with its present mission, there would be no change in land use at ORR.

RPL and Building 306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. These buildings are existing structures that would require only internal modifications. Because no additional land would be disturbed and target fabrication and processing would be compatible with their present mission, there would be no impact on land use at Hanford.

VISUAL RESOURCES. The use of FFTF would not require any external modifications that would alter the appearance of the facility. Thus, the current Visual Resource Management Class IV rating for the 400 Area would not change. Since there would be no change in the appearance of FFTF or that of the 400 Area, there would be no additional impact on visual resources.

All activities associated with neptunium-237 storage, target fabrication, and processing would take place in REDC at ORR. Because REDC is an existing facility that would require no external modifications, there would be no change in its appearance. Therefore, the current Visual Resource Management Class IV rating for the 7900 Area would not change, and there would be no impact on visual resources.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. These existing structures would require no external modifications. Because the appearance of these buildings would remain unchanged, the current Visual Resource Management Class IV rating for the 300 Area would not change; thus, there would be no impact on visual resources.

4.3.1.1.2 Noise

No new construction would be required at FFTF under Option 1. Noise sources from FFTF operations would be similar to those during standby. Therefore, the change in noise levels from operation activities would be expected to be small. FFTF operations would not be expected to result in any change in noise impacts on wildlife around the 400 Area, and offsite noise impacts would also be minor because the nearest site boundary is 7 kilometers (4.3 miles) to the east. Operations would be expected to result in a minimal change in noise impacts on people near Hanford as a result of changes in employee and truck traffic levels.

REDC at ORR would be used for neptunium-237 target-material storage, target fabrication, and processing. Interior modifications of these facilities in the 7900 Area of ORNL would be expected to result in little change in noise impacts on wildlife around this area. REDC operations would not be expected to result in any change in noise impacts on wildlife around the 7900 Area, and offsite noise impacts would be small because the nearest site boundary is 2.5 kilometers (1.6 miles) to the southeast. Operations would be expected to result in minimal change in noise impacts on people near ORR as a result of changes in employee and truck traffic levels.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. Interior modifications of these facilities would be expected to result in little change in noise impacts on wildlife around this area and people near Hanford. Operation of these facilities for target fabrication and processing would not be expected to result in any change in noise impacts on wildlife around the 300 Area and people near Hanford. Operations would be expected to result in minimal change in noise impacts on people near Hanford as a result of changes in employee and truck traffic levels.

4.3.1.1.3 Air Quality

There are no planned FFTF outdoor construction activities associated with the restart of FFTF. No airborne constituents are currently measured or have been required to be monitored during previous reactor operations (Nielsen 2000). Several air pollutant sources are operating at FFTF, including the gas turbine emergency generator, the diesel-driven fire pump, and the oil-fired preheaters. They would continue to operate at the existing frequency. The emergency diesel generators are not currently operated or tested. The operation of FFTF would require the emergency diesel generators to be tested approximately 30 minutes each month to ensure operability (Nielsen 2000). Criteria pollutants were modeled for a stack 9.22 meters (30.3 feet) in height at a distance of 7,200 meters (23,600 feet) east of FFTF and compared with the most stringent standards for the Hanford area. The concentrations are based on a dispersion-modeling screening analysis conducted with maximum expected emission rates and a set of worst-case meteorological conditions.

The concentrations at Hanford from FFTF attributable to this option are presented in **Table 4–13.** Only those air pollutants expected to be emitted that have ambient air quality standards are presented in the table. The concentrations were determined to be small and would be below the applicable ambient standards even when ambient monitored values and contributions from other site activities were included.

There would be no change in air quality impacts from target processing at the Hanford 300 Area. Emissions of target material would be minimal due to efficient filtration and measures taken to prevent losses of expensive target material. Fugitive dust from employee and truck traffic could increase slightly.

The concentrations at Hanford attributable to this option are compared with the Prevention of Significant Deterioration Class II increments for sulfur dioxide and nitrogen dioxide in **Table 4–14**.

Table 4–13 Incremental Hanford Concentrations Associated with Alternative 1 (Restart FFTF)—Option 1

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (micrograms per cubic meter)
Carbon monoxide	8 hours	10,000[b]	52.1
	1 hour	40,000[b]	74.4
Nitrogen dioxide	Annual	100[b]	0.0118
PM_{10}	Annual	50[c]	8.4×10^{-4}
	24 hours	150[c]	9.84
Sulfur dioxide	Annual	50[d]	7.86×10^{-4}
	24 hours	260[d]	9.1
	3 hours	1,300[b]	20.5
	1 hour	660[d]	22.8

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The 24-hour PM_{10} (particulate matter with an aerodynamic diameter less than or equal to 10 micrometers) standard is attained when the expected number of days with a 24-hour average concentration above the standard is equal to or less than 1. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Federal and state standard.
c. Federal standard currently under litigation.
d. State standard.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); additional data from Nielsen 2000.

Table 4–14 PSD Class II Increments Compared to Hanford Concentrations Associated with FFTF Under Alternative 1 (Restart FFTF)—Option 1

Pollutant	Averaging Period	Allowable PSD Increment (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Nitrogen dioxide	Annual	25	0.0118
Sulfur dioxide	Annual	20	7.86×10^{-4}
	24 hours	91	9.1
	3 hours	512	20.5

Key: PSD, Prevention of Significant Deterioration.
Source: Modeled PSD increments are based on the SCREEN3 computer code (EPA 1995).

The air pollutant concentrations at ORR attributable to this option at REDC are presented in **Table 4–15**. The concentrations are based on a dispersion-modeling screening analysis conducted with maximum expected emission rates and a set of worst-case meteorological conditions. Criteria pollutants were modeled for a stack height of 76.2 meters (250 feet) at the boundary limit of 4,988 meters (16,370 feet). Only those air pollutants expected to be emitted that have ambient air quality standards are presented in the table.

Table 4–15 Incremental ORR Concentrations[a] Associated with Alternative 1 (Restart FFTF)—Option 1

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (micrograms per cubic meter)
Nitrogen dioxide	Annual	100	1.99×10^{-4}
Sulfur dioxide	Annual	80	0.04
	24 hours	365	0.31
	3 hours	1,300	0.70

a. For comparison with ambient air quality standards.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995).

There are no Prevention of Significant Deterioration increment-consuming sources at ORR; therefore, a Prevention of Significant Deterioration increment consumption analysis was not conducted. Health effects from hazardous chemicals associated with this option are addressed in Section 4.3.1.1.9.

The change in ambient concentrations of these pollutants would be minimal compared to the baseline. Concentrations off site would be expected to stay well below the ambient standards even when ambient monitored values and the contribution from other site activities were included.

The air quality impacts of transportation are presented in Section 4.3.1.1.11.

4.3.1.1.4 Water Resources

For the restart of FFTF, an existing facility, there would be no construction-related impacts on water bodies, floodplains, or on surface water or groundwater quality.

Incremental effects on key water resource indicators under this option are summarized in **Table 4–16**. During current standby operations, annual average groundwater withdrawal by 400 Area facilities is about 197 million liters (52 million gallons). Should FFTF be restarted, FFTF operations would increase water use by about 61 million liters (16 million gallons) to a total annual withdrawal of approximately 258 million liters (68 million gallons) (Nielsen 1999:38, 41). In addition to higher process cooling demands at FFTF from cooling tower operation, this increase reflects additional staffing and associated potable and sanitary water demands in the 400 Area (DOE 2000a:11; Nielsen 1999:38, 41). This volume of 258 million liters (68 million gallons) per year is approximately 65 percent of the 400 Area groundwater production capacity of about 398 million liters (105.1 million gallons) per year (DOE 1999a:4-262). However, no impact on regional groundwater levels would be expected from increased withdrawals (Nielsen 1999:38). Resumption of groundwater withdrawals could potentially affect the direction of groundwater flow in the unconfined aquifer system on a localized basis. Surface water would not be used for operation of the 400 Area facilities; thus, there would be no impact on the availability of surface water from the Columbia River.

Table 4–16 Incremental Water Use and Wastewater Generation Associated with Operating FFTF and RPL/306–E at Hanford and REDC at ORR Under Alternative 1 (Restart FFTF)—Option 1

Indicator (million liters per year)	Hanford			
	FFTF Operations[a]	FFTF Increment Over Standby[b]	RPL/306–E	ORR REDC
Water use	258	61	~ 0.016[c]	2.86
Process wastewater generation	98	22	0.016	0.023
Sanitary wastewater generation	5.7	1.9	0	2.83

a. These estimates represent total projected operational impacts after restart.
b. Incremental impacts of FFTF restart and operation over standby operations (see Table 4–1).
c. Water use for RPL/306–E operations is estimated to be approximately equal to the process wastewater estimate, as no other additional demands on water use are expected.

Note: To convert from liters per year to gallons per year, multiply by 0.264; ~ means "approximately."
Source: DOE 2000a:11, C-3; Nielsen 1999:38, 41; Wham 1999c.

Additional staffing required to support the restart of FFTF would also increase annual sanitary wastewater generation in the 400 Area by approximately 1.9 million liters (502,000 gallons) over standby to about 5.7 million liters (1.5 million gallons) per year during operations (DOE 2000a:11). Sanitary wastewater from the 400 Area is conveyed to the Energy Northwest treatment system (Nielsen 1999:39). The Energy Northwest treatment system has a treatment capacity of approximately 235 million liters (62 million gallons) per year with sufficient excess capacity to accommodate increased flow from the 400 Area (DOE 1999a:4-41).

There are no radiological liquid effluent pathways to the environment from FFTF. Process (nonradioactive) wastewater from 400 Area facilities is discharged to the 400 Area process sewer system and ultimately to the 400 Area Pond (i.e., 4608 B/C percolation ponds). These discharges are regulated under State Waste Discharge Permit No. ST-4501. This system is further described in Section 3.4.4.1.2. Process wastewater discharges from FFTF would increase by about 22 million liters (5.8 million gallons) annually over standby to approximately 98 million liters (26 million gallons) per year during operations. Increased process wastewater volume would mainly consist of cooling tower blowdown from FFTF's eight cooling towers. However, chemical usage required to control scaling and biofouling of the cooling water systems would not increase (DOE 2000a:11; Nielsen 1999:38). Therefore, as the chemical quality of the process wastewater would not change, no impact on groundwater quality would be expected.

Small quantities of liquid, low-level radioactive waste would be generated during operations associated with washing residual sodium from reactor components in FFTF's Interim Examination and Maintenance Cell and decontamination activities at the 400 Area Maintenance and Storage Facility. Approximately 6,000 liters (1,600 gallons) of liquid, low-level radioactive waste would be generated annually, which would be collected and transported to the 200 Area Effluent Treatment Facility for treatment and disposal (DOE 2000a:7; Nielsen 1999:39, 41).

REDC in the 7900 Area of ORNL at ORR would be used for neptunium-237 storage, target fabrication, and processing in support of plutonium-238 production with proposed activities similar to the current mission of REDC. As existing facilities would be used, there would be no construction-related impacts on water bodies, floodplains, or on surface water or groundwater quality. As summarized in Table 4–16, a relatively small increase in water use and sanitary wastewater generation is projected mainly to support the additional staffing required at REDC (see Section 4.3.1.1.8). The only other measurable increase would be an additional 23,000 liters (6,100 gallons) per year of process wastewater associated with target processing (Wham 1999c). Changes in the quantity or quality, if any, of process and sanitary wastewater discharges would be very small compared to that of other activities with no radiological liquid effluent discharge to the environment under normal operations (Wham 1999a; LMER 1997). Specifically, the anticipated additional 23,000 liters (6,100 gallons) of process wastewater generated per year would be negligible relative to the total volume of process wastewater generated and treated at the ORNL Process Waste Treatment Complex, approximately 2.08 million liters (550,000 gallons) per day (Section 3.2.4.1.2). All wastewaters would be discharged to designated collection and treatment systems as described in Section 3.2.4.1.2. Overall, no measurable impact on water resources at ORR is expected.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. However, radiological activities would be confined to RPL, with Building 306–E providing support to activities not involving radioactive materials (DOE 2000a:C-1). As existing facilities would be used requiring no external modifications, there would be no construction-related impacts on water bodies, floodplains, or on surface water or groundwater quality. Little measurable increase in water use is anticipated to support target fabrication and processing for medical, industrial, and research and development isotope production, with no radiological liquid effluent discharge to the environment under normal operations (DOE 1997b:4-28, 4-29). Also, changes in the quantity or quality of process and sanitary wastewater discharges would be negligible compared to that of other RPL activities, with the only projected increase resulting from equipment washing of nonradiological target materials (DOE 1997b:4-30). Process wastewater discharge from washing activities at RPL is projected to increase by about 16,000 liters (4,200 gallons) per year from a current annual average of approximately 3.6 million liters (950,000 gallons) (DOE 2000a:C-3). This is an increase of less than 1 percent. The only increase in water use expected would be to support this minor increase in processing activity. Process wastewater is discharged to the 300 Area retention process

sewer system (Section 3.4.4.1.2). Thus, impacts on water resources at Hanford are expected to be negligible overall.

Waste management aspects of this option and their effects are further discussed in Section 4.3.1.1.13.

4.3.1.1.5 Geology and Soils

Since no new construction is planned under the proposed restart of FFTF, there would be no disturbance to either geologic or soil resources in the 400 Area of Hanford. As discussed in Section 4.2.1.2.5, hazards from large-scale geologic conditions at Hanford, such as earthquakes and volcanoes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-45). The analysis determined that these hazards present a low risk to properly or specially designed or upgraded facilities. That analysis was reviewed in the *Surplus Plutonium Disposition EIS* (DOE 1999a:4-260). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to proposed FFTF operations. This is based on the relatively low seismic risk of the area to such specially designed facilities and the expected minimal effects from postulated volcanic events in the Cascade Region.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. Because existing structures would be used, there would be no disturbance to geologic or soil resources in the 300 Area. Hazards from large-scale geologic conditions at Hanford were previously evaluated as discussed above for FFTF and determined to present a low risk to existing facilities. For the reasons previously described, the large-scale geologic conditions likewise present a low risk to the subject 300 Area facilities. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

Because the existing REDC facility would be used for neptunium-237 storage, target fabrication, and processing, geologic and soil resources in the 7900 Area of ORR would not be disturbed. Hazards from large-scale geologic conditions at ORR, such as earthquakes and volcanoes, were previously analyzed as discussed in Section 4.2.2.2.5 and determined to present a relatively low risk to REDC.

4.3.1.1.6 Ecological Resources

Terrestrial resources would not be adversely affected by the restart of FFTF because it is in the highly disturbed and fenced 400 Area and no new construction is planned. Further, as noted in Section 4.3.1.1.2, there would be no change in noise impacts on wildlife. Because additional surface water would not be used and wastewater discharge chemistry would not be expected to change, there would be no change in impacts on aquatic habitat or wetlands associated with the Columbia River (Section 4.3.1.1.4). Due to the developed nature of the area and the fact that no new construction would take place, impacts on threatened and endangered species would not occur.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. Because use of these buildings would not involve any new construction, direct disturbance to ecological resources would not occur. As noted in Section 4.3.1.1.2, wildlife would not be adversely affected by noise associated with target fabrication and processing activities. Because water usage and wastewater discharge would be small fractions of current values and discharge chemistry would not be expected to change, there would be no change in impacts on aquatic habitat or wetlands associated with the Columbia River

(Section 4.3.1.1.4). Due to the developed nature of the area and the fact that no new construction would take place, impacts on threatened and endangered species would not occur.

Consultation letters concerning threatened and endangered species were sent to the U.S. Fish and Wildlife Service, the National Marine Fisheries Service, the Washington State Department of Natural Resources, and the State of Washington Department of Fish and Wildlife (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. Both the Washington State Department of Natural Resources and the State of Washington Department of Fish and Wildlife provided lists of state species of concern that occur in the vicinity of the project area. As noted above, no impacts to any threatened or endangered species are expected, including those of concern to these agencies. While DOE has made additional contacts with the U.S. Fish and Wildlife Service and the National Marine Fisheries Service, responses are pending from these agencies. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at Hanford prior to the receipt of input from these Federal agencies.

The existing REDC at ORR would be used for neptunium-237 storage, target fabrication, and processing. No new construction would take place; thus, direct disturbance to ecological resources, including wetlands, would not occur. As noted in Section 4.3.1.1.2, there would be no change in noise impacts on wildlife. Because there are no wetlands in or directly adjacent to the 7900 Area, this resource would not be affected. There would be no change in impacts on aquatic resources because no additional water would be withdrawn from or discharged to site surface waters and discharge chemistry would not be expected to change (Section 4.3.1.1.4). Threatened and endangered species would not be impacted because an existing facility in the developed area would be used.

Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service (see Table 5–3) and resulted in the Service concluding that it does not anticipate adverse effects to federally listed endangered species that occur near the project area. DOE has also consulted with the Tennessee Department of Environment and Conservation; a response concerning state-listed species is pending from this agency. Although no state-listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the state.

4.3.1.1.7 Cultural and Paleontological Resources

Because FFTF is in the highly disturbed 400 Area and new construction would not be required, no direct impacts on cultural and paleontological resources would be expected. No prehistoric, historic, or paleontological sites have been identified either in the 400 Area or within 2 kilometers (1.2 miles) of the 400 Area. Six buildings in the 400 Area, including two FFTF structures (the Reactor Containment Building and FFTF Control Building), have been determined to be eligible for the National Register of Historic Places as contributing properties in the Historic District recommended for mitigation. The restart of FFTF would be consistent with the purpose for which the reactor was built and would not affect the status of the aforementioned structures.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. Although a number of archaeological sites have been located at least partially within the 300 Area, none would be disturbed because new construction would not be required. Additionally, both buildings have been determined eligible for the National Register of Historic Places as contributing properties within the Historic District recommended for mitigation (Section 3.4.7.2.2); however, they would not be substantially altered by use for target fabrication and processing, and thus, their status would not change. Areas near the 300 Area that are of importance to Native Americans would not be affected by the proposed action.

Consultation to comply with Section 106 of the National Historic Preservation Act was conducted with the State Historic Preservation Office (see Table 5–3) and resulted in concurrence by the State Historic Preservation Office that the proposed action would have no effect on historic properties at Hanford. Consultation was also conducted with interested Native American tribes that resulted in comments at public hearings by members representing the Nez Perce and Confederated Tribes of the Umatilla Indian Reservation. Responses to their specific comments are addressed in Volume 3.

Neptunium-237 storage, target fabrication, and processing would take place at the existing REDC facility, which is in the 7900 Area of ORNL. Because no new construction would take place, direct impacts on cultural and paleontological resources would not occur. One structure within ORNL, the Graphite Reactor, is listed on the National Register of Historic Places as a National Historic Landmark. Additionally, several other structures proposed for listing on the National Register of Historic Places are found within or near ORNL. However, neither the Graphite Reactor nor any of the other structures is in the 7900 Area and, thus, their status would not change by the use of REDC for target fabrication and processing.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). While DOE has made additional contact with the State Historic Preservation Office, a response is pending from this office. Although impacts to cultural resources are not expected as a result of the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the State Historic Preservation Office.

4.3.1.1.8 Socioeconomics

Operating FFTF and target fabrication and processing of all other targets at Hanford 300 Area facilities would require about 218 additional workers to operate these facilities (DOE 1997b; Hoyt et al. 1999). This level of employment would generate about 552 indirect jobs in the region around Hanford. The potential total employment increase of 770 direct and indirect jobs in the Hanford region represents an approximate 0.3 percent increase in the projected regional economic area workforce. It would have no noticeable impact on the regional economic area.

Additional employment resulting from this option would not have any noticeable impact on community services in the Hanford region of influence. Assuming that 91 percent of the new employment associated with this option would reside in Hanford's region of influence (Section 3.4.8), 701 new jobs could increase the region's population by approximately 1,346 persons. This increase, in conjunction with normal population growth forecasted by the State of Washington, would not have any noticeable impact on the availability of housing and/or the price of housing in the region of influence. Given the current population-to-student ratio in the region of influence, this would likely result in an increase of about 279 students, requiring local school districts to slightly increase the number of classrooms to accommodate them.

Community services in the region of influence would be expected to change to accommodate the population growth as follows: 17 new teachers would be needed to maintain the current student-to-teacher ratio of 16:1; 2 new police officers would be needed to maintain the current officer-to-population ratio of 1.5:1000; 5 new firefighters would need to be added to maintain the current firefighter-to-population ratio of 3.4:1000; and 2 new doctors would be needed to maintain the current doctor-to-population ratio of 1.4:1000. Thus, an additional 26 positions would have to be created to maintain community services at current levels. Hospitals in the region of influence would not experience any change from the 2.1 beds per 1,000 persons currently available. Additionally, the average school enrollment would increase from 92.5 percent to 93.1 percent. None of these projected changes should have a major impact on the level of community services currently offered in the region of influence.

Target fabrication and processing of neptunium-237 targets at ORR would require about 41 additional workers to operate these facilities (Wham et al. 1998). This level of employment would generate approximately 105 indirect jobs in the region around Oak Ridge. The potential total employment increase of 146 direct and indirect jobs represents less than 0.1 percent of the projected regional economic area workforce. It would have no noticeable impact on the regional economic area.

Additional employment resulting from this option would not have any noticeable impact on community services in the ORR region of influence. Assuming that 89.9 percent of the new employment associated with this option would reside in ORR's region of influence (Section 3.2.8), 146 total new jobs could increase the region's population by approximately 248 persons. This increase, in conjunction with normal population growth forecasted by the State of Tennessee, would have no noticeable effect on the availability of housing and/or the price of housing in the region of influence. The public would experience little or no change in the level of community services currently offered in the region of influence.

4.3.1.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup, processing, and operations are given in **Table 4–17** for FFTF and RPL at Hanford and REDC at ORR: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

To represent a bounding annual dose scenario at Hanford, it is assumed that a full-year's isotopic release would occur from target processing at RPL concurrently with a full-year's release from FFTF operations at 400 megawatts; the impacts presented in Table 4–17 assume a full-year's release resulting from FFTF and RPL preoperational testing and startup activities. To represent a bounding annual dose scenario at ORR, it is assumed that a full year's release would occur from neptunium-237 target processing at REDC.

As a result of annual operations, the bounding projected total incremental population dose in the year 2020 for the populations surrounding Hanford and ORR would be 0.25 person-rem. The corresponding number of latent cancer fatalities in these populations from 35 years of operations would be 0.0044. The bounding total incremental dose to the maximally exposed member of the public from annual operations at Hanford would be 0.0054 millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 9.5×10^{-8}. The incremental dose to the maximally exposed member of the public from annual operations at ORR would be 1.9×10^{-6} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 3.3×10^{-11}.

Table 4–17 Incremental Radiological Impacts on the Public Around ORR and Hanford from Operational Facilities Under Alternative 1 (Restart FFTF)—Option 1

Receptor	ORR REDC Processing[a]	Hanford FFTF Preoper. Activities[b]	Hanford RPL Preoper. Activities[b]	Hanford FFTF Operations	Hanford RPL Target Processing[a]	Hanford Operations and Processing Total[c]
Population within 80 kilometers (50 miles) in the year 2020						
Dose (person-rem)	8.8×10^{-5}	0.028	1.0[d]	0.044	0.21	0.25
1-year latent cancer fatalities	–	1.4×10^{-5}	5.0×10^{-4}	–	–	–
35-year latent cancer fatalities	1.5×10^{-6}	–	–	7.7×10^{-4}	0.0037	0.0044
Maximally exposed individual						
Annual dose (millirem)	1.9×10^{-6}	1.4×10^{-4}	0.043[c]	4.1×10^{-4}	0.0050	0.0054
1-year latent cancer fatality risk	–	6.8×10^{-11}	2.2×10^{-8}	–	–	–
35-year latent cancer fatality risk	3.3×10^{-11}	–	–	7.2×10^{-9}	8.8×10^{-8}	9.5×10^{-8}
Average exposed individual within 80 kilometers (50 miles)						
Annual dose[e] (millirem)	7.8×10^{-8}	5.7×10^{-5}	2.0×10^{-3} [c]	8.8×10^{-5}	4.2×10^{-4}	5.0×10^{-4}
1-year latent cancer fatality risk	–	2.8×10^{-11}	9.9×10^{-10}	–	–	–
35-year latent cancer fatality risk	1.4×10^{-12}	–	–	1.5×10^{-9}	7.3×10^{-9}	8.8×10^{-9}

a. Target storage, processing, and fabrication activities are performed at the facility. Impacts are for all facility target activities and are dominated by processing activity impacts.
b. For conservatism as well as consistency with other radiological impacts evaluated in this NI PEIS, these values were assessed for the year 2020 even though these activities would commence prior to that year.
c. Represents upper-bounding values.
d. Annual emissions during preoperational activities were assumed to be the same as the 1998 releases for RPL (BWHC 1999). The majority of this dose is due to tritium releases.
e. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the facilities in the year 2020 (about 505,000 for Hanford and 1,134,200 for ORR).

Source: Model results, using the GENII computer code (Napier et al. 1988).

Incremental doses to involved workers from normal operations are given in **Table 4–18**; these workers are defined as those directly associated with all process and operational activities. The incremental annual average dose to REDC workers would be 170 millirem; the incremental annual average dose to FFTF workers (during startup) would be 3.5 millirem; the incremental annual average dose to FFTF workers (during operations) would be 6.6 millirem; the incremental annual average dose to RPL workers (during startup) would be 81 millirem; and the incremental annual average dose for RPL workers (during processing) is estimated to be approximately 160 millirem. The incremental annual dose received by the total site workforce for each of these facilities (at the different phases) would be approximately 12, 0.69, 1.3, 3.2, and 4.8 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers are included in Table 4–18. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–18 Incremental Radiological Impacts on Involved REDC, FFTF, and RPL Workers Under Alternative 1 (Restart FFTF)—Option 1

Receptor—Involved Workers[a]	ORR REDC Processing[b]	Hanford FFTF Preoper. Activities	Hanford RPL Preoper. Activities	Hanford FFTF Operations	Hanford RPL Target Processing[b]	Hanford Operations & Processing Total
Total dose (person-rem per year)	12[c]	0.69[d]	3.2[e]	1.3[d]	4.8[f]	6.1
1-year latent cancer fatalities	–	2.8×10^{-4}	0.0013	–	–	–
35-year latent cancer fatalities	0.17	–	–	0.018	0.067	0.086
Average worker dose (millirem per year)	170	3.5	81	6.6	160	NA
1-year latent cancer fatality risk	–	1.4×10^{-6}	3.2×10^{-5}	–	–	–
35-year latent cancer fatality risk	0.0023	–	–	9.2×10^{-5}	0.0022	NA

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Target storage, processing, and fabrication activities are performed at this facility. Impacts, dominated by processing activities, include impacts from all facility target activities.
c. Based on an estimated 75 badged workers.
d. Based on an estimated 200 badged workers.
e. Based on an estimated 40 badged workers.
f. Based on an estimated 30 badged workers.

Key: NA, not applicable.
Source: BWHC 1999; Nielsen 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. No new hazardous chemical impacts would be expected at RPL/306–E in the 300 Area of Hanford. The quantities of chemicals used for target fabrication and processing would change little from ongoing operations in the 300 Area, and emissions and air quality impacts would be expected to be unchanged.

FFTF restart would require emergency diesel generators to be tested. Hazardous chemical impacts are summarized in **Table 4–19**.

Table 4–19 Incremental Hazardous Chemical Impacts Associated with FFTF Emergency Diesel Generators at Hanford Under Alternative 1 (Restart FFTF)—Option 1

Chemical	Modeled Annual Increment (micrograms per cubic meter)	Reference Concentration (micrograms per cubic meter)	Unit Cancer Risk (risk per micrograms per cubic meter)	Hazard Quotient	Cancer Risk
Benzene	2.5×10^{-6}	NA	7.8×10^{-6}	NA	1.96×10^{-11}
Toluene	1.10×10^{-6}	400	NA	2.74×10^{-9}	NA
Propylene	6.92×10^{-6}	NA	3.7×10^{-6}	NA	2.56×10^{-11}
Formaldehyde	3.17×10^{-6}	NA	0.000013	NA	4.12×10^{-11}
Acetaldehyde	2.06×10^{-6}	NA	2.2×10^{-6}	NA	4.53×10^{-12}

Key: NA, not applicable (the chemical is not a known carcinogen or it is a carcinogen and only unit cancer risk will apply).
Source: EPA 1999; model results, using the Screen3 computer model (EPA 1995).

At ORR, both carcinogenic and noncarcinogenic health effects from exposure to hazardous chemicals were evaluated. It was assumed that under normal operating conditions, the primary exposure pathway for members of the public would be from air emissions released through the 7911 stack. Emissions of chemicals were estimated based on anticipated chemical usage. A worst-case dispersion-modeling screening analysis was performed to estimate annual concentrations for each chemical, based on their emission rates.

The annual concentration for each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the Hazard Quotient for each chemical. The Hazard Quotients were summed to give the Hazard Index from all noncarcinogenic chemicals associated with this option. A Hazard Index of less than one indicates that adverse health effects from non-cancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical. Hazardous chemical health effects are summarized in **Table 4–20**.

Table 4–20 Incremental Hazardous Chemical Impacts on the Public Around ORR Under Alternative 1 (Restart FFTF)—Option 1

Chemical	Modeled Annual Increment (milligrams per cubic meter)	Reference Concentration Inhalation (milligrams per cubic meter)	Unit Cancer Risk (risk per milligrams per cubic meter)	Hazard Quotient	Cancer Risk
Diethyl benzene	3.37×10^{-5}	1	0.0078	3.37×10^{-5}	2.63×10^{-7}
Methanol	1.23×10^{-6}	1.75	NA	7.03×10^{-7}	NA
Nitric acid	1.53×10^{-6}	0.1225	NA	1.25×10^{-5}	NA
Tributyl phosphate	6.34×10^{-5}	0.01	NA	0.00634	NA
			Hazard Index =	0.00639	

Note: For diethyl benzene, the reference concentration for ethyl benzene and the unit cancer risk for benzene were used to estimate Hazard Quotient and cancer risk because no information was available for diethyl benzene. For tributyl phosphate, the reference concentration for phosphoric acid was used to estimate the Hazard Quotient because no information was available for tributyl phosphate. Propylene oxide cancer unit was used for propylene.
Key: NA, not applicable (the chemical is not a known carcinogen or it is a carcinogen and only unit cancer risk will apply).
Source: DOE 1996a; EPA 1999; model results, using the SCREEN3 computer code (EPA 1995).

4.3.1.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with FFTF target irradiation, REDC neptunium-237 target processing, and RPL medical, industrial, research and development isotope processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

The accident analysis incorporates external events (e.g., earthquakes, fires) as well as internal events (e.g., equipment failure, human errors). A recent external event of concern is the threat of wildfires. While two large range fires at Hanford in 1984 and in June 2000 burned very close to FFTF, neither caused any damage or operational difficulties at the facility. Several features of FFTF make it well equipped to deal with a large range fire. A more detailed discussion is provided in Section I.1.1.4.1.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the

offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are presented in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, the summation of the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–21** and **4–22**, respectively.

FFTF would operate for 21 years with a mixed oxide core followed by 14 years with a highly enriched uranium core. As shown in Table 4–21, the beyond-design-basis core melt accident would result in the largest radiological consequences among FFTF accidents. To incorporate internal and external initiators, the accident frequency of 1×10^{-6} was selected for the beyond-design-basis core melt accident. For 35 years of operation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 1.23×10^{-8} and 1.20×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00127.

For 35 years of REDC neptunium-237 target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.157.

For 35 years of RPL medical, industrial, and research and development target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 4.51×10^{-4} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.377

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 4.51×10^{-4} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.535.

The irradiation of medical, industrial, research and development, and neptunium-237 targets at FFTF would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of medical, industrial, research and development, or neptunium-237 targets at FFTF.

Processing associated with the plutonium-238 production program at REDC, including storage of neptunium-237 and plutonium-238, neptunium-237 target fabrication, postirradiation processing to extract plutonium-238 and to recycle the unconverted neptunium-237 into new targets, does not require the introduction of hazardous chemicals that are not in current use in the facility. The quantities of in-process hazardous chemicals for the plutonium-238 production program are bounded by the quantities of the material currently stored in the facility. The impacts of in-process hazardous chemical accidents associated with the plutonium-238 production are bounded by the impacts of hazardous chemical accidents for existing storage facilities at REDC.

Table 4–21 FFTF, REDC, and RPL Accident Consequences Under Alternative 1 (Restart FFTF)—Option 1

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
FFTF accidents						
Design-basis-accident primary sodium spill (MOX)	0.00113	5.65×10^{-7}	78.6	0.0393	0.00313	1.25×10^{-6}
Design-basis-accident primary sodium spill (HEU)	8.63×10^{-4}	4.32×10^{-7}	72.6	0.0363	0.00181	7.24×10^{-7}
Beyond-design-basis core melt accident (MOX)	0.679	3.40×10^{-4}	6.68×10^{4}	33.4	0.679	2.72×10^{-4}
Beyond-design-basis core melt accident (HEU)	0.481	2.41×10^{-4}	6.16×10^{4}	30.8	0.375	1.50×10^{-4}
BLTC driver fuel-handling accident (MOX)	0.00383	1.92×10^{-6}	1,280	0.639	0.357	1.43×10^{-4}
BLTC driver fuel-handling accident (HEU)	0.00384	1.92×10^{-6}	1,230	0.617	0.340	1.36×10^{-4}
BLTC neptunium-237 target-handling accident	2.61×10^{-4}	1.31×10^{-7}	25.8	0.0129	0.0279	1.12×10^{-5}
BLTC isotope target-handling accident	1.22×10^{-4}	6.10×10^{-8}	2.74	0.00137	0.0143	5.72×10^{-6}
REDC accidents						
Ion exchange explosion during neptunium-237 target fabrication	6.13×10^{-9}	3.06×10^{-12}	8.58×10^{-5}	4.29×10^{-8}	5.60×10^{-10}	2.24×10^{-13}
Target dissolver tank failure during plutonium-238 separation	1.76×10^{-7}	8.79×10^{-11}	0.00196	9.82×10^{-7}	1.69×10^{-8}	6.74×10^{-12}
Ion exchange explosion during plutonium-238 separation	4.68×10^{-4}	2.34×10^{-7}	5.23	0.00261	4.49×10^{-5}	1.79×10^{-8}
Plutonium-238 processing facility beyond-design-basis earthquake	163	0.163	8.91×10^{5}	445	1,310	1.00[c]
RPL accidents						
Medical and industrial isotopes localized solvent fire	0.0135	6.74×10^{-6}	77.8	0.0389	0.0047	1.88×10^{-6}
Medical and industrial isotopes unlikely seismic event	1.52	7.60×10^{-4}	1,350	0.675	1.50	6.00×10^{-4}
Medical and industrial isotopes glovebox explosion	50.0	0.050	4.60×10^{4}	23.0	49.0	0.0392

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–22 FFTF, REDC, and RPL Accident Risks Under Alternative 1 (Restart FFTF)—Option 1

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual FFTF risks			
Design-basis-accident primary sodium spill (MOX) (1×10^{-4})	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis-accident primary sodium spill (HEU) (1×10^{-4})	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX) (1×10^{-6})	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU) (1×10^{-6})	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX) (1×10^{-7})	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU) (1×10^{-7})	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident (1×10^{-7})	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident (1×10^{-7})	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
35-year FFTF risk	1.23×10^{-8}	0.00127	1.20×10^{-8}
Annual REDC risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	1.63×10^{-6}	0.00445	1.00×10^{-5}(c)
35-year REDC risk	5.71×10^{-5}	0.157	3.50×10^{-4}
Annual RPL risks			
Medical and industrial isotopes localized solvent fire (0.044)	2.99×10^{-7}	0.00173	8.35×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	7.60×10^{-6}	0.00675	6.00×10^{-6}
Medical and industrial isotopes glovebox explosion (1×10^{-4})	5.00×10^{-6}	0.00230	3.92×10^{-6}
35-year RPL risk	4.51×10^{-4}	0.377	3.50×10^{-4}
35-year Option risk[d]	4.51×10^{-4}	0.535	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.
d. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Processing associated with the medical, industrial, and research and development isotope production program at RPL, including target fabrication and postirradiation processing, would not require the introduction of hazardous chemicals that are not in current use in the facility. The quantities of in-process hazardous chemicals for the medical and industrial isotope production program are bounded by the quantities of the material currently stored in the facility. The impacts of in-process hazardous chemical accidents associated with the medical and industrial isotope production are bounded by the impacts of hazardous chemical accidents for existing storage facilities at RPL.

4.3.1.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to FFTF. Following irradiation in FFTF, the targets would be returned to REDC for processing. After this processing, the plutonium-238 product would be shipped to LANL. FFTF would receive highly enriched uranium fuel from a U.S. fuel fabrication facility and mixed oxide fuel from Europe. Additionally, medical and industrial isotopes would be shipped from FFTF to a local airport, and from there to locations throughout the country.

Approximately 38,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 8.0 million kilometers (5.0 million miles); at sea by ships carrying mixed oxide fuel, 96,000 kilometers (52,000 nautical miles); and in the air carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 31 person-rem; the dose to the public, 299 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.012 latent cancer fatality among transportation workers and 0.15 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.03. About half of the crew risk, about 2 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to REDC with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.19 traffic fatality. Nearly all of the radiological and traffic accident risk would result from shipping medical and industrial isotopes.

IMPACTS OF MARINE TRANSPORTATION. The potential impacts of marine transport of mixed oxide fuel on the global commons (i.e., portions of the ocean not within the territorial boundary of any nation) were

evaluated in accordance with Executive Order 12114 (44 FR 1957). Following a hypothetical severe accident, radioactive particles dispersed over the ocean would not be in large enough amounts to have a measurable impact on the environment. The risks of accidents approaching and docking at the port have been estimated to be less than 1×10^{-9} person-rem, resulting in less than 1×10^{-12} latent cancer fatality. The radiological doses associated with incident-free transportation, which include the exposure of the ship's crew to low levels of radiation during transport and handling of the packages, have been estimated to be approximately 0.03 person-rem for a route to an east coast port and 0.06 person-rem for a route to a west coast port. These doses would result in 1.2×10^{-5} and 2.4×10^{-5} latent cancer fatalities, respectively.

4.3.1.1.12 Environmental Justice

NORMAL OPERATIONS. The number of expected latent cancer fatalities among the population residing within 80 kilometers (50 miles) of REDC at ORR and FFTF and RPL at Hanford would be less than 0.005 for 35 years of normal operations (Table 4–17). As shown in Tables 4–19 and 4–20, the release of hazardous chemicals at Hanford and ORR would pose no significant risk of cancer or toxic effects among the public. As discussed in Sections K.5.2 and K.5.3, the expected latent cancer fatalities that would result from the ingestion of food that could be radiologically contaminated due to normal operations would be approximately 0.002 at Hanford and essentially zero at ORR. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near Hanford or ORR. As discussed in Section 4.3.1.1.11, incident-free transportation would not be expected to result in fatalities.

ACCIDENTS. Expected latent cancer fatalities among populations at risk due to radiological accidents listed in Table 4–22 would be approximately 0.5. In the event a radiological accident were to occur at REDC and winds were from the southwest, the predominantly minority population of the Scarboro Community adjacent to the northern boundary of ORR would lie in the path of highest potential radiological exposure (see Figure K–6). If the winds were from the west, the predominantly minority populations in Knoxville, Tennessee, would lie in the path of exposure. Because the accidents that could occur under the implementation of this option would not be expected to result in significant offsite exposures to any exposed offsite individual or populations, neither situation would result in a disproportionately high and adverse risk to any group or individuals within the population. If a radiological accident were to occur at FFTF or the 300 Area at Hanford and northeasterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Yakama Indian Reservation (see Figure K–11). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the population or maximally exposed individual residing within the boundary of the Yakama Indian Reservation.

The number of expected latent cancer fatalities resulting from transportation accidents with radiological emissions was found to be approximately 0.5. As discussed in Appendix J, this risk is driven by accidents that could occur during air transportation of medical and industrial isotopes and the conservative assumptions used in the analysis of such accidents. Such accidents could occur anywhere along the flight paths and would not place any identifiable group within the general population at disproportionate risk. As discussed in Section 4.3.2.1.11 and Appendix J, expected fatalities due to a fatal traffic collision would be approximately 0.2.

In summary, normal operations and accidents that could result from the implementation of this option would pose no significant radiological or nonradiological risks to the public, and implementation would pose no disproportionately high and adverse risks to any group within the population.

4.3.1.1.13 Waste Management

The expected generation rates of waste at Hanford that would be generated from the operation of FTFF for irradiating targets and RPL/306–E for processing and fabricating target materials for the research and development support and medical and industrial isotope production are compared with Hanford's treatment, storage, and disposal capacities in **Table 4–23**. The expected generation rates of waste at ORR that would be associated with the operation of REDC to fabricate and process the neptunium-237 targets for plutonium-238 production are compared with ORR's treatment, storage, and disposal capacities in **Table 4–24**.

Table 4–23 Incremental Waste Management Impacts of Operating FFTF and RPL/306–E at Hanford Under Alternative 1 (Restart FFTF)—Option 1

Waste Type[a]	Estimated Additional Waste Generation for FFTF (cubic meters per year)	Estimated Total Waste Generation for FFTF Operation[b] (cubic meters per year)	Estimated Additional Waste Generation for RPL/306–E (cubic meters per year)	Estimated Additional Waste Generation (both FFTF and RPL/306–E) as a Percent of[c]		
				Onsite Treatment Capacity	Onsite Storage Capacity	Onsite Disposal Capacity
High-level radioactive	0	0	0	0	0	0
Transuranic	0	0	0	0	0	0
Low-level radioactive						
Liquid	0	<6	0	0	0	0
Solid	63	80	20	NA	NA	0.17
Mixed low-level radioactive	0	<0.5	4	NA	0.83	0.98
Hazardous	0	4	<1	NA	NA	NA
Nonhazardous						
Process wastewater	22,000	98,000	16	(d)	(d)	(d)
Sanitary wastewater	1,900	5,700	0	0.81[e]	NA	NA
Solid	130	250	20	NA	NA	NA

a. See definitions in Section G.9.
b. These estimates represent the sum of the standby waste generation amounts provided for the No Action Alternative (Table 4–6) and the additional waste generation amounts given in the first column of this table (Table 4–23).
c. The estimated additional amounts of waste generated annually are compared with the annual site treatment capacities. The estimated total amounts of additional waste generated over the assumed 35-year operational period are compared with the site storage and disposal capacities.
d. Refer to the text.
e. Percent of capacity of the Energy Northwest Sewage Treatment Facility.

Note: To convert from cubic meters per year to cubic yards per year, multiply by 1.308; < means "less than."
Key: NA, not applicable (i.e., the majority of this waste is not routinely treated, stored, or disposed of on site; refer to the text in this section).
Source: DOE 2000a; Nielsen 1999.

Table 4–24 Incremental Waste Management Impacts of Operating REDC at ORR Under Alternative 1 (Restart FFTF)—Option 1

Waste Type[a]	Estimated Additional Waste Generation (cubic meters per year)	Estimated Additional Waste Generation as a Percent of [b]		
		Onsite Treatment Capacity	Onsite Storage Capacity	Onsite Disposal Capacity
Transuranic/High-level radioactive[c]	11	(c)	18	NA[d]
Low-level radioactive				
Liquid	25	0.13	24[e]	46
Solid	35	NA[f]	2.6[g]	NA[h]
Mixed low-level radioactive				
Liquid	NA[i]	NA[i]	NA[i]	NA[i]
Solid	<5	<2.2[j]	<0.57[k]	NA[h]
Hazardous	6,500 kilograms	NA[l]	NA[l]	NA[l]
Nonhazardous				
Process wastewater	23	0.0017	NA[m]	NA[m]
Sanitary wastewater	2,832	0.0068	NA	NA
Solid	148	NA[n]	NA[n]	0.42

a. See definitions in Section G.9.
b. The estimated additional amounts of waste generated annually are compared with the annual site treatment capacities. The estimated total amounts of additional waste generated over the assumed 35-year operational period are compared with the site storage and disposal capacities.
c. Refer to the text for a discussion on waste classification and treatment.
d. This waste would be stored on site pending availability of a suitable repository. It is assumed this waste would be remotely handled.
e. Liquid low-level radioactive waste is processed through an evaporator for volume reduction. The evaporator bottoms are stored as a concentrated solution.
f. The solid low-level radioactive waste would not be treated on site.
g. Refer to the text for a discussion of potential limitations of the onsite storage capacity for solid low-level radioactive waste and the probable solution.
h. It is anticipated that solid low-level radioactive waste and solid mixed low-level radioactive waste would be disposed of at an off site facility.
i. Reported as low-level radioactive waste.
j. In the short-term, the Toxic Substances Control Act Incinerator would be used for the treatment of solid mixed low-level radioactive waste. If this facility is shut down, the site's management and integration contractor would identify other options for treatment of this waste.
k. Refer to the text for a discussion of potential limitations of the onsite storage capacity for solid mixed low-level radioactive waste and the probable solution.
l. Although there is some treatment and storage capacity for hazardous waste, this waste would be shipped off site to permitted commercial facilities.
m. The nonhazardous process wastewater would be discharged to a permitted outfall or otherwise disposed of off site after onsite treatment.
n. Solid nonhazardous waste would be taken to the Oak Ridge Y–12 landfill for disposal.

Note: To convert from cubic meters per year to cubic yards per year, multiply by 1.308; to convert from kilograms to pounds, multiply by 2.20; < means "less than."
Key: NA, not applicable.
Source: Brunson 1999b; Wham 1999c, 1999d, 1999e.

The impacts on the Hanford and ORR waste management systems in terms of managing the additional waste are discussed in this section. This analysis is consistent with policy and DOE Order 435.1, that DOE radioactive waste shall be treated, stored, and in the case of low-level waste, disposed of at the site where the waste is generated, if practical, or at another DOE facility. However, if DOE determines that use of the Hanford waste management infrastructure or other DOE sites is not practical or cost effective, DOE may issue an exemption under DOE Order 435.1 for the use of non-DOE facilities (i.e., commercial facilities) to store,

treat, and dispose of such waste generated from the restart and operation of FFTF. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.3.1.1.9 through 4.3.1.1.11.

Canisters used to transport neptunium-237 to ORR would constitute a very small additional amount of solid low-level radioactive waste—less than 10 cubic meters (13.1 cubic yards) over the 35-year operational period, even if no credit is taken for volume reduction by compaction (Brunson 1999a). The annual generation of this waste would fall within the range of accuracy of the generation rate of solid low-level radioactive waste given in Table 4–24, and its management need not be addressed separately.

In accordance with the Records of Decision for the *Waste Management PEIS* (DOE 1997a), waste could be treated and disposed of on site at Hanford or at other DOE sites or commercial facilities. Based on the Record of Decision for high-level radioactive waste issued on August 12, 1999 (64 FR 46661), immobilized high-level radioactive waste would be stored on site until transfer to a geologic repository. Based on the Record of Decision for transuranic waste issued on January 20, 1998 (63 FR 3629), transuranic waste would be certified on site and eventually shipped to a suitable geologic repository for disposal. Based on the Record of Decision for hazardous waste issued on August 5, 1998 (63 FR 41810), nonwastewater hazardous waste would continue to be treated and disposed of at offsite commercial facilities. Based on the Record of Decision for low-level radioactive waste and mixed low-level radioactive waste issued on February 18, 2000 (65 FR 10061), minimal treatment of low-level radioactive waste will be performed at all sites and, to the extent practicable, onsite disposal of low-level radioactive waste will continue. Hanford and the Nevada Test Site will be made available to all DOE sites for disposal of low-level radioactive waste. Mixed low-level radioactive waste analyzed in the *Waste Management PEIS* (DOE 1997a) will be treated at Hanford, INEEL, ORR, and SRS and will be disposed of at Hanford and the Nevada Test Site.

No high-level radioactive waste or transuranic waste would be generated from merely operating FFTF or from target fabrication and processing in RPL/306–E.

Solid low-level radioactive waste generated from target irradiation at FFTF and fabrication and processing in RPL/306–E would be packaged in appropriate containers or burial casks, certified, and transferred for additional treatment and disposal in the existing onsite low-level radioactive Burial Grounds. Liquid low-level radioactive waste generated from target irradiation at FFTF and fabrication and processing in RPL/306–E would be transported to the 200 Area Effluent Treatment Facility for processing and ultimate disposal.

An additional 2,200 cubic meters (2,900 cubic yards) of solid low-level radioactive waste would be generated over the 35-year operational period as a result of target irradiation at FFTF as compared to the current standby mode for FFTF. Target fabrication and processing at RPL/306–E would generate about 700 cubic meters (920 cubic yards) of solid low-level radioactive waste over the 35-year operational period. The total amount of additional solid low-level radioactive waste resulting from operations at FFTF and RPL/306–E represents approximately 0.17 percent of the 1.74-million-cubic-meter (2.28-million-cubic-yard) capacity of the low-level radioactive Burial Grounds. Using the 3,480-cubic-meter-per-hectare (1,842-cubic-yard-per-acre) disposal land usage factor for Hanford published in the *Storage and Disposition PEIS* (DOE 1996a:E-9), 2,900 cubic meters (3,800 cubic yards) of waste would require 0.83 hectares (2.1 acres) of disposal space at Hanford. The impacts of managing this additional low-level radioactive waste at Hanford would be minimal.

There would be no increase in liquid low-level radioactive waste generation as a result of target irradiation at FFTF as compared to the current standby mode for FFTF, nor for target fabrication and processing at the RPL/306–E.

Mixed low-level radioactive waste would be stabilized, packaged, and stored on site for treatment and disposal in a manner consistent with the Tri-Party Agreement (EPA et al. 1989) for Hanford. Over the 35-year operational period, no additional mixed low-level radioactive waste would be generated as a result of target irradiation at FFTF as compared to the current standby mode. Mixed low-level radioactive waste generated at RPL/306–E associated with target fabrication and processing is estimated over the 35-year operation period to be about 140 cubic meters (180 cubic yards). This mixed low-level radioactive waste is expected to be treated at a nearby commercial facility. This additional waste is estimated to be about 0.83 percent of the 16,800-cubic-meter (22,000-cubic-yard) storage capacity of the Central Waste Complex and about 0.98 percent of the 14,200-cubic-meter (18,600-cubic-yard) planned disposal capacity of the Radioactive Mixed Waste Disposal Facility. Therefore, this additional waste would only have a minimal impact on the management of mixed low-level radioactive waste at Hanford.

Hazardous waste generated during operation would be packaged in DOT-approved containers and shipped off site to permitted commercial recycling, treatment, and disposal facilities. The additional waste load generated during the 35-year operational period would have only a minimal impact on the Hanford hazardous waste management system.

Nonhazardous solid waste would be packaged and transported in conformance with standard industrial practice. Solid waste such as office paper, metal cans, and plastic and glass bottles that can be recycled would be sent off site for that purpose. The remaining solid sanitary waste would be sent for offsite disposal. This additional waste load would have only a minimal impact on the nonhazardous solid waste management system at Hanford.

Nonhazardous sanitary wastewater from FFTF operations would be discharged to the 400 Area sanitary sewer system, which connects to the Energy Northwest Sewage Treatment Facility. Nonhazardous sanitary wastewater generated at FFTF would represent about 0.81 percent of the 235,000-cubic-meter-per-year (307,000-cubic-yard-per-year) capacity of the Energy Northwest Sewage Treatment Facility.

Nonhazardous process wastewater from FFTF would be discharged into the 400 Area Ponds. This discharge is regulated by State Waste Discharge Permit ST-4501. Nonhazardous process wastewater generated from target fabrication and processing in RPL/306–E would be discharged to the 300 Area Treated Effluent Disposal Facility.

The generation rates of waste at Hanford that would be associated with this option (refer to Table 4–23) can be compared with the current waste generation rates at the site, given in Table 3–34 (Section 3.4.11). The waste generation rates would be much smaller than the current waste generation rates at the site.

The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. Irrespective of how the waste is classified (i.e., transuranic or high-level radioactive waste), the composition and characteristics are the same, and the waste management activities (i.e., treatment and onsite storage) as described in this NI PEIS would be the same. In addition, either waste type would require disposal in a suitable repository. If it is transuranic waste, it would be nondefense waste and could not be disposed of at WIPP under current law. Because nondefense transuranic waste has no current disposal path, DOE Headquarters' approval would be necessary before a decision were made to generate such waste, as required by DOE Order 435.1. If the waste is classified as high-level radioactive waste, it is assumed for the purposes of this analysis that Yucca Mountain, Nevada, if approved, would be the final disposal site for DOE's high-level radioactive waste. The other differences between these two waste classifications are that a high-level radioactive waste repository requires a much more

rigorous waste-form qualification process than a transuranic waste repository and there is a slightly different set of requirements for high-level radioactive waste than for transuranic waste delineated in DOE Manual 435.1.

Target fabrication and processing in REDC would generate a total of 385 cubic meters (504 cubic yards) of transuranic or high-level radioactive waste over the 35-year operational period. As described in Section 3.4.5 of the *Preconceptual Design Planning for Chemical Processing to Support Pu-238 Production* (Wham 1998), the waste would be vitrified into a glass matrix at a glass melter installed within REDC. The resulting glass matrix would be stored on site pending availability of a suitable repository. This additional waste would represent approximately 18 percent of the available 2,169-cubic-meter (2,837-cubic-yard) storage capacity in facilities 7572, 7574, 7826, 7878, 7879, and 7883. The impacts of managing the additional quantities of this waste at ORR would be minimal.

Low-level radioactive waste at ORR would be treated, packaged, certified, and accumulated before transfer for additional treatment and disposal at onsite and offsite facilities. Annual liquid low-level radioactive waste generation (including mixed low-level radioactive waste—refer to Table 4–24) that would be associated with neptunium-237 target fabrication and processing in REDC is estimated to be 0.13 percent of the 19,908-cubic-meter-per-year (26,040-cubic-yard-per-year) site treatment capacity. If all the liquid low-level radioactive waste generated over the 35-year operational period were stored on site, the amount would represent 24 percent of the 3,646-cubic-meter (4,769-cubic-yard) storage capacity at ORR, and 46 percent of the estimated onsite disposal capacity of 1,894 cubic meters (2,477 cubic yards) of tank storage for liquid low-level radioactive waste from the Liquid Low-Level Waste Evaporator Facility Building 2531. Solid low-level radioactive waste would not be treated on site. If all the solid low-level radioactive waste generated over the 35-year operational period were stored on site, the amount would represent 2.6 percent of the 47,000-cubic-meter (61,500-cubic-yard) storage capacity at ORR. If account is taken of the existing inventory of solid low-level radioactive waste (41,000 cubic meters [53,600 cubic yards]) and of its present generation rate (7,000 cubic meters [9,160 cubic yards] per year), sufficient storage capacity probably would not be available. However, this should be considered only an interim situation. Arrangements are being made that would allow the solid low-level radioactive waste to be treated and disposed of off site at another DOE site or at a commercial facility, thereby eliminating any onsite storage problems, including the storage capacity limitations at ORR. A draft *Environmental Assessment for Transportation of Low-Level Radioactive Waste from the Oak Ridge Reservation to Off-Site Treatment and Disposal Facilities* (DOE 2000d) was issued by the Oak Ridge Operations Office.

The management of the additional low-level radioactive waste from 35 years of operating REDC to fabricate and process neptunium-237 targets would not have a major impact on ORR's ability to manage low-level radioactive waste.

Mixed low-level radioactive waste associated with neptunium-237 target fabrication and processing at ORR would be stabilized, packaged, and stored on site for treatment and disposal in a manner consistent with the site treatment plan. Liquid mixed low-level radioactive waste is reported as low-level radioactive waste; the generation and management of this waste are covered under the low-level radioactive waste discussion above. Solid mixed low-level radioactive waste generation is estimated to be less than 2.2 percent of the 227-cubic-meter-per-year (297-cubic-yard-per-year) site treatment capacity. If all the solid mixed low-level radioactive waste generated over the 35-year operational period were stored on site, the amount would represent less than 0.57 percent of the 30,780-cubic-meter (40,260-cubic-yard) storage capacity at ORR. However, if account is taken of the existing inventory of solid mixed low-level radioactive waste (24,964 cubic meters [32,700 cubic yards]) and of its present generation rate (801 cubic meters [1,050 cubic yards] per year), part or all of the storage capacity may not be available. As is the case for the solid low-level radioactive waste, arrangements are being made that would allow the solid mixed low-level radioactive waste to be disposed of

off site at another DOE site or at a commercial facility, thereby eliminating any onsite storage problems, including the storage capacity limitations at ORR. A draft *Environmental Assessment for Transportation of Low-Level Radioactive Mixed Waste from the Oak Ridge Reservation to Off-Site Treatment or Disposal Facilities* (DOE 2000e) was developed by the Oak Ridge Operations Office.

Managing the small additional quantities of mixed waste that would be generated at ORR would not impact ORR's management of this type of waste.

At ORR, hazardous waste associated with the fabrication and processing of neptunium-237 targets at REDC would be packaged in DOT-approved containers, and shipped off site to permitted commercial recycling, treatment, and disposal facilities. The additional waste load generated during the operational period would only have a minimal impact on ORR's management of hazardous waste.

Nonhazardous solid waste associated with neptunium-237 target fabrication and processing in REDC would be packaged in conformance with standard industrial practices and disposed of in the onsite landfills. If all the nonhazardous solid waste generated over the 35-year operational period were disposed of in Industrial Landfills V and VI, only 0.42 percent of the 1,219,000-cubic-meter (1,594,000-cubic-yard) total capacity of these landfills would be needed. Nonhazardous sanitary wastewater from REDC operations would be discharged to the sanitary wastewater treatment facility. Nonhazardous process wastewater would be processed, as necessary, in the wastewater treatment facilities before discharge to an outfall or other offsite disposition facility. The additional solid and liquid waste loads would only have a minimal impact on nonhazardous waste management at ORR.

The generation rates of waste at ORR that would be associated with this option (Table 4–24) can be compared with the current waste generation rates at the site, given in Table 3–11 (Section 3.2.11). The waste generation rates associated with plutonium-238 production would be much smaller than the current waste generation rates at the site. However, if the waste resulting from processing irradiated neptunium-237 is classified as high-level radioactive waste, although ORR does not currently manage high-level radioactive waste, the impacts to the waste management infrastructure would be minimal.

4.3.1.1.14 Spent Nuclear Fuel Management

Data on spent nuclear fuel generation and storage under all options of Alternative 1 are presented in **Table 4–25**.

Table 4–25 Data for Spent Nuclear Fuel Generation and Storage Under All Options of Alternative 1 (Restart FFTF)

Data Parameter	At FFTF
Operating duration (years)	35
Operating power level (megawatts)	100
Existing spent nuclear fuel inventory (metric tons of heavy metal)	11[a]
Method of storage	Sodium-cooled vessels and dry storage casks
Number of spent nuclear fuel assemblies generated annually	About 12 to 15 (i.e., 2 casks per year)
Spent nuclear fuel generated in 35 years (metric tons of heavy metal)	16

a. The total spent nuclear fuel inventory at Hanford is 2,133 metric tons of heavy metal.
Note: To convert from metric tons to pounds, multiply by 2,200.
Source: DOE 2000a.

The operation of FFTF would generate about 0.46 metric ton heavy metal (1,012 pounds) of spent nuclear fuel per year. For the 35-year mission at 100 megawatts, this would equate to a total of 16 metric tons of heavy

metal (35,200 pounds) of spent nuclear fuel, which is less than 1 weight-percent of the total spent nuclear fuel inventory presently stored at Hanford.

The currently authorized storage modes for the FFTF spent nuclear fuel include two sodium-filled storage vessels within the facility and the interim storage area located at the northeast corner of the FFTF site which now is capable of accommodating spent nuclear fuel in 49 aboveground dry storage casks. It is projected that these storage modes will provide enough capacity at the reactor site for 35 years of reactor operation. This projection is based on the assumption that the nonfuel irradiated components are disposed of and do not remain in storage. If it is conservatively assumed that this hardware remains in storage, the number of spaces available for spent nuclear fuel storage would be reduced. With this worst-case assumption, it is projected that the current storage modes would support 24 years of reactor operation. Since the operation of FFTF would result in the generation of 12 to 15 spent nuclear fuel assemblies per year and each dry storage cask is capable of storing 7 assemblies, the additional storage capacity for years 25 through 35 of reactor operation could be provided by loading 2 additional dry storage casks per year.

Upon cessation of reactor operation, or earlier, the spent nuclear fuel would be packaged in acceptable containers and shipped to a geologic repository for disposal. Refer to Section 4.6.1.3.13 for further information on the geologic repository.

CONSTRUCTION IMPACTS. The interim storage area is currently authorized for spent nuclear fuel storage in 49 dry storage casks. Prior to standby, 30 dry storage casks were procured, 18 of which are storing spent nuclear fuel and 12 of which are currently empty. It is anticipated that with additional cask procurement, the interim storage area, as currently authorized, would provide enough capacity for 35 years of reactor operation. As such, no construction impacts associated with expanding the dry cask storage capability of the interim storage area would be incurred.

However, based on the worst case assumption that all the irradiated nonfuel hardware would remain in storage, it is possible that the interim storage area would need to accommodate 20 additional dry storage casks. The construction impact of providing an additional concrete storage pad north of the existing concrete pad would be minimal.

OPERATIONAL IMPACTS. Operation of the sodium-filled storage vessels and the dry storage casks would not result in significant releases of radionuclides to the environment. The airborne radionuclides emitted from overall FFTF operations have always been at levels practically indistinguishable from natural background radiation. During the last year of reactor operations (1992), the overall radionuclide releases from the entire FFTF complex resulted in a total effective dose equivalent to the maximally exposed member of the public of less than 1.0×10^{-4} millirem (DOE 2000a). This dose is well below EPA's Clean Air Act standard of 10 millirem per year that is cited in DOE Order 5400.5. Any dose contribution from the storage vessels would be expected to be only a small fraction of the overall dose. No radionuclide releases from the dry cask storage system would occur because the spent nuclear fuel is contained in a sealed confinement.

Although no radionuclides are expected to be released from the dry storage cask, the cask would be a source of direct and skyline-scattered radiation that would penetrate the thick concrete shielding of the cask. The direct radiation is from neutron and gamma sources emitted from the spent nuclear fuel, with the greatest contribution coming from the gamma source. Based on the operating experience of the Independent Spent Fuel Storage Installation (ISFSI) facilities (BGE 1989; NRC 1986; Duke 1988; NRC 1985), the direct radiation dose to an individual 100 meters from the cask was calculated to be in the range of 0.01 to 0.1 millirem per hour. This direct radiation would have an effect only on onsite workers; the radiation dose is greatly reduced to insignificant levels beyond the site boundary. The whole body dose to an offsite

individual (at about or more than 1,000 meters [0.62 mile] from the site) for these ISFSIs is normally less than 1 millirem per year.

The operation of the dry storage system would generate a small quantity of decay heat, which is removed by natural air convection and would not have any effect on the offsite environment.

There would be no liquid releases to the environment associated with spent nuclear fuel management. The environmental impacts associated with the dry spent nuclear fuel storage system are summarized in **Table 4–26**. The dry spent fuel storage at the FFTF site is similar to NRC-approved methods currently being used for interim storage of commercial spent nuclear fuel.

Table 4–26 Environmental Impact of Dry Spent Nuclear Fuel Storage System Under All Options of Alternative 1 (Restart FFTF)

Environmental Parameter	Environmental Impact
Radiological impacts (normal operation)	Dose of less than 0.1 millirem per year, well below EPA's Clean Air Act standard of 10 millirem per year
Effect of decay heat on the site	Equivalent to 210 light bulbs (100 watts each); no offsite effect
Facility water use	Small
Liquid and solid radwaste generated	Small; no discharges to the environment
Chemical and biocide generated	Minimal (if any)
Effect of sanitary waste discharges	Minimal
Noise and traffic impacts	Minimal
Effect of maintenance of the electrical system	Minimal
Effect on ecology	Minimal
Socioeconomics	Small; fewer than five additional people would be employed

Source: BGE 1989; Duke 1988; NRC 1985; NRC 1986.

4.3.2 Alternative 1 (Restart FFTF)—Option 2

Option 2 involves operating FFTF at Hanford to irradiate all targets and materials associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating FDPF at INEEL to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and operating facilities in the Hanford 300 Area to fabricate and process the other targets and materials and to process the associated products. This option includes storage in Building CPP–651 or FDPF of the neptunium-237 transported from SRS to INEEL and storage in RPL/306–E of the other target materials transported to Hanford from other offsite facilities.

The transportation of the mixed oxide and highly enriched uranium fuel to Hanford for use in FFTF, the transportation of the neptunium-237 to INEEL and then to Hanford, the transportation of the other target material to Hanford, and the transportation of the product materials following irradiation and postirradiation processing are also part of this option.

Under Option 2, FFTF would operate with a mixed oxide fuel core for the first 21 years and with a highly enriched uranium fuel core for the next 14 years.

4.3.2.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations and with all transportation activities are assessed in this section.

4.3.2.1.1 Land Resources

LAND USE. The restart of FFTF would not result in impacts on land use at Hanford for the reasons described in Section 4.3.1.1.1.

Building CPP–651 and/or FDPF, which are both in the INTEC area of INEEL, would be used for neptunium-237 storage, and FDPF for target fabrication and processing. The use of either facility would require internal modifications, but no new facilities would be built. Because additional land would not be disturbed and the use of Building CPP–651 and/or FDPF would be compatible with the missions for which they were designed, there would be no change in land use at INEEL.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on land use at Hanford for the reasons described in Section 4.3.1.1.1.

VISUAL RESOURCES. The restart of FFTF would not result in impacts on visual resources at Hanford for the reasons described in Section 4.3.1.1.1.

All activities associated with neptunium-237 storage, target fabrication, and processing would take place in Building CPP–651 and/or FDPF. Because neither facility would require external modification, there would be no change in appearance. Therefore, the current Visual Resource Management Class IV rating for INTEC would not change. Because there would be no change in the appearance of either of these facilities or the INTEC area, there would be no impact on visual resources.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on visual resources at Hanford for the reasons described in Section 4.3.1.1.1.

4.3.2.1.2 Noise

For the restart of FFTF, the change in noise impacts from construction and operation would be expected to be small as described in Section 4.3.1.1.2.

Building CPP–651 and/or FDPF, both in the INTEC area of INEEL, would be used for neptunium-237 target-material storage, and FDPF for target fabrication and processing. Interior modifications of these facilities in the INTEC area of INEEL would be expected to result in little change in noise impacts on wildlife around this area. The operation of these facilities would not be expected to result in any change in noise impacts on wildlife around the INTEC area and offsite noise impacts would be small because the nearest site boundary is 12 kilometers (7.5 miles) to the south. Operation would be expected to result in minimal change in noise impacts on people near the INEEL as a result of changes in employee and truck traffic levels.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. Interior modifications of these facilities and operation would be expected to result in little change in noise impacts on wildlife around this area and people near Hanford as described in Section 4.3.1.1.2.

4.3.2.1.3 Air Quality

Under this option, air quality impacts due to the restart and operation of FFTF would be the same as under Option 1 (Section 4.3.1.1.3). Air quality impacts from target fabrication and processing in the Hanford 300 Area facility would be the same as under Option 1 (Section 4.3.1.1.3).

The concentrations at INEEL attributable to FDPF operations under this option are presented in **Table 4–27**. The concentrations are based on a dispersion-modeling screening analysis conducted with maximum expected emission rates and a set of worst-case meteorological conditions. Criteria and toxic air pollutants were modeled for a stack height of 48.8 meters (160 feet) at a boundary limit of 6,800 meters (22,300 feet). Only those air pollutants expected to be emitted that have ambient air quality standards are presented in the table. The change in concentrations of these pollutants would be small and would be below applicable ambient standards even when ambient monitored values and the contribution from other site activities were included.

Table 4–27 Incremental INEEL Concentrations[a] Associated with Alternative 1 (Restart FFTF)—Option 2

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Criteria pollutants			
Nitrogen dioxide	Annual	100	3.66×10^{-4}
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.024 0.19 0.43
Toxic air pollutants			
Methanol	24 hours	13,000	0.0048
Nitric acid	24 hours	250	0.0097
Paraffin hydrocarbons	24 hours	100	0.44
Tributyl phosphate	24 hours	110	0.25

a. For comparison with ambient air quality standards.
Note: Toxic air pollutant standards apply to new or modified sources only.
Source: 40 CFR Part 50; ID DHW 1998; modeled increments are based on the SCREEN3 computer code (EPA 1995).

The concentrations at INEEL attributed to this option are compared with the Prevention of Significant Deterioration Class II increments for nitrogen dioxide and sulfur dioxide in **Table 4–28**.

Table 4–28 PSD Class II Increments Compared to INEEL Concentrations Associated with Alternative 1 (Restart FFTF)—Option 2

Pollutant	Averaging Period	Allowable PSD Increment (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Nitrogen dioxide	Annual	25	3.66×10^{-4}
Sulfur dioxide	Annual 24 hours 3 hours	20 91 512	0.024 0.19 0.43

Key: PSD, Prevention of Significant Deterioration.
Source: Modeled PSD increments are based on the SCREEN3 computer code (EPA 1995).

Health impacts from FDPF chemical releases are discussed in Section 4.3.2.1.9.

The air quality impacts of transportation are presented in Section 4.3.2.1.11.

4.3.2.1.4 Water Resources

Impacts on water resources at Hanford associated with the restart of FFTF would be substantially the same as those described in Section 4.3.1.1.4.

Building CPP–651 and/or FDPF, which are both located within the INTEC area of INEEL, would be used for neptunium-237 storage with target fabrication and processing in support of plutonium-238 production conducted in FDPF. The projected incremental effects on key water resource indicators are summarized in **Table 4–29**. As existing facilities would be used, there would be no construction-related impacts on water bodies, floodplains, or on surface water or groundwater quality. A relatively small increase in water use and sanitary wastewater generation is projected mainly attributable to the additional staffing required at FDPF (see Section 4.3.2.1.8). The only other measurable increase would be an additional 23,000 liters (6,100 gallons) per year of process wastewater associated with target processing in FDPF (Kirkham 1999; Wham 1999c). All wastewater would be discharged to designated collection and treatment systems as described in Section 3.3.4.1.2. There would be no radiological liquid effluent discharge to the environment under normal operations, and no measurable impact on water resources at INEEL would be expected.

Table 4–29 Incremental Water Use and Wastewater Generation Associated with Operating FDPF at INEEL Under Alternative 1 (Restart FFTF)—Option 2

Indicator (million liters per year)	INEEL
	FDPF
Water use	1.68
Process wastewater generation	0.023
Sanitary wastewater generation	1.66

Note: To convert from liters per year to gallons per year, multiply by 0.264.
Source: Kirkham 1999; Wham 1999c.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. As a result, it is expected that impacts on water resources at Hanford would be negligible as previously described in Section 4.3.1.1.4.

Waste management aspects of this option and their effects are further discussed in Section 4.3.2.1.13.

4.3.2.1.5 Geology and Soils

The restart of FFTF would not be expected to result in impacts on geologic and soil resources at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.1.2.5 and 4.3.1.1.5.

Because existing facilities (i.e., Building CPP–651 and/or FDPF) would be used, there would be no disturbance to either geologic or soil resources at INTEC. Hazards from large-scale geologic conditions, such as earthquakes and volcanoes, were previously evaluated as discussed in Section 4.2.3.2.5. The analysis determined that these hazards present a low risk for neptunium-237 storage in INTEC facilities. Likewise, large-scale geologic conditions do not present a substantial risk to use of the proposed facilities for neptunium-237 storage, target fabrication, and processing.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not be expected to result in impacts on geologic resources at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.1.2.5 and 4.3.1.1.5. As necessary,

the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.3.2.1.6 Ecological Resources

The restart of FFTF would not be expected to result in impacts on ecological resources at Hanford for the reasons described in Section 4.3.1.1.6.

Because no new construction is planned, the use of Building CPP–651 and/or FDPF at INEEL would not result in direct disturbance to ecological resources. As noted in Section 4.3.2.1.2, there would be little change in noise impacts on wildlife. Because additional water usage and wastewater discharge would be small fractions of current values, and discharge chemistry would not be expected to change, there would be no impact on aquatic resources (Section 4.3.2.1.4). Due to the developed nature of the area and the fact that no new construction would take place, impacts on threatened and endangered species would not occur.

Consultation letters to comply with Section 7 of the Endangered Species Act were sent to the U.S. Fish and Wildlife Service and the Idaho Department of Fish and Game (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. The Idaho Department of Fish and Game indicated that their database contained no known occurrences of special status plants or animals near the project area. While DOE has made additional contact with the U.S. Fish and Wildlife Service, a response is pending from this agency. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at INEEL prior to the receipt of input from the Service.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on ecological resources at Hanford for the reasons described in Section 4.3.1.1.6.

4.3.2.1.7 Cultural and Paleontological Resources

The restart of FFTF would not result in impacts on cultural resources at Hanford for the reasons described in Section 4.3.1.1.7.

No new construction is planned; therefore, direct impacts on cultural and paleontological resources at INTEC would not occur. The use of Building CPP–651 and/or FDPF to store neptunium-237 or FDPF to fabricate and process neptunium-237 targets would not change the status of six historic structures located at INTEC. Also, Native American resources occurring in the vicinity of INTEC would not be impacted.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). The State Historic Preservation Office indicated that Building CPP–651 and FDPF are likely to be eligible for the National Register of Historic Places as contributory properties in a potential historic district of exceptional significance. However, at this time, the State Historic Preservation Office has determined that more information is needed prior to assisting DOE in evaluating these properties. The State Historic Preservation Office also indicated that since there would be no new construction, there is little potential for effects on archaeological properties. DOE would provide additional information as required to the Idaho State Historic Preservation Office prior to the use of any facility at INEEL for the proposed project. Consultation was conducted with interested Native American tribes; however, responses are pending.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on cultural resources at Hanford for the reasons described in Section 4.3.1.1.7.

4.3.2.1.8 Socioeconomics

The socioeconomic impacts associated with restarting and operating FFTF to irradiate all targets, and operating RPL/306–E to fabricate and process all other targets are addressed in Section 4.3.1.1.8.

Target fabrication and processing of neptunium-237 targets at INEEL would require approximately 24 additional workers (Hill et al. 1999). This level of employment could generate 64 indirect jobs in the region around INEEL. The potential total employment increase of 88 direct and indirect jobs in the INEEL region represents less than 0.1 percent of the projected regional economic area workforce. It would have no noticeable impact on the regional economic area.

Additional employment resulting from this option would not have any noticeable impact on community services in the INEEL region of influence. Assuming 94 percent of the new employment associated with this alternative would reside in INEEL's region of influence (Section 3.3.8), 83 new jobs could increase the region's population by approximately 161 persons. This increase in conjunction with normal population growth forecasted by the State of Idaho would not have any noticeable effect on the availability of housing and/or the price of housing in the region of influence. The public would experience little or no change in the level of community services currently offered in the region of influence.

4.3.2.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup, processing, and operations are given in **Table 4–30** for FFTF and RPL at Hanford and FDPF at INEEL: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

To represent a bounding annual dose scenario at Hanford, it is assumed that a full-year's isotopic release would occur from target processing at RPL concurrently with a full-year's release from FFTF operations at 400 megawatts; the impacts presented in Table 4–30 also assume a full-year's release resulting from FFTF and RPL preoperational testing and startup activities. To represent a bounding annual dose scenario at INEEL, it is assumed that a full year's release would occur from neptunium-237 target processing at FDPF.

Table 4–30 Incremental Radiological Impacts on the Public Around INEEL and Hanford from Operational Facilities Under Alternative 1 (Restart FFTF)—Option 2

Receptor	INEEL FDPF Processing[a]	Hanford Preoperational Activities[b]		Hanford FFTF Operations	Hanford RPL Target Processing[a]	Hanford Operations and Processing Total[c]
		FFTF	RPL			
Population within 80 kilometers (50 miles) in the year 2020						
Dose (person-rem)	3.9×10^{-6}	0.028	1.0[d]	0.044	0.21	0.25
1-year latent cancer fatalities	–	1.4×10^{-5}	5.0×10^{-4}	–	–	–
35-year latent cancer fatalities	6.7×10^{-8}	–	–	7.7×10^{-4}	0.0037	0.0044
Maximally exposed individual						
Annual dose (millirem)	2.6×10^{-7}	1.4×10^{-4}	0.043[c]	4.1×10^{-4}	0.0050	0.0054
1-year latent cancer fatality risk	–	6.8×10^{-11}	2.2×10^{-8}	–	–	–
35-year latent cancer fatality risk	4.6×10^{-12}	–	–	7.2×10^{-9}	8.8×10^{-8}	9.5×10^{-8}
Average exposed individual within 80 kilometers (50 miles)						
Annual dose[e] (millirem)	2.0×10^{-8}	5.7×10^{-5}	0.0020[c]	8.8×10^{-5}	4.2×10^{-4}	5.0×10^{-4}
1-year latent cancer fatality risk	–	2.8×10^{-11}	9.9×10^{-10}	–	–	–
35-year latent cancer fatality risk	3.6×10^{-13}	–	–	1.5×10^{-9}	7.3×10^{-9}	8.8×10^{-9}

a. Target storage, processing, and fabrication activities are performed at the facility. Impacts are for all facility target activities and are dominated by processing activity impacts.
b. For conservatism as well as consistency with other radiological impacts evaluated in this NI PEIS, these values were assessed for the year 2020 even though these activities would commence prior to that year.
c. Represents upper-bounding values.
d. Annual emissions during preoperational activities were assumed to be the same as the 1998 releases for RPL (BWHC 1999). The majority of this dose is due to tritium releases.
e. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the facilities in the year 2020 (about 505,000 for Hanford and 188,400 for INEEL).

Source: Model results, using the GENII computer code (Napier et al. 1988).

As a result of annual operations, the bounding projected total incremental population dose in the year 2020 for the populations surrounding Hanford and INEEL would be 0.25 person-rem. The corresponding number of latent cancer fatalities in these populations from 35 years of operations would be 0.0044. The bounding total incremental dose to the maximally exposed member of the public from annual operations at Hanford would be 0.0054 millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 9.5×10^{-8}. The incremental dose to the maximally exposed member of the public from annual operations at FDPF would be 2.6×10^{-7} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 4.6×10^{-12}.

Incremental doses to involved workers from normal operations are given in **Table 4–31**; these workers are defined as those directly associated with all process and operational activities. The incremental annual average dose to FDPF workers would be 170 millirem; the incremental annual average dose to FFTF workers (during startup) would be 3.5 millirem; the incremental annual average dose to FFTF workers (during operations) would be 6.6 millirem; the incremental annual average dose to RPL workers (during startup) would be 81 millirem; and the incremental annual average dose for RPL workers (during processing) would be approximately 160 millirem. The incremental annual dose received by the total workforce for each of these

facilities (at the different phases) would be approximately 12, 0.69, 1.3, 3.2, and 4.8 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers are included in Table 4–31. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–31 Incremental Radiological Impacts on Involved FDPF, FFTF, and RPL Workers Under Alternative 1 (Restart FFTF)—Option 2

Receptor—Involved Workers[a]	INEEL FDPF Processing[b]	Hanford Preoperational Activities		Hanford		
		FFTF	RPL	FFTF Operations	RPL Target Processing[b]	Operations and Processing Total
Total dose (person-rem per year)	12[c]	0.69[d]	3.2[e]	1.3[d]	4.8[f]	6.1
1-year latent cancer fatalities	–	2.8×10^{-4}	0.0013	–	–	–
35-year latent cancer fatalities	0.17	–	–	0.018	0.067	0.086
Average worker dose (millirem per year)	170	3.5	81	6.6	160	NA
1-year latent cancer fatality risk	–	1.4×10^{-6}	3.2×10^{-5}	–	–	–
35-year latent cancer fatality risk	0.0023	–	–	9.2×10^{-5}	0.0022	NA

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Target storage, processing, and fabrication activities are performed at this facility. Impacts, dominated by processing activities, include impacts from all facility target activities.
c. Based on an estimated 75 badged workers.
d. Based on an estimated 200 badged workers.
e. Based on an estimated 40 badged workers.
f. Based on an estimated 30 badged workers.

Key: NA, not applicable.
Source: BWHC 1999; Mecham 1999; Nielsen 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts associated with FFTF restart and target fabrication and processing in the 300 Area at Hanford were determined to be the same as for Option 1 (Section 4.3.1.1.9). Hazardous chemical impacts associated with processing in FDPF at INEEL are presented in **Table 4–32** and show little effect from air pollutant releases associated with this option.

4.3.2.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with FFTF target irradiation, FDPF neptunium-237 target processing, and RPL medical and industrial target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the

Table 4–32 Incremental Hazardous Chemical Impacts on the Public Around INEEL Under Alternative 1 (Restart FFTF)—Option 2

Chemical	Modeled Annual Increment (micrograms per cubic meter)	RfC (micrograms per cubic meter)	Unit Cancer Risk (risk per micrograms per cubic meter)	Hazard Quotient	Cancer Risk
Diethyl benzene	0.0165	1,000	7.80×10^{-6}	1.65×10^{-5}	1.29×10^{-7}
Methanol	6.02×10^{-4}	1,750	NA	3.44×10^{-7}	NA
Nitric acid	0.00121	122.5	NA	9.86×10^{-6}	NA
Tributyl phosphate	0.031	10	NA	0.0031	NA
			Hazard Index =	0.0031	

Note: For diethyl benzene, the reference concentration for ethyl benzene and the unit cancer risk for benzene were used. For tributyl phosphate, the reference concentration for phosphoric acid was used to estimate the Hazard Quotient because no information was available for tributyl phosphate.
Key: NA, not applicable (the chemical is not a known carcinogen); RfC, Reference Concentration.
Source: DOE 1996a; EPA 1999; model results, using the SCREEN3 computer code (EPA 1995).

accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are presented in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, the summation of the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–33** and **4–34**, respectively.

FFTF would operate for 21 years with a mixed oxide core followed by 14 years with a highly enriched uranium core. As shown in Table 4–33, the beyond-design-basis core melt accident would result in the largest radiological consequences among FFTF accidents. To incorporate internal and external initiators, the accident frequency of 1×10^{-6} was selected for the beyond-design-basis core melt accident. For 35 years of operation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 1.23×10^{-8} and 1.20×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00127.

For 35 years of FDPF neptunium-237 target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 1.49×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0287.

For 35 years of RPL medical, industrial, and research and development target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 4.51×10^{-4} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.377.

Table 4–33 FFTF, RPL, and FDPF Accident Consequences Under Alternative 1 (Restart FFTF)—Option 2

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
FFTF accidents						
Design-basis-accident primary sodium spill (MOX)	0.00113	5.65×10^{-7}	78.6	0.0393	0.00313	1.25×10^{-6}
Design-basis-accident primary sodium spill (HEU)	8.63×10^{-4}	4.32×10^{-7}	72.6	0.0363	0.00181	7.24×10^{-7}
Beyond-design-basis core melt accident (MOX)	0.679	3.40×10^{-4}	6.68×10^{4}	33.4	0.679	2.72×10^{-4}
Beyond-design-basis core melt accident (HEU)	0.481	2.41×10^{-4}	6.16×10^{4}	30.8	0.375	1.50×10^{-4}
BLTC driver fuel-handling accident (MOX)	0.00383	1.92×10^{-6}	1,280	0.639	0.357	1.43×10^{-4}
BLTC driver fuel-handling accident (HEU)	0.00384	1.92×10^{-6}	1,230	0.617	0.340	1.36×10^{-4}
BLTC neptunium-237 target-handling accident	2.61×10^{-4}	1.31×10^{-7}	25.8	0.0129	0.0279	1.12×10^{-5}
BLTC isotope target-handling accident	1.22×10^{-4}	6.10×10^{-8}	2.74	0.00137	0.0143	5.72×10^{-6}
FDPF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.01×10^{-9}	1.01×10^{-12}	2.49×10^{-5}	1.24×10^{-8}	7.26×10^{-9}	2.91×10^{-12}
Target dissolver tank failure during plutonium-238 separation	6.11×10^{-8}	3.05×10^{-11}	5.65×10^{-4}	2.82×10^{-7}	2.17×10^{-7}	8.69×10^{-11}
Ion exchange explosion during plutonium-238 separation	1.63×10^{-5}	8.13×10^{-9}	0.150	7.51×10^{-5}	5.79×10^{-5}	2.31×10^{-8}
Plutonium-238 processing facility beyond-design-basis earthquake	42.5	0.0425	1.64×10^{5}	82.0	1,200	1.0[c]
RPL accidents						
Medical and industrial isotopes localized solvent fire	0.0135	6.74×10^{-6}	77.8	0.0389	0.0047	1.88×10^{-6}
Medical and industrial isotopes unlikely seismic event	1.52	7.60×10^{-4}	1,350	0.675	1.50	6.00×10^{-4}
Medical and industrial isotopes glovebox explosion	50.0	0.050	4.60×10^{4}	23.0	49.0	0.0392

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–34 FFTF, RPL, and FDPF Accident Risks Under Alternative 1 (Restart FFTF)—Option 2

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual FFTF risks			
Design-basis-accident primary sodium spill (MOX) (1×10^{-4})	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis-accident primary sodium spill (HEU) (1×10^{-4})	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX) (1×10^{-6})	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU) (1×10^{-6})	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX) (1×10^{-7})	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU) (1×10^{-7})	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident (1×10^{-7})	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident (1×10^{-7})	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
35-year FFTF risk	1.23×10^{-8}	0.00127	1.20×10^{-8}
Annual FDPF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	4.25×10^{-7}	8.20×10^{-4}	$1.00\times10^{-5(c)}$
35-year FDPF risk	1.49×10^{-5}	0.0287	3.50×10^{-4}
Annual RPL risks			
Medical and industrial isotopes localized solvent fire (0.044)	2.99×10^{-7}	0.00173	8.35×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	7.60×10^{-6}	0.00675	6.00×10^{-6}
Medical and industrial isotopes glovebox explosion (1×10^{-4})	5.00×10^{-6}	0.00230	3.92×10^{-6}
35-year RPL risk	4.51×10^{-4}	0.377	3.50×10^{-4}
35-year Option risk[d]	4.51×10^{-4}	0.407	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.
d. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 4.51×10^{-4} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.407.

The irradiation of medical, industrial, research and development, and neptunium-237 targets at FFTF would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of medical, industrial, or neptunium-237 targets at FFTF.

No chemical processing activities are currently performed at FDPF and no chemicals are stored in this facility. Processing activities in support of plutonium-238 production would require the introduction of hazardous chemicals, specifically nitric acid and nitric oxide. Potential health impacts from accidental releases of nitric acid were assessed by comparing estimated airborne concentrations of the chemicals to Emergency Response Planning Guidelines (ERPG) developed by the American Industrial Hygiene Association. The ERPG-1 value (0.5 part per million) is the maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour, resulting in only mild, transient, and reversible adverse health effects. The ERPG-2 value (10 parts per million) is protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. The ERPG-3 value (25 parts per million) is indicative of potentially life-threatening health effects.

The maximum distances, in meters, needed to reach the ERPG values for nitric acid releases at FDPF for Stability Classes D and F are shown in **Table 4–35.** Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions while Stability Class F represents worst-case meteorological conditions. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they are sheltered within buildings at the time of release. Individuals at the nearest highway (5,800 meters [3.6 miles]) and at the nearest site boundary (13,952 meters [8.7 miles]) from FDPF would be exposed to levels well below ERPG-1.

Table 4–35 ERPG Distances for Nitric Acid Releases at FDPF

Evaluation Parameter	Stability Class D (meters)	Stability Class F (meters)
ERPG-3	375	450
ERPG-2	500	600
ERPG-1	2,000	3,000

Note: To convert from meters to miles, multiply by 6.22×10^{-4}.
Key: ERPG, Emergency Response Planning Guideline.

There are no ERPG values for nitric oxide. For nitric oxide accidents, the level of concern has been estimated by using one-tenth of the "Immediately Dangerous to Life and Health" level published by the National Institute for Occupational Safety and Health. The Immediately Dangerous to Life and Health value for nitric oxide is 100 parts per million. The level of concern value used for this PEIS is 10 parts per million. The level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects as a result of a single exposure for a relatively short period of time.

For FDPF, the maximum distances needed to reach the level of concern for nitric oxide releases for Stability Classes D and F are 500 and 2,000 meters (1,640 and 6,560 feet), respectively. The number of involved and noninvolved workers potentially exposed would vary with a number of factors such as the time of day and whether they are sheltered within buildings at the time of release. Individuals at the nearest highway (5,800 meters [3.6 miles]) and at the nearest site boundary (13,952 meters [8.7 miles]) from FDPF would be exposed to levels well below the level of concern for nitric oxide.

Potential health impacts from the accidental release of the hazardous chemicals were assessed for a noninvolved worker, offsite individuals who are members of the public located at the nearest site boundary, and onsite individuals who are members of the public located at the nearest highway access onsite.

The impacts associated with the accidental release of nitric acid and nitric oxide at FDPF are presented in **Table 4–36**.

Table 4–36 FDPF Hazardous Chemical Accident Impacts Under Alternative 1 (Restart FFTF)—Option 2

Receptor	Evaluation Parameter	Nitric Acid		Nitric Oxide	
		Stability Class D	Stability Class F	Stability Class D	Stability Class F
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	3.3 <ERPG-2 Mild, transient	8.4 <ERPG-2 Mild, transient	4.2 <LOC Mild, transient	67.5 >LOC Serious
Nearest highway maximally exposed individual	Parts per million Level of concern Potential health effects	0.05 < ERPG-1 None	0.15 ERPG-1 Mild, transient	0.09 < LOC None	0.87 < LOC None
Site boundary maximally exposed individual	Parts per million Level of concern Potential health effects	<<0.05 < ERPG-1 None	<<0.15 ERPG-1 Mild, transient	<<0.09 < LOC None	<<0.87 < LOC None

Note: < means "less than"; << means "much less than."
Key: ERPG, Emergency Response Planning Guideline; LOC, level of concern.
Source: Model results.

Processing associated with the medical, industrial, and research and development isotope production program at RPL, including target fabrication and postirradiation processing, would not require the introduction of hazardous chemicals that are not in current use in the facility. The quantities of in-process hazardous chemicals for the medical and industrial isotope production program are bounded by the quantities of the material currently stored in the facility. The impacts of in-process hazardous chemical accidents associated with the medical, industrial, and research and development isotope production are bounded by the impacts of hazardous chemical accidents for existing storage facilities at RPL.

4.3.2.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FDPF target fabrication facility at INEEL. DOE would transport the unirradiated neptunium-237 targets from FDPF to FFTF. Following irradiation in FFTF, the targets would be returned to FDPF for processing. After this processing, the plutonium-238 product would be shipped to LANL. FFTF would receive highly enriched uranium fuel from a U.S. fuel fabrication facility and mixed oxide fuel from Europe. Additionally, medical and industrial isotopes would be shipped from FFTF to a local airport, and from there to locations throughout the country.

Approximately 38,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 6.2 million kilometers (3.9 million miles); at sea by ships carrying mixed oxide fuel, 96,000 kilometers (52,000 nautical miles); and in the air carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impacts analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 21 person-rem; the dose to the public, 88 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.008 latent cancer fatality among transportation workers and 0.044 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.024. About half of the crew risk, about 8 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.13 traffic fatality. Nearly all of the radiological and traffic accident risk would result from shipping medical and industrial isotopes.

IMPACTS OF MARINE TRANSPORTATION. The potential impacts of marine transport of mixed oxide fuel on the global commons (i.e., portions of the ocean not within the territorial boundary of any nation) were evaluated in accordance with Executive Order 12114 (44 FR 1957). Following a hypothetical severe accident, radioactive particles dispersed over the ocean would not be in large enough amounts to have a measurable impact on the environment. The risks of accidents approaching and docking at the port have been estimated to be less than 1×10^{-9} person-rem, resulting in less than 1×10^{-12} latent cancer fatality. The radiological doses associated with incident-free transportation, which include the exposure of the ship's crew to low levels of radiation during transport and handling of the packages, have been estimated to be approximately 0.03 person-rem for a route to an east coast port and 0.06 person-rem for a route to a west coast port. These doses would result in 1.2×10^{-5} and 2.4×10^{-5} latent cancer fatalities, respectively.

4.3.2.1.12 Environmental Justice

NORMAL OPERATIONS. The number of expected latent cancer fatalities among the populations residing within 80 kilometers (50 miles) of FDPF at INEEL and FFTF and RPL at Hanford would be less than 0.005 for 35 years of normal operations (Table 4–30). As shown in Table 4–32, the release of hazardous chemicals at INEEL would pose no significant risk of cancer or toxic effects among the public. As discussed in Sections K.5.1 and K.5.3, the expected latent cancer fatalities that would result from the ingestion of food that could be radiologically contaminated due to normal operations would be approximately 0.002 at Hanford and essentially zero at INEEL. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near Hanford or INEEL. As shown in Section 4.3.2.1.11, incident-free transportation would not be expected to result in fatalities.

ACCIDENTS. The number of expected latent cancer fatalities among populations at risk due to radiological accidents listed in Table 4–34 would be approximately 0.41. In the event a radiological accident were to occur

at FDPF and northwesterly winds prevailed at the time of the accident, radiological contamination would be directed toward the Fort Hall Indian Reservation (see Figure K–2). If a radiological accident were to occur at FFTF or the 300 Area at Hanford and northeasterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Yakama Indian Reservation (see Figure K–11). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the populations or a maximally exposed individuals residing within the boundaries of the Fort Hall Indian Reservation or Yakama Indian Reservation.

The number of expected latent cancer fatalities resulting from transportation accidents with radiological emissions was found to be approximately 0.5. As discussed in Appendix J, this risk is driven by accidents that could occur from air transportation of medical and industrial isotopes and the conservative assumptions used in the analysis of such accidents. Such accidents could occur anywhere along the flight paths and would not place any identifiable group within the general population at disproportionate risk. As discussed in Section 4.3.2.1.11 and Appendix J, expected fatalities due to a traffic collision would be approximately 0.14.

In summary, normal operations and accidents that could result from the implementation of this option would pose no significant radiological or nonradiological risks to the public, and implementation would pose no disproportionately high and adverse risks to any group within the population.

4.3.2.1.13 Waste Management

The impacts of managing waste generated from irradiating targets in FFTF and processing and fabricating target materials for the research and development support and medical and industrial isotope production in RPL/306–E are assumed to be the same as for Option 1 (Section 4.3.1.1.13). This is because the same amount of plutonium-238 production, medical and industrial isotope production, and civilian nuclear energy research and development support would be accomplished annually. As discussed in that section, the impacts on Hanford's waste management systems would be minimal.

The expected generation rates of waste that would be associated with the operation of FDPF to fabricate and process neptunium-237 targets are compared with INEEL's treatment, storage, and disposal capacities in **Table 4–37**. The impacts on the INEEL waste management systems, in terms of managing the additional waste, are discussed in this section. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.3.2.1.9 through 4.3.2.1.11.

Canisters used to transport neptunium-237 to INEEL would constitute a very small additional amount of solid low-level radioactive—less than 10 cubic meters (13.1 cubic yards) over the 35-year operational period, even if no credit is taken for volume reduction by compaction (Brunson 1999a). The annual generation of this waste would fall within the range of accuracy of the generation rate of solid low-level radioactive waste given in Table 4–37, and its management need not be addressed separately.

In accordance with the Records of Decision for the *Waste Management PEIS* (DOE 1997a), waste could be treated and disposed of on site at INEEL or at other DOE sites or commercial facilities. Based on the Record of Decision for high-level radioactive waste issued on August 12, 1999 (64 FR 46661), immobilized high-level radioactive waste would be stored on site until transfer to a geologic repository. Based on the Record of Decision for transuranic waste issued on January 20, 1998 (63 FR 3629), transuranic waste would be certified on site and eventually shipped to a suitable geologic repository for disposal. Based on the Record of Decision for hazardous waste issued on August 5, 1998 (63 FR 41810), nonwastewater hazardous waste would continue to be treated and disposed of at offsite commercial facilities. Based on the Record of Decision for low-level radioactive waste and mixed low-level radioactive waste issued on February 18, 2000 (65 FR 10061), minimal

Table 4–37 Incremental Waste Management Impacts of Operating FDPF at INEEL Under Alternative 1 (Restart FFTF)—Option 2

Waste Type[a]	Estimated Additional Waste Generation (cubic meters per year)	Estimated Additional Waste Generation as a Percent of [b]		
		Onsite Treatment Capacity	Onsite Storage Capacity	Onsite Disposal Capacity
Transuranic waste/High-level radioactive[c]	7	(c)	(c)	NA
Low-level radioactive				
Liquid[d]	30	0.23	(e)	(e)
Solid	35	(e)	NA	0.093
Mixed low-level radioactive				
Liquid	(d)	(d)	(d)	(d)
Solid	<5	<0.077	<0.099	NA
Hazardous	6,500 kilograms	NA	2.4	NA
Nonhazardous				
Process wastewater	23	NA	NA	0.14[f]
Sanitary wastewater	1,658	0.00052	NA	NA
Solid	148	NA	NA	0.31

a. See definitions in Section G.9.
b. Estimated additional annual waste generation is compared with annual site treatment and disposal capacities. Additional waste generation over the assumed 35-year operational period is compared with site storage capacities.
c. Refer to the text for a discussion on waste classification, treatment, and storage. This waste would be stored on site pending availability of a suitable repository. It is assumed this waste would be remotely handled.
d. Mixed liquid low-level radioactive waste is included under liquid low-level radioactive waste because these wastes are processed together.
e. Refer to the text. The impact on the waste management system would be minimal.
f. Percent of capacity of the two INTEC percolation ponds.

Note: To convert from cubic meters per year to cubic yards per year, multiply by 1.308; to convert from kilograms to pounds, multiply by 2.20; < means "less than."
Key: INTEC, Idaho Nuclear Technology and Engineering Center; NA, not applicable (i.e., the majority of this waste is not routinely treated, or is not routinely stored, or is not routinely disposed of on site; refer to the text).
Source: Brunson 1999b; DOE 1999a; Kirkham 1999; Wham 1999d.

treatment of low-level radioactive waste will be performed at all sites and, to the extent practicable, onsite disposal of low-level radioactive waste will continue. Hanford and the Nevada Test Site will be made available to all DOE sites for disposal of low-level radioactive waste. Mixed low-level radioactive waste analyzed in the *Waste Management PEIS* will be treated at Hanford, INEEL, ORR, and SRS and will be disposed of at Hanford and the Nevada Test Site.

The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. Irrespective of how the waste is classified (i.e., transuranic or high-level radioactive waste), the composition and characteristics are the same and the waste management activities (i.e., treatment and onsite storage) as described in this NI PEIS would be the same. In addition, either waste type would require disposal in a suitable repository. If it is transuranic waste, it would be nondefense waste and could not be disposed of at WIPP under current law. Because nondefense transuranic waste has no current disposal path, DOE Headquarters' approval would be necessary before a decision were made to generate such waste, as required by DOE Order 435.1. If the waste is classified as high-level radioactive waste, it is assumed for the purposes of this analysis that Yucca Mountain, Nevada,

if approved, would be the final disposal site for DOE's high-level radioactive waste. The other differences between these two waste classifications are that a high-level radioactive waste repository requires a much more rigorous waste-form qualification process than a transuranic waste repository and there is a slightly different set of requirements for high-level radioactive waste than for transuranic waste delineated in DOE Manual 435.1.

Target fabrication and processing in FDPF would generate a total of 245 cubic meters (320 cubic yards) of transuranic or high-level radioactive waste over the 35-year operational period. As described in Sections 3.4.5 of the *Preconceptual Design Planning for Chemical Processing to Support Pu-238 Production* (Wham 1998), the waste would be vitrified into a glass matrix at a glass melter installed within FDPF. The resulting glass matrix would be stored at FDPF pending availability of the suitable repository. The impacts of managing the additional quantities of this waste at INEEL would be minimal.

At INEEL, low-level radioactive waste from neptunium-237 fabrication and processing would be packaged, certified, and accumulated at FDPF before transfer for additional treatment as necessary, by compaction, size reduction, or stabilization on site or by incineration off site and then sent for disposal in existing onsite facilities. Annual liquid low-level radioactive waste generation, including mixed liquid low-level radioactive waste that would be associated with neptunium-237 target fabrication and processing in FDPF, is estimated to be 0.23 percent of the 13,000-cubic-meter-per-year (17,000-cubic-yard-per-year) capacity of the INTEC Process Equipment Waste evaporator. The condensate from this evaporator is processed by the Liquid Effluent Treatment and Disposal System evaporator and released to the main stack as steam. After any appropriate treatment, liquid waste generated by the neptunium-237 fabrication and processing would eventually be grouted for final disposition.

The annual amount of solid low-level radioactive waste that would be generated at FDPF as the result of neptunium-237 target fabrication and processing is estimated as 0.093 percent of the 37,700-cubic-meter-per-year (49,300-cubic-yard-per-year) disposal capacity of the Radioactive Waste Management Complex. A total of 1,225 cubic meters (1,602 cubic yards) of solid low-level radioactive waste would be generated over the 35-year operational period. Using the 6,264-cubic-meter-per-hectare (3,316-cubic-yard-per-acre) disposal land usage factor for INEEL published in the *Storage and Disposition PEIS* (DOE 1996a:E-9), 1,225 cubic meters (1,602 cubic yards) of waste would require 0.20 hectares (0.48 acres) of disposal space at INEEL. At some future time, low-level radioactive waste would be disposed of off site. The impacts of managing the additional low-level radioactive waste at INEEL would be minimal.

At INEEL, mixed solid low-level radioactive waste would be stabilized, packaged, and stored on site for treatment and disposal in a manner consistent with the site treatment plan. Mixed low-level radioactive waste is currently treated on site with some waste shipped to Envirocare of Utah for disposal. The additional mixed solid low-level radioactive waste that would be generated at FDPF is estimated to be less than 0.077 percent of the 6,500-cubic-meter-per-year (8,500-cubic-yard-per-year) planned capacity of the Advanced Mixed Waste Treatment Project. Over the 35-year operational period, the amount of this waste generated would represent less than 0.099 percent of the 177,300-cubic-meter (231,900-cubic-yard) storage capacity of the Radioactive Waste Management Complex. Therefore, the management of this additional waste at INEEL would have only a minimal impact on the management of mixed low-level radioactive waste at INEEL.

Hazardous waste generated during the operation of FDPF would be packaged in DOT-approved containers and shipped off site to permitted commercial recycling, treatment, and disposal facilities. Hazardous waste generated from 35 years of operating FDPF to fabricate and process the neptunium-237 targets is estimated to represent about 2.4 percent of the 9,600-cubic-meter (12,560-cubic-yard) capacity of the hazardous waste storage buildings (including staging). Management of the additional hazardous waste at INEEL would have only a minimal impact on the hazardous waste management system.

Nonhazardous solid waste would be packaged and transported in conformance with standard industrial practice. Solid waste such as office paper, metal cans, and plastic and glass bottles that can be recycled would be sent off site for that purpose. The remaining solid sanitary waste would be sent to the onsite landfill. This additional waste load would have only a minimal impact on the nonhazardous solid waste management system at INEEL. The annual amount of nonhazardous solid waste that would be generated is estimated to represent 0.31 percent of the 48,000-cubic-meter-per-year (63,000-cubic-yard-per-year) capacity of the Central Facilities Area Landfill Complex.

At INEEL, nonhazardous process wastewater generated by FDPF would be discharged to the INTEC service waste system, which then discharges to the two INTEC percolation ponds. Nonhazardous process wastewater generated as the result of neptunium-237 target fabrication and processing is estimated to be 0.14 percent of the 16,700-cubic-meter-per-year (21,800-cubic-yard-per-year) capacity of the INTEC percolation ponds. Nonhazardous sanitary wastewater from FDPF operations would be discharged to the INTEC Sewage Treatment Plant. Sanitary wastewater generated is estimated to be 0.00052 percent of the 3,200,000-cubic-meter-per-year (4,200,000-cubic-yard-per-year) capacity of the INTEC Sewage Treatment Plant. Therefore, management of nonhazardous liquid waste at INEEL would have only a minimal impact on the management system.

The generation rates of waste at INEEL that would be associated with this option (Table 4–37) can be compared with the current waste generation rates at the site, given in Table 3–25 (Section 3.3.11). Except for transuranic waste, which currently is not being generated at INEEL, the waste generation rates associated with plutonium-238 production would be much smaller than the current waste generation rates at the site.

4.3.2.1.14 Spent Nuclear Fuel Management

Impacts associated with spent nuclear fuel management would be the same as for Option 1, and are given in Section 4.3.1.1.14.

4.3.3 Alternative 1 (Restart FFTF)—Option 3

Option 3 involves operating FFTF at Hanford to irradiate all targets and materials associated with plutonium-238 production, medical and industrial isotope production, and research and development, and also operating FMEF at Hanford to fabricate and process these targets and materials and the associated irradiated products. This option includes storage in FMEF of the neptunium-237 transported to Hanford from SRS and of the other target materials transported to Hanford from other offsite facilities.

The transportation of the mixed oxide and highly enriched uranium fuel to Hanford for use in FFTF, the transportation of the neptunium-237 and other target material to Hanford, and the transportation of the product materials following postirradiation processing are also part of this option.

Under Option 3, FFTF would operate with a mixed oxide fuel core for the first 21 years and with a highly enriched uranium fuel core for the next 14 years.

4.3.3.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations and with all transportation activities are assessed in this section.

4.3.3.1.1 Land Resources

LAND USE. The restart of FFTF would not result in impacts on land use at Hanford for the reasons described in Section 4.3.1.1.1.

FMEF, which is in the 400 Area of Hanford, would be used for target material storage, target fabrication, and processing. The use of this facility would require the construction of a new 76-meter (250-foot) stack. Because the stack would be placed on previously disturbed land, and use of FMEF would be compatible with the mission for which it was designed, land use impacts in the 400 Area would be minimal.

VISUAL RESOURCES. The restart of FFTF would not result in impacts on visual resources at Hanford for the reasons described in Section 4.3.1.1.1.

The use of FMEF for target material storage, target fabrication, and processing would involve the construction of a 76-meter (250-foot) stack. While the stack would be visible from surrounding areas, it would not change the overall appearance of the 400 Area or its Visual Resource Management Class IV rating. Thus, impacts on visual resources would be minimal.

4.3.3.1.2 Noise

The change in noise impacts from FFTF restart and operation would be expected to be small as described in Section 4.3.1.1.2.

FMEF would be used for target material storage, target fabrication, and processing. A new 76-meter (250-foot) stack would be required for neptunium-237 target processing at FMEF. Activities associated with construction of a new stack would be typical of small construction projects and would result in some temporary increase in noise. Noise sources associated with this construction would not be expected to be loud impulsive sources and would not be expected to result in disturbance of wildlife around the 400 Area. FMEF operations would not be expected to result in any change in noise impacts on wildlife around the 400 Area, and offsite noise impacts would also be minor because the nearest site boundary is 7 kilometers (4.3 miles) to the east. Operations would be expected to result in minimal change in noise impacts on people near Hanford as a result of changes in employee and truck traffic levels.

4.3.3.1.3 Air Quality

The restart and operation of FFTF under this option would have the same air quality impacts as under Option 1 (Section 4.3.1.1.3), and are presented in **Table 4–38**. The concentrations at Hanford from FMEF attributable to this option are also presented in Table 4–38. Changes in concentrations were determined to be small and would be below the applicable ambient standards even when ambient monitored values and the contributions from the other site activities were included. Hazardous chemical impacts are addressed in Section 4.3.3.1.9.

The concentrations at Hanford attributable to this option are compared with the Prevention of Significant Deterioration Class II increments for sulfur dioxide and nitrogen dioxide in **Table 4–39**.

The air quality impacts of transportation are presented in Section 4.3.3.1.11.

Table 4–38 Incremental Hanford Concentrations Associated with Alternative 1 (Restart FFTF)—Option 3

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (micrograms per cubic meter)	
			FFTF	FMEF
Criteria pollutants				
Carbon monoxide	8 hours 1 hour	10,000[b] 40,000[b]	52.1 74.4	0 0
Nitrogen dioxide	Annual	100[b]	0.0118	4.43×10^{-5}
PM_{10}	Annual 24 hours	50[c] 150[c]	8.4×10^{-4} 9.84	0 0
Sulfur dioxide	Annual 24 hours 3 hours 1 hour	50[d] 260[d] 1,300[b] 660[d]	7.86×10^{-4} 9.1 20.5 22.8	0.0087 0.069 0.16 0.17
Toxic air pollutants				
Methanol	24 hours	870	0	0.0018
Nitric acid	24 hours	17	0	0.0022
Paraffin hydrocarbons	24 hours	7	0	0.16
Tributyl phosphate	24 hours	7.3	0	0.090

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The 24-hour PM_{10} (particulate matter with an aerodynamic diameter less than or equal to 10 micrometers) standard is attained when the expected number of days with a 24-hour average concentration above the standard is equal to or less than 1. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Federal and state standard.
c. Federal standard currently under litigation.
d. State standard.

Source: 40 CFR Part 50; WDEC 1998; modeled increments are based on the SCREEN3 computer code (EPA 1995); additional data from Nielsen 2000.

Table 4–39 PSD Class II Increments Compared to Hanford Concentrations Associated with FMEF Under Alternative 1 (Restart FFTF)—Option 3

Pollutant	Averaging Period	Allowable PSD Increment (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Nitrogen dioxide	Annual	25	0.0118
Sulfur dioxide	Annual 24 hours 3 hours	20 91 512	0.00949 9.17 20.6

Key: PSD, Prevention of Significant Deterioration.
Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995).

4.3.3.1.4 Water Resources

The restart of FFTF for isotope production and the use of FMEF for target material storage, target fabrication, and processing, both existing facilities located in the Hanford 400 Area, would not have any construction-related impacts on water bodies, floodplains, or on surface water or groundwater quality.

Operational impacts on water resources associated with the restart of FFTF would be substantially the same as those discussed in Section 4.3.1.1.4, with only a small incremental impact associated with FMEF operations. Total projected 400 Area and incremental effects of this option on key water resource indicators are summarized in **Table 4–40**. Annual average groundwater withdrawal during standby by 400 Area facilities

is about 197 million liters (52 million gallons) (Section 4.3.1.1.4). The restart of FFTF combined with the use of FMEF would increase annual water use to a total of 277 million liters (73 million gallons). This is a total increase of about 80 million liters (21 million gallons) per year. This includes some 15 million liters (4 million gallons) per year to support FMEF cooling needs and approximately 3.8 million liters (1 million gallons) per year for increased sanitary and potable water needs (Chapin 2000). This volume of 277 million liters (73 million gallons) per year is approximately 70 percent of the 400 Area groundwater production capacity of about 398 million liters (105.1 million gallons) per year (DOE 1999a:4-262).

Table 4–40 Incremental Water Use and Wastewater Generation Associated with Operating FFTF and FMEF at Hanford Under Alternative 1 (Restart FFTF)—Option 3

Indicator (million liters per year)	Hanford			
	Total 400 Area[a]	FFTF Operations[b]	FFTF Increment Over Standby[c]	FMEF Increment[d]
Water use	277	258	61	19
Process wastewater generation	113	98	22	15
Sanitary wastewater generation	9.5	5.7	1.9	3.8

a. Total projected operational impacts in the Hanford 400 Area (FFTF and FMEF operations combined).
b. These estimates represent total projected operational impacts after restart (FFTF only).
c. Incremental impacts of FFTF restart and operation over standby operations (see Table 4–1).
d. Incremental impacts of FMEF operations only.
Note: To convert from liters per year to gallons per year, multiply by 0.264.
Source: Chapin 2000; DOE 2000a:11; Nielsen 1999:38, 41.

Additional staffing required to support both the restart of FFTF and use of FMEF would also increase annual sanitary wastewater generation in the 400 Area by a total of 5.7 million liters (1.5 million gallons) over standby to about 9.5 million liters (2.5 million gallons) per year during operation. FMEF alone would contribute 3.8 million liters (1 million gallons) annually to this increase (Chapin 2000). Nevertheless, the Energy Northwest treatment system has sufficient excess capacity to accommodate this increased flow from the 400 Area (Section 4.3.1.1.4).

Process (nonradioactive) wastewater discharge from the 400 Area (mainly FFTF and FMEF) would increase by a total of approximately 37 million liters (9.8 million gallons) over standby to about 113 million liters (29.8 million gallons) per year as a result of FFTF and FMEF operations. FMEF would contribute about 15 million liters (4 million gallons) annually based on a conservative estimate of cooling water discharges and blowdown from FMEF's three cooling towers (currently inactive) (Chapin 2000; Nielsen 1999:38). This additional volume includes approximately 38,000 liters (10,000 gallons) per year of process wastewater resulting from target fabrication and processing activities (Chapin 2000). This wastewater would be discharged to the 400 Area process sewer system and ultimately to the 400 Area Pond, with no impact on groundwater quality expected for the same reasons cited in Section 4.3.1.1.4.

Waste management aspects of this option and their effects are further discussed in Section 4.3.3.1.13.

4.3.3.1.5 Geology and Soils

The restart of FFTF would not be expected to result in impacts on geologic and soil resources at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.1.2.5 and 4.3.1.1.5.

FMEF would be used for target material storage, target fabrication, and processing. Additionally, a new 76-meter (250-foot) stack would be constructed (Nielsen 1999:24). Because FMEF is an existing facility and the stack would be located on previously disturbed land, impacts on geologic resources and soils would be negligible. As referenced above, and in Section 4.2.4.2.5, hazards from large-scale geologic conditions at

Hanford were previously evaluated and were reviewed in this NI PEIS and found to present a low risk to FMEF. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.3.3.1.6 Ecological Resources

The restart of FFTF would not be expected to result in impacts on ecological resources at Hanford for the reasons described in Section 4.3.1.1.6.

FMEF, an existing facility, would be used for target material storage, target fabrication, and processing. Impacts on ecological resources resulting from the use of FMEF would not occur for the same reasons noted above for FFTF, which is also in the 400 Area. While a new 76-meter (250-foot) stack would be built, it would be placed on previously disturbed land in the 400 Area; thus, no natural terrestrial habitat would be lost.

4.3.3.1.7 Cultural and Paleontological Resources

The restart of FFTF would not be expected to result in impacts on cultural resources at Hanford for the reasons described in Section 4.3.1.1.7.

Target material storage, target fabrication, and processing would take place at FMEF in the 400 Area. Impacts on cultural resources resulting from the use of FMEF would not occur for the same reasons noted above for FFTF, which is also in the 400 Area. Although a new 76-meter (250-foot) stack would be built, it would be placed on previously disturbed land in the 400 Area; thus, impacts on cultural and paleontological resources would not be expected.

4.3.3.1.8 Socioeconomics

The irradiation of all isotopes at FFTF, and the fabrication and processing of all targets at FMEF would annually require about 292 additional workers at Hanford (Hoyt et al. 1999; DOE 1997b). This level of employment would generate about 739 indirect jobs in the region around Hanford. The potential total employment increase of 1,031 direct and indirect jobs in the Hanford region represents a less than 0.5 percent increase in the projected regional economic area workforce. It would have no noticeable impact on the regional economic area.

Additional employment resulting from this option would not have any noticeable impact on community services in the Hanford region of influence. Assuming that 91 percent of the new employment would reside in Hanford's region of influence (refer to Section 3.4.8), 938 new jobs could increase the region's population by approximately 1,803 persons. This increase, in conjunction with the normal population growth forecasted by the State of Washington, would not have any noticeable impact on the availability of housing and/or the price of housing in the region of influence. Given the current population-to-student ratio in the region of influence, this would likely result in an increase of about 373 students, requiring local school districts to slightly increase the number of classrooms to accommodate them.

Community services in the region of influence would be expected to change to accommodate the population growth as follows: 23 new teachers would be needed to maintain the current student-to-teacher ratio of 16:1; 3 new police officers would need to be added to maintain the current officer-to-population ratio of 1.5:100; 6 new firefighters would need to be added to maintain the current firefighter-to-population ratio of 3.4:1000; and 3 new doctors would be added to maintain the current physician-to-population ratio of 1.4:1000. Thus, an additional 35 positions would have to be created to maintain community services at current levels. Hospitals in the region of influence would not experience any change from the 2.1 beds per 1,000 persons

currently available. Moreover, average school enrollment would increase to 94.5 percent from the current 92.5 percent unless additional classrooms were built. None of these projected changes should have a major impact on the level of community services currently offered in the region of influence.

4.3.3.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup, processing, and operations are given in **Table 4–41**: the population within 80 kilometers (50 miles) of FFTF and FMEF in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–41 Incremental Radiological Impacts on the Public Around Hanford from Operational Facilities Under Alternative 1 (Restart FFTF)—Option 3

Receptor	FFTF Preoperational Activities[a]	FFTF Operations	FMEF Target Processing[b]	Operations and Processing Total[c]
Population within 80 kilometers (50 miles) in the year 2020				
Dose (person-rem)	0.028	0.044	0.085	0.13
1-year latent cancer fatalities	1.4×10^{-5}	–	–	–
35-year latent cancer fatalities	–	7.7×10^{-4}	0.0015	0.0023
Maximally exposed individual				
Annual dose (millirem)	1.4×10^{-4}	4.1×10^{-4}	3.0×10^{-4}	7.0×10^{-4}
1-year latent cancer fatality risk	6.8×10^{-11}	–	–	–
35-year latent cancer fatality risk	–	7.2×10^{-9}	5.3×10^{-9}	1.2×10^{-8}
Average exposed individual within 80 kilometers (50 miles)				
Annual dose[d] (millirem)	5.7×10^{-5}	8.8×10^{-5}	1.7×10^{-4}	2.6×10^{-4}
1-year latent cancer fatality risk	2.8×10^{-11}	–	–	–
35-year latent cancer fatality risk	–	1.5×10^{-9}	3.0×10^{-9}	4.5×10^{-9}

a. For conservatism as well as consistency with other radiological impacts evaluated in this NI PEIS, these values were assessed for the year 2020 even though these activities would commence prior to that year.
b. Target storage, processing, and fabrication activities are performed at the facility. Impacts are for all facility target activities and are dominated by processing activity impacts.
c. Represents upper-bounding values.
d. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FFTF and FMEF in the year 2020 (about 500,000).

Source: Model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

To represent a bounding annual dose scenario, it is assumed that a full-year's isotopic release would occur from target processing at FMEF concurrently with a full-year's release from FFTF operations at 400 megawatts. The impacts presented in Table 4–41 assume a full-year's release resulting from FFTF

preoperational testing and startup activities. As a result of annual operations, the bounding projected total incremental population dose in the year 2020 would be 0.13 person-rem. The corresponding number of latent cancer fatalities in the population surrounding Hanford from 35 years of operations would be 2.3×10^{-3}. The bounding total incremental dose to the maximally exposed member of the public from annual operations of FFTF and FMEF would be 7.0×10^{-4} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 1.2×10^{-8}.

Incremental doses to involved workers from normal operations are given in **Table 4–42**; these workers are defined as those directly associated with all process and operational activities. The incremental annual average dose to FFTF workers during startup would be 3.5 millirem; the incremental annual average dose during operations, 6.6 millirem. For FMEF workers, the incremental annual average dose is estimated to be approximately 160 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be approximately 0.69, 1.3, and 17 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers are included in Table 4–42. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–42 Incremental Radiological Impacts on Involved FFTF and FMEF Workers Under Alternative 1 (Restart FFTF)—Option 3

Receptor—Involved Workers[a]	FFTF Preoperational Activities	FFTF Operations	FMEF Target Processing[b]	Operations and Processing Total
Total dose (person-rem per year)	0.69[c]	1.3[c]	17[d]	18
1-year latent cancer fatalities	2.8×10^{-4}	–	–	–
35-year latent cancer fatalities	–	0.018	0.24	0.26
Average worker dose (millirem per year)	3.5	6.6	160	NA
1-year latent cancer fatality risk	1.4×10^{-6}	–	–	–
35-year latent cancer fatality risk	–	9.2×10^{-5}	0.0023	NA

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Doses are based on a weighted average from historical data associated with plutonium processing and other radiochemical processing. Target storage, processing, and fabrication activities are performed at this facility. Impacts, dominated by processing activities, include impacts from all facility target activities.
c. Based on an estimated 200 badged workers.
d. Based on an estimated 105 badged workers.

Key: NA, not applicable.
Source: BWHC 1999; Mecham 1999; Nielsen 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. At FMEF, both carcinogenic and noncarcinogenic health effects from exposure to hazardous chemicals were evaluated and are presented in **Table 4–43**. It was assumed that under normal operating conditions, the primary exposure pathway for members of the public would be from airborne emissions released through the new 76-meter (250-foot) stack. Emissions of chemicals were estimated based on anticipated chemical usage. A worst-case dispersion-modeling screening analysis was performed to estimate annual concentrations for each chemical.

The annual concentration of each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the Hazard Quotient for each of the noncarcinogenic chemicals associated with this option. The Hazard Quotients were then summed to determine the Hazard Index. A Hazard Index of less than one indicates that adverse health effects from non-cancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical.

Table 4–43 Incremental Hazardous Chemical Impacts on the Public at Hanford Under Alternative 1 (Restart FFTF)—Option 3

Chemical	Modeled Annual Increment (micrograms per cubic meter)	Reference Concentration (micrograms per cubic meter)	Unit Cancer Risk (risk per micrograms per cubic meter)	Hazard Quotient	Cancer Risk
FFTF emergency diesel generators					
Benzene	2.5×10^{-6}	NA	7.8×10^{-6}	NA	1.96×10^{-11}
Toluene	1.10×10^{-6}	400	NA	2.74×10^{-9}	NA
Propylene	6.92×10^{-6}	NA	3.7×10^{-6}	NA	2.56×10^{-11}
Formaldehyde	3.17×10^{-6}	NA	1.3×10^{-5}	NA	4.12×10^{-11}
Acetaldehyde	2.06×10^{-6}	NA	2.2×10^{-6}	NA	4.53×10^{-12}
FMEF					
Nitric acid	2.73×10^{-4}	122.5	NA	2.22×10^{-6}	NA
Diethyl benzene	0.00601	1000	7.8×10^{-6}	6.01×10^{-6}	4.69×10^{-8}
Methanol	2.19×10^{-4}	1750	NA	1.25×10^{-7}	NA
Tributyl phosphate	0.0113	10	NA	0.00113	NA
			Hazard Index =	0.00114	

Note: For diethyl benzene, the reference concentration for ethyl benzene and the unit cancer risk for benzene were used. For tributyl phosphate, the reference concentration for phosphoric acid was used to estimate the Hazard Quotient because no information was available for tributyl phosphate. The propylene oxide unit cancer risk factor was used for propylene.
Key: NA, not applicable (the chemical is not a known carcinogen or it is a carcinogen and only unit risk will apply).
Source: DOE 1996a; EPA 1999; model results, using the SCREEN3 computer code (EPA 1995).

4.3.3.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with FFTF target irradiation and FMEF target fabrication and processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are presented in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, the summation of the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–44** and **4–45**, respectively.

Table 4–44 FFTF and FMEF Accident Consequences Under Alternative 1 (Restart FFTF)—Option 3

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
FFTF accidents						
Design-basis-accident primary sodium spill (MOX)	0.00113	5.65×10^{-7}	78.6	0.0393	0.00313	1.25×10^{-6}
Design-basis-accident primary sodium spill (HEU)	8.63×10^{-4}	4.32×10^{-7}	72.6	0.0363	0.00181	7.24×10^{-7}
Beyond-design-basis core melt accident (MOX)	0.679	3.40×10^{-4}	6.68×10^{4}	33.4	0.679	2.72×10^{-4}
Beyond-design-basis core melt accident (HEU)	0.481	2.41×10^{-4}	6.16×10^{4}	30.8	0.375	1.50×10^{-4}
BLTC driver fuel-handling accident (MOX)	0.00383	1.92×10^{-6}	1,280	0.639	0.357	1.43×10^{-4}
BLTC driver fuel-handling accident (HEU)	0.00384	1.92×10^{-6}	1,230	0.617	0.340	1.36×10^{-4}
BLTC neptunium-237 target-handling accident	2.61×10^{-4}	1.31×10^{-7}	25.8	0.0129	0.0279	1.12×10^{-5}
BLTC isotope target-handling accident	1.22×10^{-4}	6.10×10^{-8}	2.74	0.00137	0.0143	5.72×10^{-6}
FMEF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.02×10^{-9}	1.01×10^{-12}	7.26×10^{-5}	3.63×10^{-8}	6.65×10^{-10}	2.66×10^{-13}
Target dissolver tank failure during plutonium-238 separation	4.64×10^{-8}	2.32×10^{-11}	0.00169	8.47×10^{-7}	1.95×10^{-8}	7.81×10^{-12}
Ion exchange explosion during plutonium-238 separation	1.24×10^{-5}	6.18×10^{-9}	0.451	2.25×10^{-4}	5.20×10^{-6}	2.08×10^{-9}
Medical and industrial isotopes localized solvent fire	0.00276	1.38×10^{-6}	56.2	0.0281	9.51×10^{-5}	3.80×10^{-8}
Medical and industrial isotopes glovebox explosion	1.00	5.00×10^{-4}	2.95×10^{4}	14.8	24.0	0.0192
Processing facility beyond-design-basis earthquake	16.5	0.00825	6.42×10^{5}	321	922	1.00[c]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–45 FFTF and FMEF Accident Risks Under Alternative 1 (Restart FFTF)—Option 3

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual FFTF risks			
Design-basis-accident primary sodium spill (MOX) (1×10^{-4})	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis-accident primary sodium spill (HEU) (1×10^{-4})	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX) (1×10^{-6})	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU) (1×10^{-6})	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX) (1×10^{-7})	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU) (1×10^{-7})	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident (1×10^{-7})	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident (1×10^{-7})	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
35-year FFTF risk	1.23×10^{-8}	0.00127	1.20×10^{-8}
Annual FMEF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Medical and industrial isotopes localized solvent fire (0.044)	6.13×10^{-8}	0.00125	1.69×10^{-9}
Medical and industrial isotopes glovebox explosion (1×10^{-4})	5.00×10^{-8}	0.00148	1.92×10^{-6}
Processing facility beyond-design-basis earthquake (1×10^{-5})	8.25×10^{-8}	0.00321	$1.00\times10^{-5(c)}$
35-year FMEF risk	6.79×10^{-6}	0.208	4.17×10^{-4}
35-year Option Risk[d]	6.80×10^{-6}	0.209	4.17×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.
d. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

FFTF would operate for 21 years with a mixed oxide core followed by 14 years with a highly enriched uranium (HEU) core. As shown in Table 4–44, the beyond-design-basis core melt accident would result in the largest radiological consequences among FFTF accidents. In order to incorporate internal and external initiators, the accident frequency of 1×10^{-6} was selected for the beyond-design-basis core melt accident. For 35 years of operation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 1.23×10^{-8} and 1.20×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00127.

For 35 years of FMEF target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 6.79×10^{-6} and 4.17×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.208.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 6.80×10^{-6} and 4.17×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.209.

The irradiation of medical, industrial, research and development, and neptunium-237 targets at FFTF would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of medical, industrial, or neptunium-237 targets at FFTF.

No chemical processing activities are currently performed at FMEF and no chemicals are stored in this facility. Processing activities in support of medical, industrial, research and development isotope and plutonium-238 production would require the introduction of hazardous chemicals, specifically nitric acid and nitric oxide. Potential health impacts from accidental releases of nitric acid were assessed by comparing estimated airborne concentrations of the chemicals to ERPG developed by the American Industrial Hygiene Association. The ERPG-1 value (0.5 parts per million) is the maximum airborne concentration below which nearly all individuals could be exposed for up to one hour, resulting in only mild, transient, and reversible adverse health effects. The ERPG-2 value (10 parts per million) is protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. The ERPG-3 value (25 parts per million) is indicative of potentially life-threatening health effects.

The maximum distances, in meters, needed to reach the ERPG values for nitric acid releases at the FMEF for Stability Classes D and F are shown in **Table 4–46**. Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions while Stability Class F represents worst-case meteorological conditions. The number of involved and noninvolved workers potentially exposed would vary with a number of factors such as the time of day and whether they are sheltered within buildings at the time of release. Individuals at the nearest highway (7,100 meters [4.4 miles]) and at the nearest site boundary (7,210 meters [4.5 miles]) from FMEF would be exposed to levels well below ERPG-1.

Table 4–46 ERPG Distances for Nitric Acid Releases at FMEF

Evaluation Parameter	Stability Class D (meters)	Stability Class F (meters)
ERPG-3	375	450
ERPG-2	500	600
ERPG-1	2,000	3,000

Note: To convert from meters to miles, multiply by 6.22×10^{-4}.
Key: ERPG, Emergency Response Planning Guideline.

There are no ERPG values for nitric oxide. For nitric oxide accidents, the level of concern has been estimated by using one-tenth of the "Immediately Dangerous to Life and Health" level published by the National Institute for Occupational Safety and Health. The Immediately Dangerous to Life and Health value for nitric oxide is 100 parts per million. The level of concern value used for this NI PEIS is 10 parts per million. The level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects as a result of a single exposure for a relatively short period of time.

For FMEF, the maximum distances needed to reach the level of concern for nitric oxide releases for Stability Classes D and F are 500 and 1,900 meters (1,640 and 6,560 feet), respectively. The number of involved and noninvolved workers potentially exposed would vary with a number of factors such as the time of day and whether they are sheltered within buildings at the time of release. Individuals at the nearest highway (7,100 meters [4.4 miles]) and at the nearest site boundary (7,210 meters [4.5 miles]) from FMEF would be exposed to levels well below the level of concern for nitric oxide.

Potential health impacts from the accidental release of the hazardous chemicals were assessed for a noninvolved worker, offsite individuals who are members of the public located at the nearest site boundary and onsite individuals who are members of the public located at the nearest highway access.

The impacts associated with the accidental release of nitric acid and nitric oxide at FMEF are presented in **Table 4–47**.

Table 4–47 FMEF Hazardous Chemical Accident Impacts Under Alternative 1 (Restart FFTF)—Option 3

Receptor	Evaluation Parameter	Nitric Acid		Nitric Oxide	
		Stability Class D	Stability Class F	Stability Class D	Stability Class F
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	3.3 <ERPG-2 Mild, transient	8.6 <ERPG-2 Mild, transient	4.2 <LOC Mild, transient	66 >LOC Serious
Nearest highway maximally exposed individual	Parts per million Level of concern Potential health effects	0.03 < ERPG-1 None	0.1 ERPG-1 Mild, transient	0.09 < LOC None	0.55 < LOC None
Site boundary maximally exposed individual	Parts per million Level of concern Potential health effects	0.03 < ERPG-1 None	0.1 ERPG-1 Mild, transient	0.09 < LOC None	0.53 < LOC None

Note: < means "less than."
Key: ERPG, Emergency Response Planning Guideline; LOC, level of concern.
Source: Model results.

4.3.3.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FMEF target fabrication facility at Hanford. DOE would transport the unirradiated neptunium-237 targets from FMEF to FFTF. Following irradiation in FFTF, the targets would be returned to FMEF for processing. After this processing, the plutonium-238 product would be shipped to LANL. FFTF would receive highly enriched uranium fuel from a U.S. fuel fabrication facility and mixed oxide fuel from Europe. Additionally, medical and industrial isotopes would be shipped from FFTF to a local airport, and from there to locations throughout the country.

Approximately 38,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 5.6 million kilometers (3.5 million miles); at sea by ships carrying mixed oxide fuel, 96,000 kilometers (52,000 nautical miles); and in the air by aircraft carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 18 person-rem; the dose to the public, 19 person-rem.

Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0072 latent cancer fatality among transportation workers and 0.009 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.023. About half of the crew risk, about 40 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of mixed oxide fuel to FFTF with a severity Category V accident in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.40 person-rem to the public with an associated 2.0×10^{-4} latent cancer fatality, and 3.3 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.7×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated), irradiated targets or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.12 traffic fatality. Nearly all of the radiological and traffic accident risk would result from shipping medical and industrial isotopes.

IMPACTS OF MARINE TRANSPORTATION. The potential impacts of marine transport of mixed oxide fuel on the global commons (i.e., portions of the ocean not within the territorial boundary of any nation) were evaluated in accordance with Executive Order 12114 (44 FR 1957). Following a hypothetical severe accident, radioactive particles dispersed over the ocean would not be in large enough amounts to have a measurable impact on the environment. The risks of accidents approaching and docking at the port have been estimated to be less than 1×10^{-9} person-rem, resulting in less than 1×10^{-12} latent cancer fatalities. The radiological doses associated with incident-free transportation, which include the exposure of the ship's crew to low levels of radiation during transport and handling of the packages, have been estimated to be approximately 0.03 person-rem for a route to an east coast port and 0.06 person-rem for a route to a west coast port. These doses would result in 1.2×10^{-5} and 2.4×10^{-5} latent cancer fatalities, respectively.

4.3.3.1.12 Environmental Justice

NORMAL OPERATIONS. The number of expected latent cancer fatalities among the population residing within 80 kilometers (50 miles) of FFTF and FMEF would be less than 0.003 for 35 years of normal operations (Table 4–41). As shown in Table 4–43, the release of hazardous chemicals at FFTF and FMEF would pose no significant risk of cancer or toxic effects among the public. As discussed in Section K.5.3, the expected latent cancer fatalities that would result from the ingestion of food that could be radiologically contaminated due to normal operations at FFTF and FMEF would be approximately 0.001. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near Hanford. As discussed in Section 4.3.1.1.11, incident-free transportation would not be expected to result in fatalities.

ACCIDENTS. The number of expected latent cancer fatalities among the populations at risk due to radiological accidents listed in Table 4–45 would be approximately 0.2. If a radiological accident were to occur at FFTF or FMEF at Hanford and northeasterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Yakama Indian Reservation (see Figure K–11). However, accidents that could occur under the implementation of this option would not be expected to result in a latent

cancer fatality among the population or maximally exposed individual residing within the boundary of the Yakama Indian Reservation.

The number of expected latent cancer fatalities resulting from transportation accidents with radiological emissions was found to be approximately 0.5. As discussed in Appendix J, this risk is driven by accidents that could occur during the air transportation of medical and industrial isotopes and the conservative assumptions used in the analysis of such accidents. Such accidents could occur anywhere along the flight paths and would not place any identifiable group within the general population at disproportionate risk. As discussed in Section 4.3.3.1.11 and Appendix J, expected fatalities due to a traffic collision would be approximately 0.1.

In summary, normal operations and accidents that could result from the implementation of this option would pose no significant radiological or nonradiological risks to the public, and implementation would pose no disproportionately high and adverse risks to any group within the population.

4.3.3.1.13 Waste Management

The expected generation rates of waste at Hanford that would be generated from the operation of FFTF for irradiating targets and with the operation of FMEF for target fabrication and processing are compared with Hanford's treatment, storage, and disposal capacities in **Table 4–48**. The impacts on the Hanford waste management systems, in terms of managing the additional waste, are discussed in this section. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.3.3.1.9 through 4.3.3.1.11.

Canisters used to transport neptunium-237 to the site would constitute a very small additional amount of solid low-level radioactive waste—less than 10 cubic meters (13.1 cubic yards) over the 35-year operational period, even if no credit is taken for volume reduction by compaction (Brunson 1999a). The annual generation of this waste would fall within the range of accuracy of the generation rate of solid low-level radioactive waste given in Table 4–48, and its management need not be addressed separately.

In accordance with the Records of Decision for the *Waste Management PEIS* (DOE 1997a), waste could be treated and disposed of on site at Hanford or at other DOE sites or commercial facilities. Based on the Record of Decision for high-level radioactive waste issued on August 12, 1999 (64 FR 46661), immobilized high-level radioactive waste would be stored on site until transfer to a geologic repository. Based on the Record of Decision for transuranic waste issued on January 20, 1998 (63 FR 3629), transuranic waste would be certified on site and eventually shipped to a suitable geologic repository for disposal. Based on the Record of Decision for hazardous waste issued on August 5, 1998 (63 FR 41810), nonwastewater hazardous waste would continue to be treated and disposed of at offsite commercial facilities. Based on the Record of Decision for low-level radioactive waste and mixed low-level radioactive waste issued on February 18, 2000 (65 FR 10061), minimal treatment of low-level radioactive waste will be performed at all sites and, to the extent practicable, onsite disposal of low-level radioactive waste will continue. Hanford and the Nevada Test Site will be made available to all DOE sites for the disposal of low-level radioactive waste. Mixed low-level radioactive waste analyzed in the *Waste Management PEIS* will be treated at Hanford, INEEL, ORR, and SRS and will be disposed of at Hanford and the Nevada Test Site.

The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. Irrespective of how the waste is classified (i.e., transuranic or high-level radioactive waste), the composition and characteristics are the same, and the waste management activities (i.e., treatment and onsite storage) as described in this NI PEIS would

Table 4–48 Incremental Waste Management Impacts of Operating FFTF and FMEF at Hanford Under Alternative 1 (Restart FFTF)—Option 3

Waste Type[a]	Estimated Additional Waste Generation for FFTF (cubic meters per year)	Estimated Total Waste Generation for FFTF Operation[b] (cubic meters per year)	Estimated Additional Waste Generation for FMEF (cubic meters per year)	Estimated Additional Waste Generation (both FFTF and FMEF) as a Percent of[c]		
				Onsite Treatment Capacity	Onsite Storage Capacity	Onsite Disposal Capacity
Transuranic/High-level radioactive[d]	0	0	11	(d)	(d)	NA
Low-level radioactive						
Liquid	0	<6	6	(e)	(e)	(e)
Solid	63	80	74	NA	NA	0.28
Mixed low-level radioactive	0	<0.5	9	NA	1.9	2.2
Hazardous	0	4	19	NA	NA	NA
Nonhazardous						
Process wastewater	22,000	98,000	15,000	(e)	(e)	(e)
Sanitary wastewater	1,900	5,700	3,800	2.4[f]	NA	NA
Solid	130	250	170	NA	NA	NA

a. See definitions in Section G.9.
b. These estimates represent the sum of the standby waste generation amounts provided for the No Action Alternative (Table 4–6) and the additional waste generation amounts given in the first column of this table (Table 4–48).
c. The estimated additional amounts of waste generated annually are compared with the annual site treatment capacities. The estimated total amounts of additional waste generated over the assumed 35-year operational period are compared with the site storage and disposal capacities.
d. Refer to the text for a discussion on waste classification and treatment. This waste would be stored at FMEF pending availability of a suitable repository. It is assumed that this waste would be remotely handled.
e. Refer to the text.
f. Percent of capacity of the Energy Northwest Sewage Treatment Facility.

Note: To convert from cubic meters per year to cubic yards per year, multiply by 1.308; < means "less than."
Key: NA, not applicable (i.e., the majority of this waste is not routinely treated, stored, or disposed of on site).
Source: Chapin 2000; DOE 2000a; Nielsen 1999.

be the same. In addition, either waste type would require disposal in a suitable repository. If it is transuranic waste, it would be nondefense waste and could not be disposed of at WIPP under current law. Because nondefense transuranic waste has no current disposal path, DOE Headquarters' approval would be necessary before a decision were made to generate such waste, as required by DOE Order 435.1. If the waste is classified as high-level radioactive waste, it is assumed for the purposes of this analysis that Yucca Mountain, Nevada, if approved, would be the final disposal site for DOE's high-level radioactive waste. The other differences between these two waste classifications are that a high-level radioactive waste repository requires a much more rigorous waste-form qualification process than a transuranic waste repository and there is a slightly different set of requirements for high-level radioactive waste than for transuranic waste delineated in DOE Manual 435.1.

Target fabrication and processing in FMEF would generate a total of 385 cubic meters (504 cubic yards) of transuranic or high-level radioactive waste over the 35-year operational period. As described in Section 3.4.5 of the *Preconceptual Design Planning for Chemical Processing to Support Pu-238 Production* (Wham 1998), the waste would be vitrified into a glass matrix at a glass melter installed within FMEF. The resulting glass matrix would be stored at FMEF pending availability of a repository for permanent disposal. The impacts of managing the additional quantities of this waste at Hanford would be minimal.

No high-level radioactive or transuranic waste would be generated from merely operating FFTF. The waste described above would result from processing targets that had been irradiated in FFTF.

Solid low-level radioactive waste generated from target irradiation at FFTF and target fabrication and processing in FMEF would be packaged in appropriate containers or burial casks, certified, and transferred for additional treatment and disposal in the existing onsite low-level radioactive Burial Grounds.

An additional 2,200 cubic meters (2,900 cubic yards) of solid low-level radioactive waste would be generated over the 35-year operational period as a result of target irradiation at FFTF as compared to the current standby mode for FFTF. Target fabrication and processing at FMEF would generate about 2,600 cubic meters (3,400 cubic yards) of solid low-level radioactive waste over the 35-year operational period. The total amount of additional solid low-level radioactive waste resulting from operations at FFTF and FMEF represents approximately 0.28 percent of the 1.74-million-cubic-meter (2.28-million-cubic-yard) capacity of the low-level radioactive Burial Grounds. Using the 3,480-cubic-meter-per-hectare (1,842-cubic-yard-per-acre) disposal land usage factor for Hanford published in the *Storage and Disposition PEIS* (DOE 1996a:E-9), 4,800 cubic meters (6,300 cubic yards) of waste would require 1.4 hectares (3.5 acres) of disposal space at Hanford. The impacts of managing this additional low-level radioactive waste at Hanford would be minimal.

Liquid low-level radioactive waste generated from target irradiation at FFTF and target fabrication and processing in FMEF would be transported to the 200 Area Effluent Treatment Facility for processing and ultimate disposal.

There would be no increase in liquid low-level radioactive waste generation as a result of target irradiation at FFTF as compared to the current standby mode for FFTF. Target fabrication and processing at FMEF would generate about 210 cubic meters (270 cubic yards) of liquid low-level radioactive waste over the 35-year operational period. This total amount of additional liquid low-level radioactive waste resulting from operations at FFTF and FMEF represents a small amount of waste that can be managed by the 200 Area Liquid Effluent Treatment Facility, which has an operating capacity of 0.57 cubic meter (0.75 cubic yard) per minute.

Mixed low-level radioactive waste would be stabilized, packaged, and stored on site for treatment and disposal in a manner consistent with the Tri-Party Agreement (EPA et al. 1989) for Hanford. Over the 35-year operational period, no additional mixed low-level radioactive waste would be generated as a result of target irradiation at FFTF as compared to the current standby mode. Mixed low-level radioactive waste generated at FMEF that is associated with target fabrication and processing is estimated over the 35-year operation period to be about 320 cubic meters (420 cubic yards). This mixed low-level radioactive waste is expected to be treated at a nearby commercial facility. This additional waste is also estimated to be about 1.9 percent of the 16,800-cubic-meter (22,000-cubic-yard) storage capacity of the Central Waste Complex and about 2.2 percent of the 14,200-cubic-meter (18,600-cubic-yard) planned disposal capacity of the Radioactive Mixed Waste Disposal Facility. Therefore, this additional waste would only have a minimal impact on the management of mixed low-level radioactive waste at Hanford.

Hazardous waste generated during operation would be packaged in DOT-approved containers and shipped off site to permitted commercial recycling, treatment, and disposal facilities. The additional waste load generated during the 35-year operational period would have only a minimal impact on the Hanford hazardous waste management system.

Nonhazardous solid waste would be packaged and transported in conformance with standard industrial practice. Solid waste such as office paper, metal cans, and plastic and glass bottles that can be recycled would be sent off site for that purpose. The remaining solid sanitary waste would be sent for offsite disposal. This

additional waste load would have only a minimal impact on the nonhazardous solid waste management system at Hanford.

Nonhazardous process wastewater would be discharged into the 400 Area Ponds. This discharge is regulated by State Waste Discharge Permit ST-4501.

Nonhazardous sanitary wastewater would be discharged to the 400 Area sanitary sewer system, which connects to the Energy Northwest Sewage Treatment Facility. Nonhazardous sanitary wastewater generated at FFTF from target irradiation and at FMEF from target fabrication and processing would represent 2.4 percent of the 235,000-cubic-meter-per-year (307,000-cubic-yard-per-year) capacity of the Energy Northwest Sewage Treatment Facility.

The generation rates of waste at Hanford that would be associated with this option (refer to Table 4–48) can be compared with the current waste generation rates at the site, given in Table 3–34 (Section 3.4.11). The waste generation rates associated with this alternative would be much smaller than the current waste generation rates at the site.

4.3.3.1.14 Spent Nuclear Fuel Management

Impacts associated with spent nuclear fuel management would be the same as for Option 1 and are given in Section 4.3.1.1.14.

4.3.4 Alternative 1 (Restart FFTF)—Option 4

Option 4 involves operating FFTF at Hanford to irradiate all targets and materials associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating REDC at ORR to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and operating facilities in RPL/306–E to fabricate and process the other targets and materials and to process the associated products. This option includes storage in REDC of the neptunium-237 transported from SRS to ORR and storage in RPL/306–E of the other target materials transported from other offsite facilities to Hanford.

The transportation of the highly enriched uranium fuel to Hanford for use in FFTF, the transportation of the neptunium-237 to ORR and then to Hanford, the transportation of the other target material to Hanford, and the transportation of the product materials following irradiation and postirradiation processing are also part of this option.

FFTF would operate with a mixed oxide fuel core for the first 6 years and with a highly enriched uranium fuel core for the next 29 years.

4.3.4.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.3.4.1.1 Land Resources

LAND USE. The restart of FFTF would not result in impacts on land use at Hanford for the reasons described in Section 4.3.1.1.1.

Neptunium-237 target fabrication and processing at REDC would not result in impacts on land use at ORR for the reasons described in Section 4.3.1.1.1.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on land use at Hanford for the reasons described in Section 4.3.1.1.1.

VISUAL RESOURCES. The restart of FFTF would not result in impacts on visual resources at Hanford for the reasons described in Section 4.3.1.1.1.

Impacts on visual resources would not occur at ORR for the reasons described in Section 4.3.1.1.1.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on visual resources at Hanford for the reasons described in Section 4.3.1.1.1.

4.3.4.1.2 Noise

For the restart of FFTF, the change in noise impacts from construction and operation would be expected to be small as described in Section 4.3.1.1.2.

Noise impacts from neptunium-237 target fabrication and processing at the REDC at ORNL would be expected to be small as described in Section 4.3.1.1.2.

Noise impacts from research and development support and medical and industrial isotope target fabrication and processing at RPL/306–E at Hanford would be expected to be small as described in Section 4.3.1.1.2.

4.3.4.1.3 Air Quality

Air quality impacts would be the same as under Option 1 (Section 4.3.1.1.3).

4.3.4.1.4 Water Resources

Impacts on water resources at Hanford associated with the restart of FFTF would be the same as those described in Section 4.3.1.1.4.

REDC in the 7900 Area of ORNL would be used for neptunium-237 storage, target fabrication, and processing in support of plutonium-238 production with impacts on ORR water resources the same as those described in Section 4.3.1.1.4.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. Impacts on water resources at Hanford from use of RPL/306–E would be the same as those described in Section 4.3.1.1.4.

4.3.4.1.5 Geology and Soils

The restart of FFTF would not be expected to result in impacts on geology and soils at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.1.2.5 and 4.3.1.1.5.

Neptunium-237 target fabrication and processing at REDC would not likely result in impacts on geology and soils at ORR, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.2.2.5 and 4.3.1.1.5.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not be expected to result in impacts on geology or soils at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.1.2.5 and 4.3.1.1.5. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.3.4.1.6 Ecological Resources

The restart of FFTF would not result in impacts on ecological resources at Hanford for the reasons described in Section 4.3.1.1.6.

Impacts on ecological resources would not occur at ORR for the reasons described in Section 4.3.1.1.6.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on ecological resources at Hanford for the reasons described in Section 4.3.1.1.6.

4.3.4.1.7 Cultural and Paleontological Resources

The restart of FFTF would not result in impacts on cultural resources at Hanford for the reasons described in Section 4.3.1.1.7.

Impacts on cultural resources would not occur at ORR for the reasons described in Section 4.3.1.1.7.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on cultural resources at Hanford for the reasons described in Section 4.3.1.1.7.

4.3.4.1.8 Socioeconomics

Impacts associated with this option would be the same as those addressed in Section 4.3.1.1.8.

4.3.4.1.9 Public and Occupational Health and Safety—Normal Operations

Impacts associated with this option would be the same as those presented in Section 4.3.1.1.9.

4.3.4.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with FFTF target irradiation, REDC neptunium-237 target processing, and RPL medical and industrial isotope processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the

accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are presented in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, the summation of the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–49** and **4–50**, respectively.

FFTF would operate for 6 years with a mixed oxide core followed by 29 years with a highly enriched uranium core. As shown in Table 4–49, the beyond-design-basis core melt accident would result in the largest radiological consequences among FFTF accidents. In order to incorporate internal and external initiators, the accident frequency of 1×10^{-6} was selected for the beyond-design-basis core melt accident. For 35 years of operation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 1.06×10^{-8} and 9.37×10^{-9}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00122.

For 35 years of REDC neptunium-237 target processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.157.

For 35 years of RPL medical, industrial, and research and development target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 4.51×10^{-4} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.377.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 4.51×10^{-4} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.535.

The consequences associated with chemical accidents would be the same as for Option 1 (Section 4.3.1.1.10).

4.3.4.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to FFTF. Following irradiation in FFTF, the targets would be returned to REDC for processing. After this processing, the plutonium-238 product would be shipped to LANL. FFTF would receive highly enriched uranium fuel from a U.S. fuel fabrication facility. Additionally, medical and industrial isotopes would be shipped from FFTF to a local airport, and from there to locations throughout the country.

Table 4–49 FFTF, REDC, and RPL Accident Consequences Under Alternative 1 (Restart FFTF)—Option 4

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
FFTF accidents						
Design-basis-accident primary sodium spill (MOX)	0.00113	5.65×10^{-7}	78.6	0.0393	0.00313	1.25×10^{-6}
Design-basis-accident primary sodium spill (HEU)	8.63×10^{-4}	4.32×10^{-7}	72.6	0.0363	0.00181	7.24×10^{-7}
Beyond-design-basis core melt accident (MOX)	0.679	3.40×10^{-4}	6.68×10^{4}	33.4	0.679	2.72×10^{-4}
Beyond-design-basis core melt accident (HEU)	0.481	2.41×10^{-4}	6.16×10^{4}	30.8	0.375	1.50×10^{-4}
BLTC driver fuel-handling accident (MOX)	0.00383	1.92×10^{-6}	1,280	0.639	0.357	1.43×10^{-4}
BLTC driver fuel-handling accident (HEU)	0.00384	1.92×10^{-6}	1,230	0.617	0.340	1.36×10^{-4}
BLTC neptunium-237 target-handling accident	2.61×10^{-4}	1.31×10^{-7}	25.8	0.0129	0.0279	1.12×10^{-5}
BLTC isotope target-handling accident	1.22×10^{-4}	6.10×10^{-8}	2.74	0.00137	0.0143	5.72×10^{-6}
REDC accidents						
Ion exchange explosion during neptunium-237 target fabrication	6.13×10^{-9}	3.06×10^{-12}	8.58×10^{-5}	4.29×10^{-8}	5.60×10^{-10}	2.24×10^{-13}
Target dissolver tank failure during plutonium-238 separation	1.76×10^{-7}	8.79×10^{-11}	0.00196	9.82×10^{-7}	1.69×10^{-8}	6.74×10^{-12}
Ion exchange explosion during plutonium-238 separation	4.68×10^{-4}	2.34×10^{-7}	5.23	0.00261	4.49×10^{-5}	1.79×10^{-8}
Processing facility beyond-design-basis earthquake	163	0.163	8.91×10^{5}	445	1,310	1.00[c]
RPL accidents						
Medical and industrial isotopes localized solvent fire	0.0135	6.74×10^{-6}	77.8	0.0389	0.0047	1.88×10^{-6}
Medical/industrial isotopes unlikely seismic event	1.52	7.60×10^{-4}	1,350	0.675	1.50	6.00×10^{-4}
Medical and industrial isotopes glovebox explosion	50.0	0.050	4.60×10^{4}	23.0	49.0	0.0392

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–50 FFTF, REDC, and RPL Accident Risks Under Alternative 1 (Restart FFTF)—Option 4

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual FFTF risks			
Design-basis-accident primary sodium spill (MOX) (1×10^{-4})	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis-accident primary sodium spill (HEU) (1×10^{-4})	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX) (1×10^{-6})	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU) (1×10^{-6})	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX) (1×10^{-7})	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU) (1×10^{-7})	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident (1×10^{-7})	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident (1×10^{-7})	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
35-year FFTF risk	1.06×10^{-8}	0.00122	9.37×10^{-9}
Annual REDC risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	1.63×10^{-6}	0.00445	1.00×10^{-5}(c)
35-year REDC risk	5.71×10^{-5}	0.157	3.50×10^{-4}
Annual RPL risks			
Medical and industrial isotopes localized solvent fire (0.044)	2.99×10^{-7}	0.00173	8.35×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	7.60×10^{-6}	0.00675	6.00×10^{-6}
Medical and industrial isotopes glovebox explosion (1×10^{-4})	5.00×10^{-6}	0.00230	3.92×10^{-6}
35-year RPL risk	4.51×10^{-4}	0.377	3.50×10^{-4}
35-year Option risk[d]	4.51×10^{-4}	0.535	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.
d. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Approximately 38,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 7.9 million kilometers (4.9 million miles); and in the air carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 31 person-rem; the dose to the public, 298 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.012 latent cancer fatality among transportation workers and 0.15 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.029. About half of the crew risk, about 2 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to REDC with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.18 traffic fatality. Nearly all of the radiological and traffic accident risk would result from shipping medical and industrial isotopes.

4.3.4.1.12 Environmental Justice

Environmental effects that would result from the implementation of Option 4 are nearly identical to those that would result from the implementation of Option 1 (Section 4.3.1.1.12). No disproportionately high and adverse radiological or nonradiological risks to minority or low-income populations would be expected to result from the implementation of Option 4.

4.3.4.1.13 Waste Management

The impacts of managing waste associated with irradiating targets in FFTF, with processing and fabricating target materials for research and development support and medical and industrial isotope production in RPL/306–E, and with fabricating and processing neptunium-237 targets for plutonium-238 production in REDC at ORR are all assumed to be the same as for Option 1 (Section 4.3.1.1.13). This is because the waste generation would not be affected by the type of fuel used (i.e., mixed oxide or highly enriched uranium), and the same amount of plutonium-238 production, medical and industrial isotope production, and civilian nuclear energy research and development support would be accomplished annually. As discussed in that section, the impacts on Hanford and ORR's waste management systems would be minimal.

4.3.4.1.14 Spent Nuclear Fuel Management

Impacts associated with spent nuclear fuel management would be the same as for Option 1 and are given in Section 4.3.1.1.14.

4.3.5 Alternative 1 (Restart FFTF)—Option 5

Option 5 involves operating FFTF at Hanford to irradiate all targets and materials associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating FDPF at INEEL to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and RPL/306–E in the Hanford 300 Area to fabricate and process the other targets and materials and to process the associated products. This option includes storage in Building CPP–651 or FDPF of the neptunium-237 transported to INEEL from SRS and storage in RPL/306–E of the other target materials transported to Hanford from other offsite facilities.

The transportation of the highly enriched uranium to Hanford for use in FFTF, the transportation of the neptunium-237 to INEEL and then to Hanford, the transportation of the other target material to Hanford, and the transportation of the product materials following irradiation and postirradiation processing are also part of this option.

FFTF would operate with a mixed oxide fuel core for the first 6 years and with a highly enriched uranium fuel core for the next 29 years.

4.3.5.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.3.5.1.1 Land Resources

LAND USE. The restart of FFTF would not result in impacts on land use at Hanford for the reasons described in Section 4.3.1.1.1.

Neptunium-237 storage in Building CPP–651 or FDPF and target fabrication and processing in FDPF would not result in impacts on land use at INEEL for the reasons described in Section 4.3.2.1.1.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on land use at Hanford for the reasons described in Section 4.3.1.1.1.

VISUAL RESOURCES. The restart of FFTF would not result in impacts on visual resources at Hanford for the reasons described in Section 4.3.1.1.1.

Impacts on visual resources would not occur at INEEL for the reasons described in Section 4.3.2.1.1.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on visual resources at Hanford for the reasons described in Section 4.3.1.1.1.

4.3.5.1.2 Noise

For the restart of FFTF, the change in noise impacts from construction and operation would be expected to be small as described in Section 4.3.1.1.2.

Noise impacts from neptunium-237 target fabrication and processing at Building CPP–651 and/or FDPF at INEEL would be expected to be small as described in Section 4.3.2.1.2.

Noise impacts from research and development support and medical and industrial isotope target fabrication and processing at RPL/306–E at Hanford would be expected to be small as described in Section 4.3.1.1.2.

4.3.5.1.3 Air Quality

Air quality impacts would be the same as under Option 2 (Section 4.3.2.1.3).

4.3.5.1.4 Water Resources

Impacts on water resources at Hanford associated with the restart of FFTF would be the same as those described in Section 4.3.1.1.4.

Building CPP–651 and/or FDPF in the INTEC area of INEEL would be used for neptunium-237 storage, with target fabrication and processing in support of plutonium-238 production in FDPF. Impacts on water resources at INEEL would be the same as those described in Section 4.3.2.1.4.

RPL/306–E in the 300 Area of Hanford would be used for the fabrication and processing of targets associated with the medical and industrial isotope production and civilian nuclear energy research and development missions. Impacts on water resources at Hanford would be the same as those described in Section 4.3.1.1.4.

4.3.5.1.5 Geology and Soils

The restart of FFTF would not be expected to result in impacts on geology and soils at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.1.2.5 and 4.3.1.1.5.

Neptunium-237 storage at Building CPP–651 and/or FDPF and target fabrication, and processing in FDPF would not likely result in impacts on geology and soils at INEEL, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.3.2.5 and 4.3.2.1.5.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not be expected to result in impacts on geology or soils at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.1.2.5 and 4.3.1.1.5. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.3.5.1.6 Ecological Resources

The restart of FFTF would not result in impacts on ecological resources at Hanford for the reasons described in Section 4.3.1.1.6.

Impacts on ecological resources would not occur at INEEL for the reasons described in Section 4.3.2.1.6.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on ecological resources at Hanford for the reasons described in Section 4.3.1.1.6.

4.3.5.1.7 Cultural and Paleontological Resources

The restart of FFTF would not result in impacts on cultural resources at Hanford for the reasons described in Section 4.3.1.1.7.

Impacts on cultural resources would not occur at INEEL for the reasons described in Section 4.3.2.1.7.

Using RPL/306–E for research and development support and medical and industrial isotope target fabrication and processing would not result in impacts on cultural resources at Hanford for the reasons described in Section 4.3.1.1.7.

4.3.5.1.8 Socioeconomics

Impacts associated with this option would be the same as those addressed in Section 4.3.2.1.8.

4.3.5.1.9 Public and Occupational Health and Safety—Normal Operations

Impacts associated with this option would be the same as those presented in Section 4.3.2.1.9.

4.3.5.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with FFTF target irradiation, FDPF neptunium-237 target processing, and RPL medical and industrial target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 miles) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are presented in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, the summation of the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–51** and **4–52**, respectively.

Table 4–51 FFTF, FDPF, and RPL Accident Consequences Under Alternative 1 (Restart FFTF)—Option 5

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
FFTF accidents						
Design-basis-accident primary sodium spill (MOX)	0.00113	5.65×10^{-7}	78.6	0.0393	0.00313	1.25×10^{-6}
Design-basis-accident primary sodium spill (HEU)	8.63×10^{-4}	4.32×10^{-7}	72.6	0.0363	0.00181	7.24×10^{-7}
Beyond-design-basis core melt accident (MOX)	0.679	3.40×10^{-4}	6.68×10^{4}	33.4	0.679	2.72×10^{-4}
Beyond-design-basis core melt accident (HEU)	0.481	2.41×10^{-4}	6.16×10^{4}	30.8	0.375	1.50×10^{-4}
BLTC driver fuel-handling accident (MOX)	0.00383	1.92×10^{-6}	1,280	0.639	0.357	1.43×10^{-4}
BLTC driver fuel-handling accident (HEU)	0.00384	1.92×10^{-6}	1,230	0.617	0.340	1.36×10^{-4}
BLTC neptunium-237 target-handling accident	2.61×10^{-4}	1.31×10^{-7}	25.8	0.0129	0.0279	1.12×10^{-5}
BLTC isotope target-handling accident	1.22×10^{-4}	6.10×10^{-8}	2.74	0.00137	0.0143	5.72×10^{-6}
FDPF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.01×10^{-9}	1.01×10^{-12}	2.49×10^{-5}	1.24×10^{-8}	7.26×10^{-9}	2.91×10^{-12}
Target dissolver tank failure during plutonium-238 separation	6.11×10^{-8}	3.05×10^{-11}	5.65×10^{-4}	2.82×10^{-7}	2.17×10^{-7}	8.69×10^{-11}
Ion exchange explosion during plutonium-238 separation	1.63×10^{-5}	8.13×10^{-9}	0.150	7.51×10^{-5}	5.79×10^{-5}	2.31×10^{-8}
Processing facility beyond-design-basis earthquake	42.5	0.0425	1.64×10^{5}	82.0	1,200	1.0[c]
RPL accidents						
Medical and industrial isotopes localized solvent fire	0.0135	6.74×10^{-6}	77.8	0.0389	0.0047	1.88×10^{-6}
Medical and industrial isotopes unlikely seismic event	1.52	7.60×10^{-4}	1,350	0.675	1.50	6.00×10^{-4}
Medical and industrial isotopes glovebox explosion	50.0	0.050	4.60×10^{4}	23.0	49.0	0.0392

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–52 FFTF, FDPF, and RPL Accident Risks Under Alternative 1 (Restart FFTF)—Option 5

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual FFTF risks			
Design-basis-accident primary sodium spill (MOX) (1×10^{-4})	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis-accident primary sodium spill (HEU) (1×10^{-4})	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX) (1×10^{-6})	3.40×10^{-10}	$3,34\times10^{-5}$	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU) (1×10^{-6})	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX) (1×10^{-7})	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU) (1×10^{-7})	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident (1×10^{-7})	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident (1×10^{-7})	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
35-year FFTF risk	1.06×10^{-8}	0.00122	9.37×10^{-9}
Annual FDPF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	4.25×10^{-7}	8.20×10^{-4}	$1.00\times10^{-5(c)}$
35-year FDPF risk	1.49×10^{-5}	0.0287	3.50×10^{-4}
Annual RPL risks			
Medical and industrial isotopes localized solvent fire (0.044)	2.99×10^{-7}	0.00173	8.35×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	7.60×10^{-6}	0.00675	6.00×10^{-6}
Medical and industrial isotopes glovebox explosion (1×10^{-6})	5.00×10^{-6}	0.00230	3.92×10^{-6}
35-year RPL risk	4.51×10^{-4}	0.377	3.50×10^{-4}
35-year Option risk[d]	4.51×10^{-4}	0.407	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.
d. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

FFTF would operate for 6 years with a mixed oxide core followed by 29 years with a highly enriched uranium core. As shown in Table 4–51, the beyond-design-basis core melt accident would result in the largest radiological consequences among FFTF accidents. In order to incorporate internal and external initiators, the accident frequency of 1×10^{-6} was selected for the beyond-design-basis core melt accident. For 35 years of operation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 1.06×10^{-8} and 9.37×10^{-9}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00122.

For 35 years of FDPF neptunium-237 target processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 1.49×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0287.

For 35 years of RPL medical, industrial, and research and development target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 4.51×10^{-4} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.377.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 4.51×10^{-4} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.407.

The consequences associated with chemical accidents would be the same as for Option 2 (Section 4.3.2.1.10).

4.3.5.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FDPF target fabrication facility at INEEL. DOE would transport the unirradiated neptunium-237 targets from FDPF to FFTF. Following irradiation in FFTF, the targets would be returned to FDPF for processing. After this processing, the plutonium-238 product would be shipped to LANL. FFTF would receive highly enriched uranium fuel from a U.S. fuel fabrication facility. Additionally, medical and industrial isotopes would be shipped from FFTF to a local airport, and from there to locations throughout the country.

Approximately 38,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 6.1 million kilometers (3.8 million miles); and in the air carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 21 person-rem; the dose to the public, 88 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.008 latent cancer fatality among transportation workers and 0.044 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.023. About half of the crew risk, about 8 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated

neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.13 traffic fatality. Nearly all of the radiological and traffic accident risk would result from shipping medical and industrial isotopes.

4.3.5.1.12 Environmental Justice

Environmental effects that would result from the implementation of Option 5 are nearly identical to those that would result from the implementation of Option 2 (Section 4.3.2.1.12). No disproportionately high and adverse radiological or nonradiological risks to minority or low-income populations would be expected to result from the implementation of Option 5.

4.3.5.1.13 Waste Management

The impacts of managing waste associated with irradiating targets in FFTF, with processing and fabricating target materials for the research and development support and medical and industrial isotope production in RPL/306–E, and with fabricating and processing neptunium-237 targets for plutonium-238 production in FDPF, are all assumed to be the same as for Option 2 (Section 4.3.2.1.13). This is because the waste generation would not be affected by the type of fuel used (i.e., mixed oxide or highly enriched uranium) and the same amount of plutonium-238 production, medical and industrial isotope production, and civilian nuclear energy research and development support would be accomplished annually. As discussed in that section, the impacts on Hanford's and INEEL's waste management systems would be minimal.

4.3.5.1.14 Spent Nuclear Fuel Management

Impacts associated with spent nuclear fuel management would be the same as for Option 1 and are given in Section 4.3.1.1.14.

4.3.6 Alternative 1 (Restart FFTF)—Option 6

Option 6 involves operating FFTF at Hanford to irradiate all targets and materials associated with plutonium-238 production, medical and industrial isotope production, and research and development, and also operating FMEF at Hanford to fabricate and process these targets and materials and the associated irradiated products. This option includes storage in FMEF of the neptunium-237 transported to Hanford from SRS and of the other target materials transported to Hanford from other offsite facilities.

The transportation of the highly enriched uranium fuel to Hanford for use in FFTF, the transportation of the neptunium-237 and other target material to Hanford, and the transportation of the product materials following postirradiation processing are also part of this option.

FFTF would operate with a mixed oxide fuel core for the first 6 years and with a highly enriched uranium fuel core for the next 29 years.

4.3.6.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.3.6.1.1 Land Resources

LAND USE. The restart of FFTF would not result in impacts on land use at Hanford for the reasons described in Section 4.3.3.1.1.

Impacts on land use at Hanford from target material storage, target fabrication, and processing at FMEF would be minimal for the reasons described in Section 4.3.3.1.1.

VISUAL RESOURCES. The restart of FFTF would not result in impacts on visual resources at Hanford for the reasons described in Section 4.3.3.1.1.

Impacts on visual resources at Hanford from target material storage, target fabrication, and processing at FMEF would be minimal for the reasons described in Section 4.3.3.1.1.

4.3.6.1.2 Noise

For the restart of FFTF, the change in noise impacts from construction and operation would be expected to be small as described in Section 4.3.1.1.2.

Noise impacts from target material storage, target fabrication, and processing at the FMEF would be expected to be small as described in Section 4.3.3.1.2.

4.3.6.1.3 Air Quality

Air quality impacts would be the same as under Option 3 (Section 4.3.3.1.3).

4.3.6.1.4 Water Resources

Impacts on water resources at Hanford associated with the restart of FFTF and the operation of FMEF for target material storage, target fabrication, and processing would be the same as those described in Section 4.3.3.1.4.

4.3.6.1.5 Geology and Soils

The restart of FFTF would not be expected to result in impacts on geologic and soil resources at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.1.2.5 and 4.3.1.1.5.

Impacts on geologic resources and soils at Hanford from the operation of FMEF for target material storage, target fabrication, and processing would not be expected for the reasons described in Section 4.3.3.1.5. Likewise, large-scale geologic conditions would not be expected to jeopardize FMEF. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.3.6.1.6 Ecological Resources

The restart of FFTF would not result in impacts on ecological resources at Hanford for the reasons described in Section 4.3.1.1.6.

Impacts on ecological resources at Hanford from target material storage, target fabrication, and processing at FMEF would not be expected for the reasons described in Section 4.3.3.1.6.

4.3.6.1.7 Cultural and Paleontological Resources

The restart of FFTF would not be expected to result in impacts on cultural resources at Hanford for the reasons described in Section 4.3.1.1.7.

Impacts on cultural resources at Hanford from target material storage, target fabrication, and processing at FMEF would not be expected for the reasons described in Section 4.3.3.1.7.

4.3.6.1.8 Socioeconomics

Impacts associated with this option would be the same as those presented in Section 4.3.3.1.8.

4.3.6.1.9 Public and Occupational Health and Safety—Normal Operations

Impacts associated with this option would be the same as those presented in Section 4.3.3.1.9.

4.3.6.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with FFTF target irradiation and FMEF target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 miles) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are presented in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, the summation of the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–53** and **4–54**, respectively.

Table 4–53 FFTF and FMEF Accident Consequences Under Alternative 1 (Restart FFTF)—Option 6

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
FFTF accidents						
Design-basis-accident primary sodium spill (MOX)	0.00113	5.65×10^{-7}	78.6	0.0393	0.00313	1.25×10^{-6}
Design-basis-accident primary sodium spill (HEU)	8.63×10^{-4}	4.32×10^{-7}	72.6	0.0363	0.00181	7.24×10^{-7}
Beyond-design-basis core melt accident (MOX)	0.679	3.40×10^{-4}	6.68×10^{4}	33.4	0.679	2.72×10^{-4}
Beyond-design-basis core melt accident (HEU)	0.481	2.41×10^{-4}	6.16×10^{4}	30.8	0.375	1.50×10^{-4}
BLTC driver fuel-handling accident (MOX)	0.00383	1.92×10^{-6}	1,280	0.639	0.357	1.43×10^{-4}
BLTC driver fuel-handling accident (HEU)	0.00384	1.92×10^{-6}	1,230	0.617	0.340	1.36×10^{-4}
BLTC neptunium-237 target-handling accident	2.61×10^{-4}	1.31×10^{-7}	25.8	0.0129	0.0279	1.12×10^{-5}
BLTC isotope target-handling accident	1.22×10^{-4}	6.10×10^{-8}	2.74	0.00137	0.0143	5.72×10^{-6}
FMEF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.02×10^{-9}	1.01×10^{-12}	7.26×10^{-5}	3.63×10^{-8}	6.65×10^{-10}	2.66×10^{-13}
Target dissolver tank failure during plutonium-238 separation	4.64×10^{-8}	2.32×10^{-11}	0.00169	8.47×10^{-7}	1.95×10^{-8}	7.81×10^{-12}
Ion exchange explosion during plutonium-238 separation	1.24×10^{-5}	6.18×10^{-9}	0.451	2.25×10^{-4}	5.20×10^{-6}	2.08×10^{-9}
Medical and industrial isotopes localized solvent fire	0.00276	1.38×10^{-6}	56.2	0.0281	9.51×10^{-5}	3.80×10^{-8}
Medical and industrial isotopes glovebox explosion	1.00	5.00×10^{-4}	2.95×10^{4}	14.8	24.0	0.0192
Processing facility beyond-design-basis earthquake	16.5	0.00825	6.42×10^{5}	321	922	1.00[c]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Key: BLTC, bottom-loading transfer casks; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–54 FFTF and FMEF Accident Risks Under Alternative 1 (Restart FFTF)—Option 6

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual FFTF risks			
Design-basis-accident primary sodium spill (MOX) (1×10^{-4})	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis-accident primary sodium spill (HEU) (1×10^{-4})	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX) (1×10^{-6})	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU) (1×10^{-6})	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX) (1×10^{-7})	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU) (1×10^{-7})	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident (1×10^{-7})	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident (1×10^{-7})	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
35-year FFTF risk	1.06×10^{-8}	0.00122	9.37×10^{-9}
Annual FMEF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Medical and industrial isotopes localized solvent fire (0.044)	6.13×10^{-8}	1.25×10^{-3}	1.69×10^{-9}
Medical and industrial isotopes glovebox explosion (1×10^{-4})	5.00×10^{-8}	0.00148	1.92×10^{-6}
Processing facility beyond-design-basis earthquake (1×10^{-5})	8.25×10^{-8}	0.00321	$1.00\times10^{-5(c)}$
35-year FMEF risk	6.79×10^{-6}	0.208	4.17×10^{-4}
35-year Option risks[d]	6.80×10^{-6}	0.209	4.17×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.
d. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium core; MOX, mixed oxide core.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

FFTF would operate for 6 years with a mixed oxide core followed by 29 years with a highly enriched uranium core. As shown in Table 4–53, the beyond-design-basis core melt accident would result in the largest radiological consequences among FFTF accidents. In order to incorporate internal and external initiators, the accident frequency of 1×10^{-6} was selected for the beyond-design-basis core melt accident. For 35 years of operation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 1.06×10^{-8} and 9.37×10^{-9}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00122.

For 35 years of FMEF target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 6.79×10^{-6} and 4.17×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.208.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 6.80×10^{-6} and 4.17×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.209.

The consequences associated with chemical accidents would be the same as for Option 3 (Section 4.3.3.1.10).

4.3.6.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FMEF target fabrication facility at Hanford. DOE would transport the unirradiated neptunium-237 targets from FMEF to FFTF. Following irradiation in FFTF, the targets would be returned to FMEF for processing. After this processing, the plutonium-238 product would be shipped to LANL. FFTF would receive highly enriched uranium fuel from a U.S. fuel fabrication facility. Additionally, medical and industrial isotopes would be shipped from FFTF to a local airport, and from there to locations throughout the country.

Approximately 38,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 5.5 million kilometers (3.4 million miles); and in the air carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 18 person-rem; the dose to the public, 18 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0071 latent cancer fatality among transportation workers and 0.009 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.023. About half of the crew risk, about 40 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) would not breach the transportation package. The probability of severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.11 traffic fatality. Nearly all of the radiological and traffic accident risk would result from shipping medical and industrial isotopes.

4.3.6.1.12 Environmental Justice

Environmental effects that would result from the implementation of Option 6 are nearly identical to those that would result from the implementation of Option 3 (Section 4.3.3.1.12). No disproportionately high and

adverse radiological or nonradiological risks to minority or low-income populations would be expected to result from the implementation of Option 6.

4.3.6.1.13 Waste Management

Impacts of managing waste associated with irradiating targets in FFTF, and with processing and fabricating target materials for the research and development support and medical and industrial isotope production and plutonium-238 production in FMEF, are assumed to be the same as for Option 3 (Section 4.3.3.1.13). This is because the waste generation would not be affected by the type of fuel used (i.e., mixed oxide or highly enriched uranium) and the same amount of plutonium-238 production, medical and industrial isotope production and civilian nuclear energy research and development support would be accomplished annually. As discussed in that section, the impacts on Hanford's waste management systems would be small.

4.3.6.1.14 Spent Nuclear Fuel Management

Impacts associated with spent nuclear fuel management would be the same as for Option 1 and are given in Section 4.3.1.1.14.

4.4 ALTERNATIVE 2—USE ONLY EXISTING OPERATIONAL FACILITIES

Under Alternative 2, DOE would use existing operating DOE reactors or U.S. commercial nuclear power plants to produce plutonium-238 for future space missions. The production of medical and industrial isotopes and support of civilian nuclear energy research and development in DOE reactors and accelerators would continue at the No Action Alternative levels. However, the currently operating DOE reactors, HFIR and ATR, cannot fully meet the projected long-term needs for medical isotope production and civilian nuclear energy research and development with or without adding the plutonium-238 production mission.

Depending on the combination of facilities used in Alternative 2, HFIR and ATR could continue their current support of the medical and industrial isotope and research and development missions, including some near-term growth, while accommodating the production of plutonium-238. Under other scenarios, some of the near-term growth in medical and industrial isotope production and civilian nuclear energy research and development, possible in these reactors, could be limited by the addition of the plutonium-238 production. In any case, non-DOE use of these facilities would be affected by the addition of the plutonium-238 mission. If a commercial reactor were used for plutonium-238 production, the DOE facilities would be unaffected and would continue operating as discussed under the No Action Alternative.

Another component of Alternative 2 is permanent deactivation of FFTF. Permanent deactivation of FFTF (Alternative 5) could occur in conjunction with any of the options under Alternatives 2, 3, or 4. Ongoing operations at existing facilities as described in Chapter 3, Affected Environment, would continue under Alternative 2.

Targets for plutonium-238 production would be fabricated in one of three facilities at ORNL, INEEL, or Hanford. The material needed for target fabrication (neptunium-237) would be processed and transported from SRS to the fabrication facilities. The targets would be irradiated at existing reactor facilities (HFIR, ATR, and a commercial light water reactor [CLWR] as described in Section 2.3.1) and would be transported back to the fabricating facilities for postirradiation processing.

Under Alternative 2, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for storage, target fabrication, target irradiation, and postirradiation processing, as well as transportation of the plutonium-238 product to LANL.

Nine options are proposed under this alternative. Options 1 through 3 involve the irradiation of targets in ATR at INEEL. Options 4 through 6 involve the irradiation of targets in a generic CLWR. Options 7 through 9 involve the irradiation of targets in both INEEL's ATR and ORNL's HFIR. These options and the associated target fabrication, postirradiation processing, and transportation activities are discussed below.

- **Option 1.** REDC at ORNL would be used to store the neptunium-237 transported from SRS to ORNL and to fabricate and process the targets irradiated at ATR. Option 1 also involves transportation of the neptunium-237 targets from ORNL to INEEL for irradiation in ATR, transportation of the irradiated targets from INEEL back to ORNL for postirradiation processing, and subsequent transportation of the plutonium-238 product from ORNL to LANL following postirradiation processing.

- **Option 2.** FDPF at INEEL would be used to store the neptunium transported from SRS to INEEL and to fabricate and process the targets (irradiated at ATR). Building CPP–651 would also be used for storage. Option 2 also involves transportation of the plutonium-238 product from INEEL to LANL following postirradiation processing.

- **Option 3.** FMEF at Hanford would be used to fabricate and process the targets (irradiated at ATR) and to store the neptunium-237 transported from SRS to Hanford. Option 3 also involves transportation of the neptunium-237 to Hanford for target fabrication, transportation of the targets from Hanford to INEEL for irradiation, transportation of the irradiated targets back to Hanford for postirradiation processing in FMEF, and subsequent transportation of the plutonium-238 product from Hanford to LANL.

- **Option 4.** REDC at ORNL would be used to store the neptunium-237 transported from SRS to ORNL and to fabricate and process the targets (irradiated at a generic CLWR). Option 4 also involves transportation of the neptunium-237 targets from ORNL to the generic CLWR location for irradiation, transportation of the irradiated targets back to ORNL for postirradiation processing, and transportation of the plutonium-238 product from ORNL to LANL.

- **Option 5.** FDPF at INEEL would be used to store the neptunium transported from SRS to INEEL and to fabricate and process the targets (irradiated at a generic CLWR). Building CPP–651 would also be used for storage. In addition, Option 5 involves transportation of the neptunium-237 targets from INEEL to the generic CLWR location for irradiation, transportation of the irradiated targets back to INEEL for postirradiation processing, and transportation of the plutonium-238 product from INEEL to LANL.

- **Option 6.** FMEF at Hanford would be used to store the neptunium-237 transported from SRS to Hanford and to fabricate and process the targets (irradiated at a generic CLWR). Option 6 also involves transportation of neptunium-237 to Hanford for target fabrication, transportation of the targets from Hanford to the generic CLWR location for irradiation, transportation of the irradiated targets back to Hanford for postirradiation processing, and transportation of the plutonium-238 product from Hanford to LANL.

- **Option 7.** REDC at ORNL would be used to store the neptunium-237 transported from SRS to ORNL and to fabricate and process the targets (irradiated at ATR and HFIR). Option 7 also involves transportation of the neptunium-237 targets from ORNL to the reactors for irradiation, transportation of the irradiated targets back to ORNL for processing, and transportation of the plutonium-238 product from ORNL to LANL.

- **Option 8.** FDPF at INEEL would be used to store the neptunium transported from SRS to INEEL and to fabricate and process the targets (irradiated at ATR and HFIR). Building CPP–651 would also be used for storage. Option 8 also involves transportation of the neptunium-237 targets from INEEL to the reactors for irradiation, transportation of the irradiated targets back to INEEL for postirradiation processing, and transportation of the plutonium-238 product from INEEL to LANL.

- **Option 9.** FMEF at Hanford would be used to store the neptunium-237 transported from SRS to Hanford and to fabricate and process the targets (irradiated at ATR and HFIR). Option 9 also involves transportation of neptunium-237 to Hanford for target fabrication, transportation of the targets from Hanford to the reactors for irradiation, transportation of the irradiated targets back to Hanford for postirradiation processing, and transportation of the plutonium-238 product from Hanford to LANL.

4.4.1 Alternative 2 (Use Only Existing Operational Facilities)—Option 1

Option 1 involves operating the Advanced Test Reactor (ATR) at INEEL to irradiate neptunium-237 targets to produce plutonium-238, and operating the REDC facility at ORR to both fabricate and process these targets and to store the neptunium-237 transported to ORR from SRS.

The transportation of the neptunium-237 from SRS to ORR for processing and fabrication into neptunium-237 targets in REDC, the transportation of these targets from ORR to INEEL for irradiation in ATR, the transportation of the irradiated targets from INEEL back to ORR for postirradiation processing in REDC, and the transportation of the plutonium-238 product from ORR to LANL also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.1.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.4.1.1.1 Land Resources

LAND USE. ATR is an operating facility in the Test Reactor Area at INEEL; use of the facility for neptunium-237 target irradiation would be compatible with its current mission. Further, because it is an existing facility, no new construction would be required, and thus, there would be no change in land use in the Test Reactor Area or INEEL.

REDC would be used for neptunium-237 storage, target fabrication, and processing. REDC is an existing operating facility in the 7900 Area of ORNL, and the use of this facility would require internal modifications, but no new facilities would be built. Because no additional land would be disturbed and the use of REDC for neptunium-237 target fabrication and processing would be compatible with its present mission, there would be no change in land use at ORR.

VISUAL RESOURCES. The irradiation of neptunium-237 targets would take place in the existing ATR at INEEL. The use of ATR would not require any external modifications that would alter the appearance of the facility. Therefore, the current Visual Resource Management Class IV rating for the Test Reactor Area would not change. Because there would be no change in the appearance of ATR or the Test Reactor Area, there would be no additional impact on visual resources.

All activities associated with neptunium-237 storage, target fabrication, and processing would take place in REDC at ORR. Because REDC is an existing facility that would require no external modifications, there would be no change in its appearance. Therefore, the current Visual Resource Management Class IV rating for the 7900 Area would not change, and there would be no impact on visual resources.

4.4.1.1.2 Noise

Noise associated with neptunium-237 target irradiation in ATR would be similar to sound levels generated by current reactor operations, as well as other operations in the Test Reactor Area. Onsite noise impacts would be expected to be minimal, and changes in offsite noise levels would not be noticeable because the nearest site boundary is 11 kilometers (6.8 miles) to the northwest. Noise levels associated with increased traffic going to and from the facility would be low, and would result in only minor changes to existing onsite and offsite noise levels. Neptunium-237 target irradiation in ATR would not produce any sudden loud noises that would adversely affect wildlife.

Noise associated with neptunium-237 storage, target fabrication, and processing would be similar to sound levels generated by present REDC operations, as well as other operations in the 7900 Area. Onsite noise impacts would be expected to be minimal, and changes in offsite noise levels would not be noticeable because the nearest site boundary is 2.5 kilometers (1.6 miles) to the southeast. Changes in traffic volume going to and

from REDC would be minor, and would not lead to noticeable changes in noise levels either on site or off site. There would be no loud noises associated with target fabrication and processing that would adversely impact wildlife.

4.4.1.1.3 Air Quality

It is estimated that there would be no measurable increases in nonradiological air pollutant emissions at INEEL associated with this option (Moor and Peterson 1999). The baseline air quality at INEEL would be unchanged.

The air pollutant concentrations at ORR attributable to REDC are presented in **Table 4–55**. The concentrations are based on a dispersion-modeling screening analysis conducted with maximum expected emission rates and a set of worst-case meteorological conditions. Only those air pollutants expected to be emitted that have ambient air quality standards are presented in the table. The changes in concentrations were determined to be small and would be below the applicable standard even when ambient monitored values and the contribution from other site activities were included. There are no Prevention of Significant Deterioration increment-consuming sources at ORR; therefore, a Prevention of Significant Deterioration increment consumption analysis was not conducted. Health effects from hazardous chemicals associated with this option are addressed in Section 4.4.1.1.9.

Table 4–55 Incremental ORR Concentrations[a] Associated with Alternative 2 (Use Only Existing Operational Facilities)—Option 1

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Nitrogen dioxide	Annual	100	1.99×10^{-4}
Sulfur dioxide	Annual	80	0.04
	24 hours	365	0.31
	3 hours	1,300	0.70

a. For comparison with ambient air quality standards.
Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995).

The air quality impacts of transportation among SRS, INEEL, ORR, and LANL are presented in Section 4.4.1.1.11.

4.4.1.1.4 Water Resources

The production of plutonium-238 would not measurably increase groundwater usage from the Snake River Plain aquifer or measurably affect the quantity or quality of effluents discharged from ATR (Moor and Peterson 1999:6). Information on current water usage, effluent discharge, and water quality for INEEL is presented in Section 3.3.4.

REDC, an existing facility in the 7900 Area of ORNL at ORR, would be used for neptunium-237 storage, target fabrication, and processing in support of plutonium-238 production with impacts on ORR water resources indicators the same as those described in Section 4.3.1.1.4. In summary, a small increase in water use and sanitary wastewater generation is anticipated, mainly attributable to increased staffing levels. Also, there would be a very small increase in process wastewater generation, but there would be no radiological liquid effluent discharge to the environment under normal operations.

4.4.1.1.5 Geology and Soils

ATR, an existing facility, would be used for the irradiation of neptunium-237 targets. Since no new construction is planned, there would be no disturbance to either geologic or soil resources in the Test Reactor Area. As previously summarized in Section 4.2.3.2.5, hazards from large-scale geologic conditions at INEEL, such as earthquakes and volcanoes, were evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-148). The analysis determined that these hazards present a low risk to INEEL facilities. That analysis was reviewed in the *Surplus Plutonium Disposition EIS* (DOE 1999a: 4-267-268). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to proposed ATR operations. This is because regional seismic conditions do not preclude the safe operation of properly or specially designed or upgraded facilities and the potential for future volcanic activity is low. The potential for nontectonic events to threaten INEEL facilities is also low.

Because the existing REDC facility would be used for neptunium-237 storage, target fabrication, and processing under this option, there would be no disturbance to either geologic or soil resources in the 7900 Area of ORNL. Hazards from large-scale geologic conditions at ORR were previously analyzed as discussed in Section 4.2.2.2.5 and determined to present a low risk to REDC.

As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.1.1.6 Ecological Resources

The existing ATR facility at INEEL would be used to irradiate neptunium-237 targets. Terrestrial resources would not be adversely affected because ATR is in the highly disturbed and fenced Test Reactor Area, and no new construction is planned. Further, as noted in Section 4.4.1.1.2, there would be no sudden loud noises that would adversely affect wildlife. Because there would be no measurable increase in water use or wastewater discharge, and discharge chemistry would not be expected to change, there would be no impact on aquatic habitat (Section 4.4.1.1.4). Due to the developed nature of the area, and because no new construction would take place, impacts on threatened and endangered species would not occur.

Consultation letters to comply with Section 7 of the Endangered Species Act were sent to the U.S. Fish and Wildlife Service and the Idaho Department of Fish and Game (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. The Idaho Department of Fish and Game indicated that its database contained no known occurrences of special status plants or animals near the project area. While DOE has made additional contact with the U.S. Fish and Wildlife Service, a response is pending from this agency. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at INEEL prior to the receipt of input from the Service.

REDC, an existing facility at ORR, would be used for neptunium-237 storage, target fabrication, and processing. No new construction would take place; thus, direct disturbance to ecological resources would not occur. As noted in Section 4.4.1.1.2, there would be no sudden loud noises that would adversely affect wildlife. There would be no change in impacts on aquatic resources because additional water usage and wastewater discharge would be small fractions of current values and discharge chemistry would not be expected to change (Section 4.4.1.1.4). Threatened and endangered species would not be impacted because an existing facility in the developed area would be used.

Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service (see Table 5–3) and resulted in the Service concluding that it does not anticipate adverse effects to federally listed endangered species that occur near the project area. DOE has also consulted with the Tennessee Department of Environment and Conservation; a response concerning state-listed species is pending from this agency. Although no state-listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the state.

4.4.1.1.7 Cultural and Paleontological Resources

The irradiation of neptunium-237 targets would take place in ATR. Because no new construction is planned, impacts on cultural and paleontological resources would not occur. The Materials Test Reactor, the Engineering Test Reactor, and ATR, as well as a number of support facilities, are potentially eligible for nomination to the National Register of Historic Places. The use of ATR would not affect the potential eligibility of these structures for listing.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). The State Historic Preservation Office indicated that ATR is likely to be eligible for the National Register of Historic Places as a contributory property in a potential historic district of exceptional significance. However, at this time, the State Historic Preservation Office has determined that more information is needed prior to assisting DOE in evaluating this property. The State Historic Preservation Office also indicated that since there would be no new construction, there is little potential for effects on archaeological properties. DOE would provide additional information as required to the Idaho State Historic Preservation Office prior to the use of any facility at INEEL for the proposed project. Consultation was conducted with interested Native American tribes; however, responses are pending.

Neptunium-237 storage, target fabrication, and processing would take place at the existing REDC facility in the 7900 Area of ORNL. Because no new construction would take place, impacts on cultural and paleontological resources would not occur. One structure within ORNL, the Graphite Reactor, is listed on the National Register of Historic Places as a National Historic Landmark. Additionally, several other structures proposed for listing on the National Register of Historic Places are found within or near ORNL. However, neither the Graphite Reactor nor any of the other structures is in the 7900 Area and, thus, their status would not change by the use of REDC for target fabrication and processing.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). While DOE has made additional contact with the State Historic Preservation Office, a response is pending from this office. Although impacts to cultural resources are not expected as a result of the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the State Historic Preservation Office.

4.4.1.1.8 Socioeconomics

After facility modifications, startup, and testing of the plutonium-238 reactor operation facilities at INEEL and target fabrication/processing facilities at ORR, approximately 41 additional workers would be required to operate these facilities (none at INEEL and approximately 41 at ORR [Wham et al. 1998]). The socioeconomic impacts at ORR are the same as those addressed in Section 4.3.1.1.8.

4.4.1.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with Alternative 2, Option 1 are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–56** for INEEL and ORR: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–56 Incremental Radiological Impacts on the Public Around INEEL and ORR from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 1

Receptor	INEEL ATR	ORR REDC	Total
Population within 80 kilometers (50 miles) in the year 2020			
Dose (person-rem)	0	8.8×10^{-5}	8.8×10^{-5}
35-year latent cancer fatalities	0	1.5×10^{-6}	1.5×10^{-6}
Maximally exposed individual			
Annual dose (millirem)	0	1.9×10^{-6}	NA[a]
35-year latent cancer fatality risk	0	3.3×10^{-11}	NA[a]
Average exposed individual within 80 kilometers (50 miles)			
Annual dose[b] (millirem)	0	7.8×10^{-8}	NA[a]
35-year latent cancer fatality risk	0	1.4×10^{-12}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of REDC in the year 2020 (1,134,200).

Key: NA, not applicable.
Source: Model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual operations of ATR at INEEL and REDC at ORR, the projected total incremental population dose in the year 2020 would be 8.8×10^{-5} person-rem. The corresponding number of latent cancer fatalities in the populations surrounding INEEL and ORR from 35 years of operations would be 1.5×10^{-6}. The total incremental dose to the maximally exposed member of the public from annual ATR operations would be 0 millirem because there would be no increase in radiological releases to the environment from ATR associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would, therefore, be zero. The incremental dose to the maximally exposed member of the public from annual REDC operations would be 1.9×10^{-6} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 3.3×10^{-11}.

Incremental doses to involved workers from normal operations are given in **Table 4–57**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to ATR workers would be 0 millirem; for REDC workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0 and approximately 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–57. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–57 Incremental Radiological Impacts on Involved INEEL and ORR Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 1

Receptor—Involved Workers[a]	INEEL ATR	ORR REDC	Total
Total dose (person-rem per year)	0	12[b]	12
35-year latent cancer fatalities	0	0.17	0.17
Average worker dose (millirem per year)	0	170	NA[c]
35-year latent cancer fatality risk	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers.
c. Values cannot be given for the average worker because the workers would be at two different facilities and sites.
Key: NA, not applicable.
Source: Mecham 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts at INEEL would be the same as those of current site operations because no new chemicals are expected to be emitted at ATR.

At ORR, both carcinogenic and noncarcinogenic health effects from exposure to hazardous chemicals were evaluated. It was assumed that under normal operating conditions, the primary exposure pathway for members of the public would be from air emissions released through the 7911 stack. Emissions of chemicals were estimated based on anticipated chemical usage. A worst-case dispersion modeling screening analysis was performed to estimate annual concentrations for each chemical, based on the emissions.

The annual concentration for each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the Hazard Quotient for each chemical. The Hazard Quotients were summed to give the Hazard Index from all noncarcinogenic chemicals associated with this option. A Hazard Index of less than one indicates that adverse health effects from non-cancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical. Hazardous chemical health effects are summarized in **Table 4–58**.

Table 4–58 Incremental Hazardous Chemical Impacts on the Public Around ORR Under Alternative 2 (Use Only Existing Operational Facilities)—Option 1

Chemical	Modeled Annual Increment (milligrams per cubic meter)	RfC - Inhalation (milligrams per cubic meter)	Unit Cancer Risk (risk per milligram per cubic meter)	Hazard Quotient	Cancer Risk
Diethyl benzene	3.37×10^{-5}	1	0.0078	3.37×10^{-5}	2.63×10^{-7}
Methanol	1.23×10^{-6}	1.75	NA	7.03×10^{-7}	NA
Nitric acid	1.53×10^{-6}	0.1225	NA	1.25×10^{-5}	NA
Tributyl phosphate	6.34×10^{-5}	0.01	NA	0.00634	NA
			Hazard Index =	0.00639	

Note: For diethyl benzene, the reference concentration for ethyl benzene and the unit cancer risk for benzene were used to estimate Hazard Quotient and cancer risk because no information was available for diethyl benzene. For tributyl phosphate, the reference concentration for phosphoric acid was used to estimate the Hazard Quotient because no information was available for tributyl phosphate.
Key: NA, not applicable (the chemical is not a known carcinogen or it is a carcinogen and only unit risk will apply); RfC, reference concentration.
Source: DOE 1996a; EPA 1999; modeled increments are based on the SCREEN3 computer code (EPA 1995).

4.4.1.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with ATR target irradiation and REDC target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 miles) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–59** and **4–60**, respectively. Because ATR is currently operating, the consequences and risks are presented for both the current reactor configuration without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configuration. Baseline accident risks attributed to ATR operations refer to accidents that could occur under the current ATR configuration (without neptunium-237 targets). Baseline accident risks are obtained from the data in Table 4–60 by summing the annual risks in columns 2, 3, or 4 for the baseline ATR configuration (0 kilograms per year plutonium-238 production), and then multiplying the sum by 35. The baseline ATR accident risk to the public would be 0.089 latent cancer fatality. Baseline ATR accident risks to the maximally exposed offsite individual and a noninvolved worker would be 8.2×10^{-7} and 7.2×10^{-6} latent cancer fatalities, respectively.

For 35 years of ATR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.45×10^{-7} and 3.48×10^{-6}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00140.

For 35 years of REDC target processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.157.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.158.

The irradiation of neptunium-237 targets to produce plutonium-238 at ATR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at ATR.

Table 4–59 ATR and REDC Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 1

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
ATR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.465	2.33×10^{-4}	5.11×10^{4}	25.5	5.15	0.00206
Large-break LOCA with 5 kg/yr plutonium-238 production	0.604	3.02×10^{-4}	5.17×10^{4}	25.9	7.61	0.00304
Target handling with 0 kg/yr plutonium-238 production[c]	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 5 kg/yr plutonium-238 production	2.05×10^{-4}	1.03×10^{-7}	0.128	6.41×10^{-5}	0.00324	1.30×10^{-6}
REDC accidents						
Ion exchange explosion during neptunium-237 target fabrication	6.13×10^{-9}	3.06×10^{-12}	8.58×10^{-5}	4.29×10^{-8}	5.60×10^{-10}	$2.24x0^{-13}$
Target dissolver tank failure during plutonium-238 separation	1.76×10^{-7}	8.79×10^{-11}	0.00196	9.82×10^{-7}	1.69×10^{-8}	6.74×10^{-12}
Ion exchange explosion during plutonium-238 separation	4.68×10^{-4}	2.34×10^{-7}	5.23	0.00261	4.49×10^{-5}	1.79×10^{-8}
Processing facility beyond-design-basis earthquake	163	0.163	8.91×10^{5}	445	1,310	1.00[d]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. There would be no neptunium-237 targets for this zero-production case. Thus, there would be no associated accident consequences.
d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–60 ATR and REDC Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 1

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual ATR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	2.33×10^{-8}	0.00255	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production (1×10^{-4})	3.02×10^{-8}	0.00259	3.04×10^{-7}
Large-break LOCA incremental risks[c]	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target handling with 5 kg/yr plutonium-238 production[d] (0.001)	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
35-year ATR risk[e]	2.45×10^{-7}	0.00140	3.48×10^{-6}
Annual REDC risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Processing facility beyond-design-basis earthquake (1×10^{-5})	1.63×10^{-6}	0.00445	1.00×10^{-5}[f]
35-year REDC risk	5.71×10^{-5}	0.157	3.50×10^{-4}
35-year Option risk[g]	5.71×10^{-5}	0.158	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. There would be no neptunium-237 targets for the zero-production case. Thus, the 5-kg/yr production rate target-handling risks are the incremental risks.
e. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
f. Risk of an early fatality.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Processing associated with the plutonium-238 production program at REDC, including storage of neptunium-237 and plutonium-238, neptunium-237 target fabrication, postirradiation processing to extract plutonium-238 and to recycle the unconverted neptunium-237 into new targets, would not require the introduction of hazardous chemicals that are not in current use in the facility. The quantities of in-process hazardous chemicals for the plutonium-238 production program are bounded by the quantities of the material currently stored in the facility. The impacts of in-process hazardous chemical accidents associated with the

plutonium-238 production are bounded by the impacts of hazardous chemical accidents for existing storage facilities at REDC.

4.4.1.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to ATR at INEEL. Following irradiation in ATR, the targets would be returned to REDC for processing. After processing, the plutonium-238 product would be shipped to LANL. The analysis is described in Appendix J.

Approximately 689 shipments of radioactive materials would be made by DOE under this option. The total distance traveled on public roads by trucks carrying radioactive materials would be 2.2 million kilometers (1.4 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 12 person-rem; the dose to the public, 240 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.005 latent cancer fatality among transportation workers and 0.12 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0064.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to REDC with a severity category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent fatal cancer risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 were also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks are as follows: a radiological dose to the population of 0.088 person-rem, resulting in 4.4×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.06 traffic fatality.

4.4.1.1.12 Environmental Justice

NORMAL OPERATIONS. The risk of latent cancer fatalities among populations residing within 80 kilometers (50 miles) of ATR and REDC would be less than 2×10^{-6} for 35 years of normal operations (derived from information in Table 4–56). As shown in Table 4–58, the release of hazardous chemicals at ORR would pose no significant risk of cancer or toxic effects among the public. As discussed in Section K.5.1, the likelihood that a latent cancer fatality would result from the ingestion of food that could be radiologically contaminated due to normal operations would be essentially zero at INEEL and ORR. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near INEEL or ORR. As discussed in Section 4.4.1.1.11, no fatalities would be expected for incident-free transportation.

ACCIDENTS. The number of expected latent cancer fatalities among populations at risk due to radiological accidents listed in Table 4–60 would be approximately 0.16. If a radiological accident were to occur at ATR and northwesterly winds prevailed at the time of the accident, radiological contamination from the accident

would be directed toward the Fort Hall Indian Reservation (see Figure K–2). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the population or maximally exposed individual residing within the boundary of the Fort Hall Indian Reservation. In the event a radiological accident were to occur at REDC and southerly winds prevailed at the time of the accident, radiological contamination would be directed toward the predominately minority population of the Scarboro community adjacent to the northern boundary of ORR (see Figure K–6). If the winds were blowing from the west-southwest at the time of the accident, radiological contamination would be directed toward minority populations residing in Knoxville, Tennessee. Accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the minority populations or maximally exposed individuals residing in the Scarboro community or Knoxville.

As discussed in Section 4.4.1.1.11, no fatalities due to transportation accidents would be expected.

In summary, the implementation of this option would pose no significant radiological risk to persons residing in potentially affected areas or along representative transportation routes. Under the conservative assumption that all food consumed in potentially affected areas during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to the ingestion of contaminated food supplies. As discussed in other parts of Section 4.4.1.1, the implementation of this option would not result in significant nonradiological impacts on populations at risk. Thus, implementation would not pose significant and adverse environmental risks to persons residing within potentially affected areas, including minority and low-income persons.

4.4.1.1.13 Waste Management

Virtually no additional waste would be generated as a result of irradiating the neptunium-237 targets in ATR because this reactor would already be operating for other purposes. Only the devices that position the neptunium-237 targets in the core would add to the ATR waste stream. The incremental amount of this waste is anticipated to be very small (about 1 cubic meter [1.3 cubic yards] per year of solid low-level radioactive waste), and therefore, no impacts on the waste management systems at INEEL would be anticipated. However, there would be impacts on ORR's waste management systems as a result of the operation of REDC to fabricate and process the neptunium-237 targets.

The impacts of managing waste associated with neptunium-237 target fabrication and processing in REDC are assumed to be the same as for Option 1 under Alternative 1 (Section 4.3.1.1.13) because the same amount of plutonium-238 would be produced annually. As shown in that section, the impacts on the waste management systems at ORR would be minimal.

4.4.1.1.14 Spent Nuclear Fuel Management

Under all options of this alternative, no additional spent nuclear fuel would be generated from reactor operations specific to neptunium-237 target irradiation. The reactor(s) would already be operating to provide other irradiation services (refer to Appendix B). Thus, there would be no incremental impacts associated with the management of spent nuclear fuel.

4.4.1.2 Permanent Deactivation of FFTF

The environmental impacts associated with the permanent deactivation of FFTF are analyzed in *Environmental Assessment, Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington,* DOE/EA-0993 (DOE 1995a). Summaries of these impacts are given in the following sections. Activities associated with final

decontamination and decommissioning are not within the scope of this NI PEIS. They would be addressed in subsequent NEPA documentation.

4.4.1.2.1 Land Resources

LAND USE. Activities associated with the permanent deactivation of FFTF would not affect land use in the 400 Area because the industrial nature of the area would not change.

VISUAL RESOURCES. The permanent deactivation of FFTF would not involve the removal of existing structures with only minimal construction of small support structures in previously disturbed area facilities; thus, visual resources would not be affected, and the Visual Resource Management Class IV rating of the 400 Area would not change.

4.4.1.2.2 Noise

Noise associated with the permanent deactivation of FFTF would be similar to sound levels generated by current activities in the 400 Area. Onsite noise impacts from deactivation would be expected to be minimal, and changes in offsite noise levels would not be noticeable since the nearest site boundary is 6.1 kilometers (3.8 miles) to the east. Noise levels associated with traffic during deactivation may be slightly higher as a result of moving fuel assemblies, equipment, and materials. When deactivation is complete, noise levels associated with traffic may decrease somewhat if the FFTF shutdown results in a decrease in the Hanford workforce (DOE 1995a). The contribution of FFTF deactivation activities to traffic noise levels on site and off site would be minor and would not lead to noticeable changes in noise levels either on site or off site. There would be no loud noises associated with the deactivation of FFTF that would adversely affect wildlife.

4.4.1.2.3 Air Quality

Several sources of air pollutants are operated to support FFTF during standby: an emergency gas turbine generator, a diesel-driven fire pump, and oil-fired preheaters. If any of Alternatives 2 through 5 were selected for implementation, then these sources would be shut down. Concentrations of air pollutants at the Hanford Site boundary resulting from these sources were estimated from a dispersion-modeling screening analysis conducted with maximum expected emission rates and worst-case meteorological conditions. Although these sources are operated intermittently, and they are not necessarily operated simultaneously, concentrations of air pollutants from all three sources were summed to give the conservative estimate of air quality impacts of maintaining FFTF in standby shown in **Table 4–61.** Concentrations of air pollutants listed in Table 4–61 are negative to indicate that they represent a decrease in adverse impacts relative to air quality with FFTF in standby, although the decrease would be less than the conservative estimate for standby.

4.4.1.2.4 Water Resources

The permanent deactivation of FFTF would eventually result in the cessation of sanitary and process wastewater discharges (i.e., cooling tower blowdown) from the facility because auxiliary systems would be shut down following hot sodium drainage. This would eliminate the annual discharge of 76 million liters (20 million gallons) of nonradioactive process wastewater to the 400 Area process sewer system and ultimately to the 400 Area Pond (i.e., 4608 B/C percolation ponds). The FFTF component (3.8 million liters [1 million gallons] per year) of 400 Area sanitary wastewater discharges to the Energy Northwest treatment system would also be eliminated. In addition, groundwater withdrawals by 400 Area facilities during standby of approximately 197 million liters (52 million gallons) per year would be greatly reduced or eliminated entirely (see Section 4.2.1.2.4). As part of the sodium-removal process, residual sodium would be washed from fuel assemblies and other reactor components, including instrumentation assemblies from the reactor core. This

Table 4–61 Incremental Hanford Concentrations Associated with All Options of Alternatives 2 through 5

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[a]	Modeled Increment (micrograms per cubic meter)
Carbon monoxide	8 hours	10,000[b]	-3.5
	1 hour	40,000[b]	-5.1
Nitrogen dioxide	Annual	100[b]	-0.032
PM_{10}	Annual	50[c]	-0.002
	24 hours	150[c]	-0.898
Sulfur dioxide	Annual	50[d]	-0.164
	24 hours	260[d]	-29.8
	3 hours	1,300[b]	-67.0
	1 hour	660[d]	-74.4

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those for ozone, particulate matter, and lead, and those based on annual averages, are not to be exceeded more than once per year. The 24-hour PM_{10} (particulate matter with an aerodynamic diameter less than or equal to 10 micrometers) standard is attained when the expected number of days with a 24-hour average concentration above the standard is equal to or less than 1. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.
b. Federal and state standard.
c. Federal standard currently under litigation.
d. State standard.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); additional data from Nielsen 2000.

would be conducted in FFTF's Interim Examination and Maintenance Cell using the existing process and equipment designed for this purpose. Ion exchange would reduce the entire volume of radioactive wastewater generated to less than 7,600 liters (2,000 gallons). This wastewater would be disposed of at existing onsite waste management facilities; spent ion exchange resin would be packaged and properly disposed of as well (DOE 1995a:3-9, 3-15).

4.4.1.2.5 Geology and Soils

No facilities would be demolished to effect permanent deactivation of FFTF. Any necessary ground disturbance would be confined to previously disturbed areas immediately adjacent to the FFTF complex. As a result, the impact on geologic and soil resources in the 400 Area of Hanford would be expected to be negligible. Activities associated with final decontamination and decommissioning and related activities that could impact geologic or soil resources to a greater degree would be addressed in subsequent NEPA documentation.

4.4.1.2.6 Ecological Resources

Activities associated with the permanent deactivation of FFTF would not impact the limited ecological resources present in the 400 Area. No threatened and endangered species reside in the vicinity of the 400 Area; consequently, no adverse impacts on such species would occur from the proposed action.

4.4.1.2.7 Cultural and Paleontological Resources

The 400 Area is highly disturbed with little potential for the occurrence of cultural and paleontological resources. For this reason and because there would be no ground disturbance beyond previously disturbed areas associated with the permanent deactivation of FFTF, impacts on cultural and paleontological resources from the proposed action would not occur.

4.4.1.2.8 Socioeconomics

The deactivation of FFTF would result in a loss of about 242 jobs at Hanford (DOE 1997b). However, it should coincide with an increase in overall site employment at Hanford in connection with construction of the tank waste remediation system. The personnel who had worked at FFTF would be absorbed into other operations at Hanford. If this were not the case, the loss of 242 jobs would result in the loss of 613 indirect jobs in the region around Hanford. The potential employment loss of 855 direct and indirect jobs represents less than 0.4 percent of the projected regional economic area workforce and, therefore, would not result in a noticeable impact on the regional economic area.

In the region of influence, the loss of employment resulting from this alternative would not significantly impact community services in the Hanford region of influence. Assuming that 91 percent of those losing their jobs left the Hanford region of influence with their families (refer to Section 3.4.8), the region's population would decrease by approximately 1,494 persons. Given the current population-to-student ratio in the region of influence, this would likely result in a decrease of about 309 students, dropping the average school enrollment from 92.8 percent to 91.8 percent.

Community services in the region of influence may change to accommodate the population decrease as follows: 19 less teachers would be needed if the current student-to-teacher ratio of 16.0:1 was maintained; 2 less police officers would be needed to maintain the current officer-to-population ratio of 1.5:1000; 5 less firefighters would be needed to maintain the current firefighter-to-population ratio of 3.4:1000; and 2 less doctors would be needed to maintain the current physician-to-population ratio of 1.4:1000. Thus, 28 additional positions could be lost if community services were maintained at current levels. Hospitals in the region of influence would not experience any change from the 2.1 beds per 1,000 persons currently available. None of these projected changes should have a major impact on the level of community services currently offered in the region of influence.

4.4.1.2.9 Public and Occupational Health and Safety—Normal Deactivation Activities

Assessments of incremental radiological and chemical impacts associated with the permanent deactivation of FFTF are presented in this section. Supplemental information is provided in Appendix H. During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from deactivation of FFTF at Hanford are given in **Table 4–62**: the population within 80 kilometers (50 miles), the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding populations and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual deactivation activities, the projected estimated total incremental population dose is estimated to be 0.036 person-rem. The corresponding number of latent cancer fatalities in the population surrounding Hanford would be 1.8×10^{-5}. The total annual incremental dose to the maximally exposed member of the public from deactivation activities would be 2.6×10^{-4} millirem. The corresponding risk of a latent cancer fatality to this individual would be 1.3×10^{-10}.

Table 4–62 Incremental Radiological Impacts on the Public Around Hanford from FFTF Deactivation Activities

Receptor	FFTF Deactivation
Estimated population within 80 kilometers (50 miles)	
Dose (person-rem)	0.036
1-year latent cancer fatalities	1.8×10^{-5}
Maximally exposed individual	
Annual dose (millirem)	2.6×10^{-4}
1-year latent cancer fatality risk	1.3×10^{-10}
Average exposed individual within 80 kilometers (50 miles)	
Annual dose[a] (millirem)	7.2×10^{-5}
1-year latent cancer fatality risk	3.6×10^{-11}

a. Obtained by dividing the estimated population dose by the number of people projected to live within 80 kilometers (50 miles) of FFTF (about 500,000).

Source: DOE 1995a.

Estimated incremental doses to involved workers associated with annual deactivation activities are given in **Table 4–63**; these workers are defined as those directly associated with all planned deactivation activities. Under this alternative, the incremental annual average dose to FFTF deactivation workers is estimated not to exceed 6 millirem. The incremental annual dose received by the FFTF deactivation workforce is estimated not to exceed 0.06 person-rem. The risks and numbers of latent cancer fatalities among these workers from annual operations are included in Table 4–63. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–63 Incremental Radiological Impacts on Involved FFTF Workers from Deactivation Activities

Receptor	FFTF Deactivation
Involved workers[a]	
Total dose (person-rem per year)	<0.06[b]
1-year latent cancer fatalities	$<2.4\times10^{-5}$
Average worker dose (millirem per year)	<6
1-year latent cancer fatality risk	$<2.4\times10^{-6}$

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations will be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program will be enforced.

b. Based on an estimated 10 badged workers.

Note: < means "less than."

Source: DOE 1995a.

HAZARDOUS CHEMICAL IMPACTS. No hazardous chemicals are anticipated to be released in substantial quantities from activities associated with permanently deactivating FFTF when compared to the annual amount routinely generated throughout Hanford. The deactivation of FFTF would result in a decrease of both near-term and long-term exposures (DOE 1995a).

4.4.1.2.10 Public and Occupational Health and Safety—Deactivation Accidents

Impacts from a postulated accident associated with the permanent deactivation of FFTF are presented in this section. The FFTF shutdown environmental assessment (DOE 1995a) describes several accident scenarios and their consequences. Rather than a summary of the environmental assessment accidents, a reevaluation of

a limiting deactivation accident was performed. The reevaluation was performed because the current FFTF status is significantly different than at the time the environmental assessment was completed.

FFTF is currently defueled; therefore, accidents related to defueling need not be considered. Also because of defueling and decay of radioactivity over time, the current sodium radionuclide inventories are much less than when the environmental assessment was completed. Considering the current FFTF conditions, it was determined that a primary heat transport system sodium drain accident would be the accident with the highest consequences. A detailed description of the accident analysis is provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 miles) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10.

The FFTF deactivation accident is a sodium spill during the transfer of primary sodium to a treatment tank. The accident frequency is the probability of a sodium spill during the transfer process. The frequency is per event (sodium transfer) rather than per year. Since the risk remains constant for any time period, the 35-year risk is the same as the accident risk presented.

Consequences and associated risks are presented in **Tables 4–64** and **4–65**, respectively.

Table 4–64 Consequences of FFTF Deactivation Accident

	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
Accident	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
Primary heat transport system sodium drain accident	4.75×10^{-10}	2.38×10^{-13}	3.64×10^{-5}	1.82×10^{-8}	3.88×10^{-9}	1.55×10^{-12}

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table 4–65 Risks of FFTF Deactivation Accident

Accident (Frequency)[a]	Maximally Exposed Individual[b]	Population to 80 Kilometers (50 Miles)[c]	Noninvolved Worker[b]
Primary heat transport system sodium drain accident (0.10)	2.38×10^{-14}	1.82×10^{-9}	1.55×10^{-13}

a. Per event.
b. Increased likelihood of a latent cancer fatality.
c. Increased number of latent cancer fatalities.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

For an FFTF deactivation accident, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.38×10^{-14} and 1.55×10^{-13}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 1.82×10^{-9}.

Deactivating FFTF would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to deactivating FFTF.

4.4.1.2.11 Public and Occupational Health and Safety—Transportation

There would be no transportation impacts associated with permanently deactivating FFTF.

4.4.1.2.12 Environmental Justice

NORMAL OPERATIONS. For deactivation activities at Hanford, the number of expected latent cancer fatalities among populations residing within 80 kilometers (50 miles) of FFTF would be less than 2×10^{-5} (derived from information in Table 4–62). As discussed in Section 4.4.1.2.9, the release of hazardous chemicals at FFTF would pose no significant risk of cancer or toxic effects among the public. There would be no intersite transportation associated with deactivation activities, and therefore, no transportation effects on the public.

ACCIDENTS. Accidents at FFTF also pose no significant environmental risk to the public. As shown, in Table 4–65, the risk of a public fatality associated with a sodium drain accident at FFTF would be essentially zero.

In summary, deactivating FFTF would have no significant environmental effects on the public. Thus, the deactivation would pose no disproportionately high and adverse risks for minority or low-income populations.

4.4.1.2.13 Waste Management

As discussed in the *Environmental Assessment, Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington* (DOE 1995a), the hazardous materials (e.g., solvents, glycols, polychlorinated biphenyls, asbestos) which may be removed or stabilized as a result of the deactivation of FFTF would be managed and reused, recycled, or disposed of in accordance with applicable Federal and state regulations. Such materials include approximately 360,000 liters (94,000 gallons) of ethylene glycol, 32,000 liters (8,500 gallons) of polychlorinated biphenyls, transformer oil, and 370,000 liters (99,000 gallons) of fuel oil. Approximately 8,200 drums of sodium sulfate (at approximately 208 liters or 55 gallons, each) could be generated for disposal. None of the materials would be anticipated to be generated in substantial quantities when compared to the annual amount routinely generated throughout Hanford (DOE 1995a:5-12).

The inventory of bulk metallic sodium (approximately 980,000 liters [260,000 gallons]) would undergo appropriate excess evaluations to determine if alternative sponsors and/or uses were available. In the event no viable use were determined, the bulk metallic sodium would be converted to an acceptable stable form (e.g., sodium sulfate), dried, collected into containers, and transported to an appropriate facility at Hanford for disposal (DOE 1995a:ES-2).

4.4.1.2.14 Spent Nuclear Fuel Management

Under deactivation, the irradiated FFTF assemblies and pin containers have been, or would be, placed into dry storage casks and transferred to storage at the site's Interim Storage Area (ISA). Each fuel assembly or pin container would be limited to a maximum decay heat value of 250 watts (850 BTU per hour) for fuel offload handling. At this heat level, no active cooling would be required, and many of the fission products and noble gases would have decayed substantially.

A typical FFTF spent nuclear fuel–handling sequence is as follows: sodium-wetted fuel assemblies are washed using existing FFTF process equipment; the spent nuclear fuel is subjected to a moist argon atmosphere to

slowly react residual sodium in a controlled manner; several water rinses of the fuel are conducted; the fuel receives a final dry; the fuel is transferred to the dry storage casks for interim storage in the Interim Storage Area. The dry casks subsequently would be transferred to the Canister Storage Building Complex in the 200-East Area for storage of the spent nuclear fuel pending disposition (DOE 1997b). When the geologic repository becomes available, the spent nuclear fuel would be transferred from the 200-East Area to the repository for disposal.

4.4.2 Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Option 2 involves operating ATR at INEEL to irradiate neptunium-237 targets, and operating FDPF at INEEL to fabricate and process these targets. This alternative also includes storage of the neptunium-237 transported to INEEL from SRS in Building CPP–651 or FDPF.

The transportation of the neptunium-237 from SRS to INEEL for processing and fabrication into neptunium-237 targets in FDPF, and the transportation of the plutonium-238 product from INEEL to LANL following postirradiation processing in FDPF also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.2.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.4.2.1.1 Land Resources

LAND USE. The use of ATR to irradiate neptunium-237 targets would not result in impacts on land use at INEEL for the reasons described in Section 4.4.1.1.1.

Building CPP–651 or FDPF at INEEL would be used for neptunium-237 storage, and FDPF for target fabrication and processing. These are existing facilities in the INTEC area. The use of these facilities would require internal modifications, but no new facilities would be built. Because no additional land would be disturbed and use of the facilities would be compatible with the missions for which they were designed, there would be no change in land use at INEEL.

VISUAL RESOURCES. The use of ATR to irradiate neptunium-237 targets would not result in impacts on visual resources at INEEL for the reasons described in Section 4.4.1.1.1.

All activities associated with neptunium-237 storage, target fabrication, and processing would take place in existing facilities that would require no external modifications. Thus, there would be no change in appearance. The current Visual Resource Management Class IV rating for INTEC would not change, and there would be no impact on visual resources.

4.4.2.1.2 Noise

The irradiation of neptunium-237 targets in ATR would not be expected to result in noise impacts at INEEL for the reasons described in Section 4.4.1.1.2.

Neptunium-237 storage in Building CPP–651 or FDPF, and target fabrication and processing at FDPF would generate noise levels similar to those presently associated with operations in INTEC. Onsite noise impacts

would be expected to be minimal, and changes in offsite noise levels should not be noticeable because the nearest site boundary is 12 kilometers (7.5 miles) to the south. Changes in traffic volume going to and from INTEC would be small and would result in only minor changes to onsite and offsite noise levels. There would be no loud noises associated with neptunium-237 storage that would adversely impact wildlife.

4.4.2.1.3 Air Quality

The concentrations at INEEL attributable to this option are presented in **Table 4–66**. The concentrations for the option are based on a dispersion modeling screening analysis conducted with maximum expected emission rates and a set of worst-case meteorological conditions.

Table 4–66 Incremental INEEL Concentrations[a] Associated with Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Criteria pollutants			
Nitrogen dioxide	Annual	100	3.66×10^{-4}
Sulfur dioxide	Annual	80	0.024
	24 hours	365	0.19
	3 hours	1,300	0.43
Toxic air pollutants			
Methanol	24 hours	13,000	0.0048
Nitric acid	24 hours	250	0.0097
Paraffin hydrocarbons	24 hours	100	0.44
Tributyl phosphate	24 hours	110	0.25

a. For comparison with ambient air quality standards.
Source: 40 CFR Part 50; ID DHW 1998; modeled increments are based on the SCREEN3 computer code (EPA 1995).

Only those air pollutants expected to be emitted that have ambient air quality standards are presented in the table. The change in concentrations of these pollutants would be small and would be below the applicable ambient air quality standards even when ambient monitoring values and the contribution from other site activities are included.

The concentrations at INEEL attributed to this option are compared to the Prevention of Significant Deterioration Class II increments for nitrogen dioxide and sulfur dioxide in **Table 4–67**.

Table 4–67 PSD Class II Increments Compared to INEEL Concentrations Associated with Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Pollutant	Averaging Period	Allowable PSD Increment (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Nitrogen dioxide	Annual	25	3.66×10^{-4}
Sulfur dioxide	Annual	20	0.024
	24 hours	91	0.19
	3 hours	512	0.43

Key: PSD, Prevention of Significant Deterioration.
Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995).

Health effects from hazardous chemicals associated with this option are addressed in Section 4.4.2.1.9. The air quality impacts of transportation among SRS, INEEL, and LANL are presented in Section 4.4.2.1.11.

4.4.2.1.4 Water Resources

Impacts on water resources at INEEL associated with operating ATR to irradiate neptunium-237 targets would be negligible as previously described in Section 4.4.1.1.4.

Building CPP–651 and/or FDPF, existing facilities in the INTEC area of INEEL, would be used for neptunium-237 storage; FDPF would also be used for the fabrication and processing of targets in support of plutonium-238 production. Impacts on water resources indicators at INEEL would be the same as those described in Section 4.3.2.1.4. In summary, a small increase in water use and sanitary wastewater generation would be anticipated, mainly attributable to increased staffing levels. Also, there would be a very small increase in process wastewater generation, but there would be no radiological liquid effluent discharge to the environment under normal operations.

4.4.2.1.5 Geology and Soils

The irradiation of neptunium-237 targets in ATR would not be expected to result in impacts on geologic or soil resources at INEEL, nor be jeopardized by large-scale geologic conditions, for the reasons described in Section 4.4.1.1.5.

Building CPP–651 and/or FDPF would be used to store neptunium-237, and FDPF would be used to fabricate and process targets. Because both are existing facilities, there would be no disturbance to either geologic or soil resources at INTEC. Hazards from large-scale geologic conditions at INEEL, such as earthquakes and volcanoes, were previously evaluated as discussed in Section 4.2.3.2.5. The analysis determined that these hazards present a low risk for neptunium-237 storage in INTEC facilities. Likewise, large-scale geologic conditions do not present a substantial risk to use of the proposed facilities for neptunium-237 storage, target fabrication, and processing. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.2.1.6 Ecological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on ecological resources at INEEL for the reasons described in Section 4.4.1.1.6.

Because no new construction is planned, the use of Building CPP–651 and/or FDPF would not result in direct disturbance to ecological resources. As noted in Section 4.4.2.1.2, there would be no loud noises that would adversely impact wildlife. Because water usage and wastewater discharge would be small fractions of current values, there would be no impact on aquatic resources (Section 4.4.2.1.4). Due to the developed nature of the area and the fact that no new construction would take place, impacts on threatened and endangered species would not occur.

Consultation letters to comply with Section 7 of the Endangered Species Act were sent to the U.S. Fish and Wildlife Service and the Idaho Department of Fish and Game (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. The Idaho Department of Fish and Game indicated that its database contained no known occurrences of special status plants or animals near the project area. While DOE has made additional contact with the U.S. Fish and Wildlife Service, a response is pending from this agency. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at INEEL prior to the receipt of input from the Service.

4.4.2.1.7 Cultural and Paleontological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on cultural and paleontological resources at INEEL for the reasons described in Section 4.4.1.1.7.

Because no new construction would take place, impacts on cultural and paleontological resources at INTEC would not occur. Use of Building CPP–651 and/or FDPF to store neptunium-237 or FDPF to fabricate and process neptunium-237 targets would not change the status of six historic structures located at INTEC. Native American resources occurring in the vicinity of INTEC would not be impacted by neptunium-237 storage, target fabrication, or processing.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). The State Historic Preservation Office indicated that Building CPP–651 and FDPF are likely to be eligible for the National Register of Historic Places as contributory properties in a potential historic district of exceptional significance. However, at this time, the State Historic Preservation Office has determined that more information is needed prior to assisting DOE in evaluating these properties. The State Historic Preservation Office also indicated that since there would be no new construction, there is little potential for effects on archaeological properties. DOE would provide additional information as required to the Idaho State Historic Preservation Office prior to the use of any facility at INEEL for the proposed project. Consultation was conducted with interested Native American tribes; however, responses are pending.

4.4.2.1.8 Socioeconomics

After facility modifications, startup, and testing of the plutonium-238 reactor operation and target fabrication/processing facilities at INEEL, approximately 24 additional workers would be required to operate these facilities (Hill et al. 1999). The socioeconomic impacts at INEEL are the same as those addressed in Section 4.3.2.1.8.

4.4.2.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–68** for INEEL: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

Table 4–68 Incremental Radiological Impacts on the Public Around INEEL from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Receptor	INEEL ATR	INEEL FDPF	Total
Population within 80 kilometers (50 miles) in the year 2020			
Dose (person-rem)	0	3.9×10^{-6}	3.9×10^{-6}
35-year latent cancer fatalities	0	6.7×10^{-8}	6.7×10^{-8}
Maximally exposed individual			
Annual dose (millirem)	0	2.6×10^{-7}	2.6×10^{-7}
35-year latent cancer fatality risk	0	4.6×10^{-12}	4.6×10^{-12}
Average exposed individual within 80 kilometers (50 miles)			
Annual dose[a] (millirem)	0	2.0×10^{-8}	2.0×10^{-8}
35-year latent cancer fatality risk	0	3.6×10^{-13}	3.6×10^{-13}

a. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FDPF in the year 2020 (188,400).

Source: Model results, using the GENII computer code (Napier et al. 1988).

As a result of annual operations of both facilities, the projected total incremental population dose in the year 2020 would be 3.9×10^{-6} person-rem. The corresponding number of latent cancer fatalities in the population surrounding INEEL from 35 years of operations would be 6.7×10^{-8}. The total incremental dose to the maximally exposed member of the public from annual ATR operations would be 0 millirem because there would be no increase in radiological releases to the environment from ATR associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would, therefore, be zero. The incremental dose to the maximally exposed member of the public from annual FDPF operations would be 2.6×10^{-7} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 4.6×10^{-12}.

Incremental doses to involved workers from normal operations are given in **Table 4–69**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to ATR workers would be 0 millirem; for FDPF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0 and approximately 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–69. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–69 Incremental Radiological Impacts on Involved INEEL Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Receptor—Involved Workers[a]	INEEL ATR	INEEL FDPF	Total
Total dose (person-rem per year)	0	12[b]	12
35-year latent cancer fatalities	0	0.17	0.17
Average worker dose (millirem per year)	0	170	NA[c]
35-year latent cancer fatality risk	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.

b. Based on an estimated 75 badged workers.

c. Values cannot be given for the average worker because the workers would be at two different facilities.

Key: NA, not applicable.

Source: Mecham 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. At INEEL, both carcinogenic and noncarcinogenic health effects from exposure to hazardous chemicals were evaluated. It was assumed that under normal operating conditions, the primary exposure pathway for members of the public would be from air emissions released through the FDPF stack. Emissions of chemicals were estimated based on anticipated chemical usage. A worst-case dispersion modeling screening analysis was performed to estimate annual concentrations for each chemical, based on the emissions.

The annual concentration for each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the Hazard Quotient for each chemical. The Hazard Quotients were summed to give the Hazard Index from all noncarcinogenic chemicals associated with this option. A Hazard Index of less than one indicates that adverse health effects from non-cancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical. Hazardous chemical health effects are summarized in **Table 4–70**.

Table 4–70 Incremental Hazardous Chemical Impacts on the Public Around INEEL Under Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Chemical	Modeled Annual Increment (milligrams per cubic meter)	RfC - Inhalation (milligrams per cubic meter)	Unit Cancer Risk (risk per milligram per cubic meter)	Hazard Quotient	Cancer Risk
Diethyl benzene	1.65×10^{-5}	1	0.0078	1.65×10^{-5}	1.29×10^{-7}
Methanol	6.02×10^{-7}	1.75	NA	3.44×10^{-7}	NA
Nitric acid	1.21×10^{-6}	0.1225	NA	9.86×10^{-6}	NA
Tributyl phosphate	3.10×10^{-5}	0.01	NA	0.00310	NA
			Hazard Index =	0.0031	

Note: For diethyl benzene, the reference concentration for ethyl benzene and the unit cancer risk for benzene were used to estimate Hazard Quotient and cancer risk because no information was available for diethyl benzene. For tributyl phosphate, the reference concentration for phosphoric acid was used to estimate the Hazard Quotient because no information was available for tributyl phosphate.
Key: NA, not applicable (the chemical is not a known carcinogen); RfC, reference concentration.
Source: DOE 1996a; EPA 1999; modeled increments are based on the SCREEN3 computer code (EPA 1995).

4.4.2.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with ATR target irradiation and FDPF target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a uninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to

incorporate in this NI PEIS, summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–71** and **4–72**, respectively. Because ATR is currently operating, the consequences and risks are presented for both the current reactor configuration without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configuration.

Table 4–71 ATR and FDPF Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
ATR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.465	2.33×10^{-4}	5.11×10^{4}	25.5	5.15	0.00206
Large-break LOCA with 5 kg/yr plutonium-238 production	0.604	3.02×10^{-4}	5.17×10^{4}	25.9	7.61	0.00304
Target handling with 0 kg/yr plutonium-238 production[c]	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 5 kg/yr plutonium-238 production	2.05×10^{-4}	1.03×10^{-7}	0.128	6.41×10^{-5}	0.00324	1.30×10^{-6}
FDPF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.01×10^{-9}	1.01×10^{-12}	2.49×10^{-5}	1.24×10^{-8}	7.26×10^{-9}	2.91×10^{-12}
Target dissolver tank failure during plutonium-238 separation	6.11×10^{-8}	3.05×10^{-11}	5.65×10^{-4}	2.82×10^{-7}	2.17×10^{-7}	8.69×10^{-11}
Ion exchange explosion during plutonium-238 separation	1.63×10^{-5}	8.13×10^{-9}	0.150	7.51×10^{-5}	5.79×10^{-5}	2.31×10^{-8}
Processing facility beyond-design-basis earthquake	42.5	0.0425	1.64×10^{5}	82.0	1,200	1.0[d]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. There would be no neptunium-237 targets for this zero-production case. Thus, there would be no associated accident consequences.
d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–72 ATR and FDPF Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual ATR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	2.33×10^{-8}	0.00255	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production (1×10^{-4})	3.02×10^{-8}	0.00259	3.04×10^{-7}
Large-break LOCA incremental risks[c]	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target handling with 5 kg/yr plutonium-238 production (0.001)[d]	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
35-year ATR risk[e]	2.45×10^{-7}	0.00140	3.48×10^{-6}
Annual FDPF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Processing facility beyond-design-basis earthquake (1×10^{-5})	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}[f]
35-year FDPF risk	1.49×10^{-5}	0.0287	3.50×10^{-4}
35-year Option risk[g]	1.51×10^{-5}	0.0301	3.53×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. There would be no neptunium-237 targets for the zero-production case. Thus, the 5-kg/yr production rate target-handling risks are the incremental risks.
e. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
f. Risk of an early fatality.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

For 35 years of ATR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.45×10^{-7} and 3.48×10^{-6}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00140.

For 35 years of FDPF target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 1.49×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0287.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 1.51×10^{-5} and 3.53×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0301.

The irradiation of neptunium-237 targets to produce plutonium-238 at ATR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at ATR.

No chemical processing activities are currently performed at FDPF and no chemicals are stored in this facility. Processing activities in support of plutonium-238 production would require the introduction of hazardous chemicals, specifically nitric acid and nitric oxide. Potential health impacts from accidental releases of nitric acid were assessed by comparing estimated airborne concentrations of the chemicals to ERPG developed by the American Industrial Hygiene Association. The ERPG-1 value (0.5 part per million) is the maximum airborne concentration below which nearly all individuals could be exposed for up to one hour, resulting in only mild, transient, and reversible adverse health effects. The ERPG-2 value (10 parts per million) is protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. The ERPG-3 value (25 parts per million) is indicative of potentially life-threatening health effects.

The maximum distances, in meters, needed to reach the ERPG values for nitric acid releases at FDPF for Stability Classes D and F are shown in **Table 4–73**. Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions while Stability Class F represents worst-case meteorological conditions. The number of involved and noninvolved workers potentially exposed would vary with a number of factors such as the time of day and whether they are sheltered within buildings at the time of release. Individuals at the nearest highway (5,800 meters [3.8 miles]) and at the nearest site boundary (13,952 meters [8.7 miles]) from FDPF would be exposed to levels well below ERPG-1.

Table 4–73 ERPG Distances for Nitric Acid Releases at FDPF Under Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Evaluation Parameter	Stability Class D (meters)	Stability Class F (meters)
ERPG-3	375	450
ERPG-2	500	600
ERPG-1	2,000	3,000

Note: To convert from meters to miles, multiply by 6.22×10^{-4}.
Key: ERPG, Emergency Response Planning Guideline.

There are no ERPG values for nitric oxide. For nitric oxide accidents, the level of concern has been estimated by using one-tenth of the "Immediately Dangerous to Life and Health" level published by the National Institute for Occupational Safety and Health. The Immediately Dangerous to Life and Health value for nitric oxide is 100 parts per million. The level of concern value used for this NI PEIS is 10 parts per million. The level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects as a result of a single exposure for a relatively short period of time.

For FDPF, the maximum distances needed to reach the level of concern for nitric oxide releases for Stability Classes D and F are 500 and 2,000 meters (0.31 and 1.24 miles), respectively. The number of involved and noninvolved workers potentially exposed would vary with a number of factors such as the time of day and whether they are sheltered within buildings at the time of release. Individuals at the nearest highway (5,800 meters [3.6 miles]) and at the nearest site boundary (13,952 meters [8.7 miles]) from FDPF would be exposed to levels well below the level of concern for nitric oxide.

Potential health impacts from the accidental release of the hazardous chemicals were assessed for a noninvolved worker, offsite individuals who are members of the public located at the nearest site boundary and onsite individuals who are members of the public located at the nearest highway access.

The impacts associated with the accidental release of nitric acid and nitric oxide at FDPF are presented in **Table 4–74**.

Table 4–74 FDPF Hazardous Chemical Accident Impacts Under Alternative 2 (Use Only Existing Operational Facilities)—Option 2

Receptor	Evaluation Parameter	Nitric Acid		Nitric Oxide	
		Stability Class D	Stability Class F	Stability Class D	Stability Class F
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	3.3 <ERPG-2 Mild, transient	8.4 <ERPG-2 Mild, transient	4.2 <LOC Mild, transient	67.5 >LOC Serious
Nearest highway maximally exposed individual	Parts per million Level of concern Potential health effects	0.05 < ERPG-1 None	0.15 ERPG-1 Mild, transient	0.09 < LOC None	0.87 < LOC None
Site boundary maximally exposed individual	Parts per million Level of concern Potential health effects	<<0.05 < ERPG-1 None	<<0.15 ERPG-1 Mild, transient	<<0.09 < LOC None	<<0.87 < LOC None

Note: < means "less than"; << means "much less than."
Key: ERPG, Emergency Response Planning Guideline; LOC, level of concern.
Source: Model results.

4.4.2.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to INEEL for target fabrication in FDPF. DOE would transport the unirradiated neptunium-237 targets from FDPF to ATR, also on the INEEL site. Following irradiation in ATR, the targets would be returned to FDPF for processing. After this processing, the plutonium-238 product would be shipped to LANL. The analysis is described in Appendix J.

Approximately 59 intersite shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 0.15 million kilometers (0. 1 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 1.3 person-rem; the dose to the public, 8 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0005 latent cancer fatality among transportation workers and 0.004 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0007.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident (probability of occurrence: 1 in 10 million per year) would not breach the transportation package. The consequences of more severe accidents that could breach the transportation package and release radioactive material were evaluated and estimated to have probabilities of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.042 person-rem, resulting in 2.1×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.0006 traffic fatality.

4.4.2.1.12 Environmental Justice

NORMAL OPERATIONS. For 35 years of normal operations under this option, the number of expected latent cancer fatalities among populations residing within 80 kilometers (50 miles) of ATR and FDPF would be essentially zero (derived from information in Table 4–68). As shown in Table 4–70, the release of hazardous chemicals at INEEL would pose no significant risk of cancer or toxic effects among the public. As discussed in Section K.5.1, the likelihood that a latent cancer fatality would result from the ingestion of food that could be radiologically contaminated due to normal operations would be essentially zero at INEEL. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near INEEL. As discussed in Section 4.4.2.1.11, no fatalities due to transportation activities would be expected.

ACCIDENTS. The number of expected latent cancer fatalities among the populations at risk due to radiological accidents listed in Table 4–72 would be approximately 0.03. If a radiological accident were to occur at ATR or FDPF and northwesterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Fort Hall Indian Reservation (see Figure K–2). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the population or maximally exposed individual residing within the boundary of the Fort Hall Indian Reservation.

As discussed in Section 4.4.2.1.11, no fatalities due to transportation accidents would be expected.

In summary, the implementation of this option would pose no significant radiological risk to persons residing in potentially affected areas or along representative transportation routes. Under the conservative assumption that all food consumed in potentially affected areas during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to the ingestion of contaminated food supplies. As discussed in other parts of Section 4.4.2.1, the implementation of this option would not result in significant nonradiological impacts on populations at risk. Thus, implementation would not pose significant and adverse environmental risks to persons residing within potentially affected areas, including minority and low-income persons.

4.4.2.1.13 Waste Management

Only an extremely small amount of additional waste would be generated as a result of irradiating neptunium-237 targets in ATR (Section 4.4.1.1.13). However, waste would be associated with FDPF operations to fabricate and process neptunium-237 targets.

The impacts of managing waste associated with neptunium-237 target fabrication and processing in FDPF are assumed to be the same as for Option 2 under Alternative 1 (Section 4.3.2.1.13) because the same amount of plutonium-238 would be produced annually. As shown in that section, the impacts on the waste management systems at INEEL would be minimal.

4.4.2.1.14 Spent Nuclear Fuel Management

No incremental impacts would be associated with the management of spent nuclear fuel (refer to Section 4.4.1.1.14).

4.4.2.2 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.4.3 Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Option 3 involves operating ATR at INEEL to irradiate neptunium-237 targets, and operating FMEF at Hanford to fabricate and process these targets and to store the neptunium-237 transported to Hanford from SRS.

The transportation of the neptunium-237 from SRS to Hanford for processing and fabrication into neptunium-237 targets in FMEF, the transportation of these targets from Hanford to INEEL for irradiation in ATR, the transportation of the irradiated targets back to Hanford for postirradiation processing in FMEF, and the transportation of the plutonium-238 product from Hanford to LANL also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.3.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.4.3.1.1 Land Resources

LAND USE. The use of ATR to irradiate neptunium-237 targets would not result in impacts on land use at INEEL for the reasons described in Section 4.4.1.1.1.

FMEF, an existing facility in the 400 Area of Hanford, would be used for neptunium-237 storage, target fabrication, and processing. The use of FMEF would require the construction of a new 76-meter (250-foot) stack. Because the stack would be placed on previously disturbed land, and the use of FMEF for target fabrication and processing would be compatible with the mission for which is was designed, change in land use in the 400 Area would be minimal.

VISUAL RESOURCES. The use of ATR to irradiate neptunium-237 targets would not result in impacts on visual resources at INEEL for the reasons described in Section 4.4.1.1.1.

Neptunium-237 storage, target fabrication, and processing would take place in FMEF. Although FMEF is an existing facility, its use would require construction of a 76-meter (250-foot) stack. While the stack would be visible from surrounding areas, it would not change the overall appearance of the 400 Area or its Visual Resource Management Class IV rating. Thus, impacts on visual resources would be minimal.

4.4.3.1.2 Noise

The irradiation of neptunium-237 targets in ATR would not be expected to result in noise impacts at INEEL for the reasons described in Section 4.4.1.1.2.

A new 76-meter (250-foot) stack would be required for neptunium-237 target processing at FMEF. Noise associated with construction of the new stack would be typical of small construction projects and would be of short duration. During neptunium-237 target processing operations, sound levels would be similar to those associated with other operations in the 400 Area. Thus, the change in overall onsite noise impacts would be

minimal. Offsite noise impacts from these operations would also be minor because the nearest site boundary is 7 kilometers (4.3 miles) to the east and changes in traffic volume going to and from FMEF would be small. There would be no loud noises associated with neptunium-237 target processing that would adversely impact wildlife.

4.4.3.1.3 Air Quality

It is estimated that there would be no measurable increases in nonradiological air pollutant emissions at INEEL associated with this option (Moor and Peterson 1999); therefore, no increased nonradiological air quality impacts would be expected.

The concentrations at Hanford attributable to this option are presented in **Table 4–75**. The concentrations for the option are based on a dispersion modeling screening analysis conducted with maximum expected emission rates and a set of worst-case meteorological conditions. Only those air pollutants expected to be emitted that have ambient air quality standards are presented in the table. The change in ambient concentrations were determined to be small, and would be below the applicable ambient air quality standards even when ambient monitoring values and the contributions from the other site activities are included.

Table 4–75 Incremental Hanford Concentrations[a] Associated with Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Criteria pollutants			
Nitrogen dioxide	Annual	100	4.43×10^{-5}
Sulfur dioxide	Annual	50	0.0087
	24 hours	260	0.069
	3 hours	1,300	0.16
	1 hour	660	0.17
Toxic air pollutants			
Methanol	24 hours	870	0.0018
Nitric acid	24 hours	17	0.0022
Paraffin hydrocarbons	24 hours	7	0.16
Tributyl phosphate	24 hours	7.3	0.090

a. For comparison with ambient air quality standards.
Source: 40 CFR Part 50; WDEC 1998; modeled increments are based on the SCREEN3 computer code (EPA 1995).

The concentrations at Hanford attributed to this option are compared to the Prevention of Significant Deterioration Class II increments for nitrogen dioxide and sulfur dioxide in **Table 4–76**.

Table 4–76 PSD Class II Increments Compared to Hanford Concentrations Associated with Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Pollutant	Averaging Period	Allowable PSD Increment (micrograms per cubic meter)	Modeled Increment (micrograms per cubic meter)
Nitrogen dioxide	Annual	25	4.43×10^{-5}
Sulfur dioxide	Annual	20	0.0087
	24 hours	91	0.069
	3 hours	512	0.16

Key: PSD, Prevention of Significant Deterioration.
Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995).

Health effects from hazardous chemicals associated with this option are addressed in Section 4.4.3.1.9. The air quality impacts of transportation among SRS, INEEL, Hanford, and LANL are presented in Section 4.4.3.1.11.

4.4.3.1.4 Water Resources

Impacts on water resources at INEEL associated with operating ATR to irradiate neptunium-237 targets would be negligible as previously described in Section 4.4.1.1.4.

Impacts on water resources at Hanford associated with the operation of FMEF for target material storage, target fabrication, and processing would be the same as those described in Section 4.3.3.1.4. Specifically, the operation of FMEF for this purpose is projected to require approximately 19 million liters (5 million gallons) of groundwater annually. This would include approximately 15 million liters (4 million gallons) per year to primarily support FMEF cooling needs, as well as material processing activities, and an additional 3.8 million liters (1 million gallons) per year for potable and sanitary water demands due to increased staffing. However, no impact on regional groundwater levels would be expected from increased withdrawals. FMEF groundwater usage would constitute an increase of about 10 percent over the 197 million liters (52 million gallons) withdrawn annually in the 400 Area during standby operations. Sanitary wastewater discharges from FMEF would also increase by roughly 3.8 million liters (1 million gallons) per year to the Energy Northwest treatment system, which has sufficient capacity. Also, the operation of FMEF for target fabrication and processing would generate approximately 15 million liters (4 million gallons) per year of process wastewater. This wastewater would be discharged to the 400 Area process sewer system and ultimately to the 400 Area Pond (i.e., 4608 B/C percolation ponds) (Chapin 2000; Nielsen 1999:38, 39, 41). Because discharges to the pond are regulated under State Waste Discharge Permit No. ST-4501 and there are no radiological liquid effluent pathways to the environment from FMEF, the impact on groundwater quality would be negligible.

It should be noted that the increase in water use and sanitary and process wastewater discharge for FMEF operations would essentially be negated by the larger reductions in water use and wastewater generation in the 400 Area associated with the permanent deactivation of FFTF (see Section 4.4.1.2.4).

Waste management aspects of this option and their effects are further discussed in Section 4.4.3.1.13.

4.4.3.1.5 Geology and Soils

The irradiation of neptunium-237 targets in ATR would not be expected to result in impacts on geologic or soil resources at INEEL, nor be jeopardized by large-scale geologic conditions, for the reasons described in Section 4.4.1.1.5.

Because the existing FMEF would be used for neptunium-237 storage, target fabrication, and processing and the new 76-meter (250-foot) stack would be built on previously disturbed land, impacts on geologic resources and native soils would be negligible. Hazards from large-scale geologic conditions at Hanford, such as earthquakes and volcanoes, were previously evaluated as discussed in Sections 4.2.4.2.5 and 4.3.3.1.5 and found to present a low risk to FMEF operations.

As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.3.1.6 Ecological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on ecological resources at INEEL for the reasons described in Section 4.4.1.1.6.

FMEF, an existing facility at Hanford, would be used for neptunium-237 target fabrication and processing. While a new 76-meter (250-foot) stack would be built, it would be placed on previously disturbed land in the 400 Area; thus, no natural terrestrial habitat would be lost. As noted in Section 4.4.3.1.2, there would be no sudden loud noises that would adversely impact wildlife. Because additional water usage and wastewater discharge would be small fractions of current values and discharge chemistry would not be expected to change, there would be no change in impacts on aquatic habitat or wetlands associated with the Columbia River (Section 4.4.3.1.4). Due to the developed nature of the area and the fact that construction would not disturb any natural habitat, impacts on threatened and endangered species would not occur.

Consultation letters concerning threatened and endangered species were sent to the U.S. Fish and Wildlife Service, the National Marine Fisheries Service, the Washington State Department of Natural Resources, and the State of Washington Department of Fish and Wildlife (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. Both the Washington State Department of Natural Resources and the State of Washington Department of Fish and Wildlife provided lists of state species of concern that occur in the vicinity of the project area. As noted above, no impacts to any threatened or endangered species are expected, including those of concern to these agencies. While DOE has made additional contacts with the U.S. Fish and Wildlife Service and the National Marine Fisheries Service, responses are pending from these agencies. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at Hanford prior to the receipt of input from these Federal agencies.

4.4.3.1.7 Cultural and Paleontological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on cultural and paleontological resources at INEEL for the reasons described in Section 4.4.1.1.7.

Neptunium-237 storage, target fabrication, and processing would take place at FMEF, which is in the 400 Area of Hanford. Although a new 76-meter (250-foot) stack would be built, it would be placed on previously disturbed land in the 400 Area; thus, impacts on cultural and paleontological resources would not be expected. No prehistoric, historic, or paleontological sites have been identified either in the 400 Area or within 2 kilometers (1.2 miles) of the 400 Area. Six buildings in the 400 Area have been determined to be eligible for listing on the National Register of Historic Places as contributing properties within the Historic District recommended for mitigation. The use of FMEF for neptunium-237 target fabrication and processing would not affect the eligibility of these structures for the National Register of Historic Places. No Native American resources are known to occur in the 400 Area.

Consultation to comply with Section 106 of the National Historic Preservation Act was conducted with the State Historic Preservation Office (see Table 5–3) and resulted in concurrence by the State Historic Preservation Office that the proposed action would have no effect on historic properties at Hanford. Consultation was also conducted with interested Native American tribes that resulted in comments at public hearings by members representing the Nez Perce and Confederated Tribes of the Umatilla Indian Reservation. Responses to their specific comments are addressed in Volume 3.

4.4.3.1.8 Socioeconomics

After facility modifications, startup, and testing of the plutonium-238 reactor operation facilities at INEEL and target fabrication/processing facilities at Hanford, approximately 62 additional workers would be required to operate these facilities (none at INEEL and 62 at Hanford) (Hoyt et al. 1999). This level of employment would not generate any indirect jobs in the region around INEEL. At Hanford, as this option would also include deactivation of FFTF, the additional workers could potentially transfer from FFTF. If not, this option could generate about 157 indirect jobs in the region around Hanford. The potential total employment increase of 219 direct and indirect jobs in the Hanford region represents less than 0.1 percent of the projected regional economic area workforce. It would have no noticeable impact on the regional economic area.

Additional employment resulting from this option would not have any noticeable impact on community services in the Hanford region of influence. Assuming that 91 percent of the new employment associated with this option would reside in Hanford's region of influence (refer to Section 3.4.8), 199 new jobs could increase the region's population by approximately 383 persons. This increase, in conjunction with normal population growth forecasted by the State of Washington, would not have any noticeable effect on the availability of housing and/or the price of housing in the region of influence. Given the current population-to-student ratio in the region of influence, this would likely result in an increase of about 79 students, requiring local school districts to slightly increase the number of classrooms to accommodate them.

Community services in the region of influence would be expected to change to accommodate the population growth as follows: five new teachers would be needed to maintain the current student-to-teacher ratio of 16:1; one new police officer would need to be added to maintain the current officer-to-population ratio of 1.5:1000; one new firefighter would need to be added to maintain the current firefighter-to-population ratio of 3.4:1000; and one new doctor would be added to maintain the current physician-to-population ratio of 1.4:1000. Thus, an additional eight positions would have to be created to maintain community services at current levels. Hospitals in the region of influence would not experience any change from the 2.1 beds per 1,000 persons currently available. Additionally, the average school enrollment would not change. None of these projected changes should have a major impact on the level of community services currently offered in the region of influence.

4.4.3.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with Alternative 2, Option 3 are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–77** for INEEL and Hanford: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

Table 4–77 Incremental Radiological Impacts on the Public Around INEEL and Hanford from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Receptor	INEEL ATR	Hanford FMEF	Total
Population within 80 kilometers (50 miles) in the year 2020			
Dose (person-rem)	0	4.4×10^{-5}	4.4×10^{-5}
35-year latent cancer fatalities	0	7.7×10^{-7}	7.7×10^{-7}
Maximally exposed individual			
Annual dose (millirem)	0	4.7×10^{-7}	NA[a]
35-year latent cancer fatality risk	0	8.3×10^{-12}	NA[a]
Average exposed individual within 80 kilometers (50 miles)			
Annual dose[b] (millirem)	0	8.9×10^{-8}	NA[a]
35-year latent cancer fatality risk	0	1.6×10^{-12}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FMEF in the year 2020 (494,400).

Key: NA, not applicable.
Source: Model results, using the GENII computer code (Napier et al. 1988).

As a result of annual operations of ATR at INEEL and FMEF at Hanford, the projected total incremental population dose in the year 2020 would be 4.4×10^{-5} person-rem. The corresponding number of latent cancer fatalities in the populations surrounding INEEL and Hanford from 35 years of operations would be 7.7×10^{-7}. The total incremental dose to the maximally exposed member of the public from annual ATR operations would be 0 millirem because there would be no increase in radiological releases to the environment from ATR associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would, therefore, be zero. The incremental dose to the maximally exposed member of the public from annual FMEF operations would be 4.7×10^{-7} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 8.3×10^{-12}.

Incremental doses to involved workers from normal operations are given in **Table 4–78**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to ATR workers would be 0 millirem; for FMEF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0 and approximately 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–78. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–78 Incremental Radiological Impacts on Involved INEEL and Hanford Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Receptor—Involved Workers[a]	INEEL ATR	Hanford FMEF	Total
Total dose (person-rem per year)	0	12[b]	12
35-year latent cancer fatalities	0	0.17	0.17
Average worker dose (millirem per year)	0	170	NA[c]
35-year latent cancer fatality risk	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers.
c. Values cannot be given for the average worker because the workers would be at two different facilities and sites.

Key: NA, not applicable.
Source: Mecham 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts at INEEL would be the same as those of ongoing site operations because no new chemicals would be emitted at ATR.

At Hanford, both carcinogenic and noncarcinogenic health effects from exposure to hazardous chemicals were evaluated. It was assumed that under normal operating conditions, the primary exposure pathway for members of the public would be from air emissions released through the process stack. Emissions of chemicals were estimated based on anticipated chemical usage. A worst-case dispersion modeling screening analysis was performed to estimate annual concentrations for each chemical, based on the emissions.

The annual concentration for each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the Hazard Quotient for each chemical. The Hazard Quotients were summed to give the Hazard Index from all noncarcinogenic chemicals associated with this option. A Hazard Index of less than one indicates that adverse health effects from non-cancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical. Hazardous chemical health effects are summarized in **Table 4–79**.

Table 4–79 Incremental Hazardous Chemical Impacts on the Public Around Hanford Under Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Chemical	Modeled Annual Increment (milligrams per cubic meter)	RfC - Inhalation (milligrams per cubic meter)	Unit Cancer Risk (risk per milligram per cubic meter)	Hazard Quotient	Cancer Risk
Diethyl benzene	6.01×10^{-6}	1	0.0078	6.01×10^{-6}	4.69×10^{-8}
Methanol	2.19×10^{-7}	1.75	NA	1.25×10^{-7}	NA
Nitric acid	2.73×10^{-7}	0.1225	NA	2.22×10^{-6}	NA
Tributyl phosphate	1.13×10^{-5}	0.01	NA	0.00113	NA
			Hazard Index =	0.00114	

Note: For diethyl benzene, the reference concentration for ethyl benzene and the unit cancer risk for benzene were used to estimate Hazard Quotient and cancer risk because no information was available for diethyl benzene. For tributyl phosphate, the reference concentration for phosphoric acid was used to estimate the Hazard Quotient because no information was available for tributyl phosphate.
Key: NA, not applicable (the chemical is not a known carcinogen); RfC, reference concentration.
Source: DOE 1996a; EPA 1999; modeled increments are based on the SCREEN3 computer code (EPA 1995).

4.4.3.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with ATR target irradiation and FMEF target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–80** and **4–81**, respectively. Because ATR is currently operating, the consequences and risks are presented for both the current reactor configuration without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configuration.

For 35 years of ATR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.45×10^{-7} and 3.48×10^{-6}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00140.

For 35 years of FMEF target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 2.88×10^{-6} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.112.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 2.88×10^{-6} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.114.

The irradiation of neptunium-237 targets to produce plutonium-238 at ATR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at ATR.

Table 4–80 ATR and FMEF Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
ATR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.465	2.33×10^{-4}	5.11×10^{4}	25.5	5.15	0.00206
Large-break LOCA with 5 kg/yr plutonium-238 production	0.604	3.02×10^{-4}	5.17×10^{4}	25.9	7.61	0.00304
Target handling with 0 kg/yr plutonium-238 production[c]	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 5 kg/yr plutonium-238 production	2.05×10^{-4}	1.03×10^{-7}	0.128	6.41×10^{-5}	0.00324	1.30×10^{-6}
FMEF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.02×10^{-9}	1.01×10^{-12}	7.26×10^{-5}	3.63×10^{-8}	6.65×10^{-10}	2.66×10^{-13}
Target dissolver tank failure during plutonium-238 separation	4.64×10^{-8}	2.32×10^{-11}	0.00169	8.47×10^{-7}	1.95×10^{-8}	7.81×10^{-12}
Ion exchange explosion during plutonium-238 separation	1.24×10^{-5}	6.18×10^{-9}	0.451	2.25×10^{-4}	5.20×10^{-6}	2.08×10^{-9}
Processing facility beyond-design-basis earthquake	16.5	0.00823	6.41×10^{5}	321	921	1.0[d]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. There would be no neptunium-237 targets for this zero-production case. Thus, there would be no associated accident consequences.
d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–81 ATR and FMEF Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual ATR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	2.33×10^{-8}	0.00255	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production (1×10^{-4})	3.02×10^{-8}	0.00259	3.04×10^{-7}
Large-break LOCA incremental risks[c]	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target handling with 5 kg/yr plutonium-238 production (0.001)[d]	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
35-year ATR risk[e]	2.45×10^{-7}	0.00140	3.48×10^{-6}
Annual FMEF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Processing facility beyond-design-basis earthquake (1×10^{-5})	8.23×10^{-8}	0.00321	1.00×10^{-5}[f]
35-year FMEF risk	2.88×10^{-6}	0.112	3.50×10^{-4}
35-year Option risk[g]	2.88×10^{-6}	0.114	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. There would be no neptunium-237 targets for the zero-production case. Thus, the 5-kg/yr production rate target-handling risks are the incremental risks.
e. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
f. Risk of an early fatality.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

No chemical processing activities are currently performed at FMEF and no chemicals are stored in this facility. Processing activities in support of plutonium-238 production would require the introduction of hazardous chemicals, specifically nitric acid and nitric oxide. Potential health impacts from accidental releases of nitric acid were assessed by comparing estimated airborne concentrations of the chemicals to ERPG developed by the American Industrial Hygiene Association. The ERPG-1 value (0.5 part per million) is the maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour, resulting in only mild, transient, and reversible adverse health effects. The ERPG-2 value (10 parts per million) is protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. The ERPG-3 value (25 parts per million) is indicative of potentially life-threatening health effects.

The maximum distances, in meters, needed to reach the ERPG values for nitric acid releases at FMEF for Stability Classes D and F are shown in **Table 4–82**. Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions while Stability Class F represents worst-case meteorological conditions. The number of involved and noninvolved workers potentially exposed would vary with a number of factors such as the time of day and whether they are sheltered within buildings at the time of release. Individuals at the nearest highway (7,100 meters [4.4 miles]) and at the nearest site boundary (7,210 meters [4.5 miles]) from FMEF would be exposed to levels well below ERPG-1.

Table 4–82 ERPG Distances for Nitric Acid Releases at FDPF Under Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Evaluation Parameter	Stability Class D (meters)	Stability Class F (meters)
ERPG-3	375	450
ERPG-2	500	600
ERPG-1	2,000	3,000

Note: To convert from meters to miles, multiply by 6.22×10^{-4}.
Key: ERPG, Emergency Response Planning Guideline.

There are no ERPG values for nitric oxide. For nitric oxide accidents, the level of concern has been estimated by using one-tenth of the "Immediately Dangerous to Life and Health" level published by the National Institute for Occupational Safety and Health. The Immediately Dangerous to Life and Health value for nitric oxide is 100 parts per million. The level of concern value used for this NI PEIS is 10 parts per million. The level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects as a result of a single exposure for a relatively short period of time.

For FMEF, the maximum distances needed to reach the level of concern for nitric oxide releases for Stability Classes D and F are 500 and 1,900 meters (0.31 and 1.18 miles), respectively. The number of involved and noninvolved workers potentially exposed would vary with a number of factors such as the time of day and whether they are sheltered within buildings at the time of release. Individuals at the nearest highway (7,100 meters [4.4 miles]) and at the nearest site boundary (7,210 meters [4.5 miles]) from FMEF would be exposed to levels well below the level of concern for nitric oxide.

Potential health impacts from the accidental release of the hazardous chemicals were assessed for a noninvolved worker, offsite individuals who are members of the public located at the nearest site boundary and onsite individuals who are members of the public located at the nearest highway access.

The impacts associated with the accidental release of nitric acid and nitric oxide at FMEF are presented in **Table 4–83**.

Table 4–83 FMEF Hazardous Chemical Accident Impacts Under Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Receptor	Evaluation Parameter	Nitric Acid		Nitric Oxide	
		Stability Class D	Stability Class F	Stability Class D	Stability Class F
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	3.3 <ERPG-2 Mild, transient	8.6 <ERPG-2 Mild, transient	4.2 <LOC Mild, transient	66 >LOC Serious
Nearest highway maximally exposed individual	Parts per million Level of concern Potential health effects	0.03 < ERPG-1 None	0.1 ERPG-1 Mild, transient	0.09 < LOC None	0.55 < LOC None
Site boundary maximally exposed individual	Parts per million Level of concern Potential health effects	0.03 < ERPG-1 None	0.1 ERPG-1 Mild, transient	0.09 < LOC None	0.53 < LOC None

Note: < means "less than."
Key: ERPG, Emergency Response Planning Guideline; LOC, level of concern.
Source: Model results.

4.4.3.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FMEF target fabrication facility at Hanford. DOE would transport the unirradiated neptunium-237 targets from FMEF to ATR at INEEL. Following irradiation in ATR, the targets would be returned to FMEF for processing. After this processing, the plutonium-238 product would be shipped to LANL. The analysis is described in Appendix J.

Approximately 689 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 0.83 million kilometers (0.52 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 5 person-rem; the dose to the public, 81 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0020 latent cancer fatality among transportation workers and 0.040 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0014.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FMEF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 were also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.06 person-rem, resulting in 3.0×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.017 traffic fatality.

4.4.3.1.12 Environmental Justice

NORMAL OPERATIONS. For 35 years of normal operations under this option, the likelihood of an incremental latent cancer fatality among the populations residing within 80 kilometers (50 miles) of ATR and FMEF would be essentially zero (derived from information in Table 4–77). As shown in Table 4–79, the release of hazardous chemicals at Hanford would pose no significant risk of cancer or toxic effects among the public. As discussed in Sections K.5.1 and K.5.3, the number of latent cancer fatalities that would result from the ingestion of food that could be radiologically contaminated due to normal operations would be essentially zero at INEEL and approximately 0.001 at Hanford. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near INEEL or Hanford. As discussed in Section 4.4.3.1.11, no fatalities would be expected from incident-free transportation activities.

ACCIDENTS. The number of expected latent cancer fatalities among the populations at risk due to radiological accidents listed in Table 4–81 would be approximately 0.11. If a radiological accident were to occur at ATR and northwesterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Fort Hall Indian Reservation (see Figure K–2). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the population or maximally exposed individual residing within the boundary of the Fort Hall Indian Reservation. If a radiological accident were to occur at FMEF and northeasterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Yakama Indian Reservation (see Figure K–11). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the population or maximally exposed individual residing within the boundary of Yakama Indian Reservation.

As discussed in Section 4.4.3.1.11, no fatalities would be expected to result from transportation accidents. In summary, the implementation of this option would pose no significant radiological risk to persons residing in potentially affected areas or along representative transportation routes. Under the conservative assumption that all food consumed in potentially affected areas during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to the ingestion of contaminated food supplies. As discussed in other parts of Section 4.4.3.1, the implementation of this option would not result in significant nonradiological impacts on populations at risk. Thus, implementation would not pose significant and adverse environmental risks to persons residing within potentially affected areas, including minority and low-income persons.

4.4.3.1.13 Waste Management

Only an extremely small amount of additional waste would be generated as a result of irradiating neptunium-237 targets in ATR (Section 4.4.1.1.13). Therefore, no impacts on the waste management systems at INEEL would be anticipated. However, there would be impacts on Hanford's waste management systems as a result of FMEF operations to fabricate and process neptunium-237 targets for plutonium-238 production.

The expected generation rates of waste at Hanford that would be associated with the operation of FMEF for this target fabrication and processing are compared with Hanford's treatment, storage, and disposal capacities in **Table 4–84**. The impacts on the Hanford waste management systems, in terms of managing the additional waste, are discussed in this section. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.4.3.1.9 through 4.4.3.1.11.

Table 4–84 Incremental Waste Management Impacts of Operating FMEF at Hanford Under Alternative 2 (Use Only Existing Operational Facilities)—Option 3

Waste Type[a]	Estimated Additional Waste Generation (cubic meters per year)	Estimated Additional Waste Generation as a Percent of[b]		
		Onsite Treatment Capacity	Onsite Storage Capacity	Onsite Disposal Capacity
Transuranic/High-level radioactive[c]	11	(c)	(c)	NA
Low-level radioactive				
Liquid	6	(d)	(d)	(d)
Solid	54	NA	NA	0.82
Mixed low-level radioactive	<5	0.27	1.0	1.2
Hazardous	18	NA	NA	NA
Nonhazardous				
Process wastewater	15,000	(d)	(d)	(d)
Sanitary wastewater	3,800	1.6[e]	NA	NA
Solid	150	NA	NA	NA

a. See definitions in Section G.9.
b. The estimated additional amounts of waste generated annually are compared with the annual site treatment capacities. For nonhazardous liquid waste, the estimated additional annual generation rate is also compared with the annual site disposal capacity. The estimated total amounts of additional waste generated over the 35-year operational period are compared with the site's storage capacities, and, for other than nonhazardous liquid waste, with the site's disposal capacities.
c. Refer to the text for a discussion on waste classification and treatment. This waste would be stored at FMEF pending availability of a suitable repository. It is assumed that this waste would be remotely handled.
d. Refer to the text.
e. Percent of capacity of the Energy Northwest system.

Note: To convert from cubic meters per year to cubic yards per year, multiply by 1.308; < means "less than."
Key: NA, not applicable (i.e., the majority of this waste is not routinely treated or is not routinely stored or is not routinely disposed of on site; refer to the text).
Source: Chapin 2000; DOE 2000a; Hoyt et al. 1999; Nielsen 1999.

The canisters used to transport neptunium-237 to the site would constitute a very small additional amount of solid low-level radioactive waste—less than 10 cubic meters (13.1 cubic yards) over the 35-year operational period, even if no credit is taken for volume reduction by compaction (Brunson 1999a). The annual generation of this waste would fall within the range of accuracy of the solid low-level radioactive waste generation rate given in Table 4–84, and its management need not be addressed separately.

In accordance with the Records of Decision for the *Waste Management PEIS* (DOE 1997a), waste could be treated and disposed of on site at Hanford or at other DOE sites or commercial facilities. Based on the Record of Decision for high-level radioactive waste issued on August 12, 1999 (64 FR 46661), immobilized high-level radioactive waste would be stored on site until transfer to a geologic repository. Based on the Record of Decision for transuranic waste issued on January 20, 1998 (63 FR 3629), transuranic waste would be certified on site and eventually shipped to a suitable geologic repository for disposal. Based on the Record of Decision for hazardous waste issued on August 5, 1998 (63 FR 41810), nonwastewater hazardous waste would continue to be treated and disposed of at offsite commercial facilities. Based on the Record of Decision for low-level radioactive waste and mixed low-level radioactive waste issued on February 18, 2000 (65 FR 10061), minimal treatment of low-level radioactive waste will be performed at all sites and, to the extent practicable, onsite disposal of low-level radioactive waste will continue. Hanford and the Nevada Test Site will be made available to all DOE sites for disposal of low-level radioactive waste. Mixed low-level radioactive waste analyzed in the *Waste Management PEIS* will be treated at Hanford, INEEL, ORR, and SRS and will be disposed of at Hanford and the Nevada Test Site.

The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. Irrespective of how the waste is classified (i.e., transuranic or high-level radioactive waste), the composition and characteristics are the same, and the waste management activities (i.e., treatment and onsite storage) as described in this NI PEIS would be the same. In addition, either waste type would require disposal in a suitable repository. If it is transuranic waste, it would be nondefense waste and could not be disposed of at WIPP under current law. Because nondefense transuranic waste has no current disposal path, DOE Headquarters' approval would be necessary before a decision were made to generate such waste, as required by DOE Order 435.1. If the waste is classified as high-level radioactive waste, it is assumed for the purposes of this analysis that Yucca Mountain, Nevada, if approved, would be the final disposal site for DOE's high-level radioactive waste. The other differences between these two waste classifications are that a high-level radioactive waste repository requires a much more rigorous waste-form qualification process than a transuranic waste repository and there is a slightly different set of requirements for high-level radioactive waste than for transuranic waste delineated in DOE Manual 435.1.

Target fabrication and processing in FMEF would generate a total of 385 cubic meters (504 cubic yards) of transuranic or high-level radioactive waste over the 35-year operational period. As described in Section 3.4.5 of the *Preconceptual Design Planning for Chemical Processing to Support Pu-238 Production* (Wham 1998), the waste would be vitrified into a glass matrix at a glass melter installed within FMEF. The resulting glass matrix would be stored at FMEF pending availability of a repository for permanent disposal. The impacts of managing the additional quantities of this waste at Hanford would be minimal.

Solid low-level radioactive waste would be packaged, certified, and accumulated at FMEF before transfer for additional treatment and disposal in the existing onsite low-level radioactive Burial Grounds. Neptunium-237 target fabrication and processing would generate 1,890 cubic meters (2,470 cubic yards) of low-level radioactive waste over the 35-year operational period. This amount of low-level radioactive waste represents approximately 0.11 percent of the 1.74 million-cubic-meter (2.28 million-cubic-yard) capacity of the low-level radioactive waste Burial Grounds and 0.82 percent of the 230,000-cubic-meter (301,000-cubic-yard) capacity of the Grout Vaults. Using the 3,480-cubic-meter-per-hectare (1,842-cubic-yard-per-acre) disposal land usage factor for Hanford published in the *Storage and Disposition PEIS* (DOE 1996a:E-9), 1,890 cubic meters (2,470 cubic yards) of waste would require 0.54 hectares (1.3 acres) of disposal space at Hanford. The impacts of managing this additional low-level radioactive waste at Hanford would be minimal.

Liquid low-level radioactive waste associated with target fabrication and processing at FMEF would be transported to the 200 Area Effluent Treatment Facility for processing and ultimate disposal. Target fabrication and processing at FMEF would generate about 210 cubic meters (270 cubic yards) of liquid low-level radioactive waste over the 35-year operational period. This total amount of additional liquid low-level radioactive waste represents a small amount of waste which can be managed by the 200 Area Liquid Effluent Treatment Facility with an operating capacity of 0.57 cubic meter (0.75 cubic yard) per minute.

Mixed low-level radioactive waste would be stabilized, packaged, and stored on site for treatment and disposal in a manner consistent with the Tri-Party Agreement (EPA et al. 1989) for Hanford. Over the 35-year operational period, 175 cubic meters (229 cubic yards) of mixed low-level radioactive waste would be generated at FMEF associated with neptunium-237 target fabrication and processing. This mixed low-level radioactive waste is expected to be treated at a nearby commercial facility. However, if this waste were treated on site, it is estimated to be 0.27 percent of the 1,820-cubic-meter-per-year (2,380-cubic-yard-per-year) capacity of the Waste Receiving and Processing Facility. This waste also represents 1.0 percent of the 16,800-cubic-meter (22,000-cubic-yard) storage capacity of the Central Waste Complex and 1.2 percent of the

14,200-cubic-meter (18,600-cubic-yard) planned disposal capacity of the Radioactive Mixed Waste Disposal Facility. Therefore, this additional waste would only have a minimal impact on the management of mixed low-level radioactive waste at Hanford.

Hazardous waste generated during operation would be packaged in DOT-approved containers and shipped off site to permitted commercial recycling, treatment, and disposal facilities. The additional waste load generated during the operational period would have only a minimal impact on the Hanford hazardous waste management system.

Nonhazardous solid waste would be packaged and transported in conformance with standard industrial practice. Solid waste such as office paper, metal cans, and plastic and glass bottles that can be recycled would be sent off site for that purpose. The remaining solid sanitary waste would be sent for offsite disposal. This additional waste load would have only a minimal impact on the nonhazardous solid waste management system at Hanford.

Nonhazardous process wastewater would be discharged into the 400 Area Ponds. This discharge is regulated by State Waste Discharge Permit ST-4501.

Nonhazardous sanitary wastewater would be discharged to the 400 Area sanitary sewer system, which connects to the Energy Northwest Sewage Treatment Facility. Nonhazardous sanitary wastewater generated from neptunium-237 target fabrication and processing in FMEF would represent 1.6 percent of the 235,000-cubic-meter-per-year (307,000-cubic-yard-per-year) capacity of the Energy Northwest Sewage Treatment Facility and would be well within the 138,000-cubic-meter-per-year (181,000-cubic-yard-per-year) excess capacity of this facility (DOE 1999a). Management of nonhazardous liquid waste at Hanford would only have a minimal impact on the treatment system.

The generation rates of waste at Hanford that would be associated with this option (refer to Table 4–84) can be compared with the current waste generation rates at the site, given in Table 3–34 (Section 3.4.11). The waste generation rates associated with plutonium-238 production would be much smaller than the current waste generation rates at the site.

4.4.3.1.14 Spent Nuclear Fuel Management

No incremental impacts would be associated with the management of spent nuclear fuel (refer to Section 4.4.1.1.14).

4.4.3.2 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.4.4 Alternative 2 (Use Only Existing Operational Facilities)—Option 4

Option 4 involves operating a CLWR at an unspecified location to irradiate neptunium-237 targets, and operating the REDC facility at ORR to fabricate and process these targets and to store the neptunium-237 transported to ORR from SRS.

The transportation of the neptunium-237 from SRS to ORR for processing and fabrication into neptunium-237 targets in REDC, the transportation of the targets from ORR to the generic CLWR site for irradiation, the transportation of the irradiated targets back to ORR for postirradiation processing in REDC, and the transportation of the plutonium-238 product from ORR to LANL also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.4.1 Operations and Transportation

Environmental impacts associated with storage, processing, and irradiation operations and with all transportation activities are assessed in this section.

4.4.4.1.1 Land Resources

LAND USE. A currently operating CLWR would be used to irradiate neptunium-237 targets. There would be no impacts on land use because no new construction would be required, and use of the facility for target irradiation would be compatible with is current function.

There would be no impacts on land use at ORR from neptunium-237 storage, target fabrication, and processing at REDC for the reasons described in Section 4.4.1.1.1.

VISUAL RESOURCES. There would be no impacts on visual resources because use of a CLWR for neptunium-237 target irradiation would not require any external modifications that would alter the appearance of the facility.

There would be no impacts on visual resources at ORR from neptunium-237 storage, target fabrication, and processing at REDC for the reasons described in Section 4.4.1.1.1.

4.4.4.1.2 Noise

Noise associated with the irradiation of neptunium-237 targets at a CLWR site would be indistinguishable from other noises generated during normal operation of the facility. Noise associated with increased traffic going to and from the facility would be low and would result in only minor changes to existing onsite and offsite noise levels. Neptunium-237 target irradiation in a CLWR would not produce any sudden loud noises that would adversely affect wildlife.

Noise impacts at ORR would be minimal from neptunium-237 storage, target fabrication, and processing at REDC and changes in traffic noise would be minimal for the reasons described in Section 4.4.1.1.2.

4.4.4.1.3 Air Quality

It is expected that there would be no measurable increases in nonradiological air pollutant emissions at a CLWR site associated with this option; therefore, no changes in nonradiological air quality impacts would be expected.

Impacts for this option at ORR would be the same as those described for Option 1 (Section 4.4.1.1.3).

The air quality impacts of transportation among SRS, the generic site, ORR, and LANL are presented in Section 4.4.4.1.11.

4.4.4.1.4 Water Resources

No measurable impact on water resources at a CLWR site is expected under this option, because neptunium-237 target irradiation would not measurably increase water use or change the quantity or quality of effluent discharges. Information on water resources for the generic CLWR site is presented in Section 3.5.4.

Impacts for this option at ORR would be substantially the same as described for Option 1 (Section 4.4.1.1.4).

4.4.4.1.5 Geology and Soils

This option involves the irradiation of neptunium-237 targets in a CLWR. Because no new construction would take place, geologic and soil resources within the site area would not be disturbed. Assessment of hazards from large-scale geologic conditions for reactor sites, including assessment of seismic and nonseismic features, is governed by 10 CFR Part 100 and is beyond the scope of this analysis. Information on geology and soils for the generic CLWR site is presented in Section 3.5.5.

Neptunium-237 storage, target fabrication, and processing at REDC would not be expected to impact geologic and soil resources at ORR, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.2.2.5 and 4.4.1.1.5. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.4.1.6 Ecological Resources

A currently operating CLWR would be used to irradiate neptunium-237 targets. Terrestrial resources and wetlands would not be adversely affected because no new construction would be required. Further, as noted in Section 4.4.4.1.2, there would be no loud noises that would adversely affect wildlife. The irradiation of neptunium-237 targets would not impact aquatic resources because there would be no measurable change in water withdrawal or wastewater discharge (Section 4.4.4.1.4). Threatened and endangered species would not be impacted for the reasons noted above.

Impacts on ecological resources at ORR would not result from neptunium-237 storage, target fabrication, and processing at REDC for the reasons described in Section 4.4.1.1.6.

4.4.4.1.7 Cultural and Paleontological Resources

The irradiation of neptunium-237 targets would take place in a currently operating CLWR. Because no new construction would take place, impacts on cultural and paleontological resources would not occur.

Impacts on cultural and paleontological resources at ORR would not result from neptunium-237 storage, target fabrication, and processing at REDC for the reasons described in Section 4.4.1.1.7.

4.4.4.1.8 Socioeconomics

Reactor operations at a CLWR site would not require additional workers. Target fabrication and processing of plutonium-238 at ORR would require approximately 41 additional workers (Wham et al. 1998). The socioeconomic impacts at ORR are the same as those addressed in Section 4.3.1.1.8.

4.4.4.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–85** for the generic CLWR site and ORR: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–85 Incremental Radiological Impacts on the Public Around the Generic CLWR Site and ORR from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 4

Receptor	Generic CLWR	ORR REDC	Total
Population within 80 kilometers (50 miles) in the year 2020			
Dose (person-rem)	0	8.8×10^{-5}	8.8×10^{-5}
35-year latent cancer fatalities	0	1.5×10^{-6}	1.5×10^{-6}
Maximally exposed individual			
Annual dose (millirem)	0	1.9×10^{-6}	NA[a]
35-year latent cancer fatality risk	0	3.3×10^{-11}	NA[a]
Average exposed individual within 80 kilometers (50 miles)			
Annual dose[b] (millirem)	0	7.8×10^{-8}	NA[a]
35-year latent cancer fatality risk	0	1.4×10^{-12}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of REDC in the year 2020 (1,134,200).

Key: NA, not applicable.
Source: Model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

Target irradiation in a CLWR would not result in any incremental radiological emissions during normal operations or increased worker exposures. Therefore, the incremental impact of CLWR target irradiation is zero.

As a result of annual operations of the generic CLWR and REDC, the projected total incremental population dose in the year 2020 would be 8.8×10^{-5} person-rem. The corresponding number of latent cancer fatalities in the populations surrounding the generic CLWR site and ORR from 35 years of operations would be 1.5×10^{-6}. The total incremental dose to the maximally exposed member of the public from annual generic CLWR operations would be 0 millirem because there would be no increase in radiological releases to the environment from the generic CLWR associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would, therefore, be zero. The incremental dose to the maximally exposed member of the public from annual REDC operations would be 1.9×10^{-6} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 3.3×10^{-11}.

Incremental doses to involved workers from normal operations are given in **Table 4–86**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to generic CLWR workers would be 0 millirem; for REDC workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0 and approximately 12 person-rem, respectively. The risks and numbers of latent

cancer fatalities among the different workers from 35 years of operations are included in Table 4–86. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–86 Incremental Radiological Impacts on Involved CLWR and ORR Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 4

Receptor—Involved Workers[a]	Generic CLWR	ORR REDC	Total
Total dose (person-rem per year)	0	12[b]	12
35-year latent cancer fatalities	0	0.17	0.17
Average worker dose (millirem per year)	0	170	NA[c]
35-year latent cancer fatality risk	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with REDC operations at a DOE facility would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced at all facilities.
b. Based on an estimated 75 badged workers.
c. Values cannot be given for the average worker because the workers would be at two different facilities and sites.
Key: NA, not applicable.
Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts at the generic CLWR site would be the same as those of ongoing site operations because no new chemicals would be emitted.

Hazardous chemical impacts for this option at ORR were determined to be the same as described for Option 1 (Section 4.4.1.1.9).

4.4.4.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with target irradiation in a generic CLWR and REDC target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual and the offsite population within 80 kilometers (50 miles) of the facility. Consequences to a noninvolved worker are not included for the generic CLWR analysis. Details regarding the exclusion of a noninvolved worker are provided in Section I.1.2.

Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality for an individual and as the increased number of latent cancer fatalities in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are plant specific,

summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–87** and **4–88**, respectively. Certain extremely unlikely or incredible severe accidents at commercial nuclear reactors could result in doses sufficiently high to cause early fatalities. The early fatality consequences and risks are presented in **Table 4–89**. The early fatalities shown in Table 4–89 are considered to be conservative estimates based upon the assumption that some individuals very close to the reactor do not evacuate. Because the generic CLWR is operational, the consequences and risks are presented for both the current reactor configuration without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configuration. Baseline accident risks attributed to generic CLWR operations refer to accidents that could occur under the current CLWR configuration (without neptunium-237 targets). Baseline accident risks are obtained from the data in Tables 4–88 and 4–89 by summing the annual risks for the baseline CLWR configuration (0 kilograms per year plutonium-238 production), and then multiplying the sum by 35. The baseline CLWR accident risk to the public would be 0.073 latent cancer fatality. Baseline CLWR accident risks to the maximally exposed offsite individual would be 5.7×10^{-5} latent cancer fatality. Baseline risk to noninvolved workers is discussed in Appendix I.

For 35 years of CLWR target fabrication and irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual would be 1.93×10^{-9}. The increased number of latent cancer fatalities in the surrounding population would be 0.00305. The increased risk of an early fatality in the surrounding population would be 2.07×10^{-6}.

For 35 years of REDC target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.157.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.160.

The irradiation of neptunium-237 targets at the generic CLWR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at the generic CLWR.

Processing associated with the plutonium-238 production program at REDC, including storage of neptunium-237 and plutonium-238, neptunium-237 target fabrication, postirradiation processing to extract plutonium-238 and to recycle the unconverted neptunium-237 into new targets, does not require the introduction of hazardous chemicals that are not in current use in the facility. The quantities of in-process hazardous chemicals for the plutonium-238 production program are bounded by the quantities of the material currently stored in the facility. The impacts of in-process hazardous chemical accidents associated with the plutonium-238 production are bounded by the impacts of hazardous chemical accidents for existing storage facilities at REDC.

Table 4–87 Generic CLWR and REDC Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 4

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
Generic CLWR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.0312	1.56×10^{-5}	186	0.0931	NA[c]	NA
Large-break LOCA with 5 kg/yr plutonium-238 production	0.0313	1.57×10^{-5}	187	0.0935	NA	NA
Early containment failure with 0 kg/yr plutonium-238 production	3,350	1.00[d]	1.80×10^{6}	1,250	NA	NA
Early containment failure with 5 kg/yr plutonium-238 production	3,670	1.00[d]	1.90×10^{6}	1,340	NA	NA
Late containment failure with 0 kg/yr plutonium-238 production	1.11	5.55×10^{-4}	1.06×10^{5}	53.6	NA	NA
Late containment failure with 5 kg/yr plutonium-238 production	1.12	5.60×10^{-4}	1.06×10^{5}	53.6	NA	NA
Containment bypass with 0 kg/yr plutonium-238 production	1,540	1.00[d]	1.45×10^{6}	922	NA	NA
Containment bypass with 5 kg/yr plutonium-238 production	1,680	1.00[d]	1.52×10^{6}	978	NA	NA
REDC accidents						
Ion exchange explosion during neptunium-237 target fabrication	6.13×10^{-9}	3.06×10^{-12}	8.58×10^{-5}	4.29×10^{-8}	5.60×10^{-10}	2.24×10^{-13}
Target dissolver tank failure during plutonium-238 separation	1.76×10^{-7}	8.79×10^{-11}	0.00196	9.82×10^{-7}	1.69×10^{-8}	6.74×10^{-12}
Ion exchange explosion during plutonium-238 separation	4.68×10^{-4}	2.34×10^{-7}	5.23	0.00261	4.49×10^{-5}	1.79×10^{-8}
Processing facility beyond-design-basis earthquake	163	0.163	8.91×10^{5}	445	1,310	1.00[d]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities. The MACCS2 computer code calculates the dose to each exposed individual in the population, applies the appropriate cancer risk factor, and then sums the individual probabilities to determine the number of latent cancer fatalities.
c. Not applicable (refer to Appendix I). Evacuation of noninvolved workers and other noninvolved worker issues are addressed in Appendix I.
d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident; NA, not applicable.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–88 Generic CLWR and REDC Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 4

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual generic CLWR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (4.65×10^{-5})	7.25×10^{-10}	4.33×10^{-6}	NA[c]
Large-break LOCA with 5 kg/yr plutonium-238 production (4.65×10^{-5})	7.30×10^{-10}	4.35×10^{-6}	NA
Large-break LOCA incremental risks[d]	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure with 0 kg/yr plutonium-238 production (7.92×10^{-8})	$7.92\times10^{-8(e)}$	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production (7.92×10^{-8})	$7.92\times10^{-8(e)}$	1.06×10^{-4}	NA
Early containment failure incremental risks	0.0	7.10×10^{-6}	NA
Late containment failure with 0 kg/yr plutonium-238 production (1.07×10^{-5})	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production (1.07×10^{-5})	5.99×10^{-9}	5.74×10^{-4}	NA
Late containment failure incremental risks	5.00×10^{-11}	0.00	NA
Containment bypass with 0 kg/yr plutonium-238 production (1.53×10^{-6})	$1.53\times10^{-6(e)}$	0.00141	NA
Containment bypass with 5 kg/yr plutonium-238 production (1.53×10^{-6})	$1.53\times10^{-6(e)}$	0.00149	NA
Containment bypass incremental risks	0.0	8.00×10^{-5}	NA
35-year CLWR risk[f]	1.93×10^{-9}	0.00305	NA
Annual REDC risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Processing facility beyond-design-basis earthquake (1×10^{-5})	1.63×10^{-6}	0.00445	$1.00\times10^{-5(e)}$
35-year REDC risk	5.71×10^{-5}	0.157	3.50×10^{-4}
35-year Option risk[g]	5.71×10^{-5}	0.160	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.

c. Not applicable (refer to Appendix I). Evacuation of noninvolved workers and other noninvolved worker issues are addressed in Appendix I.
d. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
e. Risk of an early fatality.
f. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident; NA, not applicable.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–89 Generic CLWR Early Fatalities and Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 4

Accident (Frequency)	Population to 80 Kilometers (50 Miles)	
	Early Fatalities[a]	Annual Risk[b]
Annual generic CLWR risks		
Early containment failure with 0 kg/yr plutonium-238 production (7.92×10^{-8})	8.65	6.85×10^{-7}
Early containment failure with 5 kg/yr plutonium-238 production (7.92×10^{-8})	8.76	6.94×10^{-7}
Early containment failure incremental risk[c]	NA	9.00×10^{-9}
Containment bypass with 0 kg/yr plutonium-238 production (1.53×10^{-6})	3.48	5.32×10^{-6}
Containment bypass with 5 kg/yr plutonium-238 production (1.53×10^{-6})	3.51	5.37×10^{-6}
Containment bypass incremental risk	NA	5.00×10^{-8}
35-year CLWR risk[d]	NA	2.07×10^{-6}

a. Number of early fatalities assuming that the accident has occurred.
b. Risk of an early fatality.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; NA, not applicable.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

4.4.4.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to a CLWR. Following irradiation in a CLWR, the targets would be returned to REDC for processing. After this processing, the plutonium-238 product would be shipped to LANL. The impact analysis, described in Appendix J, assumes the most distant CLWR is used for target irradiation.

Approximately 689 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 2.6 million kilometers (1.6 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 14 person-rem; the dose to the public, 299 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.006 latent cancer fatality among transportation workers and 0.15 latent cancer fatality in the total affected

population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0056.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to REDC with a severity category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 were also evaluated and estimated to have a probability of lower than 1 in 10 million per year.

Estimates of the total ground transportation accident risks are as follows: a radiological dose to the population of 0.088 person-rem, resulting in 4.4×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.074 traffic fatality.

4.4.4.1.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in a CLWR at an unspecified site. Target fabrication and processing would be performed at REDC located at ORR. Activities at REDC were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.1.1.12) and found to pose no significant radiological or other risks to minority and low-income populations. The analysis of accidents at specific sites shows that accidents at the fabrication and target facilities would result in radiological risks to the public that are small, but which are several orders of magnitude larger than those that would result from accidents at specific reactor sites (see Section 2.7.1.1). It is plausible that a similar difference would exist between accident risks at an unspecified CLWR site and the fabrication and processing facilities. However, evaluations of environmental justice are necessarily site specific and cannot be performed for unspecified locations. In the event that this option were selected for implementation and a specific CLWR were selected for irradiation services, additional evaluation of environmental justice at the CLWR site would be performed prior to implementation.

4.4.4.1.13 Waste Management

There would be no change in the amounts of waste generated as the result of irradiating neptunium-237 targets in a CLWR. Thus, there would be no impact on the CLWR site's waste management systems as the result of target irradiation.

The impacts of managing waste associated with neptunium-237 target fabrication and processing in REDC are assumed to be the same as for Option 1 under Alternative 1 (Section 4.3.1.1.13) because the same amount of plutonium-238 would be produced annually. As shown in that section, the impacts on the waste management systems at ORR would be minimal.

4.4.4.1.14 Spent Nuclear Fuel Management

No incremental impacts would be associated with the management of spent nuclear fuel (refer to Section 4.4.1.1.14).

4.4.4.2 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.4.5 Alternative 2 (Use Only Existing Operational Facilities)—Option 5

This option involves operating a generic CLWR at a generic site to irradiate neptunium-237 targets, and operating FDPF at INEEL to fabricate and process these targets. This option also includes storage of the neptunium-237 transported to INEEL from SRS, in either Building CPP–651 or FDPF.

The transportation of the neptunium-237 from SRS to INEEL for processing and fabrication into neptunium-237 targets in FDPF, the transportation of the targets from INEEL to the generic CLWR site for irradiation in the CLWR, the transportation of the irradiated targets back to INEEL for postirradiation processing in FDPF, and the transportation of the plutonium-238 product from INEEL to LANL also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.5.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all intersite transportation activities, are assessed in this section.

4.4.5.1.1 Land Resources

LAND USE. The use of a CLWR to irradiate neptunium-237 targets would not be expected to result in impacts on land use for the reasons described in Section 4.4.4.1.1.

Impacts on land use at INEEL from neptunium-237 storage, target fabrication, and processing would not result for the reasons described in Section 4.4.2.1.1.

VISUAL RESOURCES. The use of a CLWR to irradiate neptunium-237 targets would not result in impacts on visual resources for the reasons described in Section 4.4.4.1.1.

Impacts on visual resources at INEEL from neptunium-237 storage, target fabrication, and processing would not result for the reasons described in Section 4.4.2.1.1.

4.4.5.1.2 Noise

The irradiation of neptunium-237 targets in a CLWR would not be expected to result in noise impacts for the reasons described in Section 4.4.4.1.2.

Noise impacts at INEEL would not be expected from neptunium-237 storage, target fabrication, and processing and changes in traffic noise would be small for the reasons described in Section 4.4.2.1.2.

4.4.5.1.3 Air Quality

Impacts for this option at a generic CLWR site would be the same as those described for Option 4 (Section 4.4.4.1.3).

Impacts for this option at INEEL would be the same as those described for Option 2 (Section 4.4.2.1.3).

The air quality impacts of transportation among SRS, the generic CLWR site, INEEL, and LANL are presented in Section 4.4.5.1.11.

4.4.5.1.4 Water Resources

Impacts for this option at a generic CLWR site would be negligible as described for Option 4 (Section 4.4.4.1.4).

Impacts for this option at INEEL would be the same as described for Option 2 (Section 4.4.2.1.4).

4.4.5.1.5 Geology and Soils

The irradiation of neptunium-237 targets in a CLWR would not be expected to result in impacts on geologic or soil resources for the reasons described in Section 4.4.4.1.5. Assessment of hazards from large-scale geologic conditions for reactor sites, including assessment of seismic and nonseismic features, is governed by 10 CFR 100 and is beyond the scope of this analysis.

Neptunium-237 storage, target fabrication, and processing would not be expected to impact geologic and soil resources at INEEL, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.3.2.5 and 4.4.2.1.5. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.5.1.6 Ecological Resources

The irradiation of neptunium-237 targets in a CLWR would not result in impacts on ecological resources for the reasons described in Section 4.4.4.1.6.

Impacts on ecological resources at INEEL would not result from neptunium-237 storage, target fabrication, and processing for the reasons described in Section 4.4.2.1.6.

4.4.5.1.7 Cultural and Paleontological Resources

The irradiation of neptunium-237 targets in a CLWR would not result in impacts on cultural and paleontological resources for the reasons described in Section 4.4.4.1.7.

Impacts on cultural and paleontological resources at INEEL would not result from neptunium-237 storage, target fabrication, and processing for the reasons described in Section 4.4.2.1.7.

4.4.5.1.8 Socioeconomics

Reactor operations at a CLWR site would not require additional workers. Target fabrication and processing of plutonium-238 at INEEL would require approximately 24 additional workers (Hill et al. 1999). The socioeconomic impacts at INEEL are the same as those assessed in Section 4.3.2.1.8.

4.4.5.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–90** for the generic CLWR site and INEEL: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–90 Incremental Radiological Impacts on the Public Around the Generic CLWR Site and INEEL from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 5

Receptor	Generic CLWR	INEEL FDPF	Total
Population within 80 kilometers (50 miles) in the year 2020			
Dose (person-rem)	0	3.9×10^{-6}	3.9×10^{-6}
35-year latent cancer fatalities	0	6.7×10^{-8}	6.7×10^{-8}
Maximally exposed individual			
Annual dose (millirem)	0	2.6×10^{-7}	NA[a]
35-year latent cancer fatality risk	0	4.6×10^{-12}	NA[a]
Average exposed individual within 80 kilometers (50 miles)			
Annual dose[b] (millirem)	0	2.0×10^{-8}	NA[a]
35-year latent cancer fatality risk	0	3.6×10^{-13}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FDPF in the year 2020 (188,400).

Key: NA, not applicable.
Source: Model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

Target irradiation in a CLWR would not result in any increased radiological emissions during normal operations or increased worker exposure. Therefore, the incremental impact of CLWR target irradiation is zero.

As a result of annual operations of the generic CLWR and FDPF, the projected total incremental population dose in the year 2020 would be 3.9×10^{-6} person-rem. The corresponding number of latent cancer fatalities in the populations surrounding the generic CLWR site and INEEL from 35 years of operations would be 6.7×10^{-8}. The total incremental dose to the maximally exposed member of the public from annual generic CLWR operations would be 0 millirem because there would be no increase in radiological releases to the environment from the generic CLWR associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would, therefore, be zero. The incremental dose

to the maximally exposed member of the public from annual FDPF operations would be 2.6×10^{-7} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 4.6×10^{-12}.

Incremental doses to involved workers from normal operations are given in **Table 4–91**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to CLWR workers would be 0 millirem; for FDPF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0 and approximately 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–91. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–91 Incremental Radiological Impacts on Involved CLWR and INEEL Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 5

Receptor—Involved Workers[a]	Generic CLWR	INEEL FDPF	Total
Total dose (person-rem per year)	0	12[b]	12
35-year latent cancer fatalities	0	0.17	0.17
Average worker dose (millirem per year)	0	170	NA[c]
35-year latent cancer fatality risk	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with FDPF operations at a DOE facility would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced at all facilities.
b. Based on an estimated 75 badged workers.
c. Values cannot be given for the average worker because the workers would be at two different facilities and sites.

Key: NA, not applicable.
Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts at the generic CLWR site for this option would be the same as those of ongoing site operations because no new chemicals would be emitted.

Hazardous chemical impacts at INEEL for this option would be the same as those described for Option 2 (Section 4.4.2.1.9).

4.4.5.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with target irradiation in a generic CLWR and FDPF target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual and the offsite population within 80 kilometers (50 miles) of the facility. Consequences to a noninvolved worker are not included for the generic CLWR analysis. Details regarding the exclusion of a noninvolved worker are provided in Appendix I.

Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality for an individual and as the increased number of latent cancer fatalities in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality,

given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are plant specific, summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–92** and **4–93**, respectively. Certain extremely unlikely or incredible severe accidents at commercial nuclear reactors could result in doses sufficiently high to cause early fatalities. The early fatality consequences and risks are presented in **Table 4–94**. The early fatalities shown in Table 4–94 are considered to be conservative estimates based upon the assumption that some individuals very close to the reactor do not evacuate. Because the CLWR is currently operating, the consequences and risks are presented for both the current reactor configuration without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configuration.

For 35 years of CLWR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual would be 1.93×10^{-9}. The increased number of latent cancer fatalities in the surrounding population would be 0.00305. The increased risk of an early fatality in the surrounding population would be 2.07×10^{-6}.

For 35 years of FDPF target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 1.49×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0287.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 1.49×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0318.

The irradiation of neptunium-237 targets at the generic CLWR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at the generic CLWR.

No chemical processing activities are currently performed at FDPF and no chemicals are stored in this facility. Processing activities in support of plutonium-238 production would require the introduction of hazardous chemicals, specifically nitric acid and nitric oxide. Potential health impacts from accidental releases of nitric acid were assessed by comparing estimated airborne concentrations of the chemicals to ERPG developed by the American Industrial Hygiene Association. The ERPG-1 value (0.5 part per million) is the maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour, resulting in only mild, transient, and reversible adverse health effects. The ERPG-2 value (10 parts per million) is protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. The ERPG-3 value (25 parts per million) is indicative of potentially life-threatening health effects.

Table 4–92 Generic CLWR and FDPF Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 5

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
Generic CLWR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.0312	1.56×10^{-5}	186	0.0931	NA[c]	NA
Large-break LOCA with 5 kg/yr plutonium-238 production	0.0313	1.57×10^{-5}	187	0.0935	NA	NA
Early containment failure with 0 kg/yr plutonium-238 production	3,350	1.00[d]	1.80×10^{6}	1,250	NA	NA
Early containment failure with 5 kg/yr plutonium-238 production	3,670	1.00[d]	1.90×10^{6}	1,340	NA	NA
Late containment failure with 0 kg/yr plutonium-238 production	1.11	5.55×10^{-4}	1.06×10^{5}	53.6	NA	NA
Late containment failure with 5 kg/yr plutonium-238 production	1.12	5.60×10^{-4}	1.06×10^{5}	53.6	NA	NA
Containment bypass with 0 kg/yr plutonium-238 production	1,540	1.00[d]	1.45×10^{6}	922	NA	NA
Containment bypass with 5 kg/yr plutonium-238 production	1,680	1.00[d]	1.52×10^{6}	978	NA	NA
FDPF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.01×10^{-9}	1.01×10^{-12}	2.49×10^{-5}	1.24×10^{-8}	7.26×10^{-9}	2.91×10^{-12}
Target dissolver tank failure during plutonium-238 separation	6.11×10^{-8}	3.05×10^{-11}	5.65×10^{-4}	2.82×10^{-7}	2.17×10^{-7}	8.69×10^{-11}
Ion exchange explosion during plutonium-238 separation	1.63×10^{-5}	8.13×10^{-9}	0.150	7.51×10^{-5}	5.79×10^{-5}	2.31×10^{-8}
Processing facility beyond-design-basis earthquake	42.5	0.0425	1.64×10^{5}	82.0	1,200	1.0[d]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities. The MACCS2 computer code calculates the dose to each exposed individual in the population, applies the appropriate cancer risk factor, and then sums the individual probabilities to determine the number of latent cancer fatalities.
c. Not applicable (refer to Appendix I). Evacuation of noninvolved workers and other noninvolved worker issues are addressed in Appendix I.
d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–93 Generic CLWR and FDPF Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 5

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual generic CLWR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (4.65×10^{-5})	7.25×10^{-10}	4.33×10^{-6}	NA[c]
Large-break LOCA with 5 kg/yr plutonium-238 production (4.65×10^{-5})	7.30×10^{-10}	4.35×10^{-6}	NA
Large-break LOCA incremental risks[d]	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure with 0 kg/yr plutonium-238 production (7.92×10^{-8})	7.92×10^{-8}(e)	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production (7.92×10^{-8})	7.92×10^{-8}(e)	1.06×10^{-4}	NA
Early containment failure incremental risks	0.0	7.10×10^{-6}	NA
Late containment failure with 0 kg/yr plutonium-238 production (1.07×10^{-5})	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production (1.07×10^{-5})	5.99×10^{-9}	5.74×10^{-4}	NA
Late containment failure incremental risks	5.00×10^{-11}	0.0	NA
Containment bypass with 0 kg/yr plutonium-238 production (1.53×10^{-6})	1.53×10^{-6}(e)	0.00141	NA
Containment bypass with 5 kg/yr plutonium-238 production (1.53×10^{-6})	1.53×10^{-6}(e)	0.00149	NA
Containment bypass incremental risks	0.0	8.00×10^{-5}	NA
35-year CLWR risk[f]	1.93×10^{-9}	0.00305	NA
Annual FDPF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Processing facility beyond-design-basis earthquake (1×10^{-5})	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}(e)
35-year FDPF risk	1.49×10^{-5}	0.0287	3.50×10^{-4}
35-year Option risk[g]	1.49×10^{-5}	0.0318	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Not applicable (refer to Appendix I). Evacuation of noninvolved workers and other noninvolved worker issues are addressed in Appendix I.
d. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.

e. Risk of an early fatality.
f. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident; NA, not applicable.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–94 Generic CLWR Early Fatalities and Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 5

Accident (Frequency)	Population to 80 Kilometers (50 Miles)	
	Early Fatalities[a]	Annual Risk[b]
Annual generic CLWR risks		
Early containment failure with 0 kg/yr plutonium-238 production (7.92×10^{-8})	8.65	6.85×10^{-7}
Early containment failure with 5 kg/yr plutonium-238 production (7.92×10^{-8})	8.76	6.94×10^{-7}
Early containment failure incremental risk[c]	NA	9.00×10^{-9}
Containment bypass with 0 kg/yr plutonium-238 production (1.53×10^{-6})	3.48	5.32×10^{-6}
Containment bypass with 5 kg/yr plutonium-238 production (1.53×10^{-6})	3.51	5.37×10^{-6}
Containment bypass incremental risk	NA	5.00×10^{-8}
35-year CLWR risk[d]	NA	2.07×10^{-6}

a. Number of early fatalities assuming that the accident has occurred.
b. Risk of an early fatality.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. The 35-year risk is determined by summing the incremental annual risks and multiplying by 35.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; NA, not applicable.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

The maximum distances, in meters, needed to reach the ERPG values for nitric acid releases at FDPF for Stability Classes D and F are shown in **Table 4–95**. Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions, while Stability Classes F represents worst-case meteorological conditions. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they were sheltered within buildings at the time of release. Individuals at the nearest highway (5,800 meters [3.6 miles]) and at the nearest site boundary (13,952 meters [8.7 miles]) from FDPF would be exposed to levels well below ERPG-1.

Table 4–95 ERPG Distances for Nitric Acid Releases at FDPF Under Alternative 2 (Use Only Existing Operational Facilities)—Option 5

Evaluation Parameter	Stability Class D (meters)	Stability Class F (meters)
ERPG-3	375	450
ERPG-2	500	600
ERPG-1	2,000	3,000

Note: To convert from meters to miles, multiply by 6.22×10^{-4}.
Key: ERPG, Emergency Response Planning Guideline.

There are no ERPG values for nitric oxide. For nitric oxide accidents, the level of concern has been estimated by using one-tenth of the "Immediately Dangerous to Life and Health" level published by the National Institute for Occupational Safety and Health. The Immediately Dangerous to Life and Health value for nitric oxide is 100 parts per million. The level of concern value used for this NI PEIS is 10 parts per million. The level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects as a result of a single exposure for a relatively short period of time.

For FDPF, the maximum distances needed to reach the level of concern for nitric oxides releases for Stability Class D and F are 500 and 2,000 meters (0.31 and 1.24 miles), respectively. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they were sheltered within buildings at the time of release. Individual at the nearest highway (5,800 meters [3.6 miles]) and the nearest site boundary (13,952 meters [8.7 miles]) from FDPF would be exposed to levels well below the level of concern for nitric oxide.

Potential health impacts from the accidental release of the hazardous chemicals were assessed for a noninvolved worker, offsite individuals who are members of the public located at the nearest site boundary and onsite individuals who are members of the public located at the nearest highway access.

The impacts associated with the accidental release of nitric acid and nitric oxide at FDPF are presented in **Table 4–96**.

Table 4–96 FDPF Hazardous Chemical Accident Impacts Under Alternative 2 (Use Only Existing Operational Facilities)—Option 5

Receptor	Evaluation Parameter	Nitric Acid		Nitric Oxide	
		Stability Class D	Stability Class F	Stability Class D	Stability Class F
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	3.3 <ERPG-2 Mild, transient	8.4 <ERPG-2 Mild, transient	4.2 <LOC Mild, transient	67.5 >LOC Serious
Nearest highway maximally exposed individual	Parts per million Level of concern Potential health effects	0.05 < ERPG-1 None	0.15 ERPG-1 Mild, transient	0.09 < LOC None	0.87 < LOC None
Site boundary maximally exposed individual	Parts per million Level of concern Potential health effects	<<0.05 < ERPG-1 None	<<0.15 ERPG-1 Mild, transient	<<0.09 < LOC None	<<0.87 < LOC None

Note: < means "less than"; << means "much less than."
Key: ERPG, Emergency Response Planning Guideline; LOC, level of concern.
Source: Model results.

4.4.5.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the target fabrication facility at INEEL. DOE would transport the unirradiated neptunium-237 targets from FDPF to a CLWR. Following irradiation in the CLWR, the targets would be returned to FDPF for processing. After this processing, the plutonium-238 product would be shipped to LANL. The impact analysis, described in Appendix J, assumes the most distant CLWR is used for target irradiation.

Approximately 689 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 3.1 million kilometers (1.9 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 17 person-rem; the dose to the public, 357 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.007 latent cancer fatality among transportation workers and 0.18 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0066.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 were also evaluated and estimated to have a probability of lower than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.0042 person-rem, resulting in 2.1×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.088 traffic fatality.

4.4.5.1.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in a CLWR at an unspecified site. Target fabrication and processing would be performed at FDPF located at INEEL. Activities at FDPF were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.2.1.12) and found to pose no significant radiological or other risks to minority and low-income populations. The analysis of accidents at specific sites shows that accidents at the fabrication and target facilities would result in radiological risks to the public that are small, but which are several orders of magnitude larger than those that would result from accidents at specific reactor sites (see Section 2.7.1.1). It is plausible that a similar difference would exist between accident risks at an unspecified CLWR site and the fabrication and processing facilities. However, evaluations of environmental justice are necessarily site specific and cannot be performed for unspecified locations. In the event that this option were selected for implementation and a specific CLWR were selected for irradiation services, additional evaluation of environmental justice at the CLWR site would be performed prior to implementation.

4.4.5.1.13 Waste Management

There would be no change in the amounts of waste generated as the result of irradiating neptunium-237 targets in the CLWR. Thus, there would be no impact on the CLWR site's waste management systems as the result of target irradiation.

The impacts of managing waste associated with neptunium-237 target fabrication and processing in FDPF are assumed to be the same as for Option 2 under Alternative 1 (Section 4.3.2.1.13) because the same amount of plutonium-238 would be produced annually. As shown in that section, the impacts on the waste management systems at INEEL would be minimal.

4.4.5.1.14 Spent Nuclear Fuel Management

No incremental impacts would be associated with the management of spent nuclear fuel (refer to Section 4.4.1.1.14).

4.4.5.2 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.4.6 Alternative 2 (Use Only Existing Operational Facilities)—Option 6

This option involves operating a generic CLWR at a generic site to irradiate neptunium-237 targets, and operating FMEF at Hanford to both fabricate and process these targets and to store the neptunium-237 transported to Hanford from SRS.

The transportation of the neptunium-237 from SRS to Hanford for processing and fabrication into neptunium-237 targets in FMEF, the transportation of the targets from Hanford to the generic CLWR site for irradiation in the CLWR, the transportation of the irradiated targets back to Hanford for postirradiation processing in FMEF, and the transportation of the plutonium-238 product from Hanford to LANL also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.6.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all intersite transportation activities, are assessed in this section.

4.4.6.1.1 Land Resources

LAND USE. The use of a CLWR to irradiate neptunium-237 targets would not result in impacts on land use for the reasons described in Section 4.4.4.1.1.

Impacts on land use at Hanford from neptunium-237 storage, target fabrication, and processing at FMEF would be expected to be minimal for the reasons described in Section 4.4.3.1.1.

VISUAL RESOURCES. The use of a CLWR to irradiate neptunium-237 targets would not result in impacts on visual resources for the reasons described in Section 4.4.4.1.1.

Impacts on visual resources at Hanford from neptunium-237 target fabrication and processing at FMEF would be expected to be minimal for the reasons described in Section 4.4.3.1.1.

4.4.6.1.2 Noise

The irradiation of neptunium-237 targets in a CLWR would not result in noise impacts for the reasons described in Section 4.4.4.1.2.

Noise impacts at Hanford would be minimal from neptunium-237 storage, target fabrication, and processing at FMEF, and changes in traffic noise would be small for the reasons described in Section 4.4.3.1.2.

4.4.6.1.3 Air Quality

Impacts for this option at the generic CLWR site would be the same as those described for Option 4 (Section 4.4.4.1.3).

Impacts for this option at Hanford would be the same as those described for Option 3 (Section 4.4.3.1.3).

The air quality impacts of transportation among SRS, the generic CLWR site, Hanford, and LANL are presented in Section 4.4.6.1.11.

4.4.6.1.4 Water Resources

Impacts for this option at a generic CLWR site would be negligible as described for Option 4 (Section 4.4.4.1.4).

Impacts for this option at Hanford would be the same as described for Option 3 (Section 4.4.3.1.4). Groundwater withdrawals and the discharge of process and sanitary effluents by FMEF would increase.

4.4.6.1.5 Geology and Soils

The irradiation of neptunium-237 targets in a CLWR would not be expected to result in impacts on geologic or soil resources for the reasons described in Section 4.4.4.1.5. Assessment of hazards from large-scale geologic conditions for reactor sites, including assessment of seismic and nonseismic features, is governed by 10 CFR Part 100 and is beyond the scope of this analysis.

Neptunium-237 storage, target fabrication, and processing at FMEF would not be expected to impact geologic and soil resources at Hanford, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.4.2.5 and 4.4.3.1.5. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.6.1.6 Ecological Resources

The irradiation of neptunium-237 targets in a CLWR would not result impacts on ecological resources for the reasons described in Section 4.4.4.1.6.

Impacts on ecological resources at Hanford would not result from neptunium-237 storage, target fabrication, and processing at FMEF for the reasons described in Section 4.4.3.1.6.

4.4.6.1.7 Cultural and Paleontological Resources

The irradiation of neptunium-237 targets in a CLWR would not result in impacts on cultural and paleontological resources for the reasons described in Section 4.4.4.1.7.

Impacts on cultural and paleontological resources at Hanford would not result from neptunium-237 storage, target fabrication, and processing at FMEF for the reasons described in Section 4.4.3.1.7.

4.4.6.1.8 Socioeconomics

Reactor operations at a CLWR site would not require additional workers. Target fabrication and processing of plutonium-238 at Hanford would require approximately 62 additional workers (Hoyt et al. 1999). The socioeconomic impacts at Hanford are the same as those addressed in Section 4.4.3.1.8.

4.4.6.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–97** for the generic CLWR site and Hanford: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–97 Incremental Radiological Impacts on the Public Around the Generic CLWR Site and Hanford from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 6

Receptor	Generic CLWR	Hanford FMEF	Total
Population within 80 kilometers (50 miles) in the year 2020			
Dose (person-rem)	0	4.4×10^{-5}	4.4×10^{-5}
35-year latent cancer fatalities	0	7.7×10^{-7}	7.7×10^{-7}
Maximally exposed individual			
Annual dose (millirem)	0	4.7×10^{-7}	NA[a]
35-year latent cancer fatality risk	0	8.3×10^{-12}	NA[a]
Average exposed individual within 80 kilometers (50 miles)			
Annual dose[b] (millirem)	0	8.9×10^{-8}	NA[a]
35-year latent cancer fatality risk	0	1.6×10^{-12}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FMEF in the year 2020 (494,400).

Key: NA, not applicable.
Source: Model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

Target irradiation in a CLWR would not result in any incremental radiological emissions during normal operations or in increased worker exposures. Therefore, the incremental impact of CLWR target irradiation is zero.

As a result of annual operations of the generic CLWR and FMEF, the projected total incremental population dose in the year 2020 would be 4.4×10^{-5} person-rem. The corresponding number of latent cancer fatalities in the populations surrounding the generic CLWR site and Hanford from 35 years of operations would be 7.7×10^{-7}. The total incremental dose to the maximally exposed member of the public from annual generic CLWR operations would be 0 millirem because there would be no increase in radiological releases to the environment from the generic CLWR associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would, therefore, be zero. The incremental dose to the maximally exposed member of the public from annual FMEF operations would be 4.7×10^{-7} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 8.3×10^{-12}.

Incremental doses to involved workers from normal operations are given in **Table 4–98**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to CLWR workers would be 0 millirem; for FMEF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0 and approximately 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–98. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–98 Incremental Radiological Impacts on Involved CLWR and Hanford Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 6

Receptor—Involved Workers[a]	Generic CLWR	Hanford FMEF	Total
Total dose (person-rem per year)	0	12[b]	12
35-year latent cancer fatalities	0	0.17	0.17
Average worker dose (millirem per year)	0	170	NA[c]
35-year latent cancer fatality risk	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with FMEF operations at DOE facilities would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced at all facilities.
b. Based on an estimated 75 badged workers.
c. Values cannot be given for the average worker because the workers would be at two different facilities and sites.

Key: NA, not applicable.
Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts at the generic CLWR site for this option would be the same as those of current site operations because no new chemicals would be emitted.

Hazardous chemical impacts for this option at Hanford would be the same as those described for Option 3 (Section 4.4.3.1.9).

4.4.6.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with target irradiation in a generic CLWR and FMEF target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual and the offsite population within 80 kilometers (50 miles) of the facility. Consequences to a noninvolved worker are

not included for the generic CLWR analysis. Details regarding the exclusion of a noninvolved worker are provided in Appendix I.

Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality for an individual and as the increased number of latent cancer fatalities in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are plant specific, summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–99** and **4–100**, respectively. Certain extremely unlikely or incredible severe accidents at commercial nuclear reactors could result in doses sufficiently high to cause early fatalities. The early fatality consequences and risks are presented in **Table 4–101**. The early fatalities shown in Table 4–101 are considered to be conservative estimates based upon the assumption that some individuals very close to the reactor do not evacuate. Because the generic CLWR is currently operating, the consequences and risks are presented for both the current reactor configuration without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configuration.

For 35 years of CLWR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual would be 1.93×10^{-9}. The increased number of latent cancer fatalities in the surrounding population would be 0.00305. The increased risk of an early fatality in the surrounding population would be 2.07×10^{-6}.

For 35 years of FMEF target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and of an early fatality to a noninvolved worker would be 2.88×10^{-6} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.112.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 2.88×10^{-6} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.115.

The irradiation of neptunium-237 targets at the generic CLWR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at the generic CLWR.

Table 4–99 Generic CLWR and FMEF Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 6

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
Generic CLWR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.0312	1.56×10^{-5}	186	0.0931	NA[c]	NA
Large-break LOCA with 5 kg/yr plutonium-238 production	0.0313	1.57×10^{-5}	187	0.0935	NA	NA
Early containment failure with 0 kg/yr plutonium-238 production	3,350	1.00[d]	1.80×10^{6}	1,250	NA	NA
Early containment failure with 5 kg/yr plutonium-238 production	3,670	1.00[d]	1.90×10^{6}	1,340	NA	NA
Late containment failure with 0 kg/yr plutonium-238 production	1.11	5.55×10^{-4}	1.06×10^{5}	53.6	NA	NA
Late containment failure with 5 kg/yr plutonium-238 production	1.12	5.60×10^{-4}	1.06×10^{5}	53.6	NA	NA
Containment bypass with 0 kg/yr plutonium-238 production	1,540	1.00[d]	1.45×10^{6}	922	NA	NA
Containment bypass with 5 kg/yr plutonium-238 production	1,680	1.00[d]	1.52×10^{6}	978	NA	NA
FMEF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.02×10^{-9}	1.01×10^{-12}	7.26×10^{-5}	3.63×10^{-8}	6.65×10^{-10}	2.66×10^{-13}
Target dissolver tank failure during plutonium-238 separation	4.64×10^{-8}	2.32×10^{-11}	0.00169	8.47×10^{-7}	1.95×10^{-8}	7.81×10^{-12}
Ion exchange explosion during plutonium-238 separation	1.24×10^{-5}	6.18×10^{-9}	0.451	2.25×10^{-4}	5.20×10^{-6}	2.08×10^{-9}
Processing facility beyond-design-basis earthquake	16.5	0.00823	6.41×10^{5}	321	921	1.0[d]

a. Likelihood of a latent cancer fatality.

b. Number of latent cancer fatalities. The MACCS2 computer code calculates the dose to each exposed individual in the population, applies the appropriate cancer risk factor, and then sums the individual probabilities to determine the number of latent cancer fatalities.

c. Not applicable (refer to Appendix I). Evacuation of noninvolved workers and other noninvolved worker issues are addressed in Appendix I.

d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident; NA, not applicable.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–100 Generic CLWR and FMEF Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 6

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles) [b]	Noninvolved Worker[a]
Annual generic CLWR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (4.65×10^{-5})	7.25×10^{-10}	4.33×10^{-6}	NA[c]
Large-break LOCA with 5 kg/yr plutonium-238 production (4.65×10^{-5})	7.30×10^{-10}	4.35×10^{-6}	NA
Large-break LOCA incremental risks[d]	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure with 0 kg/yr plutonium-238 production (7.92×10^{-8})	7.92×10^{-8}(e)	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production (7.92×10^{-8})	7.92×10^{-8}(e)	1.06×10^{-4}	NA
Early containment failure incremental risks	0.0	7.10×10^{-6}	NA
Late containment failure with 0 kg/yr plutonium-238 production (1.07×10^{-5})	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production (1.07×10^{-5})	5.99×10^{-9}	5.74×10^{-4}	NA
Late containment failure incremental risks	5.00×10^{-11}	0.0	NA
Containment bypass with 0 kg/yr plutonium-238 production (1.53×10^{-6})	1.53×10^{-6}(d)	0.00141	NA
Containment bypass with 5 kg/yr plutonium-238 production (1.53×10^{-6})	1.53×10^{-6}(e)	0.00149	NA
Containment bypass incremental risks	0.0	8.00×10^{-5}	NA
35-year CLWR risk[f]	1.93×10^{-9}	0.00305	NA
Annual FMEF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Processing facility beyond-design- basis earthquake (1×10^{-5})	8.23×10^{-8}	0.00321	1.00×10^{-5}(e)
35-year FMEF risk	2.88×10^{-6}	0.112	3.50×10^{-4}
35-year Option risk[g]	2.88×10^{-6}	0.115	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Not applicable (refer to Appendix I). Evacuation of noninvolved workers and other noninvolved worker issues are addressed in Appendix I.
d. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
e. Risk of an early fatality.
f. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident; NA, not applicable.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–101 Generic CLWR Early Fatalities and Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 6

Accident (Frequency)	Population to 80 Kilometers (50 Miles)	
	Early Fatalities[a]	Annual Risk[b]
Annual generic CLWR risks		
Early containment failure with 0 kg/yr plutonium-238 production (7.92×10^{-8})	8.65	6.85×10^{-7}
Early containment failure with 5 kg/yr plutonium-238 production (7.92×10^{-8})	8.76	6.94×10^{-7}
Early containment failure incremental risk[c]	NA	9.00×10^{-9}
Containment bypass with 0 kg/yr plutonium-238 production (1.53×10^{-6})	3.48	5.32×10^{-6}
Containment bypass with 5 kg/yr plutonium-238 production (1.53×10^{-6})	3.51	5.37×10^{-6}
Containment bypass incremental risks	NA	5.00×10^{-8}
35-year CLWR risk[d]	NA	2.07×10^{-6}

a. Number of early fatalities assuming that the accident has occurred.
b. Risk of an early fatality.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; NA, not applicable.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

No chemical processing activities are currently performed at FMEF and no chemicals are stored in this facility. Processing activities in support of plutonium-238 production would require the introduction of hazardous chemicals, specifically nitric acid and nitric oxide. Potential health impacts from accidental releases of nitric acid were assessed by comparing estimated airborne concentrations of the chemicals to ERPG developed by the American Industrial Hygiene Association. The ERPG-1 value (0.5 part per million) is the maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour, resulting in only mild, transient, and reversible adverse health effects. The ERPG-2 value (10 parts per million) is protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. The ERPG-3 value (25 parts per million) is indicative of potentially life-threatening health effects.

The maximum distances, in meters, needed to reach the ERPG values for nitric acid releases at FMEF for Stability Classes D and F are shown in **Table 4–102**. Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions, while Stability Class F represents worst-case meteorological conditions. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they were sheltered within buildings at the time of release. Individuals at the nearest highway (7,100 meters [4.4 miles]) and at the nearest site boundary (7,210 meters [4.5 miles]) from FMEF would be exposed to levels well below ERPG-1.

Table 4–102 ERPG Distances for Nitric Acid Releases at FMEF Under Alternative 2 (Use Only Existing Operational Facilities)—Option 6

Evaluation Parameter	Stability Class D (meters)	Stability Class F (meters)
ERPG-3	375	450
ERPG-2	500	600
ERPG-1	2,000	3,000

Note: To convert from meters to miles, multiply by 6.22×10^{-4}.
Key: ERPG, Emergency Response Planning Guideline.

There are no ERPG values for nitric oxide. For nitric oxide accidents, the level of concern has been estimated by using one-tenth of the "Immediately Dangerous to Life and Health" level published by the National Institute for Occupational Safety and Health. The Immediately Dangerous to Life and Health value for nitric oxide is 100 parts per million. The level of concern value used for this NI PEIS is 10 parts per million. The level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects as a result of a single exposure for a relatively short period of time.

For FMEF, the maximum distances needed to reach the level of concern for nitric oxides releases for Stability Classes D and F are 500 and 1,900 meters (0.31 and 1.18 miles), respectively. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they were sheltered within buildings at the time of release. Individual at the nearest highway (7,100 meters [4.4 miles]) and the nearest site boundary (7,210 meters [4.5 miles]) from FMEF would be exposed to levels well below the level of concern for nitric oxide.

Potential health impacts from the accidental release of the hazardous chemicals were assessed for a noninvolved worker, offsite individuals who are members of the public located at the nearest site boundary and onsite individuals who are members of the public located at the nearest highway access. Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions, while Stability Class F represents worst-case meteorological conditions.

The impacts associated with the accidental release of nitric acid and nitric oxide at FMEF are presented in **Table 4–103**.

Table 4–103 FMEF Hazardous Chemical Accident Impacts Under Alternative 2 (Use Only Existing Operational Facilities)—Option 6

Receptor	Evaluation Parameter	Nitric Acid		Nitric Oxide	
		Stability Class D	Stability Class F	Stability Class D	Stability Class F
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	3.3 <ERPG-2 Mild, transient	8.6 <ERPG-2 Mild, transient	4.2 <LOC Mild, transient	66 >LOC Serious
Nearest highway maximally exposed individual	Parts per million Level of concern Potential health effects	0.03 < ERPG-1 None	0.1 ERPG-1 Mild, transient	0.09 < LOC None	0.55 < LOC None
Site boundary maximally exposed individual	Parts per million Level of concern Potential health effects	0.03 < ERPG-1 None	0.1 ERPG-1 Mild, transient	0.09 < LOC None	0.53 < LOC None

Note: < means "less than."
Key: ERPG, Emergency Response Planning Guideline; LOC, level of concern.
Source: Model results.

4.4.6.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FMEF target fabrication facility at Hanford. DOE would transport the unirradiated neptunium-237 targets from FMEF to a CLWR. Following irradiation in the CLWR, the targets would be returned to FMEF for processing. After this processing, the plutonium-238 product would be shipped to LANL. The impact analysis, described in Appendix J, assumes the most distant CLWR is used for target irradiation.

Approximately 689 shipments of radioactive materials would be made by DOE under this option. The total distance traveled on public roads by trucks carrying radioactive materials would be 3.6 million kilometers (2.2 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 20 person-rem; the dose to the public, 411 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.008 latent cancer fatality among transportation workers and 0.21 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0075.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FMEF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 were also evaluated and estimated to have a probability of lower than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.06 person-rem, resulting in 3.0×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.10 traffic fatality.

4.4.6.1.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in a CLWR at an unspecified site. Target fabrication and processing would be performed at FMEF located at Hanford. Activities at FMEF were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.3.1.12) and found to pose no significant radiological or other risks to minority and low-income populations. The analysis of accidents at specific sites shows that accidents at the fabrication and target facilities would result in radiological risks to the public that are small, but which are several orders of magnitude larger than those that would result from accidents at specific reactor sites (see Section 2.7.1.1). It is plausible that a similar difference would exist between accident risks at an unspecified CLWR site and the fabrication and processing facilities. However, evaluations of environmental justice are necessarily site specific and cannot be performed for unspecified locations. In the event that this option were selected for implementation and a specific CLWR were selected for irradiation services, additional evaluation of environmental justice at the CLWR site would be performed prior to implementation.

4.4.6.1.13 Waste Management

There would be no change in the amounts of waste generated as the result of irradiating neptunium-237 targets in the CLWR. Thus, there would be no impact on the CLWR site's waste management systems as the result of target irradiation.

The impacts of managing waste associated with neptunium-237 target fabrication and processing in FMEF are assumed to be the same as for Option 3 (Section 4.4.3.1.13) because the same amount of plutonium-238 would be produced annually. As shown in that section, the impacts on the waste management systems at Hanford would be minimal.

4.4.6.1.14 Spent Nuclear Fuel Management

No incremental impacts would be associated with the management of spent nuclear fuel (refer to Section 4.4.1.1.14).

4.4.6.2 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.4.7 Alternative 2 (Use Only Existing Operational Facilities)—Option 7

This option involves operating both the High Flux Isotope Reactor (HFIR) at ORR and ATR at INEEL to irradiate neptunium-237 targets, and operating the REDC facility at ORR to both fabricate and process these targets and to store the neptunium-237 transported to ORR from SRS.

The transportation of the neptunium-237 from SRS to ORR for processing and fabrication into neptunium-237 targets in REDC, the transportation of a portion of these targets from ORR to INEEL for irradiation in ATR, the transportation of the irradiated targets back to ORR for postirradiation processing in REDC, and the transportation of the entire plutonium-238 product from ORR to LANL also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.7.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.4.7.1.1 Land Resources

LAND USE. The use of ATR to irradiate neptunium-237 targets would not result in impacts on land use at INEEL for the reasons described in Section 4.4.1.1.1.

The irradiation of neptunium-237 targets would also take place at HFIR. HFIR is an existing facility in the 7900 Area of ORNL. Use of the facility for target irradiation would not involve any new construction. Because no additional land would be disturbed and the target irradiation would be compatible with the present mission of the reactor, there would be no change in impacts on land use at ORR.

There would be no impacts on land use at ORR from neptunium-237 storage, target fabrication, and processing at REDC for the reasons described in Section 4.4.1.1.1.

VISUAL RESOURCES. The use of ATR to irradiate neptunium-237 targets would not result in visual impacts at INEEL for the reasons described in Section 4.4.1.1.1.

The irradiation of neptunium-237 targets would also take place within HFIR at ORR. Because HFIR is an existing facility that would require no external modifications, there would be no change in its appearance. Therefore, the current Visual Resource Management Class IV rating for the 7900 Area would not change. Because there would be no change in the appearance of HFIR or the 7900 Area, there would be no impact on visual resources.

Neptunium-237 storage, target fabrication, and processing at REDC would not impact visual resources at ORR for the reasons described in Section 4.4.1.1.1.

4.4.7.1.2 Noise

The irradiation of neptunium-237 targets in ATR would not result in noise impacts at INEEL for the reasons described in Section 4.4.1.1.2.

Noise generated during the irradiation of neptunium-237 targets in HFIR would be similar to sound levels associated with current reactor operations, as well as other operations conducted within the 7900 Area. Onsite noise impacts would be expected to be minimal, and changes in offsite noise levels would not be noticeable, because the nearest site boundary is 2.5 kilometers (1.6 miles) to the southeast. Changes in traffic volume going to and from HFIR would be small, and would result in only minor changes to onsite and offsite noise levels. There would be no loud noises associated with neptunium-237 target irradiation that would adversely impact wildlife.

Noise impacts at ORR would not be expected from neptunium-237 storage, target fabrication, and processing at REDC and changes in traffic noise would be small for the reasons described in Section 4.4.1.1.2.

4.4.7.1.3 Air Quality

Impacts for this option at INEEL would be the same as those described for Option 1 (Section 4.4.1.1.3).

Impacts for this option at ORR would be the same as those described for Option 1 (Section 4.4.1.1.3). There would be no measurable nonradiological air pollutant emissions associated with the operation of HFIR.

The air quality impacts of transportation among SRS, INEEL, ORR, and LANL are presented in Section 4.4.7.1.11.

4.4.7.1.4 Water Resources

The irradiation of neptunium-237 targets for plutonium-238 production in ATR at INEEL would have no measurable impact on water resources as previously described for Option 1 (Section 4.4.1.1.4). Under this option, neptunium-237 target irradiation would also be conduced in the HFIR at ORR. Similar to ATR, impacts on water resources associated with the dual operation of HFIR in the 7900 Area of ORR would not be expected to impact water resources as plutonium-238 production would not measurably increase water use or change the quality or quantity of effluents discharged. Both facilities would already be operating for other purposes so dual operation should not have any measurable cumulative impact.

REDC at ORR would be used for neptunium-237 storage, target fabrication, and processing. Impacts on water resources of this activity were determined to be the same as previously described for Option 1 (see Section 4.4.1.1.4). Impacts of this option on water resources are expected to be negligible overall.

4.4.7.1.5 Geology and Soils

The use of ATR to irradiate neptunium-237 targets would not be expected to result in impacts on geologic or soil resources, nor be jeopardized by large-scale geologic conditions, for the reasons described in Section 4.4.1.1.5.

HFIR would also be used to irradiate neptunium-237 targets. Because there would be no construction, there would be no disturbance to either geologic or soil resources in the 7900 Area of ORR. Impacts on geologic and soil resources at ORR would not be expected from neptunium-237 storage, target fabrication, and processing at REDC for the reasons described in Section 4.4.1.1.5. Hazards from large-scale geologic conditions at ORR, such as earthquakes and volcanoes, were evaluated as summarized in Section 4.2.2.2.5. The analysis determined that these hazards present a low risk to specially designed or upgraded facilities (such as HFIR and REDC), and is not revisited here.

As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards will be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.7.1.6 Ecological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on ecological resources at INEEL for the reasons described in Section 4.4.1.1.6.

The irradiation of neptunium-237 targets would also take place in the existing HFIR facility at ORR. No new construction would occur that could cause direct disturbance to ecological resources, including wetlands. As noted in Section 4.4.7.1.2, there would be no loud noises that would adversely impact wildlife. There would be no change in impacts on aquatic resources because additional water would not be withdrawn from or discharged to site surface waters and effluent chemistry would not measurably change (Section 4.4.1.1.4). Due to the developed nature of the area and because no new construction would take place, impacts on threatened and endangered species would not occur.

Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service (see Table 5–3) and resulted in the Service concluding that it does not anticipate adverse effects to federally listed endangered species that occur near the project area. DOE has also consulted with the Tennessee Department of Environment and Conservation; a response concerning state-listed species is pending from this agency. Although no state-listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the state.

There would be no impacts on ecological resources at ORR from neptunium-237 storage, target fabrication, and processing at REDC for the reasons described in Section 4.4.1.1.6.

4.4.7.1.7 Cultural and Paleontological Resources

The use of ATR to irradiate neptunium-237 targets at INEEL would not result in impacts on cultural and paleontological resources for the reasons described in Section 4.4.1.1.7.

The irradiation of neptunium-237 targets would also take place in the existing HFIR facility at ORR. No new construction would take place. Therefore, direct impacts on cultural and paleontological resources would not occur. One structure located within ORNL, the Graphite Reactor, is listed on the National Register of Historic Places as a National Historic Landmark. Additionally, several other structures proposed for listing on the National Register of Historic Places are found within or near ORNL. However, neither the Graphite Reactor nor any of the other structures is located within the 7900 Area, and therefore, their status would not change by the use of HFIR for the irradiation of neptunium-237 targets.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). While DOE has made additional contact with the State Historic Preservation Office, a response is pending from this office. Although impacts to cultural resources are not

expected as a result of the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the State Historic Preservation Office.

Impacts on cultural and paleontological resources at ORR would not result from neptunium-237 target fabrication and processing at REDC for the reasons described in Section 4.4.1.1.7.

4.4.7.1.8 Socioeconomics

After facility modifications, startup, and testing of the plutonium-238 reactor operation facilities at INEEL, and reactor operation and target fabrication/processing facilities at ORR, approximately 41 additional workers would be required to operate these facilities (none at INEEL and approximately 41 at ORR) (Wham et al. 1998). The socioeconomic impacts at ORR are the same as those addressed in Section 4.3.1.1.8.

4.4.7.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–104** for INEEL and ORR: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–104 Incremental Radiological Impacts on the Public Around INEEL and ORR from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 7

Receptor	INEEL ATR	ORR			Two-Site Total
		HFIR	REDC	Total	
Population within 80 kilometers (50 miles) in the year 2020					
Dose (person-rem)	0	0	8.8×10^{-5}	8.8×10^{-5}	8.8×10^{-5}
35-year latent cancer fatalities	0	0	1.5×10^{-6}	1.5×10^{-6}	1.5×10^{-6}
Maximally exposed individual					
Annual dose (millirem)	0	0	1.9×10^{-6}	1.9×10^{-6}	NA[a]
35-year latent cancer fatality risk	0	0	3.3×10^{-11}	3.3×10^{-11}	NA[a]
Average exposed individual within 80 kilometers (50 miles)					
Annual dose[b] (millirem)	0	0	7.8×10^{-8}	7.8×10^{-8}	NA[a]
35-year latent cancer fatality risk	0	0	1.4×10^{-12}	1.4×10^{-12}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of HFIR and REDC in the year 2020 (1,134,200).

Key: NA, not applicable.
Source: Model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual operations of ATR at INEEL and HFIR and REDC at ORR, the projected incremental total population dose in the year 2020 would be 8.8×10^{-5} person-rem. The corresponding number of latent cancer fatalities in the populations surrounding INEEL and ORR from 35 years of operations would be 1.5×10^{-6}. The total incremental dose to the maximally exposed member of the public from annual ATR and HFIR operations would be 0 millirem because there would be no increase in radiological releases to the environment from either of these reactors associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would, therefore, be zero. The incremental dose to the maximally exposed member of the public from annual HFIR and REDC operations would be 1.9×10^{-6} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 3.3×10^{-11}.

Incremental doses to involved workers from normal operations are given in **Table 4–105**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to ATR workers would be 0 millirem; for HFIR workers, the incremental annual average dose would also be 0 millirem; for REDC workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0, 0, and approximately 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–105. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–105 Incremental Radiological Impacts on Involved INEEL and ORR Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 7

Receptor—Involved Workers[a]	INEEL	ORR		Total
	ATR	HFIR	REDC	
Total dose (person-rem per year)	0	0	12[b]	12
35-year latent cancer fatalities	0	0	0.17	0.17
Average worker dose (millirem per year)	0	0	170	NA[c]
35-year latent cancer fatality risk	0	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers.
c. Values cannot be given for the average worker because the workers would be in three different facilities at two different sites.
Key: NA, not applicable.
Source: Mecham 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. No new hazardous chemicals would be emitted at HFIR. Therefore, impacts for this option at both INEEL and ORR would be the same as those described for Option 1 (Section 4.4.1.1.9).

4.4.7.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with ATR and HFIR target irradiation and REDC target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the

probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed offsite individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–106** and **4–107**, respectively. Because ATR and HFIR are currently operating, the consequences and risks are presented for both the current reactor configurations without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configurations. Baseline accident risks attributed to ATR and HFIR operations refer to accidents that could occur under the current ATR and HFIR configurations (without neptunium-237 targets). Baseline accident risks are obtained from the data in Table 4–107 by summing the annual risks for the baseline reactor configuration (0 kilogram per year plutonium-238 production), and then multiplying the sum by 35. The baseline ATR accident risk to the public would be 0.089 latent cancer fatality. Baseline ATR accident risks to the maximally exposed offsite individual and a noninvolved worker would be 8.2×10^{-7} and 7.2×10^{-6} latent cancer fatalities, respectively. Similarly, the baseline HFIR accident risk to the public would be 0.0052 latent cancer fatality. Baseline HFIR accident risks to the maximally exposed offsite individual and a noninvolved worker would be 4.2×10^{-6} and 2.4×10^{-5} latent cancer fatalities, respectively.

For 35 years of ATR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 1.49×10^{-7} and 1.95×10^{-6}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 7.01×10^{-4}.

For 35 years of HFIR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 8.68×10^{-9} and 3.43×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 4.09×10^{-5}.

For 35 years of REDC target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and of an early fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.157.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.157.

The irradiation of neptunium-237 targets at ATR and HFIR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at ATR and HFIR.

Table 4–106 ATR, HFIR, and REDC Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 7

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
ATR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.465	2.33×10^{-4}	5.11×10^{4}	25.5	5.15	0.00206
Large-break LOCA with 3 kg/yr plutonium-238 production	0.549	2.75×10^{-4}	5.15×10^{4}	25.7	6.52	0.00261
Target handling with 0 kg/yr plutonium-238 production[c]	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 3 kg/yr plutonium-238 production	1.23×10^{-4}	6.15×10^{-8}	0.0786	3.93×10^{-5}	0.00195	7.80×10^{-7}
HFIR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	2.41	0.00121	2,990	1.49	17.2	0.00688
Large-break LOCA with 2 kg/yr plutonium-238 production	2.41	0.00121	3,000	1.50	17.2	0.00688
Target handling with 0 kg/yr plutonium-238 production	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 2 kg/yr plutonium-238 production	4.96×10^{-4}	2.48×10^{-7}	0.335	1.68×10^{-4}	0.00245	9.80×10^{-7}
REDC accidents						
Ion exchange explosion during neptunium-237 target fabrication	6.13×10^{-9}	3.06×10^{-12}	8.58×10^{-5}	4.29×10^{-8}	5.60×10^{-10}	2.24×10^{-13}
Target dissolver tank failure during plutonium-238 separation	1.76×10^{-7}	8.79×10^{-11}	0.00196	9.82×10^{-7}	1.69×10^{-8}	6.74×10^{-12}
Ion exchange explosion during plutonium-238 separation	4.68×10^{-4}	2.34×10^{-7}	5.23	0.00261	4.49×10^{-5}	1.79×10^{-8}
Processing facility beyond-design-basis earthquake	163	0.163	8.91×10^{5}	445	1,310	1.00[d]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. There would be no neptunium-237 targets for this zero-production case. Thus, there would be not associated accident consequences.
d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–107 ATR, HFIR, and REDC Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 7

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual ATR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	2.33×10^{-8}	0.00255	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production (1×10^{-4})	2.75×10^{-8}	0.00257	2.61×10^{-7}
Large-break LOCA incremental risks[c]	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target handling with 3 kg/yr plutonium-238 production[d] (0.001)	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
35-year ATR risk[e]	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
Annual HFIR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production (1×10^{-4})	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risks[c]	0.0	1.00×10^{-6}	0.0
Neptunium-237 target handling with 2 kg/yr plutonium-238 production[d] (0.001)	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
35-year HFIR risk[e]	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
Annual REDC risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Processing facility beyond-design-basis earthquake (1×10^{-5})	1.63×10^{-6}	0.00445	1.00×10^{-5}[f]
35-year REDC risk	5.71×10^{-5}	0.157	3.50×10^{-4}
35-year Option risk[g]	5.71×10^{-5}	0.157	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. There would be no neptunium-237 targets for the zero-production case. Thus, the (3 kg/yr at ATR, 2 kg/yr at HFIR) production rate target-handling risks are the incremental risks.
e. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
f. Risk of an early fatality.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Processing associated with the plutonium-238 production program at REDC, including storage of neptunium-237 and plutonium-238, neptunium-237 target fabrication, postirradiation processing to extract plutonium-238 and to recycle the unconverted neptunium-237 into new targets, would not require the introduction of hazardous chemicals that are not in current use in the facility. The quantities of in-process hazardous chemicals for the plutonium-238 production program are bounded by the quantities of the material currently stored in the facility. The impacts of in-process hazardous chemical accidents associated with the plutonium-238 production are bounded by the impacts of hazardous chemical accidents for existing storage facilities at REDC.

4.4.7.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to HFIR, also at ORR, and to ATR at INEEL. Following irradiation in HFIR or ATR, the targets would be returned to REDC for processing. After this processing, the plutonium-238 product would be shipped to LANL. The analysis is described in Appendix J.

Approximately 563 intersite shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 1.8 million kilometers (1.1 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 10 person-rem; the dose to the public, 192 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.004 latent cancer fatality among transportation workers and 0.096 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0052.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to REDC with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 were also evaluated and estimated to have a probability of lower than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 0.088 person-rem, resulting in 4.4×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.048 traffic fatality.

4.4.7.1.12 Environmental Justice

NORMAL OPERATIONS. For 35 years of normal operations under this option, the radiological risk among the population residing within 80 kilometers (50 miles) of ATR, HFIR, and REDC would be less than 2×10^{-6} latent cancer fatalities. As shown in Sections 4.4.1.1.9 and 4.4.7.1.9, the release of hazardous chemicals at ORR and at INEEL would pose no significant risk of cancer or toxic effects among the public. As discussed in Sections K.5.1 and K.5.2, the likelihood that a latent cancer fatality would result from the ingestion of food that could be radiologically contaminated due to normal operations would be essentially zero

at INEEL and ORR. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near INEEL or ORR. As discussed in Section 4.4.7.1.11, no fatalities would be expected to result from incident-free transportation.

ACCIDENTS. The number of expected latent cancer fatalities among populations at risk due to accidents listed in Table 4–107 would be approximately 0.16. If a radiological accident were to occur at ATR and northwesterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Fort Hall Indian Reservation (see Figure K–2). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the population or maximally exposed individual residing within the boundary of the Fort Hall Indian Reservation. In the event a radiological accident were to occur at REDC or HFIR and southerly winds prevailed at the time of the accident, radiological contamination would be directed toward the predominately minority population of the Scarboro community adjacent to the northern boundary of ORR (see Figure K–6). If the winds were blowing from the west-southwest at the time of the accident, radiological contamination would be directed toward minority populations residing in Knoxville, Tennessee. Accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the minority populations or maximally exposed individuals residing in the Scarboro community or Knoxville.

As discussed in Section 4.4.7.1.11, no fatalities would be expected to result from transportation accidents.

In summary, the implementation of this option would pose no significant radiological risk to persons residing in potentially affected areas or along representative transportation routes. Under the conservative assumption that all food consumed in potentially affected areas during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to the ingestion of contaminated food supplies. As discussed in other parts of Section 4.4.7.1, the implementation of this option would not result in significant nonradiological impacts on populations at risk. Thus, implementation would not pose significant and adverse environmental risks to persons residing within potentially affected areas, including minority and low-income persons.

4.4.7.1.13 Waste Management

Only very small amounts of additional waste would be generated as a result of irradiating neptunium-237 targets in ATR and HFIR because these reactors would already be operating for other purposes. The anticipated incremental generation of waste from ATR operations is discussed in Section 4.4.1.1.13. The operation of HFIR is expected to increase the generation of solid low-level radioactive waste by less than 1 cubic meter (1.3 cubic yards) per year. There would be virtually no impacts on either site's waste management systems as the result of neptunium-237 target irradiation.

The impacts of managing waste associated with neptunium-237 target fabrication and processing in REDC are assumed to be the same as for Option 1 under Alternative 1 (Section 4.3.1.1.13) because the same amount of plutonium-238 would be produced annually. As shown in that section, the impacts on the waste management systems at ORR would be minimal.

4.4.7.1.14 Spent Nuclear Fuel Management

No incremental impacts would be associated with the management of spent nuclear fuel (refer to Section 4.4.1.1.14).

4.4.7.2 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.4.8 Alternative 2 (Use Only Existing Operational Facilities)—Option 8

This option involves operating both the HFIR at ORR and ATR at INEEL to irradiate neptunium-237 targets, and operating FDPF at INEEL to fabricate and process these targets. This option also includes storage of the neptunium-237 transported to INEEL from SRS, in either Building CPP–651 or FDPF.

The transportation of the neptunium-237 from SRS to INEEL for processing and fabrication into neptunium-237 targets in FDPF, the transportation of a portion of these targets from INEEL to ORR for irradiation in HFIR, the transportation of the irradiated targets back to INEEL for postirradiation processing in FDPF, and the transportation of the entire plutonium-238 product from INEEL to LANL also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.8.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.4.8.1.1 Land Resources

LAND USE. The irradiation of neptunium-237 targets in ATR would not result in impacts on land use at INEEL for the reasons described in Section 4.4.1.1.1.

The irradiation of neptunium-237 targets would also take place in the existing HFIR facility. There would be no impacts on land use at ORR for the reasons described in Section 4.4.7.1.1.

There would be no impacts on land use at INEEL from neptunium-237 storage, target fabrication, and processing for the reasons described in Section 4.4.2.1.1.

VISUAL RESOURCES. The irradiation of neptunium-237 targets in ATR would not result in impacts on visual resources at INEEL for the reasons described in Section 4.4.1.1.1.

The irradiation of neptunium-237 targets would also take place within the existing HFIR facility. There would be no impacts on visual resources at ORR for the reasons described in Section 4.4.7.1.1.

There would be no impacts on visual resources at INEEL from neptunium-237 storage, target fabrication, and processing for the reasons described in Section 4.4.2.1.1.

4.4.8.1.2 Noise

The irradiation of neptunium-237 targets in ATR would not result in noise impacts at INEEL for the reasons described in Section 4.4.1.1.2.

The irradiation of neptunium targets would also take place in HFIR. No change in noise impacts at ORR would be expected for the reasons described in Section 4.4.7.1.2.

Noise impacts at INEEL would not be expected from neptunium-237 storage, target fabrication, and processing and changes in traffic noise would be small for the reasons described in Section 4.4.2.1.2.

4.4.8.1.3 Air Quality

Impacts for this option at INEEL would be the same as those described for Option 2 (Section 4.4.2.1.3).

It is expected that there would be no measurable increases in nonradiological air pollutant emissions at ORR associated with HFIR operations; therefore, no changes in nonradiological air quality impacts would be expected (Wham 1999a).

The air quality impacts of transportation among SRS, INEEL, ORR, and LANL are presented in Section 4.4.8.1.11.

4.4.8.1.4 Water Resources

Impacts for this option at INEEL would be the same as those described for Option 2 (Section 4.4.2.1.4).

The irradiation of neptunium-237 targets would also take place in the existing HFIR facility at ORR. No measurable impact on water resources at ORR would be expected for the same reasons as described in Section 4.4.7.1.4.

4.4.8.1.5 Geology and Soils

The use of ATR to irradiate neptunium-237 targets at INEEL would not be expected to result in impacts on geologic or soil resources, nor be jeopardized by large-scale geologic conditions, for the reasons described in Section 4.4.1.1.5.

Dual use of HFIR at ORR to irradiate neptunium-237 targets would also not be expected to result in impacts on geologic and soil resources, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.2.2.5 and 4.4.7.1.5.

Neptunium-237 storage, target fabrication, and processing in FDPF would not be expected to impact geologic and soil resources at INEEL, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.3.2.5 and 4.4.2.1.5.

As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards will be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.8.1.6 Ecological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on ecological resources at INEEL for the reasons described in Section 4.4.1.1.6.

The irradiation of neptunium targets would also take place in HFIR. There would be no impacts on ecological resources at ORR for the reasons described in Section 4.4.7.1.6.

There would be no impacts on ecological resources at INEEL from neptunium-237 storage, target fabrication, and processing for the reasons described in Section 4.4.2.1.6.

4.4.8.1.7 Cultural and Paleontological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on cultural and paleontological resources at INEEL for the reasons described in Section 4.4.1.1.7.

The irradiation of neptunium-237 targets would also take place in HFIR. Impacts on cultural and paleontological resources at ORR would not be expected for the reasons described in Section 4.4.7.1.7.

Impacts on cultural and paleontological resources at INEEL would not be expected from neptunium-237 storage, target fabrication, and processing for the reasons described in Section 4.4.2.1.7.

4.4.8.1.8 Socioeconomics

After facility modifications, startup, and testing of the plutonium-238 reactor operation facilities at INEEL and ORR and target fabrication/processing facilities at INEEL, approximately 24 additional workers would be required to operate these facilities (24 at INEEL and none at ORR) (Hill et al. 1999). The socioeconomic impacts at INEEL are the same as those addressed in Section 4.3.2.1.8.

4.4.8.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–108** for INEEL and ORR: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual operations of HFIR at ORR and ATR and FDPF at INEEL, the projected incremental total population dose in the year 2020 would be 3.9×10^{-6} person-rem. The corresponding number of latent cancer fatalities in the populations surrounding INEEL and ORR from 35 years of operations would be 6.7×10^{-8}. The incremental total dose to the maximally exposed member of the public from annual ATR and HFIR operations would be 0 millirem because there would be no increase in radiological releases to the environment from either of these reactors associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would, therefore, be zero. The total incremental dose to the maximally exposed member of the public from annual ATR and FDPF operations would be 2.6×10^{-7} millirem. From 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 4.6×10^{-12}.

Table 4–108 Incremental Radiological Impacts on the Public Around ORR and INEEL from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 8

Receptor	ORR HFIR	INEEL			Two-Site Total
		ATR	FDPF	Total	
Population within 80 kilometers (50 miles) in the year 2020					
Dose (person-rem)	0	0	3.9×10^{-6}	3.9×10^{-6}	3.9×10^{-6}
35-year latent cancer fatalities	0	0	6.7×10^{-8}	6.7×10^{-8}	6.7×10^{-8}
Maximally exposed individual					
Annual dose (millirem)	0	0	2.6×10^{-7}	2.6×10^{-7}	NA[a]
35-year latent cancer fatality risk	0	0	4.6×10^{-12}	4.6×10^{-12}	NA[a]
Average exposed individual within 80 kilometers (50 miles)					
Annual dose[b] (millirem)	0	0	2.0×10^{-8}	2.0×10^{-8}	NA[a]
35-year latent cancer fatality risk	0	0	3.6×10^{-13}	3.6×10^{-13}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FDPF in the year 2020 (188,400).

Key: NA, not applicable.
Source: Model results, using the GENII computer code (Napier et al. 1988).

Incremental doses to involved workers from normal operations are given in **Table 4–109**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to ATR workers would be 0 millirem; for HFIR workers, the incremental annual average dose would also be 0 millirem; for FDPF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0, 0, and approximately 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–109. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–109 Incremental Radiological Impacts on Involved ORR and INEEL Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 8

Receptor—Involved Workers[a]	ORR HFIR	INEEL		Total
		ATR	FDPF	
Total dose (person-rem per year)	0	0	12[b]	12
35-year latent cancer fatalities	0	0	0.17	0.17
Average worker dose (millirem per year)	0	0	170	NA[c]
35-year latent cancer fatality risk	0	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers.
c. Values cannot be given for the average worker because the workers would be in three different facilities at two different sites.

Key: NA, not applicable.
Source: Mecham 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts for this option at INEEL would be the same as those described for Option 2 (Section 4.4.2.1.9).

Hazardous chemical impacts at ORR would be the same as those of ongoing site operations because no new chemicals are expected to be emitted from operating HFIR.

4.4.8.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with ATR and HFIR target irradiation and FDPF target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed offsite individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–110** and **4–111**, respectively. Because ATR and HFIR are currently operating, the consequences and risks are presented for both the current reactor configurations without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configurations.

For 35 years of ATR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 1.49×10^{-7} and 1.95×10^{-6}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 7.01×10^{-4}.

For 35 years of HFIR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 8.68×10^{-9} and 3.43×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 4.09×10^{-5}.

For 35 years of FDPF target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and an early fatality to a noninvolved worker would be 1.49×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0287.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 1.50×10^{-5} and 3.52×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0295.

The irradiation of neptunium-237 targets at ATR and HFIR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at ATR and HFIR.

Table 4–110 ATR, HFIR, and FDPF Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 8

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
ATR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.465	2.33×10^{-4}	5.11×10^{4}	25.5	5.15	0.00206
Large-break LOCA with 3 kg/yr plutonium-238 production	0.549	2.75×10^{-4}	5.15×10^{4}	25.7	6.52	0.00261
Target handling with 0 kg/yr plutonium-238 production[c]	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 3 kg/yr plutonium-238 production	1.23×10^{-4}	6.15×10^{-8}	0.0786	3.93×10^{-5}	0.00195	7.80×10^{-7}
HFIR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	2.41	0.00121	2,990	1.49	17.2	0.00688
Large-break LOCA with 2 kg/yr plutonium-238 production	2.41	0.00121	3,000	1.50	17.2	0.00688
Target handling with 0 kg/yr plutonium-238 production	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 2 kg/yr plutonium-238 production	4.96×10^{-4}	2.48×10^{-7}	0.335	1.68×10^{-4}	0.00245	9.80×10^{-7}
FDPF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.01×10^{-9}	1.01×10^{-12}	2.49×10^{-5}	1.24×10^{-8}	7.26×10^{-9}	2.91×10^{-12}
Target dissolver tank failure during plutonium-238 separation	6.11×10^{-8}	3.05×10^{-11}	5.65×10^{-4}	2.82×10^{-7}	2.17×10^{-7}	8.69×10^{-11}
Ion exchange explosion during plutonium-238 separation	1.63×10^{-5}	8.13×10^{-9}	0.150	7.51×10^{-5}	5.79×10^{-5}	2.31×10^{-8}
Processing facility beyond-design-basis earthquake	42.5	0.0425	1.64×10^{5}	82.0	1,200	1.0[d]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. There would be no neptunium-237 targets for this zero-production case. Thus, there would be no associated accident consequences.
d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–111 ATR, HFIR, and FDPF Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 8

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual ATR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	2.33×10^{-8}	0.00255	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production (1×10^{-4})	2.75×10^{-8}	0.00257	2.61×10^{-7}
Large-break LOCA incremental risks[c]	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target handling with 3 kg/yr plutonium-238 production[d] (0.001)	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
35-year ATR risk[e]	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
Annual HFIR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production (1×10^{-4})	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risks[c]	0.0	1.00×10^{-6}	0.0
Neptunium-237 target handling with 2 kg/yr plutonium-238 production[d] (0.001)	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
35-year HFIR risk[e]	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
Annual FDPF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Processing facility beyond-design-basis earthquake (1×10^{-5})	4.25×10^{-7}	8.20×10^{-4}	$1.00\times10^{-5(f)}$
35-year FDPF risk	1.49×10^{-5}	0.0287	3.50×10^{-4}
35-year Option risk[g]	1.50×10^{-5}	0.0295	3.52×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. There would be no neptunium-237 targets for the zero-production case. Thus, the (3 kg/yr at ATR, 2 kg/yr at HFIR) production rate target-handling risks are the incremental risks.
e. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
f. Risk of an early fatality.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

No chemical processing activities are currently performed at FDPF and no chemicals are stored in this facility. Processing activities in support of plutonium-238 production would require the introduction of hazardous chemicals, specifically nitric acid and nitric oxide. Potential health impacts from accidental releases of nitric acid were assessed by comparing estimated airborne concentrations of the chemicals to ERPG developed by the American Industrial Hygiene Association. The ERPG-1 value (0.5 part per million) is the maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour, resulting in only mild, transient, and reversible adverse health effects. The ERPG-2 value (10 parts per million) is protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. The ERPG-3 value (25 parts per million) is indicative of potentially life-threatening health effects.

The maximum distances, in meters, needed to reach the ERPG values for nitric acid releases at FDPF for Stability Classes D and F are shown in **Table 4–112**. Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions while Stability Class F represents worst-case meteorological conditions. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they were sheltered within buildings at the time of release. Individuals at the nearest highway (5,800 meters [3.6 miles]) and at the nearest site boundary (13,952 meters [8.7 miles]) from FDPF would be exposed to levels well below ERPG-1.

Table 4–112 ERPG Distances for Nitric Acid Releases at FDPF Under Alternative 2 (Use Only Existing Operational Facilities)—Option 8

Evaluation Parameter	Stability Class D (meters)	Stability Class F (meters)
ERPG-3	375	450
ERPG-2	500	600
ERPG-1	2,000	3,000

Note: To convert from meters to miles, multiply by 6.22×10^{-4}.
Key: ERPG, Emergency Response Planning Guideline.

There are no ERPG values for nitric oxide. For nitric oxide accidents, the level of concern has been estimated by using one-tenth of the "Immediately Dangerous to Life and Health" level published by the National Institute for Occupational Safety and Health. The Immediately Dangerous to Life and Health value for nitric oxide is 100 parts per million. The level of concern value used for this NI PEIS is 10 parts per million. The level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects as a result of a single exposure for a relatively short period of time.

For FDPF, the maximum distances needed to reach the level of concern for nitric oxides releases for Stability Classes D and F are 500 and 2,000 meters (0.31 and 1.24 miles), respectively. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they were sheltered within buildings at the time of release. Individual at the nearest highway (5,800 meters [3.6 miles]) and the nearest site boundary (13,952 meters [8.7 miles]) from FDPF would be exposed to levels well below the level of concern for nitric oxide.

Potential health impacts from the accidental release of the hazardous chemicals were assessed for a noninvolved worker, offsite individuals who are members of the public located at the nearest site boundary and onsite individuals who are members of the public located at the nearest highway access.

The impacts associated with the accidental release of nitric acid and nitric oxide at FDPF are presented in **Table 4–113**.

Table 4–113 FDPF Hazardous Chemical Accident Impacts Under Alternative 2 (Use Only Existing Operational Facilities)—Option 8

Receptor	Evaluation Parameter	Nitric Acid		Nitric Oxide	
		Stability Class D	Stability Class F	Stability Class D	Stability Class F
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	3.3 <ERPG-2 Mild, transient	8.4 <ERPG-2 Mild, transient	4.2 <LOC Mild, transient	67.5 >LOC Serious
Nearest highway maximally exposed individual	Parts per million Level of concern Potential health effects	0.05 < ERPG-1 None	0.15 ERPG-1 Mild, transient	0.09 < LOC None	0.87 < LOC None
Site boundary maximally exposed individual	Parts per million Level of concern Potential health effects	<<0.05 < ERPG-1 None	<<0.15 ERPG-1 Mild, transient	<<0.09 < LOC None	<<0.87 < LOC None

Note: < means "less than"; << means "much less than."
Key: ERPG, Emergency Response Planning Guideline; LOC, level of concern.
Source: Model results.

4.4.8.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FDPF target fabrication facility at INEEL. DOE would transport the unirradiated neptunium-237 targets from FDPF to HFIR at ORR, and to ATR at INEEL. Following irradiation in HFIR or ATR, the targets would be returned to FDPF for processing. After this processing, the plutonium-238 product would be shipped to LANL. The analysis is described in Appendix J.

Approximately 311 intersite shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 0.99 million kilometers (0.62 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 6 person-rem; the dose to the public, 103 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0024 latent cancer fatality among transportation workers and 0.052 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0030.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 were also evaluated and estimated to have a probability of lower than 1 in 10 million per year.

Estimates of the total ground transportation accident risks are as follows: a radiological dose to the population of 0.088 person-rem, resulting in 4.4×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.024 traffic fatality.

4.4.8.1.12 Environmental Justice

NORMAL OPERATIONS. For 35 years of normal operations under this option, the likelihood of an incremental latent cancer fatality among the population residing within 80 kilometers (50 miles) of HFIR, ATR, and FDPF would be essentially zero (derived from information in Table 4–108). As shown in Sections 4.4.2.1.9 and 4.4.8.1.9, the release of hazardous chemicals at INEEL would pose no significant risk of cancer or toxic effects among the public. As discussed in Sections K.5.1 and K.5.2, the likelihood that a latent cancer fatality would result from the ingestion of food that could be radiologically contaminated due to normal operations would be essentially zero at INEEL and ORR. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near INEEL or ORR. The likelihood of a latent cancer fatality among the public due to incident-free transportation during the 35-year project would be approximately 1 in 19, and the likelihood of a nonradiological fatality due to vehicular emissions would be approximately 1 in 330 (derived from information in Section 4.4.8.1.11).

ACCIDENTS. The number of expected latent cancer fatalities among the populations at risk due to accidents listed in Table 4–111 would be approximately 0.03. If a radiological accident were to occur at ATR or FDPF and northwesterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Fort Hall Indian Reservation (see Figure K–2). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the population or maximally exposed individual residing within the boundary of the Fort Hall Indian Reservation. In the event a radiological accident were to occur at HFIR and southerly winds prevailed at the time of the accident, radiological contamination would be directed toward the predominately minority population of the Scarboro community adjacent to the northern boundary of ORR (see Figure K–6). If the winds were blowing from the west-southwest at the time of the accident, radiological contamination would be directed toward minority populations residing in Knoxville, Tennessee. Accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the minority populations or maximally exposed individuals residing in the Scarboro community or Knoxville.

The radiological risk of a public fatality due to incident-free transportation of radioactive material would be approximately 0.052 latent cancer fatality and the risk of a fatal traffic collision during 35 years of shipments would be approximately 0.024 fatality (Section 4.4.8.1.11).

In summary, the implementation of this option would pose no significant radiological risk to persons residing in potentially affected areas or along representative transportation routes. Under the conservative assumption that all food consumed in potentially affected areas during the 35-year operational period would be radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to the ingestion of contaminated food supplies. As discussed in other parts of Section 4.4.8.1, the implementation of this option would not result in significant nonradiological impacts on populations at risk. Thus, implementation would not pose significant and adverse environmental risks to persons residing within potentially affected areas, including minority and low-income persons.

4.4.8.1.13 Waste Management

Only very small amounts of additional waste would be generated as a result of irradiating neptunium-237 targets in ATR and HFIR because these reactors would already be operating for other purposes. The anticipated incremental generation of waste from ATR operations is discussed in Section 4.4.1.1.13. The anticipated incremental generation of waste from HFIR operations is discussed in Section 4.4.7.1.13. There would be virtually no impacts on either site's waste management system as the result of neptunium-237 target irradiation.

The impacts of managing waste associated with neptunium-237 target fabrication and processing in FDPF are assumed to be the same as for Option 2 under Alternative 1 (Section 4.3.2.1.13) because the same amount of plutonium-238 would be produced annually. As shown in that section, the impacts on the waste management systems at INEEL would be minimal.

4.4.8.1.14 Spent Nuclear Fuel Management

No incremental impacts would be associated with the management of spent nuclear fuel (refer to Section 4.4.1.1.14).

4.4.8.2 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.4.9 Alternative 2 (Use Only Existing Operational Facilities)—Option 9

This option involves operating HFIR at ORR and ATR at INEEL to irradiate neptunium-237 targets, and operating FMEF at Hanford to both fabricate and process these targets and to store the neptunium-237 transported to Hanford from SRS.

The transportation of the neptunium-237 from SRS to Hanford for processing and fabrication into neptunium-237 targets in FMEF, the transportation of targets from Hanford to both INEEL and ORR for irradiation in ATR and HFIR, respectively, the transportation of the irradiated targets back to Hanford for postirradiation processing in FMEF, and the transportation of the plutonium-238 product from Hanford to LANL also constitute part of this option.

All options under this alternative include the permanent deactivation of FFTF at Hanford.

4.4.9.1 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.4.9.1.1 Land Resources

LAND USE. The irradiation of neptunium-237 targets in ATR would not result in impacts on land use at INEEL for the reasons described in Section 4.4.1.1.1.

The irradiation of neptunium-237 targets would also take place within the existing HFIR facility at ORR. Impacts on land use at ORR would not result for the reasons described in Section 4.4.7.1.1.

Impacts on land use at Hanford from neptunium-237 storage, target fabrication, and processing at FMEF would be expected to be minimal for the reasons described in Section 4.4.3.1.1.

VISUAL RESOURCES. The irradiation of neptunium-237 targets in ATR would not result in impacts on visual resources at INEEL for the reasons described in Section 4.4.1.1.1.

The irradiation of neptunium-237 targets would also take place within the existing HFIR facility at ORR. There would be no impacts on visual resources at ORR for the reasons described in Section 4.4.7.1.1.

Impacts on visual resources at Hanford from neptunium-237 storage, target fabrication, and processing at FMEF would be expected to be minimal for the reasons described in Section 4.4.3.1.1.

4.4.9.1.2 Noise

The irradiation of neptunium-237 targets in ATR would not result in a change in noise impacts at INEEL for the reasons described in Section 4.4.1.1.2.

The irradiation of neptunium targets would also take place in HFIR. No change in noise impacts at ORR would be expected for the reasons described in Section 4.4.7.1.2.

Noise impacts at Hanford would be expected to be minimal from neptunium-237 storage, target fabrication, and processing at FMEF and changes in traffic noise would be small for the reasons described in Section 4.4.3.1.2.

4.4.9.1.3 Air Quality

Impacts for this option at INEEL would be the same as those described for Option 3 (Section 4.4.3.1.3).

Impacts for this option at ORR would be the same as those described for Option 8 (Section 4.4.8.1.3).

Impacts for this option at Hanford would be the same as those described for Option 3 (Section 4.4.3.1.3).

The air quality impacts of transportation among SRS, INEEL, ORR, Hanford, and LANL are presented in Section 4.4.9.1.11.

4.4.9.1.4 Water Resources

Impacts for this option at INEEL would be the same as those described for Option 1 (Section 4.4.1.1.4).

Impacts for this option at ORR would be the same as those described for Option 7 (Section 4.4.7.1.4).

Impacts for this option at Hanford would be the same as those described for Option 3 (Section 4.4.3.1.4).

4.4.9.1.5 Geology and Soils

The use of ATR to irradiate neptunium-237 targets at INEEL would not be expected to result in impacts on geologic or soil resources, nor be jeopardized by large-scale geologic conditions, for the reasons described in Section 4.4.1.1.5.

Dual use of HFIR at ORR to irradiate neptunium-237 targets would also not be expected to result in impacts on geologic and soil resources, nor be jeopardized by large-scale geologic conditions, for the reasons described in Sections 4.2.2.2.5 and 4.4.7.1.5.

Impacts on geologic and soil resources at Hanford would not be expected from neptunium-237 storage, target fabrication, and processing at FMEF for the reasons described in Sections 4.3.3.1.5 and 4.4.3.1.5. Large-scale geologic conditions also present a low risk to FMEF operations, as further discussed in Section 4.2.4.2.5. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards will be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.4.9.1.6 Ecological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on ecological resources at INEEL for the reasons described in Section 4.4.1.1.6.

The irradiation of neptunium targets would also take place in HFIR. There would be no impacts on ecological resources at ORR for the reasons described in Section 4.4.7.1.6.

There would be no impacts on ecological resources at Hanford from neptunium-237 storage, target fabrication, and processing at FMEF for the reasons described in Section 4.4.3.1.6.

4.4.9.1.7 Cultural and Paleontological Resources

The irradiation of neptunium-237 targets in ATR would not result in impacts on cultural and paleontological resources at INEEL for the reasons described in Section 4.4.1.1.7.

The irradiation of neptunium targets would also take place in HFIR. There would be no impacts on cultural and paleontological resources at ORR for the reasons described in Section 4.4.7.1.7.

There would be no impacts on cultural and paleontological resources at Hanford from neptunium-237 target fabrication and processing at FMEF for the reasons described in Section 4.4.3.1.7.

4.4.9.1.8 Socioeconomics

After facility modifications, startup, and testing of the plutonium-238 reactor operation facilities at INEEL and ORR and target fabrication/processing facilities at Hanford, approximately 62 additional workers would be required to operate these facilities (none at INEEL and ORR and 62 at Hanford) (Hoyt et al. 1999). The socioeconomic impacts at Hanford are the same as those addressed in Section 4.3.3.1.8.

4.4.9.1.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from operations are given in **Table 4–114** for INEEL, ORR, and Hanford: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

Table 4–114 Incremental Radiological Impacts on the Public Around INEEL, ORR, and Hanford from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 9

Receptor	INEEL ATR	ORR HFIR	Hanford FMEF	Three-Site Total
Population within 80 kilometers (50 miles) in the year 2020				
Dose (person-rem)	0	0	4.4×10^{-5}	4.4×10^{-5}
35-year latent cancer fatalities	0	0	7.7×10^{-7}	7.7×10^{-7}
Maximally exposed individual				
Annual dose (millirem)	0	0	4.7×10^{-7}	NA[a]
35-year latent cancer fatality risk	0	0	8.3×10^{-12}	NA[a]
Average exposed individual within 80 kilometers (50 miles)				
Annual dose[b] (millirem)	0	0	8.9×10^{-8}	NA[a]
35-year latent cancer fatality risk	0	0	1.6×10^{-12}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at three different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FMEF in the year 2020 (494,400).

Key: NA, not applicable.
Source: Model results, using the GENII computer code (Napier et al. 1988).

As a result of annual operations of ATR at INEEL, HFIR at ORR, and FMEF at Hanford, the projected incremental total population dose in the year 2020 would be 4.4×10^{-5} person-rem. The corresponding number of latent cancer fatalities in the populations surrounding INEEL, ORR, and Hanford from 35 years of operations would be 7.7×10^{-7}. The total incremental dose to the maximally exposed members of the public from annual ATR and HFIR operations would be 0 millirem because there would be no increase in radiological releases to the environment from either of these reactors associated with this option. From 35 years of operations, the corresponding risk of a latent cancer fatality to these individuals would, therefore, be zero. The incremental dose to the maximally exposed member of the public from annual FMEF operations would be 4.7×10^{-7} millirem. From 35 years of operations, the corresponding risk of a latent fatal cancer to this individual would be 8.3×10^{-12}.

Incremental doses to involved workers from normal operations are given in **Table 4–115**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to ATR and HFIR workers would be 0 millirem; for FMEF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities would be 0, 0, and approximately 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–115. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–115 Incremental Radiological Impacts on Involved INEEL, ORR, and Hanford Workers from Operational Facilities Under Alternative 2 (Use Only Existing Operational Facilities)—Option 9

Receptor—Involved Workers[a]	INEEL ATR	ORR HFIR	Hanford FMEF	Three-Site Total
Total dose (person-rem per year)	0	0	12[b]	12
35-year latent cancer fatalities	0	0	0.17	0.17
Average worker dose (millirem per year)	0	0	170	NA[c]
35-year latent cancer fatality risk	0	0	0.0023	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers.
c. Values cannot be given for the average worker because the workers would be at three different facilities and sites.

Key: NA, not applicable.
Source: Mecham 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts for this option at INEEL would be the same as those of ongoing site operations because no new chemicals are expected to be emitted at ATR.

Hazardous chemical impacts for this option at ORR were determined to be the same as those of ongoing site operations because no new chemicals are expected to be emitted at HFIR.

Hazardous chemical impacts for this option at Hanford would be the same as those described for Option 3 (Section 4.4.3.1.9).

4.4.9.1.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with ATR and HFIR target irradiation and FMEF target processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Estimates of radiological consequences have been developed for the maximally exposed individual, the offsite population within 80 kilometers (50 miles) of the facility, and a noninvolved worker at a distance of 640 meters (0.4 mile) from the release point. Consequences are presented in terms of radiological dose (in rem) and the probability that the dose would result in a latent cancer fatality. Accident risk is defined as the product of the accident probability (i.e., accident frequency) and the accident consequence. In this NI PEIS, risk is expressed as the increased likelihood of a latent cancer fatality per year for an individual (the maximally exposed offsite individual or a noninvolved worker), and as the increased number of latent cancer fatalities per year in the offsite population. The probability coefficients for determining the likelihood of a latent cancer fatality, given a dose, are given in Section 4.2.1.2.10. Consequences to involved workers are addressed in Section I.1.7.

To provide a better indication of risks from the postulated accidents, the risks are summed for each facility and also for each option. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk from accidents, a full-scope probabilistic risk analysis would be required for each facility. Since full-scope probabilistic risk analyses are not available to incorporate in this NI PEIS, summing the spectrum of accident risks was considered appropriate for the purposes of this NI PEIS. Details of the risk summation calculations are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–116** and **4–117**, respectively. Because ATR and HFIR are currently operating, the consequences and risks are presented for both the current reactor configurations without neptunium-237 targets and for the worst-case neptunium-237 target-loading reactor configurations.

For 35 years of ATR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 1.49×10^{-7} and 1.95×10^{-6}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 7.01×10^{-4}.

Table 4–116 ATR, HFIR, and FMEF Accident Consequences Under Alternative 2 (Use Only Existing Operational Facilities)—Option 9

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
ATR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	0.465	2.33×10^{-4}	5.11×10^{4}	25.5	5.15	0.00206
Large-break LOCA with 3 kg/yr plutonium-238 production	0.549	2.75×10^{-4}	5.15×10^{4}	25.7	6.52	0.00261
Target handling with 0 kg/yr plutonium-238 production[c]	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 3 kg/yr plutonium-238 production	1.23×10^{-4}	6.15×10^{-8}	0.0786	3.93×10^{-5}	0.00195	7.80×10^{-7}
HFIR accidents						
Large-break LOCA with 0 kg/yr plutonium-238 production	2.41	0.00121	2,990	1.49	17.2	0.00688
Large-break LOCA with 2 kg/yr plutonium-238 production	2.41	0.00121	3,000	1.50	17.2	0.00688
Target handling with 0 kg/yr plutonium-238 production	0.0	0.0	0.0	0.0	0.0	0.0
Target handling with 2 kg/yr plutonium-238 production	4.96×10^{-4}	2.48×10^{-7}	0.335	1.68×10^{-4}	0.00245	9.80×10^{-7}
FMEF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.02×10^{-9}	1.01×10^{-12}	7.26×10^{-5}	3.63×10^{-8}	6.65×10^{-10}	2.66×10^{-13}
Target dissolver tank failure during plutonium-238 separation	4.64×10^{-8}	2.32×10^{-11}	0.00169	8.47×10^{-7}	1.95×10^{-8}	7.81×10^{-12}
Ion exchange explosion during plutonium-238 separation	1.24×10^{-5}	6.18×10^{-9}	0.451	2.25×10^{-4}	5.20×10^{-6}	2.08×10^{-9}
Processing facility beyond-design-basis earthquake	16.5	0.00823	6.41×10^{5}	321	921	1.0[d]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. There would be no neptunium-237 targets for this zero-production case. Thus, there would be no associated accident consequences.
d. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–117 ATR, HFIR, and FMEF Accident Risks Under Alternative 2 (Use Only Existing Operational Facilities)—Option 9

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual ATR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	2.33×10^{-8}	0.00255	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production (1×10^{-4})	2.75×10^{-8}	0.00257	2.61×10^{-7}
Large-break LOCA incremental risks[c]	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target handling with 3 kg/yr plutonium-238 production[d] (0.001)	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
35-year ATR risk[e]	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
Annual HFIR risks			
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production (1×10^{-4})	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risks[c]	0.0	1.00×10^{-6}	0.0
Neptunium-237 target handling with 2 kg/yr plutonium-238 production[d] (0.001)	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
35-year HFIR risk[e]	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
FMEF accidents			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Processing facility beyond-design-basis earthquake (1×10^{-5})	8.23×10^{-8}	0.00321	1.00×10^{-5}[f]
35-year FMEF risk	2.88×10^{-6}	0.112	3.50×10^{-4}
35-year Option risk[g]	2.88×10^{-6}	0.113	3.50×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. The incremental risk from irradiation of neptunium-237 targets in a currently operating reactor is determined by subtracting the risk of operating without targets from the risk of operating with targets.
d. There would be no neptunium-237 targets for the zero-production case. Thus, the (3 kg/yr at ATR, 2kg/yr at HFIR) production rate target-handling risks are the incremental risks.
e. The 35-year risk is determined by summing the incremental annual risks and then multiplying by 35.
f. Risk of an early fatality.
g. Individual risks are summed only for colocated individuals. The highest individual risk was used to represent the 35-year option risk.

Note: To convert from kilograms per year to pounds per year, multiply by 2.20.
Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

For 35 years of HFIR target irradiation, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be 8.68×10^{-9} and 3.43×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 4.09×10^{-5}.

For 35 years of FMEF target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed offsite individual and of an early fatality to a noninvolved worker would be 2.88×10^{-6} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.112.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 3.04×10^{-6} and 3.52×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.113.

The irradiation of neptunium-237 targets at ATR and HFIR would not introduce any additional operations that require the use of hazardous chemicals. Thus, there are no postulated hazardous chemical accidents attributable to the irradiation of neptunium-237 targets at ATR and HFIR.

No chemical processing activities are currently performed at FMEF and no chemicals are stored in this facility. Processing activities in support of plutonium-238 production would require the introduction of hazardous chemicals, specifically nitric acid and nitric oxide. Potential health impacts from accidental releases of nitric acid were assessed by comparing estimated airborne concentrations of the chemicals to ERPG developed by the American Industrial Hygiene Association. The ERPG-1 value (0.5 part per million) is the maximum airborne concentration below which nearly all individuals could be exposed for up to 1 hour, resulting in only mild, transient, and reversible adverse health effects. The ERPG-2 value (10 parts per million) is protective of irreversible or serious health effects or impairment of an individual's ability to take protective action. The ERPG-3 value (25 parts per million) is indicative of potentially life-threatening health effects.

The maximum distances, in meters, needed to reach the ERPG values for nitric acid releases at FMEF for Stability Classes D and F are shown in **Table 4–118**. Two separate atmospheric conditions were evaluated, Stability Classes D and F. Stability Class D represents average meteorological conditions while Stability Class F represents worst-case meteorological conditions. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they were sheltered within buildings at the time of release. Individuals at the nearest highway (7,100 meters [4.4 miles]) and at the nearest site boundary (7,210 meters [4.5 miles]) from FDPF would be exposed to levels well below ERPG-1.

Table 4–118 ERPG Distances for Nitric Acid Releases at FMEF Under Alternative 2 (Use Only Existing Operational Facilities)—Option 9

Evaluation Parameter	Stability Class D (meters)	Stability Class F (meters)
ERPG-3	375	450
ERPG-2	500	600
ERPG-1	2,000	3,000

Note: To convert from meters to miles, multiply by 6.22×10^{-4}.
Key: ERPG, Emergency Response Planning Guideline.

There are no ERPG values for nitric oxide. For nitric oxide accidents, the level of concern has been estimated by using one-tenth of the "Immediately Dangerous to Life and Health" level published by the National Institute for Occupational Safety and Health. The Immediately Dangerous to Life and Health value for nitric oxide is 100 parts per million. The level of concern value used for this NI PEIS is 10 parts per million. The level of

concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects as a result of a single exposure for a relatively short period of time.

For FMEF, the maximum distances needed to reach the level of concern for nitric oxides releases for Stability Classes D and F are 500 and 1,900 meters (0.31 and 1.18 miles), respectively. The number of involved and noninvolved workers potentially exposed would vary with a number of factors, such as the time of day and whether they were sheltered within buildings at the time of release. Individual at the nearest highway (7,100 meters [4.4 miles]) and the nearest site boundary (7,210 meters [4.5 miles]) from FMEF would be exposed to levels well below the level of concern for nitric oxide.

Potential health impacts from the accidental release of the hazardous chemicals were assessed for a noninvolved worker, offsite individuals who are members of the public located at the nearest site boundary and onsite individuals who are members of the public located at the nearest highway access.

The impacts associated with the accidental release of nitric acid and nitric oxide at FMEF are presented in **Table 4–119**.

Table 4–119 FMEF Hazardous Chemical Accident Impacts Under Alternative 2 (Use Only Existing Operational Facilities)—Option 9

Receptor	Evaluation Parameter	Nitric Acid		Nitric Oxide	
		Stability Class D	Stability Class F	Stability Class D	Stability Class F
Noninvolved worker (640 meters)	Parts per million Level of concern Potential health effects	3.3 <ERPG-2 Mild, transient	8.4 <ERPG-2 Mild, transient	4.2 <LOC Mild, transient	66 >LOC Serious
Nearest highway maximally exposed individual	Parts per million Level of concern Potential health effects	0.03 < ERPG-1 None	0.1 ERPG-1 Mild, transient	0.09 < LOC None	0.55 < LOC None
Site boundary maximally exposed individual	Parts per million Level of concern Potential health effects	0.03 < ERPG-1 None	0.1 ERPG-1 Mild, transient	0.09 < LOC None	0.53 < LOC None

Note: < means "less than."
Key: ERPG, Emergency Response Planning Guideline; LOC, level of concern.
Source: Model results.

4.4.9.1.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FMEF target fabrication facility at Hanford. DOE would transport the unirradiated neptunium-237 targets from FMEF to HFIR at ORR, and to ATR at INEEL. Following irradiation in HFIR or ATR, the targets would be returned to FMEF for processing. After this processing, the plutonium-238 product would be shipped to LANL. The analysis is described in Appendix J.

Approximately 689 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 1.6 million kilometers (0.99 million miles).

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 9 person-rem; the dose to the public, 167 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0036 latent cancer fatality among transportation workers and 0.084 latent cancer fatality in the total affected

population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option is 0.0037.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FMEF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 were also evaluated and estimated to have a probability of lower than 1 in 10 million per year.

Estimates of the total ground transportation accident risks are as follows: a radiological dose to the population of 0.06 person-rem, resulting in 3.0×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.04 traffic fatality.

4.4.9.1.12 Environmental Justice

NORMAL OPERATIONS. For 35 years of normal operations under this option, the likelihood of an incremental latent cancer fatality among the population residing within 80 kilometers (50 miles) of HFIR, ATR, and FMEF would be essentially zero (derived from information in Table 4–114). As shown in Sections 4.4.3.1.9 and 4.4.9.1.9, the release of hazardous chemicals at Hanford would pose no significant risk of cancer or toxic effects among the public. As discussed in Sections K.5.1, K.5.2, and K.5.3, the risk that would result from the ingestion of food that could be radiologically contaminated due to normal operations would be essentially zero at INEEL and ORR, and approximately 0.001 latent cancer fatality at Hanford. No credible pattern of food consumption by persons residing in potentially affected areas would result in significant health risks due to radiological contamination of food supplies near INEEL, ORR, and Hanford. As discussed in Section 4.4.9.1.11, no fatalities would be expected to result from incident-free transportation activities.

ACCIDENTS. The number of expected latent cancer fatalities among populations at risk due to accidents listed in Table 4–117 would be approximately 0.11. If a radiological accident were to occur at ATR and northwesterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Fort Hall Indian Reservation (see Figure K–2). In the event a radiological accident were to occur at HFIR and southerly winds prevailed at the time of the accident, radiological contamination would be directed toward the predominately minority population of the Scarboro community adjacent to the northern boundary of ORR (see Figure K–6). If the winds were blowing from the west-southwest at the time of the accident, radiological contamination would be directed toward minority populations residing in Knoxville, Tennessee. If a radiological accident were to occur at FMEF and northeasterly winds prevailed at the time of the accident, radiological contamination from the accident would be directed toward the Yakama Indian Reservation (see Figure K–11). However, accidents that could occur under the implementation of this option would not be expected to result in a latent cancer fatality among the populations or maximally exposed individuals residing near or within the boundaries of the Fort Hall Indian Reservation, the Scarboro community, Knoxville, or the Yakama Indian Reservation.

As discussed in Section 4.4.9.1.11, no fatalities would be expected to result from transportation accidents.

In summary, the implementation of this option would pose no significant radiological risk to persons residing in potentially affected areas or along representative transportation routes. Under the conservative assumption that all food consumed in potentially affected areas during the 35-year operational period would be

radioactively contaminated, no credible pattern of food consumption would pose a significant radiological health risk due to the ingestion of contaminated food supplies. As discussed in other parts of Section 4.4.9.1, the implementation of this option would not result in significant nonradiological impacts on populations at risk. Thus, implementation would not pose significant and adverse environmental risks to persons residing within potentially affected areas, including minority and low-income persons.

4.4.9.1.13 Waste Management

Only very small amounts of additional waste would be generated as a result of irradiating neptunium-237 targets in ATR and HFIR because these reactors would already be operating for other purposes. The anticipated incremental generation of waste from ATR operations is discussed in Section 4.4.1.1.13. The anticipated incremental generation of waste from HFIR operations is discussed in Section 4.4.7.1.13. There would be virtually no impacts on either site's waste management systems as the result of neptunium-237 target irradiation.

The impacts of managing waste associated with neptunium-237 target fabrication and processing in FMEF are assumed to be the same as for Option 3 (Section 4.4.3.1.13) because the same amount of plutonium-238 would be produced annually. As shown in that section, the impacts on the waste management systems at Hanford would be minimal.

4.4.9.1.14 Spent Nuclear Fuel Management

No incremental impacts would be associated with the management of spent nuclear fuel (refer to Section 4.4.1.1.14).

4.4.9.2 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.5 ALTERNATIVE 3—CONSTRUCT NEW ACCELERATOR(S)

Under Alternative 3, one or two new accelerators would be used for target irradiation for the evaluation period of 35 years. The new accelerator(s) which would be constructed at an existing DOE site,[1] would be used to irradiate all of the targets (i.e., for production of plutonium-238, isotopes for medical and industrial uses, and materials testing for research and development). Ongoing operations at existing facilities as described in Chapter 3, Affected Environment, would continue.

The targets for plutonium-238 production would be fabricated in one of the three candidate facilities at ORNL, INEEL, or Hanford. The material needed for the target fabrication (neptunium-237) would be transported from SRS to the fabrication facilities. The targets would be irradiated at the new high-energy accelerator facility and transported back to the target fabrication facilities for postirradiation processing.

Targets for medical and industrial isotope production would be fabricated in a new support facility located at the same site as the low-energy accelerator. The targets would be irradiated in the low-energy accelerator and returned to the new support facility for postirradiation processing. Site selection for Alternative 3 is not evaluated as part of this NI PEIS. Because Alternative 3 is evaluated at a generic DOE site, no credit was taken for any support infrastructure existing at the site and it was postulated that a new support facility would be required to support operation of the low-energy accelerator and its missions and the high-energy accelerator civilian nuclear energy research and development missions if both accelerators are located on the same site. While this approach bounds the environmental impact assessment for the implementation of Alternative 3, it overstates the impacts because this NI PEIS integrates the impacts associated with constructing a new support facility and infrastructure that may be available at the existing DOE site. In the event that Alternative 3 or the low-energy accelerator alone is selected by the Record of Decision for subsequent consideration, follow-on NEPA assessments would evaluate potential locations for either both accelerators or one of the accelerators. It is unlikely that DOE would consider locating the new low-energy or high-energy accelerator on a DOE site that does not have existing infrastructure capable of supporting all or most of the mission requirements. If the accelerator(s) were built on a DOE site with existing support facilities, the environmental impacts of such implementation could be determined by subtracting the construction and decommissioning impacts associated with the new support facility from the total impacts given for this alternative.

Under Alternative 3, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for storage, target fabrication, target irradiation, postirradiation processing, and the final destination of the plutonium-238. Alternative 3 also would include decontamination and decommissioning of the accelerator(s) and the support facility when the missions are over, as well as deactivation of FFTF at Hanford.

The low-energy accelerator would serve as a dedicated isotope production facility. Due to the nature of this type of accelerator, it could only produce a limited number of isotopes (listed in Table 1–1), has no ability to satisfy the plutonium-238 needs, and a very limited ability to support the proposed nuclear-based research and development needs. The preconceptual design of the high-energy accelerator presented in Appendix F focused on supporting the plutonium-238 production mission. The design of the high-energy accelerator could be refined and expanded to perform additional missions such as the production of a select set of medical and industrial radioisotopes. In addition, DOE is aware of longer-term concepts that would apply high-energy accelerators to produce "tuneable" neutrons in a subcritical assembly. Such a facility could be used to address some of the missions more familiar to reactor facilities and may hold considerable promise for future science

[1] If two accelerators were constructed, they could be located at different sites. However, to bound the environmental impacts at a generic site, the assessments in this section assume their location at a single site.

and technology research. A facility of this nature could provide unique capabilities in areas such as the testing of many different nuclear system coolant, fuel, and material interactions. The changes required to add additional capability to the high-energy accelerator could be provided, but they would increase the size of the facility, add complexity to the facility design and operation, increase the cost of construction and operation, and potentially require more time for design and construction.

The three options under this alternative and their associated target fabrication, postirradiation processing, and transportation activities are discussed below.

- **Option 1.** REDC at ORNL would be used to fabricate and process the neptunium-237 targets required for plutonium-238 production. The neptunium-237 transported from SRS to ORNL would be stored at REDC. The plutonium-238 product would be transported from ORNL to LANL for use in radioisotope power systems for future U.S. space missions. A new support facility at an existing DOE site would be used to fabricate and process the targets required for the production of medical and industrial and research isotopes and to store the materials needed for target fabrication.

- **Option 2.** FDPF at INEEL would be used to fabricate and process the neptunium-237 targets associated with plutonium-238 production. The neptunium-237 transported from SRS to INEEL would be stored in FDPF or Building CPP–651 at INEEL. The plutonium-238 product would be transported from INEEL to LANL for use in radioisotope power systems for future U.S. space missions. A new support facility at an existing DOE site would be used to fabricate and process the targets required to produce medical and industrial and research isotopes and to store the materials needed for target fabrication.

- **Option 3.** FMEF at Hanford would be used to fabricate and process the neptunium-237 targets for plutonium-238 production. The neptunium-237 transported from SRS to Hanford would be stored in FMEF. The plutonium-238 product would be transported from Hanford to LANL. A new support facility at an existing DOE site would be used to fabricate and process the targets required for the production of medical and industrial and research isotopes and to store the materials needed for target fabrication.

The incremental environmental impacts associated with each option are presented separately for the high-energy accelerator, the low-energy accelerator, and the support facility because combinations of these facilities may be selected for implementation. This segmentation assists in the selection of facilities from any of the possible combinations.

4.5.1 Alternative 3 (Construct New Accelerator[s])—Option 1

Option 1 involves constructing and operating one or two accelerators to irradiate all targets associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating REDC at ORR to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and conducting and operating the support facility to fabricate and process the other targets and materials and to process the associated products. This option includes storage in REDC of the neptunium-237 transported to ORR from SRS and storage in the new support facility of the other target materials transported to the generic site from other offsite facilities.

The transportation of the neptunium-237 from SRS to ORR and then to the generic site, the transportation of the other target materials to the generic site, and the transportation of plutonium-238 and other product materials following irradiation and postirradiation processing constitute part of this option.

All options under this alternative include the decontamination and decommissioning of the accelerator(s) and support facility at the generic site following their operational lifetimes, and also the permanent deactivation of FFTF at Hanford.

4.5.1.1 Construction of the New Accelerator(s) and Support Facility

The environmental impacts associated with the construction of one or two new accelerators and a support facility at the generic DOE site are assessed in this section. If the accelerator(s) were built on a site with existing support facilities, there would be no impacts associated with constructing a new support facility.

4.5.1.1.1 Land Resources

LAND USE. The construction of a low-energy accelerator, a high-energy accelerator, and a support facility would require 4 hectares (10 acres), 20.2 hectares (50 acres), and 2.4 hectares (6 acres), respectively (TechSource 2000; Herrington 2000; SAIC 2000). Since the exact nature of the construction site for any of these facilities is not known at this time (e.g., whether it has been previously disturbed or not), potential effects on land use cannot be determined. In general, if a location in a previously developed portion of a generic DOE site were selected, impacts on land use would be minimal. However, if an undisturbed location were chosen, land use would change from its present designation to industrial. If the accelerator(s) alternative were selected, tiered NEPA documentation would permit an exact determination of impacts on land use.

VISUAL RESOURCES. Impacts from construction of one or two accelerators and a support facility to visual resources at a generic DOE site would depend on the specific location selected. Impacts could include a change in the present Visual Resource Management rating of the site and/or increase in visibility of the site from offsite locations due to the presence of new structures. If construction took place on undeveloped land, the Visual Resource Management rating could change from Class II or III (ratings typical of undeveloped portions of many DOE sites) to Class IV. If a previously developed location were chosen for the accelerator(s), the Visual Resource Management rating would remain Class IV. In either case, new facilities may impact the view from offsite locations by increasing the industrial nature of the viewshed. This impact would be more likely at a western site due to the generally level terrain and sparse vegetation. Specific impacts on visual resources would be determined in tiered NEPA documentation if the accelerator(s) alternative were selected.

4.5.1.1.2 Noise

The construction of high-energy and/or low-energy accelerators would result in some increase in noise levels from the use of earthmoving, materials handling, and impact equipment; employee vehicles; and truck traffic. Noise from construction activities, especially impulsive noise, would be expected to disturb wildlife in the immediate area of the construction site. The change in noise levels in areas outside the DOE site would be dependent on the location selected and the exact nature of the construction location and activities required. However, generally if the location selected were within one of the larger DOE sites and more centrally located within the site, offsite noise impacts from construction activities would be small. Construction employee vehicles and truck traffic would result in an increase in traffic noise along roads used to access the site. However, this increase in traffic noise would be small unless the construction traffic volume were as large as the existing site traffic. Site-specific analysis would be conducted in tiered NEPA documentation if the accelerator(s) alternative were selected.

The construction of a support facility would result in some increase in noise levels from the use of earthmoving, materials handling, and impact equipment; employee vehicles; and truck traffic. Noise from construction activities, especially impulsive noise, would be expected to disturb wildlife in the immediate area of the construction site. The change in noise levels in areas outside the DOE site would be dependent on the

location selected and the exact nature of the construction location and activities required. However, generally if the location selected were within one of the larger DOE sites and more centrally located within the site, offsite noise impacts from construction activities would be small. Construction employee vehicles and truck traffic would result in an increase in traffic noise along roads used to access the site. However, this increase in traffic noise would be small unless the construction traffic volume were as large as the existing site traffic. Site-specific analysis would be conducted in tiered NEPA documentation if the accelerator(s) alternative were selected.

4.5.1.1.3 Air Quality

High-Energy Accelerator. Construction of the high-energy accelerator would result in an increase in air quality impacts from employee vehicles, trucks, and construction equipment. Criteria pollutant concentrations for construction of the high-energy accelerator were modeled and compared to the most stringent standards (**Table 4–120**). The maximum ground-level concentrations that would result from high-energy accelerator construction would be well below the ambient air quality standards, although concentrations of some pollutants (i.e., PM_{10} and nitrogen oxide) would be relatively high. Therefore, if the accelerator were in an area that already had high background pollutant concentrations, resultant pollutant concentrations could approach or exceed the ambient standards. Regulatory compliance would need to be assessed on a case-by-case basis. Hazardous chemical emissions from construction activities have not been identified.

Table 4–120 Incremental Concentrations Associated with High-Energy Accelerator Construction Under All Options of Alternative 3 (Construct New Accelerator[s])

Pollutant	Averaging Period	Most Stringent Standard or Guideline (microgram per cubic meter)[a]	Modeled Increment (microgram per cubic meter)
Carbon monoxide	8 hours 1 hour	10,000 40,000	436 623
Nitrogen oxide	Annual	100	42
PM_{10}	Annual 24 hours	50 150	3 69
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	3 64 143

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

Source: Modeled increments are based on SCREEN3 computer code (EPA 1995); data from TechSource 2000.

Modeling was based on a construction area of 100,000 square meters (1,080,000 square feet). The site was modeled as an area source with emissions occurring at a height of 3 meters (9.8 feet). A boundary limit of 3,200 meters (2 miles) was assumed for a generic site.

Low-Energy Accelerator. Given the small size of the low-energy accelerator (about 20 percent of the construction size of the high-energy accelerator), emissions of air pollutants from construction would be very small in comparison with the high-energy accelerator. The maximum ground level concentrations that would result from low-energy accelerator construction would be well below the ambient air quality standards. Nevertheless, regulatory compliance would need to be assessed on a case-by-case basis.

Support Facility for Accelerator(s). Given the small size of the support facility in comparison with the accelerators (about 10 percent of the construction size of the high-energy accelerator), emissions of air

pollutants from construction would be very small in comparison with the high-energy accelerator construction. The maximum ground level concentrations that would result from accelerator support facility construction would be 10 percent of the values given in Table 4–120.

4.5.1.1.4 Water Resources

The estimated effects on key water resource indicators associated with constructing the new accelerator(s) and support facility are presented in **Table 4–121.**

Table 4–121 Estimated Water Use and Wastewater Generation Associated with Constructing New Accelerator(s) and Support Facility Under All Options of Alternative 3 (Construct New Accelerator[s])

Indicator (million liters per year)	Accelerators[a]		
	Low-Energy	High-Energy	New Support Facility[a]
Water use	14.0	22.7	14.6
Sanitary wastewater generation	1.5	11.4	3.6

a. These estimates are annualized values based on projected construction/preoperational testing periods for the low-energy accelerator, high-energy accelerator, and new support facility of 3 years, 4.5 years, and 4.5 years, respectively.

Note: To convert from liters per year to gallons per year, multiply by 0.264.

Source: SAIC 2000; Snead 2000; TechSource 2000.

Water would be expected to be required for such uses as mixing concrete, dust control, washing activities, and for potable and sanitary needs. These estimates are annual average values over the forecasted construction periods; these values do not include dewatering of excavations that could be required at some sites. The exact impact of these withdrawals on the resource would depend on the water source (surface water or groundwater) and its relative abundance. These factors would be used to determine the impact on the local and/or regional availability of the resource. Impacts would be expected to be small to negligible due to the relatively small volumes of water required for construction compared to expected site availability.

Sanitary wastewater would be generated by construction personnel and also by facility staff during preoperational testing. Process wastewater could also be generated during construction, associated with facility cold-startup and testing of auxiliary systems as construction progresses (e.g., cooling towers). The site selected would use existing infrastructure; nearby wastewater treatment facilities would be used to the extent possible, supplemented by portable or temporary facilities during construction, as necessary. The potential impact on water resources would depend on the availability and capacity of appropriate treatment facilities. All wastewater would be disposed of in accordance with applicable regulatory requirements with discharges to surface waters in accordance with National Pollutant Discharge Elimination System (NPDES) effluent limitations.

Ground disturbance and runoff from denuded areas could potentially impact surface water quality near construction areas (Section 4.5.1.1.6). However, appropriate spill prevention practices and soil erosion and sediment control measures (e.g., silt fences, mulching disturbed areas) would be employed during construction to minimize water quality impacts.

Some locations on a generic DOE site could potentially be affected by flooding requiring appropriate siting decisions (Section 3.6.4). Applicable regulatory requirements would be followed in siting facilities including Executive Order 11988, *Floodplain Management*.

Although specific impacts on water resources cannot be determined at this time, site-specific analysis would be conducted in tiered NEPA documentation if the accelerator(s) alternative were selected.

4.5.1.1.5 Geology and Soils

Construction of the high-energy accelerator would disturb a total of approximately 20.2 hectares (50 acres) of land, with construction of the low-energy accelerator disturbing about 4 hectares (10 acres) of land (TechSource 2000). Construction of the support facility would disturb an additional 2.4 hectares (6 acres) of land (Herrington 2000; SAIC 2000). Construction impacts on geologic and soil resources cannot be determined at this time since they are site specific in nature. However, impacts would be expected to be less if previously disturbed land were used than if an undeveloped area were selected for construction.

In general, construction activities would likely require appreciable quantities of sand and gravel and possibly other geologic materials and, depending on the site chosen, could temporarily deplete local deposits or stockpiles of these materials. Soil erosion potential is also closely related to the amount of land disturbed.

As discussed in Section 3.6.5, the proposed facilities could be located at a generic DOE site with seismic activity ranging from low to moderate. Known capable faults could be located within 19 kilometers (12 miles). However, no known large-scale geologic conditions are present at any generic DOE site that would preclude the construction and operation of properly designed facilities. Appropriate activities and subsurface investigations would be conducted to identify geologic hazards including seismic and volcanic features and other natural hazards (landslide areas, sinkholes, unstable soils) as part of the site selection process. As stated in DOE Order 420.1, DOE requires that nuclear or nonnuclear facilities be designed, constructed, and operated so that the public, the workers, and the environment are protected from the adverse impacts of natural phenomena hazards, including earthquakes. DOE Order 420.1, Section 4.4, as supplemented by DOE Guide 420.1-2, stipulates the natural phenomena hazards mitigation requirements for DOE facilities. Further, the natural phenomena hazards mitigation requirements of DOE Order 420.1 are consistent with the guidance for seismic design and construction contained in the National Earthquake Hazards Reduction Program 1997 provisions (BSSC 1997). In addition, DOE Guide 420.1-2 was recently issued to recognize the consolidation of the three previous U.S. model building codes, including the Uniform Building Code, into the *International Building Code* (ICC 2000). The DOE requirements for seismic engineering have followed the Uniform Building Code, unless the importance of achieving a high level of protection warrants the use of more demanding methods and criteria (DOE Guide 420.1-2). Thus, new facilities would be designed and sited in accordance with DOE Order 420.1.

Site-specific analysis would be conducted in tiered NEPA documentation if the accelerator(s) alternative were selected.

4.5.1.1.6 Ecological Resources

If the accelerator(s) alternative were selected, tiered NEPA documentation would be undertaken to determine the exact nature of construction impacts on ecological resources. During that process, impacts on individual species and habitats that are sensitive to disturbance would be determined. This would include consideration of wetlands and threatened and endangered species. Wetland delineations and consultation with the U.S. Fish and Wildlife Service and state wildlife agency would take place, as necessary, to ensure that these resources would be protected.

Construction impacts on ecological resources are site specific. The nature of these impacts would be expected to vary depending on whether the site was located in the eastern or western portion of the United States. In fact, depending on the site location, impacts on some resources may not occur. Additionally, construction impacts on ecological resources would depend on whether the selected location was within an already disturbed portion of the site. In general impacts on terrestrial resources, wetlands, aquatic resources, and threatened and endangered species described below are applicable to an undeveloped site.

Terrestrial Resources. The construction of a low-energy accelerator, a high-energy accelerator, and a support facility would require 4 hectares (10 acres), 20.2 hectares (50 acres), and 2.4 hectares (6 acres), respectively (Herrington 2000; TechSource 2000; SAIC 2000). If these facilities were constructed at an undeveloped location, it is likely that woodland habitat would be lost at an eastern generic DOE site and shrubland would be disturbed at a western site. Land clearing activities would affect animal populations. Less mobile animals within the project area, such as reptiles and small mammals, would not be expected to survive. Construction activities and noise would cause larger mammals and birds in the construction and adjacent areas to move to similar habitat nearby. If the area to which they moved was below its carrying capacity, these animals would be expected to survive. However, if the area were already supporting the maximum number of individuals, the additional animals would compete for limited resources that could lead to habitat degradation and eventual loss of the excess population. Nests and young animals living within the disturbed area may not survive.

Wetlands. Clearing and grading operations could result in the direct loss of wetlands, although proper placement of the accelerator(s) and support facility within the overall generic DOE site would eliminate or reduce the potential for such loss. Indirect impacts could also result from stormwater runoff carrying sediments to wetlands located adjacent to the site. Changes in hydrology, water quality, and soils could occur as a result of alterations in water levels, runoff, and the buildup of sediments. These changes could, in turn, alter the vegetative composition of the wetland. In general, both direct and indirect impacts would be more likely to occur at an eastern site due to the greater abundance of wetlands. If preliminary analysis determined that wetlands could be impacted by development, a wetland delineation would be required. Impacts on wetlands could also lead to the implementation of mitigation measures.

Aquatic Resources. During construction of the accelerator(s) and a support facility, impacts on aquatic resources could result from stormwater runoff. Runoff could alter flow rates, increase turbidity, and lead to sedimentation of streambeds. These impacts could, in turn, cause temporary and permanent changes in species composition and density, and alter breeding habitats. The implementation of erosion and sediment control procedures would lessen construction impacts.

Threatened and Endangered Species. The construction of one or two accelerators and a support facility would have the potential to impact threatened and endangered species. Sources of impacts would be similar to those discussed above for terrestrial resources, wetlands, and aquatic resources. The primary difference is that the resource of concern involves individual species that are sensitive to disturbance and whose existence may be threatened by development. Consultations with the U.S. Fish and Wildlife Service and appropriate state agency would be conducted at the site-specific level, as appropriate.

4.5.1.1.7 Cultural and Paleontological Resources

The construction of a low-energy accelerator, a high-energy accelerator, and a support facility would require 4 hectares (10 acres), 20.2 hectares (50 acres), and 2.4 hectares (6 acres), respectively (SAIC 2000; TechSource 2000). Since the exact nature of the construction site for any of these facilities is not known at this time (e.g., whether it has previously been disturbed or not), potential effects on cultural resources cannot be determined. In general, if a location in a previously developed portion of a DOE generic site were selected, impacts on cultural resources may not occur. However, if an undisturbed location were chosen, cultural resources could be impacted. If the accelerator(s) alternative were selected, prehistoric and historic resources, including those that are or may be eligible for listing on the National Register of Historic Places, would be identified. These resources would be identified through site surveys and consultation with the State Historic Preservation Officer. Specific concerns about the presence, type, and location of Native American resources would be addressed through consultation with the potentially affected tribes in accordance with the *National Historic Preservation Act*, the *Native American Graves Protection and Repatriation Act*, and the *American Indian Religious Freedom Act*.

4.5.1.1.8 Socioeconomics

It is estimated that 410 workers would be needed to construct the new accelerator(s) and a support facility at a generic DOE site during the peak year of construction. The impact from this influx of workers upon the site's region of influence and regional economic area would depend on whether the site were located near a large urbanized area or in a remote rural area. Since the population for the region of influence for a generic site could range from nearly 2.0 million people for a site in a large metropolitan area, to less than 200,000 for a site in a small rural community, the socioeconomic impacts of constructing new accelerator(s) and a support facility would vary greatly. Therefore, if DOE were to select the new accelerator(s) alternative, additional NEPA documentation would be required to select the specific DOE site to locate the new accelerator(s) and support facility. In that document, DOE would perform a thorough evaluation of the socioeconomic impacts of the sites under consideration.

4.5.1.1.9 Public and Occupational Health and Safety—Normal Construction Activities

Assessments of incremental radiological and chemical impacts associated with the construction of the new accelerator(s) and support facility are presented in this section. Supplemental information is provided in Appendix H.

RADIOLOGICAL IMPACTS. During construction operations, it is not anticipated that there would be any resulting radiological releases to the environment; therefore no additional dose to the public is expected. Furthermore, construction workers are not expected to receive exposures above natural background levels which exist within the construction areas. However, as a precautionary measure, workers would be badged as deemed appropriate.

HAZARDOUS CHEMICAL IMPACTS. No hazardous chemical releases have been identified for construction activities. The painting activities would result in very small emissions of noncarcinogenic chemicals, which would produce minimal impact. Therefore, minimal hazardous chemical impacts are associated with construction.

4.5.1.1.10 Public and Occupational Health and Safety—Construction Accidents

There are no radiological or hazardous chemical accidents postulated during the construction phases of the new accelerator(s) or the support facility. Workers could experience industrial accidents commonly associated with the construction of large facilities.

4.5.1.1.11 Environmental Justice

Environmental effects due to construction activities that would be expected to occur at an unspecified accelerator(s) and support facility site are addressed in Section 4.5.1.1. The analysis shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected areas would not be significant. Unless there are patterns of food consumption among minority or low-income residents surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that construction activities would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new accelerator(s) and support facility, an additional evaluation of environmental justice at the accelerator(s) and support facility site during construction would be performed prior to implementation.

4.5.1.1.12 Waste Management

The expected generation rates of waste at a generic DOE site that would be associated with the construction of new accelerator(s) to irradiate targets and a support facility to fabricate and process medical and industrial isotope targets and to meet research and development needs are provided in **Table 4–122**. These estimates represent the total amount of waste generated during the construction period. These generation rates cannot be compared at this time with site treatment, storage, and disposal capacities because a DOE site has not yet been chosen for these facilities. Site-specific analyses would be conducted if this alterative were chosen, and appropriate NEPA documentation would be prepared.

Table 4–122 Estimated Waste Generation Associated with Constructing New Accelerator(s) and Support Facility Under All Options of Alternative 3 (Construct New Accelerator[s])

Waste Type[a]	Estimated Waste Generation for New Accelerator(s) (total cubic meters)		Estimated Waste Generation for New Support Facility (total cubic meters)
	Low-Energy	High-Energy	
High-level radioactive	0	0	0
Transuranic	0	0	0
Low-level radioactive			
Liquid	0	2	0
Solid	0	115	0
Mixed low-level radioactive	0	6	0
Hazardous			
Liquid	0	4	1
Solid	0	7	3
Nonhazardous			
Process wastewater	0	0	0
Sanitary wastewater	4,500	51,000	16,000
Solid	500	3,900	650

a. See definitions in Section G.9.
Note: To convert from cubic meters to cubic yards, multiply by 1.308.
Source: SAIC 2000; TechSource 2000.

Section 3.6.11.1 provides DOE site ranges for each waste type that include volume currently stored, projected generation, and for some types of waste, disposal volume. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.5.1.1.9 through 4.5.1.1.10.

4.5.1.2 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.5.1.2.1 Land Resources

LAND USE. The operation of one or two accelerators and a support facility at a generic DOE site would not be expected to affect land use. This is because none of the anticipated operational impacts (e.g., air emissions) are expected to affect this resource.

REDC would be used for neptunium-237 storage, target fabrication, and processing. The use of REDC for this purpose would not change land use at the site since REDC is currently operating and its proposed use would be compatible with its present mission.

VISUAL RESOURCES. The primary source of impacts on visual resources from the operation of one or two accelerators and a support facility would be air emissions. Releases from stacks associated with this alternative would be controlled and, therefore, would be unlikely to exceed Bureau of Land Management Visual Resource Management objectives. However, the operation of cooling towers could result in a visible plume. The extent and visibility of the plume would depend on site meteorological conditions and terrain features. While plume formation would be favored by meteorological conditions at an eastern generic DOE site, terrain features would tend to mask it from offsite locations; the opposite would tend to be true at a western site. If the accelerator(s) alternative were selected, the visual impact of the cooling tower plume would be determined in tiered NEPA documentation.

All activities associated with neptunium-237 storage, target fabrication, and processing would take place within REDC. Operations associated with the proposed activities would not result in any impact on visual resources or change in the current Visual Resource Management Class IV rating of the 7900 Area. This is because none of the anticipated operational impacts (e.g., air emissions) would be expected to affect this resource.

4.5.1.2.2 Noise

The operation of high-energy and/or low-energy accelerators at a generic DOE site would result in some increase in noise levels from equipment (e.g., cooling systems, vents, motors, generators, compressors, pumps, and material-handling equipment), employee vehicles, and truck traffic. Noise from operation activities could disturb wildlife outside the facility fence line. The change in noise levels in areas outside the DOE site would be dependent on the location selected and the equipment. However, generally if the location selected were within one of the larger DOE sites and were more centrally located within the site, offsite noise impacts from operation would be expected to be small. Operation employee vehicles and truck traffic would result in an increase in traffic noise along roads used to access the site. However, this increase in traffic noise would be small unless the operation traffic volume were as large as the existing site traffic. Site-specific analysis would be conducted in tiered NEPA documentation if the accelerator(s) alternative were selected.

The operation of a support facility at a generic DOE site would result in some increase in noise levels from equipment (e.g., cooling systems, vents, motors, generators, compressors, pumps, and material-handling equipment), employee vehicles, and truck traffic. Noise from operation activities could disturb wildlife outside the facility fence line. The change in noise levels in areas outside the DOE site would be dependent on the location selected and the equipment. However, generally if the location selected were within one of the larger DOE sites and were more centrally located within the site, offsite noise impacts from operation would be expected to be small. Operation employee vehicles and truck traffic would result in an increase in traffic noise along roads used to access the site. However, this increase in traffic noise would be small unless the operation traffic volume were as large as the existing site traffic. Site-specific analysis would be conducted in tiered NEPA documentation if the accelerator(s) alternative were selected.

This option also involves using REDC for neptunium-237 target material storage, target fabrication, and processing. Interior modifications of these facilities in the 7900 Area of ORNL would be expected to result in little change in noise impacts on wildlife around this area. The operation of REDC would not result in any change in noise impacts on wildlife around the 7900 Area and offsite noise impacts would be small because the nearest site boundary is 2.5 kilometers (1.6 miles) to the southeast. Operation would be expected to result in a minimal change in noise impacts on people near the ORR as a result of changes in employee and truck traffic levels.

4.5.1.2.3 Air Quality

High-Energy Accelerator. The operation of a new high-energy accelerator would result in some increase in air quality impacts due to the operation of emergency diesel generators. Criteria pollutants were modeled and compared to the most stringent standards (**Table 4–123**). The maximum ground-level pollutant concentrations that would result from high-energy accelerator operation would be well below the ambient air quality standards. However, if the accelerator is in an area that already had high background pollutant concentrations, resultant pollutant concentrations could approach or exceed the ambient standards for some pollutants. As a result, regulatory compliance would need to be assessed on case-by-case basis. Hazardous chemical impacts are addressed in Section 4.5.1.2.9.

Table 4–123 Incremental Concentrations Associated with High-Energy Accelerator Operation[a] Under Alternative 3 (Construct New Accelerator[s])—Option 1

Pollutant	Averaging Period	Most Stringent Standard or Guideline (microgram per cubic meter)[b]	Modeled Increment (microgram per cubic meter)
Carbon monoxide	8 hours 1 hour	10,000 40,000	94 135
Nitrogen oxide	Annual	100	0.47
PM_{10}	Annual 24 hours	50 150	0.03 17.7
Sulfur dioxide	Annual 24 hours 3 hours	80 365 1,300	0.03 16.5 37.2

a. From operation of two emergency diesel generators.
b. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); TechSource 2000.

Modeling was based on design consideration of two emergency diesel generators. These were modeled as a point source with emissions occurring at a stack height of 3 meters (9.8 feet). A boundary limit of 3,200 meters (2 miles) was assumed for a generic site.

Air quality impacts at ORR from target fabrication and processing associated with this option were determined to be the same as described for Alternative 2, Option 1 (Section 4.4.1.1.3).

Low-Energy Accelerator. The operation of a low-energy accelerator would not require emergency diesel generators. Thus, there would be no increase in air quality impacts due to the operation of the low-energy accelerator.

Support Facility for Accelerator(s). The operation of the support facility would result in air pollutant emissions similar to FMEF operating in support of FFTF. Thus, there would be an increase in air quality impacts that would be assessed and appropriate NEPA documentation prepared if this option were selected for implementation.

4.5.1.2.4 Water Resources

The estimated effects on key water resource indicators associated with operating the new accelerator(s) and support facility are presented in **Table 4–124.** Operation of the high-energy accelerator at a generic DOE site

would have the highest estimated water demand of any facility considered, requiring 1,904 million liters (503 million gallons) of water per year. In general, water would be required by the high-energy accelerator and other facilities to support such uses as process cooling, material processing, and potable and sanitary needs with the high water use of the high-energy accelerator attributable to cooling tower operation. The exact impact of these withdrawals on the resource would depend on the water source (surface water or groundwater) and its relative abundance. These factors would be used to determine the impact on the local and/or regional availability of the resource. For surface water, a dedicated surface water intake may have to be constructed if the generic site's existing distribution system is inadequate to meet the increased demands of the facilities. For groundwater, additional wells may have to be developed to supply the facilities (particularly the high-energy accelerator) directly or to provide increased production capacity for the generic site's existing supply system.

Table 4–124 Estimated Water Use and Wastewater Generation Associated with Operating Accelerator(s) and Support Facility Under Alternative 3 (Construct New Accelerator[s])—Option 1

Indicator (million liters per year)	Accelerators		
	Low-Energy	High-Energy	New Support Facility
Water use	1.9	1,904	6.92
Process wastewater generation	0	284	0.016[a]
Sanitary wastewater generation	0.9	11.4	6.91

a. Assume process wastewater generated at the same incremental rate as the Hanford 300 Area facilities (RPL/306–E).
Note: To convert from liters per year to gallons per year, multiply by 0.264.
Source: SAIC 2000; TechSource 2000.

The operation of the high-energy accelerator is estimated to generate approximately 284 million liters (75 million gallons) of process wastewater per year. This process effluent would mainly consist of cooling tower blowdown. The operation of the low-energy accelerator would not generate process wastewater as process cooling water would be recirculated within a closed-loop system. The support facility would generate a very small quantity of process wastewater mainly as a result of material processing. There would be no radiological liquid effluent discharge to the environment under normal operations. Sanitary wastewater would be generated as a result of operation of the accelerator(s) and support facility based on facility staff use of lavatory, shower, and kitchen facilities, and from miscellaneous potable and sanitary uses. Waste management activities and their effects are further detailed in Section 4.5.1.2.13. The potential impact on water resources would depend on the availability and capacity of appropriate treatment facilities. Process and sanitary wastewater would be discharged to either existing site wastewater treatment facilities or to new facilities constructed specifically to serve the proposed facilities. All wastewater would be disposed of in accordance with applicable regulatory requirements with discharges to surface waters in accordance with NPDES effluent limitations.

Although specific impacts on water resources cannot be determined at this time, site-specific analysis would be conducted in tiered NEPA documentation if the accelerator(s) alternative were selected.

REDC, an existing facility in the 7900 Area of ORNL at ORR, would be used for neptunium-237 storage, target fabrication, and processing in support of plutonium-238 production with impacts on ORR water resources indicators the same as those described in Section 4.3.1.1.4. In summary, a small increase in water use and sanitary wastewater generation is anticipated, mainly attributable to increased staffing levels. Also, there would be a very small increase in process wastewater generation, but there would be no radiological liquid effluent discharge to the environment under normal operations.

4.5.1.2.5 Geology and Soils

The operation of the accelerator(s) and support facility would not be expected to result in impacts on geologic and soils resources at a generic DOE site. If cooling towers are used, the potential exists for salt deposition to alter soil chemistry. While high rainfall at an eastern site would tend to keep salt from accumulating in the soil, the potential exists that salt could accumulate at a western site where rainfall is sparse. If the accelerator(s) alternative were selected, impacts on geology and soils would be determined in tiered NEPA documentation. As discussed in Section 4.5.1.1.5, the proposed facilities would be designed and constructed in accordance with DOE Order 420.1 and sited to minimize the risk from geologic hazards. Thus, site geologic conditions would be unlikely to affect the facilities.

The use of REDC for neptunium-237 storage, target fabrication, and processing would not be expected to impact either geologic or soil resources, nor be jeopardized by large-scale geologic conditions. Hazards from large-scale geologic conditions at ORR, such as earthquakes, volcanoes, and sinkholes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-260) as discussed in Section 4.2.2.2.5. The analysis determined that these hazards present a low risk to long-term storage facilities. Further review of the data and analyses presented in that document and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to REDC operations.

As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards will be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.5.1.2.6 Ecological Resources

If the accelerator(s) alternative were selected, tiered NEPA documentation would be undertaken to determine the exact nature of operational impacts on ecological resources. During this process, impacts on individual species and habitats that are sensitive to disturbance would be determined. This would include consideration of wetlands and threatened and endangered species.

While the exact nature of operational impacts on ecological resources cannot be determined until a specific site is selected, certain general types of impacts are possible. The nature and extent of these impacts would be expected to vary depending on whether the selected site was located in the eastern or western portion of the United States.

Terrestrial Resources. Activities associated with operations, such as noise and human presence, could affect wildlife living adjacent to the accelerator(s) and support facility. These disturbances could cause some species to move from the area. Preventing workers from entering undisturbed areas would minimize impacts on wildlife living adjacent to the facilities. Emissions to the air and water, both nonradiological and radiological, could impact both plants and animals. Plants and animals could be exposed to pollutants via a number of pathways including direct exposure, contact with contaminated soil, ingestion, and inhalation. Further, bioaccumulation could affect species that consume exposed plants or animals. While regulatory limits would act to limit the effects of air emissions and effluent discharges, impacts would be analyzed once site and facility specific information became available.

Wetlands. Impacts from the operation of one or two accelerators and a support facility at a western generic DOE site would not be expected to affect wetlands since discharges would be to an evaporation pond. At an eastern site, wastewater and cooling tower blowdown would be discharged to an onsite water body. While these discharges would be through permitted outfalls, the potential exists that wetlands could be affected. Potential impacts, such as changes in water levels and plant species composition, would depend on outfall location, water volume, discharge temperature, and water chemistry. Since these factors depend on site

location and facility engineering design, operational impacts on onsite wetlands would have to be analyzed once these factors are known.

Aquatic Resources. Operational impacts on aquatic resources at a western site would not be expected because groundwater would be used and wastewater and cooling tower blowdown would be discharged to an evaporation pond. At an eastern site, potential impacts on aquatic resources could occur as a result of water withdrawal and discharge. Water withdrawal could lead to the loss of aquatic organisms through impingement and entrainment. The discharge of cooling water could result in alterations in aquatic communities. Alterations could include changes in aquatic vegetation and the loss of fish and benthic macroinvertebrates. Additionally, radionuclides and chemicals in the discharge water have the potential to impact aquatic organisms. The extent of potential impacts on the aquatic environment would depend on site and facility specific information.

Threatened and Endangered Species. The operation of one or two accelerators and a support facility would have the potential to impact threatened and endangered species. Sources of impacts would be similar to those discussed above for terrestrial resources, wetlands, and aquatic resources. The primary difference is that the resources of concern involve individual species that are sensitive to disturbance and whose existence may be threatened by development.

REDC would be used for neptunium-237 storage, target fabrication, and processing. As noted in Section 4.5.1.2.2, wildlife would not be adversely affected by noise associated with facility operation. There would be no change in impacts on wetlands or aquatic resources because additional water usage and wastewater discharge would be small fractions of current values. Further, this option would not result in any new contaminants in existing discharges (Section 4.5.1.2.4). No threatened and endangered species have been identified within the 7900 Area; therefore, operational impacts on this resource are not expected.

Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service (see Table 5–3) and resulted in the Service concluding that it does not anticipate adverse effects to federally listed endangered species that occur near the project area. DOE has also consulted with the Tennessee Department of Environment and Conservation; a response concerning state-listed species is pending from this agency. Although no state-listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the state.

4.5.1.2.7 Cultural and Paleontological Resources

Impacts on cultural and paleontological resources from the operation of one or two accelerators and a support facility at a generic DOE site would depend on the relative location of such resources to the site and/or transportation routes. While impacts would be expected to be nonexistent or small, they cannot be ruled out. For example, noise related to plant operation or traffic to and from the facility or alterations in the viewshed could adversely affect visitor enjoyment of an historic site. Since impacts on cultural resources are site dependent, specific operational impacts cannot be determined until a site were selected. The operation of the accelerator(s) and support facility would not be expected to impact paleontological resources.

The operation of REDC for neptunium-237 storage, target fabrication, and processing would not affect the status of cultural and paleontological resources at ORR. The Graphite Reactor, which is located within ORNL, is listed on the National Register of Historic Places as a National Historic Landmark. Additionally, several other structures proposed for listing on the National Register of Historic Places are found within or near ORNL. However, neither the Graphite Reactor nor any of the other structures is located within the 7900 Area, thus, the use of REDC for target fabrication and processing would not change their status.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). While DOE has made additional contact with the State Historic Preservation Office, a response is pending from this office. Although impacts to cultural resources are not expected as a result of the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the State Historic Preservation Office.

4.5.1.2.8 Socioeconomics

It is estimated that 325 workers would be needed to operate the new accelerator(s) and support facility at a generic DOE site. The impact from this influx of workers upon the site's region of influence and regional economic area would depend on whether the site were located near a large urbanized area or in a remote rural area. Since the population for the region of influence for a generic site could range from nearly 2.0 million people for a site in a large metropolitan area, to less than 200,000 for a site in a small rural community, the socioeconomic impacts of operating a new accelerator and support facility would vary greatly. Therefore, if DOE were to select this option, additional NEPA documentation would be required to determine the specific socioeconomic impacts.

The socioeconomic impacts associated with neptunium-237 target fabrication and processing at ORR are addressed in Section 4.3.1.1.8.

4.5.1.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup and operations are given in **Table 4–125** for the generic DOE site and ORR: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

Table 4–125 Incremental Radiological Impacts on the Public Around the Generic DOE Site and ORR from Operational Facilities Under Alternative 3 (Construct New Accelerator[s])—Option 1

Receptor	ORR REDC	Accelerators Preoperational Startup[a]		Generic Site Operations				Two-Site Total
				Accelerators		Accelerator(s) Support Facility	Total	
		Low-Energy	High-Energy	Low-Energy	High-Energy			
Population within 80 kilometers (50 miles) in the year 2020								
Dose (person-rem)	8.8×10^{-5}	0.0024	0.035	0.0043	0.055	0.14	0.20	0.20
35-year latent cancer fatalities	1.5×10^{-6}	2.4×10^{-6}[b]	3.5×10^{-5}[b]	7.5×10^{-5}	9.6×10^{-4}	0.0025	0.0035	0.0035
Maximally exposed individual								
Annual dose (millirem)	1.9×10^{-6}	1.4×10^{-5}	1.8×10^{-4}	1.1×10^{-4}	8.7×10^{-4}	0.0025	0.0035	NA[c]
35-year latent cancer fatality risk	3.3×10^{-11}	1.4×10^{-11}[b]	1.8×10^{-10}[b]	1.9×10^{-9}	1.5×10^{-8}	4.4×10^{-8}	6.1×10^{-8}	NA[c]
Average exposed individual within 80 kilometers (50 miles)								
Annual dose[d] (millirem)	7.8×10^{-8}	1.6×10^{-6}	2.3×10^{-5}	2.8×10^{-6}	3.6×10^{-5}	9.1×10^{-5}	1.3×10^{-4}	NA[c]
35-year latent cancer fatality risk	1.4×10^{-12}	1.6×10^{-12}[b]	2.3×10^{-11}[b]	4.9×10^{-11}	6.3×10^{-10}	1.6×10^{-9}	2.3×10^{-9}	NA[c]

a. For conservatism as well as consistency with other radiological impacts evaluated in this NI PEIS, these values were assessed for the year 2020 even though these activities would commence prior to that year.
b. Preoperational activities last 2 years. Number is a 2-year latent cancer fatality risk.
c. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
d. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of REDC or the generic site in the year 2020 (1,134,200 and 1,538,100, respectively).

Key: NA, not applicable.
Source: SAIC 2000; TechSource 2000; model results, using the GENII computer code (Napier et al. 1988).

As a result of annual operations of the accelerator facilities (a high-energy accelerator, a low-energy accelerator, and an accelerator support facility) and REDC, the projected incremental total population dose in the year 2020 would be 0.20 person-rem; the corresponding number of latent cancer fatalities in the populations surrounding the generic DOE site and ORR from 35 years of operations would be 0.0035. The incremental total dose to the maximally exposed member of the public from annual operations of the accelerator(s) and support facility at the generic site would be 0.0035 millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 6.1×10^{-8}. Estimated annual risks are also presented for preoperational testing and startup phase activities anticipated for the accelerator(s) and support facility. The incremental dose to the maximally exposed member of the public from annual REDC operations would be 1.9×10^{-6} millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 3.3×10^{-11}.

Incremental doses to involved workers from normal operations are given in **Table 4–126**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to the high-energy and low-energy accelerator workers during startup and operations would be 150 millirem; for support facility workers, the incremental annual average dose operations would be 114 millirem for REDC workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities is estimated to be 30 (high-energy accelerator startup and operation), 15 (low-energy accelerator startup and operation), 11 (support facility operation), and 12 person-rem (REDC operation). The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–126. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–126 Incremental Radiological Impacts on Involved Workers at the Generic DOE Site and ORR from Operational Facilities Under Alternative 3 (Construct New Accelerator[s])—Option 1

Receptor—Involved Workers[a]	ORR REDC	Accelerators Preoperational Startup		Generic Site Operations			Two-Site Total
				Accelerators		Accelerator(s) Support Facility	
		Low-Energy	High-Energy	Low-Energy	High-Energy		
Total dose (person-rem per year)	12[b]	23[b]	45[b]	15[b]	30[b]	11[b]	69
35-year latent cancer fatalities	0.17	0.018[c]	0.036[c]	0.21	0.42	0.16	0.96
Average worker dose (millirem per year)	170	150	150	150	150	114	NA[d]
35-year latent cancer fatality risk	0.0023	1.2×10^{-4}[c]	1.2×10^{-4}[c]	0.0021	0.0021	0.0016	NA[d]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that facilities adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.

b. Based on an estimated 75 badged workers at ORR REDC, 90 radiological workers (and 100 total workers) at the accelerator(s) support facility, 200 workers at the high-energy accelerator (300 during startup), and 100 workers at the low-energy accelerator (150 during startup).

c. Preoperational startup testing lasts 2 years. Number is a 2-year latent cancer fatality risk.

d. Values cannot be given for the average worker because the workers would be in three different facilities at two different sites.

Key: NA, not applicable.

Source: DOE 1999b; Nielsen 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICALS IMPACTS

High-Energy Accelerator. The operation of a high-energy accelerator would result in some increase in emissions of hazardous chemicals from diesel fuel burning equipment used for operation. The operation of the accelerator would require the emergency diesel generators to be tested approximately 1 hour each month and 12 hours once a year to ensure operability. Chemical releases were modeled based on 48 hours of operation. The source was modeled as a point source with emissions occurring at a stack height of 3 meters (9.8 feet). A boundary limit of 3,200 meters (2 miles) was assumed for a generic site. Resulting concentrations were determined to be very small and would have no incremental impact on the current conditions at the site (**Table 4–127**).

Hazardous chemical impacts at ORR from target fabrication and processing associated with this option were determined to be the same as described in Option 1 under Alternative 2 (Section 4.4.1.1.9).

Low-Energy Accelerator. The operation of a low-energy accelerator would not require emergency diesel generators. Thus, there would be no increase in hazardous chemical impacts due to the operation of the low-energy accelerator.

Support Facility for Accelerator(s). The operation of the support facility would result in air pollutant emissions similar to FMEF operating in support of FFTF. Thus, there would be an increase in hazardous chemical impacts that would be assessed and appropriate NEPA documentation prepared if this option were selected for implementation.

Table 4–127 Incremental Hazardous Chemical Impacts on the Public Around a Generic Site from High-Energy Accelerator Operation Under Alternative 3 (Construct New Accelerator[s])—Option 1

Chemicals	Modeled Annual Increment (microgram per cubic meter)	RfC (microgram per cubic meter)	Unit Cancer Risk (Risk per Microgram per cubic meter)	Hazard Quotient	Cancer Risk
Acetaldehyde	0.0000811	NA	2.20×10^{-6}	NA	1.79×10^{-10}
Benzene	0.0000987	NA	7.80×10^{-6}	NA	7.70×10^{-10}
Formaldehyde	0.000125	NA	1.30×10^{-5}	NA	1.62×10^{-9}
Toluene	0.0000433	400	NA	1.08×10^{-7}	NA
Propylene	0.000273	NA	3.70×10^{-6}	NA	1.01×10^{-9}

Note: Propylene oxide cancer unit was used for propylene.
Key: RfC, Reference concentration; NA, not applicable; The chemical is not a known carcinogen, or it is a carcinogen and only unit cancer will apply.
Source: Data from TechSource 2000; EPA 1999; modeled increments are based on the SCREEN3 computer code (EPA 1995).

4.5.1.2.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with accelerator target irradiation; support facility fabrication and processing of medical, industrial, and research and development isotopes; and REDC target fabrication and processing of neptunium-237 targets are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–128** and **4–129**, respectively.

For 35 years of high-energy accelerator target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.05×10^{-6} and 5.15×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0063.

For 35 years of low-energy accelerator target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.45×10^{-9} and 3.07×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 3.66×10^{-5}.

For 35 years of medical, industrial, and research and development target fabrication and processing at the support facility, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 3.26×10^{-5} and 9.85×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.056.

For 35 years of neptunium-237 target fabrication and processing at REDC, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.157.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 9.18×10^{-5} and 5.00×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.219.

Table 4–128 New Accelerator(s), Support Facility, and REDC Accident Consequences Under Alternative 3 (Construct New Accelerator[s])—Option 1

Accident	Maximally Exposed Individual		Population to 80 kilometers (50 miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
High-energy accelerator accidents						
Design-basis target accident	2.93×10^{-4}	1.47×10^{-7}	0.980	4.90×10^{-4}	9.35×10^{-4}	3.74×10^{-7}
Beyond-design-basis earthquake	11.7	0.00585	3.01×10^{4}	18	184	0.147
Low-energy accelerator accidents						
Design-basis target accident	8.05×10^{-5}	4.03×10^{-8}	17.7	0.00885	0.00112	4.48×10^{-7}
Beyond-design-basis earthquake	0.0132	6.60×10^{-6}	32.4	0.0162	0.208	8.32×10^{-5}
Support facility accidents						
Medical and industrial isotopes localized solvent fire	0.0194	9.72×10^{-6}	31.1	0.0156	0.00530	2.12×10^{-6}
Medical and industrial isotopes unlikely seismic event	0.0750	3.75×10^{-5}	136	0.0680	0.510	2.04×10^{-4}
Medical and industrial isotopes glovebox explosion	2.50	0.00125	4,600	2.30	17.0	0.00680
REDC accidents						
Ion exchange explosion during neptunium-237 target fabrication	6.13×10^{-9}	3.06×10^{-12}	8.58×10^{-5}	4.29×10^{-8}	5.60×10^{-10}	$2.24x0^{-13}$
Target dissolver tank failure during plutonium-238 separation	1.76×10^{-7}	8.79×10^{-11}	0.00196	9.82×10^{-7}	1.69×10^{-8}	6.74×10^{-12}
Ion exchange explosion during plutonium-238 separation	4.68×10^{-4}	2.34×10^{-7}	5.23	0.00261	4.49×10^{-5}	1.79×10^{-8}
Processing facility beyond-design-basis earthquake	163	0.163	8.91×10^{5}	445	1,310	1.00[c]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–129 New Accelerator(s), Support Facility, and REDC Accident Risks Under Alternative 3 (Construct New Accelerator[s])—Option 1

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual high-energy accelerator risks			
Design basis target accident (1×10^{-4})	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk	2.05×10^{-6}	0.00630	5.15×10^{-5}
Annual low-energy accelerator risks			
Design basis target accident (1×10^{-4})	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
35-year low-energy accelerator risk	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
Annual support facility risks			
Medical and industrial isotopes localized solvent fire (0.044)	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical and industrial isotopes glovebox explosion (1.00×10^{-4})	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
35-year support facility risk	3.26×10^{-5}	0.056	9.85×10^{-5}
Annual REDC risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Processing facility beyond-design-basis earthquake (1×10^{-5})	1.63×10^{-6}	0.00445	$1.00\times10^{-5(c)}$
35-year REDC risk	5.71×10^{-5}	0.157	$3.50\times10^{-4(c)}$
35-year Option risk	9.18×10^{-5}	0.219	5.00×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

There are no hazardous chemical accidents associated with the new accelerator(s) or new support facility. The irradiation of neptunium-237, medical, industrial, and research and development isotopes in the new accelerator(s) would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The fabrication and processing of medical, industrial, and research and development isotopes at the new support facility would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The hazardous chemical accident impacts at REDC are the same as those presented in Section 4.4.4.1.10.

4.5.1.2.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to the accelerator(s) site. Following

irradiation in the accelerator(s), the targets would be returned to REDC for processing. After this processing, the plutonium-238 product would be shipped to LANL. Additionally, medical and industrial isotopes would be shipped from the accelerator(s) site to a local airport, and from there to locations throughout the country.

Approximately 37,000 shipments of radioactive materials would be made by DOE in support of the low-energy accelerator. The total distance traveled on public roads by trucks carrying radioactive materials would be 4.8 million kilometers (3.0 million miles); and in the air carrying medical and industrial isotopes, 23 million kilometers (14 million miles).

Approximately 269 shipments of radioactive materials would be made by DOE in support of the high-energy accelerator. The total distance traveled on public roads by trucks carrying radioactive materials would be 0.94 million kilometers (0.59 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION FOR THE LOW-ENERGY ACCELERATOR. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 15 person-rem; the dose to the public, 7 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0059 latent cancer fatality among transportation workers and 0.0037 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.02.

IMPACTS OF INCIDENT-FREE TRANSPORTATION FOR THE HIGH-ENERGY ACCELERATOR. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 5 person-rem; the dose to the public, 101 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0020 latent cancer fatality among transportation workers and 0.050 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.0022.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION FOR THE LOW-ENERGY ACCELERATOR. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: more than 1 in 10 million per year) would not breach the transportation package. The consequences of more severe accidents that could breach the transportation package and release radioactive material were evaluated and estimated to have probabilities of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting 0.11 traffic fatality.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION FOR THE HIGH-ENERGY ACCELERATOR. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 0.16 person-rem, resulting in 8.1×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.025 traffic fatality.

4.5.1.2.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in one or two new accelerators that would be constructed at a site yet to be specified. Fabrication and processing of neptunium-237 targets for plutonium-238 production would be performed at REDC located at ORR. A new support facility would be constructed at the same unspecified site for fabrication and processing targets not used for plutonium-238 production.

Activities at REDC were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.1.1.12) and found to pose no significant radiological or other risks to minority and low-income populations. The environmental analysis of operations at the new accelerator(s) and support facility site shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected area would not be significant. Unless there are patterns of food consumption among minority or low-income residents surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that operations at the site would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site-specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specified site selected for the new accelerator(s) and support facility, an additional evaluation of environmental justice at the accelerator(s) and support facility site during operation would be performed prior to implementation.

4.5.1.2.13 Waste Management

The expected annual generation of waste that would be generated from the operation of new accelerator(s) to irradiate targets and a support facility to fabricate and process medical and industrial isotope targets and to meet research and development needs are provided in **Table 4–130**. These generation rates cannot be compared at this time with site treatment, storage, and disposal capacities because a DOE site has not yet been chosen for these facilities. Section 3.6.11.1 provides DOE site ranges for each waste type that include volume currently stored, projected generation, and for some types of waste, disposal volume. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.5.1.2.9 through 4.5.1.2.11.

In accordance with the Records of Decision for the *Waste Management PEIS* (DOE 1997a), waste could be treated and disposed of on site or at other DOE sites or commercial facilities. No high-level radioactive waste or transuranic waste would be generated from irradiating targets in the new accelerator(s) or from target fabrication or processing in the new support facility.

Table 4–130 Estimated Waste Generation Rates of Operating New Accelerator(s) and Support Facility Under Alternative 3 (Construct New Accelerator[s])—Option 1

Waste Type[a]	Estimated Waste Generation for New Accelerator(s) (cubic meters per year)		Estimated Waste Generation for New Support Facility (cubic meters per year)
	Low-Energy	High-Energy	
High-level radioactive	0	0	0
Transuranic	0	0	0
Low-level radioactive			
Liquid	0	1	0
Solid	5	54	20
Mixed low-level radioactive	0.20	3	4
Hazardous	0.10	2	<1
Nonhazardous			
Process wastewater	0	280,000	16[b]
Sanitary wastewater	910	11,300	6,900
Solid	8	31	80

a. See definitions in Section G.9.
b. Assume process wastewater generated at the same incremental rate as the Hanford 300 Area facilities (RPL/306–E).
Note: To convert from cubic meters to cubic yards, multiply by 1.308, < mean "less than."
Source: SAIC 2000; TechSource 2000.

Currently, DOE sites that manage low-level radioactive waste treat and/or dispose of the waste on site or off site, either at another DOE facility or a commercial facility. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision issued on February 18, 2000 (65 FR 10061), states that for the management of low-level radioactive waste, minimal treatment will be performed at all sites, and disposal will continue, to the extent practicable, on site at INEEL, LANL, ORR, and SRS. In addition, Hanford and the Nevada Test Site will be available to all DOE sites for low-level radioactive waste disposal. An estimated 35 cubic meters (46 cubic yards) of liquid low-level radioactive waste and 2,100 cubic meters (2,750 cubic yards) of solid low-level radioactive waste would be generated over a 35-year period as a result of target irradiation at the new accelerator(s). Target fabrication and processing at the new support facility would generate about 700 cubic meters (920 cubic yards) of solid low-level radioactive waste. The minor amounts of low-level radioactive waste (less than 10 cubic meters [13.1 cubic yards]) (Brunson 1999a) generated from the decontamination of the shipping containers used to transport neptunium-237 from SRS to REDC (or FDPF or FMEF, depending on the option) for storage could easily be managed under the existing waste management practices and are not included in the table.

Most of DOE's mixed low-level radioactive waste is being stored on site awaiting the development of treatment methods. DOE is subject to the requirements mandated by the Federal Facility Compliance Act of 1992, and most DOE facilities that currently store or generate mixed low-level radioactive waste have either a state-approved or EPA region-approved site treatment plan or another type of agreement. Each site treatment plan or agreement requires the treatment of mixed waste, including mixed low-level radioactive waste, in accordance with its provisions. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision, issued on February 18, 2000 (65 FR 10061), states that mixed low-level radioactive waste will be treated at Hanford, INEEL, ORR, and SRS and disposed of at Hanford and the Nevada Test Site. Over the 35-year operational period, an estimated 110 cubic meters (140 cubic yards) of mixed low-level radioactive waste would be generated as a result of target irradiation at the new accelerator(s). Target fabrication and processing at the new support facility would generate about 140 cubic meters (180 cubic yards) of mixed low-level radioactive waste.

The hazardous waste Record of Decision, issued on August 5, 1998 (63 FR 41810), states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of nonwastewater hazardous waste, with ORR and SRS continuing to treat some of their own nonwastewater hazardous waste on site in existing facilities where this is economically favorable. Wastewater, which is about 99 percent of DOE's hazardous waste, is treated on site. An estimated 74 cubic meters (97 cubic yards) of hazardous waste would be generated during the 35-year operational period at the accelerator(s) and less than 35 cubic meters (46 cubic yards) at the new support facility.

DOE currently manages sanitary and industrial waste on a site-by-site basis. Some DOE sites dispose of this waste in onsite landfills that have permits issued by appropriate state agencies, while other sites use commercial landfills (DOE 1997a:1-29). Solid waste such as office paper, metal cans, and plastic and glass bottles that can be recycled would be sent off site for that purpose. Over the 35-year operational period, an estimated 9.8 million cubic meters (12.8 million cubic yards) of process wastewater, 427,000 cubic meters (558,000 cubic yards) of sanitary wastewater, and 1,400 cubic meters (1,800 cubic yards) of solid nonhazardous waste would be generated as a result of target irradiation at the new accelerator(s). Target fabrication and processing at the new support facility would generate about 560 cubic meters (730 cubic yards) of process wastewater, 241,500 cubic meters (316,000 cubic yards) of sanitary wastewater, and 2,800 cubic meters (3,700 cubic yards) of solid nonhazardous waste.

The impacts of managing waste associated with fabricating and processing neptunium-237 targets for plutonium-238 production in REDC at ORR are assumed to be the same as for Option 1 under Alternative 1 (Section 4.3.1.1.13). As shown in that section, the impacts on the waste management systems at ORR would be small.

4.5.1.3 Decontamination and Decommissioning of the Accelerator(s) and Support Facility

The environmental impacts associated with the decontamination and decommissioning of the accelerator(s) and support facility at the generic DOE site are assessed in this section. If the accelerator(s) were built on a site with existing support facilities, there would be no impacts associated with decommissioning a "new" support facility.

4.5.1.3.1 Land Resources

LAND USE. Decontamination and decommissioning of the accelerator(s) and a support facility would not involve the removal of any major structures, although some smaller facilities and pieces of equipment could be removed. Thus, the industrial nature of the land would not change.

VISUAL RESOURCES. Decontamination and decommissioning of the accelerator(s) and a support facility would not impact visual resources since no major structures would be removed. Thus, the Visual Resource Management Class IV rating of the site would remain unchanged.

4.5.1.3.2 Noise

Decontamination and decommissioning of the high-energy and/or low-energy accelerators and support facility would result in some increase in noise levels from the use of construction type equipment, materials handling and impact equipment, employee vehicles, and truck traffic. Actual noise levels would depend on the decontamination and decommissioning activities selected. Noise from these activities, especially impulsive noise, would be expected to disturb wildlife in the immediate area of the facilities. The change in noise levels in areas outside the DOE site would depend on the location selected and the exact nature of the activities required. However, generally if the accelerator(s) and support facility location were within one of the large

DOE sites and were more centrally located within the site, offsite noise impacts from decontamination and decommissioning would be expected to be small. Employee vehicles and truck traffic would result in an increase in traffic noise along roads used to access the site. However, this increase in traffic noise would be small unless the decontamination and decommissioning traffic volume were as large as the traffic from facility operation and other site activities. Site-specific analysis would be conducted in tiered NEPA documentation if the accelerator(s) alternative were selected.

4.5.1.3.3 Air Quality

The potential for air quality impacts due to decommissioning and deactivation of the accelerator(s) and support facility would not be expected to be any higher than those associated with their construction and operation. Some decrease in air quality impacts may occur when generators and pumps supporting operations of the accelerator(s) are shut down.

4.5.1.3.4 Water Resources

Decontamination and decommissioning of the accelerator(s) and support facility would involve permanent shutdown, stabilization, and monitoring of the deactivated facilities. As a result, processing and auxiliary systems would be shutdown and process and sanitary wastewater discharges would cease from the vacated facilities. This would eliminate the annual discharge of approximately 284 million liters (75 million gallons) of nonradioactive process wastewater from the high-energy accelerator and 0.016 million liters (0.004 million gallons) from the support facility to onsite treatment facilities. Also, the discharge of sanitary wastewater to onsite treatment facilities would be eliminated, including 11.4 million liters (3 million gallons) per year from the high-energy accelerator, 0.9 million liters (0.24 million gallons) from the low-energy accelerator, and 6.91 million liters (1.82 million gallons) annually from the support facility. The effects of decontamination and decommissioning on waste management are further detailed in Section 4.5.1.3.13. Site water withdrawals to supply the facilities would also be reduced by an estimated 1,904 million liters (503 million gallons) per year for the high-energy accelerator, 1.9 million liters (0.50 million gallons) for the low-energy accelerator, and 6.92 million liters (1.83 million gallons) annually for the support facility (SAIC 2000; TechSource 2000).

4.5.1.3.5 Geology and Soils

No major structures would be demolished to effect decontamination and decommissioning of the accelerator(s) and support facility. Some ground disturbance could occur associated with removal of some smaller facilities and pieces of equipment. However, ground disturbance would be confined to previously disturbed areas immediately adjacent to the accelerator(s) and support facility, with the impact on geologic and soil resources expected to be negligible overall.

4.5.1.3.6 Ecological Resources

Since no major structures would be demolished during the decontamination and decommissioning of the accelerator(s) and a support facility, the area would continue to be of limited value to wildlife. Noise from decontamination and decommissioning activities would be expected to disturb wildlife in the immediate area; however, this disturbance would be of limited duration. Water use would decrease at the generic site with the decommissioning of the accelerator(s) and support facility. This would result in a decrease in impingement and entrainment of aquatic organisms, as well as a decrease in impacts from effluent discharge at a site where surface water bodies are used. At a site where water is withdrawn from groundwater and discharged to an evaporation pond, the cessation of discharge from the accelerator(s) and support facility could result in a reduction in the size of the pond or its possible elimination. This could, in turn, result in the loss (or elimination) of associated aquatic and terrestrial wildlife, as well as wetland habitat. The response of any threatened or endangered species to decontamination and decommissioning of the accelerator(s) and support facility could vary from positive (e.g., due to a decrease in human presence and emissions) to negative (e.g., due to the elimination of aquatic or wetland habitat), depending on the species involved.

4.5.1.3.7 Cultural and Paleontological Resources

Decontamination and decommissioning of the accelerator(s) and a support facility would not change the status of cultural and paleontological resources. This is because any required ground disturbance would be confined to previously disturbed areas immediately adjacent to the accelerator(s) and support facility.

4.5.1.3.8 Socioeconomics

Decommissioning of the accelerator(s) and support facility would result in a negative impact on the socioeconomic characteristics of the DOE site at which they were located. This impact would depend on whether the candidate site was located near a large urbanized area or in a remote rural area. Since the population for the region of influence for a generic DOE site could range from nearly 2.0 million people for a site in a large metropolitan area, to less than 200,000 for a site in a small rural community, the socioeconomic impacts of decommissioning would vary greatly. Therefore, if DOE were to select the new accelerator(s) alternative, additional NEPA documentation would be required to evaluate the specific socioeconomic impacts of decommissioning.

4.5.1.3.9 Public and Occupational Health and Safety—Normal Decontamination and Decommissioning Activities

Assessments of incremental radiological and chemical impacts associated with the decontamination and decommissioning of the accelerator(s) and support facility are presented in this section. Supplemental information is provided in Appendix H.

During decontamination and decommissioning operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers are described below.

RADIOLOGICAL IMPACTS. In the *Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities,* NUREG-0586 (NRC 1988), NRC determined that the health impact to the public from the decommissioning of research reactors was "negligible." In the same NUREG, NRC also concluded that the public health impact from radiological releases associated with the decommissioning and decontamination of the research reactor support facility was also "negligible." Decommissioning and decontamination of the accelerator(s) and support facility would involve less radioactive materials and thus less radioactive emissions,

than those associated with the research reactor and support facility. Based on these conclusions, the environmental impact on the public health and safety from the routine release of radionuclides during the decontamination and decommissioning of the accelerator(s) and support facility addressed in this NI PEIS are deemed to be negligible.

Incremental doses to involved workers from decontamination and decommissioning operations are given in **Table 4–131**; these workers are defined as those directly associated with all decontamination and decommissioning activities. The incremental annual average dose to involved workers during decontamination and decommissioning operations at the accelerator(s) would be 160 millirem; for support facility workers, the incremental annual average dose during decontamination and decommissioning operations would be 100 millirem. The incremental annual dose received by the total site workforce for each of these facilities is estimated to be 17 (total for both accelerators) and 4 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from annual decontamination and decommissioning operations are included in Table 4–131; a probability coefficient of 4×10^{-4} latent cancer fatality per rem was applied for workers (ICRP 1991). Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–131 Incremental Radiological Impacts on Involved Workers at the Generic DOE Site from Accelerator(s) and Support Facility Decontamination and Decommissioning Activities Under All Options of Alternative 3 (Construct New Accelerator[s])

	Generic Site Decontamination and Decommissioning Activities			
	Accelerators		Accelerator(s) Support Facility	Total
Receptor—Involved Workers[a]	Low-Energy	High-Energy		
Total dose (person-rem per year)	5.6[b]	11[b]	4[b]	21
1-year latent cancer fatalities	0.0022	0.0045	0.0016	0.0083
Average worker dose (millirem per year)	160	160	100	143
1-year latent cancer fatality risk	6.5×10^{-5}	6.5×10^{-5}	0.0004	5.7×10^{-5}

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.

b. Based on 105 badged workers at the accelerator(s) (35 at low-energy and 70 at high-energy) and 40 badged workers at the support facility.

Source: Calculational results.

HAZARDOUS CHEMICAL IMPACTS. No additional hazardous chemical release is expected from activities associated with decontamination and decommissioning the accelerator(s) and its support facility.

4.5.1.3.10 Public and Occupational Health and Safety—Decontamination and Decommissioning Accidents

There are no radiological or hazardous chemical accidents postulated during the decontamination and decommissioning phases of the new accelerator(s) or the new support facility. Involved workers could experience industrial accidents commonly associated with these types of activities.

4.5.1.3.11 Environmental Justice

Environmental effects due to decontamination and decommissioning activities that would be expected to occur at an unspecified accelerator(s) and support facility site are addressed in Section 4.5.1.3. The environmental analysis of decontamination and decommissioning activities at the new accelerator(s) and support facility site shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected

areas would not be significant. Unless there are patterns of food consumption among minority or low-income residents surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that decontamination and decommissioning activities at the site would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new accelerator(s) and support facility, an additional evaluation of environmental justice at the accelerator(s) and support facility site during decontamination and decommissioning would be performed prior to implementation.

4.5.1.3.12 Waste Management

The decontamination and decommissioning of the new accelerator(s) and support facility could generate numerous types of waste. The materials that may be removed or stabilized as a result of decontamination and decommissioning would be managed and reused, recycled, or disposed of in accordance with applicable Federal and state regulations. No analysis of waste management impacts, however, can be formulated at this time. Once proposals concerning decontamination and decommissioning activities were developed, DOE would undertake any additional NEPA analysis that may be necessary or appropriate.

4.5.1.4 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.5.2 Alternative 3 (Construct New Accelerator[s])—Option 2

Option 2 involves constructing and operating one or two accelerators to irradiate all targets associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating FDPF at INEEL to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and conducting and operating the support facility to fabricate and process the other targets and materials and to process the associated products. This option includes storage in Building CPP–651 or FDPF of the neptunium-237 transported to INEEL from SRS and storage in the new support facility of the other target materials transported to the generic site from other offsite facilities.

The transportation of the neptunium-237 from SRS to INEEL and then to the generic site, the transportation of the other target materials to the generic site, and the transportation of plutonium-238 and other product materials following irradiation and postirradiation processing constitute part of this option.

All options under this alternative include the decontamination and decommissioning of the accelerator(s) and support facility at the generic DOE site following their operating lifetimes, and also the permanent deactivation of FFTF at Hanford.

4.5.2.1 Construction of the New Accelerator(s) and Support Facility

The environmental impacts associated with the construction of one or two new accelerators and a support facility at the generic DOE site are assessed in Section 4.5.1.1.

4.5.2.2 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.5.2.2.1 Land Resources

LAND USE. Impacts on land use associated with the operation of one or two accelerators and a support facility are addressed in Section 4.5.1.2.1.

Building CPP–651 and/or FDPF, which are located at INTEC, would be used for neptunium-237 storage, and FDPF would be used for target fabrication and processing. Use of these facilities would not change land use at the site since both are currently operating and their proposed use would be compatible with their present mission.

VISUAL RESOURCES. Impacts on visual resources associated with the operation of one or two accelerators and a support facility are addressed in Section 4.5.1.2.1.

All activities associated with neptunium-237 storage would take place within Building CPP–651 and/or FDPF, and target fabrication and processing would be in FDPF. Operations associated with the proposed activities would not result in any impact on visual resources or change in the current Visual Resource Management Class IV designation of INTEC. This is because none of the anticipated operational impacts (e.g., air emissions) would be expected to affect this resource.

4.5.2.2.2 Noise

Noise impacts associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.2.

This option also involves using Building CPP–651 and/or FDPF, both in the INTEC area of INEEL, for neptunium-237 target material storage, and FDPF for target fabrication and processing. Interior modifications of these facilities would be expected to result in little change in noise impacts on wildlife around this area. The operation of these facilities would not be expected to result in any change in noise impacts on wildlife around the INTEC area and offsite noise impacts would be small because the nearest site boundary is 12 kilometers (7.5 miles) to the south. Operation would result in a minimal change in noise impacts on people near the INEEL as a result of changes in employee and truck traffic levels.

4.5.2.2.3 Air Quality

Air quality impacts associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.3.

Impacts associated with this option at INEEL were determined to be the same as under Option 2 of Alternative 2 (Section 4.4.2.1.3).

The air quality impacts of transportation are presented in Section 4.5.2.2.11.

4.5.2.2.4 Water Resources

Impacts on water resources associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.4.

Building CPP–651 and/or FDPF, existing facilities in the INTEC area of INEEL, would be used for neptunium-237 storage; FDPF would also be used for the fabrication and processing of targets in support of plutonium-238 production. Impacts on water resources indicators at INEEL would be the same as those

described in Section 4.3.2.1.4. In summary, a small increase in water use and sanitary wastewater generation would be anticipated, mainly attributable to increased staffing levels. Also, there would be a very small increase in process wastewater generation, but there would be no radiological liquid effluent discharge to the environment under normal operations.

4.5.2.2.5 Geology and Soils

Impacts on geology and soils associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.5. As discussed in Section 4.5.1.1.5, the proposed facilities would be designed and constructed in accordance with DOE Order 420.1 and sited to minimize the risk from geologic hazards. Thus, site geologic conditions would be unlikely to affect the facilities.

The use of Building CPP–651 and/or FDPF for neptunium-237 storage, and FDPF for target fabrication and processing would not be expected to impact geologic resources, nor be jeopardized by large-scale geologic conditions, at INEEL. Hazards from large-scale geologic conditions at INEEL, such as earthquakes and volcanoes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-148) as discussed in Section 4.2.3.2.5. The analysis determined that these hazards present a low risk to long-term storage facilities. That analysis was reviewed in the *Surplus Plutonium Disposition EIS* (DOE 1999a:4-267-268). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to the proposed use of the INTEC facilities. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards will be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.5.2.2.6 Ecological Resources

Impacts on ecological resources associated with the operation of one or two accelerators and a support facility are addressed in Section 4.5.1.2.6.

Building CPP–651 and/or FDPF would be used for neptunium-237 storage, and FDPF for target fabrication and processing. As noted in Section 4.5.2.2.2, there would be little change in noise impacts on wildlife. Because additional water usage and wastewater discharge would be small fractions of current values, there would be no impact on aquatic resources (Section 4.5.2.2.4). Threatened and endangered species would not be affected by operation because an existing facility(s) within an already developed area would be used.

Consultation letters to comply with Section 7 of the Endangered Species Act were sent to the U.S. Fish and Wildlife Service and the Idaho Department of Fish and Game (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. The Idaho Department of Fish and Game indicated that its database contained no known occurrences of special status plants or animals near the project area. While DOE has made additional contact with the U.S. Fish and Wildlife Service, a response is pending from this agency. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at INEEL prior to the receipt of input from the Service.

4.5.2.2.7 Cultural and Paleontological Resources

Impacts on cultural and paleontological resources associated with the operation of one or two accelerators and a support facility are addressed in Section 4.5.1.2.7.

Although six historic structures are associated with INTEC, their status would not be affected by the operation of Building CPP–651 and/or FDPF for neptunium-237 storage, and FDPF for target fabrication and processing. Also, the status of Native American and paleontological resources occurring in the vicinity of INTEC would not be affected by operation of these facilities.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). The State Historic Preservation Office indicated that Building CPP-651 and FDPF are likely to be eligible for the National Register of Historic Places as contributory properties in a potential historic district of exceptional significance. However, at this time, the State Historic Preservation Office has determined that more information is needed prior to assisting DOE in evaluating these properties. The State Historic Preservation Office also indicated that since there would be no new construction, there is little potential for effects on archaeological properties. DOE would provide additional information as required to the Idaho State Historic Preservation Office prior to the use of any facility at INEEL for the proposed project. Consultation was conducted with interested Native American tribes; however, responses are pending.

4.5.2.2.8 Socioeconomics

The socioeconomic impacts associated with the operation of the accelerator(s) and support facility at a generic DOE site are addressed in Section 4.5.1.2.8.

The socioeconomic impacts associated with neptunium-237 target fabrication and processing at INEEL are addressed in Section 4.3.2.1.8.

4.5.2.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup and operations are given in **Table 4–132** for the generic DOE accelerator(s) site and INEEL: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–132 Incremental Radiological Impacts on the Public Around the Generic DOE Site and INEEL from Operational Facilities Under Alternative 3 (Construct New Accelerator[s])—Option 2

Receptor	INEEL FDPF	Accelerators Preoperational Startup[a]		Generic Site Operations				Two-Site Total
				Accelerators		Accelerator(s) Support Facility	Total	
		Low-Energy	High-Energy	Low-Energy	High-Energy			
Population within 80 kilometers (50 miles) in the year 2020								
Dose (person-rem)	3.9×10^{-6}	0.0024	0.035	0.0043	0.055	0.14	0.20	0.20
35-year latent cancer fatalities	6.7×10^{-8}	$2.4\times10^{-6(b)}$	$3.5\times10^{-5(b)}$	7.5×10^{-5}	9.6×10^{-4}	0.0025	0.0035	0.0035
Maximally exposed individual								
Annual dose (millirem)	2.6×10^{-7}	1.4×10^{-5}	1.8×10^{-4}	1.1×10^{-4}	8.7×10^{-4}	0.0025	0.0035	NA[c]
35-year latent cancer fatality risk	4.6×10^{-12}	$1.4\times10^{-11(b)}$	$1.8\times10^{-10(b)}$	1.9×10^{-9}	1.5×10^{-8}	4.4×10^{-8}	6.1×10^{-8}	NA[c]
Average exposed individual within 80 kilometers (50 miles)								
Annual dose[d] (millirem)	2.0×10^{-8}	1.6×10^{-6}	2.3×10^{-5}	2.8×10^{-6}	3.6×10^{-5}	9.1×10^{-5}	1.3×10^{-4}	NA[c]
35-year latent cancer fatality risk	3.6×10^{-13}	$1.6\times10^{-12(b)}$	$2.3\times10^{-11(b)}$	4.9×10^{-11}	6.3×10^{-10}	1.6×10^{-9}	2.3×10^{-9}	NA[c]

a. For conservatism as well as consistency with other radiological impacts evaluated in this NI PEIS, these values were assessed for the year 2020 even though these activities would commence prior to that year.
b. Preoperational activities last 2 years. Number is a 2-year latent cancer fatality risk.
c. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
d. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FDPF or the generic site in the year 2020 (188,400 and 1,538,100, respectively).

Key: NA, not applicable.
Source: SAIC 2000; TechSource 2000; model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual operations of the accelerator facilities (a high-energy accelerator, a low-energy accelerator, and a support facility) and FDPF, the projected incremental total population dose in the year 2020 would be 0.20 person-rem; the corresponding number of latent cancer fatalities in the populations surrounding the generic DOE site and INEEL from 35 years of operations would be 0.0035. The incremental total dose to the maximally exposed member of the public from annual operations of the accelerator(s) and support facility at the generic DOE site would be 0.0035 millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 6.1×10^{-8}. Estimated annual risks are also presented for pre-operational testing/startup phase activities anticipated for the accelerator(s) and support facility. The incremental dose to the maximally exposed member of the public from annual FDPF operations would be 2.6×10^{-7} millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 4.6×10^{-12}.

Incremental doses to involved workers from normal operations are given in **Table 4–133**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to the high-energy and low-energy accelerator workers during startup and operations would be 150 millirem; for support facility workers, the incremental annual average dose during operations would be 114 millirem; for FDPF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities is estimated to be 30 (high-energy accelerator startup and operation), 15 (low-energy accelerator startup and operation), 11 (support facility operation), and 12 person-rem (FDPF operation). The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–133. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–133 Incremental Radiological Impacts on Involved Workers at the Generic DOE Site and INEEL from Operational Facilities Under Alternative 3 (Construct New Accelerator[s])—Option 2

Receptor—Involved Workers[a]	INEEL FDPF	Accelerators Preoperational Startup		Generic Site Operations			Two-Site Total
				Accelerators		Accelerator(s) Support Facility	
		Low-Energy	High-Energy	Low-Energy	High-Energy		
Total dose (person-rem per year)	12[b]	23[b]	45[b]	15[b]	30[b]	11[b]	69
35-year latent cancer fatalities	0.17	0.018[c]	0.036[c]	0.21	0.42	0.16	0.96
Average worker dose (millirem per year)	170	150	150	150	150	114	NA[d]
35-year latent cancer fatality risk	0.0023	1.2×10^{-4}[(c)]	1.2×10^{-4}[(c)]	0.0021	0.0021	0.0016	NA[d]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that each facility adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers at INEEL FDPF, 200 at the high-energy accelerator (300 during startup), 100 at the low-energy accelerator (150 during startup), and 100 total workers at the accelerator(s) support facility.
c. Preoperational startup testing lasts 2 years. Number is a 2-year latent cancer fatality risk.
d. Values cannot be given for the average worker because the workers would be in three different facilities at two different sites.

Key: NA, not applicable.
Source: DOE 1999b; Nielsen 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.9.

Impacts from hazardous chemicals at INEEL were determined to be the same as under Option 2 of Alternative 2 (Section 4.4.2.1.9).

4.5.2.2.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with accelerator target irradiation; support facility fabrication and processing of medical, industrial, and research and development isotopes; and FDPF target fabrication and processing of neptunium-237 targets are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–134** and **4–135**, respectively.

For 35 years of high-energy accelerator target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.05×10^{-6} and 5.15×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0063.

For 35 years of low-energy accelerator target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.45×10^{-9} and 3.07×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 3.66×10^{-5}.

Table 4–134 New Accelerator(s), Support Facility, and FDPF Accident Consequences Under Alternative 3 (Construct New Accelerator[s])—Option 2

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
High-energy accelerator accidents						
Design-basis target accident	2.93×10^{-4}	1.47×10^{-7}	0.980	4.90×10^{-4}	9.35×10^{-4}	3.74×10^{-7}
Beyond-design-basis earthquake	11.7	0.00585	3.01×10^{4}	18	184	0.147
Low-energy accelerator accidents						
Design-basis target accident	8.05×10^{-5}	4.03×10^{-8}	17.7	0.00885	0.00112	4.48×10^{-7}
Beyond-design-basis earthquake	0.0132	6.60×10^{-6}	32.4	0.0162	0.208	8.32×10^{-5}
Support facility accidents						
Medical and industrial isotopes localized solvent fire	0.0194	9.72×10^{-6}	31.1	0.0156	0.00530	2.12×10^{-6}
Medical and industrial isotopes unlikely seismic event	0.0750	3.75×10^{-5}	136	0.0680	0.510	2.04×10^{-4}
Medical and industrial isotopes glovebox explosion	2.50	0.00125	4,600	2.30	17.0	0.00680
FDPF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.01×10^{-9}	1.01×10^{-12}	2.49×10^{-5}	1.24×10^{-8}	7.26×10^{-9}	2.91×0^{-12}
Target dissolver tank failure during plutonium-238 separation	6.11×10^{-8}	3.05×10^{-11}	5.65×10^{-4}	2.82×10^{-7}	2.17×10^{-7}	8.69×10^{-11}
Ion exchange explosion during plutonium-238 separation	1.63×10^{-5}	8.13×10^{-9}	0.150	7.51×10^{-5}	5.79×10^{-5}	2.31×10^{-8}
Processing facility beyond-design-basis earthquake	42.5	0.0425	1.64×10^{5}	82.0	1,200	1.00[c]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–135 New Accelerator(s), Support Facility, and FDPF Accident Risks Under Alternative 3 (Construct New Accelerator[s])—Option 2

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual high-energy accelerator risks			
Design-basis target accident (1×10^{-4})	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk	2.05×10^{-6}	0.00630	5.15×10^{-5}
Annual low-energy accelerator risks			
Design-basis target accident (1×10^{-4})	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
35-year low-energy accelerator risk	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
Annual support facility risks			
Medical and industrial isotopes localized solvent fire (0.044)	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical and industrial isotopes glovebox explosion (1.00×10^{-4})	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
35-year support facility risk	3.26×10^{-5}	0.056	9.85×10^{-5}
Annual FDPF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Processing facility beyond-design-basis earthquake (1×10^{-5})	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}(c)
35-year FDPF risk	1.49×10^{-5}	0.0287	3.50×10^{-4}(c)
35-year Option risk	4.95×10^{-5}	0.0911	5.00×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

For 35 years of medical, industrial, and research and development target fabrication and processing at the support facility, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 3.26×10^{-5} and 9.85×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.056.

For 35 years of neptunium-237 target fabrication and processing at FDPF, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 1.49×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0287.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 4.95×10^{-5} and 5.00×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0911.

There are no hazardous chemical accidents associated with the new accelerator(s) or new support facility. The irradiation of neptunium-237, medical, industrial, and research and development isotopes in the new accelerator(s) would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The fabrication and processing of medical, industrial, and research and development isotopes at the new support facility would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The hazardous chemical accident impacts at FDPF are the same as those presented in Section 4.4.5.1.10.

4.5.2.2.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to the accelerator(s) site. Following irradiation in the accelerator(s), the targets would be returned to REDC for processing. After this processing, the plutonium-238 product would be shipped to LANL. Additionally, medical and industrial isotopes would be shipped from the accelerator(s) site to a local airport, and from there to locations throughout the country.

Approximately 37,000 shipments of radioactive materials would be made by DOE in support of the low-energy accelerator. The total distance traveled on public roads by trucks carrying radioactive materials would be 4.8 million kilometers (3.0 million miles); and in the air carrying medical and industrial isotopes, 23 million kilometers (14 million miles).

Approximately 269 shipments of radioactive materials would be made by DOE in support of the high-energy accelerator. The total distance traveled on public roads by trucks carrying radioactive materials would be 0.99 million kilometers (0.62 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION FOR THE LOW-ENERGY ACCELERATOR. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 15 person-rem; the dose to the public, 7 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0059 latent cancer fatality among transportation workers and 0.0037 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.02.

IMPACTS OF INCIDENT-FREE TRANSPORTATION FOR THE HIGH-ENERGY ACCELERATOR. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 6 person-rem; the dose to the public, 107 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.002 latent cancer fatality among transportation workers and 0.054 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.0023.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION FOR THE LOW-ENERGY ACCELERATOR. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: more than 1 in 10 million per year) would not breach the transportation package. The consequences of more severe accidents

that could breach the transportation package and release radioactive material were evaluated and estimated to have probabilities of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting 0.11 traffic fatality.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION FOR THE HIGH-ENERGY ACCELERATOR. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 0.16 person-rem, resulting in 8.1×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.025 traffic fatality.

4.5.2.2.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in one or two new accelerators that would be constructed at a site yet to be specified. Fabrication and processing of neptunium-237 targets for plutonium-238 production would be performed at FDPF located at INEEL. A new support facility would be constructed at the same site for fabrication and processing targets not used for plutonium-238 production.

Activities at FDPF were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.2.1.12) and found to pose no significant radiological or other risks to minority and low-income populations. The environmental analysis of operations at the new accelerator(s) and support facility site shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected areas would not be significant. Unless there are patterns of food consumption among minority or low-income residents surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that operations at the site would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new accelerator(s) and support facility, an additional evaluation of environmental justice at the accelerator(s) and support facility site during operation would be performed prior to implementation.

4.5.2.2.13 Waste Management

The impacts of managing waste associated with the operation of new accelerator(s) to irradiate targets and a support facility to fabricate and process medical and industrial isotope targets and to meet research and development needs are assumed to be the same as for Option 1 (Section 4.5.1.2.13). Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.5.2.2.9 through 4.5.2.2.11.

The impacts of managing waste associated with fabricating and processing neptunium-237 targets for plutonium-238 production in FDPF at INEEL are assumed to be the same as for Option 2 under Alternative 1 (Section 4.3.2.1.13). As shown in that section, the impacts on the waste management systems at INEEL would be small.

4.5.2.3 Decontamination and Decommissioning of the Accelerator(s) and Support Facility

The environmental impacts associated with the decontamination and decommissioning of the accelerator(s) and support facility at the generic DOE site are assessed in Section 4.5.1.3.

4.5.2.4 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.5.3 Alternative 3 (Construct New Accelerator[s])—Option 3

Option 3 involves constructing and operating one or two accelerators to irradiate all targets associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating FMEF at Hanford to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and conducting and operating the support facility to fabricate and process the other targets and materials and to process the associated products. This option includes storage in FMEF of the neptunium-237 transported to Hanford from SRS and storage in the new support facility of the other target materials transported to the generic site from other offsite facilities.

The transportation of the neptunium-237 from SRS to Hanford and then to the generic site, the transportation of the other target materials to the generic site, and the transportation of plutonium-238 and other product materials following irradiation and postirradiation processing constitute part of this option.

All options under this alternative include the decontamination and decommissioning of the accelerator(s) and support facility at the generic site following their operating lifetimes, and also the permanent deactivation of FFTF at Hanford.

4.5.3.1 Construction of the New Accelerator(s) and Support Facility

Environmental impacts associated with the construction of one or two new accelerators and support facility at the generic DOE site are assessed in Section 4.5.1.1.

4.5.3.2 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.5.3.2.1 Land Resources

LAND USE. Impacts on land use associated with the operation of the accelerator(s) and a support facility are addressed in Section 4.5.1.2.1.

FMEF would be used for neptunium-237 storage, target fabrication, and processing. Land use within the 400 Area would not change since the use of FMEF would be compatible with the mission for which it was designed.

VISUAL RESOURCES. Impacts on visual resources associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.1.

All activities associated with neptunium-237 storage, target fabrication, and processing would take place within FMEF. Operations associated with the proposed activities would not result in any change to visual resources; thus, the current Visual Resource Management Class IV rating of the 400 Area. This is because none of the anticipated operational impacts (e.g., air emissions) would be expected to affect this resource.

4.5.3.2.2 Noise

Noise impacts associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.2.

This option also involves using FMEF for target material storage, target fabrication, and processing. Activities associated with construction of a new stack would be typical of small construction projects and would result in some temporary increase in noise. Noise sources associated with this construction would not be expected to be loud impulsive sources and are not expected to result in disturbance of wildlife around the 400 Area. The operation of FMEF would not be expected to result in any change in noise impacts on wildlife around the 400 Area and offsite noise impacts would also be minor because the nearest site boundary is 7 kilometers (4.3 miles) to the east. Operation would be expected to result in a minimal change in noise impacts on people near the Hanford site as a result of changes in employee and truck traffic levels.

4.5.3.2.3 Air Quality

Air quality impacts associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.3.

Air quality impacts at Hanford associated with this option were determined to be the same as under Option 3 of Alternative 2 (Section 4.4.3.1.3).

The air quality impacts of transportation are presented in Section 4.5.3.2.11.

4.5.3.2.4 Water Resources

Impacts on water resources associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.4.

FMEF in the 400 Area of Hanford would be used for neptunium-237 storage, target fabrication, and processing in support of plutonium-238 production. The operation of FMEF for this purpose is projected to require approximately 19 million liters (5 million gallons) of groundwater annually. This includes approximately 15 million liters (4 million gallons) per year to support FMEF cooling needs and an additional 3.8 million liters (1 million gallons) per year for potable and sanitary water demands due to increased staffing. However, no impact on regional groundwater levels would be expected from increased withdrawals. FMEF groundwater usage would constitute an increase of about 10 percent over the 197 million liters (52 million gallons) withdrawn annually in the 400 Area during standby operations. Sanitary wastewater discharges from FMEF would also increase by roughly 3.8 million liters (1 million gallons) per year to the Energy Northwest treatment system, which has sufficient capacity. Also, the operation of FMEF for target fabrication and processing would generate approximately 15 million liters (4 million gallons) per year of process wastewater. This wastewater would be discharged to the 400 Area process sewer system and ultimately to the 400 Area Pond (i.e., 4608 B/C percolation ponds) (Chapin 2000; Nielsen 1999:38, 39, 41). As discharges to the pond are

regulated under State Waste Discharge Permit No. ST-4501 and there are no radiological liquid effluent pathways to the environment from FMEF, the impact on groundwater quality would be negligible.

It should be noted that the increase in water use and sanitary and process wastewater discharge for FMEF operations would essentially be negated by the larger reductions in water use and wastewater generation in the 400 Area associated with the permanent deactivation of FFTF (see Section 4.4.1.2.4).

4.5.3.2.5 Geology and Soils

Impacts on geology and soils associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.5. As discussed in Section 4.5.1.1.5, the proposed facilities would be designed and constructed in accordance with DOE Order 420.1 and sited to minimize the risk from geologic hazards. Thus, site geologic conditions would be unlikely to affect the facilities.

The use of FMEF for neptunium-237 storage, target fabrication, and processing would not be expected to impact geologic resources, nor be jeopardized by large-scale geologic conditions. Hazards from large-scale geologic conditions at Hanford, such as earthquakes and volcanoes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-45) as discussed in Section 4.2.4.2.5. The analysis determined that these hazards present a low risk to long-term storage facilities. That analysis was reviewed in the *Surplus Plutonium Disposition EIS* (DOE 1999a:4-260). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to FMEF operations. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards will be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.5.3.2.6 Ecological Resources

Impacts on ecological resources associated with the operation of the accelerator(s) and a support facility are addressed in Section 4.5.1.2.6.

This option also involves using FMEF for neptunium-237 storage, target fabrication, and processing. As noted in Section 4.5.3.2.2, there would be no change in noise impacts on wildlife. Because additional water usage and wastewater discharge would be small fractions of current values, there would be no change in impacts on aquatic habitat or wetlands associated with the Columbia River (Section 4.5.3.2.4). Threatened and endangered species would not be affected by operation because an existing facility within an already developed area would be used.

Consultation letters concerning threatened and endangered species were sent to the U.S. Fish and Wildlife Service, the National Marine Fisheries Service, the Washington State Department of Natural Resources, and the State of Washington Department of Fish and Wildlife (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. Both the Washington State Department of Natural Resources and the State of Washington Department of Fish and Wildlife provided lists of state species of concern that occur in the vicinity of the project area. As noted above, no impacts to any threatened or endangered species are expected, including those of concern to these agencies. While DOE has made additional contacts with the U.S. Fish and Wildlife Service and the National Marine Fisheries Service, responses are pending from these agencies. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at Hanford prior to the receipt of input from these Federal agencies.

4.5.3.2.7 Cultural and Paleontological Resources

Impacts on cultural and paleontological resources associated with the operation of the accelerator(s) and a support facility are addressed in Section 4.5.1.2.7.

Neptunium-237 storage, target fabrication, and processing would take place at FMEF, which is in the 400 Area. No prehistoric, historic, or paleontological sites have been identified either within the 400 Area or within 2 kilometers (1.2 miles) of the 400 Area. Six buildings located within the 400 Area, including two FFTF structures (the Reactor Containment Building and FFTF Control Building), have been determined to be eligible for the National Register as contributing properties within the Historic District recommended for mitigation. The operation of FMEF would not affect the status of these structures. No Native American resources are known to occur within the 400 Area.

Consultation to comply with Section 106 of the National Historic Preservation Act was conducted with the State Historic Preservation Office (see Table 5–3) and resulted in concurrence by the State Historic Preservation Office that the proposed action would have no effect on historic properties at Hanford. Consultation was also conducted with interested Native American tribes that resulted in comments at public hearings by members representing the Nez Perce and Confederated Tribes of the Umatilla Indian Reservation. Responses to their specific comments are addressed in Volume 3.

4.5.3.2.8 Socioeconomics

The socioeconomic impacts associated with the operation of the new accelerator(s) and support facility at a generic DOE site are addressed in Section 4.5.1.2.8.

Target fabrication and processing of neptunium-237 targets at FMEF at Hanford would require about 62 additional workers (Hoyt et al. 1999). The socioeconomic impacts at Hanford are the same as those addressed in Section 4.4.3.1.8.

4.5.3.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup and operations are given in **Table 4–136** for the generic DOE site and Hanford: the population within 80 kilometers (50 miles)in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

Table 4–136 Incremental Radiological Impacts on the Public Around the Generic DOE Site and Hanford from Operational Facilities Under Alternative 3 (Construct New Accelerator[s])—Option 3

Receptor	Hanford FMEF	Accelerators Preoperational Startup[a]		Generic Site Operations				Two-Site Total
				Accelerators		Accelerator(s) Support Facility	Total	
		Low-Energy	High-Energy	Low-Energy	High-Energy			
Population within 80 kilometers (50 miles) in the year 2020								
Dose (person-rem)	4.4×10^{-5}	0.0024	0.035	0.0043	0.055	0.14	0.20	0.20
35-year latent cancer fatalities	7.7×10^{-7}	$2.4\times10^{-6(b)}$	$3.5\times10^{-5(b)}$	7.5×10^{-5}	9.6×10^{-4}	0.0025	0.0035	0.0035
Maximally exposed individual								
Annual dose (millirem)	4.7×10^{-7}	1.4×10^{-5}	1.8×10^{-4}	1.1×10^{-4}	8.7×10^{-4}	0.0025	0.0035	NA[c]
35-year latent cancer fatality risk	8.3×10^{-12}	$1.4\times10^{-11(b)}$	$1.8\times10^{-10(b)}$	1.9×10^{-9}	1.5×10^{-8}	4.4×10^{-8}	6.1×10^{-8}	NA[c]
Average exposed individual within 80 kilometers (50 miles)								
Annual dose[d] (millirem)	8.9×10^{-8}	1.6×10^{-6}	2.3×10^{-5}	2.8×10^{-6}	3.6×10^{-5}	9.1×10^{-5}	1.3×10^{-4}	NA[c]
35-year latent cancer fatality risk	1.6×10^{-12}	$1.6\times10^{-12(b)}$	$2.3\times10^{-11(b)}$	4.9×10^{-11}	6.3×10^{-10}	1.6×10^{-9}	2.3×10^{-9}	NA[c]

a. For conservatism as well as consistency with other radiological impacts evaluated in this NI PEIS, these values were assessed for the year 2020 even though these activities would commence prior to that year.
b. Preoperational activities last 2 years. Number is a 2-year latent cancer fatality risk.
c. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
d. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FMEF or the generic site in the year 2020 (494,400 and 1,538,100, respectively).

Key: NA, not applicable.
Source: SAIC 2000; TechSource 2000; model results, using the GENII computer code (Napier et al. 1988).

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual operations of the accelerator facilities (the high- and low-energy accelerators and the support facility) and FMEF, the projected incremental total population dose in the year 2020 would be 0.20 person-rem; the corresponding number of latent cancer fatalities in the populations surrounding the generic DOE site and Hanford from 35 years of operations would be 0.0035. The incremental total dose to the maximally exposed member of the public from annual operations of the accelerator(s) and support facility at the generic DOE site would be 0.0035 millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 6.1×10^{-8}. Estimated annual risks are also presented for preoperational testing/startup phase activities anticipated for the accelerator(s) and support facility. The incremental dose to the maximally exposed member of the public from annual FMEF operations would be 4.7×10^{-7} millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 8.3×10^{-12}.

Incremental doses to involved workers from normal operations are given in **Table 4–137**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to the high-energy and low-energy accelerator workers during startup and operations would be 150 millirem; for support facility workers, the incremental annual average dose during operations would be 114 millirem; for FMEF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities is estimated to be 45 (total for both accelerators), 11, and 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–137. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–137 Incremental Radiological Impacts on Involved Workers at the Generic DOE Site and Hanford from Operational Facilities Under Alternative 3 (Construct New Accelerator[s])—Option 3

Receptor—Involved Workers[a]	Hanford FMEF	Accelerators Preoperational Startup		Generic Site Operations			Two-Site Total
				Accelerators		Accelerator(s) Support Facility	
		Low-Energy	High-Energy	Low-Energy	High-Energy		
Total dose (person-rem per year)	12[b]	23[b]	45[b]	15[b]	30[b]	11[b]	69
35-year latent cancer fatalities	0.17	0.018[c]	0.036[c]	0.21	0.42	0.16	0.96
Average worker dose (millirem per year)	170	150	150	150	150	114	NA[d]
35-year latent cancer fatality risk	0.0023	1.2×10^{-4}[(c)]	1.2×10^{-4}[(c)]	0.0021	0.0021	0.0016	NA[d]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that each facility adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.

b. Based on an estimated 75 badged workers at FMEF, 200 at the high-energy accelerator (300 during startup), 100 at the low-energy accelerator (150 during startup), and 100 at the accelerator(s) support facility.

c. Preoperational startup testing lasts 2 years. Number is a 2-year latent cancer fatality risk.

d. Values cannot be given for the average worker because the workers would be in three different facilities at two different sites.

Key: NA, not applicable.

Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts associated with the operation of the accelerator(s) and support facility are addressed in Section 4.5.1.2.9.

Impacts from hazardous chemicals at Hanford were determined to be the same as under Option 3 of Alternative 2 (Section 4.4.3.1.9).

4.5.3.2.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with accelerator target irradiation; support facility fabrication and processing of medical, industrial, and research and development isotopes; and FMEF target fabrication and processing of neptunium-237 targets are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–138** and **4–139**, respectively.

For 35 years of high-energy accelerator target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.05×10^{-6} and 5.15×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0063.

Table 4–138 New Accelerator(s), Support Facility, and FMEF Accident Consequences Under Alternative 3 (Construct New Accelerator[s])—Option 3

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
High-energy accelerator accidents						
Design-basis target accident	2.93×10^{-4}	1.47×10^{-7}	0.980	4.90×10^{-4}	9.35×10^{-4}	3.74×10^{-7}
Beyond-design-basis earthquake	11.7	0.00585	3.01×10^{4}	18	184	0.147
Low-energy accelerator accidents						
Design-basis target accident	8.05×10^{-5}	4.03×10^{-8}	17.7	0.00885	0.00112	4.48×10^{-7}
Beyond-design-basis earthquake	0.0132	6.60×10^{-6}	32.4	0.0162	0.208	8.32×10^{-5}
Support facility accidents						
Medical and industrial isotopes localized solvent fire	0.0194	9.72×10^{-6}	31.1	0.0156	0.00530	2.12×10^{-6}
Medical and industrial isotopes unlikely seismic event	0.0750	3.75×10^{-5}	136	0.0680	0.510	2.04×10^{-4}
Medical and industrial isotopes glovebox explosion	2.50	0.00125	4,600	2.30	17.0	0.00680
FMEF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.02×10^{-9}	1.01×10^{-12}	7.26×10^{-5}	3.63×10^{-8}	6.65×10^{-10}	2.66×10^{-13}
Target dissolver tank failure during plutonium-238 separation	4.64×10^{-8}	2.32×10^{-11}	0.00169	8.47×10^{-7}	1.95×10^{-8}	7.81×10^{-12}
Ion exchange explosion during plutonium-238 separation	1.24×10^{-5}	6.18×10^{-9}	0.451	2.25×10^{-4}	5.20×10^{-6}	2.08×10^{-9}
Processing facility beyond-design-basis earthquake	16.5	0.00823	6.41×10^{5}	321	921	1.00[c]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–139 New Accelerator(s), Support Facility, and FMEF Accident Risks Under Alternative 3 (Construct New Accelerator[s])—Option 3

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual high-energy accelerator risks			
Design-basis target accident (1×10^{-4})	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk	2.05×10^{-6}	0.00630	5.15×10^{-5}
Annual low-energy accelerator risks			
Design-basis target accident (1×10^{-4})	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
35-year low-energy accelerator risk	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
Annual support facility risks			
Medical and industrial isotopes localized solvent fire (0.044)	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical and industrial isotopes glovebox explosion (1.00×10^{-4})	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
35-year support facility risk	3.26×10^{-5}	0.056	9.85×10^{-5}
Annual FMEF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Processing facility beyond-design-basis earthquake (1×10^{-5})	8.23×10^{-8}	0.00321	$1.00\times10^{-5(c)}$
35-year FMEF risk	2.88×10^{-6}	0.112	$3.50\times10^{-4(c)}$
35-year Option risk	3.76×10^{-5}	0.175	5.00×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

For 35 years of low-energy accelerator target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.45×10^{-9} and 3.07×10^{-8}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 3.66×10^{-5}.

For 35 years of medical, industrial, and research and development target fabrication and processing at the support facility, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 3.26×10^{-5} and 9.85×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.056.

For 35 years of neptunium-237 target fabrication and processing at FMEF, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 2.88×10^{-6} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.112.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 3.76×10^{-5} and 5.00×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.175.

There are no hazardous chemical accidents associated with the new accelerator(s) or new support facility. The irradiation of neptunium-237, medical, industrial, and research and development isotopes in the new accelerator(s) would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The fabrication and processing of medical, industrial, and research and development isotopes at the new support facility would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The hazardous chemical accident impacts at FMEF are the same as those presented in Section 4.4.6.1.10.

4.5.3.2.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-37 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to the accelerator(s) site. Following irradiation in the accelerator(s), the targets would be returned to REDC for processing. After this processing, the plutonium-238 product would be shipped to LANL. Additionally, medical and industrial isotopes would be shipped from the accelerator(s) site to a local airport, and from there to locations throughout the country.

Approximately 37,000 shipments of radioactive materials would be made by DOE in support of the low-energy accelerator. The total distance traveled on public roads by trucks carrying radioactive materials would be 4.8 million kilometers (3.0 million miles); and in the air carrying medical and industrial isotopes, 23 million kilometers (14 million miles).

Approximately 269 shipments of radioactive materials would be made by DOE in support of the high-energy accelerator. The total distance traveled on public roads by trucks carrying radioactive materials would be 1.1 million kilometers (0.71 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION FOR THE LOW-ENERGY ACCELERATOR. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 15 person-rem; the dose to the public, 7 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.0059 latent cancer fatality among transportation workers and 0.0037 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.02.

IMPACTS OF INCIDENT-FREE TRANSPORTATION FOR THE HIGH-ENERGY ACCELERATOR. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 7 person-rem; the dose to the public, 123 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.003 latent cancer fatality among transportation workers and 0.0061 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.0026.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION FOR THE LOW-ENERGY ACCELERATOR. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: more than 1 in 10 million per year) would not breach the transportation package. The consequences of more severe accidents that could breach the transportation package and release radioactive material were evaluated and estimated to have probabilities of less than 1 in 10 million per year.

Estimates of the total ground transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting 0.11 traffic fatality.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION FOR THE HIGH-ENERGY ACCELERATOR. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 0.14 person-rem, resulting in 7.2×10^{-5} latent cancer fatality; and traffic accidents resulting in 0.03 traffic fatality.

4.5.3.2.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in one or two new accelerators that would be constructed at a site yet to be specified. Fabrication and processing of neptunium-237 targets for plutonium-238 production would be performed at FMEF located at Hanford. A new support facility would be constructed at the same unspecified site for fabrication and processing targets not used for plutonium-238 production.

Activities at FMEF were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.3.1.12) and found to pose no significant radiological or other risks to minority and low-income populations. The environmental analysis of operations at the new accelerator(s) and support facility site shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected area would not be significant. Unless there are patterns of food consumption among minority or low-income residents surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that operations at the site would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site-specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new accelerator(s) and support facility, an additional evaluation of environmental justice at the accelerator(s) and support facility site during operation would be performed prior to implementation.

4.5.3.2.13 Waste Management

The impacts of managing waste associated with the operation of new accelerator(s) to irradiate targets and a support facility to fabricate and process medical and industrial isotope targets and to meet research and development needs are assumed to be the same as for Option 1 (Section 4.5.1.2.13). Radiological and

chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.5.3.2.9 through 4.5.3.2.11.

The impacts of managing waste associated with fabricating and processing neptunium-237 targets for plutonium-238 production in FMEF at Hanford are assumed to be the same as for Option 3 under Alternative 2 (Section 4.4.3.1.13). As shown in that section, the impacts on the waste management systems at Hanford would be small.

4.5.3.3 Decontamination and Decommissioning of the Accelerator(s) and Support Facility

The environmental impacts associated with the decontamination and decommissioning of the accelerator(s) and support facility at the generic DOE site are assessed in Section 4.5.1.3.

4.5.3.4 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.6 ALTERNATIVE 4—CONSTRUCT NEW RESEARCH REACTOR

Under Alternative 4, a new research reactor would be used for target irradiation for the evaluation period of 35 years. The new research reactor, to be constructed at an existing DOE site, would be used to irradiate all targets (i.e., for the production of plutonium-238, isotopes for medical and industrial uses, and materials testing for civilian nuclear energy research and development). Ongoing operations at existing facilities as described in Chapter 3, Affected Environment, would continue.

The targets for plutonium-238 production would be fabricated in one of the three candidate facilities at ORNL, INEEL, or Hanford. The material needed for the target fabrication (neptunium-237) would be transported from SRS to the fabrication facilities. The targets would be irradiated at the new research reactor facility and transported back to the target fabrication facilities for postirradiation processing.

Targets for medical and industrial isotope production would be fabricated in a new support facility located at the same site as the new research reactor. The targets would be irradiated in the new research reactor and returned to the new support facility for postirradiation processing.

Alternative 4 site selection is not evaluated as part of this NI PEIS. Because Alternative 4 is evaluated at a generic DOE site, no credit was taken for any existing support infrastructure existing at the site and it was postulated that a new support facility would be required to support operation of the new research reactor and its missions. While this approach bounds the environmental impact assessment for the implementation of Alternative 4, it overstates the impacts because this NI PEIS integrates the impacts associated with constructing a new support facility and infrastructure that may be available at the existing DOE site. In the event that Alternative 4 is selected by the Record of Decision for subsequent consideration, follow-up NEPA assessments would evaluate potential locations for the new research reactor. It is unlikely that DOE would consider locating the new research reactor on a DOE site that does not have existing infrastructure capable of supporting all or most of the medical and industrial isotope production and civilian nuclear energy research and development mission requirements. If the reactor were built on a DOE site with existing support facilities, the environmental impacts of such implementation could be determined by subtracting the construction and decommissioning impacts associated with the new support facility from the total impacts given for this alternative.

Under Alternative 4, nonirradiated targets, irradiated targets, and processed materials would be transported between the locations selected for storage, target fabrication, target irradiation, postirradiation processing, and the final destination of the plutonium-238. Alternative 4 also would include the decontamination and decommissioning of both the research reactor and the support facility when the missions are over, as well as deactivation of FFTF at Hanford.

The proposed new research reactor would provide ample neutrons for the production of plutonium-238 and for many of the isotopes listed in Table 1–1. The thermal flux would limit the new research reactor's ability to produce a number of isotopes requiring fast or high-energy neutrons. Its lower flux levels (10^{13} neutrons per square centimeter per second) and predominantly thermal flux would limit its ability to support many of the projected nuclear-based research and development needs.

The three options under this alternative and their associated target fabrication, postirradiation processing, and transportation activities are discussed below.

- **Option 1.** REDC at ORNL would be used to fabricate and process the neptunium-237 targets associated with plutonium-238 production. The neptunium-237 transported from SRS to ORNL would be stored at REDC. The plutonium-238 product would be transported from ORNL to LANL.

A new support facility at an existing DOE site would be used to fabricate and process the targets required for the production of medical and industrial and research isotopes and to store the materials needed for target fabrication.

- **Option 2.** FDPF at INEEL would be used to fabricate and process the neptunium-237 targets associated with plutonium-238 production. The neptunium-237 transported from SRS to INEEL would be stored in FDPF or Building CPP–651. The plutonium-238 product would be transported from INEEL to LANL. A new support facility at an existing DOE site would be used to fabricate and process the targets required for the production of medical and industrial and research isotopes and to store the materials needed for target fabrication.

- **Option 3.** FMEF at Hanford would be used to fabricate and process neptunium-237 targets for plutonium-238 production. The neptunium-237 transported from SRS to Hanford would be stored in FMEF. The plutonium-238 product would be transported from Hanford to LANL. A new support facility at an existing DOE site would be used to fabricate and process the targets required for the production of medical and industrial and research isotopes and to store the materials needed for target fabrication.

The incremental environmental impacts associates with each option are presented separately for the research reactor and the support facility because both facilities may not be selected together. This segmentation assists in the selection of facilities from the two possible combinations, that is, research reactor plus support facility or research reactor only.

As described in Section 1.2.3, the civilian nuclear energy research and development initiatives requiring an enhanced DOE nuclear infrastructure fall into three basic categories: materials research, nuclear fuels research, and advanced reactor development.

- Materials research involves irradiating materials in a high-flux field to determine the radiation effect during reactor normal operating conditions or to perform accelerated life-cycle testing. This form of testing would not introduce material into the research reactor that would result in additional releases during normal operation or accident conditions.

- Nuclear fuels research involves irradiating test fuel pellets, fuel pins, and fuel assemblies in high-temperature environments expected in future reactor designs. When the test specimens are inserted into the research reactor, there would be no significant increase of fissile material in the reactor core inventory that would result in additional releases during normal operation or accident conditions.

- Advanced reactor development involves test loop experiments under prototypical reactor conditions. When the test loop is operating in the research reactor core, there would be no significant increase of fissile material in the reactor core inventory that would result in additional releases during normal operation or accident conditions.

The environmental impacts associated with implementation of the proposed civilian nuclear energy research and development mission cannot be distinguished from the impacts of operating the new research reactor without the civilian nuclear energy research and development mission.

4.6.1 Alternative 4 (Construct New Research Reactor)—Option 1

Option 1 involves constructing and operating the research reactor to irradiate all targets associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating

REDC at ORR to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and constructing and operating the support facility to fabricate and process the other targets and materials and to process the associated products. This option includes storage in REDC of the neptunium-237 transported to ORR from SRS and storage in the new support facility of the other target materials transported to the generic site from other offsite facilities.

The transportation of the low-enriched uranium fuel for use in the research reactor, the transportation of the neptunium-237 to ORR and then to the generic site, the transportation of the other target materials to the generic site, and the transportation of all product materials following irradiation and postirradiation processing are also part of the option.

All options under this alternative include the decontamination and decommissioning of the research reactor and support facility at the generic DOE site following their operating lifetimes, and also the permanent deactivation of FFTF at Hanford.

4.6.1.1 Construction of the New Research Reactor and Support Facility

The environmental impacts associated with the construction of a new research reactor and support facility at the generic DOE site are assessed in this section. If the research reactor were built on a site with existing support facilities, there would be no impacts associated with constructing a new support facility.

4.6.1.1.1 Land Resources

LAND USE. The construction of a research reactor and support facility at a generic DOE site would disturb 1.6 hectares (4 acres) and 2.4 hectares (6 acres), respectively. Since the exact nature of the construction site is not known at this time (e.g., whether it has been previously disturbed or not), potential effects on land use cannot be determined. In general, if a location in a previously developed portion of a generic DOE site were selected, impacts on land use would be minimal. However, if an undisturbed location were chosen, land use would change from its present designation to industrial. If the reactor alternative were selected, tiered NEPA documentation would permit an exact determination of impacts on land use.

VISUAL RESOURCES. Impacts from construction of a research reactor and support facility to visual resources at a generic DOE site would depend on the specific location selected. Impacts could include a change in the present Visual Resource Management rating of the site and/or increase in visibility of the site from offsite locations due to the presence of new structures. If construction took place on undeveloped land, the Visual Resource Management rating could change from Class II or III (ratings typical of undeveloped portions of many DOE sites) to Class IV. If a previously developed location were chosen for the reactor, the Visual Resource Management rating would remain Class IV. In either case, new facilities may impact the view from off site locations by increasing the industrial nature of the viewshed. This impact would be more likely at a western site due to the generally level terrain and sparse vegetation. Specific impacts on visual resources would be determined in tiered NEPA documentation if the reactor alternative were selected.

4.6.1.1.2 Noise

Construction of a research reactor and support facility would result in some increase in noise levels from the use of earthmoving, materials handling, and impact equipment; employee vehicles; and truck traffic. Noise from construction activities, especially impulsive noise, would be expected to disturb wildlife in the immediate area of the construction site. The change in noise levels in areas outside the DOE site would be dependent on the location selected and the exact nature of the construction location and activities required. However, generally if the location selected were within one of the larger DOE sites and more centrally located within

the site, offsite noise impacts from construction activities can be expected to be small. Construction employee vehicles and truck traffic would result in an increase in traffic noise along roads used to access the site. However, this increase in traffic noise would be small unless the construction traffic volume is as large as the existing site traffic. Site-specific analysis would be conducted in tiered NEPA documentation if the reactor alternative were selected.

4.6.1.1.3 Air Quality

Construction of the new research reactor and support facility would result in an increase in employee vehicles and truck traffic. Criteria pollutants were modeled and compared to the most stringent standards (**Table 4–140**). The maximum ground-level concentrations that would result from reactor construction would be below the ambient air quality standards, although concentrations of some pollutants (i.e., PM_{10}) would be relatively high. Therefore, if the reactor were in an area that already had high background pollutant concentrations, resultant pollutant concentrations could approach or exceed the ambient standards. As a result, regulatory compliance would need to be assessed on case-by-case basis. Hazardous chemical emissions from construction have not been identified.

Table 4–140 Incremental Concentrations Associated with Research Reactor Construction Under All Options of Alternative 4 (Construct New Research Reactor)

Pollutant	Averaging Period	Most Stringent Standard or Guideline (microgram per cubic meter)[a]	Modeled Increment (microgram per cubic meter)
Carbon monoxide	8 hours 1 hour	10,000 40,000	72 103
Nitrogen oxide	Annual	100	1
PM_{10}	Annual 24 hours	50 150	3 88

a. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standards.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); data from Appendix E.

4.6.1.1.4 Water Resources

The impacts on key water resource indicators associated with constructing the new research reactor and support facility are presented in **Table 4–141**. Water is expected to be required for such uses as mixing concrete, dust control, washing activities, and potable and sanitary needs. These estimates are annual average values over the forecasted construction periods; these values do not include dewatering of excavations that could be required at some sites. The exact impact of these withdrawals on the resource would depend on the water source (surface water or groundwater) and its relative abundance. These factors would be used to determine the impact on the local and/or regional availability of the resource. Impacts would be expected to be small to negligible due to the relatively small volumes of water required for construction compared to expected site availability.

Sanitary wastewater would be generated by construction personnel and also by facility staff during preoperational testing. Process wastewater could also be generated during construction associated with facility cold-startup and testing of auxiliary systems as construction progresses (e.g., cooling towers). Site selection would make use of existing infrastructure and nearby wastewater treatment facilities would be used to the

Table 4–141 Estimated Water Use and Wastewater Generation Associated with Constructing a New Research Reactor and Support Facility Under All Options of Alternative 4 (Construct New Research Reactor)

Indicator (million liters per year)	New Research Reactor[a]	New Support Facility[a]
Water use	11.7	14.6
Sanitary wastewater generation	11.4	3.6

a. These estimates are annualized values based on projected construction/preoperational testing periods for the new research reactor and the new support facility of 4 years and 4.5 years, respectively.

Note: To convert from liters per year to gallons per year, multiply by 0.264.

Source: Appendix E; SAIC 2000.

extent possible and would be supplemented by portable or temporary facilities during construction as necessary. The potential impact on water resources would depend on the availability and capacity of appropriate treatment facilities. All wastewater would be disposed of in accordance with applicable regulatory requirements with discharges to surface waters in accordance with NPDES effluent limitations.

Ground disturbance and runoff from denuded areas could potentially impact surface water quality near construction areas (Section 4.6.1.1.6). However, appropriate spill prevention practices and soil erosion and sediment control measures (e.g., silt fences, mulching disturbed areas) would be employed during construction to minimize water quality impacts.

Some locations on a generic DOE site could potentially be affected by flooding requiring appropriate siting decisions (refer to Section 3.6.4). Applicable regulatory requirements would be followed in siting facilities including Executive Order 11988, *Floodplain Management*.

Although specific impacts on water resources cannot be determined at this time, site-specific analysis would be conducted in tiered NEPA documentation if the research reactor alternative were selected.

4.6.1.1.5 Geology and Soils

Construction of the research reactor would disturb a total of approximately 1.6 hectares (4 acres) of land. Construction of the support facility would disturb an additional 2.4 hectares (6 acres) of land (Appendix E; Herrington 2000; SAIC 2000). Related impacts on geologic and soil resources cannot be determined at this time since they are site specific in nature. However, impacts would be expected to be less if previously disturbed land were used than if an undeveloped area were selected for construction. In general, construction activities would likely require appreciable quantities of sand and gravel and possibly other geologic materials and, depending on the site chosen, could temporarily deplete local deposits or stockpiles of these materials. Soil erosion potential is also closely related to the amount of land disturbed.

As discussed in Section 3.6.5, the proposed facilities could be located at a generic DOE site with seismic activity ranging from low to moderate. Known capable faults could be located within 19 kilometers (12 miles). However, no known large-scale geologic conditions are present at any generic DOE site that would preclude the construction and operation of properly designed facilities. Appropriate activities and subsurface investigations would be conducted to identify geologic hazards including seismic and volcanic features and other natural hazards (landslide areas, sinkholes, unstable soils) as part of the site selection process. As stated in DOE Order 420.1, DOE requires that nuclear or nonnuclear facilities be designed, constructed, and operated so that the public, the workers, and the environment are protected from the adverse impacts of natural phenomena hazards, including earthquakes. DOE Order 420.1, Section 4.4, as supplemented by DOE Guide 420.1-2, stipulates the natural phenomena hazards mitigation requirements for DOE facilities. Further, the natural phenomena hazards mitigation requirements of DOE Order 420.1 are consistent with the

guidance for seismic design and construction contained in the National Earthquake Hazards Reduction Program (NEHRP) 1997 provisions (BSSC 1997). In addition, DOE Guide 420.1-2 was recently issued to recognize the consolidation of the three previous U.S. model building codes, including the Uniform Building Code (UBC), into the International Building Code (ICC 2000). The DOE requirements for seismic engineering have followed the UBC, unless the importance of achieving a high level of protection warrants the use of more demanding methods and criteria (DOE Guide 420.1-2). Thus, new facilities would be designed and sited in accordance with DOE Order 420.1.

Site-specific analysis would be conducted in tiered NEPA documentation if the research reactor alternative were selected.

4.6.1.1.6 Ecological Resources

If the research reactor alternative were selected, tiered NEPA documentation would be undertaken to determine the exact nature of construction impacts on ecological resources. During that process, impacts on individual species and habitats that are sensitive to disturbance would be determined. This would include consideration of wetlands and threatened and endangered species. Wetland delineations and consultation with the U.S. Fish and Wildlife Service and state wildlife agency would take place, as necessary, to ensure that these resources would be protected.

Construction impacts on ecological resources are site specific. The nature of these impacts would be expected to vary depending on whether the site was located in the eastern or western portion of the United States. In fact, depending on the site location, impacts on some resources may not occur. Additionally, construction impacts on ecological resources would depend on whether the selected location was within an already disturbed portion of the site. In general impacts on terrestrial resources, wetlands, aquatic resources, and threatened and endangered species described below are applicable to an undeveloped site.

Terrestrial Resources. Construction of a research reactor and support facility would disturb 1.6 hectares (4 acres) and 2.4 hectares (6 acres), respectively, of terrestrial habitat. If these facilities were constructed at an undeveloped location, it is likely that woodland habitat would be lost at an eastern generic DOE site and shrubland would be disturbed at a western site. Land clearing activities would affect animal populations. Less mobile animals within the project area, such as reptiles and small mammals, would not be expected to survive. Construction activities and noise would cause larger mammals and birds in the construction and adjacent areas to move to similar habitat nearby. If the area to which they moved was below its carrying capacity, these animals would be expected to survive. However, if the area were already supporting the maximum number of individuals, the additional animals would compete for limited resources that could lead to habitat degradation and eventual loss of the excess population. Nests and young animals living within the disturbed area may not survive.

Wetlands. Clearing and grading operations could result in the direct loss of wetlands, although proper placement of the research reactor and support facility within the overall generic DOE site would eliminate or reduce the potential for such loss. Indirect impacts could also result from stormwater runoff carrying sediments to wetlands located adjacent to the site. Changes in hydrology, water quality, and soils could occur as a result of alterations in water levels, runoff, and the buildup of sediments. These changes could, in turn, alter the vegetative composition of the wetland. In general, both direct and indirect impacts would be more likely to occur at an eastern site due to the greater abundance of wetlands. If preliminary analysis determined that wetlands could be impacted by development, a wetland delineation would be required. Impacts on wetlands could also lead to the implementation of mitigation measures.

Aquatic Resources. During construction of a research reactor and support facility, impacts on aquatic resources could result from stormwater runoff. Runoff could alter flow rates, increase turbidity, and lead to sedimentation of streambeds. These impacts could, in turn, cause temporary and permanent changes in species composition and density, and alter breeding habitats. The implementation of erosion and sediment control procedures would lessen construction impacts.

Threatened and Endangered Species. Construction of a research reactor and support faculty would have the potential to impact threatened and endangered species. Sources of impacts would be similar to those discussed above for terrestrial resources, wetlands, and aquatic resources. The primary difference is that the resource of concern involves individual species that are sensitive to disturbance and whose existence may be threatened by development. Consultations with the U. S. Fish and Wildlife Service and appropriate state agency would be conducted at the site-specific level, as appropriate.

4.6.1.1.7 Cultural and Paleontological Resources

The construction of a research reactor and support facility at a generic DOE site would disturb 1.6 hectares (4 acres) and 2.4 hectares (6 acres), respectively. Since the exact nature of the construction site is not known at this time (e.g., whether it has previously been disturbed or not), potential effects on cultural resources cannot be determined. In general, if a location in a previously developed portion of a DOE generic site were selected, impacts on cultural resources may not occur. However, if an undisturbed location were chosen, cultural resources could be impacted. If the reactor alternative were selected, prehistoric and historic resources, including those that are or may be eligible for listing on the National Register of Historic Places, would be identified. These resources would be identified through site surveys and consultation with the State Historic Preservation Officer. Specific concerns about the presence, type, and location of Native American resources would be addressed through consultation with the potentially affected tribes in accordance with the *National Historic Preservation Act*, the *Native American Graves Protection and Repatriation Act*, and the *American Indian Religious Freedom Act*.

4.6.1.1.8 Socioeconomics

It is estimated that 160 workers would be needed each of the 4 years required to construct the research reactor and support facility at a generic DOE site. The impact of this influx of workers upon the site's region of influence and regional economic area would depend on whether the site were located near a large urbanized area or in a remote rural area. Since the population for the region of influence for a generic site could range from nearly 2.0 million people for a site in a large metropolitan area, to less than 200,000 for a site in a small rural community, the socioeconomic impacts of constructing a new research reactor and support facility would vary greatly. Therefore, if DOE were to select the new research reactor alternative, additional NEPA documentation would be required to select the specific DOE site to locate the new research reactor and support facility. In that document, DOE would perform a thorough evaluation of the socioeconomic impacts of the sites under consideration.

4.6.1.1.9 Public and Occupational Health and Safety—Normal Construction Activities

RADIOLOGICAL IMPACTS. During construction operations, it is not anticipated that there would be any resulting radiological releases to the environment; therefore, no additional dose to the public is expected. Furthermore, construction workers are not expected to receive exposures above natural background levels that exist within the construction areas. However, as a precautionary measure, workers would be badged as deemed appropriate.

HAZARDOUS CHEMICAL IMPACTS. No hazardous chemical releases have been identified from construction activities. Therefore, minimal hazardous chemical impacts would be associated with construction.

4.6.1.1.10 Public and Occupational Health and Safety—Construction Accidents

There are no radiological or hazardous chemical accidents postulated during the construction phases of the new research reactor or support facility. Workers could experience industrial accidents commonly associated with the construction of large facilities.

4.6.1.1.11 Environmental Justice

Section 4.6.1.1 addresses environmental effects due to construction activities that would be expected to occur at an unspecified reactor and support facility site. The analysis shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected areas are not significant. Unless there are patterns of food consumption among minority or low-income residents surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that construction activities would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site-specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new research reactor and support facility, an additional evaluation of environmental justice at the research reactor and support facility site during construction would be performed prior to implementation.

4.6.1.1.12 Waste Management

The expected generation rates of waste at a generic DOE site that would be associated with the construction of a new research reactor to irradiate targets and a support facility to fabricate and process medical and industrial isotope targets and to meet research and development needs are provided in **Table 4–142**. These estimates represent the total amount of waste generated during the construction period. These generation rates cannot be compared at this time with site treatment, storage, and disposal capacities because a DOE site has not yet been chosen for these facilities. Site-specific analyses would be conducted if this alternative were chosen, and appropriate NEPA documentation would be prepared.

Section 3.6.11.1 provides DOE site ranges for each waste type that include volume currently stored, projected generation, and for some types of waste, disposal volume. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.6.1.1.9 through 4.6.1.1.11.

4.6.1.2 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.6.1.2.1 Land Resources

LAND USE. The operation of a research reactor and support facility at a generic DOE site would not be expected to affect land use. This is because none of the anticipated operational impacts (e.g., air emissions) are expected to affect this resource.

Table 4–142 Estimated Waste Generation Associated with Constructing a New Research Reactor and Support Facility Under All Options of Alternative 4 (Construct New Research Reactor)

Waste Type[a]	Estimated Waste Generation for New Research Reactor (total cubic meters)	Estimated Waste Generation for New Support Facility (total cubic meters)
High-level radioactive	0	0
Transuranic	0	0
Low-level radioactive		
Liquid	0	0
Solid	0	0
Mixed low-level radioactive	0	0
Hazardous		
Liquid	1	1
Solid	3	3
Nonhazardous		
Process wastewater	0	0
Sanitary wastewater	44,000	16,000
Solid (kilograms)	1,230,000	230,000

a. See definitions in Section G.9.
Note: To convert from cubic meters to cubic yards, multiply by 1.308.
Source: Appendix E; SAIC 2000.

REDC would be used for neptunium-237 storage, target fabrication, and processing. The use of REDC for this purpose would not change land use at the site since REDC is currently operating and its proposed use would be compatible with its present mission.

VISUAL RESOURCES. The primary source of impacts on visual resources from the operation of a research reactor and support facility would be air emissions. Releases from stacks associated with this alternative would be controlled and, therefore, would be unlikely to exceed Bureau of Land Management Visual Resource Management objectives. However, the operation of cooling towers could result in a visible plume. The extent and visibility of the plume would depend on site meteorological conditions and terrain features. While plume formation would be favored by meteorological conditions at an eastern generic DOE site, terrain features would tend to mask it from offsite locations; the opposite would tend to be true at a western site. If the reactor alternative were selected, the visual impact of the cooling tower plume would be determined in tiered NEPA documentation.

All activities associated with neptunium-237 storage, target fabrication, and processing would take place within REDC. Operations associated with the proposed activities would not result in any impact on visual resources or change in the current Visual Resource Management Class IV rating of the 7900 Area. This is because none of the anticipated operational impacts (e.g., air emissions) would be expected to affect this resource.

4.6.1.2.2 Noise

The operation of a reactor and support facility at a generic DOE site would result in some increase in noise levels from equipment (e.g., cooling systems, vents, motors, generators, compressors, pumps, and material-handling equipment), employee vehicles, and truck traffic. Noise from operation activities could disturb wildlife outside the facility fence line. The change in noise levels in areas outside the DOE site would be dependent on the location selected and the equipment. However, generally if the location selected is within one of the larger DOE sites and is more centrally located within the site, offsite noise impacts from operation can be expected to be small. Operation employee vehicles and truck traffic would result in an increase in traffic noise along roads used to access the site. However, this increase in traffic noise would be small unless

the operation traffic volume were as large as the existing site traffic. Site-specific analysis would be conducted in tiered NEPA documentation if the reactor alternative were selected.

This option also involves using REDC for neptunium-237 target material storage, target fabrication, and processing. Interior modifications of these facilities in the 7900 Area of ORNL would be expected to result in little change in noise impacts on wildlife around this area. The operation of REDC would not be expected to result in any change in noise impacts on wildlife around the 7900 Area and offsite noise impacts would be small because the nearest site boundary is 2.5 kilometers (1.6 miles) to the southeast. Operation would be expected to result in a minimal change in noise impacts on people near ORR as a result of changes in employee and truck traffic levels.

4.6.1.2.3 Air Quality

The operation of a new research reactor and support facility would result in some increase in air quality impacts due to operation of emergency diesel generators. Criteria pollutants were modeled and compared to the most stringent standards (**Table 4–143**). The maximum ground-level pollutant concentrations that would result from reactor operation would be well below the ambient air quality standards. However, if the reactor were in an area that already had high background pollutant concentrations, resultant pollutant concentrations could approach or exceed the ambient standards for some pollutants. As a result, regulatory compliance would need to be assessed on case-by-case basis. Hazardous chemical impacts are addressed in Section 4.6.1.2.9.

Table 4–143 Incremental Concentrations Associated with Research Reactor Operation[a] Under Alternative 4 (Construct New Research Reactor)—Option 1

Criteria Pollutant	Averaging Period	Most Stringent Standard or Guideline (micrograms per cubic meter)[b]	Modeled Increment (micrograms per cubic meter)
Carbon monoxide	8 hours	10,000	89.5
	1 hour	40,000	128
Nitrogen oxide	Annual	100	0.198
PM_{10}	Annual	50	0.0035
	24 hours	150	3.46
Sulfur dioxide	Annual	80	0.062
	24 hours	365	61.2
	3 hours	1,300	138

a. From operation of two emergency diesel generators.

b. The more stringent of the Federal and state standards is presented if both exist for the averaging period. The National Ambient Air Quality Standards (NAAQS) (40 CFR Part 50), other than those based on annual averages, are not to be exceeded more than once per year. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

Source: Modeled increments are based on the SCREEN3 computer code (EPA 1995); data from Appendix E.

Air quality impacts associated with this option at ORR were determined to be the same as under Option 1 of Alternative 2 (Section 4.4.1.1.3).

The air quality impacts of transportation are presented in Section 4.6.1.2.11.

4.6.1.2.4 Water Resources

The estimated impacts on key water resource indicators associated with operating the new research reactor and support facility are presented in **Table 4–144.** Operation of the research reactor at a generic DOE site would have the highest water demand under this alternative and the second highest of any production facility

considered, requiring 807 million liters (213.2 million gallons) of water per year. In general, water would be required by the new research reactor and support facility to support such uses as process cooling, potable, and sanitary needs. For the new research reactor, the single largest system use would be for cooling tower operation and associated evaporative losses. The exact impact of these withdrawals on the resource would depend on the water source (surface water or groundwater) and its relative abundance. These factors would be used to determine the impact on the local and/or regional availability of the resource. For surface water, a dedicated surface water intake may have to be constructed if the generic site's existing distribution system is inadequate to meet the increased demands of the facilities. For groundwater, additional wells may have to be developed to supply the facilities directly or to provide increased production capacity for the generic site's existing supply system.

Table 4–144 Estimated Water Use and Wastewater Generation Associated with Operating a New Research Reactor and Support Facility Under Alternative 4 (Construct New Research Reactor)—Option 1

Indicator (million liters per year)	New Research Reactor[a]	New Support Facility[a]
Water use	807	6.92
Process wastewater generation	7.9	0.016[a]
Sanitary wastewater generation	11.6	6.91

a. Assume process wastewater generated at the same incremental rate as the Hanford 300 Area facilities (RPL/306–E).
Note: To convert from liters per year to gallons per year, multiply by 0.264.
Source: Appendix E; SAIC 2000.

The operation of the research reactor is estimated to generate approximately 7.9 million liters (2.1 million gallons) of process wastewater per year. It is expected that this process effluent would mainly consist of cooling tower blowdown. The support facility would generate a very small amount of process wastewater, mainly as a result of material processing. There would be no radiological liquid effluent discharge to the environment from either facility under normal operations. Sanitary wastewater would be generated as a result of operations of the new research reactor and support facility based on facility staff use of lavatory, shower, and kitchen facilities and from miscellaneous potable and sanitary uses. Waste management activities and their effects are further detailed in Section 4.6.1.2.13. The potential impact on water resources would depend on the availability and capacity of appropriate treatment facilities. Process and sanitary wastewater would be discharged to either existing site wastewater treatment facilities or to new facilities constructed specifically to serve the proposed facilities. All wastewater would be disposed of in accordance with applicable regulatory requirements with discharges to surface waters in accordance with NPDES effluent limitations.

Although specific impacts on water resources cannot be determined at this time, site-specific analysis would be conducted in tiered NEPA documentation if the research reactor alternative were selected.

REDC, an existing facility in the 7900 Area of ORNL at ORR, would be used for neptunium-237 storage, target fabrication, and processing in support of plutonium-238 production with impacts on ORR water resources indicators the same as those described in Section 4.3.1.1.4. In summary, a small increase in water use and sanitary wastewater generation is anticipated, mainly attributable to increased staffing levels. Also, there would be a very small increase in process wastewater generation, but there would be no radiological liquid effluent discharge to the environment under normal operations.

4.6.1.2.5 Geology and Soils

The operation of a research reactor and support facility would not be expected to result in impacts on geologic and soils resources at a generic DOE site. If cooling towers are used, the potential exists for salt deposition

to alter soil chemistry. While high rainfall at an eastern site would tend to keep salt from accumulating in the soil, the potential exists that salt could accumulate at a western site where rainfall is sparse. If the reactor alternative were selected, impacts on geology and soils would be determined in tiered NEPA documentation. As discussed in Section 4.6.1.1.5, the proposed facilities would be designed and constructed in accordance with DOE Order 420.1 and sited to minimize the risk from geologic hazards. Thus, site geologic conditions would be unlikely to affect the facilities.

The use of REDC for neptunium-237 storage, target fabrication, and processing would not be expected to impact either geologic or soil resources, nor be jeopardized by large-scale geologic conditions. Hazards from large-scale geologic conditions at ORR, such as earthquakes, volcanoes, and sinkholes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-260) as discussed in Section 4.2.2.2.5. The analysis determined that these hazards present a low risk to long-term storage facilities. Further review of the data and analyses presented in that document and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to REDC operations.

As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.6.1.2.6 Ecological Resources

If the new research reactor alternative were selected, tiered NEPA documentation would be undertaken to determine the exact nature of operational impacts on ecological resources. During this process, impacts on individual species and habitats that are sensitive to disturbance would be determined. This would include consideration of wetlands and threatened and endangered species.

While the exact nature of operational impacts on ecological resources cannot be determined until a specific site is selected, certain general types of impacts are possible. The nature and extent of these impacts would be expected to vary depending on whether the selected site was located in the eastern or western portion of the United States.

Terrestrial Resources. Activities associated with operations, such as noise and human presence, could affect wildlife living adjacent to the research reactor and support facility. These disturbances could cause some species to move from the area. Preventing workers from entering undisturbed areas would minimize impacts on wildlife living adjacent to the facilities. Emissions to the air and water, both nonradiological and radiological, could impact both plants and animals. Plants and animals could be exposed to pollutants via a number of pathways including direct exposure, contact with contaminated soil, ingestion, and inhalation. Further, bioaccumulation could affect species that consume exposed plants or animals. While regulatory limits would act to limit the effects of air emissions and effluent discharges, impacts would be analyzed once site and facility specific information became available.

Wetlands. Impacts from the operation of a research reactor and support facility at a western generic DOE site would not be expected to affect wetlands since discharges would be to an evaporation pond. At an eastern site, wastewater and cooling tower blowdown would be discharged to an onsite waterbody. While these discharges would be through permitted outfalls, the potential exists that wetlands could be affected. Potential impacts, such as changes in water levels and plant species composition, would depend on outfall location, water volume, discharge temperature, and water chemistry. Since these factors depend on site location and facility engineering design, operational impacts on site wetlands would have to be analyzed once these factors are known.

Aquatic Resources. Operational impacts on aquatic resources at a western site would not be expected since groundwater would be used and wastewater and cooling tower blowdown would be discharged to an evaporation pond. At an eastern site, potential impacts on aquatic resources could occur as a result of water withdrawal and discharge. Water withdrawal could lead to the loss of aquatic organisms through impingement and entrainment. The discharge of cooling water could result in alterations in aquatic communities. Alterations could include changes in aquatic vegetation and the loss of fish and benthic macroinvertebrates. Additionally, radionuclides and chemicals in the discharge water have the potential to impact aquatic organisms. The extent of potential impacts on the aquatic environment would depend upon site and facility specific information.

Threatened and Endangered Species. The operation of a research reactor and support facility would have the potential to impact threatened and endangered species. Sources of impacts would be similar to those discussed above for terrestrial resources, wetlands, and aquatic resources. The primary difference is that the resources of concern involve individual species that are sensitive to disturbance and whose existence may be threatened by development.

REDC would be used for neptunium-237 storage, target fabrication, and processing. As noted in Section 4.6.1.2.2, there would be little change in noise impacts on wildlife. There would be no change in impacts on wetlands or aquatic resources because additional water usage and wastewater discharge would be small fractions of current values. Further, this option would not result in any new contaminants in existing discharges (Section 4.6.1.2.4). Threatened and endangered species would not be affected by operation because an existing facility within an already developed area would be used.

Consultation to comply with Section 7 of the Endangered Species Act was conducted with the U.S. Fish and Wildlife Service (see Table 5–3) and resulted in the Service concluding that it does not anticipate adverse effects to federally listed endangered species that occur near the project area. DOE has also consulted with the Tennessee Department of Environment and Conservation; a response concerning state-listed species is pending from this agency. Although no state-listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the state.

4.6.1.2.7 Cultural and Paleontological Resources

Impacts on cultural and palenontological resources from the operation of a research reactor and support facility at a generic DOE site would depend on the relative location of such resources to the site and/or transportation routes. While impacts would be expected to be nonexistent or small, they cannot be ruled out. For example, noise related to plant operation or traffic to and from the facility or alterations in the viewshed could adversely affect visitor enjoyment of an historic site. Since impacts on cultural resources are site dependent, specific operational impacts cannot be determined until a site were selected. The operation of a reactor and support facility would not be expected to impact paleontological resources.

The operation of REDC for neptunium-237 storage, target fabrication, and processing would not change the status of cultural and paleontological resources at ORR. The Graphite Reactor, which is located within ORNL, is listed on the National Register of Historic Places as a National Historic Landmark. Additionally, several other structures proposed for listing on the National Register of Historic Places are found within or near ORNL. However, neither the Graphite Reactor nor any of the other structures is located within the 7900 Area; thus, the use of REDC for target fabrication and processing would not change their status.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). While DOE has made additional contact with the State Historic Preservation Office, a response is pending from this office. Although impacts to cultural resources are not

expected as a result of the proposed action, no action would be taken relative to the use of facilities at ORR prior to the receipt of input from the State Historic Preservation Office.

4.6.1.2.8 Socioeconomics

It is estimated that 220 workers would be needed each year to operate the research reactor and support facility at a generic DOE site. The impact of this influx of workers upon the site's region of influence and regional economic area would depend on whether the site were located near a large urbanized area or in a remote rural area. Since the population for the region of influence for a generic site could range from nearly 2.0 million people for a site in a large metropolitan area, to less than 200,000 for a site in a small rural community, the socioeconomic impacts of operating a new research reactor and support facility would vary greatly. Therefore, if DOE were to select this option, additional NEPA documentation would be required to determine the specific socioeconomic impacts.

The socioeconomic impacts associated with neptunium-237 target fabrication and processing at ORR are addressed in Section 4.3.1.1.8.

4.6.1.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup and operations are given in **Table 4–145** for the generic DOE site and ORR: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. Radiological impacts from startup operations prior to fuel loading would be zero. After fuel loading, these impacts would be expected to be bounded by normal operation impacts. Therefore, startup impacts have not been treated separately from normal operational impacts. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual operations of the research reactor facilities and REDC, the projected incremental total population dose in the year 2020 would be 0.14 person-rem; the corresponding number of latent cancer fatalities in the populations surrounding the generic DOE site and ORR from 35 years of operations would be 0.0025. The incremental total dose to the maximally exposed member of the public from annual operations of the research reactor and support facility at the generic DOE site would be 0.0026 millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 4.5×10^{-8}.

Table 4–145 Incremental Radiological Impacts on the Public Around the Generic DOE Site and ORR from Operational Facilities Under Alternative 4 (Construct New Research Reactor)—Option 1

Receptor	ORR REDC	Generic Site			Two-Site Total
		Research Reactor Operations	Reactor Support Facility Operations	Total	
Population within 80 kilometers (50 miles) in the year 2020					
Dose (person-rem)	8.8×10^{-5}	0.0023	0.14	0.14	0.14
35-year latent cancer fatalities	1.5×10^{-6}	4.0×10^{-5}	0.0025	0.0025	0.0025
Maximally exposed individual					
Annual dose (millirem)	1.9×10^{-6}	6.8×10^{-5}	0.0025	0.0026	NA[a]
35-year latent cancer fatality risk	3.3×10^{-11}	1.2×10^{-9}	4.4×10^{-8}	4.5×10^{-8}	NA[a]
Average exposed individual within 80 kilometers (50 miles)					
Annual dose[b] (millirem)	7.8×10^{-8}	1.5×10^{-6}	9.1×10^{-5}	9.3×10^{-5}	NA[a]
35-year latent cancer fatality risk	1.4×10^{-12}	2.6×10^{-11}	1.6×10^{-9}	1.6×10^{-9}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of REDC or the generic site in the year 2020 (1,134,200 and 1,538,100, respectively).

Key: NA, not applicable.
Source: Appendix E; model results, using the GENII computer code (Napier et al. 1988).

Incremental doses to involved workers from normal operations are given in **Table 4–146**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to research reactor workers during startup and operations would be 100 millirem; for support facility workers, the incremental annual average dose during startup and operations would be 114 millirem; for REDC workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities is estimated to be 12, 11, and 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–146. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–146 Incremental Radiological Impacts on Involved Workers at the Generic DOE Site and ORR from Operational Facilities Under Alternative 4 (Construct New Research Reactor)—Option 1

Receptor—Involved Workers[a]	ORR REDC	Generic Site		Two-Site Total
		Research Reactor Operations	Reactor Support Facility Operations	
Total dose (person-rem per year)	12[b]	12[b]	11[b]	36
35-year latent cancer fatalities	0.17	0.17	0.16	0.50
Average worker dose (millirem per year)	170	100	114	NA[c]
35-year latent cancer fatality risk	0.0023	0.0014	0.0016	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that each facility adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers at REDC, 120 research reactor workers, and 100 workers at the reactor support facility.
c. Values cannot be given for the average worker because the workers would be in three different facilities at two different sites.

Key: NA, not applicable.
Source: Nielsen 1999; Wham 1999b, 2000.

HAZARDOUS CHEMICALS IMPACTS. The operation of a new reactor and associated support facility would result in some increase in emissions of hazardous chemicals due to diesel fuel burning from different sources of equipment used for operation. The operation of the reactor would require the emergency diesel generators to be tested approximately 1 hour each month and 24 hours once a year to ensure operability. Chemical releases were modeled based on 72 hours of operation. Resulting concentrations were determined to be very small and would have no incremental impact on the site current conditions (**Table 4–147**).

Table 4–147 Incremental Hazardous Chemical Impacts from New Research Reactor Diesel Generator Operation Under Alternative 4 (Construct New Research Reactor)—Option 1

Chemicals	Modeled Annual Increment (micrograms per cubic meter)	RfC (micrograms per cubic meter)	Unit Cancer Risk (risk per micrograms per cubic meter)	Hazard Quotient	Cancer Risk
Benzene	4.83×10^{-5}	NA	0.0000078	NA	3.77×10^{-10}
Naphthalene	6.83×10^{-6}	3	NA	2.28×10^{-6}	NA
Toluene	1.75×10^{-5}	400	NA	4.38×10^{-8}	NA
Propylene	1.73×10^{-4}	NA	0.0000037	NA	6.42×10^{-10}

Note: Propylene oxide cancer unit was used for propylene.
Key: NA, not applicable (the chemical is not a known carcinogen, or it is a carcinogen and only unit cancer would apply); RfC, Reference Concentration.
Source: Data from Appendix E; EPA 1999; modeled increments are based on the SCREEN3 computer code (EPA 1995).

Hazardous chemicals impacts for this option at ORR were determined to be the same as described in Alternative 2, Option 1 (Section 4.4.1.1.9).

4.6.1.2.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with research reactor target irradiation; support facility medical, industrial, and research and development isotope fabrication and processing; and REDC neptunium-237 target fabrication and processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–148** and **4–149**, respectively.

For 35 years of research reactor target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.49×10^{-9} and 8.41×10^{-9}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 2.26×10^{-5}.

For 35 years of support facility medical, industrial, and research and development target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 3.26×10^{-5} and 9.85×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.056.

For 35 years of REDC neptunium-237 target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 5.71×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.157.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 8.98×10^{-5} and 4.49×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.213.

Table 4–148 New Research Reactor, Support Facility, and REDC Accident Consequences Under Alternative 4 (Construct New Research Reactor)—Option 1

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
New research reactor accidents						
Design-basis accident	1.33×10^{-6}	6.65×10^{-10}	0.00241	1.20×10^{-6}	5.49×10^{-6}	2.20×10^{-9}
Beyond-design-basis earthquake	0.00373	1.87×10^{-6}	27.6	0.0138	0.0531	2.12×10^{-5}
Fuel-handling accident	1.90×10^{-9}	9.50×10^{-13}	6.79×10^{-6}	3.40×10^{-9}	5.83×10^{-9}	2.33×10^{-12}
Neptunium-237 target-handling accident	5.42×10^{-8}	2.71×10^{-11}	8.95×10^{-5}	4.47×10^{-8}	2.43×10^{-7}	9.72×10^{-11}
Medical isotope target-handling accident	1.04×10^{-5}	5.20×10^{-9}	0.101	5.06×10^{-5}	6.76×10^{-6}	2.70×10^{-9}
Support facility accidents						
Medical and industrial isotopes localized solvent fire	0.0194	9.72×10^{-6}	31.1	0.0156	0.00530	2.12×10^{-6}
Medical and industrial isotopes unlikely seismic event	0.0750	3.75×10^{-5}	136	0.0680	0.510	2.04×10^{-4}
Medical and industrial isotopes glovebox explosion	2.50	0.00125	4,600	2.30	17.0	0.00680
REDC accidents						
Ion exchange explosion during neptunium-237 target fabrication	6.13×10^{-9}	3.06×10^{-12}	8.58×10^{-5}	4.29×10^{-8}	5.60×10^{-10}	2.24×10^{-13}
Target dissolver tank failure during plutonium-238 separation	1.76×10^{-7}	8.79×10^{-11}	0.00196	9.82×10^{-7}	1.69×10^{-8}	6.74×10^{-12}
Ion exchange explosion during plutonium-238 separation	4.68×10^{-4}	2.34×10^{-7}	5.23	0.00261	4.49×10^{-5}	1.79×10^{-8}
Plutonium-238 processing facility beyond-design-basis earthquake	163	0.163	8.91×10^{5}	445	1,310	1.00[c]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–149 New Research Reactor, Support Facility, and REDC Accident Risks Under Alternative 4 (Construct New Research Reactor)—Option 1

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual new research reactor risks			
Design-basis accident (1×10^{-4})	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis earthquake (1×10^{-5})	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident (0.01)	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident (0.01)	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical isotope target-handling accident (0.01)	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
35-year new research reactor risk	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Annual support facility risks			
Medical and industrial isotopes localized solvent fire (0.044)	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical and industrial isotopes glovebox explosion (0.01)	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
35-year support facility risk	3.26×10^{-5}	0.056	9.85×10^{-5}
Annual REDC risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	1.63×10^{-6}	0.00445	$1.00\times10^{-5(c)}$
35-year REDC risk	5.71×10^{-5}	0.157	$3.50\times10^{-4(c)}$
35-year Option risk	8.98×10^{-5}	0.213	4.49×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

There are no hazardous chemical accidents associated with the new research reactor or new support facility. The irradiation of neptunium-237, medical, industrial, and research and development isotopes in the new research reactor would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The fabrication and processing of medical, industrial, and research and development isotopes at the new support facility would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The hazardous chemical accident impacts at REDC are the same as those presented in Section 4.4.4.1.10.

4.6.1.2.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the REDC target fabrication facility at ORR. DOE would transport the unirradiated neptunium-237 targets from REDC to the reactor site. Following irradiation in the reactor, the targets would be returned to REDC for processing. After this processing, the plutonium-238 product would be shipped to LANL. The reactor would receive low enriched uranium fuel from a U.S. fuel fabrication facility. Additionally, medical and industrial isotopes would be shipped from the reactor site to a local airport, and from there to locations throughout the country.

Approximately 37,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 7.5 million kilometers (4.7 million miles); and in the air carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 28.6 person-rem; the dose to the public, 308 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.011 latent cancer fatality among transportation workers and 0.15 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.026. About half of the crew risk, about 2 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.19 traffic fatality. Nearly all of the radiological and about 59 percent of the traffic accident risk would result from shipping medical and industrial isotopes.

4.6.1.2.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in a new reactor that would be constructed at a site yet to be specified. Fabrication and processing of neptunium-237 targets for plutonium-238 production would be performed at REDC located at ORR. A new support facility would be constructed at an unspecified site for fabrication and processing targets not used for plutonium-238 production.

Activities at REDC were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.1.1.12) and found to pose no significant radiological or other risks to minority and low-income populations. The environmental analysis of operations at the new research reactor and support facility site shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected

area would not be significant. Unless there are patterns of food consumption among minority or low-income resident surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that operations at the site would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site-specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new research reactor and support facility, an additional evaluation of environmental justice at the reactor and support facility site during operation would be performed prior to implementation.

4.6.1.2.13 Waste Management

The expected annual generation of waste that would be generated from the operation of a new research reactor to irradiate targets and a support facility to fabricate and process medical and industrial isotope targets and to meet research and development needs are provided in **Table 4–150**. These generation rates cannot be compared at this time with site treatment, storage, and disposal capacities because a DOE site has not yet been chosen for these facilities. Section 3.6.11.1 provides DOE site ranges for each waste type that include volume currently stored, projected generation, and for some types of waste, disposal volume. Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.6.1.2.9 through 4.6.1.2.11.

Table 4–150 Estimated Waste Generation Rates of Operating a New Research Reactor and Support Facility Under Alternative 4 (Construct New Research Reactor)—Option 1

Waste Type[a]	Estimated Waste Generation for New Research Reactor (cubic meters per year)	Estimated Waste Generation for New Support Facility (cubic meters per year)
High-level radioactive	0	0
Transuranic	0	0
Low-level radioactive		
Liquid	<6	0
Solid	50	20
Mixed low-level radioactive	<0.5	4
Hazardous	4	<1
Nonhazardous		
Process wastewater	7,950	16[b]
Sanitary wastewater	11,600	6,900
Solid	250	80

a. See definitions in Section G.9.
b. Assumes process wastewater generated at the same incremental rate as Hanford 300 facilities.
Note: To convert from cubic meters per year to cubic yards per year, multiply by 1.308; < means "less than."
Source: Appendix E; SAIC 2000.

Depending in part on decisions in the Records of Decision for the *Waste Management PEIS* (DOE 1997a), waste could be treated and disposed of on site or at other DOE sites or commercial facilities. No high-level radioactive waste or transuranic waste would be generated from irradiating targets in the new research reactor or from target fabrication or processing in the new support facility.

Currently, DOE sites that manage low-level radioactive waste treat and/or dispose of the waste on site or off site, either at another DOE facility or a commercial facility. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision issued on February 18, 2000 (65 FR 10061), states that for the management of low-level radioactive waste, minimal treatment will be performed at all sites, and disposal will

continue, to the extent practicable, on site at INEEL, LANL, ORR, and SRS. In addition, Hanford and the Nevada Test Site will be available to all DOE sites for low-level radioactive waste disposal. Less than 210 cubic meters (275 cubic yards) of liquid low-level radioactive waste and 1,750 cubic meters (2,300 cubic yards) of solid low-level radioactive waste would be generated over a 35-year period as a result of target irradiation at the new research reactor. Target fabrication and processing at the new support facility would generate about 700 cubic meters (920 cubic yards) of solid low-level radioactive waste. The minor amounts of low-level radioactive waste (less than 10 cubic meters [13.1 cubic yards]) (Brunson 1999a) generated from the decontamination of the shipping containers used to transport neptunium-237 from SRS to REDC (or FDPF or FMEF, depending on the option) for storage could easily be managed under the existing waste management practices and are not included in the table.

Most of DOE's mixed low-level radioactive waste is being stored on site awaiting the development of treatment methods. DOE is subject to the requirements mandated by the Federal Facility Compliance Act of 1992, and most DOE facilities that currently store or generate mixed low-level radioactive waste have either a state-approved or EPA region-approved site treatment plan or another type of agreement. Each site treatment plan or agreement requires the treatment of mixed waste, including mixed low-level radioactive waste, in accordance with its provisions. The low-level radioactive waste and mixed low-level radioactive waste Record of Decision, issued on February 18, 2000 (65 FR 10061), states that mixed low-level radioactive waste will be treated at Hanford, INEEL, ORR, and SRS and disposed of at Hanford and the Nevada Test Site. Over the 35-year operational period, less than 18 cubic meters (24 cubic yards) of mixed low-level radioactive waste would be generated as a result of target irradiation at the new research reactor. Target fabrication and processing at the new support facility would generate about 140 cubic meters (180 cubic yards) of mixed low-level radioactive waste.

The hazardous waste Record of Decision, issued on August 5, 1998 (63 FR 41810), states that most DOE sites will continue to use offsite facilities for the treatment and disposal of major portions of nonwastewater hazardous waste, with ORR and SRS continuing to treat some of their own nonwastewater hazardous waste on site in existing facilities where this is economically favorable. Wastewater, which is about 99 percent of DOE's hazardous waste, is treated on site. An estimated 140 cubic meters (180 cubic yards) of hazardous waste would be generated during the 35-year operational period at the research reactor and less than 35 cubic meters (46 cubic yards) at the new support facility.

DOE currently manages sanitary and industrial waste on a site-by-site basis. Some DOE sites dispose of this waste in onsite landfills that have permits issued by appropriate state agencies, while other sites use commercial landfills (DOE 1997a:1-29). Solid waste such as office paper, metal cans, and plastic and glass bottles that can be recycled would be sent off site for that purpose. Over the 35-year operational period, an estimated 280,000 cubic meters (370,000 cubic yards) of process wastewater, 406,000 cubic meters (531,000 cubic yards) of sanitary wastewater, and 8,800 cubic meters (12,000 cubic yards) of solid nonhazardous waste would be generated as a result of target irradiation at the new research reactor. Target fabrication and processing at the new support facility would generate about 560 cubic meters (730 cubic yards) of process wastewater, 241,500 cubic meters (316,000 cubic yards) of sanitary wastewater, and 2,800 cubic meters (3,700 cubic yards) of solid nonhazardous waste.

The impacts of managing waste associated with fabricating and processing neptunium-237 targets for plutonium-238 production in REDC at ORR are assumed to be the same as for Option 1 under Alternative 1 (Section 4.3.1.1.13). As shown in that section, the impacts on the waste management systems at ORR would be minimal.

4.6.1.2.14 Spent Nuclear Fuel Management

The operation of the proposed new research reactor would generate about 0.31 metric ton heavy metal (682 pounds) of spent nuclear fuel per year, a total of about 11 metric tons heavy metal (24,200 pounds) from 35 years of operation. This spent nuclear fuel would be stored at the spent nuclear fuel pool at the reactor site. The spent nuclear fuel pool would be designed to provide enough capacity for 35 years of operation, that is, have a total capacity of about 11 metric tons (24,200 pounds). No dry fuel storage is anticipated at the site; therefore there are no environmental impacts associated with the construction of a dry fuel storage facility. The environmental impacts associated with the normal operation of the proposed new research reactor (which includes spent nuclear fuel storage) would result in an annual dose to the maximally exposed individual member of the public of 6.8×10^{-5} millirem from total site operations. This dose is well below the EPA's Clean Air Act standard of 10 millirem per year that is cited in DOE Order 5400.5. The environmental impacts associated with spent nuclear fuel management would be small.

4.6.1.3 Decontamination and Decommissioning of the Research Reactor and Support Facility

The environmental impacts associated with the decontamination and decommissioning of the research reactor and support facility at the generic DOE site are assessed in this section. If the research reactor were built on a site with existing support facilities, there would be no impacts associated with decommissioning a "new" support facility.

4.6.1.3.1 Land Resources

LAND USE. Decontamination and decommissioning of a research reactor and support facility would not involve the removal of any major structures, although some smaller facilities and pieces of equipment could be removed. Thus, the industrial nature of the land would not change.

VISUAL RESOURCES. Decontamination and decommissioning of a research reactor and support facility would not impact visual resources since no major structures would be removed. Thus, the Visual Resource Management Class IV rating of the site would remain unchanged.

4.6.1.3.2 Noise

Decontamination and decommissioning of a reactor and support facility would result in some increase in noise levels from the use of construction type equipment, materials handling and impact equipment, employee vehicles, and truck traffic. Actual noise levels would depend on the decontamination and decommissioning activities selected. Noise from these activities, especially impulsive noise, would be expected to disturb wildlife in the immediate area of the facilities. The change in noise levels in areas outside the DOE site would depend on the location selected and the exact nature of the activities required. However, generally if the reactor and support facility location were within one of the larger DOE sites and were more centrally located within the site, offsite noise impacts from decontamination and decommissioning activities would be expected to be small. Employee vehicles and truck traffic would result in an increase in traffic noise along roads used to access the site. However, this increase in traffic noise would be small unless the decontamination and decommissioning traffic volume were as large as the traffic from facility operation and other site activities. Site-specific analysis would be conducted in tiered NEPA documentation if the reactor alternative were selected.

4.6.1.3.3 Air Quality

Deactivation and decommissioning of the new reactor and support facility would result in some change in the air quality impacts. However, they would not be expected to be higher than the impacts associated with construction and operation.

4.6.1.3.4 Water Resources

Decontamination and decommissioning of the research reactor and support facility would involve permanent shutdown, stabilization, and monitoring of the deactivated facilities. As a result, processing and auxiliary systems would be shutdown and process and sanitary wastewater discharges would cease from the vacated facilities. This would eliminate the annual discharge of approximately 7.9 million liters (2.1 million gallons) of nonradioactive process wastewater from the research reactor and 0.016 million liters (0.004 million gallons) from the support facility on an annual basis. The discharge of 11.6 million liters (3.07 million gallons) per year of sanitary wastewater from the research reactor and 6.91 million liters (1.82 million gallons) from the support facility would be eliminated to onsite treatment facilities. The effects of decontamination and decommissioning on waste management are further detailed in Section 4.6.1.3.13. Site water withdrawals to supply the facilities would also be reduced by approximately 807 million liters (213.2 million gallons) per year for the research reactor and 6.92 million liter (1.83 million gallons) annually for the support facility (Appendix E; SAIC 2000).

4.6.1.3.5 Geology and Soils

No major structures would be demolished to effect decontamination and decommissioning of the research reactor and support facility. Some ground disturbance could occur associated with removal of some smaller facilities and pieces of equipment. However, ground disturbance would be confined to previously disturbed areas immediately adjacent to the reactor complex and support facility, with the impact on geologic and soil resources expected to be negligible overall.

4.6.1.3.6 Ecological Resources

Since no major structures would be demolished during decontamination and decommissioning of a research reactor and support facility, the area would continue to be of limited value to wildlife. Noise from decontamination and decommissioning activities would be expected to disturb wildlife in the immediate area; however, this disturbance would be of limited duration. Water use would decrease at the generic site with the decommissioning of a research reactor and support facility. This would result in a decrease in impingement and entrainment of aquatic organisms, as well as a decrease in impacts from effluent discharge at a site where surface water bodies are used. At a site where water is withdrawn from groundwater and discharged to an evaporation pond, the cessation of discharge from a reactor and support facility could result in a reduction in the size of the pond or its possible elimination. This could, in turn, result in the loss (or elimination) of associated aquatic and terrestrial wildlife, as well as wetland habitat. The response of any threatened or endangered species to decontamination and decommissioning of a reactor and support facility could vary from positive (e.g., due to a decrease in human presence and emissions) to negative (e.g., due to the elimination of aquatic or wetland habitat), depending on the species involved.

4.6.1.3.7 Cultural and Paleontological Resources

Decontamination and decommissioning of a research reactor and support facility would not change the status of cultural and paleontological resources. Any required ground disturbance would be confined to previously disturbed areas immediately adjacent to the reactor and support facility.

4.6.1.3.8 Socioeconomics

Decommissioning of the research reactor and support facility would result in a negative impact on the socioeconomic characteristics of the DOE site at which they were located. This impact would depend on whether the candidate site was located near a large urbanized area or in a remote rural area. Since the population for the region of influence for a generic DOE site could range from nearly 2.0 million people for a site in a large metropolitan area, to less than 200,000 for a site in a small rural community, the socioeconomic impacts of decommissioning would vary greatly. If DOE were to select the new research reactor alternative, additional NEPA documentation would be required to evaluate the specific socioeconomic impacts of the decommissioning.

4.6.1.3.9 Public and Occupational Health and Safety—Normal Decontamination and Decommissioning Activities

Assessments of incremental radiological and chemical impacts associated with the decontamination and decommissioning of the research reactor and support facility are presented in this section. Supplemental information is provided in Appendix H.

During decontamination and decommissioning operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers are described below.

RADIOLOGICAL IMPACTS. In the *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities*, NUREG-0586, (NRC 1988), NRC determined that the health impact to the public from the decommissioning of research reactors was "negligible." This statement was based on the analysis of a 60-megawatt thermal light water pool reactor fueled with TRIGA fuel. The generic reactor facility used in the analysis of generic research reactor environmental impacts is a 50-megawatt thermal light water pool reactor fueled with TRIGA fuel. In the same NUREG, NRC also concluded that the public health impact from radiological releases associated with the decommissioning and decontamination of process facilities similar to the generic research reactor support facility was also "negligible." Based on these NRC conclusions, the environmental impact on the public health and safety from the routine release of radionuclides during the decontamination and decommissioning of the generic research reactor and its support facility addressed in this NI PEIS are deemed to be negligible.

Incremental doses to involved workers from decontamination and decommissioning operations are given in **Table 4–151**; these workers are defined as those directly associated with all decontamination and decommissioning activities. The incremental annual average dose to involved workers during decontamination and decommissioning operations at the research reactor would be 275 millirem; for support facility workers, the incremental annual average dose during decontamination and decommissioning operations would be 25 millirem. The incremental annual dose received by the total site workforce for each of these facilities is estimated to be 11 and 1 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from annual decontamination and decommissioning operations are included in Table 4–151; a probability coefficient of 4×10^{-4} latent cancer fatality per rem was applied for workers (ICRP 1991). Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–151 Incremental Radiological Impacts on Involved Workers at the Generic DOE Site from the Research Reactor and Support Facility Decontamination and Decommissioning Activities Under All Options of Alternative 4 (Construct New Research Reactor)

Receptor—Involved Workers[a]	Generic Site Decontamination and Decommissioning Activities		
	Research Reactor	Research Reactor Support Facility	Total
Total dose (person-rem per year)	11[b]	1[b]	12
4-year latent cancer fatalities	0.018	0.0016	0.019
Average worker dose (millirem per year)	275	25	150
4-year latent cancer fatality risk	4.4×10^{-4}	4.0×10^{-5}	2.4×10^{-4}

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that each facility adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 40 badged workers.
Source: NRC 1988.

HAZARDOUS CHEMICAL IMPACTS. Limited impacts would result from hazardous chemicals associated with deactivation and decommissioning activities.

4.6.1.3.10 Public and Occupational Health and Safety—Decontamination and Decommissioning Accidents

Impacts from postulated accidents associated with the decontamination and decommissioning of the research reactor and support facility are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–152** and **4–153**, respectively.

Table 4–152 Research Reactor and Support Facility Decontamination and Decommissioning Accident Consequences Under All Options of Alternative 4 (Construct New Research Reactor)

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
Spent nuclear fuel cask drop	7.01×10^{-12}	3.51×10^{-15}	2.78×10^{-8}	1.39×10^{-11}	1.30×10^{-11}	5.20×10^{-15}
Reactor core tank vaporization	1.55×10^{-5}	7.75×10^{-9}	0.346	1.73×10^{-4}	5.23×10^{-5}	2.09×10^{-8}

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
Source: Model results using MACCS2 (Chanin and Young 1997).

Table 4–153 Research Reactor and Support Facility Decontamination and Decommissioning Accident Risks Under All Options of Alternative 4 (Construct New Research Reactor)

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Spent nuclear fuel cask drop (5.0×10^{-6})	1.75×10^{-20}	6.95×10^{-17}	2.60×10^{-20}
Reactor core tank vaporization (1.0×10^{-4})	7.75×10^{-13}	1.73×10^{-8}	2.09×10^{-12}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
Source: Model results using MACCS2 (Chanin and Young 1997).

For a spent nuclear fuel cask drop, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 1.75×10^{-20} and 2.60×10^{-20}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 6.95×10^{-17}.

For a reactor core tank vaporization accident, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 7.75×10^{-13} and 2.09×10^{-12}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 1.73×10^{-8}.

There are no hazardous chemical accidents postulated during the decontamination and decommissioning phases of the new research reactor or the new support facility. Involved workers could experience industrial accidents commonly associated with these types of activities.

4.6.1.3.11 Environmental Justice

Environmental effects due to decontamination and decommissioning activities that would be expected to occur at the unspecified reactor and support facility site are addressed in Section 4.6.1.3. The environmental analysis of decontamination and decommissioning activities at the new reactor and support facility site shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected area are not significant. Unless there are patterns of food consumption among minority or low-income residents surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that decontamination and decommissioning activities at the site would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site-specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new research reactor and support facility, an additional evaluation of environmental justice at the reactor and support facility site during decontamination and decommissioning would be performed prior to implementation.

4.6.1.3.12 Waste Management

The decontamination and decommissioning of the new research reactor and support facility would generate numerous types of waste. The materials that may be removed or stabilized as a result of decontamination and decommissioning would be managed and reused, recycled, or disposed of in accordance with applicable Federal and state regulations. No analysis of waste management impacts, however, can be formulated at this time. Once proposals concerning decontamination and decommissioning activities are developed, DOE will undertake any additional NEPA analysis that may be necessary or appropriate.

4.6.1.3.13 Spent Nuclear Fuel Management

The Nuclear Waste Policy Act of 1982, as amended, assigned to the Secretary of Energy the responsibility for the development of a geologic repository for the ultimate disposal of high-level radioactive waste and spent nuclear fuel. When such a repository is available, spent nuclear fuel would be transferred from nuclear reactor sites to the repository for disposal. Until a repository becomes available, spent nuclear fuel associated with the operation of the research reactor would be stored in the reactor pool. Upon cessation of research reactor operations, the reactor would be decontaminated and decommissioned. At that time, spent nuclear fuel stored in the pool would be packaged in acceptable containers and shipped to the geologic repository for disposal.

4.6.1.4 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.6.2 Alternative 4 (Construct New Research Reactor)—Option 2

Option 2 involves constructing and operating the research reactor to irradiate all targets associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating FDPF at INEEL to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and constructing and operating the support facility to fabricate and process the other targets and materials and to process the associated products. This option includes storage in Building CPP–651 or FDPF of the neptunium-237 transported to INEEL from SRS and storage in the new support facility of the other target materials transported to the generic site from other offsite facilities.

The transportation of the low enriched uranium fuel for use in the research reactor, the transportation of the neptunium-237 to INEEL and then to the generic site, the transportation of the other target materials to the generic site, and the transportation of all product materials following irradiation and postirradiation processing are also part of this option.

All options under this alternative include the decontamination and decommissioning of the research reactor and support facility at the generic site following their operating lifetimes, and also the permanent deactivation of FFTF at Hanford.

4.6.2.1 Construction of the New Research Reactor and Support Facility

The environmental impacts associated with the construction of a new research reactor and support facility at the generic DOE site are assessed in Section 4.6.1.1.

4.6.2.2 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.6.2.2.1 Land Resources

LAND USE. Impacts on land use associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.1.

Building CPP–651 and/or FDPF, which are located at INTEC, would be used for neptunium-237 storage, and FDPF would be used for target fabrication and processing. Use of these facilities would not change land use

at the site since both are currently operating and their proposed use would be compatible with their present mission.

VISUAL RESOURCES. Impacts on visual resources associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.1.

All activities associated with neptunium-237 storage would take place within Building CPP–651 and/or FDPF, and target fabrication and processing in FDPF. Operations associated with the proposed activities would not result in any impact on visual resources or change in the current Visual Resource Management Class IV rating of INTEC. This is because none of the anticipated operational impacts (e.g., air emissions) would be expected to affect this resource.

4.6.2.2.2 Noise

Noise impacts associated with operation of a research reactor and support facility are addressed in Section 4.6.1.2.2.

This option also involves using the Building CPP–651 and/or FDPF, both in the INTEC area of INEEL, for neptunium-237 target material storage, and FDPF for target fabrication and processing. Interior modifications of these facilities would be expected to result in little change in noise impacts on wildlife around this area. The operation of this facility would not be expected to result in any change in noise impacts on wildlife around the INTEC area and offsite noise impacts would be small because the nearest site boundary is 12 kilometers (7.5 miles) to the south. Operation would be expected to result in minimal change in noise impacts on people near the INEEL as a result of changes in employee and truck traffic levels.

4.6.2.2.3 Air Quality

Air quality impacts associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.3.

Impacts associated with this option at INEEL were determined to be the same as under Option 2 of Alternative 2 (Section 4.4.2.1.3).

The air quality impacts of transportation are presented in Section 4.6.2.2.11.

4.6.2.2.4 Water Resources

Impacts on water resources associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.4.

Building CPP–651 and/or FDPF, existing facilities in the INTEC area of INEEL, would be used for neptunium-237 storage; FDPF would also be used for the fabrication and processing of targets in support of plutonium-238 production. Impacts on water resources indicators at INEEL would be the same as those described in Section 4.3.2.1.4. In summary, a small increase in water use and sanitary wastewater generation would be anticipated, mainly attributable to increased staffing levels. Also, there would be a very small increase in process wastewater generation, but there would be no radiological liquid effluent discharge to the environment under normal operations.

4.6.2.2.5 Geology and Soils

Impacts on geology and soils associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.5. As discussed in Section 4.6.1.1.5, the proposed facilities would be designed and constructed in accordance with DOE Order 420.1 and sited to minimize the risk from geologic hazards. Thus, site geologic conditions would be unlikely to affect the facilities.

The use of Building CPP–651 and/or FDPF for neptunium-237 storage, and FDPF for target fabrication and processing would not be expected to impact geologic resources, nor be jeopardized by large-scale geologic conditions. Hazards from large-scale geologic conditions at INEEL, such as earthquakes and volcanoes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-148) as discussed in Section 4.2.3.2.5. The analysis determined that these hazards present a low risk to long-term storage facilities. That analysis was reviewed in the *Surplus Plutonium Disposition EIS* (DOE 1999a:4-267-268). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to the proposed use of the INTEC facilities. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.6.2.2.6 Ecological Resources

Impacts on ecological resources associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.6.

Building CPP–651 and/or FDPF would be used for neptunium-237 storage, and FDPF for target fabrication and processing. As noted in Section 4.6.2.2.2, there would be no change in noise impacts on wildlife. Because additional water usage and wastewater discharge would be small fractions of current values, there would be no impact on aquatic resources (Section 4.6.2.2.4). Threatened and endangered species would not be affected by operation because an existing facility(s) within an already developed area would be used.

Consultation letters to comply with Section 7 of the Endangered Species Act were sent to the U.S. Fish and Wildlife Service and the Idaho Department of Fish and Game (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. The Idaho Department of Fish and Game indicated that its database contained no known occurrences of special status plants or animals near the project area. While DOE has made additional contact with the U.S. Fish and Wildlife Service, a response is pending from this agency. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at INEEL prior to the receipt of input from the Service.

4.6.2.2.7 Cultural and Paleontological Resources

Impacts on cultural and paleontological resources associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.7.

Although six historic structures are associated with INTEC, their status would not be affected by operation of Building CPP–651 and/or FDPF for neptunium-237 storage, and FDPF for target fabrication and processing. Also, the status of Native American and paleontological resources occurring in the vicinity of INTEC would not be affected by the operation of these facilities.

Consultation to comply with Section 106 of the National Historic Preservation Act was initiated with the State Historic Preservation Office (see Table 5–3). The State Historic Preservation Office indicated that Building CPP–651 and FDPF are likely to be eligible for the National Register of Historic Places as contributory properties in a potential historic district of exceptional significance. However, at this time, the State Historic Preservation Office has determined that more information is needed prior to assisting DOE in evaluating these properties. The State Historic Preservation Office also indicated that since there would be no new construction, there is little potential for effects on archaeological properties. DOE would provide additional information as required to the Idaho State Historic Preservation Office prior to the use of any facility at INEEL for the proposed project. Consultation was conducted with interested Native American tribes; however, responses are pending.

4.6.2.2.8 Socioeconomics

The socioeconomic impacts associated with the operation of the new research reactor and support facility at a generic DOE site are addressed in Section 4.6.1.2.8.

The socioeconomic impacts associated with neptunium-237 target fabrication and processing at INEEL are addressed in Section 4.3.2.1.8.

4.6.2.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup and operations are given in **Table 4–154** for the generic DOE site and INEEL: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public.

Radiological impacts from startup operations prior to fuel loading would be zero. After fuel loading, these impacts would be expected to be bounded by the normal operation impacts. Therefore, startup impacts have not been treated separately from normal operational impacts. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual operations of the research reactor facilities and FDPF, the projected incremental total population dose in the year 2020 would be 0.14 person-rem; the corresponding number of latent cancer fatalities in the populations surrounding the generic DOE site and INEEL from 35 years of operations would be 0.0025. The incremental total dose to the maximally exposed member of the public from annual operations of the research reactor and support facility at the generic DOE site would be 0.0026 millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 4.5×10^{-8}.

Table 4–154 Incremental Radiological Impacts on the Public Around the Generic DOE Site and INEEL from Operational Facilities Under Alternative 4 (Construct New Research Reactor)—Option 2

Receptor	INEEL FDPF	Generic Site			Two-Site Total
		Research Reactor Operations	Reactor Support Facility Operations	Total	
Population within 80 kilometers (50 miles) in the year 2020					
Dose (person-rem)	3.9×10^{-6}	0.0023	0.14	0.14	0.14
35-year latent cancer fatalities	6.7×10^{-8}	4.0×10^{-5}	0.0025	0.0025	0.0025
Maximally exposed individual					
Annual dose (millirem)	2.6×10^{-7}	6.8×10^{-5}	0.0025	0.0026	NA[a]
35-year latent cancer fatality risk	4.6×10^{-12}	1.2×10^{-9}	4.4×10^{-8}	4.5×10^{-8}	NA[a]
Average exposed individual within 80 kilometers (50 miles)					
Annual dose[b] (millirem)	2.0×10^{-8}	1.5×10^{-6}	9.1×10^{-5}	9.3×10^{-5}	NA[a]
35-year latent cancer fatality risk	3.6×10^{-13}	2.6×10^{-11}	1.6×10^{-9}	1.6×10^{-9}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FDPF or the generic site in the year 2020 (188,400 and 1,538,100, respectively).

Key: NA, not applicable.
Source: Appendix E; model results, using the GENII computer code (Napier et al. 1988).

Incremental doses to involved workers from normal operations are given in **Table 4–155**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to research reactor workers during operations would be 100 millirem; for support facility workers, the incremental annual average dose during operations would be 114 millirem; for FDPF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities is estimated to be 12, 11, and 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–155. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–155 Incremental Radiological Impacts on Involved Workers at the Generic DOE Site and INEEL from Operational Facilities Under Alternative 4 (Construct New Research Reactor)—Option 2

Receptor—Involved Workers[a]	INEEL FDPF	Generic Site		Two-Site Total
		Research Reactor Operations	Reactor Support Facility Operations	
Total dose (person-rem per year)	12[b]	12[b]	11[b]	36
35-year latent cancer fatalities	0.17	0.17	0.16	0.50
Average worker dose (millirem per year)	170	100	114	NA[c]
35-year latent cancer fatality risk	0.0023	0.0014	0.0016	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that each facility adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers at FDPF, 120 research reactor workers, and 100 workers at the reactor support facility.
c. Values cannot be given for the average worker because the workers would be in three different facilities at two different sites.

Key: NA, not applicable.
Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts associated with the operation of the research reactor and support facility are addressed in Section 4.6.1.2.9.

Impacts from hazardous chemicals at INEEL were determined to be the same as in Alternative 2, Option 2 (Section 4.4.2.1.9).

4.6.2.2.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with research reactor target irradiation; support facility medical, industrial, and research and development isotope fabrication and processing; and FDPF neptunium-237 target fabrication and processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–156** and **4–157**, respectively.

Table 4–156 New Research Reactor, Support Facility, and FDPF Accident Consequences Under Alternative 4 (Construct New Research Reactor)—Option 2

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
New research reactor accidents						
Design-basis accident	1.33×10^{-6}	6.65×10^{-10}	0.00241	1.20×10^{-6}	5.49×10^{-6}	2.20×10^{-9}
Beyond-design-basis earthquake	0.00373	1.87×10^{-6}	27.6	0.0138	0.0531	2.12×10^{-5}
Fuel-handling accident	1.90×10^{-9}	9.50×10^{-13}	6.79×10^{-6}	3.40×10^{-9}	5.83×10^{-9}	2.33×10^{-12}
Neptunium-237 target-handling accident	5.42×10^{-8}	2.71×10^{-11}	8.95×10^{-5}	4.47×10^{-8}	2.43×10^{-7}	9.72×10^{-11}
Medical isotope target-handling accident	1.04×10^{-5}	5.20×10^{-9}	0.101	5.06×10^{-5}	6.76×10^{-6}	2.70×10^{-9}
Support facility accidents						
Medical and industrial isotopes localized solvent fire	0.0194	9.72×10^{-6}	31.1	0.0156	0.00530	2.12×10^{-6}
Medical and industrial isotopes unlikely seismic event	0.0750	3.75×10^{-5}	136	0.0680	0.510	2.04×10^{-4}
Medical and industrial isotopes glovebox explosion	2.50	0.00125	4,600	2.30	17.0	0.00680
FDPF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.01×10^{-9}	1.01×10^{-12}	2.49×10^{-5}	1.24×10^{-8}	7.26×10^{-9}	2.91×0^{-12}
Target dissolver tank failure during plutonium-238 separation	6.11×10^{-8}	3.05×10^{-11}	5.65×10^{-4}	2.82×10^{-7}	2.17×10^{-7}	8.69×10^{-11}
Ion exchange explosion during plutonium-238 separation	1.63×10^{-5}	8.13×10^{-9}	0.150	7.51×10^{-5}	5.79×10^{-5}	2.31×10^{-8}
Plutonium-238 processing facility beyond-design-basis earthquake	42.5	0.0425	1.64×10^{5}	82.0	1,200	1.0 [c]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–157 New Research Reactor, Support Facility, and FDPF Accident Risks Under Alternative 4 (Construct New Research Reactor)—Option 2

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual new research reactor risks			
Design-basis accident (1×10^{-4})	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis earthquake (1×10^{-5})	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident (0.01)	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident (0.01)	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical isotope target-handling accident (0.01)	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
35-year new research reactor risk	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Annual support facility risks			
Medical and industrial isotopes localized solvent fire (0.044)	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical and industrial isotopes glovebox explosion (1.00×10^{-4})	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
35-year support facility risk	3.26×10^{-5}	0.056	9.85×10^{-5}
Annual FDPF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	4.25×10^{-7}	8.20×10^{-4}	$1.00\times10^{-5(c)}$
35-year FDPF risk	1.49×10^{-5}	0.0287	$3.50\times10^{-4(c)}$
35-year Option risk	4.75×10^{-5}	0.0848	4.49×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

For 35 years of research reactor target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.49×10^{-9} and 8.41×10^{-9}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 2.26×10^{-5}.

For 35 years of support facility medical, industrial, and research and development target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 3.26×10^{-5} and 9.85×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.056.

For 35 years of FDPF neptunium-237 target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be

1.49×10^{-5} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0287.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 4.75×10^{-5} and 4.49×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0848.

There are no hazardous chemical accidents associated with the new research reactor or new support facility. The irradiation of neptunium-237, medical, industrial, and research and development isotopes in the new research reactor would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The fabrication and processing of medical, industrial, and research and development isotopes at the new support facility would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The hazardous chemical accident impacts at FDPF are the same as those presented in Section 4.4.5.1.10.

4.6.2.2.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FDPF target fabrication facility at INEEL. DOE would transport the unirradiated neptunium-237 targets from FDPF to the reactor site. Following irradiation in the reactor, the targets would be returned to FDPF for processing. After this processing, the plutonium-238 product would be shipped to LANL. The reactor would receive low enriched uranium fuel from a U.S. fuel fabrication facility. Additionally, medical and industrial isotopes would be shipped from the reactor site to a local airport, and from there to locations throughout the country.

Approximately 37,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 7.5 million kilometers (4.7 million miles); and in the air carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 29.2 person-rem; the dose to the public, 315 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.012 latent cancer fatality among transportation workers and 0.16 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.026. About half of the crew risk, about 2 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying

neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.19 traffic fatality. Nearly all of the radiological and about 59 percent of the traffic accident risk would result from shipping medical and industrial isotopes.

4.6.2.2.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in a new reactor that would be constructed at a site yet to be specified. Fabrication and processing of neptunium-237 targets for plutonium-238 production would be performed at FDPF at INEEL. A new support facility would be constructed at an unspecified site for fabrication and processing targets not used for plutonium-238 production.

Activities at FDPF were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.2.1.12) and found to pose no significant radiological or other risks to minority and low-income populations. The environmental analysis of operations at the new research reactor and support facility site shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected areas would not be significant. Unless there are patterns of food consumption among minority or low-income resident surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that operations at the site would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site-specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new research reactor and support facility, an additional evaluation of environmental justice at the reactor and support facility site during operation would be performed prior to implementation.

4.6.2.2.13 Waste Management

The impacts of managing waste generated from the operation of a new research reactor to irradiate targets and a support facility to fabricate and process medical and industrial isotope targets and to meet research and development needs are assumed to be the same as for Option 1 (Section 4.6.1.2.13). Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.6.2.2.9 through 4.6.2.2.11.

The impacts of managing waste associated with fabricating and processing neptunium-237 targets for plutonium-238 production in FDPF at INEEL are assumed to be the same as for Option 2 under Alternative 1 (Section 4.3.2.1.13). As shown in that section, the impacts on the waste management systems at INEEL would be minimal.

4.6.2.2.14 Spent Nuclear Fuel Management

The impacts associated with spent nuclear fuel management under this option would be the same as for Option 1, and are given in Section 4.6.1.2.14.

4.6.2.3 Decontamination and Decommissioning of the Research Reactor and Support Facility

The environmental impacts associated with the decontamination and decommissioning of the research reactor and support facility at the generic DOE site are assessed in Section 4.6.1.3.

4.6.2.4 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.6.3 Alternative 4 (Construct New Research Reactor)—Option 3

Option 3 involves construction and operating the research reactor to irradiate all targets associated with plutonium-238 production, medical and industrial isotope production, and research and development; operating FMEF at Hanford to fabricate and process neptunium-237 targets and to process the plutonium-238 product; and constructing and operating the support facility to fabricate and process the other targets and materials and to process the associated products. This option includes storage in FMEF of the neptunium-237 transported to Hanford from SRS and storage in the new support facility of the other target materials transported to the generic site from other offsite facilities.

The transportation of the low enriched uranium fuel for use in the research reactor, the transportation of the neptunium-237 to Hanford and then to the generic site, the transportation of the other target materials to the generic site, and the transportation of the product materials following irradiation and postirradiation processing are also part of this option.

All options under this alternative include the contamination and decommissioning of the research reactor and support facility at the generic DOE site following their operating lifetimes, and also the permanent deactivation of FFTF at Hanford.

4.6.3.1 Construction of the New Research Reactor and Support Facility

Environmental impacts associated with the construction of a new research reactor and support facility at the generic DOE site are assessed in Section 4.6.1.1.

4.6.3.2 Operations and Transportation

The environmental impacts associated with storage, processing, and irradiation operations, and with all transportation activities, are assessed in this section.

4.6.3.2.1 Land Resources

LAND USE. Impacts on land use associated with the operation of a reactor and support facility are addressed in Section 4.6.1.2.1.

FMEF would be used for neptunium-237 storage, target fabrication, and processing. Land use within the 400 Area would not change since the use of FMEF would be compatible with the mission for which it was designed.

VISUAL RESOURCES. Impacts on visual resources associated with the operation of a reactor and support facility are addressed in Section 4.6.1.2.1.

All activities associated with neptunium-237 storage, target fabrication, and processing would take place within FMEF. Operations associated with the proposed activities would not result in any impact on visual resources or change in the current Visual Resource Management Class IV rating of the 400 Area. This is because none of the anticipated operational impacts (e.g., air emissions) would be expected to affect this resource.

4.6.3.2.2 Noise

Noise impacts associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.2.

This option also involves using FMEF for target material storage, target fabrication, and processing. Activities associated with construction of a new stack would be typical of small construction projects and would result in some temporary increase in noise. Noise sources associated with this construction would not be expected to be loud impulsive sources and would not be expected to result in disturbance of wildlife around the 400 Area. The operation of FMEF would not be expected to result in any change in noise impacts on wildlife around the 400 Area and offsite noise impacts would also be minor because the nearest site boundary is 7 kilometers (4.3 miles) to the east. Operation would be expected to result in minimal change in noise impacts on people near Hanford as a result of changes in employee and truck traffic levels.

4.6.3.2.3 Air Quality

Air quality impacts associated with the operation of the new research reactor and support facility are addressed in Section 4.6.1.2.3.

Air quality impacts at Hanford associated with this option were determined to be the same as in Alternative 2, Option 3 (Section 4.4.3.1.3).

The air quality impacts of transportation are presented in Section 4.6.3.2.11.

4.6.3.2.4 Water Resources

Impacts on water resources associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.4.

FMEF in the 400 Area of Hanford would be used for neptunium-237 storage, target fabrication, and processing in support of plutonium-238 production. The operation of FMEF for this purpose is projected to require approximately 19 million liters (5 million gallons) of groundwater annually. This includes approximately 15 million liters (4 million gallons) per year to support FMEF cooling needs and an additional 3.8 million liters (1 million gallons) per year for potable and sanitary water demands due to increased staffing. However, no impact on regional groundwater levels would be expected from increased withdrawals. FMEF groundwater usage would constitute an increase of about 10 percent over the 197 million liters (52 million gallons) withdrawn annually in the 400 Area during standby operations. Sanitary wastewater discharges from FMEF would also increase by roughly 3.8 million liters (1 million gallons) per year to the Energy Northwest treatment system, which has sufficient capacity. Also, the operation of FMEF for target fabrication and processing would generate approximately 15 million liters (4 million gallons) per year of process wastewater. This wastewater would be discharged to the 400 Area process sewer system and ultimately to the 400 Area Pond (i.e., 4608 B/C percolation ponds) (DOE 2000a:B-3; Nielsen 1999:38, 39, 41). As discharges to the pond are regulated under State Waste Discharge Permit No. ST-4501 and there are no radiological liquid effluent pathways to the environment from FMEF, the impact on groundwater quality would be negligible.

It should be noted that the increase in water use and sanitary and process wastewater discharge for FMEF operations would essentially be negated by the larger reductions in water use and wastewater discharge associated with the permanent deactivation of FFTF (see Section 4.4.1.2.4).

4.6.3.2.5 Geology and Soils

Impacts on geology and soils associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.5. As discussed in Section 4.6.1.1.5, the proposed facilities would be designed and constructed in accordance with DOE Order 420.1 and sited to minimize the risk from geologic hazards. Thus, site geologic conditions would be unlikely to affect the facilities.

The use of FMEF for neptunium-237 storage, target fabrication, and processing would not be expected to impact geologic resources, nor be jeopardized by large-scale geologic conditions. Hazards from large-scale geologic conditions at Hanford, such as earthquakes and volcanoes, were previously evaluated in the *Storage and Disposition PEIS* (DOE 1996a:4-45) as discussed in Section 4.2.4.2.5. The analysis determined that these hazards present a low risk to long-term storage facilities. That analysis was reviewed in the *Surplus Plutonium Disposition EIS* (DOE 1999a:4-260). Further review of the data and analyses presented in these referenced documents and the site-specific data presented in this NI PEIS indicates that the large-scale geologic conditions likewise present a low risk to FMEF operations. As necessary, the need to evaluate and upgrade existing DOE facilities with regard to natural geologic hazards would be assessed in accordance with DOE Order 420.1, which is described in Section 4.2.1.2.5.

4.6.3.2.6 Ecological Resources

Impacts on ecological resources associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.6.

This option also involves using FMEF for neptunium-237 storage, target fabrication, and processing. As noted in Section 4.6.3.2.2, there would be no loud noises that would disturb wildlife. Because additional water usage and wastewater discharge would be small fractions of current values, there would be no change in impacts on aquatic habitat or wetlands associated with the Columbia River (Section 4.6.3.2.4). Threatened and endangered species would not be affected by operation because an existing facility within an already developed area would be used.

Consultation letters concerning threatened and endangered species were sent to the U.S. Fish and Wildlife Service, the National Marine Fisheries Service, the Washington State Department of Natural Resources, and the State of Washington Department of Fish and Wildlife (see Table 5–3). Each agency was asked to provide information on potential impacts of the proposed action on threatened and endangered species. Both the Washington State Department of Natural Resources and the State of Washington Department of Fish and Wildlife provided lists of state species of concern that occur in the vicinity of the project area. As noted above, no impacts to any threatened or endangered species are expected, including those of concern to these agencies. While DOE has made additional contacts with the U.S. Fish and Wildlife Service and the National Marine Fisheries Service, responses are pending from these agencies. Although no federally listed species are expected to be impacted by the proposed action, no action would be taken relative to the use of facilities at Hanford prior to the receipt of input from these Federal agencies.

4.6.3.2.7 Cultural and Paleontological Resources

Impacts on cultural and paleontological resources associated with the operation of a research reactor and support facility are addressed in Section 4.6.1.2.7.

Neptunium-237 storage, target fabrication, and processing would take place at FMEF, which is in the 400 Area. No prehistoric, historic, or paleontological sites have been identified either within the 400 Area or within 2 kilometers (1.2 miles) of the 400 Area. Six buildings located within the 400 Area, including two

FFTF structures (the Reactor Containment Building and FFTF Control Building), have been determined to be eligible for the National Register as contributing properties within the Historic District recommended for mitigation. The operation of FMEF would not affect the status of these structures. No Native American resources are known to occur within the 400 Area.

Consultation to comply with Section 106 of the National Historic Preservation Act was conducted with the State Historic Preservation Office (see Table 5–3) and resulted in concurrence by the State Historic Preservation Office that the proposed action would have no effect on historic properties at Hanford. Consultation was also conducted with interested Native American tribes that resulted in comments at public hearings by members representing the Nez Perce and Confederated Tribes of the Umatilla Indian Reservation. Responses to their specific comments are addressed in Volume 3.

4.6.3.2.8 Socioeconomics

The socioeconomic impacts associated with the operation of a new research reactor and support facility at a generic DOE site are addressed in Section 4.6.1.2.8.

The socioeconomic impacts associated with neptunium-237 target fabrication and processing at Hanford are the same as those addressed in Section 4.4.3.1.8.

4.6.3.2.9 Public and Occupational Health and Safety—Normal Operations

Assessments of incremental radiological and chemical impacts associated with this option are presented in this section. Supplemental information is provided in Appendix H.

During normal operations, there would be incremental radiological and hazardous chemical releases to the environment and also incremental direct in-plant exposures. The resulting doses and potential health effects to the public and workers for this option are described below.

RADIOLOGICAL IMPACTS. Incremental radiological doses to three receptor groups from startup and operations are given in **Table 4–158** for the generic DOE site and Hanford: the population within 80 kilometers (50 miles) in the year 2020, the maximally exposed member of the public, and the average exposed member of the public. Radiological impacts from startup operations prior to fuel loading would be zero. After fuel loading, these impacts would be expected to be bound by the normal operational impacts. Therefore, startup impacts have not been treated separately from normal operational impacts. The projected number of latent cancer fatalities in the surrounding population and the latent cancer fatality risk to the maximally and average exposed individuals are also presented in the table.

A probability coefficient of 5×10^{-4} latent cancer fatality per rem is applied for the public, and a coefficient of 4×10^{-4} latent cancer fatality per rem is applied for workers (ICRP 1991). The value for workers is lower due to the absence of children and the elderly, who are more radiosensitive.

As a result of annual operations of the research reactor facilities and FMEF, the projected incremental total population dose in the year 2020 would be 0.14 person-rem; the corresponding number of latent cancer fatalities in the populations surrounding the generic DOE site and Hanford from 35 years of operations would be 0.0025. The incremental total dose to the maximally exposed member of the public from annual operations of the research reactor and support facility at the generic DOE site would be 0.0026 millirem; from 35 years of operations, the corresponding risk of a latent cancer fatality to this individual would be 4.5×10^{-8}.

Table 4–158 Incremental Radiological Impacts on the Public Around the Generic DOE Site and Hanford from Operational Facilities Under Alternative 4 (Construct New Research Reactor)—Option 3

Receptor	Hanford FMEF	Generic Site			Two-Site Total
		Research Reactor Operations	Reactor Support Facility Operations	Total	
Population within 80 kilometers (50 miles) in the year 2020					
Dose (person-rem)	4.4×10^{-5}	0.0023	0.14	0.14	0.14
35-year latent cancer fatalities	7.7×10^{-7}	4.0×10^{-5}	0.0025	0.0025	0.0025
Maximally exposed individual					
Annual dose (millirem)	4.7×10^{-7}	6.8×10^{-5}	0.0025	0.0026	NA[a]
35-year latent cancer fatality risk	8.3×10^{-12}	1.2×10^{-9}	4.4×10^{-8}	4.5×10^{-8}	NA[a]
Average exposed individual within 80 kilometers (50 miles)					
Annual dose[b] (millirem)	8.9×10^{-8}	1.5×10^{-6}	9.1×10^{-5}	9.3×10^{-5}	NA[a]
35-year latent cancer fatality risk	1.6×10^{-12}	2.6×10^{-11}	1.6×10^{-9}	1.6×10^{-9}	NA[a]

a. A "Total" cannot be given in this case because the same individual cannot be located at two different sites simultaneously.
b. Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of FMEF or the generic site in the year 2020 (494,400 and 1,538,100, respectively).

Key: NA, not applicable.
Source: Appendix E; model results, using the GENII computer code (Napier et al. 1988).

Incremental doses to involved workers from normal operations are given in **Table 4–159**; these workers are defined as those directly associated with all process activities. The incremental annual average dose to research reactor workers during operations would be 100 millirem; for support facility workers, the incremental annual average dose during startup and operations would be 114 millirem; for FMEF workers, the incremental annual average dose would be approximately 170 millirem. The incremental annual dose received by the total site workforce for each of these facilities is estimated to be 12, 11, and 12 person-rem, respectively. The risks and numbers of latent cancer fatalities among the different workers from 35 years of operations are included in Table 4–159. Doses to individual workers would be kept to minimal levels by instituting badged monitoring and ALARA programs.

Table 4–159 Incremental Radiological Impacts on Involved Workers at the Generic DOE Site and Hanford from Operational Facilities Under Alternative 4 (Construct New Research Reactor)—Option 3

Receptor—Involved Workers[a]	Hanford FMEF	Generic Site		One- or Two-Site Total
		Research Reactor Operations	Reactor Support Facility Operations	
Total dose (person-rem per year)	12[b]	12[b]	11[b]	36
35-year latent cancer fatalities	0.17	0.17	0.16	0.50
Average worker dose (millirem per year)	170	100	114	NA[c]
35-year latent cancer fatality risk	0.0023	0.0014	0.0016	NA[c]

a. The radiological limit for an individual worker is 5,000 millirem per year (10 CFR Part 835). However, the maximum dose to a worker involved with operations would be kept below the DOE Administrative Control Level of 2,000 millirem per year (DOE 1999j). Further, DOE recommends that each facility adopt a more limiting, 500 millirem per year, Administrative Control Level (DOE 1999j). To reduce doses to levels that are as low as is reasonably achievable (ALARA), an effective ALARA program would be enforced.
b. Based on an estimated 75 badged workers at FMEF, 120 research reactor workers, and 100 workers at the reactor support facility.
c. Values cannot be given for the average worker because the workers would be in three different facilities at two different sites.

Key: NA, not applicable.
Source: Wham 1999b, 2000.

HAZARDOUS CHEMICAL IMPACTS. Hazardous chemical impacts associated with the operation of the research reactor and support facility are addressed in Section 4.6.1.2.9.

Impacts from hazardous chemicals at Hanford were determined to be the same as in Alternative 2, Option 3 (Section 4.4.3.1.9).

4.6.3.2.10 Public and Occupational Health and Safety—Facility Accidents

Impacts from postulated accidents associated with research reactor target irradiation; support facility medical, industrial, and research and development isotope fabrication and processing; and FMEF neptunium-237 target fabrication and processing are presented in this section. Detailed descriptions of the accident analyses are provided in Appendix I.

Consequences and associated risks are presented in **Tables 4–160** and **4–161**, respectively.

For 35 years of research reactor target irradiation, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 2.49×10^{-9} and 8.41×10^{-9}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 2.26×10^{-5}.

For 35 years of support facility medical, industrial, and research and development target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and to a noninvolved worker would be 3.26×10^{-5} and 9.85×10^{-5}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.056.

For 35 years of FMEF neptunium-237 target fabrication and processing, the increased risk of a latent cancer fatality to the maximally exposed individual and of an early fatality to a noninvolved worker would be 2.88×10^{-6} and 3.50×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.112.

For 35 years under this option, the increased risk of a latent cancer fatality to the maximally exposed individual and of a fatality to a noninvolved worker would be 3.55×10^{-5} and 4.49×10^{-4}, respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.168.

There are no hazardous chemical accidents associated with the new research reactor or new support facility. The irradiation of neptunium-237, medical, industrial, and research and development isotopes in the new research reactor would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The fabrication and processing of medical, industrial, and research and development isotopes at the new support facility would not require the use of hazardous chemicals in amounts that exceed the Threshold Planning Quantities on the Extremely Hazardous Substances List (EPA 1998).

The hazardous chemical accident impacts at FMEF are the same as those presented in Section 4.4.6.1.10.

Table 4–160 New Research Reactor, Support Facility, and FMEF Accident Consequences Under Alternative 4 (Construct New Research Reactor)—Option 3

Accident	Maximally Exposed Individual		Population to 80 Kilometers (50 Miles)		Noninvolved Worker	
	Dose (rem)	Latent Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]	Dose (rem)	Latent Cancer Fatality[a]
New research reactor accidents						
Design-basis accident	1.33×10^{-6}	6.65×10^{-10}	0.00241	1.20×10^{-6}	5.49×10^{-6}	2.20×10^{-9}
Beyond-design-basis earthquake	0.00373	1.87×10^{-6}	27.6	0.0138	0.0531	2.12×10^{-5}
Fuel-handling accident	1.90×10^{-9}	9.50×10^{-13}	6.79×10^{-6}	3.40×10^{-9}	5.83×10^{-9}	2.33×10^{-12}
Neptunium-237 target-handling accident	5.42×10^{-8}	2.71×10^{-11}	8.95×10^{-5}	4.47×10^{-8}	2.43×10^{-7}	9.72×10^{-11}
Medical isotope target- handling accident	1.04×10^{-5}	5.20×10^{-9}	0.101	5.06×10^{-5}	6.76×10^{-6}	2.70×10^{-9}
Support facility accidents						
Medical and industrial isotopes localized solvent fire	0.0194	9.72×10^{-6}	31.1	0.0156	0.00530	2.12×10^{-6}
Medical and industrial isotopes unlikely seismic event	0.0750	3.75×10^{-5}	136	0.0680	0.510	2.04×10^{-4}
Medical and industrial isotopes glovebox explosion	2.50	0.00125	4,600	2.30	17.0	0.00680
FMEF accidents						
Ion exchange explosion during neptunium-237 target fabrication	2.02×10^{-9}	1.01×10^{-12}	7.26×10^{-5}	3.63×10^{-8}	6.65×10^{-10}	2.66×10^{-13}
Target dissolver tank failure during plutonium-238 separation	4.64×10^{-8}	2.32×10^{-11}	0.00169	8.47×10^{-7}	1.95×10^{-8}	7.81×10^{-12}
Ion exchange explosion during plutonium-238 separation	1.24×10^{-5}	6.18×10^{-9}	0.451	2.25×10^{-4}	5.20×10^{-6}	2.08×10^{-9}
Plutonium-238 processing facility beyond-design-basis earthquake	16.5	0.00823	6.41×10^{5}	321	921	1.0[c]

a. Likelihood of a latent cancer fatality.
b. Number of latent cancer fatalities.
c. Early fatality due to radiation dose. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

Table 4–161 New Research Reactor, Support Facility, and FMEF Accident Risks Under Alternative 4 (Construct New Research Reactor)—Option 3

Accident (Frequency)	Maximally Exposed Individual[a]	Population to 80 Kilometers (50 Miles)[b]	Noninvolved Worker[a]
Annual new research reactor risks			
Design-basis accident (1×10^{-4})	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis earthquake (1×10^{-5})	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident (0.01)	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident (0.01)	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical isotope target-handling accident (0.01)	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
35-year new research reactor risk	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Annual support facility risks			
Medical and industrial isotopes localized solvent fire (0.044)	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical and industrial isotopes glovebox explosion (0.01)	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
35-year support facility risk	3.26×10^{-5}	0.056	9.85×10^{-5}
Annual FMEF risks			
Ion exchange explosion during neptunium-237 target fabrication (0.01)	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	8.23×10^{-8}	0.00321	1.00×10^{-5}[c]
35-year FMEF risk	2.88×10^{-6}	0.112	3.50×10^{-4}[c]
35-year Option risk	3.55×10^{-5}	0.168	4.49×10^{-4}

a. Increased likelihood of a latent cancer fatality.
b. Increased number of latent cancer fatalities.
c. Risk of an early fatality.

Source: Model results, using the MACCS2 (Chanin and Young 1997) and GENII (Napier et al. 1988) computer codes.

4.6.3.2.11 Public and Occupational Health and Safety—Transportation

DOE would transport neptunium-237 from storage at SRS to the FMEF target fabrication facility at Hanford. DOE would transport the unirradiated neptunium-237 targets from FMEF to the reactor site. Following irradiation in the reactor, the targets would be returned to FMEF for processing. After this processing, the plutonium-238 product would be shipped to LANL. The reactor would receive low enriched uranium fuel from a U.S. fuel fabrication facility. Additionally, medical and industrial isotopes would be shipped from the reactor site to a local airport, and from there to locations throughout the country.

Approximately 37,000 shipments of radioactive materials would be made by DOE. The total distance traveled on public roads by trucks carrying radioactive materials would be 7.9 million kilometers (4.9 million miles); and in the air carrying medical isotopes, 23 million kilometers (14 million miles).

The transportation impact analysis is described in detail in Appendix J.

IMPACTS OF INCIDENT-FREE TRANSPORTATION. The dose to transportation workers from all transportation activities entailed by this option has been estimated at 31 person-rem; the dose to the public, 354 person-rem. Accordingly, incident-free transportation of radioactive material associated with this option would result in 0.012 latent cancer fatality among transportation workers and 0.18 latent cancer fatality in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this option would be 0.027. About half of the crew risk, about 2 percent of the public risk, and most of the emissions risk would result from shipping medical and industrial isotopes.

IMPACTS OF ACCIDENTS DURING TRANSPORTATION. The maximum foreseeable offsite transportation accident under this option (probability of occurrence: 1 in 10 million per year) is a shipment of irradiated neptunium-237 targets to FDPF with a severity Category V accident in an urban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatality, and 2.6 millirem to the hypothetical maximally exposed individual with a latent cancer fatality risk of 1.3×10^{-6}. No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of the accident, or occurrence while carrying neptunium-237 (unirradiated) or plutonium-238 was also evaluated and estimated to have a probability of less than 1 in 10 million per year.

Estimates of the total transportation accident risks under this option are as follows: a radiological dose to the population of 1,063 person-rem, resulting in 0.53 latent cancer fatality; and traffic accidents resulting in 0.19 traffic fatality. Nearly all of the radiological and about 58 percent of the traffic accident risk would result from shipping medical and industrial isotopes.

4.6.3.2.12 Environmental Justice

Under this option, neptunium-237 targets would be irradiated in a new reactor that would be constructed at a site yet to be specified. Fabrication and processing of neptunium-237 targets for plutonium-238 production would be performed at FMEF at Hanford. A new support facility would be constructed at an unspecified site for fabrication and processing targets not used for plutonium-238 production.

Activities at FMEF were evaluated under other alternatives and options in this NI PEIS (e.g., Section 4.4.3.1.12) and were found to pose no significant radiological or other risks to minority and low-income populations. The environmental analysis of operations at the new research reactor and support facility site shows that radiological and nonradiological risks to persons residing in the (hypothetical) potentially affected areas would not be significant. Unless there are patterns of food consumption among minority or low-income residents surrounding the actual site (yet to be determined) that would result in a significant ingestion of radiologically contaminated food, it is plausible that operations at the site would pose no significant risks to minority and low-income persons. However, evaluations of environmental justice are necessarily site-specific and cannot be performed in detail for unspecified locations. In the event that this option were selected for implementation and a specific site selected for the new research reactor and support facility, then an additional evaluation of environmental justice at the reactor and support facility site during operation would be performed prior to implementation.

4.6.3.2.13 Waste Management

The impacts of managing waste generated from the operation of a new research reactor to irradiate targets and a support facility to fabricate and process medical and industrial isotope targets and to meet research and development needs are assumed to be the same as for Option 1 (Section 4.6.1.2.13). Radiological and chemical impacts on workers and the public from waste management activities are included in the public and occupational health and safety impacts that are given in Sections 4.6.3.2.9 through 4.6.3.2.11.

The impacts of managing waste associated with fabricating and processing neptunium-237 targets for plutonium-238 production in FMEF at Hanford are assumed to be the same as for Option 3 under Alternative 1 (Section 4.3.3.1.13). As shown in that section, the impacts on the waste management systems at Hanford would be minimal.

4.6.3.2.14 Spent Nuclear Fuel Management

The impacts associated with spent nuclear fuel management under this option would be the same as for Option 1, and are given in Section 4.6.1.2.14.

4.6.3.3 Decontamination and Decommissioning of the Research Reactor and Support Facility

The environmental impacts associated with the decontamination and decommissioning of the research reactor and support facility at the generic DOE site are assessed in Section 4.6.1.3.

4.6.3.4 Permanent Deactivation of FFTF

The environmental impacts associated with permanently deactivating FFTF are addressed in Section 4.4.1.2.

4.7 ALTERNATIVE 5—PERMANENTLY DEACTIVATE FFTF (WITH NO NEW MISSIONS)

Under Alternative 5, DOE would permanently deactivate FFTF at Hanford with no new missions. Medical and industrial isotope production and civilian nuclear energy research and development missions at the existing facilities, as described in Chapter 3, Affected Environment, would continue. DOE's nuclear facilities infrastructure would not be enhanced. The environmental impacts associated with FFTF deactivation are addressed in Section 4.4.1.2.

4.8 CUMULATIVE IMPACTS

The projected incremental environmental impacts of (1) constructing (as necessary) and operating the proposed facilities to store, fabricate, irradiate, and process the various targets addressed in this NI PEIS for 35 years, and (2) deactivating FFTF were added to the environmental impacts of other present and reasonably foreseeable future actions at or near the identified candidate sites to obtain cumulative site impacts under normal conditions. The other present and reasonably foreseeable future actions at or near the candidate sites are included in the baseline impacts presented in Chapter 3. Cumulative transportation impacts were determined by analyzing the impacts along the various routes used to transport the materials associated with nuclear infrastructure activities over the 35-year period. The methodology for assessing cumulative impacts is presented in Section G.10.

In this section, cumulative site impacts are presented only for those "resources" at a site that may reasonably be expected to be affected by the storage, fabrication, irradiation, and processing of the various targets. These include site employment, electrical consumption, water usage, air quality, waste management, and public and occupational health and safety. This section also includes the cumulative impacts associated with intersite transportation.

Impacts of the following are considered in the cumulative site impacts assessment:

- Current (baseline) activities at or in the vicinity of the candidate sites
- Other onsite and offsite activities that are reasonably foreseeable and documented
- Construction (as necessary), operation, and deactivation (as necessary) of the proposed nuclear infrastructure facilities to fabricate, irradiate, and process targets

Activities whose impacts are contained in cumulative site impacts include, but are not limited to, operation of the Spallation Neutron Source Facility at ORR, implementation of the Advanced Mixed Waste Treatment Project at INEEL, and remediation of the high-level waste tanks at Hanford.

Details of activities that may be implemented in the foreseeable future at any of the nuclear infrastructure candidate sites and evaluated in the cumulative impact assessment are given in the following documents:

- *Surplus Plutonium Disposition Final Environmental Impact Statement* (DOE 1999a) (Record of Decision issued)

- *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (DOE 1996a) (Record of Decision issued)

- *Disposition of Surplus Highly Enriched Uranium Final Environmental Impact Statement* (DOE 1996b) (Record of Decision issued)

- *Final Programmatic Environmental Impact Statement for Tritium Supply and Recycling* (DOE 1995b) (Record of Decision issued)

- *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (DOE 1997a) (Final EIS issued; Records of Decision issued for the various waste types)

- *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995c) (Record of Decision issued)

- *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (DOE 1996c) (Record of Decision issued)

- *Final Programmatic Environmental Impact Statement for Stockpile Stewardship and Management* (DOE 1996d) (Record of Decision issued)

- *Final Environmental Impact Statement Management of Spent Nuclear Fuel from the K Basins at the Hanford Site* (DOE 1996h) (Record of Decision issued)

- *Advanced Mixed Waste Treatment Project Final Environmental Impact Statement* (DOE 1999c) (Record of Decision issued)

- *Final Environmental Impact Statement for the Tank Waste Remediation System, Hanford Site, Richland, Washington* (DOE 1996e) (Record of Decision issued)

- *Hanford Reach of the Columbia River Comprehensive River Conservation Study and Environmental Impact Statement* (NPS 1994) (Record of Decision issued)

- *Final Hanford Comprehensive Land-Use Plan Environmental Impact Statement* (DOE 1999d) (Record of Decision issued)

- *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE 2000b) (Record of Decision issued)

- *Final Environmental Impact Statement, Construction and Operation of the Spallation Neutron Sources* (DOE 1999e) (Record of Decision issued)

- *Final Programmatic Environmental Impact Statement for Alternative Strategies for the Long-Term Management and Use of Depleted Uranium Hexafluoride* (DOE 1999f) (Record of Decision issued)

- *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement* (DOE 1999g)

- *Final Environmental Impact Statement for Treating Transuranic (TRU)/Alpha Low-Level Waste at the Oak Ridge National Laboratory* (DOE 2000c) (Record of Decision issued)

- *Environmental Assessment Melton Valley Storage Tanks Capacity Increase Project—Oak Ridge National Laboratory* (DOE/EA-1044 and FONSI, DOE 1995d)

- *Management of Spent Nuclear Fuel on the Oak Ridge Reservation* (DOE/EA-1117 and FONSI, DOE 1996f)

- *Environmental Assessment - Management of Hanford Site Non-Defense Production Reactor Spent Nuclear Fuel* (DOE/EA-1185 and FONSI, DOE 1997c)

- *Environmental Assessment for Transportation of Low-Level Radioactive Waste from the Oak Ridge Reservation to Off-Site Treatment or Disposal Facilities* (DOE 2000d)

- *Environmental Assessment for Transportation of Mixed Low-Level Radioactive Waste from the Oak Ridge Reservation to Off-Site Treatment or Disposal Facilities* (DOE 2000e)

- *Environmental Assessment for Selection and Operation of the Proposed Field Research Centers for the Natural and Accelerated Bioremediation Field Research (NABIR) Program* (DOE 2000f) (FONSI issued April 2000).

The related programs included in the cumulative impact assessment for the potentially affected candidate sites are identified in **Table 4–162**.

Table 4–162 Other Present and Reasonably Foreseeable Actions Considered in the Cumulative Impact Assessment

Activities	ORR	INEEL	Hanford
Disposition of Surplus Plutonium	X		
Storage and Disposition of Weapons-Usable Fissile Materials	X	X	X
Disposition of Surplus Highly Enriched Uranium	X		
Waste Management PEIS	X	X	X
Spent Nuclear Fuel Management and INEL Environmental Restoration and Waste Management		X	X
Foreign Research Reactor Spent Nuclear Fuel Management		X	X
Stockpile Stewardship and Management	X		
Tank Waste Remediation			X
Radioactive Releases from WNP Nuclear Power Plant			X
Hanford Reach of the Columbia River Comprehensive River Conservation Study			X
Hanford Comprehensive Land Use Plan			X
K Basins Spent Fuel Management			X
Advanced Mixed Waste Treatment Project		X	
Treatment and Management of Sodium-Bonded Spent Nuclear Fuel		X	
Construction and Operation of the Spallation Neutron Source	X		
Long-Term Management and Use of Depleted Uranium Hexafluoride	X		
Treatment and Shipment of Transuranic Waste	X		
Management of Liquid Low-Level Radioactive Waste	X		
Management of Spent Nuclear Fuel	X		
Transportation of Low-Level Radioactive Waste to Off-Site Treatment or Disposal	X		
Transportation of Mixed Low-Level Radioactive Waste to Off-Site Treatment or Disposal	X		
Natural and Accelerated Bioremediation Field Research Center Assessment	X		
Idaho High-Level Waste and Facilities Disposition		X	

Source: Literature review.

In the tables that are included in the following sections, all relevant activities at each site are identified to the extent possible. They include existing and reasonably foreseeable activities, and those associated with nuclear infrastructure operations. The impacts associated with the latter are specifically shown as "New Nuclear Infrastructure Operations." They include the impacts from construction (as necessary), operation, and deactivation (as necessary) of the proposed target fabrication, irradiation, and processing facilities assessed in this NI PEIS.

A bounding option was analyzed for each site. The bounding option is the option that would involve the greatest amounts of operational activities and associated environmental impacts at the candidate site. For example, the bounding option for ORR is Option 7 of Alternative 2, under which both HFIR and REDC operations would be involved in plutonium-238 production.

In addition to reasonably foreseeable site activities, other activities within the regions of the candidate sites were considered in the cumulative impact analysis for the selected resources. However, because of the distances between the candidate sites and these other existing and planned facilities, there is little opportunity for interactions among them.

4.8.1 Cumulative Impacts at ORR

For ORR, the bounding option for this NI PEIS is Option 7 of Alternative 2. This option calls for the operation of HFIR to irradiate neptunium-237 targets and the operation of REDC to fabricate and process these targets and other neptunium-237 targets irradiated in ATR. The impacts associated with HFIR and REDC operations for other missions are included under "existing site activities."

4.8.1.1 Resource Requirements

Cumulative impacts on resource requirements at ORR are presented in **Table 4–163**. ORR would remain within its site capacity for all major resources. If Option 7 of Alternative 2 were implemented, the proposed nuclear infrastructure facilities would require essentially no change in the site's use of electricity or water. Cumulatively, ORR would use approximately 10 percent of its electrical capacity and 37 percent of its water capacity. Site employment would increase by approximately 41 workers.

Table 4–163 Maximum Cumulative Resource Use and Impacts at ORR

Activities[a]	Site Employment	Electrical Consumption (megawatt-hours per year)	Water Usage (million liters per year)
Existing site activities[b]	14,215	726,000	14,210
Storage and Disposition PEIS	Included above	7,260	0.24
Waste Management PEIS	1,259	84,160	394
Spallation Neutron Source	744	543,120	1,592
Treatment and Shipment of Transuranic Waste	17	3,000	3.8
New nuclear infrastructure operations[c]	41[d]	Negligible[e]	2.86
Total	16,276	~1,363,540	16,203
Total site capacity	NA	13,880,000	44,348

a. See Section 4.8 and Table 4–162 for a listing of past, present, and reasonably foreseeable actions considered.
b. Reflects current sitewide activities that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Nuclear infrastructure activities from Alternative 2, Option 7.
d. Some, or all, of these worker requirements may be filled by the reassignment of the existing site workforce.
e. Additional electricity consumption associated with this option would be negligible compared to that associated with existing facility activities.

Note: To convert from liters per year to gallons per year, multiply by 0.264; to convert from megawatt-hours to British thermal units, multiply by 3.42×10^6; ~ means "approximately" and indicates that new nuclear infrastructure operations would contribute only minimally.
Key: NA, not applicable.
Source: DOE 1996a:4-246, 4-255; 1997a:10-18, 10-32, 10-45, 10-58; 1999a:3-185; 1999e:4-45, 5-21, 5-177; 2000c:S-34, 4-60, 4-93; LMER 1997; Wham 1999a; Sections 4.4.7.1.4 and 4.4.7.1.8 of this NI PEIS.

4.8.1.2 Air Quality

Cumulative impacts on air quality at ORR are presented in **Table 4–164**. ORR is currently in compliance with all Federal and State ambient air quality standards, and would continue to be in compliance even if the cumulative effects of all activities are included. As shown in the table, the contributions of nuclear infrastructure operations to overall site concentrations would be very small.

Table 4–164 Maximum Cumulative Air Pollutant Concentrations at ORR for Comparison with Ambient Air Quality Standards

Parameter	Carbon Monoxide		Nitrogen Dioxide	PM_{10}		Sulfur Dioxide		
Averaging Period	8 hours	1 hour	Annual	Annual	24 hours	Annual	24 hours	3 hours
Activities								
Existing site activities[a] (micrograms per cubic meter)	7.75	26.5	0.98	1.6	12.6	4.76	33.4	106.4
HEU disposition[b] (micrograms per cubic meter)	11.5	53	1.33	0.03	0.37	2.46	29.3	161
Waste management program (micrograms per cubic meter)	0	0	0	3	9	2.4	11	39
Spallation Neutron Source (micrograms per cubic meter)	69	99	16	1.9	23	0.1	1	2.4
New nuclear infrastructure operations[c] (micrograms per cubic meter)	0	0	1.99×10^{-4}	0	0	0.04	0.31	0.7
Total concentration (micrograms per cubic meter)	88.3	179	18.3	6.53	45	9.76	75	310
Standard								
Most stringent standard[d] (micrograms per cubic meter)	10,000	40,000	100	50	150	80	365	1,300

a. Environmental impacts associated with existing site activities (based on 1998 emissions from the *Oak Ridge Reservation Annual Site Environmental Report 1998* [Hamilton et al. 1999]) that are anticipated to continue during part or all of the 35-year period evaluated for proposed nuclear infrastructure operations. The values in this row reflect a curtailment of stockpile stewardship management activities during this time period.
b. Highly enriched uranium disposition activities.
c. Nuclear infrastructure activities from Alternative 2, Option 7.
d. The more stringent of the Federal and state standards is presented if both exist for the averaging period.

Source: DOE 1996b; 1996d; 1997a; 1999e:5-27; Hamilton et al. 1999; modeled results from nuclear infrastructure operations are based on the SCREEN3 computer code (EPA 1995); Sections 4.4.1.1.3 and 4.4.7.1.3 of this NI PEIS.

4.8.1.3 Public and Occupational Health and Safety—Normal Operations

Cumulative impacts in terms of radiation exposure to the public and workers at ORR are presented in **Table 4–165**. There would be no increase expected in the number of latent cancer fatalities in the population from ORR site operations if nuclear infrastructure operations were to occur at HFIR and REDC. The dose limits for individual members of the public are given in DOE Order 5400.5. As discussed in that order, the dose limit from airborne emissions is 10 millirem per year, as required by the Clean Air Act; the dose limit from drinking water is 4 millirem per year, as required by the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. Therefore, as is evident in Table 4–165, the dose to the maximally exposed individual would be expected to remain well within the regulatory limits. Onsite workers would be expected to see an increase of approximately 0.17 latent cancer fatality due to radiation from nuclear infrastructure operations over the 35-year operational period.

Table 4–165 Maximum Cumulative Radiation Impacts at ORR

Impact	Maximally Exposed Individual		Population Dose Within 80 Kilometers (50 Miles) (Year 2020)		Total Site Workforce	
	Annual Dose (millirem per year)	Risk of a Latent Cancer Fatality[a]	Dose (person-rem)	Number of Latent Cancer Fatalities[a]	Dose (person-rem per year)	Number of Latent Cancer Fatalities[a]
Existing site activities[b]	4.4	7.7×10^{-5}	60.3	1.1	103	1.4
HEU disposition	0.039	6.8×10^{-7}	0.16	0.0028	11	0.16
Stockpile stewardship and management	0.2	3.5×10^{-6}	0.6	0.011	-1.8	-0.025
Waste management	0.35	6.1×10^{-6}	1.2	0.021	0.45	0.0063
New nuclear infrastructure operations at ORR[c]	1.9×10^{-6}	3.3×10^{-11}	8.8×10^{-5}	1.5×10^{-6}	12	0.168
Total	5.0[d]	8.7×10^{-5}[(d)]	62	1.1	125	1.7

a. These values are calculated based on a 35-year exposure period.
b. Environmental impacts associated with present activities at ORR that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Impacts are bounded by Option 7 of Alternative 2.
d. The same individual would not be expected to be the maximally exposed individual for all activities at ORR. The location of the maximally exposed individual depends upon where on the site an activity is performed. However, to provide an upper bound of the cumulative impacts to the maximally exposed individual, the impacts from each activity have been summed.

Source: (1) Hamilton et al. 1999:chap. 6 for impacts to the public from existing site activities and DOE 1999k:table B-1c for impacts to the site workforce from existing site activities; (2) DOE 1996a:chap. 4 and DOE 1997a:chap. 11 for all impacts associated with each of the other activities listed in the first column, except nuclear infrastructure operations; and (3) Section 4.4.7.1.9 of this NI PEIS for all impacts associated with nuclear infrastructure operations. Impacts presented in the source documents have been adjusted, as appropriate, to reflect the Records of Decision for waste management, as discussed in Chapter 3.

4.8.1.4 Waste Management

Cumulative amounts of waste generated at ORR are presented in **Table 4–166**. It is unlikely that there would be major impacts on waste management at ORR because sufficient capacity would exist to manage the site wastes. As discussed in Section 4.3.1.1.13, irrespective of how the waste from processing irradiated neptunium-237 targets is classified (i.e., transuranic or high-level radioactive), the waste composition and characteristics are the same, and the management (i.e., treatment and onsite storage), as described in this NI PEIS, would be the same. In addition, either waste type would require disposal in a suitable repository. None of the options assessed in this NI PEIS would generate more than a small amount of additional waste at ORR.

4.8.2 Cumulative Impacts at INEEL

For INEEL, the bounding option for this NI PEIS is Option 2 of Alternative 2. This option calls for the operation of ATR to irradiate neptunium-237 targets and the operation of FDPF to fabricate and process these targets. The impacts associated with ATR and FDPF operations for other missions are included under "existing site activities."

Table 4–166 Cumulative Impacts on Waste Management Activities at ORR Over 35-Year Period (cubic meters)

Waste Type	Existing Site Activities[a]	Treatment and Shipment of Transuranic Waste[b]	Surplus Plutonium Disposition[c]	Spallation Neutron Source[d]	New Nuclear Infra-structure Operations[e]	Total	Site Capacity[f]		
							Treatment (cubic meters/ year)	Storage	Disposal
Transuranic (High-level radioactive)[g]	766 (0)	607 (0)	11 (0)	0 (0)	385 (385)	1,769 (385)	4,050/5 years (0)	2,845 (0)	NA (NA)
Low-level radioactive	335,755	2,778	140	612,000	<2,145	~952,818	440,405	87,776	NA
Mixed low-level radioactive	28,035	23	1	623	<175	~28,857	263,560	234,226	NA
Hazardous[h] (kilograms)	1,260,000	0	1	1,435,000	227,500	2,922,501	1,738,803	7,312	NA
Nonhazardous									
Liquid	23,845,500	1,560	1,500	2,415	99,925	23,950,900	3,395,918	NA	NA
Solid	2,590,000	5,500	130	47,215	5,180	2,648,025	NA	NA	1,219,000

a. Total 35-year waste generation estimate was derived from annual waste generation rates based on historical data obtained from Wham 1999c.
b. Data from the *Final Environmental Impact Statement for Treating Transuranic (TRU)/Alpha Low-Level Waste at the Oak Ridge National Laboratory* (DOE 2000c:S-31) Low-Temperature Drying Alternative and selected in the Record of Decision (65 FR 48683).
c. Data from the *Surplus Plutonium Disposition Final EIS* (DOE 1999a: 4-394) postirradiation examination (2006 through 2009) and selected in Record of Decision (65 FR 1608).
d. Data from the *Spallation Neutron Source Final EIS* (DOE 1999e:3-31).
e. Option 7 of Alternative 2. This alternative would generate the most waste for all waste types.
f. Total 35-year and annual capacity derived from Table 3–13.
g. Refer to Section 4.3.1.1.13 for a discussion on the classification of waste from processing irradiated neptunium-237 targets. Volumes in parentheses represent high-level radioactive waste.
h. Assumes for hazardous waste that 353 kilograms equal 1 cubic meter (22 pounds equal 1 cubic foot).

Note: To convert from cubic meters to cubic yards, multiply by1.308; < means "less than"; ~ means "approximately;" NA, not applicable (i.e., the majority of the waste is not routinely treated, stored, or disposed of on site).
Source: DOE 1999a:4-394; 65 FR 1608; DOE 1999e:3-31; DOE 2000c:S-31; 65 FR 48683; Wham 1999c; Sections 4.3.1.1.13 and 4.4.7.1.13 of this NI PEIS.

4.8.2.1 Resource Requirements

Cumulative impacts on resource requirements at INEEL are presented in **Table 4–167**. INEEL would remain within its site capacity for all major resources. If Option 2 of Alternative 2 were implemented, the proposed nuclear infrastructure facilities would require essentially no change in the site's use of electricity or water.

Cumulatively, INEEL would use 80 percent of its electrical capacity and 13 percent of its water capacity. Site employment would increase by approximately 24 workers.

Table 4–167 Maximum Cumulative Resource Use and Impacts at INEEL

Activities[a]	Site Employment	Electrical Consumption (megawatt-hours per year)	Water Usage (million liters per year)
Existing site activities[b]	7,993	232,500	4,830
SNF Management and INEL Environmental Restoration and Waste Management	–	2,200	2
Foreign Research Reactor SNF Management	–	1,000	2
Waste Management PEIS	–	13,980	194
Advanced Mixed Waste Treatment Project	–	33,000	16
High-Level Waste and Facilities Disposition	–	33,000	351
New nuclear infrastructure operations[c]	24[d]	Negligible[e]	1.68
Total	8,017	~315,680	5,397
Total site capacity	NA	394,200	43,000

a. See Section 4.8 and Table 4–162 for descriptions of past, present, and reasonably foreseeable actions considered.
b. Reflects current sitewide activities (except that "Site Employment" value also reflects projected employment from other activities) that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Nuclear infrastructure activities from Alternative 2, Option 2.
d. Some, or all, of those worker requirements may be filled by the reassignment of the existing workforce.
e. Additional electricity consumption associated with this option would be negligible compared to that associated with existing facility activities.

Note: To convert from liters per year to gallons per year, multiply by 0.264; to convert from megawatt-hours to British thermal units, multiply by 3.42×10^6; ~ means "approximately," and indicates that new nuclear infrastructure operations would contribute only minimally.

Key: NA, not applicable; SNF, spent nuclear fuel.

Source: DOE 1995c:K-16; 1996c:4-53, 4-54, F-164; 1997a:6-18, 6-32, 6-45, 6-55, 6-67; 1999a:3-85, 4-379; 1999c:5.13-2; 1999g:5-86; French, Tallman, and Taylor 1999:v; Sections 4.4.2.1.4 and 4.4.2.1.8 of this NI PEIS.

4.8.2.2 Air Quality

Cumulative impacts on air quality at INEEL are presented in **Table 4–168**. INEEL is currently in compliance with all Federal and state ambient air quality standards, and would continue to remain in compliance, even with consideration of the cumulative effects of all activities. The contributions of nuclear infrastructure operations to overall site concentrations are expected to be very small.

Table 4–168 Maximum Cumulative Air Pollutant Concentrations at INEEL for Comparison with Ambient Air Quality Standards

Parameter	Carbon Monoxide		Nitrogen Dioxide	PM_{10}		Sulfur Dioxide		
Averaging Period	8 hours	1 hour	Annual	Annual	24 hours	Annual	24 hours	3 hours
Activities								
Existing site activities[a] (micrograms per cubic meter)	78	206	0.46	0.49	12	0.14	5.3	24
ANL–W contribution[b] (micrograms per cubic meter)	41	59	13	0.14	1.1	3.3	27	60
Advanced Mixed Waste Treatment Project[c] (micrograms per cubic meter)	0.85	115	0.34	0.006	4.6	0.012	4.5	25
HLW & FD[d] (micrograms per cubic meter)	4.2	10	0.19	0.02	0.28	0.57	8.9	42
New nuclear infrastructure operations[e] (micrograms per cubic meter)	0	0	3.66×10^{-4}	0	0	0.024	0.19	0.43
Total concentration (micrograms per cubic meter)	124	390	14	0.656	18	4.05	45.9	151
Standard								
Most stringent standard[f] (micrograms per cubic meter)	10,000	40,000	100	50	150	80	365	1,300

a. Environmental impacts associated with existing site activities (excluding activities at ANL–W) as shown in the *Idaho High-Level Waste and Facilities Disposition Draft EIS*, Table C.2-14 (DOE 1999g) and in the *Final EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*, Table 3–2 (DOE 2000b). The activities whose concentrations are given in this row, are anticipated to continue during part or all of the 35-year period evaluated for proposed nuclear infrastructure operations.
b. The contribution from existing ANL–W sources as shown the *Final EIS for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel*, Table 3–2 (DOE 2000b).
c. *Advanced Mixed Waste Treatment Project Final EIS* activities—proposed action with microencapsulation or vitrification, Table 5.7-6 (DOE 1999c).
d. High-level waste and facilities disposition site boundary contribution for planning basis option, Table C.2-14 (DOE 1999g).
e. Nuclear infrastructure activities from Alternative 2, Option 2.
f. The more stringent of the Federal and state standards is presented if both exist for the averaging period.

Key: HLW & FD = high-level waste and facilities disposition.
Source: DOE 1999c:5.7-15, 1999g, 2000b:sec. 4.11; modeled results for nuclear infrastructure operations are based on the SCREEN3 computer code (EPA 1995); Section 4.4.2.1.3 of this NI PEIS.

4.8.2.3 Public and Occupational Health and Safety—Normal Operations

Cumulative impacts in terms of radiation exposure to the public and workers at INEEL are presented in **Table 4–169**. There would be no increase expected in the number of latent cancer fatalities in the population from INEEL site operations if nuclear infrastructure operations were to occur at ATR and FDPF. The dose limits for individual members of the public are given in DOE Order 5400.5. As discussed in that order, the dose limit from airborne emissions is 10 millirem per year, as required by the Clean Air Act; the dose limit from drinking water is 4 millirem per year, as required by the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. Therefore, as is evident in Table 4–169, the dose to the maximally exposed individual would be expected to remain well within the regulatory limits. Onsite workers would be expected to see an increase of approximately 0.17 latent cancer fatality due to radiation from nuclear infrastructure operations over the 35-year operational period.

Table 4–169 Maximum Cumulative Radiation Impacts at INEEL

Impact	Maximally Exposed Individual		Population Dose Within 80 Kilometers (50 Miles) (Year 2020)		Total Site Workforce	
	Annual Dose (millirem per year)	Risk of a Latent Cancer Fatality[a]	Dose (person-rem)	Number of Latent Cancer Fatalities[a]	Dose (person-rem per year)	Number of Latent Cancer Fatalities[a]
Existing site activities[b]	0.008	1.7×10^{-7}	0.075	0.0013	64.9	0.91
Storage and disposition	1.6×10^{-6}	2.8×10^{-11}	1.8×10^{-5}	3.2×10^{-7}	25	0.35
Foreign research reactor spent nuclear fuel	5.6×10^{-4}	9.8×10^{-9}	0.0045	7.9×10^{-5}	33	0.46
Spent nuclear fuel	0.008	1.4×10^{-7}	0.19	0.0033	5.4	0.076
Advanced Mixed Waste Treatment Project	0.022	3.9×10^{-7}	0.009	1.6×10^{-4}	4.1	0.057
High-level waste and facilities disposition	0.002	3.5×10^{-8}	0.10	0.0018	59	0.83
Sodium-bonded spent nuclear fuel	0.002	3.5×10^{-8}	0.012	2.1×10^{-4}	22	0.31
New nuclear infrastructure operations at ATR and FDPF[c]	2.6×10^{-7}	4.6×10^{-12}	3.9×10^{-6}	6.8×10^{-8}	12	0.17
Total	0.043[d]	7.4×10^{-7}[(d)]	0.39	0.0068	225.4	3.16

a. These values are calculated based on a 35-year exposure period.
b. Environmental impacts associated with present activities at INEEL that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Impacts are bounded by Option 2 of Alternative 2.
d. The same individual would not be expected to be the maximally exposed individual for all activities at INEEL. The location of the maximally exposed individual depends upon where on the site an activity is performed. However, to provide an upper bound of the cumulative impacts to the maximally exposed individual, the impacts from each activity have been summed.

Source: (1) Saffle et al. 2000 for impacts to the public from existing site activities and DOE 1999k for impacts to the workforce from existing site activities; (2) DOE 1995c and DOE 1996a for all impacts associated with storage and disposition activities, foreign research reactor spent nuclear fuel activities, spent nuclear fuel activities, advanced mixed waste treatment activities, and high-level waste activities; (3) DOE 2000b for all impacts associated with sodium-bonded spent nuclear fuel activities; and (4) Section 4.4.2.1.9 of this NI PEIS for all impacts associated with infrastructure operations.

4.8.2.4 Waste Management

Cumulative amounts of waste generated at INEEL are presented in **Table 4–170**. It is unlikely that there would be major impacts on waste management at INEEL because sufficient capacity would exist to manage the site waste. As discussed in Section 4.3.2.1.13, irrespective of how the waste from processing of irradiated neptunium-237 targets is classified (i.e., transuranic or high-level radioactive), the waste composition and characteristics are the same, and the management (i.e., treatment and onsite storage), as described in this NI PEIS, would be the same. In addition, either waste type would require disposal in a suitable repository. None of the alternatives assessed in this NI PEIS would generate more than a small amount of additional waste at INEEL.

Table 4–170 Cumulative Impacts on Waste Management Activities at INEEL Over 35-Year Period (cubic meters)

Waste Type	Existing Site Activities[a]	Idaho HLW and Facility Disposition EIS[b]	Treatment and Management of Sodium-Bonded SNF[c]	New Nuclear Infrastructure Operations[d]	Total	Site Capacity[e]		
						Treatment (cubic meters/ year)	Storage	Disposal (cubic meter/ year)
Transuranic (High-level[f] radioactive)	65,000[g] (0)	110 (0)	14 (0)	245 (245)	65,369 (245)	57,794 (6,434)	190,319 (19,483)	NA (NA)
Low-level radioactive	135,600	15,325	862	<2,320	~154,107	42,363	177,493	69,530
Mixed low-level radioactive	3,767	12,837	40	<175	~16,819	157,092	187,761	NA
Hazardous	1,180	2,457	0	227,500 kilograms (644 cubic meters)[h]	4,281	NA	9,619	NA
Nonhazardous	124,905	145,262	4,960	64,015	339,142	3,200,000	NA	3,062,000

a. DOE 2000b:table 4–67 and figures 5.4–1 through 5.4–3 and input values for those figures through year 2035.
b. DOE 2000b:table 4–67, Separations Alternative. Maximum quantities for any alternative.
c. DOE 2000b:table 4–18, Alternative 1, electrometallurgically treat blanket and driver fuel at ANL–W; 12 years of operation and selected in the Record of Decision (65 FR 56565).
d. Option 2 of Alternative 2 would generate the most waste for all waste types.
e. Total 35-year and annual capacity derived from Table 3–27.
f. Refer to Section 4.3.2.1.13 for a discussion on the classification of waste from processing irradiated neptunium-237 targets. Volumes in parentheses represent high-level radioactive waste.
g. This 65,000 cubic meters is in storage at the Radioactive Waste Management Complex.
h. Assumes for hazardous waste that 353 kilograms equals 1 cubic meter (22.0 pounds equals 1 cubic foot).

Note: To convert from cubic meters to cubic yards, multiply by 1.308; HLW means high-level radioactive waste; SNF means spent nuclear fuel; < means "less than"; ~ means "approximately;" NA, not applicable (i.e., the majority of the waste is not routinely treated, stored, or disposed of on site).
Source: DOE 2000b:table 4–67 and figures 5.4–1 through 5.4–3 and input values for those figures through year 2035; Sections 4.3.2.1.13 and 4.4.1.1.13 of this NI PEIS.

4.8.3 Cumulative Impacts at Hanford

For Hanford, the bounding option for this NI PEIS depends on the parameter assessed. For example, under Public and Occupational Health and Safety, the highest radiological doses and associated latent cancer fatalities to the public would be associated with Option 1 of Alternative 1, whereas the highest doses and latent cancer fatalities to workers would be associated with Option 3 of this same alternative. Processing of targets in RPL versus processing in FMEF accounts for there being different bounding options (refer to Tables 4–41, 4–42, 4–17, and 4–18). For each of the parameters addressed in this section, a footnote is included in each of the cumulative impact tables, as necessary, to indicate the bounding alternative/option.

4.8.3.1 Resource Requirements

Cumulative impacts on resource requirements at Hanford are presented in **Table 4–171**. Hanford would remain within its site capacity for all major resources. If any of the options under Alternative 1 were implemented, the proposed nuclear infrastructure facilities would require a small increase in the site's use of electricity and water. For the bounding options identified in Table 4–171, this would reflect an increase of about 2 and 1 percent, respectively, over current baseline utilization for these resources. There would be no additional land disturbance or development. Cumulatively, Hanford would use approximately 23 percent of its electrical capacity and 38 percent of its water capacity. Site employment would increase by approximately 130 workers.

Table 4–171 Maximum Cumulative Resource Use and Impacts at Hanford

Activities[a]	Site Employment	Electrical Consumption (megawatt-hours per year)	Water Usage (million liters per year)
Existing site activities[b]	16,005	323,128	2,754[c]
Tank Waste Remediation System	–	170,000	200
Waste Management PEIS	–	13,920	133
New nuclear infrastructure operations[d]	130[e]	55,000	80
Total	16,135	562,048	3,167
Total site capacity	NA	2,484,336	8,263[c]

a. See Section 4.8 and Table 4–162 for descriptions of past, present, and reasonably foreseeable actions considered.
b. Reflects current sitewide activities. The "Site Employment" value also reflects projected employment from other activities that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Reflects domestic/potable water only and not raw water usage or availability.
d. Electrical consumption and water usage are bounded by Option 3 or 6 of Alternative 1, with the values reflecting the increase over standby operations from restart of FFTF and associated support activities in FMEF.
e. Some, or all, of these worker requirements may be filled by the reassignment of the existing site workforce.

Note: To convert from liters per year to gallons per year, multiply by 0.264; to convert from megawatt-hours to British thermal units, multiply by 3.42×10^6.
Key: NA, not applicable.
Source: DOE 1996e:5-284; 1997a:5-18, 5-32, 5-45, 5-55, 5-67; 1999a:3-45, 4-376; Sections 4.3.3.1.4 and 4.3.3.1.8 of this NI PEIS.

4.8.3.2 Air Quality

Cumulative impacts on air quality at Hanford are presented in **Table 4–172**. Hanford is currently in compliance with all Federal and state ambient air quality standards, and would continue to be in compliance even with consideration of the cumulative effects of all activities. The nuclear infrastructure contributions to overall site concentrations are expected to be very small.

4.8.3.3 Public and Occupational Health and Safety—Normal Operations

Cumulative impacts in terms of radiation exposure to the public and workers at Hanford are presented in **Table 4–173**. There would be no increase expected in the number of latent cancer fatalities in the population from Hanford site operations if nuclear infrastructure operations were to occur at FMEF. The dose limits for individual members of the public are given in DOE Order 5400.5. As discussed in that order, the dose limit from airborne emissions is 10 millirem per year, as required by the Clean Air Act; the dose limit from drinking water is 4 millirem per year, as required by the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. Therefore, as is evident in Table 4–173, the dose to the maximally exposed individual would be expected to remain well within the regulatory limits. Onsite workers would be expected to see an increase of approximately 0.26 latent cancer fatality due to radiation from nuclear infrastructure operations over the 35-year operational period.

Radiation doses listed under site activities in Table 4–173 include exposures due to activities associated with waste management (as estimated in the *Hanford Comprehensive Land-Use Plan* (DOE 1999d), the tank waste remediation system (DOE 1996e), management of spent nuclear fuel from the K Basins (DOE 1996h), disposal of decommissioned naval reactor plants (Navy 1996), and the Plutonium Finishing Plant Stabilization (DOE 1996g).

Table 4–172 Maximum Cumulative Air Pollutant Concentrations at Hanford for Comparison with Ambient Air Quality Standards

Parameter	Carbon Monoxide		Nitrogen Dioxide	PM_{10}		Sulfur Dioxide			
Averaging Period	8 hours	1 hour	Annual	Annual	24 hours	Annual	24 hours	3 hours	1 hour
Activities									
Existing site activities[a] (micrograms per cubic meter)	27.3	63.3	0.666	0.0182	1.01	0.175	30.17	69.4	79.4
Hanford tank waste remediation[b] (micrograms per cubic meter)	34	48	0.12	0.0079	0.75	0.020	1.6	3.6	4
Spent nuclear fuel management[c] (micrograms per cubic meter)	0	0	0.1	0	0	0	0	0	0
New nuclear infrastructure FFTF operations[d] (micrograms per cubic meter)	52.1	74.4	0.0118	8.39×10^{-4}	9.84	0.00785	9.11	20.5	22.8
New nuclear infrastructure FMEF operations[d] (micrograms per cubic meter)	0	0	4.43×10^{-5}	0	0	0.0087	0.069	0.16	0.17
Total concentration (micrograms per cubic meter)	113.4	185.7	0.90	0.027	11.6	0.212	40.9	93.7	106
Standard									
Most stringent standard[e] (micrograms per cubic meter)	10,000	40,000	100	50	150	50	260	1,300	660

a. Environmental impacts associated with existing site activities as described in Section 3.4.3 of this NI PEIS. The values listed are the summed values presented in Table 3–29. These activities are anticipated to continue during part or all of the 35-year period evaluated for proposed nuclear infrastructure operations.
b. *Hanford Tank Waste Remediation Final EIS* activities, vitrification facilities from Table 5.3-1, Phased Implementation – Phase II Operation (DOE 1996e).
c. *Spent Nuclear Fuel Management* – regionalization alternative (DOE 1995c:vol. 1, app. A, p. 5-43).
d. Nuclear infrastructure contributions are bounded by Alternative 1, Option 3. Periodic testing of emergency diesel generators would result in higher values for certain pollutants and time periods (Section 4.3.3.1.3).
e. The more stringent of the Federal and State standards is presented if both exist for the averaging period.

Note: The contribution from activities in the *Final Waste Management Programmatic EIS* (DOE 1997a) are small and are not shown.
Source: DOE 1995c, 1996e, 1997a; Wisness 2000; modeled results for nuclear infrastructure operations are based on the SCREEN3 computer code (EPA 1995); Sections 3.4.3 and 4.3.3.1.3 of this NI PEIS.

Table 4–173 Maximum Cumulative Radiation Impacts at Hanford

Impact	Maximally Exposed Individual		Population Dose Within 80 Kilometers (50 Miles) (Year 2020)		Total Site Workforce	
	Annual Dose (millirem per year)	Risk of a Latent Cancer Fatality[a]	Dose (person-rem)	Number of Latent Cancer Fatalities[a]	Dose (person-rem per year)	Number of Latent Cancer Fatalities[a]
Existing site activities[b]	0.02	3.5×10^{-7}	0.60	0.011	181	2.5
Waste management	0.0057	2.9×10^{-9}	0.28	0.0014	1,300	5.2
Tank remediation	(c)	2.4×10^{-6}	(c)	0.19	(c)	3.27
Spent nuclear fuel management	(c)	1.4×10^{-8}	(c)	8.0×10^{-4}	(c)	0.057
Burial of low-level waste	0	0	0	0	1,018	0.41
Plutonium Finishing Plant stabilization	0.13	3.9×10^{-7}	2.3	0.007	157	0.38
New nuclear infrastructure operations at FFTF and FMEF or RPL[d]	0.0054	9.5×10^{-8}	0.25	0.0044	18	0.26
Total	(e)	$3.3\times10^{-6(f)}$	(e)	0.21	(e)	12

a. These values are calculated based on a 35-year exposure period except for waste management (project duration for waste transfer of 10 years) and Plutonium Finishing Plant stabilization (a 6-year project).
b. Environmental impacts associated with present activities at Hanford (including activities at other non-DOE facilities at, or near, Hanford) that are anticipated to continue during all or part of the 35-year period evaluated for proposed nuclear infrastructure operations.
c. Source document provides project total; annual values are not constant.
d. Impacts on the public are bounded by Option 1 of Alternative 1; impacts on workers are bounded by Option 3 of Alternative 1.
e. Some source documents did not provide dose values, only expected latent cancer fatalities. Therefore, total dose estimates have not been developed.
f. The same individual would not be expected to be the maximally exposed individual for all activities at Hanford. The location of the maximally exposed individual depends upon where an activity is performed on the site. However, to provide an upper bound cumulative impact for the maximally exposed individual, the impacts from each activity have been summed.

Source: (1) Dirkes, Hanf, and Poston 1999:chap. 5 and DOE 1997a for impacts to the public from existing site activities and DOE 1999k:table B-1c for impacts to the workforce from existing site activities; (2) DOE 1996a:chap. 4, 1996e:table 5.11.1, 1996g:table 3-12a, 1997a:table 11.6-2, 1999d:sec. 5.6.4; Navy 1996:sec. 4.3.3.5 for impacts associated with each of the other activities listed in the first column, except nuclear infrastructure operations; and (3) Sections 4.3.1.1.9 and 4.3.3.1.9 of this NI PEIS for all impacts associated with nuclear infrastructure operations.

4.8.3.4 Waste Management

Cumulative amounts of waste generated at Hanford are presented in **Table 4–174**. It is unlikely that there would be major impacts on waste management at Hanford because sufficient capacity would exist to manage the site waste. As discussed in Sections 4.3.3.1.13 and 4.4.3.1.13, irrespective of how the waste from processing of irradiated neptunium-237 targets is classified (i.e., transuranic or high-level radioactive), the waste composition and characteristics are the same, and the management (i.e., treatment and onsite storage), as described in this NI PEIS, would be the same. In addition, either waste type would require disposal in a suitable repository. None of the alternatives assessed in this NI PEIS would generate more than a relatively small amount of additional waste at Hanford.

Table 4–174 Cumulative Impacts on Waste Management Activities at Hanford Over 35-Year Period (cubic meters)

Waste Type	Existing Site Activities[a]	New Nuclear Infrastructure Operations	Total	Site Capacity[b]		
				Treatment (cubic meters/ year)	Storage	Disposal
Transuranic (High-level radioactive)[c]	9,880 (0)	385[d] (385)	10,265 (385)	98,520 (50,000)	17,216 (146,000)	NA (NA)
Low-level radioactive	95,666	5,015[d]	100,681	398,112	99,910	1,970,000
Mixed low-level radioactive	46,207	315[d]	46,522	413,211	100,483	14,200
Hazardous	19,600	3,100[e]	22,700	NA	NA	NA
Nonhazardous						
Liquid	7,000,000	1,494,500[d]	8,494,500	120,000	NA	4,807,720
Solid	1,505,000	10,500[d]	1,515,500	NA	NA	NA

a. Total 35-year waste generation derived from DOE 1999h, except for hazardous and nonhazardous waste, which were derived from Table 3–34.
b. Total 35-year and annual capacity derived from Table 3–36.
c. Refer to Section 4.3.3.1.13 and 4.4.3.1.13 for a discussion on the classification of waste from processing irradiated neptunium-237 targets. Volumes in parentheses represent high-level radioactive waste.
d. The bounding alternative for this waste type is Alternative 1, Option 3 or 6.
e. The bounding alternative for this waste type is Alternative 2, Option 3, 6, or 9; Alternative 3, Option 3; or Alternative 4, Option 3; which all include the deactivation of FFTF and neptunium-237 target fabrication and processing at FMEF. The inventory of bulk metallic sodium (Section 4.4.1.2.13) is not included because alternative sponsors and/or users will be found for its disposition.

Note: To convert from cubic meters to cubic yards, multiply by1.308.
Source: DOE 1996a; DOE 1999h; Sections 4.3.3.1.13 and 4.4.1.2.13 of this NI PEIS.

4.8.3.5 Spent Nuclear Fuel Management

As discussed in Section 4.3.1.1.14, the operation of FFTF for the proposed activities at 100 megawatts for 35 years under Alternative 1 would produce a total of about 16 metric tons of heavy metal (35,200 pounds) of spent nuclear fuel. The existing spent nuclear fuel at Hanford is about 2,133 metric tons of heavy metal (4,700,000 pounds) (DOE 1995c). The environmental impacts associated with the existing spent nuclear fuel management at Hanford are addressed in the following documents:

- DOE 1996a—*Storage and Disposition of Weapons-Usable Fissile Materials Fuel Programmatic Environmental Impact Statement*
- DOE 1995c—*Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho Nuclear Engineering Laboratory Environmental Restoration and Waste Management Final Environmental Impact Statement*
- DOE 1997c—*Environmental Assessment - Management of Hanford Site, Non-Defense Production Reactor Spent Nuclear Fuel*
- DOE 1996h—*Final Environmental Impact Statement, Management of Spent Nuclear Fuel from the K Basins at the Hanford Site*

The management of the existing spent fuel at Hanford results in a dose of less than 0.1 millirem per year to the maximally exposed member of the public. This dose is well within the DOE dose limits cited in

Section 4.8.3.3. DOE has committed to remove the spent fuel at Hanford for ultimate disposition in a geologic repository. The restart of FFTF under Alternative 1 would generate 16 metric tons of heavy metal of spent nuclear fuel, which is less than 1 weight-percent of the total spent nuclear fuel inventory presently at Hanford. Only a small fraction of the dose shown for nuclear infrastructure operations would be attributable to the management of this spent fuel at FFTF. The doses at Hanford, including those associated with spent nuclear fuel management, would remain within the DOE dose limits.

4.8.4 Cumulative Impacts at the Generic CLWR Site

No incremental environmental impacts at the generic site would be expected from the normal operation of a CLWR to irradiate targets (refer to Sections 4.4.4 through 4.4.6). Therefore, the cumulative impacts at the generic CLWR site would not be affected by any action assessed in this NI PEIS, and are not addressed further.

4.8.5 Cumulative Impacts at the New Accelerator(s) Generic DOE Site

Cumulative impacts cannot be presented for a generic site. If Alternative 3 were selected for implementation, a subsequent site-specific analysis would be conducted for the DOE site chosen for the combination of new accelerator(s) and support facility (refer to Section 4.5), and appropriate NEPA documentation would be prepared to address the cumulative impacts for that site.

4.8.6 Cumulative Impacts at the New Research Reactor Generic DOE Site

Cumulative impacts cannot be presented for a generic site. If Alternative 4 were selected for implementation, a subsequent site-specific analysis would be conducted for the DOE site chosen for the new research reactor and support facility or research reactor only (refer to Section 4.6), and appropriate NEPA documentation would be prepared to address the cumulative impacts for that site.

4.8.7 Cumulative Impacts of Transportation

The cumulative impacts from transportation associated with nuclear infrastructure activities are identified in Appendix J. Because likely transportation routes cross many states, cumulative impacts are compared on a national basis. Under all alternatives assessed in this NI PEIS, occupational radiation exposure to transportation workers and exposure to the public are estimated to each represent less than 0.05 percent of the cumulative exposures from nationwide transportation (DOE 1999i) over the 35-year period of nuclear infrastructure activities. No additional traffic fatality is expected; the incremental increase in traffic fatalities would be less than 0.0001 percent per year.

4.9 MITIGATION MEASURES

As shown throughout Chapter 4, the impacts of all the alternatives would be small. No specific mitigation measures would be necessary because all potential environmental impacts would be below acceptable levels or applicable standards.

Nevertheless, DOE would maintain all public and worker exposures, both direct exposures and indirect exposures via airborne emissions, as low as is reasonably achievable. This is a long-standing DOE policy to control or manage radiation exposures and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. This DOE policy is not a dose limit but rather a process that has as its objective the attainment of dose levels as far below the applicable limits as practical.

Similarly, DOE has a long-standing policy to minimize waste generation. Thus, DOE would conduct all operations in a manner that generates the smallest amount of waste practical. This policy applies to all types of waste, including solid and liquid radioactive waste, hazardous waste, and mixed waste.

Alternatives 3 and 4 involve the construction of major new facilities. In these alternatives, DOE would employ modern construction practices that minimize the environmental impacts.

4.10 UNAVOIDABLE ADVERSE ENVIRONMENTAL IMPACTS

Regardless of the option selected by DOE, there may be some associated unavoidable adverse environmental impacts, although the impacts would be small. Some health risks to workers and the public would be unavoidable were the option implemented. Workers at operating sites and involved in truck shipments would be subject to the same types and frequencies of injuries and accidental deaths that workers experience across the industrial sector of the nation. Workers would also be exposed to the specific health risks of exposures to radiation and hazardous chemicals. The public would generally be at a lower risk than any of the workers involved in processing activities. During processing operations, air quality would be unavoidably affected as the result of criteria and hazardous and toxic air pollutant emissions at the site, and from worker vehicles and truck shipment vehicles. Nonradiological air quality impacts at any particular site are not expected to affect attainment status of the site's air quality control region for each criteria air pollutant.

Construction activities associated with several options (including construction of a new 76-meter (250-foot) stack at Hanford, one or two new DOE accelerators and support facility, and a new DOE nuclear research reactor and support facility) would result in short term elevated levels of particulate matter in localized areas. In addition, portions of nonsensitive terrestrial habitats would be lost when these new facilities were constructed. None of these habitat losses is expected to constitute a significant impact on the resident plant and animal species because these species have broad ranges and the amount of lost habitat would comprise only a small fraction of these communities.

If an alternative were selected which involved a DOE site not yet selected for implementation (generic DOE site), the issue of unavoidable adverse environmental impacts would be assessed as part of the site selection process. Appropriate NEPA documentation would be prepared.

4.11 RELATIONSHIP BETWEEN SHORT-TERM USE OF THE ENVIRONMENT AND LONG-TERM PRODUCTIVITY

Short-term environmental impacts are those that would occur during construction and operation of target fabrication, storage, irradiation, or processing facilities. Impacts that extend beyond the period of facility operations are considered to be long-term impacts.

The implementation of any of the options assessed in this NI PEIS would result either in the short-term use of existing facilities and environmental resources, or in the construction of new facilities and their operation and ultimate decontamination and decommissioning. Facility modifications would be required for the implementation of neptunium-237 target fabrication and postirradiation plutonium-238 processing technologies at ORR, INEEL, and Hanford. In addition, facility modifications would be required at Hanford to support target fabrication and postirradiation processing for civilian nuclear energy research and industrial and medical isotope production in FFTF and perhaps at other DOE sites for targets irradiated in either one or two new DOE accelerators or a new DOE research reactor. Some new target fabrication and postirradiation processing facilities might be required to support targets irradiated in either one or two new DOE accelerators or a new DOE research reactor.

The implementation of options that require construction of new facilities would require short-term use of the environment and a variety of resources such as land, construction materials, and labor. Development of these facilities would commit lands to those uses from the beginning of the construction period through the duration of the operation period and until such facility would be fully decommissioned. Depending upon the specific locations at sites selected for these facilities, some terrestrial habitat may be lost when the area is cleared for construction. Disturbance of this acreage would eliminate the natural productivity of the land. At the end of the operational period, these facilities could be converted to other uses or decontaminated and decommissioned and the land returned to its original use or a condition compatible with future uses.

Transportation between SRS and the candidate neptunium-237 target fabrication sites; between the candidate sites for neptunium-237 target fabrication and processing and the irradiation sites; between the candidate postirradiation plutonium-238 processing sites and LANL; between an east coast port and Hanford for shipment of mixed oxide fuel; and between a U.S. fuel fabrication facility and Hanford for shipment of highly enriched uranium fuel would occur on existing roadways. Most target fabrication and postirradiation processing for industrial and medical isotopes and for civilian nuclear energy research and development would occur on the same site as the irradiation of those targets; however, there would be air and truck transport of the irradiation products to certain distribution centers. There would also be some transport of targets containing hazardous materials to the irradiation sites. These activities would result in emissions to the atmosphere that would not measurably affect regional or global air quality. Short-term uses of the environment would have no appreciable beneficial or adverse effects on long-term productivity of the environment on, or in the vicinity of, any of the sites assessed in this NI PEIS.

4.12 IRREVERSIBLE AND IRRETRIEVABLE COMMITMENTS OF RESOURCES

This section describes the major irreversible and irretrievable commitments of resources that can be identified at this programmatic level of analysis. A commitment of resources is irreversible when primary or secondary impacts limit the future options for a resource. An irretrievable commitment refers to the use or consumption of resources neither renewable nor recoverable for future use.

Processing activities related to the production of plutonium-238 described in this NI PEIS would be conducted at existing facilities. In addition, the fabrication and processing of targets used to produce industrial and medical isotopes and to conduct civilian nuclear energy research and development using FFTF at Hanford would also be conducted in existing facilities. Modifications to existing facilities would consist of improvements required to meet current environmental standards or the installation of new processing equipment. In addition, the use of FMEF at Hanford would require construction of a 76-meter (250-foot) stack.

The implementation of several alternatives described in this NI PEIS would require the construction of new facilities to fabricate, irradiate, or process targets to produce plutonium-238 for space missions, industrial and medical isotopes, or to conduct civilian nuclear energy research and development. These alternatives would include those using one or two new accelerators or a new research reactor to irradiate the targets. To limit the cost and environmental impacts of these alternatives, DOE would consider modifying existing appropriate facilities at the irradiation sites rather than constructing new facilities.

The land that is currently occupied by either existing or new processing or irradiation facilities could ultimately be returned to open space uses if buildings, roads, and other structures were removed, areas cleaned up, and the land revegetated. Alternatively, the facilities could be modified for use in other DOE programs. The commitment of such land is irreversible in the short term, but not necessarily irreversible in the long term.

The irreversible and irretrievable commitment of material resources during the life-cycles of the activities described in this NI PEIS includes construction materials that cannot be recovered or recycled, materials that are rendered radioactive and cannot be decontaminated, and materials consumed or reduced to unrecoverable forms of waste. Where construction would be necessary, materials required include wood, concrete, sand, gravel, plastics, steel, aluminum, and other metals. These construction resources, except for those that can be recovered and recycled with present technology, would be irretrievably lost. However, none of those identified construction resources is in short supply, and all are readily available in the vicinity of locations being considered for new facilities. Materials required for the processing equipment, utilities, and fuel required for transportation options comprise the irretrievable resources required to implement the various options that use either new or existing facilities. None of the alternatives requires resources that would noticeably affect local or national supplies, or that would noticeably affect the quality of the local or global environment.

4.13 INDUSTRIAL SAFETY

Estimates of potential industrial impacts to workers during construction, irradiation, fabrication, and processing were evaluated based on DOE and Bureau of Labor Statistics data. Impacts are classified into two groups: total recordable cases and fatalities. A recordable case includes work-related death, illness, or injury which resulted in loss of consciousness, restriction of work or motion, transfer to another job, or required medical treatment beyond first aid. The industrial safety evaluation is discussed in more detail in Section I.3.

The average occupational total recordable cases and fatality rates for construction and operation activities are presented in **Table 4–175**.

Table 4–175 Average Occupational Total Recordable Cases and Fatality Rates (per worker-year)

Labor Category	Total Recordable Cases	Fatalities
Construction	0.053	1.39×10^{-4}
Operation	0.033	1.3×10^{-5}

Source: Section I.3.

The expected impacts (both annual and for the duration of the activity) to workers at each facility for construction and operation are presented in **Table 4–176**.

Table 4–176 Industrial Safety Impacts from Construction and Operation

Facility	Estimated Number of Workers	Construction or Operation Duration (years)	Expected Annual Total Recordable Cases	Expected Activity Duration Total Recordable Cases	Annual Fatalities	Activity Duration Fatalities
Construction						
Low-energy accelerator	75	3	4.0	12	0.010	0.030
High-energy accelerator	410	5	22	110	0.057	0.285
New research reactor	160	7	8.5	59.5	0.022	0.154
Operation						
ATR[a]	0	35	–	–	–	–
HFIR[a]	0	35	–	–	–	–
CLWR[a]	0	35	–	–	–	–
FFTF	242	35	8.0	280	0.0031	0.109
Low-energy accelerator	13	35	0.4	14	1.7×10^{-4}	0.00595
High-energy accelerator	225	35	7.4	259	0.0029	0.102
New research reactor	120	35	4.0	140	0.0016	0.056
REDC	116	35	3.8	133	0.0015	0.0525
FDPF	75	35	2.5	87.5	9.8×10^{-4}	0.0343
FMEF	105	35	3.5	123	0.0014	0.049
RPL/306–E	30	35	1.0	35	3.9×10^{-4}	0.0137
New support facility	100	35	3.3	116	0.0013	0.0455

a. No additional workers would be required at these facilities for the proposed activities evaluated in this NI PEIS.
Source: Section I.3.

No fatalities would be expected from either construction or operation of any facility.

4.14 REFERENCES

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Chapter 5
Applicable Laws, Regulations, and Other Requirements

Chapter 5 presents the laws, regulations, and other requirements that apply to the proposed action and alternatives. Federal, state, and U.S. Department of Energy environmental, safety, and health laws, regulations, and Executive and DOE orders are summarized in Section 5.1. Radioactive material packaging and transportation laws and regulations are discussed in Section 5.2. Emergency management and response laws, regulations, and Executive orders are discussed in Section 5.3. Consultations with Federal, state, and local agencies and federally recognized Native American groups are discussed in Section 5.4.

5.1 Environmental, Safety, and Health Laws, Regulations, Executive Orders, and U.S. Department of Energy Orders

There are a number of Federal environmental laws that affect environmental protection, compliance, or consultation at every U.S. Department of Energy (DOE) location and at commercial light water reactors (CLWRs). In addition, certain environmental requirements have been delegated to state authorities for enforcement and implementation. It is DOE policy to conduct its operations in a manner that ensures the protection of public health, safety, and the environment through compliance with all applicable Federal and state laws, regulations, orders, and other requirements.

The various action alternatives analyzed in this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]) involve the operation of existing DOE and commercial facilities, the construction and operation of new DOE facilities, and the transportation of materials. Compliance with statutes, regulations, and other Federal and state requirements may be dependent on whether a facility is newly built (preoperational) or has already been operating, as well as dependent on who owns the facility (i.e., DOE or a private entity). A brief summary of the alternatives is provided below to help the reader understand the statutory, regulatory, and other requirements that are discussed later in this chapter. Chapter 2 provides a more detailed discussion of these alternatives.

- **No Action Alternative** includes maintaining an existing facility (the Fast Flux Text Facility [FFTF]) in standby status and purchasing plutonium-238 from Russia. The continued use of existing DOE facilities to produce medical and industrial isotopes and conduct nuclear research and development activities at current operating levels is also included in this alternative, as well as the storage of neptunium-237.

- **Alternative 1** includes the use of existing DOE facilities to store nuclear materials (the Radiochemical Engineering Development Center [REDC], the Fluorinel Dissolution Process Facility [FDPF], the Fuels and Materials Examination Facility [FMEF], the Hanford Radiochemical Processing Laboratory [RPL/306–E], or the Idaho Nuclear Technology and Engineering Center [INTEC] Building 651 [CPP–651]); fabricate and process targets for plutonium-238 production, medical and industrial isotope production, and nuclear research and development needs (FMEF, REDC, Hanford RPL/306–E, FDPF); and irradiate all targets at FFTF.

- **Alternative 2** includes the use of existing operating DOE facilities to store nuclear materials (REDC, FDPF, CPP–651, or FMEF), fabricate and process neptunium-237 targets (REDC, FDPF, FMEF), and irradiate neptunium-237 targets (the Advanced Test Reactor [ATR], the High Flux Isotope Reactor [HFIR]). In addition to existing DOE facilities, the use of an existing operating CLWR to irradiate neptunium-237 targets was analyzed. The permanent deactivation of an existing DOE facility (FFTF) also is included in this alternative. Under this alternative, existing DOE facilities would

continue medical and industrial isotope production and nuclear research and development activities at current operating levels.

- **Alternative 3** includes the construction and operation of a new support facility at an existing DOE site to fabricate and process medical and industrial targets and nuclear materials for research and development purposes, and one or two new accelerators at an existing DOE site to irradiate these targets and materials, as well as neptunium-237 targets for plutonium-238 production. The use of existing DOE facilities for neptunium-237 target fabrication and processing (REDC, FDPF, FMEF) and materials storage (REDC, FDPF, CPP–651, FMEF) also was analyzed. The decontamination and decommissioning of the new accelerator(s) and support facility and the permanent deactivation of FFTF, an existing DOE facility, are included in this alternative.

- **Alternative 4** includes the construction and operation of a new support facility at an existing DOE site to fabricate and process medical and industrial targets and nuclear materials for research and development purposes, and a new research reactor at an existing DOE site to irradiate these targets and materials, as well as neptunium-237 targets for plutonium-238 production. Neptunium-237 target fabrication and processing at existing DOE facilities (REDC, FDPF, FMEF) and materials storage (REDC, FDPF, CPP–651, FMEF) were analyzed in this alternative as well. The decontamination and decommissioning of the new research reactor and support facility and the permanent deactivation of FFTF, an existing DOE facility, also are included in this alternative.

- **Alternative 5** includes the permanent deactivation of FFTF, an existing DOE facility, with no new missions at existing facilities.

The addition of plutonium-238 production, research and development, and industrial and medical isotope production missions at existing facilities would necessitate few, if any, physical or substantive changes to current activities at these sites. Continued compliance with applicable laws, regulations, and other requirements, including permits and licenses, would be required. Based on projections for air emissions and liquid effluent, no changes to the permits at these existing facilities should be necessary to accommodate the nuclear energy research and development and isotope production missions. Waste generated as a result of the expanded missions would be managed consistent with current site waste management practices and existing permits, agreements, and orders. However, as with any project, it should be noted that regulatory requirements can change over time and may impact current practices, possibly requiring changes or modifications to facility operations and applicable permits or licenses.

The construction and operation of one or two new accelerators or the research reactor and support facility at an existing DOE site would require major physical changes. These new facilities would require the appropriate licenses and permits necessary for construction and operation.

This section describes the environmental, safety, and health laws, regulations, and orders that may apply to the proposed action and alternatives. A more detailed analysis for new facilities (i.e., new accelerator(s) or new research reactor) would be needed if either of these alternatives is selected in the Record of Decision.

5.1.1 Federal Environmental, Safety, and Health Laws and Regulations

National Environmental Policy Act of 1969, as amended (42 United States Code (U.S.C.) 4321 *et seq.*)—The National Environmental Policy Act (NEPA) establishes a national policy promoting awareness of the environmental consequences of human activity on the environment and consideration of environmental impacts during the planning and decision-making stages of a project. It requires Federal

agencies to prepare a detailed environmental impact statement (EIS) for any major Federal action with potentially significant environmental impact.

This NI PEIS has been prepared in accordance with NEPA requirements, the Council on Environmental Quality regulations (40 Code of Federal Regulations (CFR) Part 1500 *et seq.*), and DOE (10 CFR Part 1021, DOE Order 451.1B) provisions for implementing the procedural requirements of NEPA. It discusses reasonable alternatives and their potential environmental consequences.

Atomic Energy Act of 1954 (42 U.S.C. 2011 *et seq.*)—The Atomic Energy Act authorizes DOE to establish standards to protect health or minimize dangers to life or property for activities under DOE's jurisdiction. Through a series of DOE orders, an extensive system of standards and requirements was established to ensure safe operation of DOE facilities. DOE regulations are found in 10 CFR.

The Atomic Energy Act also requires entities that operate nuclear power plants, such as CLWRs, to have a plant license issued by the U.S. Nuclear Regulatory Commission (NRC). The NRC regulations that implement this requirement provide for permits to be issued for the construction or alteration of such facilities. Operating licenses are applied for after completion of the construction or alteration of the facilities. Construction permits and operating licenses include detailed provisions regarding their duration and the design, safety, and quality assurance requirements for the subject facilities. The NRC regulations for permits and operating licenses are found in 10 CFR.

For alternatives involving existing DOE facilities and the construction and operation of one or two new DOE accelerators or a research reactor and support facility, an NRC license is not required and the facilities would need to comply with the appropriate DOE orders. A list of applicable DOE orders is provided in Section 5.1.3.

For the alternative involving the existing generic CLWR, the existing operating license would need to be amended to address neptunium-237 target irradiation. NRC must approve the generic CLWR's license amendment before this plant can irradiate these targets to produce plutonium-238.

For nuclear facilities in the United States, annual exposure limits to the public and radiation workers are established by NRC in 10 CFR Part 20 ("Standards for Protection Against Radiation") and 10 CFR Part 50, Appendix I ("Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluent").

Clean Air Act of 1970, as amended (42 U.S.C. 7401 *et seq.*)—The Clean Air Act is intended to "protect and enhance the quality of the Nation's air resources so as to promote the public health and welfare and the productive capacity of its population." Section 118 of the Clean Air Act (42 U.S.C. 7418) requires that each Federal agency with jurisdiction over any property or facility engaged in any activity that might result in the discharge of air pollutants comply with "all Federal, state, interstate, and local requirements" with regard to the control and abatement of air pollution.

The Clean Air Act: (1) requires the U.S. Environmental Protection Agency (EPA) to establish National Ambient Air Quality Standards as necessary to protect the public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 U.S.C. 7409 *et seq.*); (2) requires establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 U.S.C. 7411); (3) requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 U.S.C. 7470 *et seq.*); and (4) requires specific standards for releases of hazardous air pollutants (including radionuclides) (42 U.S.C. 7412). These standards are implemented

through state implementation plans developed by each state with EPA approval. The Clean Air Act requires sources to meet standards and obtain permits to satisfy these standards.

Emissions of air pollutants are regulated by EPA under 40 CFR Parts 50 through 99. Radionuclide emissions from DOE facilities are regulated under the National Emission Standards for Hazardous Air Pollutants Program under 40 CFR Part 61. Approval to construct a new facility or to modify an existing one may be required by these regulations under 40 CFR Section 61.07. These standards are not applicable to NRC-licensed facilities such as CLWRs. As cited in EPA's final rule (60 Federal Register 46206), compliance with the NRC regulations constitutes compliance with 40 CFR Part 61, Subparts H and I.

EPA also establishes radiation protection standards for members of the public from the general environment and from radioactive materials introduced into the general environment as a result of nuclear fuel cycle operations. These standards, which are applicable to CLWRs, are found in 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations."

For alternatives involving an existing DOE facility or an existing generic CLWR, no amendments to current air permits are expected, nor would approvals be needed to modify an existing facility, as required under 40 CFR Section 61.07. As discussed in the Air Resource sections of Chapter 4, the air quality standards are not expected to be exceeded.

For the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, air permits for construction and operation would need to be obtained.

Clean Water Act of 1972, as amended (33 U.S.C. 1251 *et seq.*)—The Clean Water Act, which amended the Federal Water Pollution Control Act, was enacted to "restore and maintain the chemical, physical, and biological integrity of the Nation's water." The Clean Water Act prohibits the "discharge of toxic pollutants in toxic amounts" to navigable waters of the United States. Section 313 of the Clean Water Act requires all branches of the Federal Government engaged in any activity that might result in a discharge or runoff of pollutants to surface waters to comply with Federal, state, interstate, and local requirements.

The Clean Water Act provides water quality standards for the Nation's waterways, guidelines and limitations for effluent discharges from point-source discharges, and the National Pollutant Discharge Elimination System (NPDES) permit program. The NPDES program is administered by EPA, pursuant to regulations in 40 CFR Part 122 *et seq.* Sections 401 through 405 of the Water Quality Act of 1987 added Section 402(p) to the Clean Water Act requiring that EPA establish regulations for permits for storm-water discharges associated with industrial activities. Storm-water provisions of the NPDES program are set forth at 40 CFR Section 122.26. Permit modifications are required if discharge effluent is altered.

For the alternatives involving an existing DOE facility or an existing generic CLWR, no amendments to NPDES permits are expected. As discussed in the Water Resource sections of Chapter 4, the water quality standards are not expected to be exceeded.

For the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, NPDES permits may be required.

Safe Drinking Water Act of 1974, as amended (42 U.S.C. 300(f) *et seq.*)—The primary objective of the Safe Drinking Water Act is to protect the quality of public drinking water supplies and sources of drinking water. The implementing regulations, administered by EPA unless delegated to states, establish standards applicable to public water systems. These regulations include maximum contaminant levels (including those for radioactivity) in public water systems, which are defined as water systems that have at least 15 service

connections used by year-round residents or regularly serve at least 25 year-round residents. The EPA regulations implementing the Safe Drinking Water Act are found under 40 CFR Parts 100 through 149. For radioactive material, the regulations specify that the average annual concentration of manmade radionuclides in drinking water, as delivered to the user by such a system, shall not produce a dose equivalent to the total body or an internal organ greater than 4 millirem per year beta activity (40 CFR Section 141.16(a)). Other programs established by the Safe Drinking Water Act include the Sole Source Aquifer Program, the Wellhead Protection Program, and the Underground Injection Control Program.

Activities conducted under all of the alternatives must be in compliance with the standards specified under the Safe Drinking Water Act, particularly the standard for radionuclides. As discussed in the Water Resources sections of Chapter 4, these standards are not expected to be exceeded.

Low-Level Radioactive Waste Policy Act of 1980, as amended (42 U.S.C. 2021 *et seq.*)—This act amended the Atomic Energy Act to specify that the Federal Government is responsible for disposal of low-level radioactive waste generated by its activities, and states are responsible for disposal of other low-level radioactive waste. It provides for and encourages interstate compacts to carry out the state responsibilities.

Low-level radioactive waste is expected to be generated from activities conducted under all of the alternatives. Therefore, this waste must be managed in compliance with this act. The Waste Management sections of Chapter 4 provide information on the generation and management of low-level radioactive waste for each of the alternatives.

Solid Waste Disposal Act of 1965, as amended by the Resource Conservation and Recovery Act of 1976 and the Hazardous and Solid Waste Amendments of 1984 (42 U.S.C. 6901 *et seq.*)—The Solid Waste Disposal Act of 1965, as amended, governs the transportation, treatment, storage, and disposal of hazardous and nonhazardous waste. Under the Resource Conservation and Recovery Act of 1976 (RCRA), which amended the Solid Waste Disposal Act of 1965, EPA defines and identifies hazardous waste; establishes standards for its transportation, treatment, storage, and disposal; and requires permits for persons engaged in hazardous waste activities. Section 3006 of the act (42 U.S.C. 6926) allows states to establish and administer these permit programs with EPA approval. The EPA regulations implementing RCRA are found in 40 CFR Parts 260 through 283.

Regulations imposed on a generator or on a treatment, storage, and/or disposal facility vary according to the type and quantity of material or waste generated, treated, stored, and/or disposed. The method of treatment, storage, and/or disposal also impacts the extent and complexity of the requirements.

Hazardous and mixed waste is expected to be generated from activities conducted for all of the alternatives. Therefore, these waste types must be managed in compliance with this act. The Waste Management sections of Chapter 4 provide information on the generation and management of hazardous and mixed waste for each of the alternatives.

Federal Facility Compliance Act of 1992 (42 U.S.C. 6961 *et seq.*)—The Federal Facility Compliance Act, enacted on October 6, 1992, amended RCRA. Section 102(a)(3) of the Federal Facility Compliance Act waives sovereign immunity for Federal facilities from fines and penalties for violations of RCRA, state, interstate, and local hazardous and solid waste management requirements. This waiver was delayed for 3 years following enactment for violations of the land disposal restrictions storage and prohibition (RCRA Section 3004(j)) involving mixed waste at DOE facilities. The act further delays the waiver of sovereign immunity beyond the 3-year period at a facility if DOE is in compliance with an approved plan for developing treatment capacity and technologies for mixed waste generated or stored at the facility, as well as an order requiring compliance with the plan.

Mixed low-level radioactive waste is expected to be generated from activities conducted for all of the alternatives. Therefore, this waste must be managed in compliance with this act. The Waste Management sections of Chapter 4 provide more information on the generation and management of mixed waste for each of the alternatives.

Nuclear Waste Policy Act of 1982 as amended (U.S.C. 10101 through 10271)—The Nuclear Waste Policy Act provides for research, development, and demonstration activities regarding disposal of high-level radioactive waste and spent nuclear fuel. As originally enacted, the Nuclear Waste Policy Act called for the Secretary of Energy to recommend candidate repository sites, but in 1987 it was amended to require DOE to proceed only with characterization of the Yucca Mountain Site (42 U.S.C. 10133 and 10172). The Nuclear Waste Policy Act also established the Office of Civilian Radioactive Waste Management (42 U.S.C. 10224), the Office of Nuclear Waste Negotiator (42 U.S.C. 10242), and the Nuclear Waste Fund (42 U.S.C. 10222); and it provides authority (along with the Atomic Energy Act) for EPA to develop standards for protection of the general environment from the management and disposal of spent nuclear fuel, transuranic, and high-level radioactive waste (40 CFR Part 191).

As discussed in Chapter 4, Sections 4.3.1.1.13, 4.3.2.1.13, 4.3.3.1.13, and 4.4.3.1.13, DOE is considering whether the waste from processing irradiated neptunium-237 targets should be classified as high-level radioactive waste. If this waste were to be classified as high-level radioactive waste, then this act would be applicable. In addition, the spent nuclear fuel expected to be generated under the alternatives to restart or deactivate FFTF, as discussed in Sections 4.3.1.1.14 and 4.4.1.2.14, must be managed in compliance with this act.

Pollution Prevention Act of 1990 (42 U.S.C. 13101 *et seq.*)—The Pollution Prevention Act establishes a national policy for waste management and pollution control. Source reduction is given first preference, followed by environmentally safe recycling, with disposal or releases to the environment as a last resort. In response to the policies established by the act, DOE committed to participation in the Superfund Amendments and Reauthorization Act, Section 313, EPA 33/50 Pollution Prevention Program. The goal for facilities involved in compliance with Section 313 is to achieve a 33 percent reduction (from a 1993 baseline) in the release of 17 priority chemicals by 1997. On August 3, 1993, President Clinton issued Executive Order 12856 requiring DOE to achieve a 50 percent reduction in total releases of all toxic chemicals by December 31, 1999. On November 12, 1999, U.S. Secretary of Energy Bill Richardson issued 14 pollution prevention and energy efficiency goals for DOE. These goals are designed to build environmental accountability and stewardship into DOE's decision-making process. Under these goals, DOE will strive to minimize waste and maximize energy efficiency as measured by continuous cost-effective improvements in the use of materials and energy, with the years 2005 and 2010 as interim measurement points.

Radioactive, hazardous, and nonhazardous waste types are expected to be generated from all the alternatives; therefore, efforts must be made to minimize their generation. The Waste Management sections of Chapter 4 provide more information on the generation and management of these wastes.

Toxic Substances Control Act of 1976 (15 U.S.C. 2601 *et seq.*)—The Toxic Substances Control Act provides EPA with the authority to require testing of chemical substances entering the environment and to regulate them as necessary. The law complements and expands existing toxic substance laws, such as Section 112 of the Clean Air Act and Section 307 of the Clean Water Act. The Toxic Substances Control Act requires compliance with inventory reporting and chemical control provisions of the act to protect the public from the risks of exposure to chemicals. The act also imposes strict limitations on the use and disposal of polychlorinated biphenyls, chlorofluorocarbons, asbestos, dioxins, certain metal-working fluids, and hexavalent chromium.

Activities under all of the alternatives would need to be in compliance with this act.

National Historic Preservation Act of 1996, as amended (16 U.S.C. 470 *et seq.*)—The National Historic Preservation Act provides that sites with significant national historic value be placed on the *National Register of Historic Places*, which is maintained by the Secretary of the Interior. The major provisions of the act for DOE are Sections 106 and 110. Both sections aim to ensure that historic properties are appropriately considered in planning Federal initiatives and actions. Section 106 is a specific, issue-related mandate to which Federal agencies must adhere. It is a reactive mechanism that is driven by a Federal action. Section 110, in contrast, sets out broad Federal agency responsibilities with respect to historic properties. It is a proactive mechanism with emphasis on ongoing management of historic preservation sites and activities at Federal facilities. No permits or certifications are required under the act.

Section 106 requires the head of any Federal agency having direct or indirect jurisdiction over a proposed Federal or federally assisted undertaking to ensure compliance with the provisions of the act. It compels Federal agencies to "take into account" the effect of their projects on historical and archaeological resources and to give the Advisory Council on Historic Preservation the opportunity to comment on such effects. Section 106 mandates consultation during Federal actions if the undertaking has the potential to have an effect on a historic property. This consultation normally involves the State and/or Tribal Historic Preservation Officers and may include other organizations and individuals, such as local governments, Native American tribes, and Native Hawaiian organizations. If an adverse effect is found, the consultation often ends with the execution of a memorandum of agreement that states how the adverse effects will be resolved.

The regulations implementing Section 106, found in 30 CFR Part 800, were revised on May 18, 1999 (64 FR 27043), effective June 17, 1999. This revision introduced new flexibility and options for agencies to use to meet their obligations to comply with the act.

As discussed in the Cultural and Paleontological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, no historic property resources would be impacted. Two FFTF structures (the Reactor Containment Building and FFTF Control Building) at Hanford have been determined to be eligible for the National Register of Historic Places as contributing properties within the Historic District recommended for mitigation; however, as discussed in Section 4.3.1.1.7, the restart of FFTF would be consistent with the purpose for which the reactor was built and would not affect the status of the aforementioned structures. In addition, the Materials Test Reactor, the Engineering Test Reactor, and ATR, as well as a number of support facilities at the Idaho National Engineering and Environmental Laboratory (INEEL), are potentially eligible for nomination to the National Register. However, as discussed in Section 4.4.1.1.7, the use of ATR would not affect the potential eligibility of these structures for listing.

For the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, additional information would be required to determine if there are any historic resources, including those that are eligible for listing on the National Register of Historic Places. These resources would be identified through site surveys and consultations with the State Historic Preservation Officer.

American Antiquities Act of 1906, as amended (16 U.S.C 431 to 433)—This act protects historic and prehistoric ruins, monuments, and antiquities, including paleontological resources, on federally controlled lands from appropriation, excavation, injury, and destruction without permission. On June 9, 2000, the Hanford Reach was designated as a national monument under this act.

As discussed in the Cultural and Paleontological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, no historic or prehistoric property resources would be impacted.

For the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, additional information would be required to determine whether there are any historic or prehistoric resources and the potential impacts.

Archaeological and Historic Preservation Act of 1974, as amended (16 U.S.C 469 to 469c)—This act protects sites that have historic and prehistoric importance.

As discussed in the Cultural and Paleontological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, no historic or prehistoric property resources would be impacted.

For the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, additional information would be required to determine whether there are any historic or prehistoric resources and the potential impacts.

Archaeological and Resources Protection Act of 1979, as amended (16 U.S.C. 470 *et seq.*)—This act requires a permit for any excavation or removal of archaeological resources from Federal or Native American lands. Excavations must be undertaken for the purpose of furthering archaeological knowledge in the public interest, and resources removed are to remain the property of the United States. The law requires that whenever any Federal agency finds that its activities may cause irreparable loss or destruction of significant scientific, prehistoric, or archaeological data, the agency must notify the U.S. Department of the Interior and may request that the Department undertake the recovery, protection, and preservation of such data. Consent must be obtained from the Native American tribe or the Federal agency having authority over the land on which a resource is located before issuance of a permit; the permit must contain terms and conditions requested by the tribe or Federal agency.

As discussed in the Cultural and Paleontological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new construction, except for the new 76-meter (250-foot) stack at FMEF, or land disturbance is expected, a permit would not be required.

However, for alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information would be needed to determine whether a permit would be required.

Endangered Species Act of 1973, as amended (16 U.S.C. 1531 *et seq.*)—The Endangered Species Act is intended to prevent the further decline of endangered and threatened species and to restore these species and habitats. Section 7 of the act requires Federal agencies having reason to believe that a prospective action may affect an endangered or threatened species or its habitat to consult with the U.S. Fish and Wildlife Service of the U.S. Department of the Interior or the National Marine Fisheries Service of the U.S. Department of Commerce to ensure that the action does not jeopardize the species or destroy its habitat (50 CFR Part 17). If, despite reasonable and prudent measures to avoid or minimize such impacts, the species or its habitat would be jeopardized by the action, a review process is specified to determine whether the action may proceed.

As discussed in the Ecological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new

construction, except for the new 76-meter (250-foot) stack at FMEF, or land disturbance is expected, impacts to threatened and endangered species are not expected.

However, for alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information would be needed to determine whether threatened or endangered species would be impacted.

Migratory Bird Treaty Act of 1918, as amended (16 U.S.C. 703 *et seq.*)—The Migratory Bird Treaty Act, as amended, is intended to protect birds that have common migration patterns between the United States and Canada, Mexico, Japan, and Russia. It regulates the harvest of migratory birds by specifying conditions such as the mode of harvest, hunting seasons, and bag limits. The act stipulates that it is unlawful at any time, by any means, or in any manner, to "kill ... any migratory bird." Although no permit for this project is required under the act, DOE is required to consult with the U.S. Fish and Wildlife Service regarding impacts to migratory birds, and to avoid or minimize these effects in accordance with the U.S. Fish and Wildlife Service Mitigation Policy.

As discussed in the Ecological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new construction, except for the new 76-meter (250-foot) stack at FMEF, or land disturbance is expected, impacts to migratory birds are not expected.

However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information would be needed to determine whether migratory birds would be impacted.

Bald and Golden Eagle Protection Act of 1973, as amended (16 U.S.C. 668 through 668d)—The Bald and Golden Eagle Protection Act, as amended, makes it unlawful to take, pursue, molest, or disturb bald (American) and golden eagles, their nests, or their eggs anywhere in the United States (Section 668, 668c). A permit must be obtained from the U.S. Department of the Interior to relocate a nest that interferes with resource development or recovery operations.

As discussed in the Ecological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new construction, except for the new 76-meter (250-foot) stack at FMEF, or land disturbance is expected, impacts to migratory birds are not expected.

However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information would be needed to determine whether bald or golden eagles would be impacted and whether a permit would be required.

Fish and Wildlife Coordination Act (16 U.S.C. 661 *et seq.*)—The Fish and Wildlife Coordination Act promotes more effectual planning and cooperation between Federal, state, public, and private agencies for the conservation and rehabilitation of the Nation's fish and wildlife and authorizes the U.S. Department of the Interior to provide assistance. This act requires consultation with the U.S. Fish and Wildlife Service on the possible effects on wildlife if there is construction, modification, or control of bodies of water in excess of 10 acres in surface area.

As discussed in the Ecological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new

construction, except for the 76-meter (250-foot) stack at FMEF, or land disturbance is expected, impacts to the Nation's fish and wildlife are not expected.

However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information would be needed to determine whether there would be any impacts to the Nation's fish and wildlife.

Wild and Scenic Rivers Act of 1968, as amended (16 U.S.C. 1271 *et seq.*)—This act requires consultation before construction of any new Federal project associated with a river designated as wild and scenic or under study to minimize and mitigate any adverse effects on the physical and biological properties of the river.

As discussed in Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new construction, except for the 76-meter (250-foot) stack at FMEF, or land disturbance is expected, impacts to any rivers designated as wild and scenic or under study are not expected. In addition, impacts on visual resources as a result of constructing a new stack at FMEF are expected to be minimal.

However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information would be needed to determine whether there would be any impacts to rivers designated as wild and scenic or under study.

Farmland Protection Act of 1981 (7 U.S.C. 4201 *et seq.*)—This act requires the avoidance of any adverse effects on prime and unique farmlands.

As discussed in Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new construction, except for the 76-meter (250-foot) stack at FMEF, or land disturbance is expected, impacts to any prime and unique farmlands are not expected.

However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information would be needed to determine whether there would be any impacts to prime or unique farmlands.

American Indian Religious Freedom Act of 1978 (42 U.S.C. 1996)—This act reaffirms Native American religious freedom under the First Amendment, and sets U.S. policy to protect and preserve the inherent and constitutional right of Native Americans to believe, express, and exercise their traditional religions. The act requires that Federal actions avoid interfering with access to sacred locations and traditional resources that are integral to the practice of religions.

As discussed in the Cultural and Paleontological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new construction, except for the 76-meter (250-foot) stack at FMEF, or land disturbance is expected, impacts to Native American resources are not expected.

However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information about the presence, type, and location of Native American resources would be needed to determine whether there would be any impacts to these resources.

Religious Freedom Restoration Act of 1993 (42 U.S.C. 2000(bb) *et seq.*)—This act prohibits the U.S. Government, including Federal departments, from substantially burdening the exercise of religion unless the Government demonstrates a compelling governmental interest, the action furthers a compelling government interest, and it is the least restrictive means of furthering that interest.

As discussed in the Cultural and Paleontological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new construction, except for the 76-meter (250-foot) stack at FMEF, or land disturbance is expected, impacts to Native American resources are not expected.

However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information about the presence, type, and location of Native American resources would be needed to determine whether there would be any impacts to these resources.

Native American Graves Protection and Repatriation Act of 1990 (25 U.S.C. 3001)—This act established a means for Native Americans to request the return or "repatriation" of human remains and other cultural items presently held by Federal agencies or federally assisted museums or institutions. The act also contains provisions regarding the intentional excavation and removal of, inadvertent discovery of, and illegal trafficking in Native American human remains and cultural items. Major actions under this law include: (a) establishing a review committee with monitoring and policy-making responsibilities, (b) developing regulations for repatriation, including procedures for identifying lineal descent or cultural affiliation needed for claims, (c) providing oversight of museum programs designed to meet the inventory requirements and deadlines of this law, and (d) developing procedures to handle unexpected discoveries of graves or grave goods during activities on Federal or tribal lands. All Federal agencies that manage land and/or are responsible for archaeological collections from their lands or generated by their activities must comply with the act. DOE managers of ground-disturbing activities on Federal and tribal lands should make themselves aware of the statutory provisions treating inadvertent discoveries of Native American remains and cultural objects. Regulations implementing the act are found at 43 CFR Part 10.

As discussed in the Cultural and Paleontological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities and the existing generic CLWR, due to the developed nature of these areas and the fact that no new construction, except for the 76-meter (250-foot) stack at FMEF, or land disturbance is expected, it is unlikely that human remains or other cultural items would be uncovered.

However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, disturbance of land would occur and, therefore, additional information about the presence, type, and location of Native American resources would be needed to determine whether there would be any impacts to these resources.

Occupational Safety and Health Act of 1970 (29 U.S.C. 651 *et seq.*)—The Occupational Safety and Health Act establishes standards for safe and healthful working conditions in places of employment throughout the United States. The act is administered and enforced by the Occupational Safety and Health Administration (OSHA), a U.S. Department of Labor agency. Although OSHA and EPA both have a mandate to reduce exposures to toxic substances, OSHA's jurisdiction is limited to safety and health conditions that exist in the workplace environment.

Under the act, it is the duty of each employer to furnish employees a place of employment free of recognized hazards likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and rules, regulations, and orders issued under the act. OSHA regulations

(29 CFR Part 1910) establish specific standards telling employers what must be done to achieve a safe and healthful working environment. Government agencies, including DOE, are not technically subject to OSHA regulations, but are required under 29 U.S.C. 668 to establish their own occupational safety and health programs for their places of employment which are consistent with OSHA standards. DOE places emphasis on compliance with these regulations at its facilities, and prescribes, through DOE orders, the Occupational Safety and Health Act standards that contractors shall meet, as applicable to their work at government-owned, contractor-operated facilities (DOE Order 440.1A). DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths as required by OSHA regulations.

Activities under all the alternatives would need to be conducted in compliance with this act.

Noise Control Act of 1972, as amended (42 U.S.C. 4901 *et seq.*)—Section 4 of the Noise Control Act of 1972, as amended, directs all Federal agencies to carry out "to the fullest extent within their authority" programs within their jurisdictions in a manner that furthers a national policy of promoting an environment free from noise jeopardizing health and welfare.

As discussed in the Noise sections of Chapter 4, the operation of existing DOE facilities and existing CLWR would not result in any impacts because of increased noise levels. The construction and operation of new DOE facilities would require compliance with this requirement.

5.1.2 Environmental, Safety, and Health Executive Orders

Executive Order 11514 (Protection and Enhancement of Environmental Quality)—This order (regulated by 40 CFR Parts 1500 through 1508) requires Federal agencies to continually monitor and control their activities to: (1) protect and enhance the quality of the environment, and (2) develop procedures to ensure the fullest practicable provision of timely public information and understanding of the Federal plans and programs that may have potential environmental impact so that views of interested parties can be obtained. DOE has issued regulations (10 CFR Part 1021) and DOE Order 451.1B for compliance with this Executive order.

As previously discussed under Section 5.1.1, this NI PEIS has been prepared in accordance with NEPA requirements (i.e., 40 CFR Parts 1500 through 1508, 10 CFR Part 1021, and DOE Order 451.1B).

Executive Order 11593 (National Historic Preservation, May 13, 1971)—This order directs Federal agencies to locate, inventory, and nominate properties under their jurisdiction or control to the *National Register of Historic Places*, if those properties qualify. This process requires DOE to provide the Advisory Council on Historic Preservation the opportunity to comment on the possible impacts of the proposed activity on any potential eligible or listed resources. Compliance with this Executive order is discussed in Section 5.1.1, National Historic Preservation Act of 1996, as amended.

Executive Order 11988 (Floodplain Management)—This order (regulated by 10 CFR Part 1022) requires Federal agencies to establish procedures to ensure that the potential effects of flood hazards and floodplain management are considered for any action undertaken in a floodplain, and that floodplain impacts be avoided to the extent practicable. As discussed in the Water Resource sections of Chapter 3, for the alternatives involving existing DOE facilities and the existing generic CLWR, compliance with this order has already been met. However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, additional information would be needed to determine compliance with this order.

Executive Order 11990 (Protection of Wetlands)—This order (regulated by 10 CFR Part 1022) requires Federal agencies to avoid any short- or long-term adverse impacts on wetlands wherever there is a practicable

alternative. As discussed in the Ecological Resource sections of Chapter 4, for the alternatives involving existing DOE facilities, impacts to wetlands are not expected because either there are no wetlands in the vicinity of the facility or because water usage and water discharge would not change or would be small fractions of current values, and discharge chemistry would not be expected to change. However, for the alternatives involving the construction and operation of one or two new DOE accelerators or a research reactor and support facility, additional information would be needed to determine whether there would be impacts to wetlands.

Executive Order 12088 (Federal Compliance with Pollution Control Standards, October 13, 1978, as amended by Executive Order 12580, Federal Compliance with Pollution Control Standards, January 23, 1987)—This order directs Federal agencies to comply with applicable administrative and procedural pollution control standards established by, but not limited to, the Clean Air Act, the Noise Control Act, the Clean Water Act, the Safe Drinking Water Act, the Toxic Substances Control Act, and RCRA.

Activities under all of the alternatives involving DOE facilities would need to be in compliance with this order.

Executive Order 12580 (Superfund Implementation)—This order delegates to the heads of executive departments and agencies the responsibility of undertaking remedial actions for releases or threatened releases that are not on the National Priorities List, and removal actions, other than emergencies, where the release is from any facility under the jurisdiction or control of executive departments and agencies.

Activities under all of the alternatives involving DOE facilities would need to be in compliance with this order.

Executive Order 12898 (Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations)—This order requires each Federal agency to identify and address disproportionately high and adverse human health or environmental effects of its programs, policies, and activities on minority and low-income populations. The Environmental Justice sections of Chapter 4 provide information on the compliance with this order.

Executive Order 12902 (Energy Efficiency and Water Conservation at Federal Facilities)—This order requires Federal agencies to develop and implement a program for conservation of energy and water resources.

Activities under all of the alternatives involving DOE facilities would need to be in compliance with this order.

Executive Order 13101 (Greening the Government Through Waste Prevention, Recycling, and Federal Acquisition)—This order requires each Federal agency to incorporate waste prevention and recycling in its daily operations and work to increase and expand markets for recovered materials. This order states that it is national policy to prefer pollution prevention, whenever feasible. Pollution that cannot be prevented should be recycled; pollution that cannot be prevented or recycled should be treated in an environmentally safe manner. Disposal should be employed only as a last resort.

Activities under all of the alternatives involving DOE facilities would need to be in compliance with this order.

Executive Order 13112 (Invasive Species)—This order requires Federal agencies to prevent the introduction of invasive species and provide for their control and to minimize the economic, ecological, and human health impacts that invasive species cause.

Activities under all of the alternatives involving DOE facilities would need to be in compliance with this order.

Executive Order 13148 (Greening the Government Through Leadership in Environmental Management—This order sets new goals for pollution prevention, requires all Federal facilities to have an environmental management system, and requires compliance or environmental management system audits.

Activities under all alternatives involving DOE facilities would need to be in compliance with this order.

5.1.3 DOE Environmental, Safety, and Health Regulations and Orders

The Atomic Energy Act authorizes DOE to establish standards to protect health or minimize dangers to life or property from activities under DOE's jurisdiction. Through a series of DOE orders and regulations, an extensive system of standards and requirements has been established to ensure safe operation of DOE facilities. DOE regulations and orders do not apply to activities regulated by NRC (10 CFR Sections 830.2(a) and 835.1(b)). Thus, DOE regulations and orders would not apply to CLWR facilities.

DOE regulations are found in 10 CFR. These regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purpose of this NI PEIS, relevant regulations include: "Procedural Rules for DOE Nuclear Activities" (10 CFR Part 820), "Nuclear Safety Management" (10 CFR Part 830), "Occupational Radiation Protection" (10 CFR Part 835), "Compliance with the National Environmental Policy Act" (10 CFR Part 1021), and "Compliance with Floodplains/Wetlands Environmental Review Requirements" (10 CFR Part 1022).

DOE orders are issued in support of environmental, safety, and health programs. Many DOE orders have been revised and reorganized to reduce duplication and to eliminate obsolete provisions. The new DOE order organization is by series, with each number identified by three digits, and is intended to include all DOE orders, policies, manuals, requirement documents, notices, and guides. The remaining DOE orders, which are identified by four digits, are expected to be revised and converted to the new DOE numbering system. The major DOE orders pertaining to the alternatives are listed in **Table 5–1**.

Table 5–1 Relevant DOE Orders (as of October 26, 2000)

DOE Order	Subject
Leadership/Management Planning	
O 151.1	Comprehensive Emergency Management System (09/25/95; Change 2, 08/21/96)
Information and Analysis	
O 231.1	Environment, Safety and Health Reporting (09/30/95; Change 2, 11/07/96)
O 232.1A	Occurrence Reporting and Processing of Operations Information (07/21/97)
Work Processes	
O 414.1A	Quality Assurance (09/29/99)
O 420.1	Facility Safety (10/13/95; Change 2, 10/24/96)
O 430.1A	Life Cycle Asset Management (10/14/98)
O 435.1	Radioactive Waste Management (07/09/99)
O 440.1A	Worker Protection Management for DOE Federal and Contractor Employees (03/27/98)
O 451.1B	National Environmental Policy Act Compliance Program (10/26/00)
O 460.1A	Packaging and Transportation Safety (10/02/96)
O 460.2	Departmental Materials Transportation and Packaging Management (09/27/95; Change 1, 10/26/95)
O 470.1	Safeguards and Security Program (09/28/95; Change 1, 06/21/96)
O 470.2A	Oversight and Performance Assurance Program (03/01/00)
O 473.2	Protective Force Program (06/30/00)
O 474.1	Control and Accountability of Nuclear Materials (08/11/99)
External Relationships	
1230.2	American Indian Tribal Government Policy (04/08/92)
Personnel Relations and Services	
3790.1B	Federal Employee Occupational Safety and Health Program (01/07/93)
Real Property Management	
4330.4B	Maintenance Management Program (02/10/94)
Project Management	
4700.1	Project Management System (03/06/87; Change 1, 06/02/92)
Environmental Quality and Impact	
5400.1	General Environmental Protection Program (11/09/88; Change 1, 06/29/90)
5400.5	Radiation Protection of the Public and the Environment (02/08/90; Change 2, 01/07/93)
5480.4	Environmental Protection, Safety, and Health Protection Standards (05/15/84; Change 4, 01/07/93)
5480.19	Conduct of Operations Requirements for DOE Facilities (07/09/90; Change 1, 05/18/92)
5480.20A	Personnel Selection, Qualification, and Training Requirements for DOE Nuclear Facilities (11/15/94)
5480.21	Unreviewed Safety Questions (12/24/91)
5480.22	Technical Safety Requirements (02/25/92; Change 2, 01/23/96)
5480.23	Nuclear Safety Analysis Reports (04/10/92; Change 1, 03/10/94)
5480.30	Nuclear Reactor Safety Design Criteria (01/19/93)
Emergency Preparedness	
5530.3	Radiological Assistance Program (01/14/92; Change 1, 04/10/92)
5530.5	Federal Radiological Monitoring and Assessment Center (07/10/92; Change 1, 12/02/92)
Defense Programs	
5610.14	Transportation Safeguards System Program Operations (05/12/93)
5632.1C	Protection and Control of Safeguards and Security Interests (07/15/94)
5660.1B	Management of Nuclear Materials (05/26/94)
Design	
6430.1A	General Design Criteria (04/06/89)

5.1.4 State Environmental Laws, Regulations, and Agreements

Certain environmental requirements, including many discussed in Section 5.1.1, have been delegated to state authorities for implementation and enforcement. It is DOE policy to conduct its operations in an environmentally safe manner, in compliance with all applicable laws, regulations, and standards, including state laws and regulations. A list of applicable state laws, regulations, and agreements is provided in **Table 5–2**. This list is not exhaustive and other state laws and regulations may be applicable. In addition, other state laws and regulations may be applicable for the CLWR and the construction and operation of new accelerator(s) or a research reactor and support facilities, but are not specifically identified in Table 5–2 because the specific locations of these facilities have not been determined.

Table 5–2 State Environmental Laws, Regulations, and Agreements

Law/Regulation/Agreement	Citation	Requirements
HANFORD, WASHINGTON		
Washington Clean Air Act	Revised Code of Washington (RCW), Chapter 70.94	Provides for development of air pollution control and permitting regulations.
Washington State Air Pollution Control Regulations	Washington Administrative Code (WAC), Chapters 173-400 through 173-495	Requires permitting of source and control of toxic air pollutants, radionuclides, and other pollutants.
Water Pollution Control Act of 1945	RCW, Chapter 90.48	Establishes a permit system to license and control the discharge of pollutants into waters of the state. Permits are required for both point-source and non-point-source discharges.
Surface Water Quality Standards	WAC, Chapter 173-201A	Establishes water quality standards for surface waters at levels protective of aquatic life.
Washington State Department of Health Radiation Protection Requirements	WAC, Chapter 246-247	Establishes requirements for all facilities with the potential to emit airborne radioactivity, including Federal facilities.
Hazardous Waste Management Act	RCW, Chapter 70.105	Requires permits for various activities involving hazardous waste.
Radioactive Waste Storage and Transport Act of 1980	RCW, Chapter 70.99	Establishes various requirements for handling, storage, and transportation of radioactive waste.
Dangerous Waste Regulations	WAC, Chapter 173-303	Establishes hazardous waste treatment, storage, and disposal standards and permit requirements. These requirements cover a larger universe of materials than the Federal hazardous waste program.
Department of Fish and Wildlife	WAC, Chapter 232-12	Defines the requirements that the Department of Game must take to protect endangered or threatened wildlife.

Law/Regulation/Agreement	Citation	Requirements
Hanford Federal Facility Agreement and Consent Order (Tri-Party Agreement)	May 15, 1989 (amended December 31, 1998)	An enforceable agreement, which details work necessary to comply with state and Federal hazardous waste management requirements and the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). This is an agreement among DOE, EPA, and the Washington State Department of Ecology.
IDAHO NATIONAL ENGINEERING AND ENVIRONMENTAL LABORATORY, IDAHO		
Idaho Environmental Protection and Health Act	ID Code, Title 39, Chapter 1	Provides for development of air pollution control permitting regulations.
Rules for the Control of Air Pollution in Idaho	Idaho Administrative Procedure Act (IDAPA) 58, Title 1, Chapter 1	Requires permitting of sources and control of toxic air pollutants and other pollutants.
Idaho Water Pollution Control Act	ID Code, Title 39, Chapter 36	Establishes a program to enhance and preserve the quality and the value of water resources.
Idaho Hazardous Waste Management Act	ID Code, Title 39, Chapter 44	Requires permit prior to construction or modification of a hazardous waste disposal facility.
Rules and Standards for Hazardous Waste	IDAPA 58, Title 1, Chapter 5 (58.01.03)	Requires permit prior to construction or modification of a hazardous waste disposal facility.
Various Acts Regarding Fish and Game	ID Code, Title 36, Chapters 9, 11, and 24	Requires consultation with responsible agency.
Idaho Historic Preservation Act	ID Code, Title 67, Chapter 46	Requires consultation with responsible local governing body.
Spent Fuel Settlement Agreement (also known as the Governor's Agreement)	October 16, 1995	Allows INEEL to receive spent nuclear fuel and mixed waste from off site and establishes schedules for the treatment of existing high-level radioactive waste, transuranic waste, mixed waste, and removal of spent nuclear fuel from the state. (This agreement is not applicable to the alternatives because only new waste will be generated by the proposed action. This newly generated waste, if determined to be mixed waste, will be covered by the INEEL Site Treatment Plan.)
Consent Order for Federal Facility Compliance Plan	November 1, 1995	Addresses compliance with the Federal Facility Compliance Act and mixed waste treatment issues by implementing the INEEL Site Treatment Plan.

Law/Regulation/Agreement	Citation	Requirements
OAK RIDGE NATIONAL LABORATORY, TENNESSEE		
Tennessee Air Pollution Control Act	TN Code, Title 68, Chapter 201 (Part 1)	Provides for permitting to construct, modify, or operate an air contaminant source.
Tennessee Air Pollution Control Regulations	TN Rule, Chapter 1200-3	Requires a permit to construct, modify, or operate an air contaminant source. Also sets fugitive dust requirements.
Tennessee Water Quality Control Act	TN Code, Title 69, Chapter 3	Provides authority to issue new or modify existing NPDES permits required for a water discharge source.
Tennessee Water Pollution Control Regulations	TN Rule, Chapter 1200-4	Requires a new or modification of an existing NPDES permit for a water discharge source.
Tennessee Hazardous Waste Management Act	TN Code, Title 68, Chapter 212 (Part 1)	Requires permit for any construction or modification of a hazardous waste facility.
Tennessee State Executive Order on Wetlands	TN State Wetlands Conservation Strategy	Provides guidance from the Governor's Interagency Wetlands Committee.
Tennessee Nongame and Endangered or Threatened Wildlife Species Conservation Act of 1974	TN Code, Title 70, Chapter 8 (Part 1)	Requires consultation with responsible agency.
Tennessee Department of Environmental Conservation Order	October 1, 1995	Requires DOE to comply with the Site Treatment Plan for the management and treatment of mixed radioactive waste.

5.2 RADIOACTIVE MATERIAL PACKAGING AND TRANSPORTATION REGULATIONS

Transportation of hazardous and radioactive materials and substances is governed by the U.S. Department of Transportation (DOT) and NRC. The Hazardous Material Transportation Act of 1975 (49 U.S.C. 5105 *et seq.*) requires DOT to prescribe uniform national regulations for transportation of hazardous materials (including radioactive materials). Most state and local regulations regarding such transportation that are not substantively the same as DOT regulations are preempted (i.e., rendered void) (49 U.S.C. 5125). This, in effect, allows state and local governments only to enforce the Federal regulations, not to change or expand upon them.

This program is administered by the Research and Special Programs Administration of DOT, which coordinates its regulations with those of NRC (under the Atomic Energy Act) and with EPA (under RCRA) when covering the same activities.

DOT regulations, which may be found under 49 CFR Parts 171 through 178, and 49 CFR Parts 383 through 397, contain requirements for identifying a material as hazardous or radioactive. These regulations interface with the NRC regulations for identifying material, but DOT hazardous material regulations govern the hazard communication (such as marking, hazard labeling, vehicle placarding, and emergency response telephone number) and shipping requirements.

The NRC regulations applicable to radioactive materials transportation may be found under 10 CFR Part 71. These regulations include detailed packaging design requirements and package certification testing requirements. Complete documentation of design and safety analysis, and the results of the required testing, are submitted to NRC to certify the package for use. This certification testing involves the following components: heat, physical drop onto an unyielding surface, water submersion, puncture by dropping the package onto a steel bar, and gas tightness.

Transportation casks, which are used to transport radioactive material, are subject to numerous inspections and tests (10 CFR Section 71.87). These tests are designed to ensure that cask components are properly assembled and meet applicable safety requirements. Tests and inspections are clearly identified in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks are loaded and inspected by registered users in compliance with approved quality assurance programs. Operations involving the casks are conducted in compliance with 10 CFR Section 71.91. Reports of defects or accidental mishandling are submitted to NRC.

5.3 EMERGENCY MANAGEMENT AND RESPONSE LAWS, REGULATIONS, AND EXECUTIVE ORDERS

This section discusses the laws, regulations, and Executive orders that address the protection of public health and worker safety and require the establishment of emergency plans. These laws, regulations, and Executive orders relate to the operation of facilities, including DOE facilities and CLWRs that engage directly or indirectly in the production of special nuclear material.

5.3.1 Emergency Management and Response Federal Laws

Emergency Planning and Community Right-to-Know Act of 1986 (U.S.C. 11001 *et seq.*) (also known as "SARA Title III")—This act requires emergency planning, and notice to communities and government agencies, of the presence and release of specific chemicals. EPA implements this act under regulations found in 40 CFR Parts 355, 370, and 372. Under Subtitle A of this act, Federal facilities are required to provide various information (such as inventories of specific chemicals used or stored and releases that occur from these sites) to the state emergency response commission and to the local emergency planning committee to ensure that emergency plans are sufficient to respond to unplanned releases of hazardous substances. Implementation of the provisions of this act began voluntarily in 1987, and inventory and annual emissions reporting began in 1988. DOE requires compliance with Title III as a matter of DOE policy at its contractor-operated facilities.

Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (42 U.S.C. 9604(I) (also know as "Superfund")—This act provides authority for Federal and state governments to respond directly to hazardous substance incidents. The act requires reporting of spills, including radioactive spills, to the National Response Center.

Robert T. Stafford Disaster Relief and Emergency Assistance Act of 1988 (42 U.S.C. 5121)—This act, as amended, provides an orderly and continuing means of assistance by the Federal Government to state and local governments in managing their responsibilities to alleviate suffering and damage resulting from disasters. The President, in response to a state Governor's request, may declare an "emergency" or "major disaster" to provide Federal assistance under this act. The President, in Executive Order 12148, delegated all functions, except those in Sections 301, 401, and 409, to the Director of the Federal Emergency Management Agency. The act provides for the appointment of a Federal coordinating officer who will operate in the designated area with a state coordinating officer for the purpose of coordinating state and local disaster assistance efforts with those of the Federal Government.

Justice Assistance Act of 1984 (42 U.S.C. 3701-3799)—This act establishes the Emergency Federal Law Enforcement Assistance, which provides assistance to state and local governments in responding to a law enforcement emergency. The act defines the term "law enforcement emergency" as an uncommon situation which requires law enforcement, which is or threatens to become of serious or epidemic proportions, and with respect to which state and local resources are inadequate to protect the lives and property of citizens or to enforce the criminal law. Emergencies that are not of an ongoing or chronic nature (for example, the Mount Saint Helens volcanic eruption) are eligible for Federal law enforcement assistance. This assistance includes funds, equipment, training, intelligence information, and personnel.

Price-Anderson Act (42 U.S.C. 2210)—This act allows DOE to indemnify its contractors if the contract involves the risk of public liability from a nuclear incident.

5.3.2 Emergency Management and Response Federal Regulations

Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR Section 30.72, Schedule C)—This section of the regulations provides a list which is the basis for both the public and private sector to determine if the radiological materials they handle must have an emergency response plan for unscheduled releases, and is one of the threshold criteria documents for DOE hazards assessments required by DOE Order 5500.3A, "Planning and Preparedness for Operational Emergencies." The "Federal Radiological Emergency Response Plan," dated November 1995, primarily discusses offsite Federal response in support of state and local governments with jurisdiction during a peacetime radiological emergency.

Commercial Nuclear Power Plant Emergency Preparedness Planning (44 CFR Part 352)—These regulations generally establish the policies, procedures, and responsibilities of the Federal Emergency Management Agency, NRC, and DOE as guidance for implementing a federal emergency preparedness program.

Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations, and Worker Right to Know (29 CFR Part 1910)—This regulation establishes OSHA requirements for employee safety in a variety of working environments. It addresses employee emergency and fire prevention plans (Section 1910.38), hazardous waste operations and emergency response (Section 1920.120), and hazards communication (Section 1910.1200) that enable employees to be aware of the dangers they face from hazardous materials at their workplace. These regulations do not directly apply to Federal agencies. However, Section 19 of the Occupational Safety and Health Act (29 U.S.C. 668) requires all Federal agencies to have occupational safety programs "consistent" with Occupational Safety and Health Act standards.

Emergency Management and Assistance (44 CFR Section 1.1)—This regulation contains the policies and procedures for the Federal Emergency Management Act, National Flood Insurance Program, Federal Crime Insurance Program, Fire Prevention and Control Program, Disaster Assistance Program, and Preparedness Program, including radiological planning and preparedness.

Hazardous Materials Tables and Communications, Emergency Response Information Requirements (49 CFR Part 172)—This regulation defines the regulatory requirements for marking, labeling, placarding, and documenting hazardous material shipments. The regulation also specifies the requirements for providing hazardous material information and training.

5.3.3 Emergency Response and Management Executive Orders

Executive Order 12148 (Federal Emergency Management, July 20, 1979)—This order transfers functions and responsibilities associated with Federal emergency management to the Director of the Federal Emergency Management Agency. The order assigns the Director the responsibility to establish Federal policies for, and to coordinate all civil defense and civil emergency planning, management, mitigation, and assistance functions of, Executive agencies.

Executive Order 12656 (Assignment of Emergency Preparedness Responsibilities, November 1988)—This order assigns emergency preparedness responsibilities to Federal departments and agencies.

5.4 CONSULTATIONS WITH FEDERAL, STATE, AND LOCAL AGENCIES AND FEDERALLY RECOGNIZED NATIVE AMERICAN GROUPS

Certain laws, such as the Endangered Species Act, the Fish and Wildlife Coordination Act, and the National Historic Preservation Act, require consultation and coordination by DOE with other governmental entities including other Federal agencies, state and local agencies, and federally recognized Native American groups. These consultations must occur on a timely basis and are generally required before any land disturbance can begin. Most of these consultations are related to biotic resources, cultural resources, and Native American rights.

The biotic resource consultations generally pertain to the potential for activities to disturb sensitive species or habitats. Cultural resource consultations relate to the potential for disruption of important cultural resources and archaeological sites. Native American consultations are concerned with the potential for disturbance of ancestral Native American sites and the traditional practices of Native Americans.

DOE initiated the required consultations with the appropriate State Historic Preservation Officers, as required by NEPA and Section 106 of the National Historic Preservation Act; the U.S. Fish and Wildlife Service and the National Marine Fisheries Service, as required by the Endangered Species Act of 1973, the Bald and Golden Eagle Protection Act, and the Migratory Bird Treaty Act; and the appropriate state regulators, as required by state laws or regulations. DOE also initiated the required consultations with the appropriate Native American tribal governments, as required by the Executive Memorandum (dated April 29, 1994) entitled "Government-to-Government Relations with Native American Tribal Governments" and DOE Order 1230.2, "American Indian Tribal Government." A list of those organizations contacted is included in **Table 5–3**. No major issues were identified as a result of the consultation process. The specific results of the consultation process are presented in Chapter 4.

Table 5–3 Organizations Contacted During the Consultation Process

DOE Site	Subject	Addressed To (Date of Letter)
ORR	Ecological resources	Mr. James Widlak U.S. Fish and Wildlife Service (July 10, 2000)
	Ecological resources	Mr. Reginald G. Reeves Tennessee Department of Environment and Conservation (July 10, 2000)
	Cultural resources	Mr. Ollie Keller State Historic Preservation Officer (July 10, 2000)
INEEL	Ecological resources	Mr. Mike Donahoo U.S. Fish and Wildlife Service (July 10, 2000)
	Ecological resources	Mr. George Stephens Idaho Department of Fish and Game (July 10, 2000)
	Cultural resources	Dr. Robert M. Yohe State Historic Preservation Officer (July 10, 2000)
	Native American	The Honorable Lionel Boyer Shoshone-Bannock Tribes (July 10, 2000)
	Native American	Ms. Diana Yupe Shoshone-Bannock Tribes (July 10, 2000)
Hanford	Ecological resources	Mr. James Michaels U.S. Fish and Wildflife Service (July 10, 2000)
	Ecological resources	Mr. Steve Landino National Marine Fisheries Service (July 10, 2000)
	Ecological resources	Ms. Sandy Swope Moody Washington Department of Natural Resources (July 10, 2000)
	Ecological resources	Ms. Lori Guggenmos Washington Department of Wildlife (July 10, 2000)
	Cultural resources	Dr. Allyson Brooks State Historic Preservation Officer (July 10, 2000)
	Native American	Ms. Lenora Seelatsee Wanapum Band (July 10, 2000)
	Native American	Mr. Russell Jim Yakama Nation (July 10, 2000)
	Native American	The Honorable William Burke Confederated Tribes of the Umatilla Indian Reservation (July 10, 2000)
	Native American	Mr. Patrick Sobotta Nez Perce Tribe (July 10, 2000)

Chapter 6
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Chapter 7
Distribution List

The U.S. Department of Energy is providing copies of this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]) (or Summary) to Federal, state, and local elected and appointed government officials and agencies; Native American representatives; Federal, state, and local environmental and public interest groups; and other organizations and individuals as listed. Approximately 1,500 copies of the NI PEIS and 7,500 copies of the Summary of the NI PEIS were sent to interested parties. Copies will be provided to others on request.

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Hanford Watch

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Institute for Science and Sociology
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Chapter 8
Glossary

absorbed dose – For ionizing radiation, the energy imparted to matter by ionizing radiation per unit mass of the irradiated material (e.g., biological tissue). The units of absorbed dose are the rad and the gray. (See rad and gray.)

accelerator – A device that accelerates charged particles (e.g., electrons or protons) to high velocities so they have high kinetic energy (i.e., the energy associated with motion); it focuses the charged particles into a beam and directs them against a target.

accident sequence – With regard to nuclear facilities, an initiating event followed by system failures or operator errors, which can result in significant core damage, confinement system failure, and/or radionuclide releases.

actinide – Any member of the group of elements with atomic numbers from 89 (actinium) to 103 (lawrencium) including uranium and plutonium. All members of this group are radioactive.

activation products – Nuclei, usually radioactive, formed by the bombardment and absorption in material with neutrons, protons, or other nuclear particles.

Advanced Test Reactor – A light-water cooled and moderated test reactor located within the Test Reactor Area of Idaho National Engineering and Environmental Laboratory. It is fueled with uranium enriched with uranium-235 and has a full power level of 250 megawatts, but typically operates at 140 megawatts or less.

air pollutant – Generally, an airborne substance that could, in high enough concentrations, harm living things or cause damage to materials. From a regulatory perspective, an air pollutant is a substance for which emissions or atmospheric concentrations are regulated or for which maximum guideline levels have been established due to potential harmful effects on human health and welfare.

air quality control region – Geographic subdivisions of the United States, designed to deal with pollution on a regional or local level. Some regions span more than one state.

alpha activity – The emission of alpha particles by radioactive materials.

alpha particle – A positively charged particle ejected spontaneously from the nuclei of some radioactive elements. It is identical to a helium nucleus and has a mass number of 4 and an electrostatic charge of +2. It has low penetrating power and a short range (a few centimeters in air). (See alpha radiation.)

alpha radiation – A strongly ionizing, but weakly penetrating, form of radiation consisting of positively charged alpha particles emitted spontaneously from the nuclei of certain elements during radioactive decay. Alpha radiation is the least penetrating of the three common types of ionizing radiation (alpha, beta, and gamma). Even the most energetic alpha particle generally fails to penetrate the dead layers of cells covering the skin and can be easily stopped by a sheet of paper. Alpha radiation is most hazardous when an alpha-emitting source resides inside an organism. (See alpha particle.)

ambient – Surrounding.

ambient air – The surrounding atmosphere as it exists around people, plants, and structures.

ambient air quality standards – The level of pollutants in the air prescribed by regulations that may not be exceeded during a specified time in a defined area. Air quality standards are used to provide a measure of the health-related and visual characteristics of the air.

aquatic – Living or growing in, on, or near water.

aquatic biota – The sum total of living organisms within any designated aquatic area.

aquifer – An underground geological formation, group of formations, or part of a formation that is capable of yielding a significant amount of water to wells or springs.

aquitard - A less-permeable geologic unit that inhibits the flow of water.

archaeological sites (resources) – Any location where humans have altered the terrain or discarded artifacts during either prehistoric or historic times.

artifact – An object produced or shaped by human workmanship of archaeological or historical interest.

as low as is reasonably achievable (ALARA) – An approach to radiation protection to manage and control worker and public exposures (both individual and collective) and releases of radioactive material to the environment to as far below applicable limits as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit but a process for minimizing doses to as far below limits as is practicable.

attainment area – An area that the U.S. Environmental Protection Agency has designated as being in compliance with one or more of the National Ambient Air Quality Standards for sulfur dioxide, nitrogen dioxide, carbon monoxide, ozone, lead, and particulate matter. An area may be in attainment for some pollutants but not for others. (See National Ambient Air Quality Standards, nonattainment area, and particulate matter.)

atmospheric dispersion – The process of air pollutants being dispersed in the atmosphere. This occurs by the wind that carries the pollutants away from their source, by turbulent air motion that results from solar heating of the Earth's surface, and air movement over rough terrain and surfaces.

Atomic Energy Act – A law originally enacted in 1946 and amended in 1954 that placed nuclear production and control of nuclear materials within a civilian agency, originally the Atomic Energy Commission. The functions of the Atomic Energy Commission were replaced by the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy.

Atomic Energy Commission – A five-member commission, established by the Atomic Energy Act of 1946, to supervise nuclear weapons design, development, manufacturing, maintenance, modification, and dismantlement. In 1974, the Atomic Energy Commission was abolished, and all functions were transferred to the Nuclear Regulatory Commission and the Administrator of the Energy Research and Development Administration. The Energy Research and Development Administration was later terminated, and functions vested by law in the Administrator were transferred to the Secretary of Energy.

atomic number – The number of positively charged protons in the nucleus of an atom or the number of electrons on an electrically neutral atom.

background radiation – Radiation from (1) cosmic sources, (2) naturally occurring radioactive materials, including radon (except as a decay product of source or special nuclear material), and (3) global fallout as it exists in the environment (e.g., from the testing of nuclear explosive devices).

badged worker – A worker who has the potential to be exposed to occupational radiation, and is equipped with a dosimeter to measure his/her dose.

barrier – Any material or structure that prevents or substantially delays movement of radionuclides toward the accessible environment.

baseline – The existing environmental conditions against which impacts of the proposed action and its alternatives can be compared. For this NI PEIS, the environmental baseline is the site environmental conditions as they exist or are estimated to exist in the absence of the proposed action.

beam expander – A device designed to expand the proton beam in an accelerator to a larger cross-sectional area.

beamstop – A device designed to absorb the full beam of an accelerator.

becquerel – A unit of radioactivity equal to one disintegration per second. Thirty-seven billion becquerels is equal to 1 curie.

BEIR V – Biological Effects of Ionizing Radiation; referring to the fifth in a series of committee reports from the National Research Council.

benthic – Plants and animals dwelling at the bottom of oceans, lakes, rivers, and other surface waters.

beryllium – An extremely lightweight element with the atomic number 4. It is metallic and is used in reactors as a neutron reflector.

beta emitter – A radioactive substance that decays by releasing a beta particle.

beta particle – A particle emitted in the radioactive decay of many radionuclides. A beta particle is identical to an electron. It has a short range in air and a small ability to penetrate other materials.

beyond-design-basis accident – An accident postulated for the purpose of generating large consequences by exceeding the functional and performance requirements for safety structures, systems, and components. (See design-basis accident.)

beyond-design-basis events – Postulated disturbances in process variables due to external events or multiple component or system failures that can potentially lead to beyond-design-basis accidents. (See design-basis events.)

biodiversity – The diversity of life in all its forms and all levels of organization. Also termed "biological diversity."

biota (biotic) – The plant and animal life of a region (pertaining to biota).

blanket – Blanket assembly in an accelerator is a material where the generated neutrons would be moderated to be absorbed (captured) in the target material to produce a new isotope.

block – U.S. Bureau of the Census term describing small areas bounded on all sides by visible features or political boundaries; used in tabulation of census data.

boron-10 – An isotope of the element boron that has a high capture cross section for neutrons. It is used in reactor absorber rods for reactor control.

bounded – Producing the greatest consequences of any assessment of impacts associated with normal or abnormal operations.

burial ground – With regard to radioactive wastes, a place for burying unwanted (i.e., radioactive) materials in which the Earth acts as a receptacle to prevent the escape of radiation and the dispersion of wastes into the environment.

calcination – A process in which a material is heated to a high temperature to drive off volatile matter (to remove organic material) or to effect changes (as oxidation or pulverization, or to convert to nodular form). The temperature is kept below the fusion point.

cancer – The name given to a group of diseases characterized by uncontrolled cellular growth with cells having invasive characteristics such that the disease can transfer from one organ to another.

canister – A general term for a container, usually cylindrical, used in handling, storage, transportation, or disposal of waste.

capable fault – A fault that has exhibited one or more of the following characteristics: (1) movement at or near the ground surface at least once within the past 35,000 years, or movement of a recurring nature within the past 500,000 years; (2) macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault; (3) a structural relationship to a capable fault according to characteristic (1) or (2) above, such that movement

on one could be reasonably expected to be accompanied by movement on the other.

capacity factor – The ratio of the annual average power production of a power plant to its rated capacity.

carbon dioxide – A colorless, odorless gas that is a normal component of ambient air; it results from fossil fuel combustion, and is an expiration production.

carbon monoxide – A colorless, odorless, poisonous gas produced by incomplete fossil fuel combustion.

cask – A heavily shielded container used to store or ship radioactive materials.

cation – A positively charged ion.

cell – See hot cell.

Chalfont container 9975 – A shielded Type B container with primary and secondary containment features that is used to store or ship radioactive materials. (See cask and Type B packaging.)

cladding – The outer metal jacket of a nuclear fuel element or target. It prevents fuel corrosion and retains fission products during reactor operation and subsequent storage, as well as providing structural support. Zirconium alloys, stainless steel, and aluminum are common cladding materials. In general, a metal coating bonded onto another metal.

Class I areas – A specifically designated area where the degradation of air quality is stringently restricted (e.g., many national parks, wilderness areas). (See Prevention of Significant Deterioration.)

Class II areas – Most of the country not designated as Class I is designated as Class II. Class II areas are generally cleaner than air quality standards require, and moderate increases in new pollution are allowed after a regulatory mandated impacts review.

Clean Air Act – This act mandates and provides for enforcement of regulations to control air pollution from various sources.

Clean Air Act Amendments of 1990 – Expands the U.S. Environmental Protection Agency's enforcement powers, and adds restrictions on air toxics, ozone depleting chemicals, stationary and mobile emissions sources, and emissions implicated in acid rain and global warming.

Clean Water Act of 1972, 1987 – This act regulates the discharge of pollutants from a point source into navigable waters of the United States in compliance with a National Pollutant Discharge Elimination System permit, and regulates discharges to or dredging of wetlands.

Code of Federal Regulations – All Federal regulations in effect are published in codified form in the Code of Federal Regulations.

collective dose – The sum of the individual doses received in a given period of time by a specified population from exposure to a specified source of radiation. Collective dose is expressed in units of person-rem or person-sievert.

commercial light water reactor – The term used to describe commercially operated power producing U.S. reactors that use “light” (as opposed to heavy) water for cooling and neutron moderation.

committed dose equivalent – The dose equivalent to organs or tissues that will be received by an individual during the 50-year period following the intake of radioactive material. It does not include contributions from radiation sources external to the body. Committed dose equivalent is expressed in units of rems or sieverts.

committed effective dose equivalent – The dose value obtained by (1) multiplying the committed dose equivalents for the organs or tissues that are irradiated and the weighting factors applicable to those organs or tissues and (2) summing all the resulting products. Committed effective dose equivalent is expressed in units of rem or sievert.

(See committed dose equivalent and weighting factor.)

community (biotic) – All plants and animals occupying a specific area under relatively similar conditions.

community (environmental justice definition) – A group of people or a site within a spatial scope exposed to risks that potentially threaten health, ecology, or land values; or are exposed to industry that stimulates unwanted noise, smell, industrial traffic, particulate matter, or other nonaesthetic impacts.

conformity – Conformity is defined in the Clean Air Act as the action's compliance with an implementation plan's purpose of eliminating or reducing the severity and number of violations of the National Ambient Air Quality Standards, and achieving expeditious attainment of such standards; and that such activities will not: (1) cause or contribute to any new violation of any standard in any area; (2) increase the frequency or severity of any existing violation of any standard in any area; or (3) delay timely attainment of any standard or any required interim emission reduction, or other milestones in any area.

contact-handled waste – Radioactive waste or waste packages whose external dose rate is low enough to permit contact handling by humans during normal waste management activities, (e.g., waste with a surface dose rate not greater than 200 millirem per hour). (See remote-handled waste.)

container – With regard to radioactive wastes, the metal envelope in the waste package that provides the primary containment function of the waste package, which is designed to meet the containment requirements of 10 CFR Part 60.

containment design basis – For a nuclear reactor, those bounding conditions for the design of the containment, including temperature, pressure, and leakage rate. Because the containment is provided as an additional barrier to mitigate the consequences of accidents involving the release of radioactive materials, the containment design basis may include an additional specified margin above those conditions expected to result from the plant design-basis accidents to ensure that the containment design can mitigate unlikely or unforeseen events.

contamination – The deposition of undesirable radioactive material on the surfaces of structures, areas, objects, or personnel.

control rod – A rod containing material such as boron that is used to control the power of a nuclear reactor. By absorbing excess neutrons, a control rod prevents the neutrons from causing further fissions, i.e., increasing power.

coolant – A substance, either gas or liquid, circulated though a nuclear reactor or processing plant to remove heat.

cooperating agency – Any Federal agency other than a lead agency which has jurisdiction by law or special expertise with respect to any environmental impact involved in a proposal (or a reasonable alternative) for legislation or other major Federal action significantly affecting the quality of the human environment.

credible accident – An accident that has a probability of occurrence greater than or equal to once in a one million year time period.

criteria pollutants – An air pollutant that is regulated by National Ambient Air Quality Standards. The U.S. Environmental Protection Agency must describe the characteristics and potential health and welfare effects that form the basis for setting, or revising, the standard for each regulated pollutant. Criteria pollutants include sulfur dioxide, nitrogen dioxide, carbon monoxide, ozone, lead, and two size classes of particulate matter, less than or equal to 10 micrometers (0.0004 inch) in diameter, and less than or equal to 2.5 micrometers (0.0001 inch) in diameter. New pollutants may be added to, or removed from, the list of criteria pollutants as more information becomes available. (See National Ambient Air Quality Standards.)

critical habitat – Habitat essential to the conservation of an endangered or threatened species that has been designated as critical by the U.S. Fish and Wildlife Service or the National Marine Fisheries Service following the procedures outlined in the Endangered Species Act and its implementing regulations (50 CFR Part 424). (See endangered species and threatened species.) The lists of Critical Habitats can be found in 50 CFR Section 17.95 (fish and wildlife), 50 CFR Section 17.96 (plants), and 50 CFR Part 226 (marine species).

criticality – The condition in which a system is capable of sustaining a nuclear chain reaction.

chain reaction: A reaction that initiates its own repetition. In nuclear fission, a chain reaction occurs when a neutron induces a nucleus to fission and the fissioning nucleus releases one or more neutrons which induce other nuclei to fission.

critical mass: The smallest mass of fissionable material that will support a self-sustaining nuclear chain reaction.

cultural resources – Archaeological sites, historical sites, architectural features, traditional use areas, and Native American sacred sites.

cumulative impacts – The impacts on the environment that result from the incremental impacts of the action when added to other past, present, and reasonably foreseeable future actions, regardless of the agency or person who undertakes such other actions. Cumulative impacts can result from individually minor but collectively significant actions taking place over a period of time. (40 CFR Section 1508.7)

curie – A unit of radioactivity equal to 37 billion disintegrations per second (i.e., 37 billion becquerels); also a quantity of any radionuclide or mixture of radionuclides having 1 curie of radioactivity.

day-night average sound level – The 24-hour, A-weighted equivalent sound level expressed in decibels. A 10-decibel penalty is added to sound levels between 10:00 p.m. and 7:00 a.m. to account for increased annoyance due to noise during night hours.

deactivation – The placement of a facility in a radiologically and industrially safe shutdown condition that is suitable for a long-term surveillance and maintenance phase prior to final decontamination and decommissioning.

decay (radioactive) – The decrease in the amount of any radioactive material with the passage of time, due to spontaneous nuclear disintegration (i.e., emission from atomic nuclei of charged particles, photons, or both).

decay heat (radioactivity) – The heat produced by the decay of radionuclides.

decibel – A unit for expressing the relative intensity of sounds on a logarithmic scale where zero is below human perception and 130 is above the threshold of pain to humans. For traffic and industrial noise measurements, the A-weighted decibel, a frequency-weighted noise unit, is widely used. The A-weighted decibel scale corresponds approximately to the frequency response of the human ear and thus correlates well with loudness.

deciduous – Trees which shed leaves at a certain season.

decommissioning – Retirement of a facility, including any necessary decontamination and/or dismantlement.

decontamination – The actions taken to reduce or remove substances that pose a substantial present or potential hazard to human health or the environment, such as radioactive or chemical contamination from facilities, equipment, or soils by washing, heating, chemical or electrochemical action, mechanical cleaning, or other techniques.

depleted uranium – Uranium whose content of the fissile isotope uranium-235 is less than the 0.7 percent (by weight) found in natural uranium, so that it contains more uranium-238 than natural uranium.

deposition – In geology, the laying down of potential rock-forming materials; sedimentation. In atmospheric transport, the settling out on ground and building surfaces of atmospheric aerosols and particles ("dry deposition"), or their removal from the air to the ground by precipitation ("wet deposition" or "rainout").

design basis – For nuclear facilities, information that identifies the specific functions to be performed by a structure, system, or component, and the specific values (or ranges of values) chosen for controlling parameters for reference bounds for design. These values may be: (1) restraints derived from generally accepted state-of-the-art practices for achieving functional goals; (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals; or (3) requirements derived from Federal safety objectives, principles, goals, or requirements.

design-basis accident – An accident postulated for the purpose of establishing functional and performance requirements for safety structures, systems, and components. (See beyond-design-basis accident.)

design-basis events – Postulated disturbances in process variables that can potentially lead to design-basis accidents. (See beyond-design-basis events.)

direct jobs – The number of workers required at a site to implement an alternative.

disposition – The ultimate "fate" or end use of a surplus U.S. Department of Energy facility following the transfer of the facility to the Office of the Assistant Secretary for Environmental Waste Management.

DOE orders – Requirements internal to the U.S. Department of Energy that establish Department policy and procedures, including those for compliance with applicable laws.

dose (or radiation dose) – A generic term that means absorbed dose, effective dose equivalent, committed effective dose equivalent, or total effective dose equivalent, as defined elsewhere in this glossary.

dose equivalent – A measure of radiological dose that correlates with biological effect on a common scale for all types of ionizing radiation. Defined as a quantity equal to the absorbed dose in tissue multiplied by a quality factor (the biological effectiveness of a given type of radiation) and all other necessary modifying factors at the location of interest. The units of dose equivalent are the rem and sievert.

dose rate – The radiation dose delivered per unit of time (e.g., rem per year).

dosimeter – A small device (instrument) carried by a radiation worker that measures cumulative radiation dose (e.g., a film badge or ionization chamber).

drinking water standards – The level of constituents or characteristics in a drinking water supply specified in regulations under the Safe Drinking Water Act as the maximum permissible.

ecology – A branch of science dealing with the interrelationships of living organisms with one another and with their nonliving environment.

ecosystem – A community of organisms and their physical environment interacting as an ecological unit.

effective dose equivalent – The dose value obtained by multiplying the dose equivalents received by specified tissues or organs of the body by the appropriate weighting factors applicable to the tissues or organs irradiated, and then summing all of the resulting products. It includes the dose from radiation sources internal and external to the body. The effective dose equivalent is expressed in units of rems or sieverts. (See committed dose equivalent and committed effective dose equivalent.)

effluent – A waste stream flowing into the atmosphere, surface water, ground water, or soil.

Most frequently the term applies to wastes discharged to surface waters.

electron – An elementary particle with a mass of 9.107×10^{-28} gram (or 1/1,837 of a proton) and a negative charge. Electrons surround the positively charged nucleus and determine the chemical properties of the atom.

emission – A material discharged into the atmosphere from a source operation or activity.

emission standards – Legally enforceable limits on the quantities and/or kinds of air contaminants that can be emitted into the atmosphere.

endangered species – Plants or animals that are in danger of extinction through all or a significant portion of their ranges and that have been listed as endangered by the U.S. Fish and Wildlife Service or the National Marine Fisheries Service following the procedures outlined in the Endangered Species Act and its implementing regulations (50 CFR Part 424). (See threatened species.) The lists of endangered species can be found in 50 CFR Section 17.11 for wildlife, 50 CFR Section 17.12 for plants, and 50 CFR Section 222.23(a) for marine organisms.

engineered safety features – For a nuclear facility, features that prevent, limit, or mitigate the release of radioactive material from its primary containment.

enriched uranium – Uranium whose content of the fissile isotope uranium-235 is greater than the 0.7 percent (by weight) found in natural uranium. (See uranium, natural uranium, and highly enriched uranium.)

entrainment – The involuntary capture and inclusion of organisms in streams of flowing water, a term often applied to the cooling water systems of power plants and reactors. The organisms involved may include phyto-and zooplankton, fish eggs and larvae (ichthyoplankton), shellfish larvae, and other forms of aquatic life.

Environment, Safety, and Health Program – In the context of the U.S. Department of Energy, encompasses those requirements, activities, and functions in the conduct of all Department and Department-controlled operations that are concerned with: impacts to the biosphere; compliance with environmental laws, regulations, and standards controlling air, water, and soil pollution; limiting the risks to the well-being of both the operating personnel and the general public; and protecting property against accidental loss and damage. Typical activities and functions related to this program include, but are not limited to, environmental protection, occupational safety, fire protection, industrial hygiene, health physics, occupational medicine, process and facilities safety, nuclear safety, emergency preparedness, quality assurance, and radioactive and hazardous waste management.

environmental impact statement – The detailed written statement that is required by section 102(2)(C) of the National Environmental Policy Act for a proposed major Federal action significantly affecting the quality of the human environment. A U.S. Department of Energy environmental impact statement is prepared in accordance with applicable requirements of the Council on Environmental Quality National Environmental Policy Act regulations in 40 CFR Parts 1500–1508 and the U.S. Department of Energy National Environmental Policy Act regulations in 10 CFR Part 1021. The statement includes, among other information, discussions of the environmental impacts of the proposed action and all reasonable alternatives, adverse environmental effects that can not be avoided should the proposal be implemented, the relationship between short-term uses of the human environment and enhancement of long-term productivity, and any irreversible and irretrievable commitments of resources.

environmental justice – The fair treatment and meaningful involvement of all people regardless of race, color, national origin, or income with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies. Fair treatment means that no group of people, including racial, ethnic, or socioeconomic groups, should bear a disproportionate share of the negative

environmental consequences resulting from industrial, municipal, and commercial operations or the execution of Federal, state, local, and tribal programs and policies. Executive Order 12898 directs Federal agencies to make achieving environmental justice part of their missions by identifying and addressing disproportionately high and adverse effects of agency programs, policies, and activities on minority and low-income populations. (See minority population and low-income population.)

epidemiology – Study of the occurrence, causes, and distribution of disease or other health-related states and events in human populations, often as related to age, sex, occupation, ethnic, and economic status, in order to identify and alleviate health problems and promote better health.

exposure limit – The level of exposure to a hazardous chemical (set by law or a standard) at which or below which adverse human health effects are not expected to occur.

Reference dose is the chronic-exposure dose (milligram or kilogram per day) for a given hazardous chemical at which or below which adverse human non-cancer health effects are not expected to occur.

Reference concentration is the chronic exposure concentration (milligram per cubic meter) for a given hazardous chemical at which or below which adverse human non-cancer health effects are not expected to occur.

extrusion – A type of process in which a material (e.g., metal or plastic) is forced through a die or very small hole to give it a certain shape.

Fast Flux Test Facility – A liquid-metal (sodium) cooled test reactor and moderated test reactor at the Hanford Site. It is fueled with a mixture of plutonium-uranium dioxide, and has a power level of 400 megawatts. It is presently in a standby status.

fault – A fracture or a zone of fractures within a rock formation along which vertical, horizontal, or transverse slippage has occurred. A normal fault occurs when the hanging wall has been depressed in relation to the footwall. A reverse fault occurs when the hanging wall has been raised in relation to the footwall.

fissile materials – Although sometimes used as a synonym for fissionable material, this term has acquired a more restricted meaning, namely, any material fissionable by thermal (slow) neutrons. The three primary fissile materials are uranium-233, uranium-235, and plutonium-239.

fission products – Nuclei (fission fragments) formed by the fission of heavy elements, plus the nuclides formed by the fission fragments' radioactive decay.

fissionable material – Commonly used as a synonym for fissile material, the meaning of this term has been extended to include material that can be fissioned by fast neutrons, such as uranium-238.

floodplain – The lowlands and relatively flat areas adjoining inland and coastal waters and the flood prone areas of offshore islands. Floodplains include, at a minimum, that area with at least a 1.0 percent chance of being inundated by a flood in any given year.

The *base floodplain* is defined as the area which has a 1.0 percent or greater chance of being flooded in any given year. Such a flood is known as a 100-year flood.

The *critical action floodplain* is defined as the area which has at least a 0.2 percent chance of being flooded in any given year. Such a flood is known as a 500-year flood. Any activity for which even a slight chance of flooding would be too great (e.g., the storage of highly volatile, toxic, or water reactive materials) should not occur in the critical action floodplain.

The *probable maximum flood* is the hypothetical flood that is considered to be the most severe reasonably possible flood, based on the comprehensive hydrometeorological application of maximum precipitation and other hydrological factors favorable for maximum flood runoff (e.g., sequential storms and snowmelts). It is usually

several times larger than the maximum recorded flood.

Fluorinel Dissolution Process Facility – A processing facility at the Idaho Nuclear Technology and Engineering Center on the Idaho National Engineering and Environmental Laboratory that is designed to handle highly radioactive material using remote handling equipment. This facility was originally intended to process spent nuclear fuel.

flux – Rate of flow through a unit area; in reactor operation, the apparent flow of neutrons in a defined energy range (see neutron flux).

formation – In geology, the primary unit of formal stratigraphic mapping or description. Most formations possess certain distinctive features.

fuel assembly – A cluster of fuel rods or plates. Also called a fuel element. Approximately 200 fuel assemblies make up a reactor core.

fuel rod – A nuclear reactor component that includes the fissile material.

Fuels and Materials Examination Facility – A processing facility at the Hanford Site that is designed to handle highly radioactive materials using remote handling equipment. This facility was originally intended to process spent nuclear fuel and irradiated targets from the Fast Flux Test Facility.

fugitive emissions – (1) Emissions that do not pass through a stack, vent, chimney, or similar opening where they could be captured by a control device. (2) Any air pollutant emitted to the atmosphere other than from a stack. Sources of fugitive emissions include pumps; valves; flanges; seals; area sources such as ponds, lagoons, landfills, piles of stored material (e.g., coal); and road construction areas or other areas where earthwork is occurring.

gamma radiation – High-energy, short wavelength, electromagnetic radiation emitted from the nucleus of an atom during radioactive decay. Gamma radiation frequently accompanies alpha and beta emissions and always accompanies fission. Gamma rays are very penetrating and are best stopped or shielded by dense materials, such as lead or depleted uranium. Gamma rays are similar to, but are usually more energetic than, x-rays.

genetic effects – Inheritable changes (chiefly mutations) produced by exposure, to ionizing radiation or other chemical or physical agents, of the parts of cells that control biological reproduction and inheritance.

geologic repository – A place to dispose of radioactive waste deep beneath the Earth's surface.

geology – The science that deals with the Earth: the materials, processes, environments, and history of the planet, including rocks and their formation and structure.

gigaelectron volts – 1,000 million electron volts (MeV). (See MeV.)

glovebox – Large enclosure that separates workers from equipment used to process hazardous material, while allowing the workers to be in physical contact with the equipment; normally constructed of stainless steel, with large acrylic/lead glass windows. Workers have access to equipment through the use of heavy-duty, lead-impregnated rubber gloves, the cuffs of which are sealed in portholes in the glovebox windows.

gray – The SI (International System of Units) unit of absorbed dose. One gray is equal to an absorbed dose of 1 joule per kilogram (1 gray is equal to 100 rads). (The joule is the SI unit of energy.) (See absorbed dose.)

ground shine – The radiation dose received from an area on the ground where radioactivity has been deposited by a radioactive plume or cloud.

groundwater – Water below the ground surface in a zone of saturation.

habitat – The environment occupied by individuals of a particular species, population, or community.

half-life – The time in which one-half of the atoms of a particular radioactive isotope disintegrate to another nuclear form. Half-lives vary from millionths of a second to billions of years.

Hazard Index – A summation of the Hazard Quotients for all chemicals now being used at a site, and those proposed to be added, to yield cumulative levels for a site. A Hazard Index value of 1.0 or less means that no adverse human health effects (noncancer) are expected to occur.

Hazard Quotient – The value used as an assessment of non-cancer associated toxic effects of chemicals, e.g., kidney or liver dysfunction. It is a ratio of the estimated exposure to that exposure at which it would be expected that adverse health effects would begin to be produced. It is independent of a cancer risk, which is calculated only for those chemicals identified as carcinogens.

hazardous air pollutants – Air pollutants not covered by ambient air quality standards but which may present a threat of adverse human health effects or adverse environmental effects. Those specifically listed in 40 CFR Section 61.01 are asbestos, benzene, beryllium, coke oven emissions, inorganic arsenic, mercury, radionuclides, and vinyl chloride. More broadly, hazardous air pollutants are any of the 189 pollutants listed in or pursuant to section 112(b) of the Clean Air Act. Very generally, hazardous air pollutants are any air pollutants that may realistically be expected to pose a threat to human health or welfare.

hazardous chemical – Under 29 CFR Part 1910, Subpart Z, hazardous chemicals are defined as "any chemical which is a physical hazard or a health hazard." Physical hazards include combustible liquids, compressed gases, explosives, flammables, organic peroxides, oxidizers, pyrophorics, and reactives. A health hazard is any chemical for which there is good evidence that acute or chronic health effects occur in exposed employees. Hazardous chemicals include carcinogens, toxic or highly toxic agents, reproductive toxins, irritants, corrosives, sensitizers, hepatotoxins, nephrotoxins, agents that act on the hematopoietic system, and agents that damage the lungs, skin, eyes, or mucous membranes.

hazardous material – A material, including a hazardous substance, as defined by 49 CFR Section 171.8, which poses a risk to health, safety, and property when transported or handled.

hazardous substance – Any substance subject to the reporting and possible response provisions of the Clean Water Act and the Comprehensive Environmental Response, Compensation, and Liability Act.

hazardous waste – A category of waste regulated under the Resource Conservation and Recovery Act. To be considered hazardous, a waste must be a solid waste under the Resource Conservation and Recovery Act and must exhibit at least one of four characteristics described in 40 CFR Section 261.20 through 40 CFR Section 261.24 (i.e., ignitability, corrosivity, reactivity, or toxicity) or be specifically listed by the U.S. Environmental Protection Agency in 40 CFR Section 261.31 through 40 CFR Section 261.33.

high-efficiency particulate air filter – An air filter capable of removing at least 99.97 percent of particles 0.3 micrometer (about 0.00001 inch) in diameter. These filters include a pleated fibrous medium (typically fiberglass) capable of capturing very small particles.

High Flux Isotope Reactor – A light-water cooled and moderated test reactor in the Oak Ridge National Laboratory area of the Oak Ridge Reservation. It is fueled with uranium highly enriched with uranium-235 and has a full authorized power level of 85 million watts.

high-level radioactive waste – High-level waste is the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation.

HIGHWAY – A computer code used for predicting routes for transporting radioactive material in the United States and calculating route-specific population density statistics.

highly enriched uranium – Uranium whose content of the fissile isotope uranium-235 has been increased through enrichment to 20 percent or more (by weight). (See natural uranium, enriched uranium, and depleted uranium.)

historic resources – Archaeological sites, architectural structures, and objects produced after the advent of written history, dating to the time of the first Euro-American contact in an area.

hot cell – A shielded facility that requires the use of remote manipulators for handling radioactive materials.

hydrology – The science dealing with the properties, distribution, and circulation of natural water systems.

impingement – The process by which aquatic organisms too large to pass through the screens of a water intake structure become caught on the screens and are unable to escape.

incident-free risk – The radiological or chemical impacts resulting from emissions during normal operations and packages aboard vehicles in normal transport. This includes the radiation or hazardous chemical exposure of specific population groups and workers.

indirect jobs – Within a regional economic area, jobs generated or lost in related industries as a result of a change in direct employment.

injection wells – A well that takes water from the surface into the ground, either through gravity or by mechanical means.

injector – A device that provides protons for an accelerator by heating hydrogen gas to a plasma state in which the hydrogen atoms lose their electrons, thereby giving the hydrogen nuclei (protons) a positive charge. An electric voltage removes the protons from the injector.

ion – An atom that has too many or too few electrons, causing it to be electrically charged.

ion exchange – A unit physiochemical process that removes anions and cations, including radionuclides, from liquid streams (usually water) for the purpose of purification or decontamination.

ion exchange resin – An organic polymer that functions as an acid or base. These resins are used to remove ionic material from a solution. Cation exchange resins are used to remove positively charged particles (cations), and anion exchange resins are used to remove negatively charged particles (anions).

ionizing radiation – Alpha particles, beta particles, gamma rays, high-speed electrons, high-speed protons, and other particles or electromagnetic radiation that can displace electrons from atoms or molecules, thereby producing ions.

irradiated – Exposure to ionizing radiation. The condition of reactor fuel elements and other materials in which atoms bombarded with nuclear particles have undergone nuclear changes.

isotopes – Any of two or more variations of an element in which the nuclei have the same number of protons (i.e., the same atomic number) but different numbers of neutrons so that their atomic masses differ. Isotopes of a single element possess almost identical chemical properties, but often different physical properties (e.g., carbon-12 and -13 are stable, carbon-14 is radioactive).

joule – A metric unit of energy, work, or heat, equivalent to one watt-second, 0.737 foot-pound, or 0.239 calories.

landscape character – The arrangement of a particular landscape as formed by the variety and intensity of the landscape features (land, water, vegetation, and structures) and the four basic elements (form, line, color, and texture). These factors give an area a distinctive quality that distinguishes it from its immediate surroundings.

latent cancer fatalities – Deaths from cancer occurring some time after, and postulated to be due to, exposure to ionizing radiation or other carcinogens.

license amendment – Changes to an existing reactor's operating license that are approved by the U.S. Nuclear Regulatory Commission.

light water – The common form of water (a molecule with two hydrogen atoms and one oxygen atom, H_2O), in which the hydrogen atom consists completely of the normal hydrogen isotope (one proton).

light water reactor – A nuclear reactor in which circulating light water is used to cool the reactor core and to moderate (reduce the energy of) the neutrons created in the core by the fission reactions.

long-lived radionuclides – Radioactive isotopes with half-lives greater than 30 years.

loss-of-coolant accident – An accident that results from the loss of reactor coolant because of a break in the reactor coolant system.

low-enriched uranium – Uranium whose content of the fissile isotope uranium-235 has been increased through enrichment to more than 0.7 percent but less than 20 percent by weight. Most nuclear power reactor fuel contains low-enriched uranium containing 3 to 5 percent uranium-235.

low-income population – Low-income populations, defined in terms of Bureau of the Census annual statistical poverty levels (Current Population Reports, Series P-60 on Income and Poverty), may consist of groups or individuals who live in geographic proximity to one another or who are geographically dispersed or transient (such as migrant workers or Native Americans), where either type of group experiences common conditions of environmental exposure or effect. (See environmental justice and minority population.)

low-level radioactive waste – Waste that contains radioactivity but is not classified as high-level waste, transuranic waste, spent nuclear fuel, or by-product material as defined by Section 11e (2) of the Atomic Energy Act of 1954, as amended. Test specimens of fissionable material irradiated for research and development only, and not for the production of power or plutonium, may be classified as low-level waste, provided the concentration of transuranic waste is less than 100 nanocuries per gram.

maximally exposed offsite individual – A hypothetical individual whose location and habits result in the highest total radiological or chemical exposure (and thus dose) from a particular source for all exposure routes (e.g., inhalation, ingestion, direct exposure).

maximum contaminant level – The designation for U.S. Environmental Protection Agency standards for drinking water quality under the Safe Drinking Water Act. The maximum contaminant level for a given substance is the maximum permissible concentration of that substance in water delivered by a public water system. The primary maximum contaminant levels (40 CFR Part 141) are intended to protect public health and are federally enforceable. They are based on health factors, but are also required by law to reflect the technological and economic feasibility of removing the contaminant from the water supply. Secondary maximum contaminant levels (40 CFR Part 143) are set by the U.S. Environmental Protection Agency to protect the public welfare. The secondary drinking water regulations control substances in drinking water that primarily affect aesthetic qualities (such as taste, odor, and color) relating to the public acceptance of water. These regulations are not federally enforceable, but are intended as guidelines for the states.

megawatt – A unit of power equal to 1 million watts. Megawatt thermal is commonly used to define heat produced, while megawatt electric defines electricity produced.

meteorology – The science dealing with the atmosphere and its phenomena, especially as relating to weather.

MeV (million electron volts) – A unit used to quantify energy. In this NI PEIS, it describes a particle's kinetic energy, which is an indicator of particle speed.

micron – One-millionth of one meter.

migration – The natural movement of a material through the air, soil, or groundwater; also, seasonal movement of animals from one area to another.

Migratory Bird Treaty Act – This act states that it is unlawful to pursue, take, attempt to take, capture, possess, or kill any migratory bird, or any part, nest, or egg of any such bird other than permitted activities.

millirem – One-thousandth of one rem.

minority population – Minority populations exist where either: (a) the minority population of the affected area exceeds 50 percent or (b) the minority population percentage of the affected area is meaningfully greater than in the general population or other appropriate unit of geographic analysis (such as a governing body's jurisdiction, a neighborhood, census tract, or other similar unit). "Minority" refers to individuals who are members of the following population groups: American Indian or Alaskan Native; Asian or Pacific Islander; Black, not of Hispanic origin; or Hispanic. "Minority populations" include either a single minority group or the total of all minority persons in the affected area. They may consist of groups of individuals living in geographic proximity to one another or a geographically dispersed/transient set of individuals (such as migrant workers or Native Americans), where either type of group experiences common conditions of environmental exposure or effect. (See environmental justice and low-income population.)

mitigate – Mitigation includes: (1) avoiding an impact altogether by not taking a certain action or parts of an action; (2) minimizing impacts by limiting the degree or magnitude of an action and its implementation; (3) rectifying an impact by repairing, rehabilitating, or restoring the affected environment; (4) reducing or eliminating the impact over time by preservation and maintenance operations during the life of an action; or (5) compensating for an impact by replacing or providing substitute resources or environments.

mixed oxide fuel – Reactor fuel made with a physical blend of different fissionable materials, such as uranium dioxide and plutonium dioxide.

mixed waste – Waste that contains both nonradioactive hazardous waste and radioactive waste, as defined in this glossary.

moderator – A material used to decelerate neutrons in a reactor from high energies to low energies.

molar – A chemical term relating to the mole, or gram-molecular weight. A 1-molar solution would have 1 mole of solute per liter of solution.

National Ambient Air Quality Standards – Standards defining the highest allowable levels of certain pollutants in the ambient air (i.e., the outdoor air to which the public has access). Because the U.S. Environmental Protection Agency must establish the criteria for setting these standards, the regulated pollutants are called *criteria* pollutants. Criteria pollutants include sulfur dioxide, nitrogen dioxide, carbon monoxide, ozone, lead, and two size classes of particulate matter (less than or equal to 10 micrometers [0.0004 inch] in diameter and less than or equal to 2.5 micrometers [0.0001 inch] in diameter). Primary standards are established to protect public health; secondary standards are established to protect public welfare (e.g., visibility, crops, animals, buildings). (See criteria pollutant.)

National Emission Standards for Hazardous Air Pollutants – Emissions standards set by the U.S. Environmental Protection Agency for air pollutants which are not covered by National Ambient Air Quality Standards and which may, at sufficiently high levels, cause increased fatalities, irreversible health effects, or incapacitating illness.

These standards are given in 40 CFR Parts 61 and 63. National Emission Standards for Hazardous Air Pollutants are given for many specific categories of sources (e.g., equipment leaks, industrial process cooling towers, dry cleaning facilities, petroleum refineries). (See hazardous air pollutants.)

National Environmental Policy Act of 1969 – This act is the basic national charter for protection of the environment. It establishes policy, sets goals (Section 101), and provides means (Section 102) for carrying out policy. Section 102(2) contains "action-forcing" provisions to ensure that Federal agencies follow the letter and spirit of the act. For major Federal actions significantly affecting the quality of the human environment, Section 102(2)(C) of the National Environmental Policy Act requires Federal agencies to prepare a detailed statement that includes the environmental impacts of the proposed action and other specified information.

National Historic Preservation Act – This act provides that property resources with significant national historic value be placed on the National Register of Historic Places. It does not require any permits, but pursuant to Federal code, if a proposed action might impact a historic property resource, it mandates consultation with the proper agencies.

National Pollutant Discharge Elimination System – A provision of the Clean Water Act which prohibits discharge of pollutants into waters of the United States unless a special permit is issued by the U.S. Environmental Protection Agency, a state, or, where delegated, a tribal government on an Indian reservation. The National Pollutant Discharge Elimination System permit lists either permissible discharges, the level of cleanup technology required for wastewater, or both.

National Register of Historic Places – The official list of the Nation's cultural resources that are worthy of preservation. The National Park Service maintains the list under direction of the Secretary of the Interior. Buildings, structures, objects, sites, and districts are included in the National Register for their importance in American history, architecture, archeology, culture, or engineering. Properties included on the National Register range from large-scale, monumentally proportioned buildings to smaller scale, regionally distinctive buildings. The listed properties are not just of nationwide importance; most are significant primarily at the state or local level. Procedures for listing properties on the National Register are found in 36 CFR Part 60.

natural phenomena accidents – Accidents that are initiated by phenomena such as earthquakes, tornadoes, floods, etc.

natural uranium – Uranium with the naturally occurring distribution of uranium isotopes (approximately 0.7-weight percent uranium-235, and the remainder essentially uranium-238). (See uranium, depleted uranium, enriched uranium, highly enriched uranium, and low-enriched uranium.)

neptunium – An element, mostly manmade, with the atomic number 93. Pure neptunium is a silvery metal. The neptunium-237 isotope has a half-life of 2.14 million years. When neptunium-237 is bombarded by neutrons, it is transformed to neptunium-238, which in turn undergoes radioactive decay to become plutonium-238. When neptunium-237 undergoes radioactive decay, it emits alpha particles and gamma rays.

neutron – An uncharged elementary particle with a mass slightly greater than that of the proton. Neutrons are found in the nucleus of every atom heavier than hydrogen-1.

neutron flux – The product of neutron number density and velocity (energy), giving an apparent number of neutrons flowing through a unit area per unit time.

nitrogen – A natural element with the atomic number 7. It is diatomic in nature and is a colorless and odorless gas that constitutes about four-fifths of the volume of the atmosphere.

nitrogen oxides – Refers to the oxides of nitrogen, primarily nitrogen oxide and nitrogen dioxide.

These are produced in the combustion of fossil fuels and can constitute an air pollution problem. Nitrogen dioxide emissions contribute to acid deposition and formation of atmospheric ozone.

noise – Any sound that is undesirable because it interferes with speech and hearing, or is intense enough to damage hearing, or is otherwise annoying or undesirable.

nonattainment area – An area that the U.S. Environmental Protection Agency has designated as not meeting (i.e., not being in attainment of) one or more of the National Ambient Air Quality Standards for sulfur dioxide, nitrogen dioxide, carbon monoxide, ozone, lead, and particulate matter. An area may be in attainment for some pollutants, but not for others. (See attainment area, National Ambient Air Quality Standards, and particulate matter.)

normal operation – All normal (incident-free) conditions and those abnormal conditions that frequency estimation techniques indicate occur with a frequency greater than 0.1 events per year.

Notice of Intent – Announces the scoping process. The Notice of Intent is usually published in the Federal Register and a local newspaper. The scoping process includes holding at least one public meeting and requesting written comments on issues and environmental concerns that an environmental impact statement should address.

nuclear criticality – See criticality.

nuclear facility – A facility that is subject to requirements intended to control potential nuclear hazards. Defined in DOE directives as any nuclear reactor or any other facility whose operations involve radioactive materials in such form and quantity that a significant nuclear hazard potentially exists to the employees or the general public.

nuclear fuel cycle – The path followed by the nuclear fuel in its various states from mining the ore to waste disposal. The basic fuel materials for the generation of nuclear power are the elements uranium and thorium.

nuclear grade – Material of a quality adequate for use in a nuclear application.

nuclear material – Composite term applied to: (1) special nuclear material; (2) source material such as uranium or thorium or ores containing uranium or thorium; and (3) by-product material, which is any radioactive material that is made radioactive by exposure to the radiation incident to the process of producing or using special nuclear material.

nuclear radiation – Particles (alpha, beta, neutrons) or photons (gamma) emitted from the nucleus of unstable radioactive atoms as a result of radioactive decay.

nuclear reactor – A device that sustains a controlled nuclear fission chain reaction that releases energy in the form of heat.

Nuclear Regulatory Commission – The Federal agency that regulates the civilian nuclear power industry in the United States.

nuclide – A species of atom characterized by the constitution of its nucleus and hence by the number of protons, the number of neutrons, and the energy content.

Occupational Safety and Health Administration – Oversees and regulates workplace health and safety; created by the Occupational Safety and Health Act of 1970.

offsite – The term denotes a location, facility, or activity occurring outside of the site boundary.

outfall – The discharge point of a drain, sewer, or pipe as it empties into a body of water.

ozone – The triatomic form of oxygen; in the stratosphere, ozone protects the Earth from the sun's ultraviolet rays, but in lower levels of the atmosphere, ozone is considered an air pollutant.

package – For radioactive materials, the packaging, together with its radioactive contents, as presented for transport (the packaging plus the radioactive contents equals the package).

packaging – With regard to hazardous or radionuclide materials, the assembly of components necessary to ensure compliance with Federal regulations. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle tie-down system and auxiliary equipment may be designated as part of the packaging.

particulate matter – Any finely divided solid or liquid material, other than uncombined (i.e., pure) water. A subscript denotes the upper limit of the diameter of particles included. Thus, PM_{10} includes only those particles equal to or less than 10 micrometers (0.0004 inch) in diameter; $PM_{2.5}$ includes only those particles equal to or less than 2.5 micrometers (0.0001 inch) in diameter.

permeability – In geology, the ability of rock or soil to transmit a fluid.

person-rem – A unit of collective radiation dose applied to populations or groups of individuals (see collective dose); that is, a unit for expressing the dose when summed across all persons in a specified population or group. One person-rem equals 0.01 person-sieverts.

phenolic protective coating – A coating material made from the chemical, phenol.

plume – The elongated volume of contaminated water or air originating at a pollutant source such as an outlet pipe or a smokestack. A plume eventually diffuses into a larger volume of less contaminated material as it is transported away from the source.

plutonium – A heavy, radioactive, metallic element with the atomic number 94. It is produced artificially by neutron bombardment of uranium. Plutonium has 15 isotopes with atomic masses ranging from 232 to 246 and half-lives from 20 minutes to 76 million years.

plutonium-238 – An isotope with a half-life of 87.74 years used as the heat source for radioisotope power systems. When plutonium-238 undergoes radioactive decay, it emits alpha particles and gamma rays.

plutonium-239 – An isotope with a half-life of 24,110 years and is the primary radionuclide in weapons-grade plutonium. When plutonium-239 decays, it emits alpha particles.

population dose – See collective dose.

pounds per square inch – A measure of pressure; atmospheric pressure is about 14.7 pounds per square inch.

Prevention of Significant Deterioration – Regulations established to prevent significant deterioration of air quality in areas that already meet National Ambient Air Quality Standards. Specific details of Prevention of Significant Deterioration are found in 40 CFR Section 51.166. Among other provisions, cumulative increases in sulfur dioxide, nitrogen dioxide, and PM_{10} levels after specified baseline dates must not exceed specified maximum allowable amounts. These allowable increases, also known as increments, are especially stringent in areas designated as Class I areas (e.g., national parks, wilderness areas) where the preservation of clean air is particularly important. All areas not designated as Class I are currently designated as Class II. Maximum increments in pollutant levels are also given in 40 CFR Section 51.166 for Class III areas, if any such areas should be so designated by EPA. Class III increments are less stringent than those for Class I or Class II areas. (See National Ambient Air Quality Standards.)

primary system – With regard to nuclear reactors, the system that circulates a coolant (e.g., water) through the reactor core to remove the heat of reaction.

prime farmland – Land that has the best combination of physical and chemical characteristics for producing food, feed, fiber, forage, oil-seed, and other agricultural crops with minimum inputs of fuel, fertilizer, pesticides, and labor, without intolerable soil erosion, as determined by the Secretary of Agriculture

(Farmland Protection Act of 1981, 7 CFR Part 7, paragraph 658).

probabilistic risk assessment – A comprehensive, logical, and structured methodology that accounts for population dynamics and human activity patterns at various levels of sophistication, considering time-space distributions and sensitive subpopulations. The probabilistic method results in a more complete characterization of the exposure information available, which is defined by probability distribution functions. This approach offers the possibility of an associated quantitative measure of the uncertainty around the value of interest.

process – Any method or technique designed to change the physical or chemical character of the product.

protactinium – An element that is produced by the radioactive decay of neptunium-237. The pure metal has a bright metallic luster. The protactinium-233 isotope has a half-life of 27 days and emits beta particles and gamma rays during radioactive decay.

proton – An elementary nuclear particle with a positive charge equal in magnitude to the negative charge of the electron; it is a constituent of all atomic nuclei, and the atomic number of an element indicates the number of protons in the nucleus of each atom of that element.

PUREX – An acronym for Plutonium-Uranium Extraction, the name of the chemical process usually used to remove plutonium and uranium from spent nuclear fuel, irradiated targets, and other nuclear materials.

purpose-built vessel – A vessel specifically designed to carry nuclear fuel casks.

rad – See radiation absorbed dose.

radiation (ionizing) – See ionizing radiation.

radiation absorbed dose (rad) – The basic unit of absorbed dose equal to the absorption of 0.01 joule per kilogram (100 ergs per gram) of absorbing material.

radioactive waste – In general, waste that is managed for its radioactive content. Waste material that contains source, special nuclear, or by-product material is subject to regulation as radioactive waste under the Atomic Energy Act. Also, waste material that contains accelerator-produced radioactive material or a high concentration of naturally occurring radioactive material may be considered radioactive waste.

radioactivity – *Defined as a process:* The spontaneous transformation of unstable atomic nuclei, usually accompanied by the emission of ionizing radiation.

Defined as a property: The property of unstable nuclei in certain atoms to spontaneously emit ionizing radiation during nuclear transformations.

Radiochemical Engineering Development Center – A chemical processing facility at the Oak Ridge National Laboratory used for processing highly radioactive materials in hot cells using remote handling equipment. The REDC complex consists of Buildings 7920 and 7930.

radioisotope or radionuclide – An unstable isotope that undergoes spontaneous transformation, emitting radiation. (See isotopes.)

radon – A gaseous, radioactive element with the atomic number 86, resulting from the radioactive decay of radium. Radon occurs naturally in the environment and can collect in unventilated enclosed areas, such as basements. Large concentrations of radon can cause lung cancer in humans.

RADTRAN – A computer code combining user-determined meteorological, demographic, transportation, packaging, and material factors with health physics data to calculate the expected radiological consequences and accident risk of transporting radioactive material.

reactor accident – See design-basis accident and severe accident.

reactor coolant system – The system used to transfer energy from the reactor core either directly or indirectly to the heat rejection system.

reactor core – The fuel assemblies, fuel and target rods, control rods, blanket assemblies, and coolant/moderator. Fissioning takes place in this part of the reactor.

reactor facility – Unless it is modified by words such as containment, vessel, or core, the term reactor facility includes the housing, equipment, and associated areas devoted to the operation and maintenance of one or more reactor cores. Any apparatus that is designed or used to sustain nuclear chain reactions in a controlled manner, including critical and pulsed assemblies and research, tests, and power reactors, is defined as a reactor. All assemblies designed to perform subcritical experiments that could potentially reach criticality are also considered reactors.

Record of Decision – A document prepared in accordance with the requirements of 40 CFR Section 1505.2 and 10 CFR Section 1021.315 that provides a concise public record of the U.S. Department of Energy's decision on a proposed action for which an environmental impact statement was prepared. A Record of Decision identifies the alternatives considered in reaching the decision, the environmentally preferable alternative, factors balanced by the U.S. Department of Energy in making the decision, whether all practicable means to avoid or minimize environmental harm have been adopted, and, if not, the reason why they were not.

reductant – A chemical that is used to reduce the oxidation state (ionic charge) of another chemical.

reference concentration – It is an estimate of a toxic chemical daily inhalation of the human population (including sensitive subgroups) that is likely to be without an appreciable risk of deleterious effects during a lifetime. Those effects are both to the respiratory system (portal-of-entry) and the peripheral to the respiratory system (extra-respiratory effects). It is expressed in units of microgram per cubic meter.

refueling outage – The period of time that a reactor is shut down for refueling operations.

region of influence – A site-specific geographic area in which the principal direct and indirect effects of actions are likely to occur, and are expected to be of consequence for local jurisdictions.

regional economic area – A geographic area consisting of an economic node and the surrounding counties that are economically related, and include the places of work and residences of the labor force. Each regional economic area is defined by the U.S. Bureau of Economic Analysis.

regulated substances – A general term used to refer to materials other than radionuclides that may be regulated by other applicable Federal, state, or local requirements.

rem (roentgen equivalent man) – A unit of dose equivalent. The dose equivalent in rems equals the absorbed dose in rads in tissue multiplied by the appropriate quality factor and possibly other modifying factors. Derived from "roentgen equivalent man," referring to the dosage of ionizing radiation that will cause the same biological effect as one roentgen of x-ray or gamma-ray exposure. One rem equals 0.01 sievert. (See absorbed dose and dose equivalent.)

remediation – The process, or a phase in the process, of rendering radioactive, hazardous, or mixed waste environmentally safe, whether through processing, entombment, or other methods.

remote-handled waste – In general, refers to radioactive waste that must be handled at a distance to protect workers from unnecessary exposure (e.g., waste with a dose rate of 200 millirem per hour or more at the surface of the waste package). (See contact-handled waste.)

resin – See ion exchange resin.

Resource Conservation and Recovery Act, as Amended – A law that gives the U.S. Environmental Protection Agency the

authority to control hazardous waste from "cradle to grave" (i.e., from the point of generation to the point of ultimate disposal), including its minimization, generation, transportation, treatment, storage, and disposal. The Resource Conservation and Recovery Act also sets forth a framework for the management of nonhazardous solid wastes. (See hazardous waste.)

riparian – Of, on, or relating to the banks of a natural course of water.

risk – The probability of a detrimental effect from exposure to a hazard. Risk is often expressed quantitatively as the probability of an adverse event occurring multiplied by the consequence of that event (i.e., the product of these two factors). However, separate presentation of probability and consequence is often more informative.

risk assessment (chemical or radiological) – The qualitative and quantitative evaluation performed in an effort to define the risk posed to human health and/or the environment by the presence or potential presence and/or use of specific chemical or radiological materials.

roentgen – A unit of exposure to ionizing x- or gamma radiation equal to or producing one electrostatic unit of charge per cubic centimeter of air.

runoff – The portion of rainfall, melted snow, or irrigation water that flows across the ground surface, and eventually enters streams.

Safe Drinking Water Act – This act protects the quality of public water supplies, water supply and distribution systems, and all sources of drinking water.

safe, secure trailer – A specially modified semi-trailer, pulled by an armored tractor truck, which DOE uses to transport nuclear weapons, nuclear weapons components, or special nuclear material over public highways.

safeguards – An integrated system of physical protection, material accounting, and material control measures designed to deter, prevent, detect, and respond to unauthorized access, possession, use, or sabotage of nuclear materials.

safety analysis report – A report that systematically identifies potential hazards within a nuclear facility, describes and analyzes the adequacy of measures to eliminate or control identified hazards, and analyzes potential accidents and their associated risks. Safety analysis reports are used to ensure that a nuclear facility can be constructed, operated, maintained, shut down, and decommissioned safely and in compliance with applicable laws and regulations. Safety analysis reports are required for the U.S. Department of Energy nuclear facilities and as a part of applications for U.S. Nuclear Regulatory Commission licenses. The U.S. Nuclear Regulatory Commission regulations or DOE orders and technical standards that apply to the facility type provide specific requirements for the content of safety analysis reports. (See nuclear facility.)

safety evaluation report – A document prepared by the U.S. Nuclear Regulatory Commission that evaluates documentation (i.e., technical specifications, safety analysis reports, and special safety reviews and studies) submitted by a reactor licensee for its approval. This ensures all of the safety aspects of part or all of the activities conducted at a reactor are formally and thoroughly analyzed, evaluated, and recorded.

sanitary waste – Wastes generated by normal housekeeping activities, liquid or solid (includes sludge), which are not hazardous or radioactive.

scope – In a document prepared pursuant to the National Environmental Policy Act of 1969, the range of actions, alternatives, and impacts to be considered.

scoping – An early and open process for determining the scope of issues to be addressed in an environmental impact statement and for identifying the significant issues related to a proposed action. The scoping period begins after publication in the Federal Register of a Notice of Intent to prepare an environmental impact statement. The public scoping process is that portion of the process where the public is invited to

participate. The U.S. Department of Energy also conducts an early internal scoping process for environmental assessments or environmental impact statements. For environmental impact statements, this internal scoping process precedes the public scoping process. The U.S. Department of Energy's scoping procedures are found in 10 CFR Section 1021.311.

security – An integrated system of activities, systems, programs, facilities, and policies for the protection of restricted data and other classified information or matter, nuclear materials, nuclear weapons and nuclear weapons components, and/or DOE contractor facilities, property, and equipment.

seismic – Pertaining to any Earth vibration, especially an earthquake.

severe accident – An accident with a frequency rate of less than 10^{-6} per year that would have more severe consequences than a design-basis accident, in terms of damage to the facility, offsite consequences, or both. Also called "beyond-design-basis reactor accidents" in this NI PEIS.

sewage – The total organic waste and wastewater generated by an industrial establishment or a community.

shielding – With regard to radiation, any material of obstruction (bulkheads, walls, or other construction) that absorbs radiation in order to protect personnel or equipment.

short-lived activation products – An element formed from neutron interaction that has a relatively short half-life, which is not produced from the fission reaction (e.g., a cobalt isotope formed from impurities in the metal of the reactor piping).

short-lived nuclides – Radioactive isotopes with half-lives no greater than about 30 years (e.g., cesium-137 and strontium-90).

shutdown – For a U.S. Department of Energy reactor, the condition in which a reactor has ceased operation, and DOE has officially declared that it does not intend to operate it further.

sievert – The SI (International System of Units) unit of radiation dose equivalent. The dose equivalent in sieverts equals the absorbed dose in grays multiplied by the appropriate quality factor (1 sievert is equal to 100 rem). (See gray.)

silica gel – An amorphous, highly adsorbent form of silicon dioxide.

silt – A sedimentary material consisting of fine mineral particles, intermediate in size between sand and clay.

sinter – A process in which particles are bonded together by pressure and heating below the melting point.

solvent extraction – A process that uses two solvents that do not mix (usually water and an organic solvent) to separate chemicals. An organic soluble chemical is usually added to the organic solvent to selectively extract a chemical from the aqueous solution into the organic solution when they are mixed. After the settling, the two solvents are separated from one another, and the desired chemical is removed from the organic solvent.

source term – The amount of a specific pollutant (e.g., chemical, radionuclide) emitted or discharged to a particular environmental medium (e.g., air, water) from a source or group of sources. It is usually expressed as a rate (i.e., amount per unit time).

spallation – A nuclear reaction in which light particles are ejected as the result of bombardment (as by high-energy protons).

special nuclear materials – As defined in Section 11 of the Atomic Energy Act of 1954, special nuclear material means: (1) plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the U.S. Nuclear Regulatory Commission determines to be special nuclear material; or (2) any material artificially enriched by any of the above. Tritium is not a special nuclear material.

sulfur oxides – Common air pollutants, primarily sulfur dioxide, a heavy, pungent, colorless gas

(formed in the combustion of fossil fuels, considered a major air pollutant) and sulfur trioxide. Sulfur dioxide is involved in the formation of acid rain. It can also irritate the upper respiratory tract and cause lung damage.

supernatant – The liquid that stands over a precipitated material.

surface water – All bodies of water on the surface of the earth and open to the atmosphere, such as rivers, lakes, reservoirs, ponds, seas, and estuaries.

target – A tube, rod, or other form containing material that on being irradiated in a nuclear reactor or an accelerator would produce a desired end product.

technical specifications – With regard to U.S. Nuclear Regulatory Commission regulations, part of a U.S. Nuclear Regulatory Commission license authorizing the operation of a nuclear reactor facility. A technical specification establishes requirements for items such as safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

threatened species – Any plants or animals that are likely to become endangered species within the foreseeable future throughout all or a significant portion of their ranges and which have been listed as threatened by the U.S. Fish and Wildlife Service or the National Marine Fisheries Service following the procedures set out in the Endangered Species Act and its implementing regulations (50 CFR Part 424). (See endangered species.)

The lists of threatened species can be found at 50 CFR Sections 17.11 (wildlife), 17.12 (plants), and 227.4 (marine organisms).

threshold limit values – The recommended highest concentrations of contaminants to which workers may be exposed according to the American Conference of Governmental Industrial Hygienists.

total effective dose equivalent – The sum of the effective dose equivalent from external exposures and the committed effective dose equivalent from internal exposures.

Toxic Substances Control Act of 1976 – This act authorizes the U.S. Environmental Protection Agency to secure information on all new and existing chemical substances and to control any substances determined to cause an unreasonable risk to public health or the environment. This law requires that the health and environmental effects of all new chemicals be reviewed by the U.S. Environmental Protection Agency before they are manufactured for commercial purposes.

transients – Events that could cause a change or disruption of plant thermal, hydraulic, or neutronic behavior.

transuranic – Refers to any element whose atomic number is higher than that of uranium (atomic number 92), including neptunium, plutonium, americium, and curium. All transuranic elements are produced artificially and are radioactive.

transuranic waste – Radioactive waste that is not classified as high-level radioactive waste and that contains more than 100 nanocuries (3700 becquerels) per gram of alpha-emitting transuranic isotopes with half-lives greater than 20 years.

Type B packaging – A regulatory category of packaging for transportation of radioactive material. The U.S. Department of Transportation and U.S. Nuclear Regulatory Commission require Type B packaging for shipping highly radioactive material. Type B packages must be designed and demonstrated to retain their containment and shielding integrity under severe accident conditions, as well as under the normal conditions of transport. The current U.S. Nuclear Regulatory Commission testing criteria for Type B package designs (10 CFR Part 71) are intended to simulate severe accident conditions, including impact, puncture, fire, and immersion in water. The most widely recognized Type B packages are the massive casks used for transporting spent nuclear fuel. Large-capacity cranes and mechanical lifting

equipment are usually needed to handle Type B packages.

Type B shipping cask – A U.S. Nuclear Regulatory Commission-certified cask with a protective covering that contains and shields radioactive materials, dissipates heat, prevents damage to the contents, and prevents criticality during normal shipment and accident conditions. It is used for transport of highly radioactive materials and is tested under severe, hypothetical accident conditions that demonstrate resistance to impact, puncture, fire, and submersion in water.

unit cancer risk – The likelihood that the substance is a human carcinogen and quantitatively gives an estimate of risk from oral exposure or from inhalation exposure. This estimate can be in terms of either risk per microgram per liter of drinking water or risk per microgram per cubic meter of air breathed.

uranium – A radioactive, metallic element with the atomic number 92; one of the heaviest naturally occurring elements. Uranium has 14 known isotopes, of which uranium-238 is the most abundant in nature. Uranium-235 is commonly used as a fuel for nuclear fission. (See natural uranium, enriched uranium, highly enriched uranium, and depleted uranium.)

viewshed – The extent of an area that may be viewed from a particular location. Viewsheds are generally bounded by topographic features such as hills or mountains.

visual resource management class – Any of the classifications of visual resources established through application of the Visual Resources Management process of the Bureau of Land Management. Four classifications are employed to describe different degrees of modification to landscape elements: Class I, areas where the natural landscape is preserved, including national wilderness areas and the wild sections of national wild and scenic rivers; Class II, areas with very limited land development activity, resulting in visual contrasts that are seen but do not attract attention; Class III, areas in which development may attract attention, but the natural landscape still dominates; Class IV, areas in which development activities may dominate the view and may be the major focus in the landscape.

volatile organic compounds – A broad range of organic compounds, often halogenated, that vaporize at ambient or relatively low temperatures, such as benzene, chloroform, and methyl alcohol. With regard to air pollution, any organic compound that participates in atmospheric photochemical reaction, except for those designated by the U.S. Environmental Protection Agency administrator as having negligible photochemical reactivity.

waste acceptance criteria – The requirements specifying the characteristics of waste and waste packaging acceptable to a disposal facility, and the documents and processes the generator needs to certify that the waste meets applicable requirements.

waste classification – Wastes are classified according to DOE Order 435.1, *Radioactive Waste Management,* and include high-level, transuranic, and low-level wastes.

Waste Isolation Pilot Plant – A U.S. Department of Energy facility designed and authorized to permanently dispose of transuranic radioactive waste in a mined underground facility in deep geologic salt beds. It is located in southeastern New Mexico, 42 kilometers (26 miles) east of the city of Carlsbad.

waste management – The planning, coordination, and direction of those functions related to generation, handling, treatment, storage, transportation, and disposal of waste, as well as associated surveillance and maintenance activities.

waste minimization and pollution prevention – An action that economically avoids or reduces the generation of waste and pollution by source reduction, reducing the toxicity of hazardous waste and pollution, improving energy use, or recycling. These actions will be consistent with the general goal of minimizing present and future threats to human health, safety, and the environment.

weighting factor – Generally, a method of attaching different importance values to different items or characteristics. In the context of radiation protection, the proportion of the risk of effects resulting from irradiation of a particular organ or tissue to the total risk of effects when the whole body is irradiated uniformly (e.g., the organ dose weighting factor for the lung is 0.12, compared to 1.0 for the whole body). Weighting factors are used for calculating the effective dose equivalent.

whole-body dose – With regard to radiation, dose resulting from the uniform exposure of all organs and tissues in a human body. (See effective dose equivalent.)

wind rose – A circular diagram showing, for a specific location, the percentage of the time the wind is from each compass direction. A wind rose for use in assessing consequences of airborne releases also shows the frequency of different wind speeds for each compass direction.

Zircaloy – An alloy of zirconium containing tin, iron, chromium, and nickel.

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DOE/EIS-0310
December 2000

FINAL Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility

Volume 2 Appendixes

United States Department of Energy
Office of Nuclear Energy, Science and Technology
Washington, DC

Cover photograph and illustration identification, beginning at top center and going clockwise:

- Radioisotope tagged monoclonal antibodies, “smart bullets,” target malignant cells for diagnosis and treatment of diseases
- The Fast Flux Test Facility at the Hanford Site near Richland, Washington
- Illustration of a satellite that could use radioisotope power systems
- The High Flux Isotope Reactor at the Oak Ridge National Laboratory near Oak Ridge, Tennessee
- The Advanced Test Reactor at the Idaho National Engineering and Environmental Laboratory near Idaho Falls, Idaho
- Tip of a remote-handling arm, used for work in developing industrial and medical isotopes

AVAILABILITY OF THE FINAL NI PEIS

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This PEIS is accessible on the Office of Nuclear Energy, Science and Technology web site at www.nuclear.gov.

Printed with soy ink on recycled paper

Cover Sheet

Responsible Agency: United States Department of Energy (DOE)

Title: *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (NI PEIS)

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Abstract: Under the authority of the Atomic Energy Act of 1954, as amended, the DOE is responsible for ensuring the availability of isotopes for medical, industrial and research applications, meeting the nuclear material needs of other Federal agencies, and undertaking research and development activities related to development of nuclear power for civilian use. To meet these responsibilities, DOE maintains nuclear infrastructure capabilities that support various missions. Current estimates for the future needs of medical and industrial isotopes, plutonium-238, and research requirements indicate that the current infrastructure may soon be insufficient to meet the projected demands. DOE proposes to enhance these capabilities to provide for: (1) production of isotopes for medical and industrial uses, (2) production of plutonium-238 for use in advanced radioisotope power systems for future National Aeronautics and Space Administration (NASA) space exploration missions, and (3) the Nation's nuclear research and development needs for civilian application.

This NI PEIS evaluates the environmental impacts of a No Action Alternative (maintaining status quo), four alternative strategies to accomplish this mission, and an alternative to permanently deactivate the Fast Flux Test Facility (FFTF), with no new missions. Alternatives 2, 3, and 4 also include permanent deactivation of FFTF. The alternatives are:

No Action
1. Restart FFTF at Hanford, Washington
2. Use only existing operational facilities
3. Construct one or two new accelerators
4. Construct a new research reactor
5. Permanently deactivate FFTF (with no new missions)

The Preferred Alternative is Alternative 2, Option 7, Use Only Existing Operational Facilities. DOE would reestablish domestic production of plutonium-238, as needed, using the Advanced Test Reactor in Idaho and the High Flux Isotope Reactor in Tennessee, and would process irradiated plutonium-238 targets at the Radiochemical Engineering Development Center in Tennessee. DOE would permanently deactivate FFTF under the Preferred Alternative.

Public Comments: The Draft NI PEIS was issued for public review and comment on July 21, 2000. The comment period ended on September 18, 2000, although late comments were considered to the extent practicable. Public hearings were held to obtain comments on the Draft NI PEIS in Oak Ridge, Tennessee; Idaho Falls, Idaho; Hood River and Portland, Oregon; Seattle and Richland, Washington; and Arlington, Virginia. All comments were considered by DOE in preparing the Final NI PEIS, which also incorporates any new information received since issuance of the Draft NI PEIS. In response to comments on the Draft NI PEIS and as a result of information that was unavailable at the time of the issuance of the Draft PEIS, the Final PEIS contains revisions and new information, indicated by a sidebar in the margin. Volume 3 contains the comments received during the public review period for the Draft NI PEIS and DOE's responses to these comments. DOE will use the analyses presented in the Final NI PEIS as well as other information, including public input, costs, nonproliferation impacts, schedules, technical assurance, and other policy and programmatic objectives, in preparing the Record of Decision for accomplishing expanded civilian nuclear energy research and development and isotope production missions in the United States, including the role of FFTF. DOE will issue the Record of Decision no sooner than 30 days after the U.S. Environmental Protection Agency publishes a notice of availability of the Final NI PEIS in the Federal Register.

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List of Acronyms

AAA	Advanced Accelerator Applications
ALARA	as low as is reasonably achievable
ALOHA	Area Locations of Hazardous Atmospheres
ATR	Advanced Test Reactor
ATW	Accelerator Transmutation of Waste
BEIR	Biological Effects of Ionizing Radiation
BLIP	Brookhaven LINAC Isotope Producer
CEQ	Council on Environmental Quality
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
CLWR	commercial light water reactor
CPP–651	Chemical Processing Plant 651 (Building CPP–651)
dBA	decibels A-weighted
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
FDPF	Fluorinel Dissolution Process Facility
FFTF	Fast Flux Test Facility
FMEF	Fuels and Materials Examination Facility
FONSI	Finding of No Significant Impact
FPF	Fuel Processing Facility
FR	Federal Register
g	gravitational acceleration
Hanford	Hanford Site
HB	horizontal beam
HEPA	high-efficiency particulate air (filter)
HFIR	High Flux Isotope Reactor
HVAC	heating, ventilating, and air conditioning
I^4	International Isotopes Idaho, Inc.
IAEA	International Atomic Energy Agency
IEM	Interim Examination and Maintenance Cell
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
IPF	Isotope Production Facility
ISC3	Industrial Source Complex
ISCST3	Industrial Source Complex (short-term model)
LANL	Los Alamos National Laboratory
LANSCE	Los Alamos Neutron Science Center
LCF	latent cancer fatality
LINAC	linear accelerator
MASF	Maintenance and Storage Facility
NAAQS	National Ambient Air Quality Standards
NASA	National Aeronautics and Space Administration
NEPA	National Environmental Policy Act
NEPO	Nuclear Energy Plant Optimization

NERAC	Nuclear Energy Research Advisory Committee
NERI	Nuclear Energy Research Initiative
NESHAPs	National Emissions Standards for Hazardous Air Pollutants
NI PEIS	*Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility*
NPDES	National Pollutant Discharge Elimination System
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
OSHA	Occupational Safety and Health Administration
PCAST	President's Committee of Advisors on Science and Technology
PEIS	programmatic environmental impact statement
P.L.	Public Law
$PM_{2.5}$	particulate matter with an aerodynamic diameter less than or equal 2.5 microns
PM_{10}	particulate matter with an aerodynamic diameter less than or equal to 10 microns
PNNL	Pacific Northwest National Laboratory
RCRA	Resource Conservation and Recovery Act
REDC	Radiochemical Engineering Development Center
RPL	Radiochemical Processing Laboratory
SNL	Sandia National Laboratories
SRPS	Stirling radioisotope power systems
SRS	Savannah River Site
SRTG	Small Radioisotope Thermoelectric Generator
SST/SGT	safe, secure trailer/SafeGuards Transport
TEDF	Treated Effluent Disposal Facility
TRIGA	training, research, isotopes General Atomics (reactor)
UZrH	uranium-zirconium-hydride
U.S.C.	United States Code
USGS	U.S. Geological Survey
WAG	waste area grouping
WIPP	Waste Isolation Pilot Plant
Y–12	Y–12 Plant

Appendix A
Neptunium-237 Target Fabrication and Processing Operations for Plutonium-238 Production

This appendix includes a description of the Oak Ridge National Laboratory (ORNL) Radiochemical Engineering Development Center (REDC), the Idaho National Engineering and Environmental Laboratory (INEEL) Fluorinel Dissolution Process Facility (FDPF), the Hanford Site (Hanford) Fuels and Materials Examination Facility (FMEF), and the proposed processing facilities and technologies that would be used to store neptunium-237, fabricate neptunium-237 targets, process irradiated targets for plutonium-238 production, recycle neptunium-237, and ship plutonium-238 oxide to Los Alamos National Laboratory (LANL). The material presented in this appendix is based primarily on *Preconceptual Design Planning for Chemical Processing to Support Pu-238 Production* (Wham et al. 1998), except where noted.

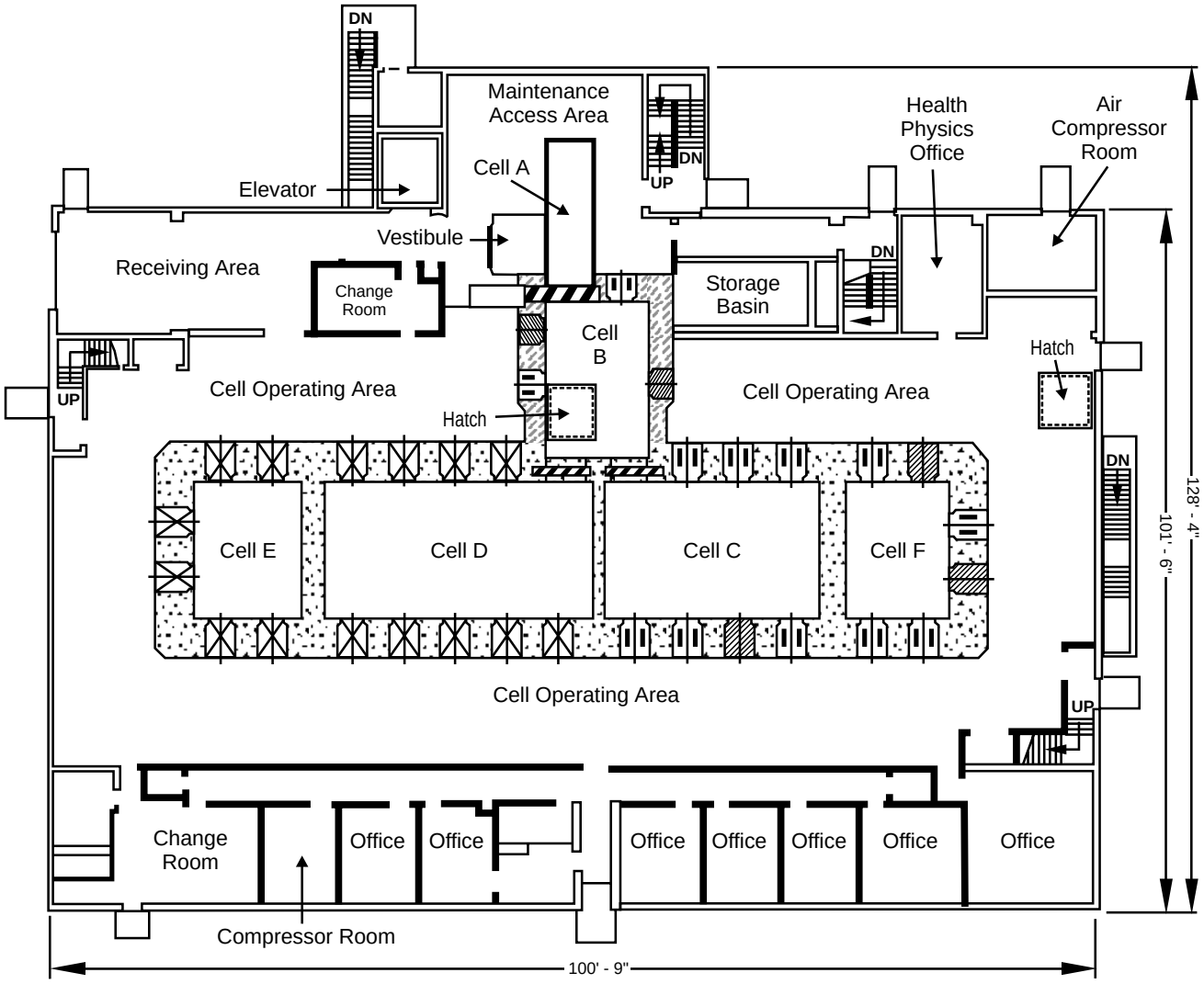
A.1 RADIOCHEMICAL ENGINEERING DEVELOPMENT CENTER

A.1.1 Facility Description

REDC is part of the Melton Valley 7900 Complex at ORNL. The High Flux Isotope Reactor (HFIR) is also part of the Melton Valley 7900 Complex. The REDC Complex includes a hot cell facility, Building 7930, which is specifically designed to address the problems associated with the containment of actinide element isotopes and their daughter isotopes. This facility was also designed to protect workers from high dose rates of penetrating radiation, including fast neutrons from spontaneous fission that require thick shielding and the capability for remote operation using manipulators. Building 7930, the proposed site for the plutonium-238 production project, has been used for a variety of transuranium actinide element projects, most notably the processing of californium-252, curium-244, and curium-248 for use as high-intensity neutron sources and research radioisotopes. The current californium-252 operations in Building 7930 would continue and coexist with the plutonium-238 production project.

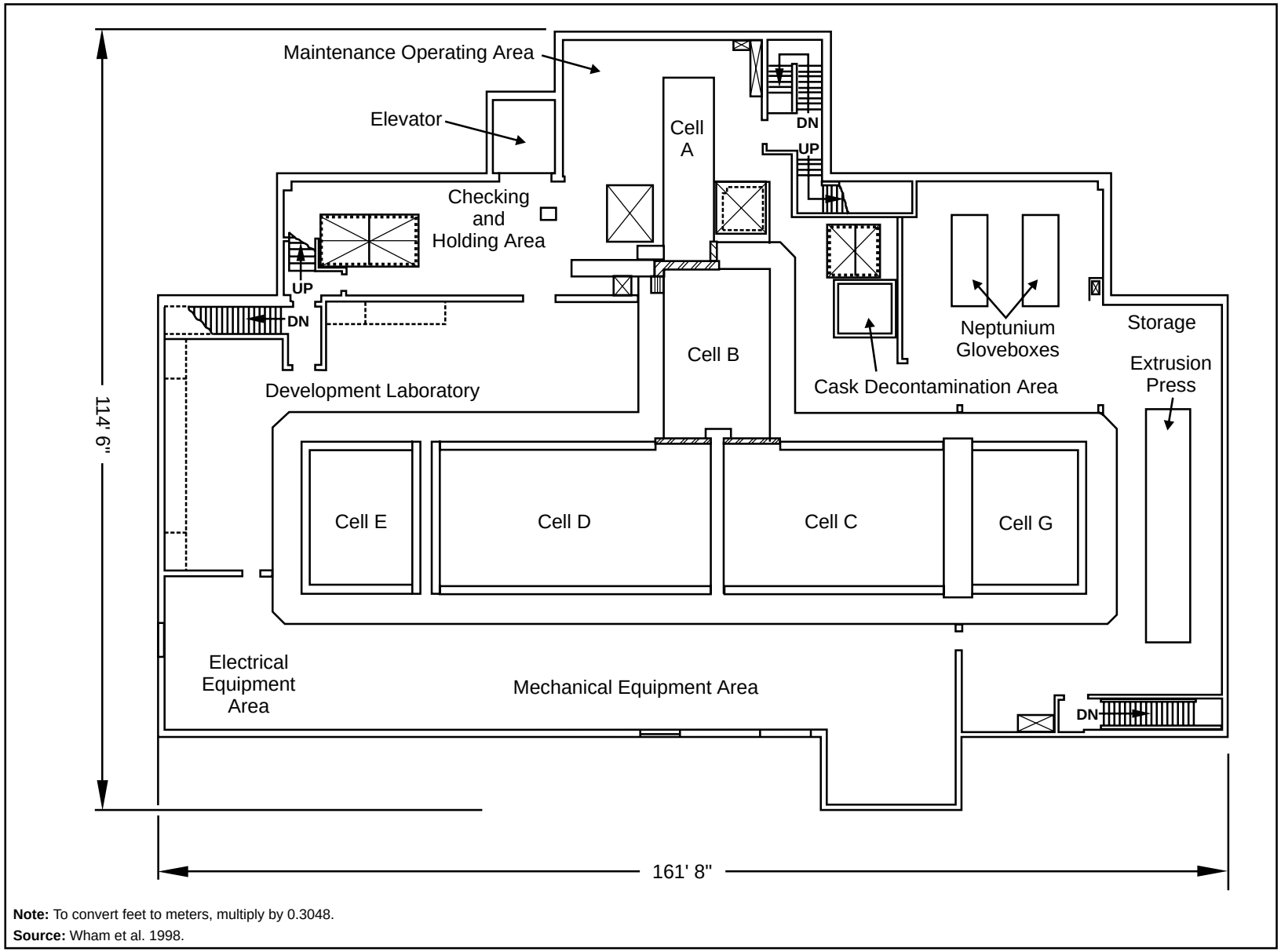
Building 7930 is a three-story structure with a partial basement, constructed of structural steel, reinforced concrete, and masonry. Perimeter walls are reinforced concrete block. Floors are reinforced concrete slabs that are either poured on compacted aggregate or are supported by structural steel. The roof, replaced in the summer of 1997, is metal decking covered with built-up roofing. The building has a gross floor area of 3,062 square meters (32,950 square feet), exclusive of hot cells. The cell complex adds 286 square meters (3,080 square feet). The total enclosed volume is 18,295 cubic meters (646,000 cubic feet). The building is divided into four major areas: (1) the hot cell complex, consisting of six shielded cells and one unshielded cell; (2) maintenance and service areas; (3) an operating control area; and (4) an office area. Also included are utility services, ventilation systems, crane and manipulator systems, and liquid waste systems. The first and second floor plans of Building 7930 are shown in **Figures A–1** and **A–2**, respectively.

Cells D and E and space on both the second and third floors would be used for the plutonium-238 project. Cells D and E are both clean and empty and could be used for this work with minimal modifications. Cell D activities would include receipt of irradiated targets, target dissolution, chemical separation of neptunium and plutonium from fission products, partitioning and purification of the neptunium, and transuranic waste processing. Cell E would contain processing equipment to purify the plutonium-238, prepare plutonium oxide, and transfer the oxide into shipping containers. Cell E also would provide temporary storage of the neptunium oxide from the Savannah River Site (SRS). Cell F is also a possible interim location for storing neptunium. Neptunium-237 target fabrication would be completed on the second floor outside the cell, but inside a glovebox.



Note: To convert feet to meters, multiply by 0.3048.
Source: Wham et al. 1998.

Figure A–1 First Floor Plan for Building 7930



Note: To convert feet to meters, multiply by 0.3048.
Source: Wham et al. 1998.

Figure A–2 Second Floor Plan for Building 7930

Cells D and E are adjacent and separated by a 1.2-meter-thick (4-foot-thick) shielding wall. The exterior walls are reinforced concrete 1.7 meters (5.5 feet) thick to a height of 3.4 meters (11 feet) above the first floor, and 1.4 meters (4.5 feet) thick from there to the roof, which is 1.5 meters (5 feet) thick. The cell floors are lined with Type 304L stainless steel, as are the walls and ceiling of Cell D. Currently, only the lower 0.3 meters (12 inches) of the Cell E walls are lined with stainless steel; above that, the walls are concrete with a 0.051-centimeter-thick (0.020-inch-thick) modified phenolic protective coating. For the plutonium-238 project, the walls in Cell E must be completely lined with stainless steel to improve containment and facilitate decontamination.

The lower section of each cell has operating modules, each with a large penetration for a viewing window and a pair of sleeves through which manipulators can be installed. Each cell is also equipped with transfer hatches and ports in the roof through which tools and equipment can be inserted.

Within a cell, operations would take place primarily with equipment contained in a workstation. A workstation is a stainless steel pan with short walls, and is open to the surrounding cell. The workstation has dimensions such that the manipulators can reach all equipment, and the entire area is viewable to the operators through the viewing window. A servomanipulator would be used to transfer materials between workstations. Transfer of items within workstations would be accomplished using the manipulators at each workstation.

Cell D has interior dimensions of 6.1 meters (20 feet) wide, 12.5 meters (41 feet) long, and 7.3 meters (24 feet) high. Currently, five chemical processing workstations and one analytical chemistry workstation are planned for Cell D. One window location would be used for transfer lines to bring in process solutions.

Cell E has interior dimensions of 6.1 meters (20 feet) wide, 4.9 meters (16 feet) long, and 9.1 meters (30 feet) high. A storage facility would be added, and three window locations would be used for chemical processing workstations. One of the workstations would be an enclosed stainless steel box. This workstation would be used for handling plutonium-238 oxide as a powder. Use of an enclosed area in this manner would minimize migration of plutonium-238 oxide powder and contamination.

All cells would be ventilated by air drawn from the occupied areas of the building through high-efficiency particulate air filters, and then through the cells on a once-through basis. The air leaving the cells would be filtered at the point of exit by high-capacity roughing filters, and then by two banks of high-efficiency particulate air filters in succession before being released to the atmosphere from Stack 7911.

Currently, Cells D and E have heat detectors, but are not equipped with fire suppression systems.

Target fabrication would be carried out on the second floor of Building 7930. The area is currently used as a maintenance shop and storage area. The walls between the maintenance shop and storage area would be removed to allow room for gloveboxes. The gloveboxes would be used to convert aqueous neptunium solutions into a form suitable for target fabrication. The storage area would be moved to another location within REDC and would be used for target fabrication equipment.

A.1.2 Neptunium-237 Storage

The neptunium-237 from SRS would arrive as neptunium dioxide, the most stable of the neptunium oxides. Upon arrival at REDC, the neptunium oxide would be removed from the shipping container(s). The product canister containing the neptunium oxide would remain in the containment vessel for storage in a designated shielded storage area until initiation of the purification process to remove protactinium-233.

After the neptunium-237 is processed into targets, the targets would be placed in shielded containers. The product canisters would be placed into containment vessels, and loaded directly into shipping containers for transport to the irradiation facility.

A.1.3 Neptunium-237 Target Fabrication Process Description

The fabrication of neptunium-237 targets for plutonium-238 production would require dissolving the neptunium oxide (if necessary), purification of neptunium to remove radioactive decay products, conversion of the neptunium to an oxide, and fabrication of neptunium oxide into targets for irradiation. The neptunium-237 to be used in the targets would come from two sources: (1) neptunium that had been separated previously during spent nuclear fuel processing at SRS, converted to an oxide, and then shipped to ORNL for storage (Section A.1.2); and (2) neptunium that would be recovered from irradiated neptunium-237 targets and recycled for use in new targets (Section A.1.4).

Initially, all of the neptunium-237 required for target fabrication would come from the neptunium oxide in storage. After postirradiation processing begins, most of the neptunium requirements would be met by using recycled neptunium, and only a small quantity of the stored neptunium would be needed to replace the neptunium transformed to plutonium-238 during irradiation.

All target fabrication activities at ORNL would be conducted in REDC Building 7930. The first stage of the target fabrication process would involve neptunium purification to remove protactinium-233 (a product of neptunium-237 alpha decay) and would be conducted in shielded facilities to minimize radiation exposure. Oxidation of the purified neptunium, mixing neptunium oxide with a suitable diluent, and preparing a billet for extrusion for Advanced Test Reactor (ATR) or HFIR targets, or preparing wafers or pellets for Fast Flux Test Facility (FFTF) targets, would be conducted in shielded gloveboxes on the second floor of Building 7930. The mechanical operations involved in the final target fabrication would be conducted in open boxes located in the target fabrication room.

The fabrication process for neptunium-237 targets is shown in **Figure A–3**. The neptunium would be brought from storage or recycling in preparation for purification. The neptunium would be dissolved (if necessary), treated to remove protactinium-233, and converted to an oxide. The oxide then would be transferred to the target fabrication line, where it would be formed into neptunium-237 targets.

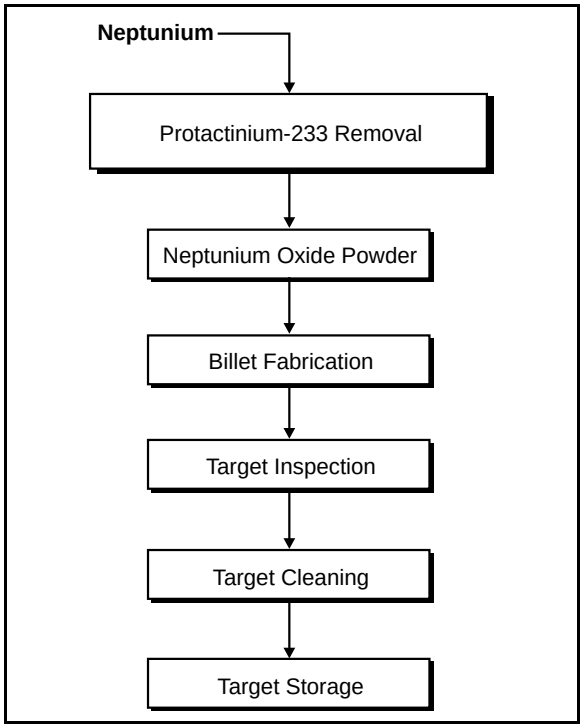


Figure A–3 Steps Required for Target Fabrication

A.1.3.1 Neptunium-237 Purification

Neptunium-237 is a radioactive isotope that decays to protactinium-233 through loss of an alpha particle. The protactinium reaches 90 percent of the equilibrium activity in approximately 10 weeks. Protactinium-233 has a short half-life (27 days) and decays to uranium-233, releasing gamma rays. The decay of this protactinium ingrowth would contribute significantly to the radiation doses to workers in the target fabrication

line. Removal of the protactinium prior to oxide conversion and target fabrication would result in less radiation exposure for personnel involved in the target fabrication activities.

Neptunium oxide would be removed from its storage location in Cell E and transferred to dissolution equipment, which also would be located in Cell E. This dissolution may also take place in Cell D. The equipment would be sized to dissolve kilogram-size batches of neptunium-237. The neptunium would be dissolved in nitric acid and the solution would be passed through a column of silica gel, which would adsorb the protactinium. The purified neptunium-bearing solution, obtained in the third solvent extraction step during processing of irradiated neptunium-237 targets, would become the primary source of neptunium-237 for targets, once irradiated target processing began (Section A.1.4). Additional neptunium needed to supplement this source would come from the neptunium oxide stored in Cell E. The protactinium adsorbed in the silica gel then would be converted to a solid waste form. Because of the relatively short half-life of the protactinium-233, and its decay to uranium-233 (a relatively stable isotope), the radioactivity from this waste would be small after about 1 year of storage. The purified neptunium-237 solution would then be transferred to the target fabrication glovebox line for conversion of the neptunium to oxide.

A.1.3.2 Neptunium Oxide Production

The desired form of neptunium oxide for target fabrication is currently assumed to be oxide microspheres. The neptunium-237 solution would be passed through a cation-exchange column containing a resin, such as Dowex 50W-X8, of selected particle-size range (typically 60 to 80 microns [0.000024 to 0.000031 inches]). The loaded resin would be washed with dilute acid and dried with an air stream pulled through the column via a vacuum.

To convert the resin to oxide, a multistep heating cycle with ramp and hold times using both air and 4 percent hydrogen/argon streams would be employed. The preferred method would use heated air to burn the resin and to form the neptunium oxide microspheres. A typical cycle would be as follows: initially heat the resin with air (7 to 10 millimeters [0.28 to 0.39 inches] per second superficial velocity) at 150 °C (300 °F) for 1 hour; ramp to 450 °C (840 °F) at 5 °C (9 °F) per minute and hold for 1 hour with air; ramp to 800 °C (1,470 °F) at 10 °C (18 °F) per minute and hold for 4 hours with air (most of the carbon should be removed in this step); and switch to 4 percent hydrogen/argon and continue at 800 °C (1,470 °F) for a final 4-hour period to complete conversion to the oxide. The oxide then would be cooled for handling, transferred to a crucible, and sintered at 1,200 °C (2,190 °F) in air for 10 hours to complete the oxidation. This material would be weighed and characterized (tap density, radiochemical analysis, and particle size) to determine the blends for fabrication into pellets.

An alternative procedure for producing neptunium oxide would be precipitation of neptunium oxalate, followed by filtration and calcination to form neptunium oxide. The neptunium-bearing solution would be mixed with a solution containing oxalic acid. After mixing for at least 30 minutes, the supernatant would be decanted and filtered into a holding tank. The neptunium oxalate would be drained into a filter boat. The tank would be washed with 0.1 molar (*M*) oxalic acid and drained through the filter. The oxalate would dry on the filter and then be transferred to a platinum-lined furnace can. The filtrate would be sampled and sent to liquid waste treatment. The furnace can containing the neptunium oxalate would be placed into a calciner and heated to approximately 400 °C (750 °F) for 1 hour to decompose the neptunium oxalate to neptunium oxide and carbon dioxide.

A.1.3.3 Neptunium-237 Target Fabrication

Fabrication of the targets would take place in dedicated gloveboxes in Building 7930. The target for the ATR and HFIR reactors consists of the neptunium oxide blended with an inert filler such as aluminum powder, pressed into a target core, and clad with aluminum. This type of target has been used historically in nearly all of the U.S. Department of Energy (DOE) water-cooled and -moderated production and research reactors used to produce isotopes, including plutonium-238. Three different fabrication techniques could be employed in making these types of targets for ATR and HFIR:

- Blending the neptunium oxide and aluminum powders, pressing the mixture into individual pellets, loading the pellets into aluminum target tubes, and seal welding and hydrostatic compression of the tubes

- Blending the neptunium oxide and aluminum powders, pressing the mixture into compacts, roll milling the compacts between aluminum-clad material, and seal welding the aluminum-clad neptunium dioxide and aluminum plates

- Blending the neptunium oxide and aluminum powders; pressing the mixture into billets; assembling the billets into welded, evacuated containers; and coextruding the neptunium oxide and aluminum mixture with the aluminum container to produce target tubes

Another fabrication technique that may be considered is mixing the neptunium oxide with a high-temperature diluent (other than aluminum), pressing the mixture into pellets or tubes, and sealing it into Zircaloy tubing. Targets with Zircaloy or stainless steel cladding would be used in targets for the commercial light water reactor (CLWR) or a high-energy accelerator due to higher operating temperatures. This fabrication technique would use similar gloveboxes for target fabrication and quality assurance tests of the targets. The proposed target for the FFTF reactor would consist of alternating wafers or pellets of neptunium oxide and yttrium hydride (moderator) sealed in stainless or ferritic alloy steel tubing to make target pins. These pins, up to 2.5 centimeters (1 inch) in diameter by 2.4 meters (8 feet) long, would subsequently be inserted into an inlet nozzle, duct, and handling socket assembly for handling and insertion in the FFTF reactor. Fabrication and assembly operations would be done in shielded gloveboxes to minimize personnel radiation exposure. The completed targets then would be stored in Building 7930 until shipment to a reactor for irradiation.

A.1.4 Postirradiation Target Processing Description

Postirradiation processing of neptunium-237 targets at ORNL would involve dissolution; separation of the actinides from the fission products; separation of neptunium from plutonium; a third solvent extraction process to purify the remaining neptunium; purification of plutonium; precipitation of plutonium oxalate; calcination of plutonium oxalate to plutonium oxide; and exchange of oxygen-17 and -18 by oxygen-16 in the plutonium oxide. The dissolution and purification processes would be conducted in Cell D of Building 7930, and the plutonium purification, plutonium oxide preparation, oxygen exchange reaction, and transfer of the plutonium oxide to shipping containers would be conducted in Cell E of Building 7930.

The postirradiation target processing steps are shown in **Figure A–4**. The irradiated targets would be cooled at the irradiation site for at least 120 days to allow time for decay of short-lived fission products.

A.1.4.1 Target Dissolution

Irradiated targets from ATR or HFIR would be brought to Building 7930 in a Type B shipping cask with sufficient shielding to meet U.S. Department of Transportation (DOT) requirements and transferred to Cell D,

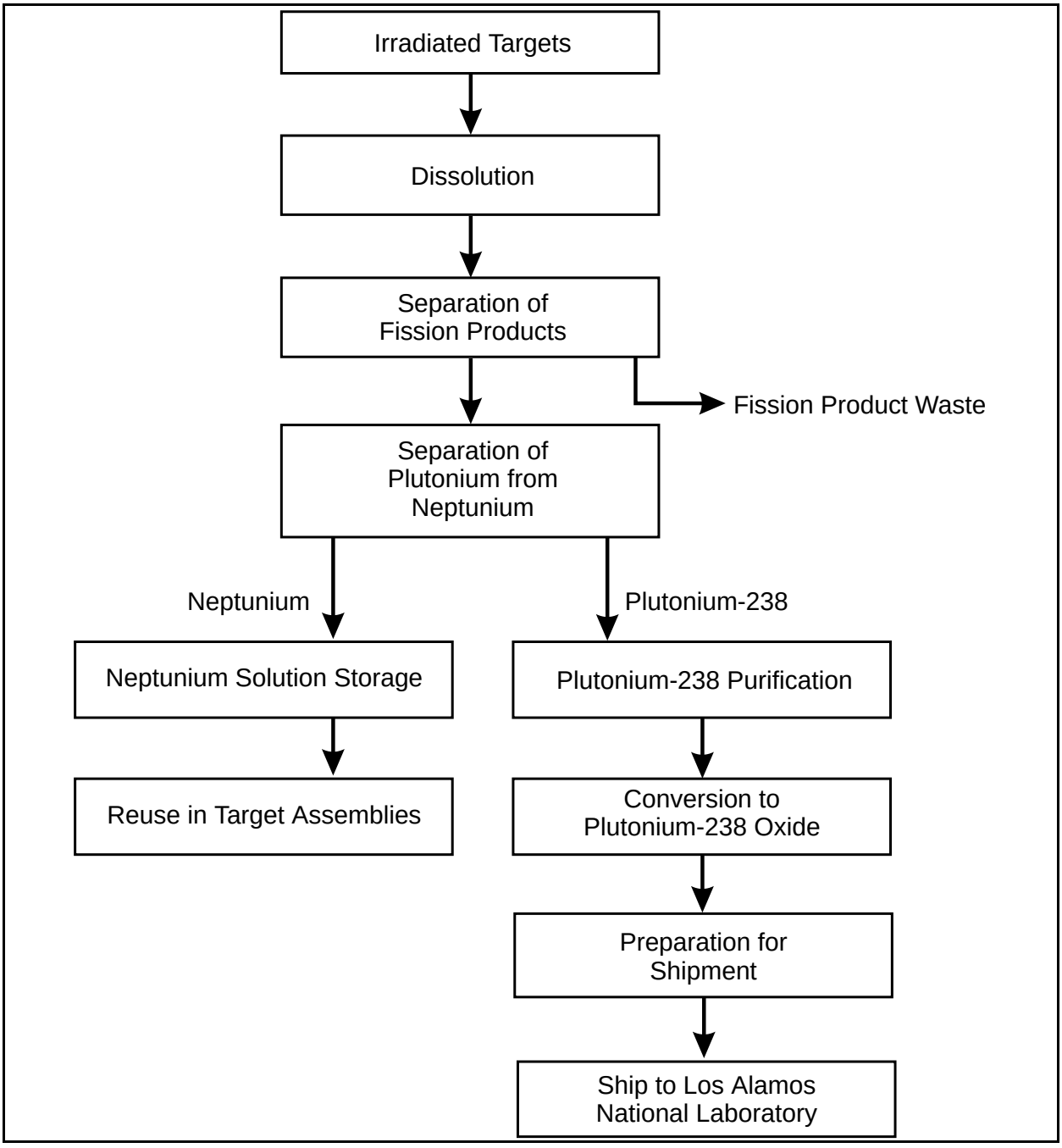


Figure A–4 Irradiated Neptunium-237 Target Processing

where postirradiation processing of the targets would be conducted. Dissolution of the aluminum-clad irradiated targets from HFIR or ATR would be accomplished using a two-step process. In the first step, the irradiated targets would be submerged in a 2.25 *M* sodium nitrate solution and heated to 90 to 95 °C (194 to 203 °F). A 10 *M* sodium hydroxide solution then would be added at a controlled rate to sustain the dissolution reaction. The aluminum-bearing caustic solution would be pumped through parallel sintered stainless steel filters and discarded as low-level waste. The filter then would be backflushed to the dissolver tank, where the remaining solids would be digested in a solution consisting of 8 *M* nitric acid and 0.02 *M* sodium fluoride. This solution would dissolve the actinides and most of the remaining fission products. This solution would

be filtered to remove any remaining solids (primarily silica fission products) and produce an acid product for solvent extraction.

Stainless steel- or Zircaloy-clad targets irradiated in the CLWR or in FFTF would be cut into small pieces and leached with nitric acid or other suitable solution to dissolve the neptunium, plutonium, and fission products away from the insoluble cladding. The solution would be filtered and the undissolved cladding would be discarded as low-level waste.

A.1.4.2 Plutonium Separation and Neptunium Recycling

The solution containing neptunium and plutonium would be processed in a series of three solvent extraction steps. In the first step, the neptunium and plutonium would be separated from fission products by extraction into an organic phase consisting of tributyl phosphate dissolved in normal paraffin hydrocarbon. Fission products and other contaminants would remain in the aqueous phase. After separation of the aqueous and organic phases, neptunium and plutonium would be stripped from the organic phase into another aqueous phase using a solution of 0.1 *M* nitric acid containing 0.1 *M* hydroxylamine nitrate as a reducing agent. Reduction would convert neptunium into the +4 oxidation state and plutonium into the +3 oxidation state. This solution then would flow to the second solvent extraction stage.

In the second solvent extraction step, neptunium (which would be in the +4 oxidation state) would be selectively extracted back into an organic phase consisting of 30 percent tributyl phosphate in normal paraffin hydrocarbon, while plutonium (which would be in the +3 oxidation state) would remain in the aqueous phase. Control of the oxidation states would be accomplished by the presence of hydroxylamine nitrate in the feed solution and the use of an aqueous hydroxylamine nitrate scrub stream to maintain the proper oxidation states within the solvent extraction contactor. Neptunium then would be stripped from the organic phase using a solution of 0.1 *M* hydroxylamine nitrate and 0.2 *M* nitric acid. The acidity of the neptunium-bearing solution would be adjusted to 2 *M* nitric acid and would be routed to a third solvent extraction step, where the solution would be further purified. The plutonium-bearing stream would be transferred to Cell E, where it would be further purified, if necessary, and converted to an oxide.

In the third solvent extraction step, any plutonium remaining in the neptunium-bearing solution would be separated from the neptunium. The neptunium-bearing solution from the second solvent extraction step would be mixed with a solution of 30 percent tributyl phosphate in normal paraffin hydrocarbon and a solution of 0.1 *M* nitric acid and 0.1 *M* ferrous sulfamate. The ferrous sulfamate would act as a reductant to ensure that the plutonium would remain in the +3 oxidation state and in the aqueous phase. The neptunium would be extracted into the organic phase. After separation of the aqueous and organic phases, neptunium would be stripped from the organic phase using a solution of 0.2 *M* nitric acid and 0.1 *M* hydroxylamine nitrate. The purified neptunium-bearing solution then would be stored in a tank below Cell D until needed for conversion to an oxide for fabrication into targets, as described in Section A.1.3.

A.1.4.3 Plutonium Purification and Preparation of Plutonium Oxide

Although the necessity for further plutonium purification from the second solvent extraction has not yet been fully determined, provisions would be made to purify the plutonium-bearing solution using an anion exchange process. This process has been used previously at REDC to purify plutonium products in preparation for precipitation and calcination to an oxide product. If the plutonium product solution from the second-cycle solvent extraction process meets the desired specifications, no anion exchange processing would be needed and the solution would be sent directly to the oxalate precipitation process. If purification were required, the feed solution would be adjusted to a high acid concentration (approximately 8 *M* nitric acid) and the oxidation states of the actinides would be adjusted to form the metal nitrate complex that loads on the anion exchange

resin. The oxidation state of the metals would be adjusted using a reductant such as ferrous sulfate or ferrous sulfamate. After loading the plutonium onto the resin, the resin would be washed with an acid solution to remove impurities, and the plutonium would be selectively stripped with a dilute acid solution or a dilute acid solution containing reductants (hydroxylamine nitrate, nitrous acid, or hydrazine) to reduce the oxidation state of the loaded complex and strip it from the anion resin column.

The plutonium product from the second-cycle solvent extraction process or the anion exchange process would be adjusted to give a final solution of 1 *M* nitric acid with plutonium in the +4 oxidation state. Ascorbic acid would be used as the reductant to adjust the plutonium oxidation state because it would not add any extraneous ions to the final product upon conversion to the oxide. Two additions of 1 *M* oxalic acid would be made to the plutonium solution to quantitatively precipitate plutonium oxalate. The solution would be filtered and the plutonium oxalate collected on a sintered platinum/Inconel filter for calcination to the oxide. The precipitated plutonium oxalate and filter then would be transferred to a furnace and calcined in air at 735 °C (1,355 °F) for 2 hours to produce plutonium oxide.

The alpha-neutron reaction, which occurs when alpha particles emitted from plutonium-238 interact with atoms of naturally occurring oxygen, results in a high neutron emission rate from plutonium oxide, and may cause a high neutron exposure to workers. To reduce this exposure, an oxygen exchange process would be used to replace the higher cross-section oxygen isotopes (oxygen-17 and oxygen-18) with oxygen-16, which has a very small cross section for the alpha-neutron reaction. To accomplish this exchange, a stream of oxygen-16 enriched gas would be passed though the plutonium oxide product from the calcination step, above, at a temperature of approximately 800 °C (1,470 °F) for approximately 4 hours. The progress of the exchange reaction would be constantly monitored with a neutron detector located adjacent to the plutonium oxide. After completion of the exchange reaction, the plutonium oxide would be cooled in an inert atmosphere and immediately transferred to a container (e.g., EP–60) for final packaging (see Section J.3.3.3 for a further discussion of this packaging system).

A.1.5 Plutonium-238 Storage Description

A container (e.g., EP–60) containing plutonium-238 would be placed into a primary containment vessel (e.g., EP–61), which then would be placed into a secondary containment vessel (e.g., EP–62) and stored until shipment to LANL. The secondary containment vessel then would be loaded into a shipping package (e.g., 5320B). DOE anticipates about four shipments per year to LANL, as described in Appendix J.

A.2 FLUORINEL DISSOLUTION PROCESS FACILITY

A.2.1 Facility Description

The Idaho Nuclear Technology and Engineering Center (INTEC) is located northeast of the Central Facilities Area at INEEL and approximately 3.2 kilometers (2 miles) southeast of ATR. Two buildings at INTEC are proposed storage and processing sites for plutonium-238 production: Building CPP–651, the Unirradiated Fuel Storage Facility, and Building CPP–666, FDPF and Fuel Storage Facility.

Building CPP–651 was originally designed for the storage of special nuclear materials to support Defense Programs and is quite flexible in terms of the size and shape of special nuclear materials that it can receive and store. The 100 storage positions in the vault use the existing structural barriers of Building CPP–651 (earth and concrete) and provide supplemental security protection via their in-ground concrete storage silo design. Each storage position houses a rack that holds seven highly enriched uranium product cans. Racks are raised and lowered in their storage positions via an overhead 1-ton hoist.

Building CPP–666 is divided into two parts, the Fuel Storage Facility and FDPF. The Fuel Storage Facility consists of receiving and unloading areas, a fuel unloading pool, and six storage pools for storing nuclear fuel.

FDPF was designed and built to process Navy fuel via three dissolver trains. When fuel reprocessing was discontinued, uranium and hazardous materials were flushed from FDPF, and the facility is currently under consideration for new missions. FDPF consists of a large hot cell and supporting areas with a total area of approximately 3,700 square meters (40,000 square feet). The facility is divided into five levels identified by their elevation relative to ground level (Hochhalter 1982). A floor plan of the +28-foot level, the proposed location for the neptunium glovebox train and the target fabrication glovebox, is shown in **Figure A–5**.

The FDPF cell is approximately 6 meters (20 feet) wide, 30 meters (100 feet) long, and 15 meters (50 feet) deep (Sire et al. 1992) with 1.8-meter-thick (6-foot-thick) concrete walls. A plan view of the FDPF cell is shown in **Figure A–6**. The cell includes manipulators, three dissolvers, off-gas cleanup systems, complexing vessels, process makeup vessels, pumps, valves, piping, and instrumentation (Hochhalter 1982). The interim storage rack area located in the south end of the cell could be used to store irradiated targets. A new 12.5-liter (3.3-gallon) or 200-liter (53-gallon) dissolution vessel would need to be installed. This dissolution vessel would be located in proximity to the existing Train One dissolver to facilitate the use of the existing dissolver off-gas system (Kirkham 1999).

A.2.2 Neptunium Storage

Neptunium-237 oxide would be shipped from SRS to INEEL in double-sealed containers loaded in Type B packages. At INEEL, it would be stored in the Building CPP–651 vault. There are 100 in-ground concrete-shielded storage positions (each approximately 25 centimeters [10 inches] in diameter by 2.4 meters [8 feet] in length) in this vault. Each storage position houses a rack that holds seven highly enriched uranium product cans, with each can containing about 8 to 10 kilograms (18 to 22 pounds) of highly enriched uranium (the total mass of the rack and cans is about 148 kilograms [325 pounds]). The design-basis radiation level for these cans is about 800 millirem per hour at contact. The rack that fits into the storage position can be redesigned for neptunium oxide containers if its existing dimensions are not adequate. Alternatively, neptunium-237 oxide may be stored in the processing hot cell (Cook and Hill 1999).

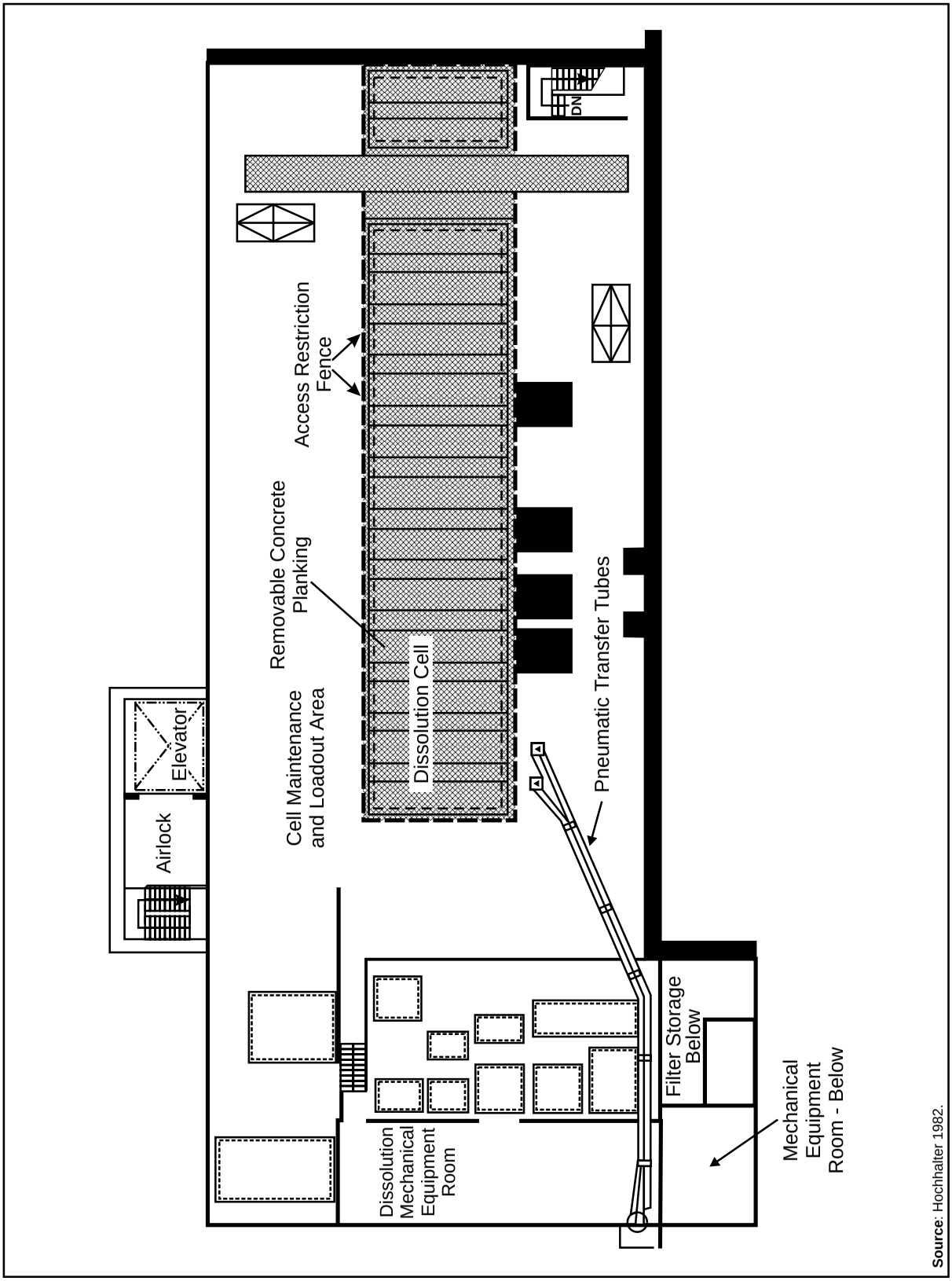
A.2.3 Neptunium-237 Target Fabrication Process Description

The neptunium-237 target fabrication process used at INEEL would be similar to that described in Section A.1.3 for target fabrication in REDC at ORNL and in more detail in *Preconceptual Design Planning for Chemical Processing to Support Pu-238 Production* (Wham et al. 1998). In addition to the target design considered in that document, INEEL could produce targets suitable for a commercial reactor or for an accelerator. The latter target is significantly longer than those considered by ORNL. The target fabrication process would include neptunium-237 purification, neptunium oxide production, and target fabrication.

A.2.3.1 Neptunium-237 Purification

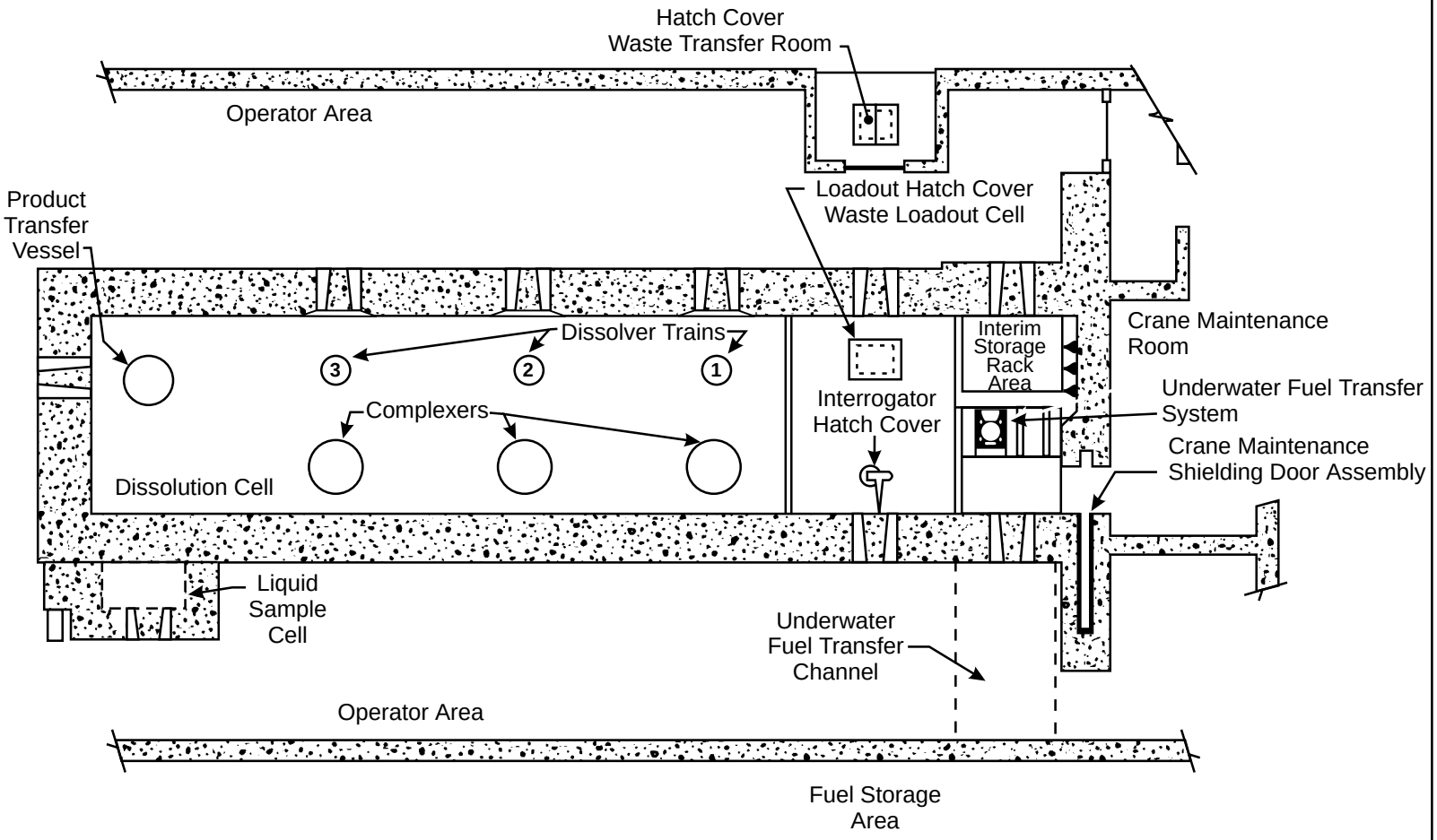
Neptunium-237 oxide retrieved from storage would be purified by dissolving the material in nitric acid and passing the neptunium nitrate solution through a silica gel bed. During this process, the protactinium would adsorb onto the silica gel, leaving a protactinium-233-free neptunium nitrate solution. This process would either be conducted in the hot cell near the third-cycle solvent extraction equipment or within the neptunium processing glovebox train at the +28-foot level.

Neptunium nitrate recovered from postirradiation processing should not be contaminated by protactium-233, as this isotope should have been removed in the solvent extraction process. However, if neptunium-237 target



Source: Hochhalter 1982.

Figure A–5 Fluorinel Dissolution Process Facility +28-Foot Level



Source: Sire et al. 1992.

Figure A–6 Plan View of the Fluorinel Dissolution Process Facility Cell

fabrication were delayed, the ingrowth of protactinium-233 would require that the solution be passed through a silica gel bed to remove the protactinium prior to formation of neptunium oxide (Kirkham 1999).

A.2.3.2 Neptunium Oxide Production

Neptunium oxide production would be accomplished using the same process described for ORNL (Section A.1.3.2 and Wham et al. 1998). Conversion of the purified neptunium nitrate solution would be accomplished by adsorption of the neptunium on a cation resin bed in a glovebox train located on the cell maintenance level (+28-foot level). Solutions from the hot cell would be pumped from in-cell storage through a K-plug installed in an existing penetration. The routing would be through doubly contained lines run through the +17- and +28-foot levels from the operating level (0-foot level). (Note: The glovebox train may be located on any level on either side of the cell. The +28-foot level has ample area available for the glovebox train; however, it may be desirable to locate the glovebox train elsewhere because of better shielding and protection to the outside environment.) The resin then would be calcined to create microspheres of neptunium dioxide powder of the required morphology. All waste solutions from this process would be transferred to the cell waste collection system. The off-gas from the glovebox train would be filtered with a high-efficiency particulate air filtering system and tied into the existing cell off-gas system (Kirkham 1999).

A.2.3.3 Neptunium-237 Target Fabrication

Neptunium-237 target fabrication would be accomplished using the same target fabrication and assembly methods described for ORNL (Section A.1.3.3 and Wham et al. 1998). The proposed target design for ATR or HFIR consists of neptunium dioxide blended with aluminum powder, pressed into a target core, and clad with aluminum. The target used for ATR would be similar in appearance, but longer than the target that would be used for HFIR. (The ATR target length would be sized for the 1.2-meter [4-foot] active core length of ATR rather than the 51-centimeter [20-inch] active length of HFIR.) Target fabrication would be performed in an area adjacent to the neptunium glovebox train on the +28-foot level. The target fabrication operations would be performed in a shielded glovebox to provide adequate personnel protection from the varying dose rate. The target pins would be transferred to a shielded glovebox where they would be assembled into a target assembly. When completed and accepted, a target assembly would be removed from the glovebox, undergo required nondestructive assay analysis and verification, and be moved by elevator to the ground level and then by vehicle to the use location or to storage in the security area (CPP–651) (Kirkham 1999).

A.2.4 Postirradiation Processing Description

Postirradiation processing of neptunium-237 targets would use a similar process to the one described for ORNL in Section A.1.4 and in more detail in the preconceptual design study (Wham et al. 1998). An exception to this process is that aluminum-clad target dissolution would use a one-step dissolution using a nitric acid-fluoroboric acid solution instead of the two-step process that would be used by ORNL. There is adequate shielded cell floor space in FDPF to do the recovery operation.

A.2.4.1 Target Dissolution

Irradiated targets would be unloaded in the Fuel Storage Facility pool from the transfer cask. (This could be one of several approved existing casks used to transfer spent nuclear fuel.) The target container would be loaded under water into the transfer cart and moved up the incline channel into the FDPF cell. The targets may be placed in interim storage in the existing fuel storage ports in the cell or loaded directly into the dissolver. The dissolution would take place in a new 12.5-liter (3.3-gallon) vessel with an overflow into an existing dissolver vessel for continuous dissolution. Alternatively, a new vessel with about a 200-liter (53-gallon) capacity could be used for a batch process. This equipment would be located near the existing Train One

dissolver. Each aluminum-clad target from ATR or HFIR would be dissolved in 120 liters (31.7 gallons) of 7.5 *M* nitric acid and 0.15 *M* fluoroboric acid at 100 °C (212 °F). This dissolution results in the neptunium and plutonium being highly complexed with fluoride. Therefore, to perform a solvent extraction separation, the fluoride would have to be complexed to free the actinide ions. This could be accomplished by adding zirconium (IV) nitrate (Cook and Hill 1999).

At ORNL and Hanford, targets with a stainless steel or Zircaloy cladding (e.g., CLWR, FFTF, or high-energy accelerator targets) would be cut into pieces and leached with nitric acid or another suitable solution to dissolve the neptunium, plutonium, and fission products away from the insoluble cladding. The solution would be filtered and the undissolved cladding would be discarded as waste. At INEEL, the stainless steel- or Zircaloy-clad targets would be dissolved in a one-step process using a suitable solution.

A.2.4.2 Plutonium Separation and Neptunium Recycling

The process solution would be transferred by a positive displacement pump to the solvent extraction system, which would use centrifugal contactors. The solvent extraction system would be located at the south end of the cell near manipulators. Crucial components would be skid-mounted and within reach of manipulators. Because of the small flow rates needed, reagents would be fed into the separation system from small feed vessels located on the +28-foot level or in the operating corridor. Chemical makeup would take place in the existing makeup area with transfers to these smaller vessels. The acidic target solution would be treated with tributyl phosphate dissolved in normal paraffin hydrocarbon in three trains of centrifugal contactors and the plutonium and neptunium would be extracted into the tributyl phosphate/normal paraffin hydrocarbon phase.

Four separate transfer lines connect the FDPF hot cell with Building CPP–601, where the waste can be transferred to Building CPP–604. Building CPP–604 houses the Process Waste Evaporator, which would be used to evaporate the liquid waste for subsequent disposal. Aqueous waste streams would be collected in one of the existing complexer vessels, transferred to the existing product transfer vessel, and bled off to the Process Equipment Waste Evaporator system. An intercycle evaporator might be necessary to concentrate the first cycle strip before the second cycle. It would be located in the cell area south of and near the Train One dissolver or complexer vessel and suspended below the grating at the 0-foot level. The condensate would be routed to waste collection, with the concentrated solution going to the second solvent extraction cycle to separate neptunium and plutonium from each other. This extraction cycle also would use centrifugal contactors. The partitioned neptunium would go to the third solvent extraction cycle where it would be purified of any remaining fission products and stored for conversion to neptunium oxide (Section A.2.3). The plutonium-bearing solution would be purified by ion exchange and stored in the cell for subsequent processing to plutonium oxide.

A.2.4.3 Preparation of Plutonium Oxide

The conversion of plutonium nitrate solution to plutonium oxide would be accomplished in the same manner identified in the ORNL preconceptual design study for plutonium-238 production support operations (Wham et al. 1998). Plutonium oxide conversion would occur in shielded gloveboxes on the +28-foot level. The plutonium nitrate solution would be treated to adjust the plutonium oxidation state and then precipitated as an oxalate. The plutonium oxalate would be washed with dilute acid and calcined at the required temperature. The required oxygen-16 exchange would be done as part of the calcination step. The oxide would be packaged in a manner to assure maintenance of the desired degree of oxygen-16 exchange. The packaged material would undergo nondestructive assay analysis for accountability control and placed into storage pending packaging for shipment.

A.2.5 Plutonium-238 Storage Description

Plutonium-238 oxide would be transferred to the security area in Building CPP–651 for storage prior to shipment to LANL. Storage quantities would be expected to be small, as shipments would be made on a regular basis.

A.3 FUELS AND MATERIALS EXAMINATION FACILITY

A.3.1 Facility Description

FMEF is located in the 400 Area on Hanford adjacent to the FFTF. Constructed in the late 1970s and early 1980s to perform fuel fabrication and development, and postirradiation examination of breeder reactor fuels, FMEF is being maintained in a condition suitable for a future mission. The building is clean and uncontaminated, as no nuclear materials have been introduced. FMEF has been well maintained for potential future missions (Hoyt et al. 1999).

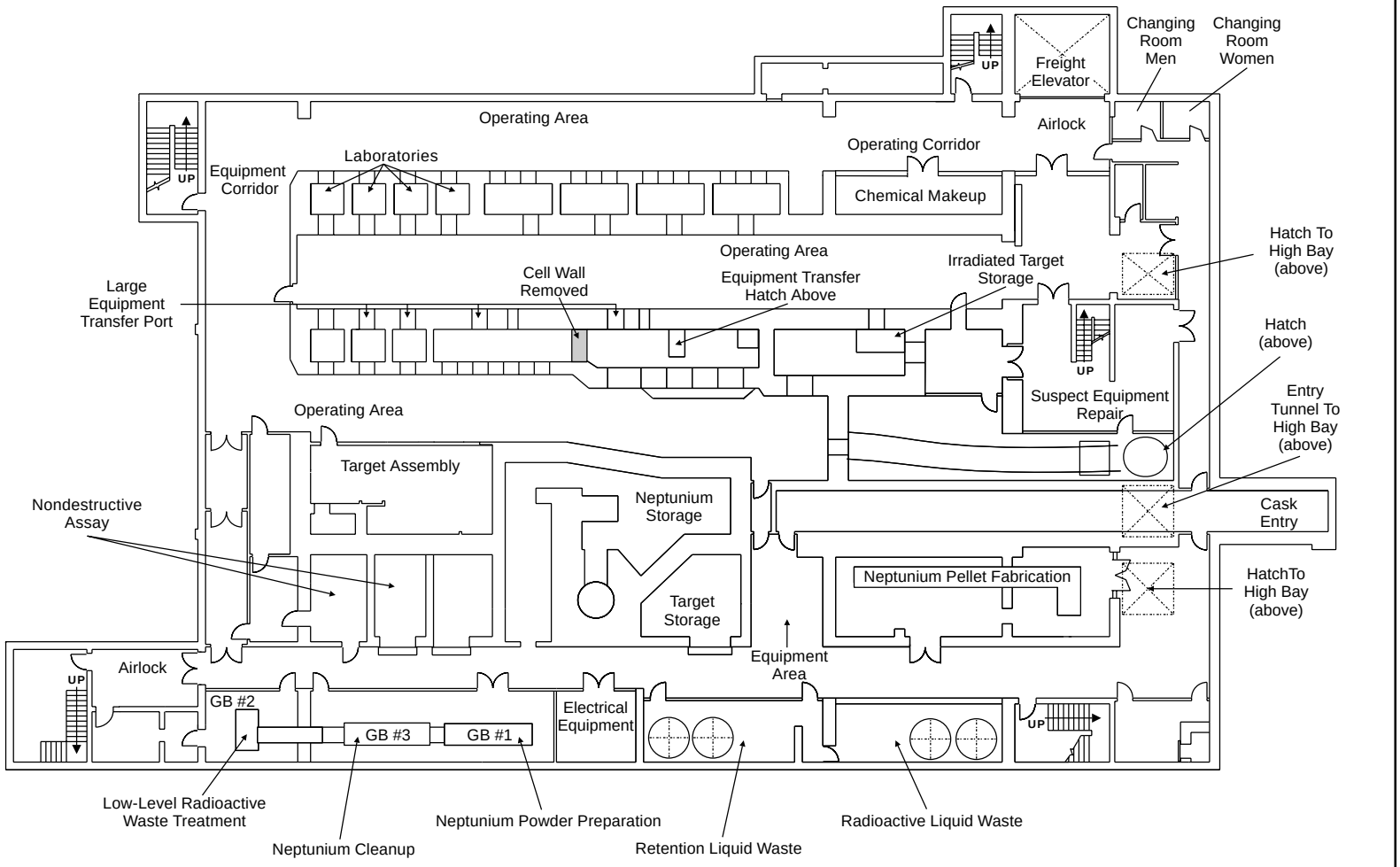
FMEF consists of the Process Building with an attached Mechanical Equipment Wing on the west side and an Entry Wing across the south side of the building. The Mechanical Equipment Wing houses facility utility and support equipment. The Entry Wing provides space for reactor fuel assembly, a lunchroom, change rooms, a security station, office space, and administrative support areas (DOE 1995).

The Process Building is 53.3 meters (175 feet) wide by 82.3 meters (270 feet) long and extends from 10.7 meters (35 feet) below grade to 29.7 meters (98 feet) above grade. The total operating space is approximately 17,400 square meters (188,000 square feet). The building is divided into six operating floors, or levels, which are identified by their elevation relative to ground level (DOE 1995). The Process Building contains several large interconnected hot cells and many smaller connected hot cells. However, most cranes, windows, and manipulators were not installed because construction was halted prior to completing work on the hot cell complex (Hoyt et al. 1999).

FMEF has the physical attributes required to process, handle, and store large quantities of special nuclear material. It is a massive, reinforced-concrete, hardened structure with safety-related equipment and systems, designed as a seismic Category 1 to withstand the Hanford design-basis earthquake, tornado, high-wind, and volcanic ashfall events. FMEF was also designed to meet the physical safeguards and security requirements for processing and storing Category 1 quantities of special nuclear material (Hoyt et al. 1999).

Ample space exists in FMEF for plutonium-238 production support, and numerous facility configurations are possible. In the absence of a detailed engineering study, it was decided that the process support would be located at the -35-foot level using the process support cells to house the irradiated target processing. This configuration also would contain this project, with its relatively modest requirements, to as few levels as possible. Alternative facility configurations can be found in the *Summary of Strategy for Implementing Plutonium-238 Production Support Activities in FMEF* (Hoyt et al. 1999). A floor plan of the -35-foot level is shown in **Figure A–7**.

The shipping and receiving bay located on the 0-foot level would be used to support the shipment and receipt of safe, secure trailer/SafeGuards Transports and irradiated target cask transporters. Additional facilities on the 0-foot level would be used to transfer irradiated targets into the storage area, decontaminate, and prepare equipment for maintenance, and package remote-handled solid waste for disposal. On the -17-foot level, the entry tunnel transporter would be used, as well as existing facility systems, as needed (Hoyt et al. 1999).



Notes: To convert feet to meters, multiply by 0.3048;
GB, gloveboxes.
Source: Hoyt et al. 1999.

Figure A–7 Layout of the Fuels and Materials Examination Facility

The -35-foot level would house most of the processing and storage functions for plutonium-238 production. Neptunium-237 storage, target fabrication, and assembly would be located in rooms on the south side of the -35-foot level and include Rooms 112, 114, 124, and 128 (Hoyt et al. 1999).

The south bank of process support cells would be dedicated to target processing. Located on the -35-foot level, the 14 process support cells are arranged in two parallel rows along a horizontal transfer corridor. The process support cell complex is approximately 12.1 meters (40 feet) wide by 30.2 meters (99 feet) long. With the exception of Cell 146, each of the process support cells is 4.3 meters (14 feet) high and lined with stainless steel. Cell 146 extends to the 0-foot level and would be lined with stainless steel for the proposed project. The process support cell area is heavily shielded with either 122 centimeters (48 inches) or 81 centimeters (32 inches) of high-density concrete. Work in the cells would be performed using remotely operated equipment (DOE 1997).

Irradiated neptunium targets would be lowered through a hatch into Cell 147 and stored awaiting processing. Target processing would begin in Cell 146 and proceed through to plutonium-238 oxide conversion, storage, and loadout in Cell 142. The main target processing activities would occur in Cell 146. Existing wastewater collection systems would be used, and hot repair facilities also would be available on this level (Hoyt et al. 1999).

A.3.2 Neptunium Storage

Neptunium-237 oxide would be shipped from SRS to Hanford in double-sealed containers loaded in Type B packages. At Hanford, it would be stored in a vertical tube rack located in Room 114 on the -35-foot level of FMEF. This room was designed earlier to contain the TRIGA (training, research, isotopes General Atomics) reactor spent nuclear fuel and would provide excellent shielding capabilities. The room would be modified to provide storage racks to hold the 3013 containers. Individual 3013 container transfers would be accomplished remotely with an overhead crane to minimize personnel exposure.

A.3.3 Neptunium-237 Target Fabrication Process Description

The neptunium-237 target fabrication process used at Hanford would be similar to that described in Section A.1.3 for target fabrication in REDC at ORNL and in more detail in *Preconceptual Design Planning for Chemical Processing to Support Pu-238 Production* (Wham et al. 1998). In addition to the target designs considered in that document, Hanford could produce targets suitable for FFTF, a commercial reactor, or an accelerator. The latter target is significantly longer than those considered by ORNL. The target fabrication process would include neptunium-237 purification, neptunium oxide production, and target fabrication.

A.3.3.1 Neptunium-237 Purification

Neptunium-237 oxide retrieved from storage would be purified by dissolving the material in nitric acid and passing the neptunium-nitrate solution through a silica gel bed. During this process, the protactinium would adsorb onto the silica gel, leaving a neptunium nitrate solution that is free of protactinium-233.

For neptunium nitrate recovered from postirradiation processing, the protactinium-233 should have been removed in the solvent extraction process. However, if any additional purification were required, this would be achieved either through an anion exchange or an oxalate precipitation process. If oxalate precipitation were to be used, the neptunium oxalate precipitate would be washed to remove entrained liquor, and redissolved in nitric acid (Kirkham 1999).

A.3.3.2 Neptunium Oxide Production

Conversion of the purified neptunium nitrate solution would be accomplished by adsorption of the neptunium on a cation resin bed. The resin then would be calcined to create microspheres of neptunium dioxide powder of the required morphology.

A.3.3.3 Neptunium-237 Target Fabrication

Neptunium-237 target fabrication would be accomplished using the same target fabrication and assembly methods described for ORNL (Section A.1.3.3 and Wham et al. 1998). Target fabrication would be performed in FMEF Room 124, which would be modified by removing a wall in the area to create sufficient unencumbered floor space. The target fabrication operations would be performed in a shielded glovebox to provide adequate personnel protection from the varying dose rate. The target pins would be transferred to a shielded glovebox in Room 128, where they would be assembled into a target assembly. When completed and accepted, a target assembly would be removed from the glovebox, undergo required nondestructive assay analysis and verification, and be packaged for interim storage in Room 112 on the -35-foot level pending shipment to the reactor site. Fuel storage tubes located in the Fuel Assembly Area also could be used. Final selection would depend on protection requirements and the optimized building configuration (Kirkham 1999).

A.3.4 Postirradiation Processing Description

Postirradiation processing of neptunium-237 targets would use the same process described for ORNL in Section A.1.4 and in more detail in the preconceptual design study (Wham et al. 1998). There is adequate shielded cell floor space on the -35-foot level in FMEF to do the recovery operation. The cells would have either 122 centimeters (48 inches) or 81 centimeters (32 inches) of high-density concrete as shielding walls.

A.3.4.1 Target Dissolution

The targets would be transferred from their storage position in Cell 147 to the dissolving cell, Cell 146. There the extraneous hardware would be removed and the target pins would be sheared into small segments and placed into a dissolver vessel located near the shear in Cell 146. Segments from targets irradiated in HFIR or ATR would be agitated in a caustic solution to dissolve the aluminum cladding and aluminum target diluent. After removal of the caustic aluminum waste solution, the neptunium target would be treated with nitric acid to dissolve the neptunium, plutonium, and fission products.

Stainless steel- or Zircaloy-clad targets irradiated in the CLWR, FFTF, or the high-energy accelerator would be cut into small pieces and leached with nitric acid or other suitable solution to dissolve the neptunium, plutonium, and fission products away from the insoluble cladding. The solution would be filtered and the undissolved cladding would be discarded as waste.

A.3.4.2 Plutonium Separation and Neptunium Recycling

The acidic target solution would be treated with tributyl phosphate dissolved in normal paraffin hydrocarbon in a mixer-settler vessel and the plutonium and neptunium would be extracted into the tributyl phosphate/normal paraffin hydrocarbon phase. The aqueous phase containing the mixed fission products would be treated again to remove trace quantities of neptunium and plutonium. Waste-handling equipment would be used to minimize the activity in low-level radioactive liquid waste and to stabilize solid waste into an acceptable waste form. This equipment would be included in the hot cells used for the chemical processing of irradiated targets for plutonium-238 production. The neptunium and plutonium would be extracted from the tributyl phosphate/normal paraffin hydrocarbon solution as an aqueous nitrate solution. The neptunium

and plutonium next would be separated from each other by additional solvent extraction and stored in their respective storage tanks pending further purification and conversion to oxides.

Alternatively, an anion exchange process may be added to the existing FMEF for separating fission products and purifying neptunium-237 and plutonium-238 from irradiated targets using a series of ion exchange columns.

A.3.4.3 Preparation of Plutonium Oxide

The conversion of plutonium nitrate solution to plutonium oxide would be accomplished in the same manner identified in the ORNL preconceptual design study for plutonium-238 production support operations (Wham et al. 1998). Plutonium oxide conversion would occur in Cell 144. The plutonium nitrate solution would be treated to adjust the plutonium oxidation state and then precipitated as an oxalate. The plutonium oxalate would be washed with dilute acid to remove entrained liquor, transferred to a boat and calcined at the required temperature. The required oxygen-16 exchange would be done as part of the calcination step. The oxide would be packaged in a manner to assure maintenance of the desired degree of oxygen-16 exchange. The packaged material would undergo nondestructive assay analysis for accountability control and then be placed into storage pending packaging for shipment.

A.3.5 Plutonium-238 Storage Description

Plutonium-238 oxide would be stored in racks in Cell 143 on the -35-foot level. Storage quantities would be expected to be small, since shipments would be made to LANL on a regular basis. Alternatively, the special nuclear material storage vault located in Rooms 428 and 429 on the 21-foot level could be used. In addition, any of several hardened rooms within FMEF could be modified to be vault-type rooms for storage of plutonium-238 oxide packaged in shipping containers awaiting shipment.

A.4 REFERENCES

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Appendix B
Neptunium-237 Target Irradiation Operations in Currently Operating Reactors for Plutonium-238 Production

Neptunium-237 targets can be irradiated with neutrons to produce the plutonium-238 used in heat sources that support National Aeronautics and Space Administration (NASA) missions. The U.S. Department of Energy (DOE) has identified two of its currently operating reactors that have the potential to provide these irradiation services—the Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL) and the High Flux Isotope Reactor (HFIR) at the Oak Ridge Reservation (ORR). A currently operating commercial light water reactor (CLWR) of generic pressurized water reactor design has also been proposed by DOE as a possible source for the irradiation of the neptunium-237 targets. This possibility evolved as a result of DOE's request for Expressions of Interest posted in the January 4, 1999, issue of *Commerce Business Daily* (DOE 1999a).

It is anticipated that the plutonium-238 needed for the NASA missions would be produced for a period of approximately 35 years. The production of this plutonium-238 would not affect the capability of the DOE reactors to support other existing DOE missions or of the CLWR to produce commercial electricity.

Each of the reactor sites has security measures in place, including access control, and procedures to ensure the adequate protection of all materials processed and stored. Descriptions of the reactors and the plutonium-238 production processes specific to each reactor are provided in this appendix.

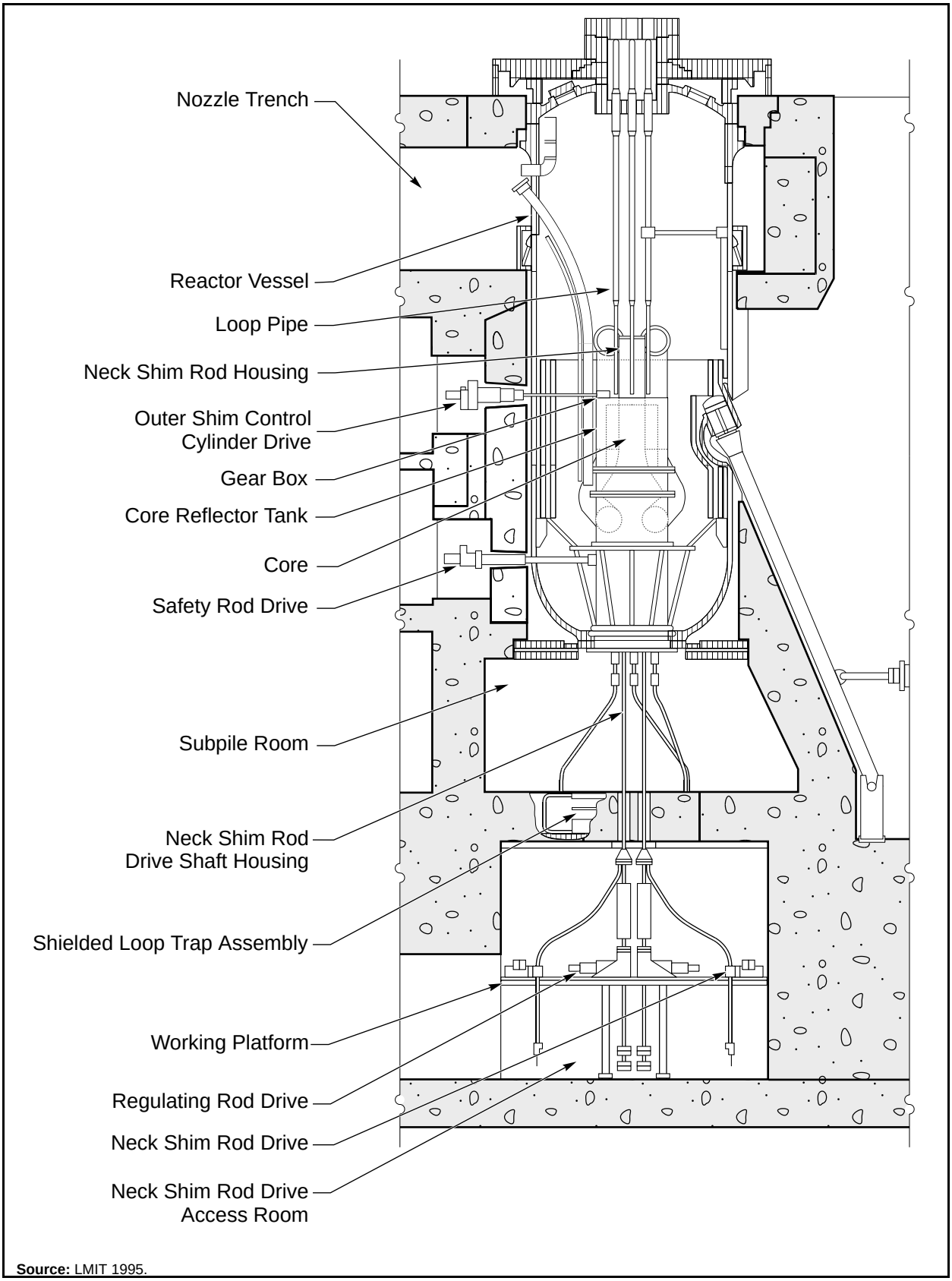
B.1 ADVANCED TEST REACTOR

B.1.1 Facility Description

ATR, located at INEEL, is one of the world's largest and most technologically advanced reactor test facilities. Special features of ATR include high neutron flux levels (ranging from 1×10^{15} neutrons per square centimeter per second in the flux traps to 1×10^{13} neutrons per square centimeter per second in the outer reflector positions) and the ability to vary power to fit different experiment needs in different test positions. The main purpose of ATR is to provide a prototypical reactor test environment for the study of radiation effects on materials and fuel. It is also used to produce radioisotopes for medical, industrial, and research uses. This facility description is based on information provided in the *Advanced Test Reactor, Upgraded Final Safety Analysis Report* (LMIT 1997) and *Capabilities of the Test Reactor Area Featuring the Advanced Test Reactor* (LMIT 1995).

ATR is located within the Test Reactor Area in the southwest portion of INEEL. The reactor, its primary coolant system, control room, and much of its auxiliary and experimental support equipment are located in Building 670. ATR began operation in 1967 and is expected to continue operating for several decades. The reactor vessel is entirely stainless steel and the core internals are replaced every 7 to 9 years, bringing the reactor to an essentially like-new condition (the most recent changeout was completed in 1994 [LMIT 1995]). Buildings and structures in other parts of the Test Reactor Area provide additional support functions.

ATR is a light-water-moderated and -cooled reactor with a design thermal power of 250 megawatts. The reactor typically operates at approximately 140 megawatts or less. Typical operating cycles are 42 days or 49 days at power followed by a 7-day outage for refueling and changeout of experiments and isotope production targets. The core is 1.2 meters (4 feet) high and is surrounded by a 1.3-meter-diameter (4.25-foot-diameter) beryllium reflector. Beryllium is an excellent neutron reflector and is used to enhance the neutron flux essential to a test reactor. The location of the core in the ATR vessel is shown in **Figure B–1**. ATR has nine flux traps in its core and achieves a close integration of flux traps and fuel by means of a



Source: LMIT 1995.

Figure B–1 Vertical Cross Section of the ATR Vessel

serpentine fuel arrangement (**Figure B–2**). When viewed from above, the ATR fuel region resembles a four-leaf clover. The four flux traps positioned within the four lobes of the reactor core are almost entirely surrounded by fuel, as is the center position. Four other flux trap positions between the lobes of the core have fuel on three sides. The ATR's unique control device design permits large power shifts among the nine flux traps. Testing can be performed in test loops installed in some flux traps with individual flow and temperature control or in reflector irradiation positions with primary fluid as coolant. The curved fuel arrangement brings the fuel closer on all sides of the test loops than is possible in a rectangular grid.

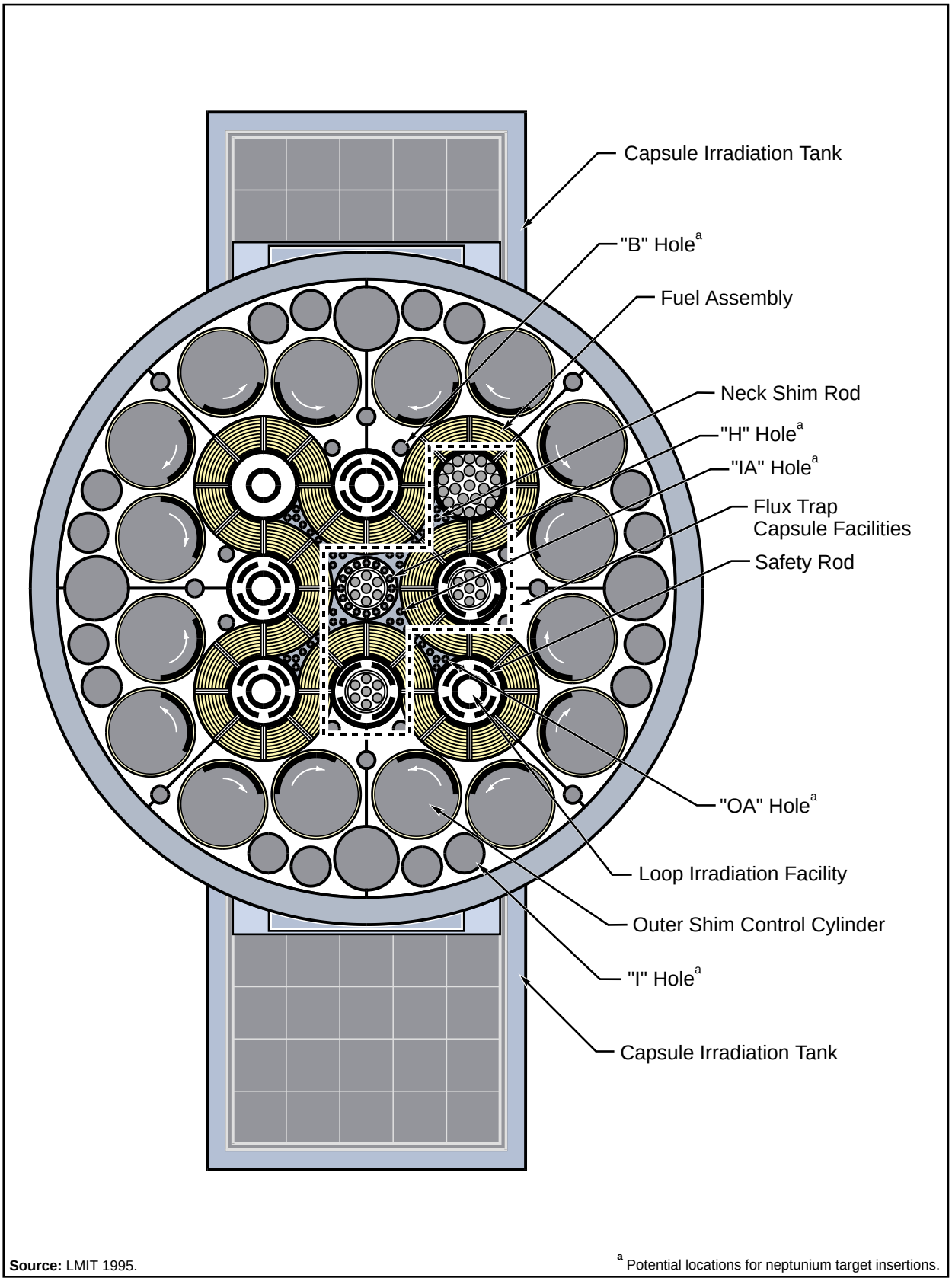
Of the nine flux traps, five are configured with pressurized-water loops that allow for individual temperature, pressure, flow, and chemistry controls. The five test loops are used by the Naval Reactors program. Of the remaining four flux traps, one is dedicated to the Naval Reactors program, one is used for isotope production, one is used for low-specific-activity cobalt production, and the fourth has recently had the Irradiation Test Vehicle installed. The Irradiation Test Vehicle can be described as three small pressurized-gas test loops. Use of one of these three has recently been purchased by a British corporation; negotiations for use of the other two are currently under way.

In addition to the primary flux trap irradiation positions, there are some 70 irradiation positions in the beryllium reflector (and aluminum support structure) that are available for experiment irradiation and isotope production. These position diameters range from 1.6 to 12.7 centimeters (0.625 to 5 inches) with thermal neutron flux levels ranging from 1×10^{15} to 1×10^{13} neutrons per square centimeter per second.

Approximately 25 percent of the high-flux test positions (A holes, B holes, and H holes) are currently used for iridium-192 production. The majority of the remaining high-flux test positions are used for cobalt-60 production. Occasionally, additional isotopes (e.g., strontium-89, nickel-63) are generated in small quantities. A private company leases the space for the production of these isotopes. A small number of positions are used by other companies or Government programs for other material irradiation projects. For the production of plutonium-238, neptunium-237 targets would be placed in the beryllium reflector positions. The proposed target design consists of neptunium dioxide blended with aluminum powder, pressed into a target core, and clad with aluminum. The basic ATR target should be similar in appearance to, but longer than, the typical transuranic isotope production target shown in **Figure B–3**. The ATR target length would be sized for the 1.2-meter (4-foot) active core length of ATR rather than the 0.51-meter (20-inch) active length of the HFIR target. Beryllium reflector position sizes range from 1.6 to 12.7 centimeters (0.625 to 5 inches) in diameter.

ATR is equipped with numerous safety features, including extensive plant protective systems, standby power sources, experiment interlocks, computerized surveillance, confinement systems, safety rods, and an emergency firewater injection system. ATR's six safety rods provide fast shutdown of the reactor if potentially damaging conditions develop. A sudden rise in power or coolant temperature, a sudden drop in coolant flow or pressure, or the overheating of a test sample are examples of approximately 360 conditions that would automatically drop the safety rods into the core. The firewater injection system provides emergency core cooling and flooding of the reactor vessel in the event of a loss of primary coolant.

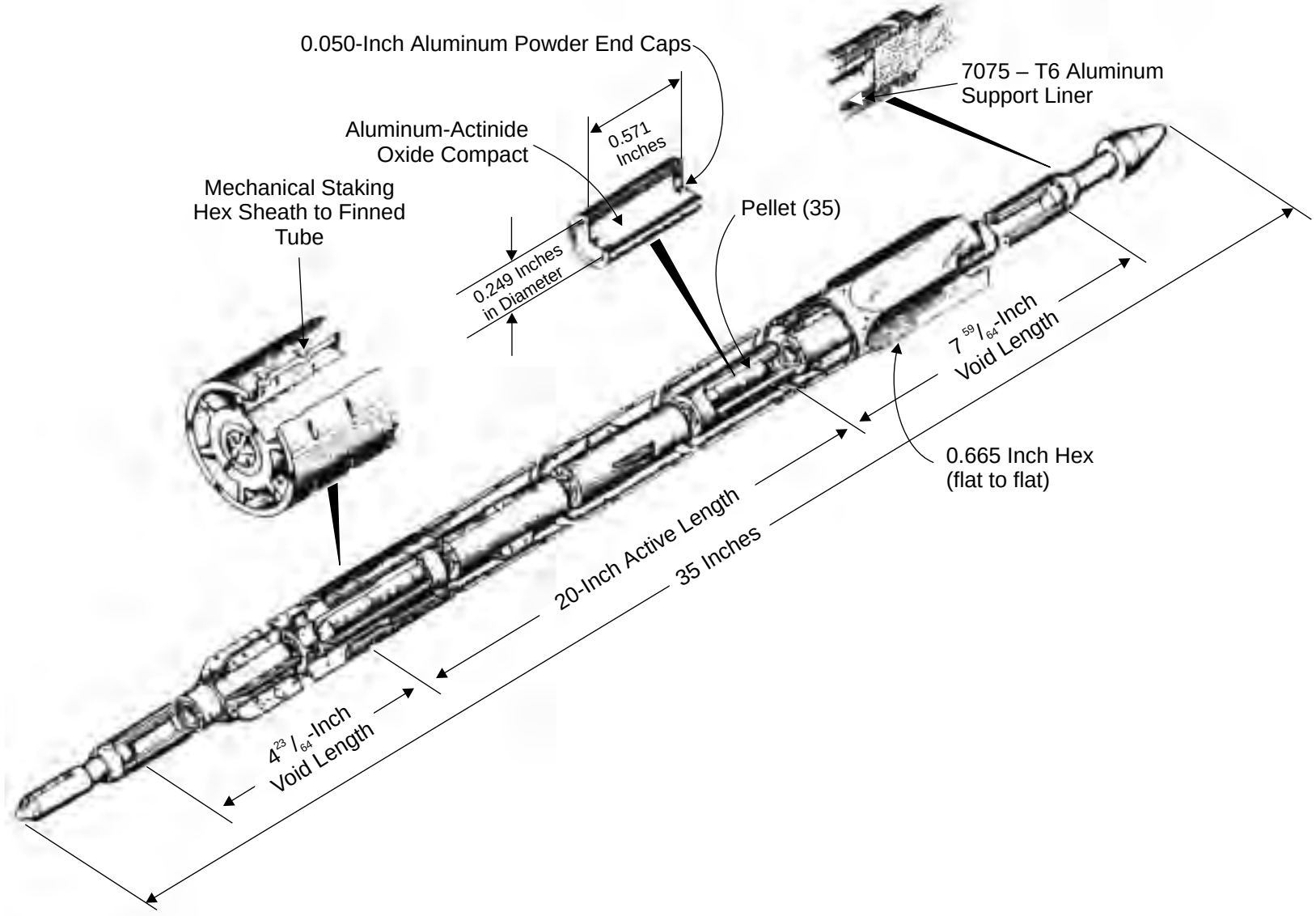
ATR is connected by a water canal to the ATR Critical Facility. The ATR Critical Facility is a low-power, full-size nuclear duplicate of ATR, and is used to provide data, as needed, for experiment loadings prior to irradiation of the actual experiments in ATR.



Source: LMIT 1995.

[a] Potential locations for neptunium target insertions.

Figure B–2 ATR Core Configuration



Source: LMER 1998.

Note: To convert inches to centimeters, multiply by 2.54.

Figure B–3 Typical Transuranic Isotope Production Target

B.1.2 Process Description

The target irradiation operations using ATR at INEEL are illustrated in **Figure B–4**.

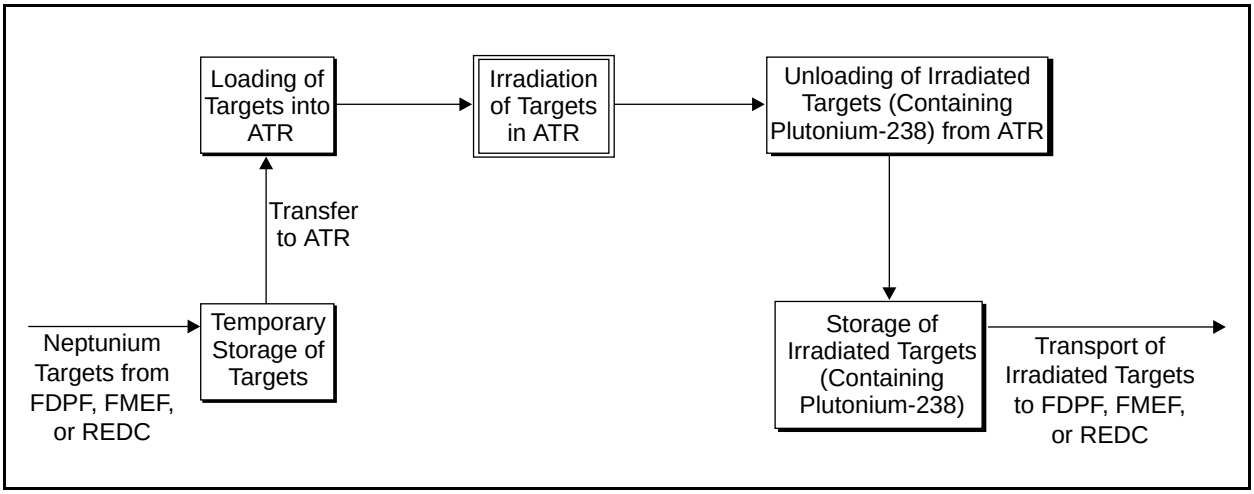


Figure B–4 Target Irradiation Operations Using ATR at INEEL

Following neptunium target fabrication at the Fluorinel Dissolution Process Facility (FDPF) at INEEL, the Fuels and Materials Examination Facility (FMEF) at Hanford, or the Radiochemical Engineering Development Center (REDC) at ORR (Appendix A), the targets would be transported to INEEL for temporary underwater storage in the ATR reactor canal pending insertion into the reactor. Each target would contain, on average, approximately 750 grams (26 ounces) (approximately 0.5 curie) of neptunium-237 and up to an equivalent curie amount of protactinium-233, depending on the elapsed time following the neptunium-237 purification. The targets then would be manually transferred underwater to ATR and inserted into the beryllium reflector area of the reactor. This loading would take about 2 to 4 hours to complete. Nominally, 94 targets would be irradiated concurrently in ATR for a period of about 6 months to 2 years. The length of irradiation depends on the positions of the targets in the reactor. Following irradiation, the targets, each nominally containing, on average, 63 grams (2 ounces) of plutonium-238, smaller amounts of plutonium isotopes with higher atomic weights, and larger amounts of neptunium-237 (Schnitzler 1999), would be removed from ATR using the same underwater manual transfer system used during loading and would be stored in the reactor canal. This unloading would take approximately 2 to 4 hours to complete. The irradiated targets would be stored for a period of approximately 4 to 6 months to allow for the decay of short-lived radionuclides generated during irradiation.

After storage, the irradiated targets would be transported to FDPF, FMEF, or REDC for processing to separate the plutonium-238 product. A discussion of the postirradiation activities at these facilities is provided in Appendix A.

B.2 HIGH FLUX ISOTOPE REACTOR

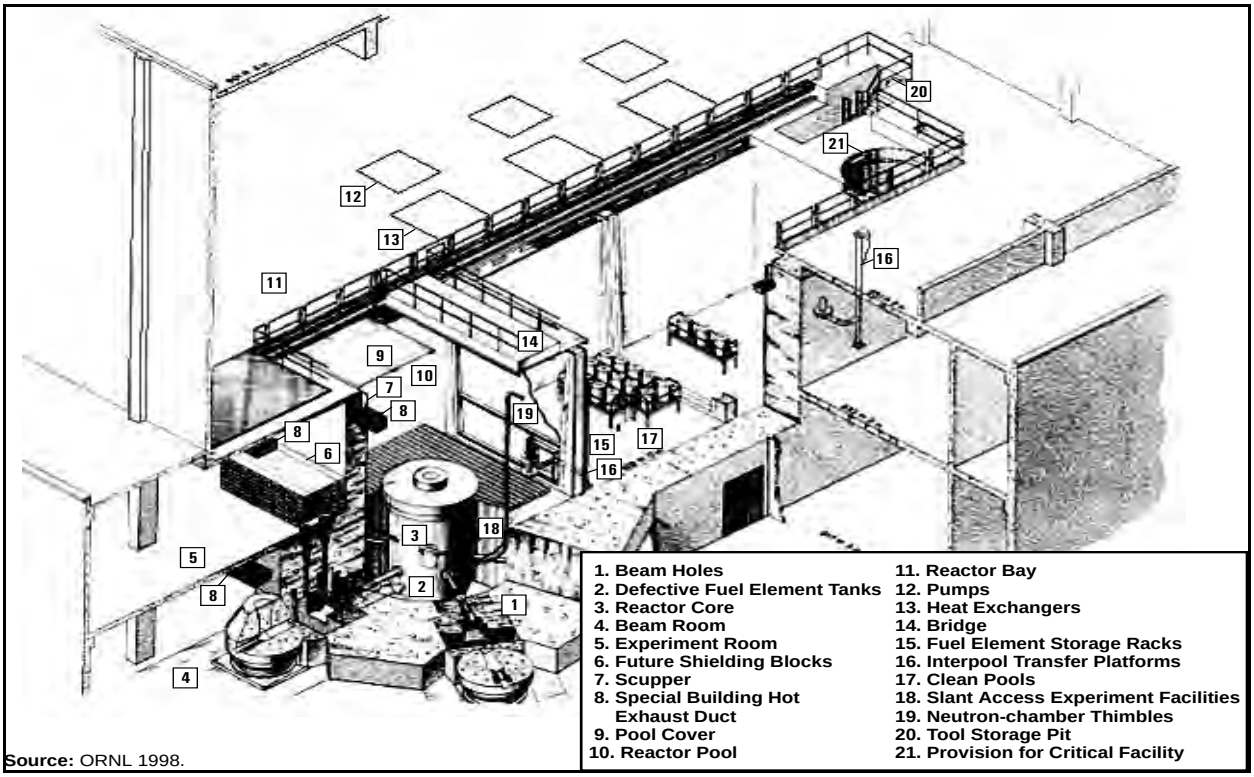
B.2.1 Facility Description

HFIR is located in Building 7900 of the Melton Valley 7900 Complex at ORR. REDC is also part of this complex. HFIR provides one of the highest steady-state neutron fluxes available in any of the world's reactors. The reactor is operated primarily for Neutron Science Research. It is also used to produce isotopes on a not-to-interfere basis and is the Western World's sole producer of heavy transuranium isotopes. This facility

description is based on information provided in the *High Flux Isotope Reactor Safety Analysis Report* (LMER 1998) and the *High Flux Isotope Reactor Facility Description* (ORNL 1998).

HFIR is used for the production of californium-252 and other transuranic isotopes for research, industrial, and medical applications, as well as for a variety of irradiation tests and experiments on a not-to-interfere basis with its primary mission of neutron science research. Each year, approximately 150 to 200 researchers use the experimental facilities at HFIR.

HFIR is a beryllium-reflected, light-water-moderated and -cooled reactor. Originally designed to operate at 100 megawatts, the currently authorized power level is 85 megawatts. The reactor core is 0.61 meter (2 feet) high, and is contained in a 2.44-meter-diameter (8-foot-diameter) pressure vessel located in a pool of water. The top of the pressure vessel is 5.18 meters (17 feet) below the pool surface, and the reactor horizontal midplane is 8.38 meters (27.5 feet) below the pool surface. The pools and experiment facilities are illustrated in **Figure B–5**.

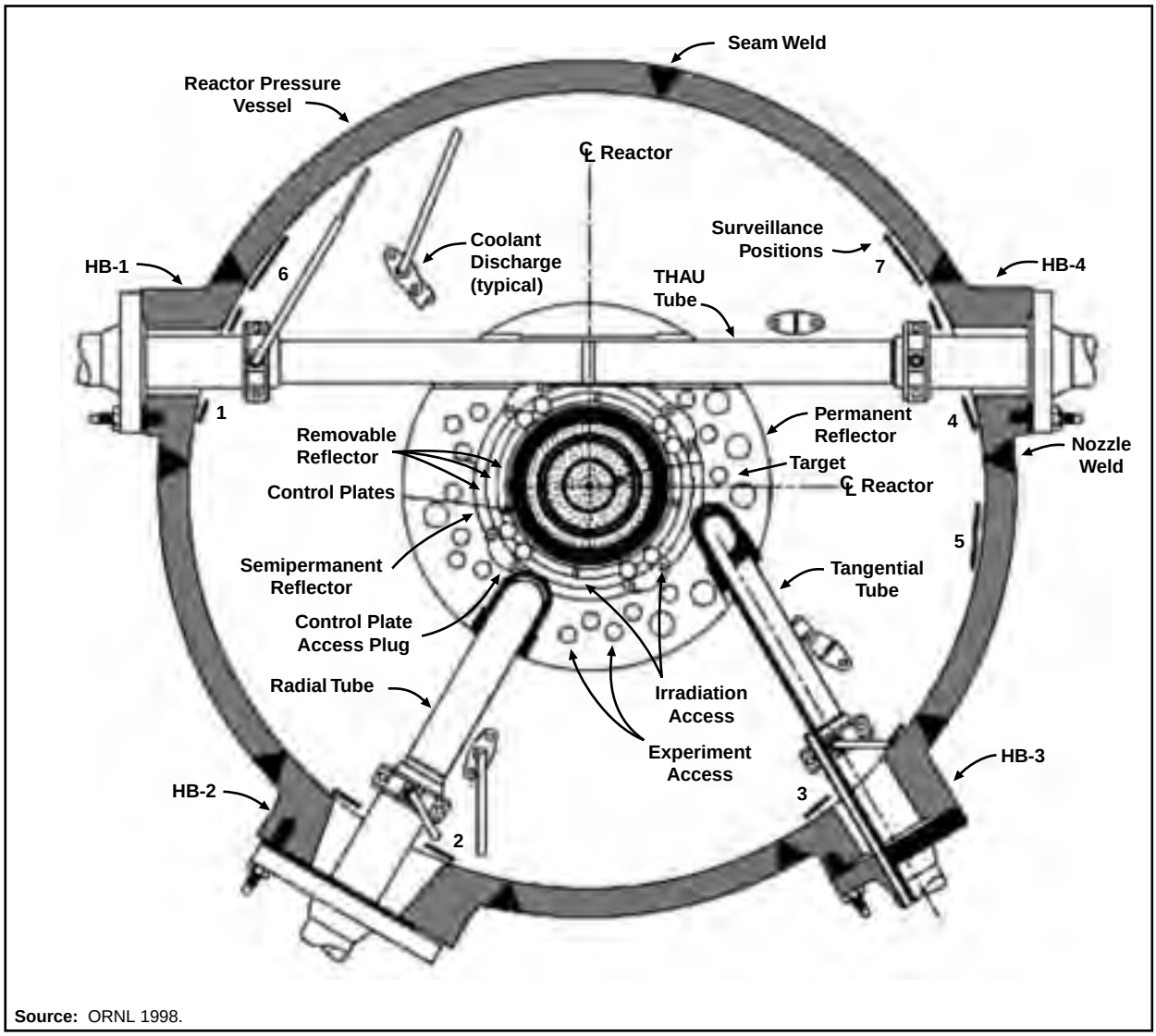


Source: ORNL 1998.

Figure B–5 Pools and Experiment Facilities

HFIR fuel contains enriched uranium-235. The reactor core consists of concentric annular regions. A sectional plan view of the core is shown in **Figure B–6.** The flux trap forms a hole at the center of the core and is 12.70 centimeters (5 inches) in diameter. The fuel region surrounding the flux trap is composed of two concentric fuel elements. The inner element contains 171 fuel plates and the outer element contains 369 fuel plates. The aluminum-clad fuel plates are curved in the shape of an involute (curled spirally), thus providing a constant coolant channel width. A fuel element is illustrated in **Figure B–7**.

The fuel region is surrounded by a concentric ring of beryllium reflector approximately 0.3 meter (1 foot) thick. The beryllium is surrounded by a water reflector of effectively infinite thickness. In the axial direction, the reactor is reflected by water.

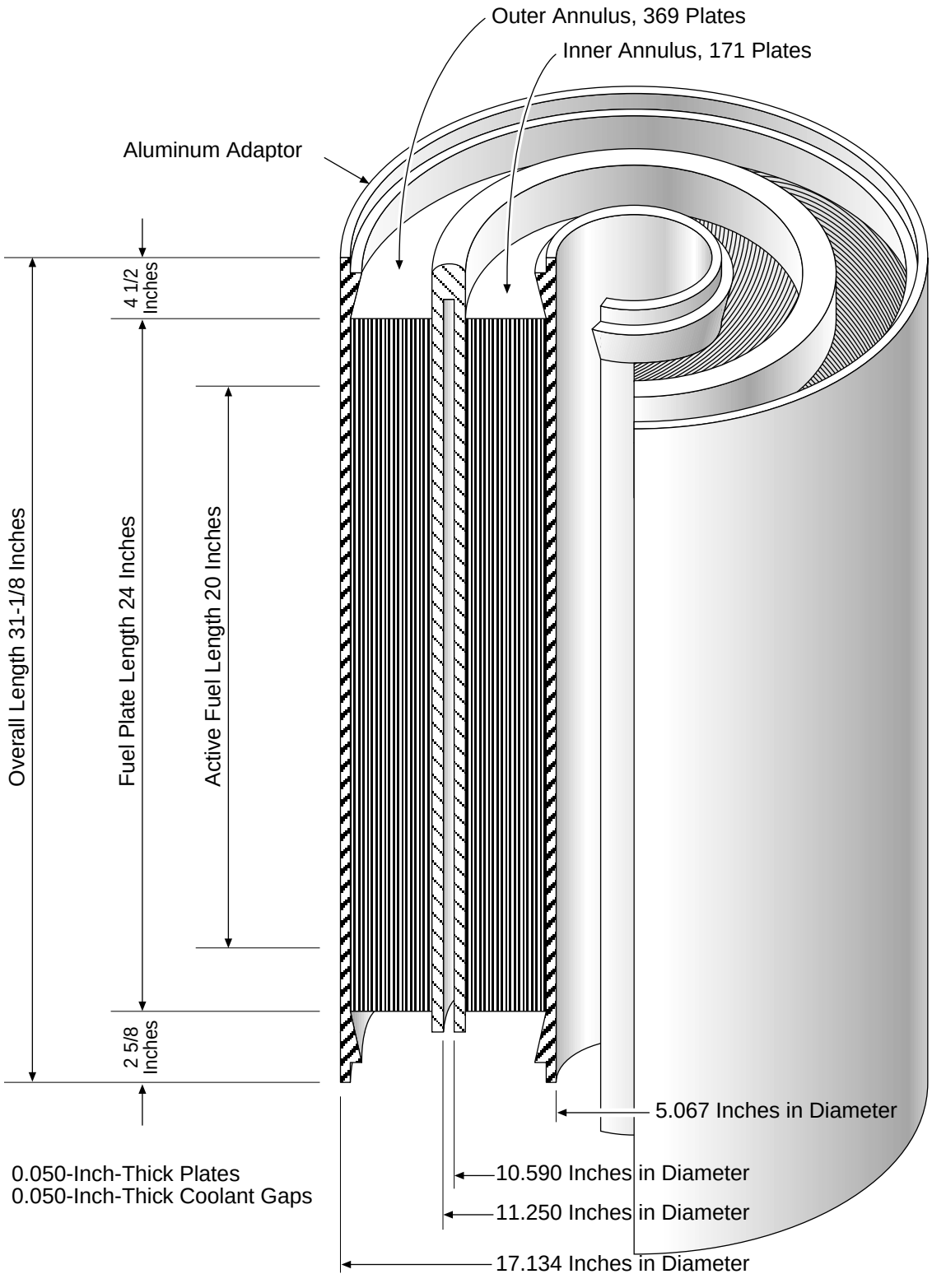


Source: ORNL 1998.

Figure B–6 Plan View (Cross Section) of HFIR

In the flux trap in the center of the HFIR fuel element, a thermal neutron flux of 2×10^{15} neutrons per square centimeter per second is available to irradiate target material. Target rods or experiments are loaded into the target holder assembly and positioned in the flux trap. There are 31 target positions in the flux trap. There are 6 peripheral target positions located at the outer edge of the flux trap, and these positions have the highest fast neutron fluxes. In addition, numerous vertical irradiation facilities of various sizes are located throughout the beryllium reflector. These are the circles of various diameters shown in the "Permanent Reflector" and "Removable Reflector" in Figure B–6. These facilities would be used for the irradiation of the neptunium-237 targets to produce the plutonium-238.

The control plates, in the form of two thin, europium-bearing concentric cylinders, are located in the region between the outer fuel element and the beryllium reflector. These plates are driven in opposite directions. Reactivity is increased by downward motion of the inner cylinder, which is used only for shimming (fine adjusting) and regulation; it has no fast safety function. Reactivity is increased as the outer plates are raised. Any single control element is capable of shutting the reactor down.



Note: 1. Not to scale.
2. To convert inches to centimeters, multiply by 2.54.
Source: LMER 1998.

Figure B–7 HFIR Fuel Element

There are a variety of experimental facilities at HFIR. These are described below.

Hydraulic Tube Facility. The HFIR hydraulic tube facility consists of the necessary piping, valving, and instrumentation to move aluminum capsules containing materials to be irradiated into and from the flux trap and the capsule loading station during reactor operation. The capsule loading station is located in one of the storage pools adjacent to the pool containing the reactor vessel.

Flux Trap Target Positions. Thirty target positions are provided in the flux trap. These positions are usually occupied by target rods used for the production of transuranium elements; however, experiments can, in principle, be irradiated in any of these positions.

Peripheral Target Positions. Six peripheral target positions are provided for experiments located at the outer radial edge of the flux trap. Fast-neutron fluxes in these positions are the highest accessible in the reactor, although a steep radial gradient in the thermal-neutron flux exists at this location.

Large Removable Beryllium Facilities. There are eight large removable beryllium facilities. In generic terms, these are referred to as the removable beryllium positions. They are located in the removable beryllium near the control region. Either instrumented or noninstrumented experiments can be irradiated in these facilities. Instrument leads and access tubes are accommodated through penetrations in the upper shroud flange and through special penetrations in the pressure vessel upper cover. These positions can accommodate spectral-tailored (i.e., shielded) experiments, making them well suited for fusion material irradiation. When not in use, these facilities contain beryllium or aluminum plugs.

To date, these particular facilities have been used primarily for three types of irradiations: (1) high-temperature gas-cooled reactor fuel irradiations, (2) production of radioisotopes, and (3) fusion materials irradiation program.

Small Removable Beryllium Facilities. Four small removable beryllium facilities are located in the removable beryllium near the control region. These unlined facilities have an inside diameter of 1.27 centimeters (0.5 inch). When not in use, these facilities contain beryllium plugs. These facilities have been used primarily for the production of radioisotopes.

Control Rod Access Plug Facilities. The semipermanent beryllium contains four control rod access plugs, the removal of which provides access to the coupling between the safety rods and their associated drive mechanisms. Each standard control rod access plug contains two 1.27-centimeter (0.5-inch) inside diameter unlined irradiation facilities, making a total of eight in the reactor. Normally, these facilities accommodate standard target rods of the type and configuration usually irradiated in the small removable beryllium facilities, although, in principle, experiments having other configurations can be accommodated. Only noninstrumented experiments can be irradiated in these facilities, because no provision is made for accommodating instrument leads and/or access tubes. When not in use, these facilities contain beryllium plugs.

Small Vertical Experiment Facilities. The permanent reflector is penetrated by 16 vertical holes, referred to as the small vertical experiment facilities, which extend completely through the beryllium. Each of these facilities has a permanent aluminum liner having an inside diameter of 4.01 centimeters (1.584 inches). Those located on the outer circle (five in all) are referred to as the outer small vertical experiment facilities. Vertical Experiment Facility–7 contains one of the pneumatic irradiation facilities and is unavailable for other use. Normally, noninstrumented experiments are irradiated in these facilities.

Large Vertical Experiment Facilities. The permanent reflector is penetrated by six vertical holes referred to as the large vertical experiment facilities. These facilities are similar (as to characteristics and capabilities) to the small vertical experiment facilities described in the preceding section except for location, size, and available neutron fluxes. The aluminum liners in the large vertical experiment facilities have an inside diameter of 7.2 centimeters (2.83 inches), and the facilities are located concentric with the core on a circle of radius 46.3 centimeters (18.23 inches). When not in use, these facilities contain beryllium or aluminum plugs.

Neutron Activation Analysis Laboratory and Pneumatic Tube Facilities. Two pneumatic tube facilities are available in HFIR. These facilities are designed and built to introduce sample containers (rabbits) into the flight tubes, and irradiation stations to which the rabbits move to be irradiated. The inner diameter of the flight tubes is 15.88 millimeters (0.62 inch), and the outer diameter of the rabbit is 14.48 millimeters (0.56 inch). Capsules are inserted into the reactor and returned to shielded loading stations in the laboratory. The facilities are used to measure the trace element content in a variety of materials by neutron activation analysis. About 65 of the chemical elements can be measured in the range of 10^{-6} to 10^{-15} gram (3.5×10^{-6} to 3.5×10^{-17} ounce).

The Neutron Activation Analysis systems support Oak Ridge National Laboratory (ORNL) (DOE) programs, are used in work-for-others projects, and are available for use by students and faculty of universities through Oak Ridge Associated Universities and other programs. Several students and faculty members have used the system.

From 1975 to 1985, approximately 100,000 samples were analyzed for uranium by the delayed neutron counter in the ORR system. Most of those samples were generated by the National Uranium Resources Evaluation and the remaining ones from the Formerly Utilized Sites Remedial Action Program.

Neutron Activation Analysis at ORNL was also used to analyze evidence related to the 1961 to 1962 French-connection heroin case and the 1962 assassination of President Kennedy. More recently, it has been used in environmental analysis and in determining levels of uranium in materials used in the semiconductor industry.

Slant Engineering Facilities. Provision has been made for installation of up to four engineering facilities to provide additional neutron beams for experiments. These facilities consist of tubes that are inclined upward from horizontal. The inner ends of the tubes terminate at the outer periphery of the beryllium and have lower flux than the main beam tubes. The upper ends of the tubes terminate at the outer face of the pool wall in an experiment room one floor above the main beam room.

Gamma Irradiation Facility. Since 1968, a large variety of materials have been irradiated in the HFIR pool with the gamma flux generated from the decay of fission products in the spent HFIR fuel assemblies. In 1972, two 7.6-centimeter (3-inch) diameter stainless steel tubes were fabricated to provide a conduit through which specimens and any associated equipment could be inserted and withdrawn from the gamma fields. Facilities are also available wherein the flux trap sections of depleted fuel elements can be used to irradiate experiments requiring a high gamma flux. To date, such experiments have included studies of the effects of gamma radiation on various salts, insulating materials, paint samples, and a variety of other materials.

Horizontal Beam Holes. The reactor has four horizontal beam (HB) tubes with inner diameters of 10 centimeters (3.94 inches) that extend outward from the reactor core at the midplane of the reactor. Beam tube HB–2 extends radially from the reactor centerline, and beam tube HB–3, which extends tangentially from the core, is offset 34 centimeters (13.4 inches) from the reactor center. A third tube is aligned on a tangential line 39 centimeters (15.4 inches) from the reactor centerline with both ends extending outward from the reactor to allow for the installation of two individual facilities. The tow ends of this tube are designated HB–1 and HB–4.

The average thermal flux at the end of the beam tubes for a power level of 85 megawatts is above 1×10^{15} neutrons per square centimeters per second.

Many neutron scattering facilities are found on the main experiment floor. All the neutron scattering spectrometers are located on this floor except that of the Center for Small Angle Scattering Research, which is located on the first floor above this area. The main spectrometer at each beam port uses a large monochromator shield that can be rotated under computer control to vary the orientation of the monochromator crystal.

The following paragraphs describe the instrumentation.

HB–1. The HB–1 spectrometer is generally operated as a polarized-beam spectrometer for elastic scattering studies, but it has all the capabilities of a three-axis spectrometer.

HB–1A. The HB–1A spectrometer is a recently installed triple-axis spectrometer constructed through a collaboration with Ames Laboratory at Iowa State University. The spectrometer has a fixed incident energy of 14.7 million electron volts so that high-order wavelength contamination (λn, n = 2, 3, ...) can be removed by a pyrolytic graphite filter. The monochromator is a double-crystal system with vertically focusing and flat pyrolytic graphite crystals. Analyzer crystals include pyrolytic graphite, germanium, or beryllium. The helium-3 detector is mounted vertically so that a large vertical divergence of the scatter beam may be used if desired.

HB–2. The HB–2 spectrometer is a very flexible three-axis spectrometer. The incident energy can be continuously varied by changing the monochromator angle. Four vertically focusing monochromator crystals are mounted on an apparatus that provides computer control of the focusing radius and of the selection of the type of crystal used in an experiment. Pyrolytic graphite, beryllium, copper, and silicon crystals are used as monochromators, depending on the desired incident energy and resolution. These same crystals are also used as analyzers. The analyzer angle is continuously variable, allowing experiments to be carried out either with fixed-incident energy or with fixed-scattered energy.

HB–3. The HB–3 spectrometer is a three-axis spectrometer that is nearly identical to the HB–2 spectrometer. The incident neutron energy is continuously variable by changing the monochromator angle, 2Θ(sub)*M*; and four vertically focusing monochromator crystals of pyrolytic graphite, beryllium, copper, and silicon are available. A sapphire filter for the HB–3 primary reactor beam is located in the shutter, and the collimator C1 is a separate unit.

HB–3A. The HB–3A spectrometer is a small-angle scattering spectrometer that uses perfect silicon crystals to obtain high angular resolution. The angular resolution in the horizontal plane is very high, but the vertical resolution is poor; thus, the spectrometer is most useful for studies of filamentary structures. This is the structure of a fluxoid lattice aligned in the vertical direction by a magnetic field, and a number of interesting studies have been performed on superconductors using this instrument.

HB–4. The HB–4 spectrometer is a very flexible time-of-flight spectrometer. The incident neutron energy is continuously variable by changing the angle for a silicon monochromator. The silicon monochromator is also used to pulse the neutron beam by being excited with high-power ultrasonic waves. The ultrasonic frequency is high, about 10 megahertz; therefore, the time resolution is very good. Because the beam is pulsed electronically, a pulse can occur at any time, and the cross-correlation technique can be used to give a high signal-to-noise ratio. Pseudorandom pulse codes are stored in an online computer and can be of various lengths and duty cycles. The neutrons scattered from a sample are timed over a 1.5-meter (4.92-foot) flight path and can be collected simultaneously in 70 helium-3 detectors that can be placed at any position over an

angular range between 10 and 130 degrees. The spectrometer can also be used as a polarized-beam time-of-flight spectrometer in which the beam polarization is varied according to a pseudorandom code. This procedure makes the spectrometer very useful for measuring magnetic excitations in ordered ferromagnetic systems because all phonon and elastic scattering is avoided by use of the polarized beam.

A modification of the HB–4 spectrometer, which will allow it to be operated as a high-resolution powder diffractometer, was constructed recently. The new design involved the addition of a bank of 32 detectors, each with a 1.83-meter (6-foot) Soller collimator, so that a complete diffraction pattern covering a scattering-angle range of 115 degrees can be obtained by step-scanning only 3.6 degrees. Changing from the time-of-flight mode of operation to the powder-diffractometer mode is under computer control.

HB–4A. The primary spectrometer located at HB–4A is a wide-angle neutron diffractometer that is operated under a United States–Japan Cooperative Program on Neutron Scattering Research. The wide-angle neutron diffractometer uses a curved linear position-sensitive helium-3 detector that subtends a 130-degree scattering angle with a resolution of about 0.6 degree. Between the sample and the detector is a collimator, with radial cadmium-plated steel blades placed at 5-degree intervals, which oscillates back and forth during measurements to reduce background. Data are taken only when the collimator moves so that the collimator shadow is uniformly distributed over the entire detector. A beryllium monochromator provides a beam having a wavelength of 1.537 angstroms (1.537×10^{-8} centimeter) from the (101) planes. The wide-angle neutron diffractometer was designed to provide two specialized data collection capabilities: (1) time-resolved measurements of powder diffraction patterns, and (2) measurements of diffuse scattering in single crystals using the flat-cone diffraction geometry.

30-Meter Small-Angle Neutron Scattering Spectrometer. The 30-meter small-angle scattering spectrometer, constructed with funds supplied by the National Science Foundation, is also located at HB–4. This spectrometer uses pinhole geometry with collimating slits of 0.5 to 3 centimeters (0.2 to 1.18 inches) in diameter separated by a distance of 10 meters (32.8 feet). The detector, with an active area of 64 by 64 centimeters (25.2 by 25.2 inches) and resolution element dimensions of 0.5 by 0.5 centimeter (0.2 by 0.2 inch), can be positioned at any distance from 1.5 to 19 meters (4.92 to 62.32 feet) from the specimen by moving a motor-driven detector carrier along rails in the evacuated flight path. The standard incident wavelength, provided by a bank of pyrolytic graphite crystals, is 4.75 angstroms (4.75×10^{-8} centimeter). This can be changed to 2.38 angstroms by substituting graphite for the cold beryllium filter normally in position. The changeover time is less than 5 minutes, and the procedure permits experiments to be performed with increased flux over a wider range of scattering angles.

The specimen chamber is designed to accommodate standard samples, or, if necessary, it can be fitted with specialized ancillary equipment (e.g., furnaces, automatic sample changers, a Displex unit, goniometers). The monochromatic beam leaving the upper crystal bank has the approximate dimensions of the projected area of the crystals, 35 by 40 millimeters (1.4 by 1.6 inches). Three horizontal beam guide sections, each 2 meters (6.56 feet) long and with the above cross-sectional area, serve to transport the effective source to distances of 5.5, 3.5, or 1.5 meters (18.0, 11.48, or 4.92 feet) from the sample with overall gain for the appropriate experiment of a factor of about 6. The instrument is interfaced to a personal computer with a 20-megahertz, 80386 microprocessor that acts as a user interface allowing menu-driven spectrometer operations; it also permits users to transfer data on disks directly to their own laboratories after completion of an experiment.

B.2.2 Process Description

The target irradiation operations using HFIR at ORR are illustrated in **Figure B–8**.

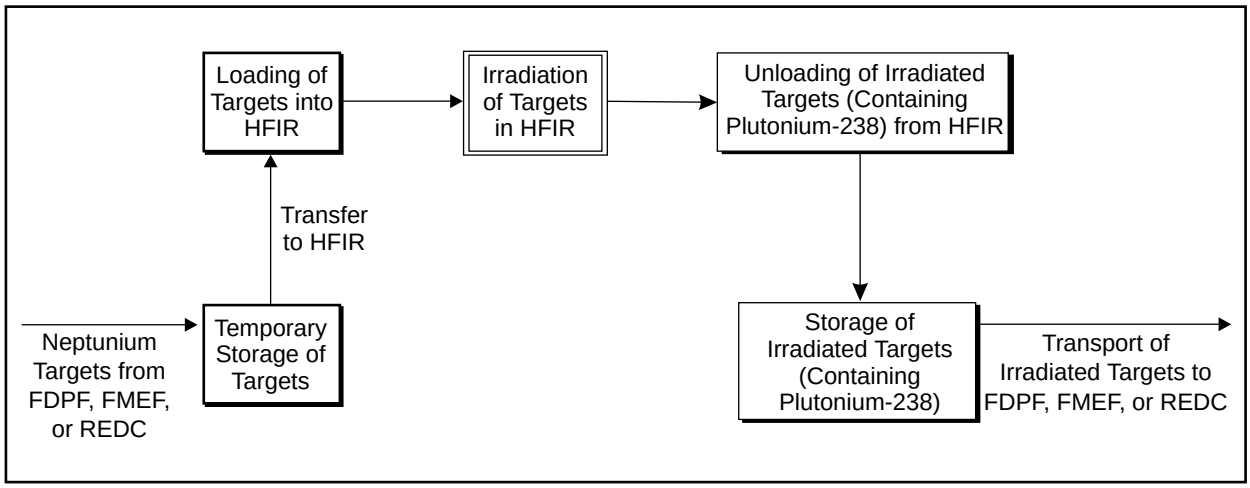


Figure B–8 Target Irradiation Operations Using HFIR at ORR

Following neptunium-237 target fabrication at FDPF, FMEF, or REDC (Appendix A), the targets would be transported to a shielded location in Building 7930 at ORR for temporary storage. Each target would contain, on average, approximately 750 grams (26 ounces) (approximately 0.5 curie) of neptunium-237 and up to an equivalent curie amount of protactinium-233, depending on the elapsed time following neptunium-237 purification (Wham 1999). The targets then would be transferred in transport casks to the HFIR spent fuel storage pool, where they would be stored for a period of approximately 28 to 30 days. Using a handling and underwater transfer system, the targets then would be transferred from the storage pool and inserted into the beryllium reflector (permanent or removable) area of HFIR. This loading would take approximately 2 to 4 hours to complete.

Targets would be irradiated concurrently in HFIR for a period of approximately 6 months to 2 years. The length of irradiation depends on the positions of the targets in the reactor.

Following irradiation, each target contains, on average, approximately 5 grams (0.18 ounce) of plutonium-238 along with smaller amounts of plutonium isotopes with higher atomic weights (i.e., approximately 0.65 gram [0.023 ounce] of plutonium-239 and approximately 0.06 gram [0.002 ounce] of plutonium-240) and also approximately 35 grams (1.2 ounces) of neptunium-237 (Wham 1999). The irradiated targets would be removed from HFIR using the same handling and underwater transfer system as used during loading and would be stored in the spent fuel storage pool. This unloading would take approximately 2 to 4 hours to complete. The storage of the irradiated targets would be for a period of 4 to 6 months to allow for the decay of short-lived radionuclides generated during irradiation.

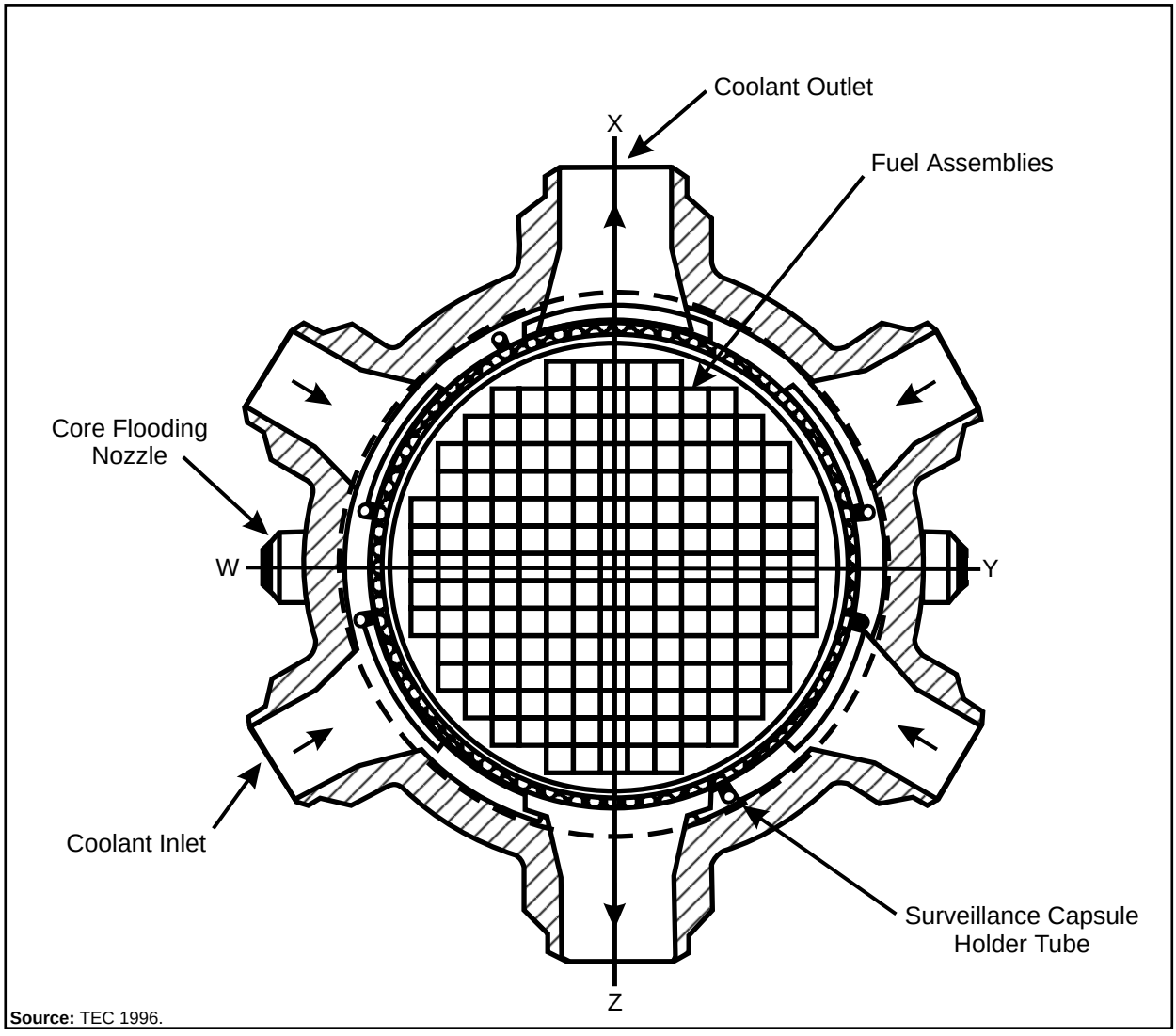
After storage, the irradiated targets would be transferred to FDPF, FMEF, or REDC for processing to separate the plutonium-238 product. A discussion of the postirradiation activities at these facilities is given in Appendix A.

B.3 COMMERCIAL LIGHT WATER REACTOR

B.3.1 Facility Description

The facility description is based on information provided in the *Final Environmental Impact Statement for the Production of Tritium in a Commercial Light Water Reactor* (DOE 1999b) and in the *Final Programmatic Environmental Impact Statement for Tritium Supply and Recycling* (DOE 1995).

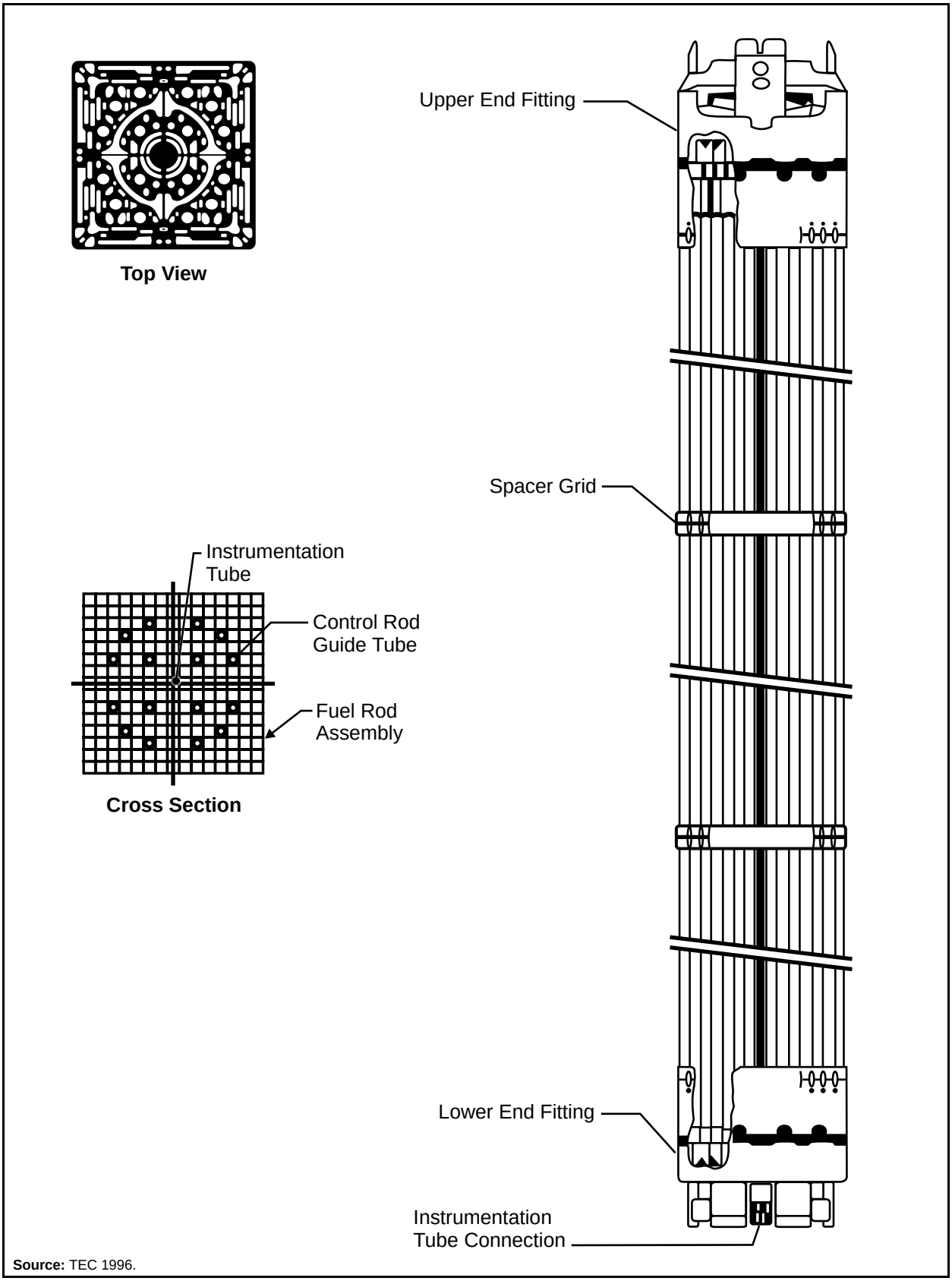
The generic pressurized water reactor core holds 177 fuel assemblies, as illustrated in **Figure B–9**, arranged in three regions that are rotated at about 18-month intervals (i.e., the inner region is removed as spent nuclear fuel to the spent fuel pool, the other two regions are moved inward, and fresh fuel is loaded into the outer region). The fuel assemblies each use a 15-by-15 lattice of fuel rods consisting of slightly enriched uranium dioxide pellets clad and sealed in Zircaloy tubing. All fuel assemblies (**Figure B–10**) are identical in mechanical construction and are interchangeable in any core location. The basic fuel assembly is normally composed of 208 fuel rods, 16 control rod guide tubes, and one centrally located position for instrumentation, all within the 15-by-15 lattice. The fuel assembly is approximately 20 centimeters by 20 centimeters (8 inches by 8 inches) in cross section, with an overall length of 420 centimeters (165 inches).



Source: TEC 1996.

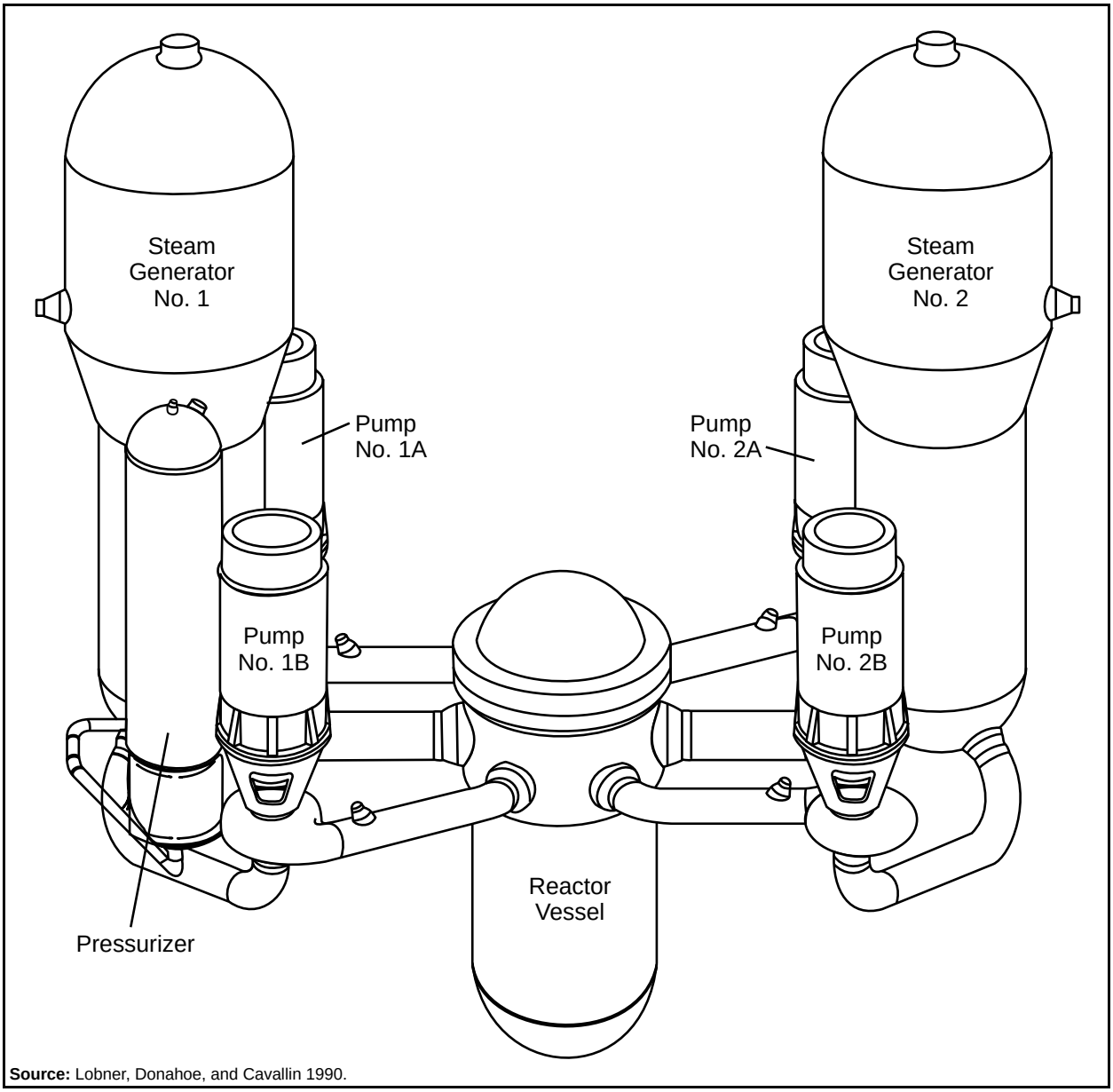
Figure B–9 Plan View (Cross Section) of a Generic CLWR

In addition to the nuclear reactor and its surrounding reactor vessel, a nuclear steam supply system contains equipment and components to remove the heat of fission and to convert it into steam to drive a turbine (in test or isotope production reactors such as ATR and HFIR, the heat arising from nuclear fission is merely passed from primary to secondary coolant systems for rejection to the atmosphere through cooling towers). A schematic drawing of a pressurized water reactor nuclear steam supply system is shown in **Figure B–11**. A pressurizer keeps the water reactor coolant under sufficient pressure to prevent it from boiling. The pressurized



Source: TEC 1996.

Figure B–10 CLWR Basic Fuel Assembly



Source: Lobner, Donahoe, and Cavallin 1990.

Figure B–11 Schematic of a Typical Pressurized Water Reactor's Nuclear Safety Steam Supply System

water (primary coolant) is pumped by the primary coolant pumps to the reactor, where it is heated by the fissioning of uranium fuel taking place within the core. The secondary coolant pumps send the heated coolant to the tube-side of the steam generators, where heat is transferred to the shell-side, boiling off secondary coolant as steam to drive turbine generators. The turbine generators are outside of the nuclear steam supply system and are major components of the "balance-of-plant" systems. Exhaust steam from the turbines is condensed to lower the back-pressure on the turbines and provide heat recovery.

The particular nuclear steam supply system illustrated in Figure B–11 is arranged as two heat transport loops, each with two primary coolant circulating pumps and one steam generator. In addition to serving as a heat transport medium, the primary coolant also serves as a neutron moderator and reflector, and as a solvent for soluble boron used in chemical "shim" control (fine adjustment of the "reactivity," or power level of the

reactor). A domed steel containment vessel envelops many nuclear steam supply system components in addition to the reactor vessel. These include the reactor coolant piping, the pressurizer, the pressurizer quench tank and coolers, reactor primary coolant pumps, steam generators, core flooding tanks, and letdown coolers. Safety systems directly associated with this vessel include the containment spray system, the containment air cooling system, and the containment isolation system.

A second level of containment exists as the reinforced concrete shield building, which surrounds the containment vessel. It is designed to provide biological shielding during normal operation and hypothetical accident conditions. The building provides for the collection and filtration of fission product leakage from the containment vessel following a hypothetical accident by means of its emergency ventilation system. In addition, the building provides environmental protection for the containment vessel from adverse atmospheric conditions and external missiles.

B.3.2 Process Description

The target irradiation operations using a generic CLWR are illustrated in **Figure B–12**.

The neptunium-237 targets can be placed in numerous locations within the reactor core (i.e., fuel assembly region) and outside the reactor core region to be irradiated for the production of plutonium-238. Three potential target arrangements were considered for evaluation in this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (NI PEIS): (1) all targets located in the center fuel assembly in the reactor core, (2) all targets distributed in select in-core locations, and (3) all targets distributed in select out-of-core locations. Locating all targets in the center fuel assembly in the reactor core was selected for evaluation in this NI PEIS because it was assumed that this would be the worst-case location during postulated beyond-design-basis accident conditions. The beyond-design-basis accident analysis postulated that the cladding of all targets in the center fuel assembly position would fail during the accident sequence. In the event that the NI PEIS Record of Decision selects the CLWR for the production of plutonium-238, the actual target arrangement in the reactor vessel will be defined during subsequent design and operational tradeoffs, which will consider reactor safety, impact on reactor core design and operation, plutonium-238 purity and generation rates, target handling, worker dose, and target design.

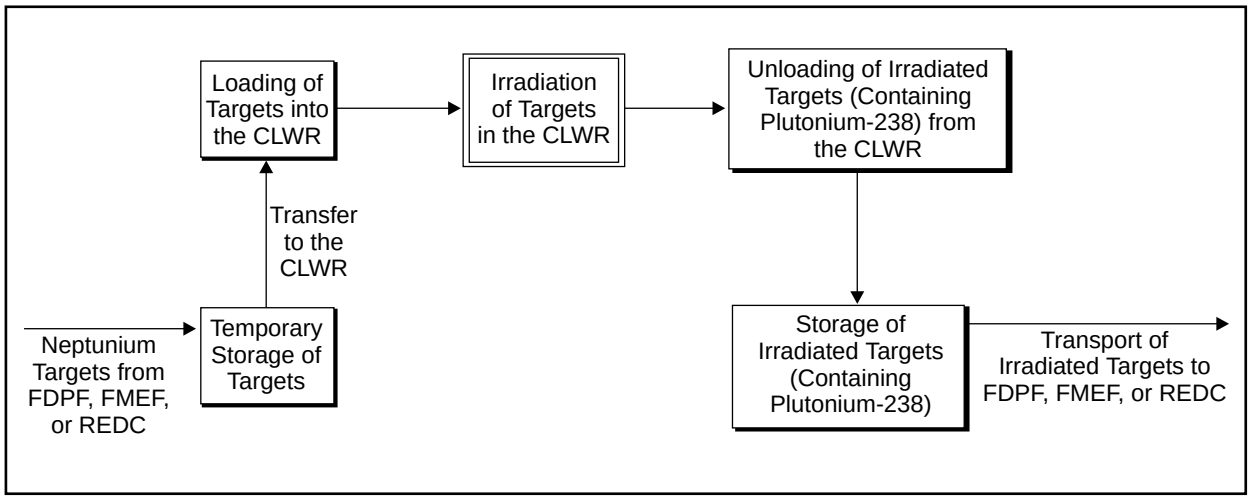


Figure B–12 Target Irradiation Operations Using a CLWR at a Generic Site

Following neptunium-237 target fabrication at FDPF, FMEF, or REDC (Appendix A), the targets would be transported to the spent fuel pool at the generic CLWR site for temporary storage. Because design of the neptunium-237 targets for the CLWR is still in the conceptual stage, the amount of neptunium-237 in each target has not yet been determined. During the period of reactor refueling, the neptunium-237 targets would be placed in a fuel assembly that was in the reactor core during the previous operating cycle. The targets would replace fuel rod positions in the fuel assembly. The removal of fuel rods from an irradiated fuel assembly and the substitution of neptunium-237 targets in the fuel rod positions would be performed in the spent fuel pool under approximately 6.1 meters (20 feet) of water to limit radiation doses to the involved workers. The Zircaloy-clad target dimensions are similar to a fuel rod, approximately 1 centimeter (0.4 inch) in diameter by 3.7 meters (12 feet) long. **Figure B–13** presents a cross-sectional view of the CLWR neptunium-237 target. The fuel assembly containing the neptunium-237 targets would be transferred from the spent fuel pool to the reactor using the refueling canal and placed in the center fuel assembly position in the reactor core (refer to Figure B–9). The neptunium-237 targets would remain in the reactor core for a complete operating cycle, nominally 18 months.

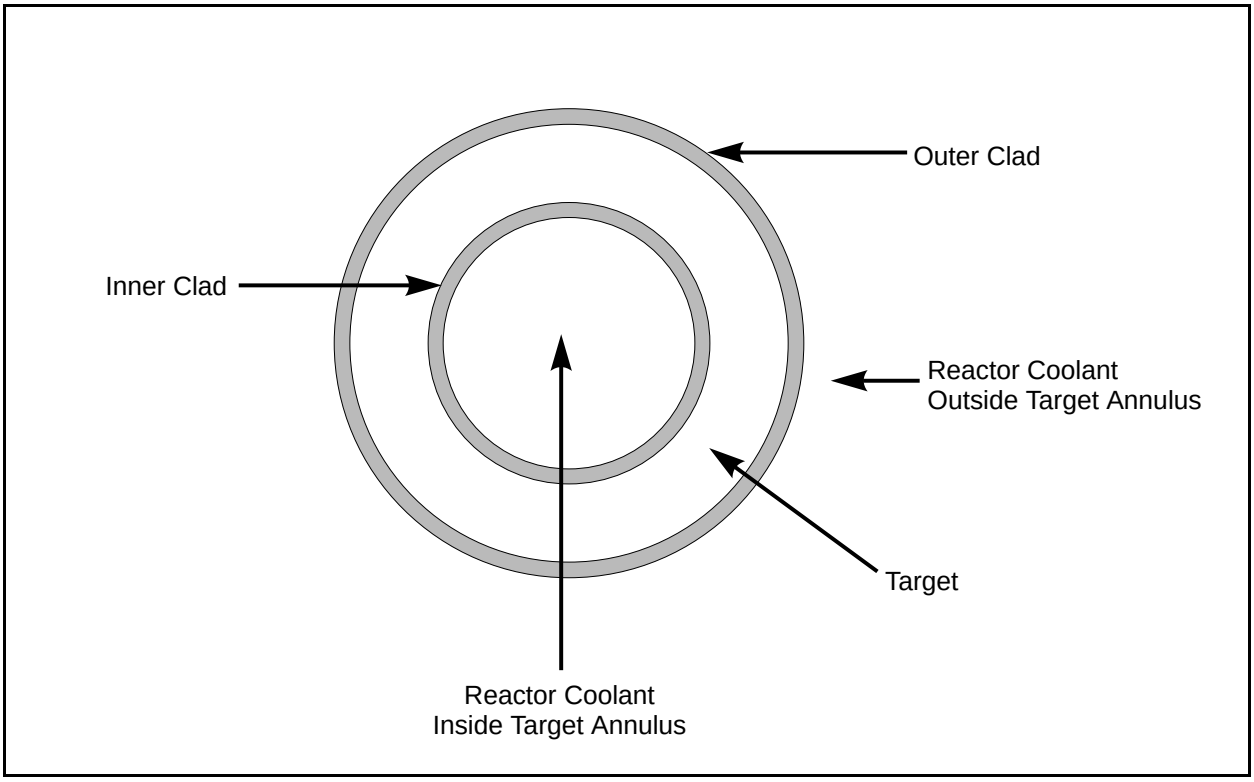


Figure B–13 Radial View (Cross Section) of a CLWR Neptunium-237 Target

The substitution of neptunium-237 target rods for fuel rod positions in the center fuel assembly would have a minimal impact on reactor operations. The fuel rods located in the center fuel assembly position do not normally contain fresh fuel (i.e., fuel placed in the core for utilization during the first 18-month operating cycle of the reactor), but contain fuel that is in the second or third operating cycle. The normal power distribution within the core and reactor coolant flow and its distribution within the core would remain within existing technical specification limits.

Following irradiation for a complete operating cycle, the targets would be removed from the reactor using the same transfer system used during loading and would be stored in the spent fuel pool. The irradiated targets

would be stored for a period of approximately 4 to 6 months to allow for the decay of short-lived radionuclides generated during irradiation.

After storage, the irradiated targets would be loaded in a spent fuel cask and transported from the reactor site to FDPF, FMEF, or REDC for processing to separate the plutonium-238 product and to recycle the remaining neptunium-237 for fabrication of new targets. A discussion of the postirradiation activities at these facilities is provided in Appendix A.

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Appendix C
Medical and Industrial Isotope Target Fabrication and Processing Operations and Civilian Nuclear Research and Development Targets

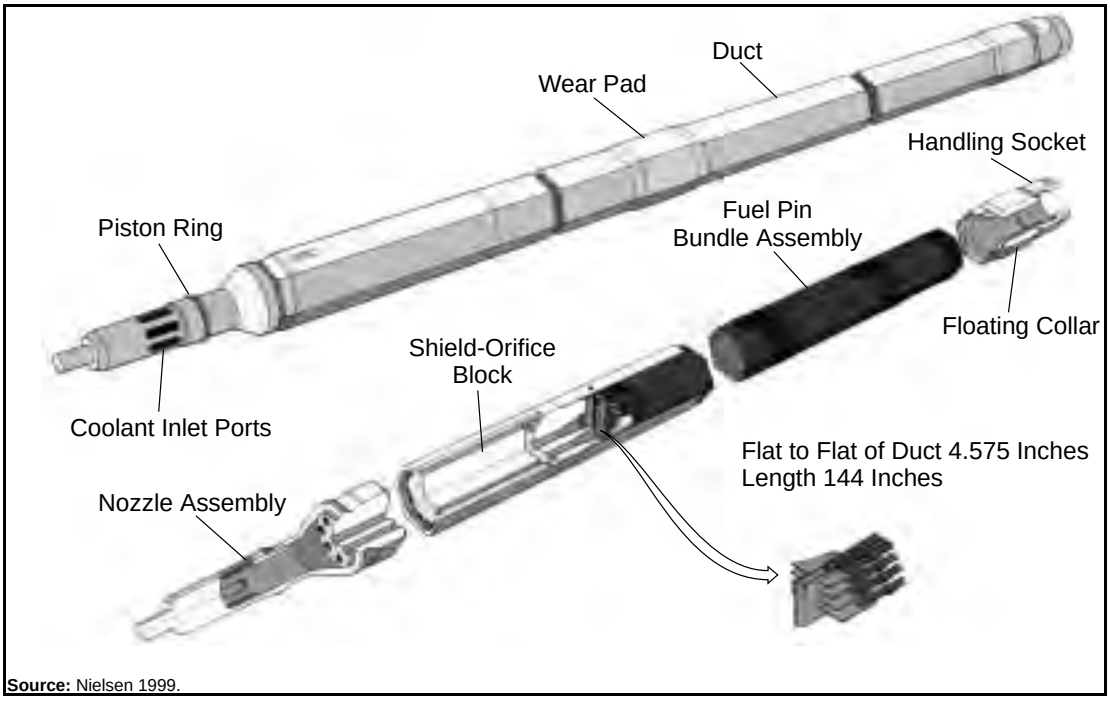
This appendix describes the technologies used to fabricate targets, process irradiated targets, recycle unconverted materials, and ship the product medical and industrial isotopes. It uses FFTF specific irradiation systems (vehicles) as examples for clarification and illustrative purposes.

C.1 TARGET FABRICATION AND PROCESSING FACILITIES

Target fabrication and processing facilities are described in Chapter 2.

C.2 DESCRIPTION OF THE TARGET FABRICATION PROCESS

The production of medical and industrial isotopes involves fabricating specially designed targets, irradiating the targets in the reactor core or the accelerator target caves to generate specific medical isotopes, and processing the targets to prepare medical isotopes for shipment to customers. At reactors, Long-Term Irradiation Vehicles would be used for irradiating materials to produce long-lived isotopes, and Rapid Radioisotope Retrieval Systems would be used for short-lived isotopes. The Long-Term Irradiation Vehicle would consist of a bundle of target pins installed inside a nozzle, duct, and handling socket assembly similar in appearance to a fuel assembly (**Figure C–1**). Depending on the isotopes to be produced, the pin bundle could contain moderator pins and neutron shield pins to provide the desired flux in specific core locations for isotope production. A design that would allow reuse of the Long-Term Irradiation Vehicle assembly nozzle, duct, and handling socket hardware would be considered during the design process in an effort to reduce costs and waste generation. **Figure C–2** is a picture of the cobalt-60 test vehicle prior to installation of the outer hexagonal duct. While cobalt-60 is not on the list of representative candidate medical isotopes, this assembly configuration is typical of the Long-Term Irradiation Vehicles irradiated in the Fast Flux Test Facility (FFTF).



Source: Nielsen 1999.

Figure C–1 An Example of an FFTF 12-Foot Fuel Assembly

The Rapid Radioisotope Retrieval System, which would be developed and used for the production of short-lived isotopes, would allow target materials to be inserted and withdrawn from the reactor core with the reactor operating at power. This system is described in Appendix D. Target material would be contained in individual capsules or interconnected carriers; allowing insertion and retrieval from the reactor as a string of targets. In addition to irradiating solid targets in the rapid retrieval system carrier strings, gas targets also could be irradiated to produce short-lived isotopes, as discussed in Appendix D. The system configuration presented for FFTF in Appendix D is typical of Rapid Radioisotope Retrieval System installations in other reactors engaged in medical and industrial isotope production.

Source: Nielsen 1999.

Figure C–2 A Cobalt-60 Test Assembly

The remainder of this section focuses on the first step in the isotope production process: the fabrication of the medical and industrial isotope targets.

Table C–1 presents a representative listing of 30 medical isotopes that are evaluated in this programmatic environmental impact statement. These isotopes were selected as being representative considering the current and future domestic and international demand for individual isotopes and current and future production capabilities for these products. The specific mix of isotopes produced by any irradiation facility will vary depending on the operating characteristics and authorized operating limits of the facility and the changing demands for specific isotopes during the production mission.

C.2.1 Target Materials

Each type of medical isotope would be produced using a target enriched in the appropriate isotope target material for neutron irradiation. With the exception of the radium-226 target and those using recycled materials, all of the targets use nonradioactive materials. Table C–1 presents the type and form of the target material to be used to produce each medical isotope. In cases where the target material and the product isotope are the same element, the target and the product cannot be chemically separated and the target material is shipped with the product. In cases where the target material and the isotope product are different elements, the target and isotope product can be chemically separated. After irradiation of the original target material (which in some cases may not be completely pure), there may be radioactive impurities that remain with the target material after the removal of the medical isotope product. Because of these impurities, the reuse of the material can, in some cases, create targets that are radioactive. As a result, the fabrication of targets using recycled target materials would require special handling. Shielding and special handling also would be required for the radium-226 target material (used to produce actinium-227, thorium-238, and thorium-239); it would be the only target material that would be radioactive before irradiation.

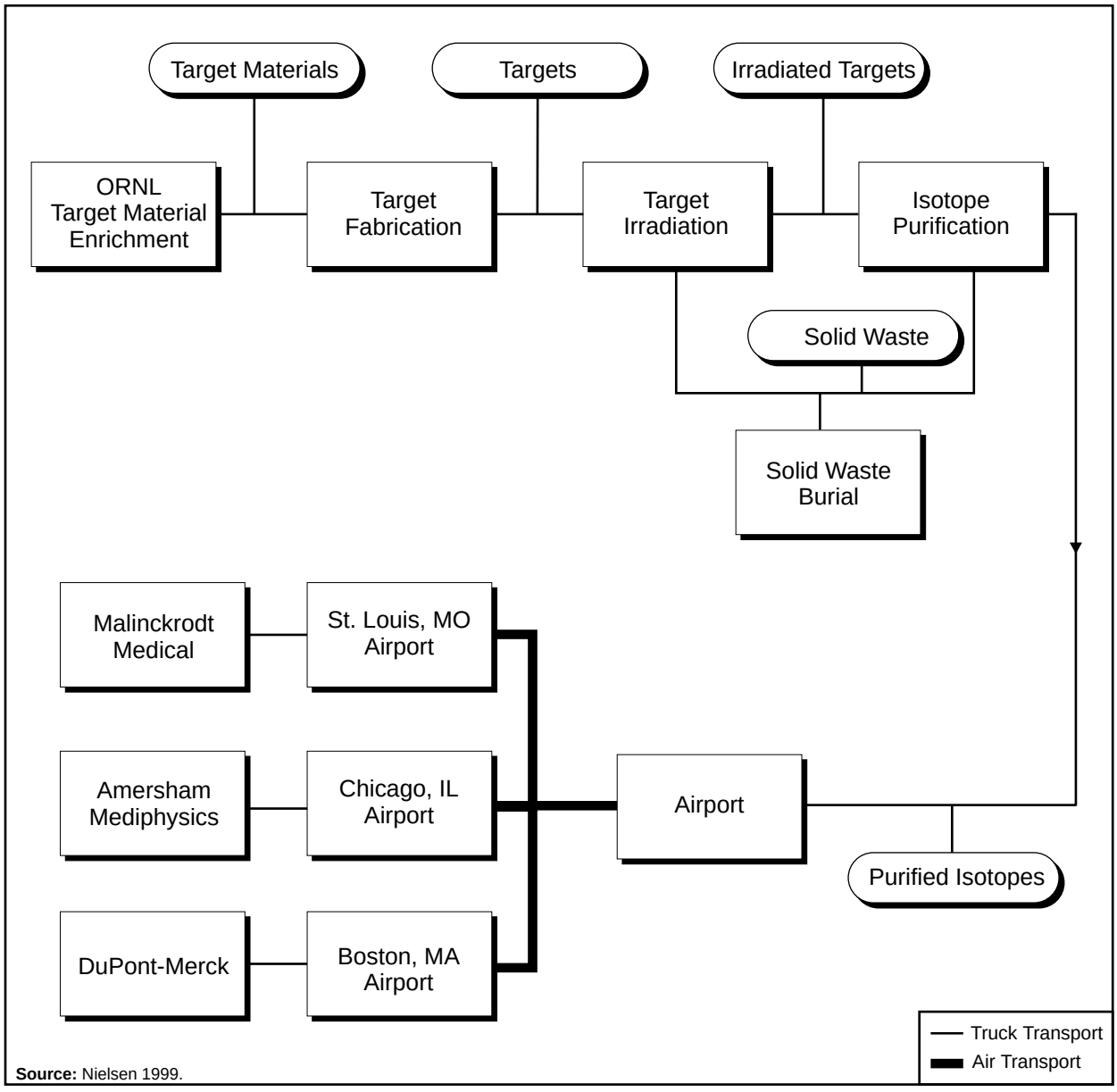
Table C–1 Representative Candidate Medical Isotopes

Product Isotope	Half-Life	Primary Target Isotope[a]	Target Vehicle	Irradiation Time (days)	Primary Target Isotope Mass (grams)	Irradiation Product Specific Activity (curies per gram)	Product Isotope Activity Produced (curies)	Product Isotope Activity Delivered (curies per assembly)
Actinium-227[b]	21.8 years	Radium-226[c]	LTIV-H	100	1.5×10^{1}	7.2×10^{1}	3.4×10^{1}	3.4×10^{1}
Gold-198	2.69 days	Gold-197[d]	R3-H	10	3.7×10^{-1}	2.1×10^{2}	1.7×10^{2}	7.9×10^{1}
Cadmium-109	462.0 days	Cadmium-108[d]	LTIV-H	100	6.6×10^{1}	9.9×10^{0}	6.6×10^{2}	6.5×10^{2}
Copper-64	12.7 hours	Zinc-64[d]	R3	10	2.4×10^{3}	3.8×10^{6}	4.8×10^{3}	9.6×10^{1}
Copper-67	2.58 days	Zinc-67[d]	R3	10	7.8×10^{1}	7.5×10^{5}	8.2×10^{0}	3.7×10^{0}
Gadolinium-153	242 days	Natural europium	LTIV-H	100	1.2×10^{3}	4.4×10^{1}	1.1×10^{3}	1.1×10^{3}
Holmium-166	1.12 days	Holmium-165	R3-H	10	4.3×10^{-1}	4.0×10^{1}	1.1×10^{2}	1.7×10^{1}
Iodine-125	60.1 days	Xenon-124	Gas line	100	1.6×10^{0}	1.7×10^{4}	2.6×10^{3}	2.4×10^{3}
Iodine-131	8.04 days	Tellurium-130[d]	R3-H	25	1.6×10^{2}	1.2×10^{5}	3.4×10^{2}	2.6×10^{2}
Iridium-192	73.8 days	Iridium-191	LTIV	100	5.7×10^{1}	6.1×10^{1}	3.7×10^{3}	3.5×10^{3}
Lutetium-177	6.68 days	Lutetium-176	R3-H	25	2.0×10^{-3}	2.0×10^{2}	5.5×10^{-1}	4.0×10^{-1}
Molybdenum-99	2.75 days	Molybdenum-98[d]	R3-H	10	2.7×10^{1}	3.7×10^{1}	2.1×10^{3}	1.0×10^{3}
Osmium-194	6.0 years	Osmium-192[d]	LTIV	100	1.6×10^{4}	1.4×10^{-4}	2.2×10^{0}	2.2×10^{0}
Phosphorus-32	14.3 days	Sulfur-32	R3	25	1.0×10^{1}	2.8×10^{5}	4.1×10^{1}	3.5×10^{1}
Phosphorus-33	25.3 days	Sulfur-33	LTIV	100	1.3×10^{0}	1.5×10^{5}	7.8×10^{1}	6.5×10^{1}
Palladium-103	17.0 days	Palladium-102[d]	R3-H	25	2.0×10^{1}	6.0×10^{1}	1.4×10^{3}	1.2×10^{3}
Platinum-195m	4.02 days	Platinum-195[d]	R3-H	25	2.2×10^{1}	5.5×10^{0}	2.0×10^{2}	1.2×10^{2}
Rhenium-186	3.78 days	Rhenium-185[d]	R3-H	25	6.4×10^{0}	4.7×10^{2}	5.2×10^{3}	3.0×10^{3}
Scandium-47	3.35 days	Titanium-47[d]	R3	10	4.3×10^{2}	8.2×10^{5}	3.3×10^{1}	1.8×10^{1}
Selenium-75	120 days	Selenium-74[d]	LTIV-H	100	3.2×10^{-2}	5.3×10^{2}	1.8×10^{1}	1.7×10^{1}
Samarium-145	340 days	Samarium-144[d]	LTIV-H	100	2.3×10^{0}	4.7×10^{0}	1.1×10^{1}	1.1×10^{1}
Samarium,-153	1.93 days	Samarium-152[d]	R3-H	10	3.4×10^{-2}	1.0×10^{3}	1.0×10^{2}	3.5×10^{1}
Tin-117m	13.6 days	Tin-116[d]	R3-H	25	1.4×10^{0}	3.1×10^{1}	5.1×10^{1}	4.4×10^{1}
Strontium-85	64.8 days	Strontium-84[d]	LTIV-H	100	3.5×10^{1}	5.7×10^{1}	2.2×10^{3}	2.0×10^{3}
Strontium-89	50.5 days	Strontium-88[d]	LTIV-H	100	1.6×10^{2}	8.8×10^{-1}	1.5×10^{2}	1.4×10^{2}
Thorium-228[e]	1.91 years	Radium-226[c]	LTIV-H	100	6.3×10^{1}	8.1×10^{2}	1.4×10^{2}	1.4×10^{2}
Thorium-229[f]	7,300 years	Radium-226[c]	LTIV-H	100	2.0×10^{3}	1.1×10^{-2}	2.7×10^{-2}	2.7×10^{-2}
Tungsten-188	69.4 days	Tungsten-186	LTIV-H	100	1.0×10^{4}	5.5×10^{-1}	5.9×10^{3}	5.5×10^{3}
Xenon-127	36.4 days	Xenon-126[d]	LTIV	100	2.2×10^{-1}	3.0×10^{2}	7.4×10^{1}	6.5×10^{1}
Yttrium-91	58.5 days	Zirconium-91[d]	LTIV	100	5.3×10^{2}	2.5×10^{4}	1.8×10^{1}	1.7×10^{1}

a. One hundred percent enriched.
b. Actinium-227 will decay to radium-223, which is the isotope that has medical applications.
c. Target form would be either basic element, carbonate, or chloride.
d. Target form would be either the basic metallic element or a metallic oxide or other compound, dependent on availability and on engineering considerations such as material melting point, degradation characteristics, and processing methods.
e. Thorium-228 will decay to radium-224, which is the isotope that has medical applications.
f. Thorium-229 will decay to bismuth-213, which is the isotope that has medical applications.

Key: LTIV, Long-Term Irradiation Vehicle; LTIV-H, Long-Term Irradiation Vehicle–hydrided; R3, Rapid Radioisotope Retrieval System; R3-H, Rapid Radioisotope Retrieval System–hydrided.
Source: Nielsen 1999.

For the nonradioactive targets, the target material typically would be acquired from Oak Ridge National Laboratory (ORNL), where enrichment processes are conducted to produce target material (isotopes) of sufficient purity to support the generation of medical isotopes. The target form may be a metal, metallic oxide, or other chemical compound suitable for irradiation, depending on engineering considerations such as material heat transfer characteristics, melting points, and metallurgical properties. The nonradioactive target material would be transported by truck from ORNL to the Hanford Site (Hanford) or to the U.S. Department of Energy (DOE) site having the new DOE low-energy accelerator or new research reactor and support facilities. The transportation mode, the shipment origin and destination, and transportation requirements for medical isotopes to be produced at Hanford are illustrated in **Figure C–3**. With the exception of the transportation of radium-226 to the medical isotope processing facility for target fabrication, the same transportation scenario would be applicable to each isotope.



Source: Nielsen 1999.

Figure C–3 Isotope Transportation

The radioactive target material, radium-226, would not be supplied by ORNL. However, radium-226 could be supplied by a variety of sources. Once materials for the targets arrive on the site, the materials would be stored in the medical isotope processing facility until needed for fabrication into medical isotope targets. Nonradioactive target materials do not have special storage requirements and most likely would be stored at the fabrication site. Special storage conditions are required because radium-226 generates radon gas. Recycled target materials would be stored in the processing facility.

For most types of targets, a cladding or encapsulation material would be needed to contain the target material during handling and irradiation. The Long-Term Irradiation Vehicle would be submerged in the reactor coolant, and it is anticipated that the target pins would be similar in construction and cladding to fuel pins. In this case, the cladding material must be fully compatible with the reactor coolant and should have low neutron-induced swelling characteristics.

For the Rapid Radioisotope Retrieval System, the targets would be inserted into and be irradiated in one or more fixed thimbles that extend into the reactor core. The individual targets, each composed of different target material, would be encapsulated or clad as necessary to (1) facilitate target handling, (2) contain the target materials and products during irradiation, (3) avoid interactions between the various target and structural materials at the irradiation temperature, and (4) maintain target purity. Dependent on the method finally selected for target insertion and retrieval (pneumatic or mechanical and cable system), the individual targets may be contained in cylindrical target carriers. These carriers would be interconnected to form a target string. To reduce the volume and weight of shielding required on the retrieval and irradiated target handling equipment, consideration would be given to using low-activation material for the target carriers. Studies also would be done to define the impacts on irradiation conversion efficiency, worker dose, processing efficiency, and waste streams of using higher-activation material for the target carriers or cladding (Nielsen 1999).

C.2.2 Target Fabrication Operations

Solid targets for either the Long-Term Irradiation Vehicles or the Rapid Radioisotope Retrieval System would be fabricated in gloveboxes using a series of mechanical and thermal processes. For the solid targets based on a powder, it is unknown at this time if the powder would be loose or if the powder would be pressed and sintered into pellets in some cases, depending on the material. If it is preferable to have pressed and sintered pellets, this option would require separate dies and boats to press and sinter each type of solid target material to reduce the risk of cross-contamination from other target materials. For solid metallic target material, depending on the purchased form, it may only be necessary to machine or cut the material to the required size.

If pellets are used, the first major step in their preparation would be powder conditioning and pressing. This would include weighing, blending, and pressing the powder and binder into slugs. The slugs would be granulated, blended with binder addition, and pressed into pellets. The pellets would be transferred to the sintering and debind station, weighed, and subjected to a series of thermal processes to debind and sinter the pellets. The sintered pellets would be subject to characterization to ensure that specifications were met.

Acceptable pellets would be transferred to the loading and welding station to be visually inspected before inclusion into a capsule or pin. For both powder or pellet target materials, capsules and pins would be cleaned before final closure. The capsules would be leak-tested and inspected before being cleared for use (Nielsen 1999). Normally, the required characteristics and physical configuration of a target for production of a specific isotope define if the powder should be loose in the target or pressed and sintered into pellets. In the event that there is a choice between powder or pellet, all other irradiation considerations and irradiated target processing considerations being equal, the loose powder target would normally be preferred. The target fabrication process for loose powder targets is much simpler, more cost effective, and generates less waste and

emissions due to the fact that there is less material processing. In addition, less equipment would require general housekeeping and cleanup, test, calibration, and maintenance.

C.2.3 Nuclear Research and Development Targets

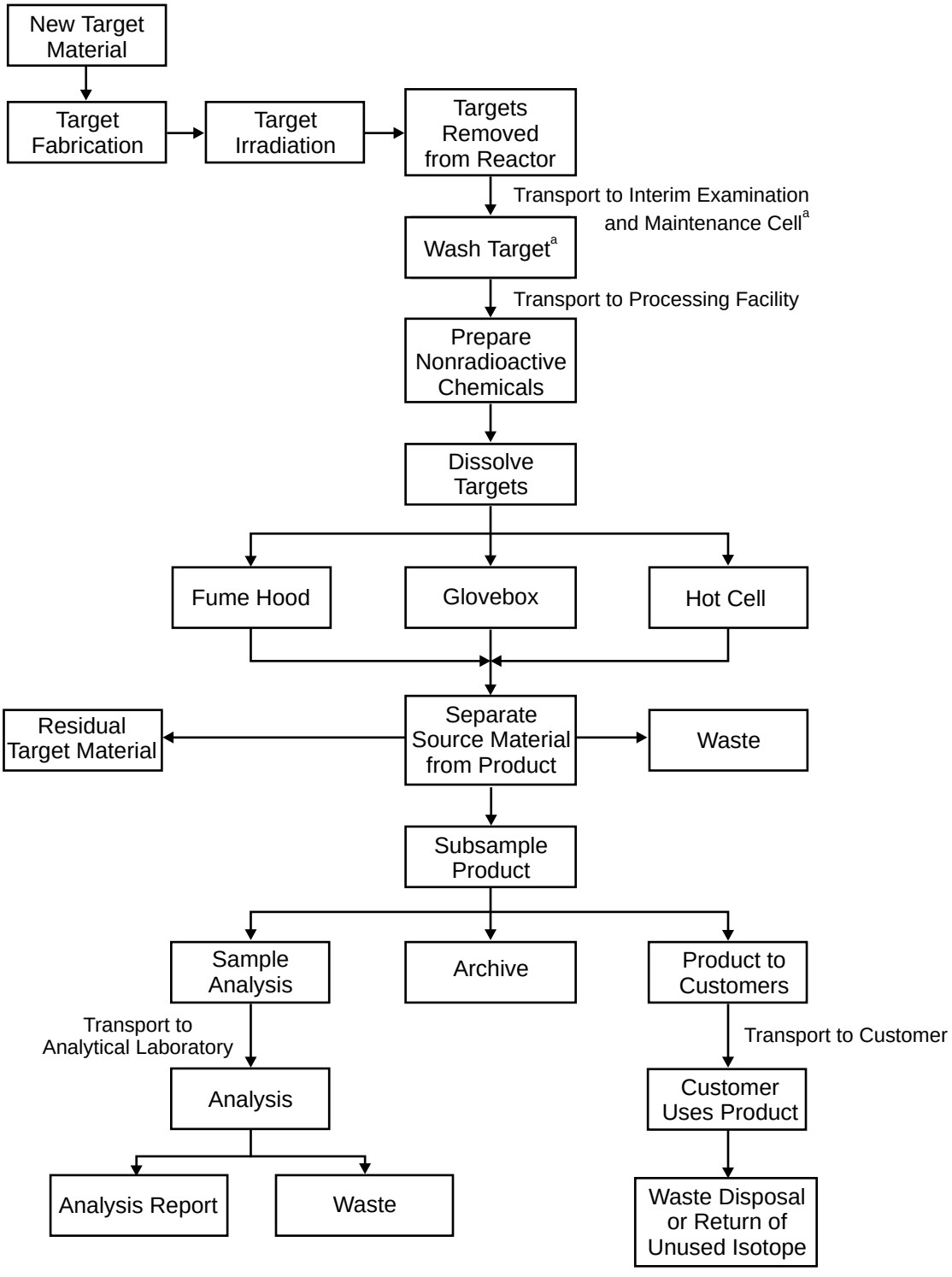
There is particular interest in materials testing associated with commercial nuclear power plant license renewals, cooperative international fusion energy research, space power technology, and transmutation of waste as a means to destroy long-lived isotopes from commercial spent nuclear fuel. Another area of interest is developing nuclear technologies that advance global nonproliferation. Target assemblies to be irradiated in support of these mission areas could be fuel or other materials configured similar to a standard driver fuel assembly (i.e., target material encased in sealed pins, and the pins placed in a ducted assembly). In the FFTF reactor, material specimens also could be installed in an existing open test assembly position within a Material Open Test Assembly. Test target assemblies or materials (e.g., for a Material Open Test Assembly) could be fabricated on site or at the customer's facility and then transported to an accelerator or reactor for irradiation.

One of these proposed testing activities consists of fuel testing for the Accelerator Transmutation of Waste (ATW) program. A fuel development activity for this program could use a reactor for irradiation testing. The tests could be fabricated by the ATW program and transported to the reactor site for irradiation. Specific test compositions and irradiation parameters for these test fuel assemblies have not been defined. It is anticipated that initial tests would involve a few pins in an assembly, while later tests could involve entire assemblies. The target pins are described as containing a matrix of zirconium and transuranic waste with a composition of 25 percent transuranic waste and 75 percent zirconium by weight. The transuranic waste likely would be light water reactor discharge fuel at a typical burnup of 33,000 megawatt days per metric ton of uranium that is stripped of essentially all uranium and fission products. Another potential testing activity could be performed at FFTF for the ATW program is irradiation testing using lead-bismuth alloy as a coolant. Lead-bismuth alloy has been proposed as the target material for production of spallation neutrons in the ATW accelerator and as the coolant for the transuranic target assemblies undergoing irradiation at the ATW facility. FFTF has the capabilities for high-temperature irradiation closed-loop testing involving liquid metal coolant isolated from the reactor coolant system and test positions in the reactor that could be adapted for this test activity (Nielsen 1999).

C.3 DESCRIPTION OF THE POSTIRRADIATION TARGET PROCESSING

Processing of irradiated targets to recover medical- and industrial-grade isotopes can be broken down into distinct steps: receipt of irradiated targets into a chemical separation facility; chemical processing of the targets (using hot cells, shielded gloveboxes, and appropriate open-faced hoods); waste handling; analysis of the products; recycling of some of the target materials; and shipment of the isotope products to customers. A representation of this process for preparation of isotopes is shown in **Figure C–4**.

The 30 representative medical isotope products shown in Table C–1 may be grouped in three categories. Thirteen of the targets would produce an isotope of the same element and would not require separation. Five of the targets would produce different isotopes of the same element but would require some processing to remove impurities. Twelve target materials would produce different elements and would require chemical separation, both for separation of the target material and unwanted impurities. These categories are discussed in more detail in Sections C.3.1, C.3.2, and C.3.3, respectively (BWHC 1999).



Source: BWHC 1999.

[a] Long-term irradiation vehicle only.

Figure C–4 Production and Radiochemical Processing Steps in the Preparation of Medical and Industrial Isotope Products

C.3.1 Same Target/Products Requiring No Separation

Thirteen of the same target element/products would not require extensive chemical separation (**Table C–2**). However, with experience, target purity and customer requirements might change, requiring additional separation and purification.

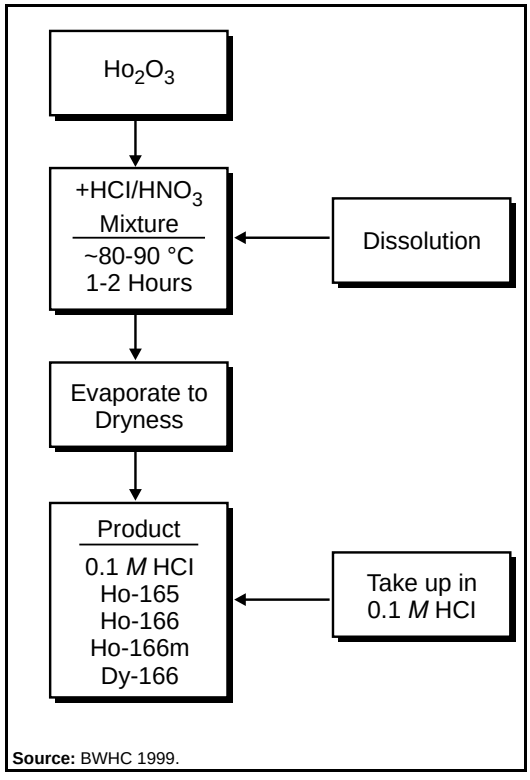
Table C–2 Products Requiring No Separation

Product	Curies per Target Campaign	Irradiation		Campaigns per Year
		Material	Time (days)	
Molybdenum-99	21,000	Molybdenum-98	10	25
Palladium-103	14,000	Palladium-102	25	10
Platinum-195m	200	Platinum-195	25	25
Rhenium-186	5,200	Rhenium-185	25	25
Selenium-75	18	Selenium-74	100	3
Samarium-153	100	Samarium-152	10	25
Tin-117m	51	Tin-116	25	10
Tungsten-188	59,000	Tungsten-186	100	3
Holmium-166	110	Holmium-165	10	25
Iridium-192	37,000	Iridium-193	100	3
Gold-198	170	Gold-197	10	25
Samarium-145	110	Samarium-144	100	3
Strontium-85	2,200	Strontium-84	100	3

Source: BWHC 1999.

A typical example of a product that would require no postirradiation separation is holmium-165/holmium-166 (**Figure C–5**). Medical isotope target carriers would be unloaded from the cask into A-Cell, where they would be separated and prepared for transport from A-Cell to isotope processing stations. The target material would be removed from the carrier and dissolved using nitric acid, hydrochloric acid, or a combination of acids. The dissolved material would be evaporated to near dryness to remove the acid. The resulting salt would be redissolved in dilute acid. The product solution would be analyzed for chemical and radionuclide purity, and aliquots of the analyzed product would be placed in appropriate containers and shipped to customers. Analytical techniques that might be used include (1) inductively coupled plasma/atomic emission spectroscopy, (2) gamma energy analysis, (3) alpha energy analysis, and (4) counting equipment.

All liquid waste would be neutralized and captured on a solid absorbent as solid waste. Solid compacted waste for each product (excluding cladding hardware) would range from 0.28 to 0.14 cubic meters (1 to 5 cubic feet) per year per product (BWHC 1999).



Source: BWHC 1999.

Figure C–5 Processing of Target Material Where the Isotope Product (Holmium-166) is Chemically Identical to the Target Isotope (Holmium-165)

C.3.2 Same Target/Product Requiring Separation

Five target materials producing other isotopes of the same element would require further processing: cadmium-109, lutetium-177, osmium-194, strontium-89, and xenon-127. A typical example of a product requiring separation would be cadmium-109 from cadmium-108. The cadmium-109 produced from enriched cadmium-108 would contain small amounts of cobalt-60, zinc-65, and silver-110m, along with cadmium-109 and cadmium-115m. Medical isotope target carriers would be unloaded from the cask into a hot cell, where they would be prepared for transport from the hot cell to the appropriate isotope processing stations. The cadmium oxide would be removed from the target carrier, dissolved in hydrochloric acid, and diluted with water to 0.1 molar (*M*) hydrochloric acid. The impurities and cadmium would be adsorbed on a Dowex 50-X8 cation exchange column. The cadmium then would be selectively eluted with 0.2 *M* hydrochloric acid. The eluate would be analyzed using inductively coupled plasma/atomic emission spectroscopy and aliquots of the product solution placed in appropriate containers and shipped to the customers.

All liquid waste would be neutralized and captured on a solid absorbent as solid waste. Solid compacted waste (excluding cladding hardware for each same target/product would be about 0.28 to 0.14 cubic meters (1 to 5 cubic feet) per year per product.

C.3.3 Processing Target Materials Containing New Elements

Twelve target materials would produce different product elements and each would require unique chemical separation. The processing steps would include separation and purification of the product isotope, retrieval of the original target material for possible reuse, and disposal of the residual waste. Four examples of process separation are discussed to show the diversity of the separation methods.

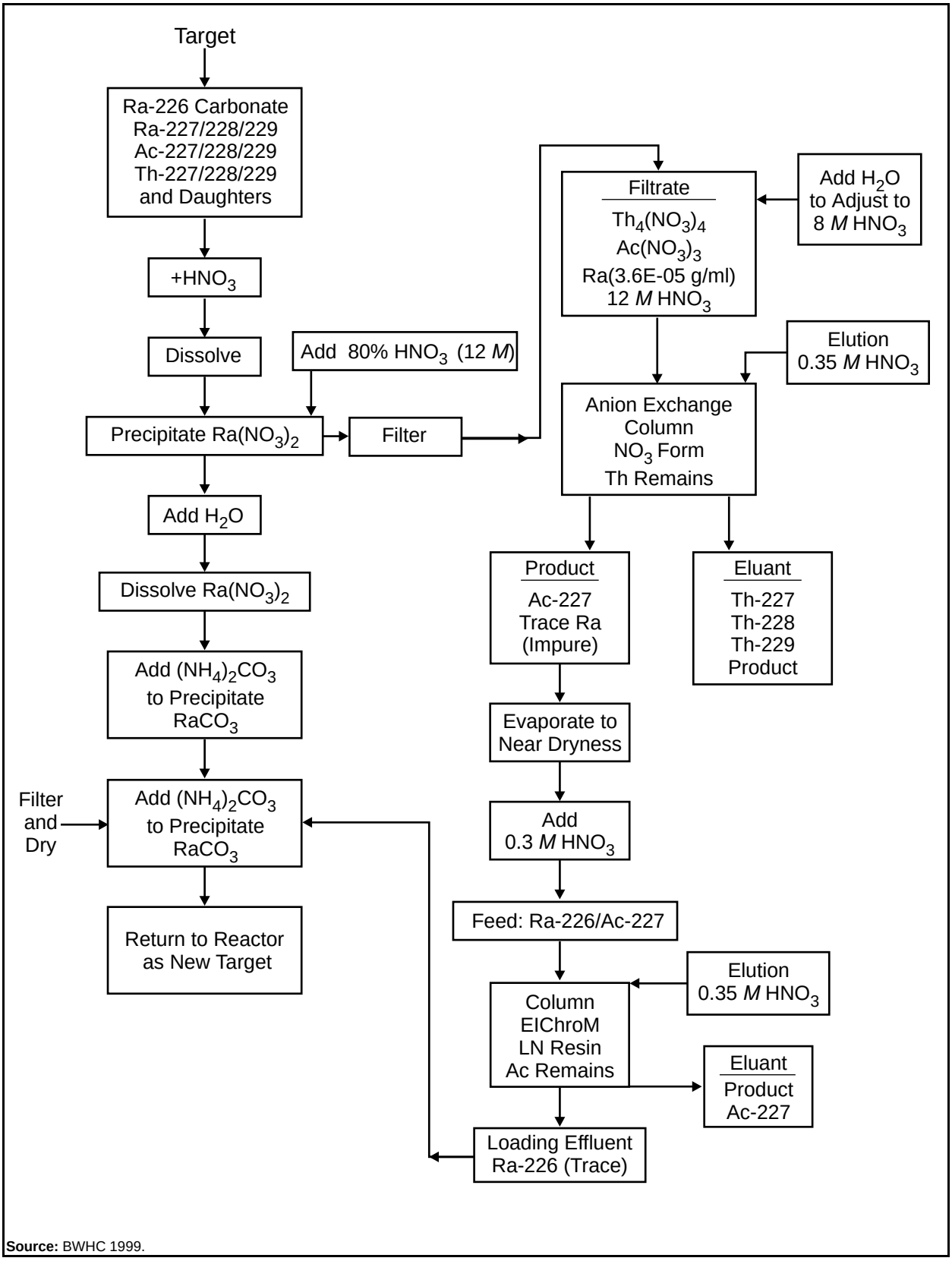
ACTINIUM-227 AND THORIUM-228/229 FROM RADIUM-226

Radium-226 would be the only target that would be radioactive before being irradiated. It would be used to produce three products: actinium-227, thorium-228, and thorium-229. The processing room would contain a high-level radiochemical cell, a leaded glovebox, and an open-face hood, all connected to a radon capture facility. The radon capture facility (temperature-controlled charcoal filter) is required to prevent release of radon gas from the facility while storing the original target material and processing the irradiated targets.

The capsule containing the irradiated target (radium carbonate) material would be transported to the hot cell in a shielded container and opened for removal of the target material by cutting the cladding. The carbonate salt would be dissolved with acid (**Figure C–6**). The radium then would be separated by nitrate precipitation and filtration. The remaining thorium, actinium, and radium in the filtrate solution would be purified using ion exchange separation.

Detailed steps in processing the radium-226 would include removal of the radium carbonate solid from the metal capsule for dissolution in dilute nitric acid. An addition of 80 percent nitric acid to the mixture then would be performed to precipitate the radium as radium nitrate, followed by filtration. The precipitated radium would be retrieved by the addition of water to redissolve the radium nitrate precipitate. Ammonium carbonate then would be added to the dissolved radium nitrate solution to precipitate radium carbonate. The carbonate precipitate would be filtered, dried, and could be reused as target material.

The 12 *M* nitric acid filtrate from the radium nitrate precipitation, containing thorium, actinium, and traces of radium, would be adjusted with water to 8 *M* acid and loaded on to an anion exchange column. The thorium



Source: BWHC 1999.

Figure C–6 Processing Procedures for Removal of Actinium-227, Thorium-228, and Thorium-229 Products from Irradiated Radium-226 Targets

would be held on the column, allowing the actinium, radium, and other impurities to be captured in the effluent. The purified thorium would be eluted from the anion exchange column with 0.35 *M* nitric acid. The thorium-228/229 product solution would be analyzed to ensure purity, and aliquots of the analyzed product would be placed in appropriate containers and shipped to customers.

The actinium-227 and traces of radium, found in the effluent solution from the anion exchange purification of thorium, would be evaporated to dryness and redissolved in 0.03 *M* nitric acid. This solution would be loaded onto an EiChroM resin column and the effluent containing the radium retrieved via carbonate precipitation. The resin column containing purified actinium-227 would be eluted with 0.35 *M* nitric acid. The actinium-227 product solution would be analyzed to ensure purity, and aliquots of the analyzed product would be placed in appropriate containers and shipped. The actinium-227 also can be used on site for recovery and shipment of radium-223.

All liquid waste would be neutralized and captured on a solid absorbent. Solid compacted waste (excluding cladding hardware and radon holdup charcoal) is estimated at 0.28 cubic meters (10 cubic feet) per year (BWHC 1999).

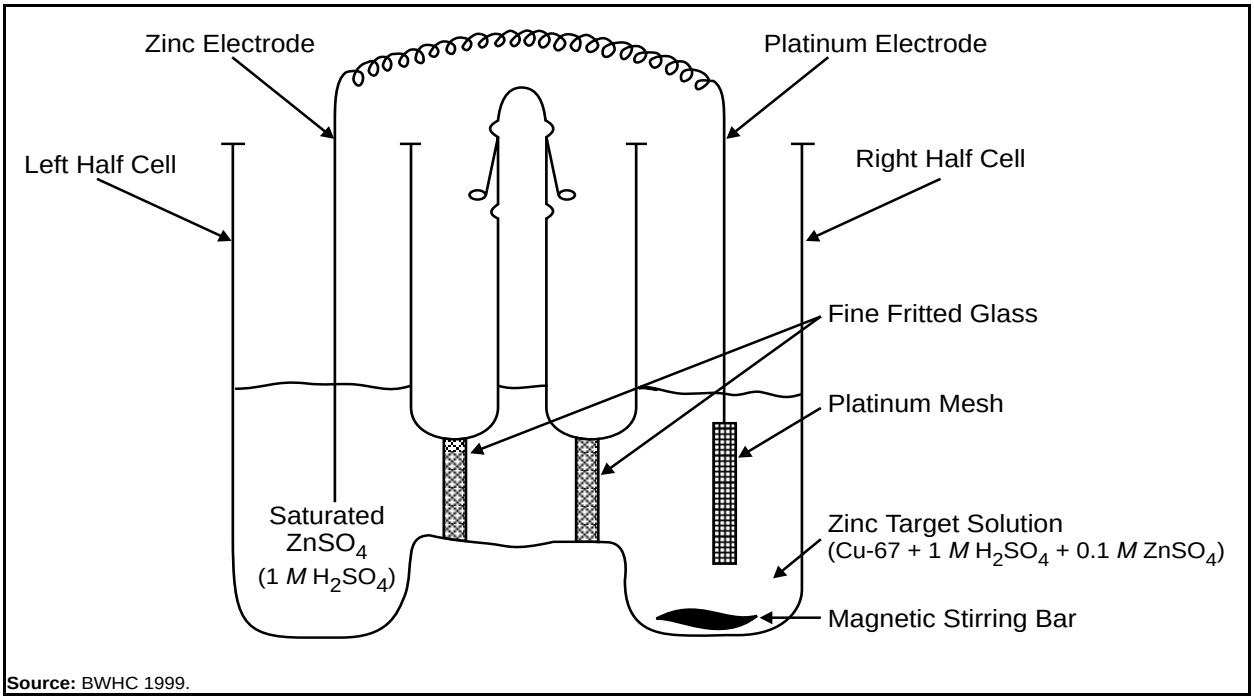
COPPER-64 AND COPPER-67 FROM ZINC-64 AND ZINC-67

The capsule containing the irradiated zinc oxide target would be transported to the receiving hot cell. Irradiated targets and carriers would be transported in shielded containers from the hot cell to an isotope processing station containing a hot cell, a lead-shielded glovebox, a fume hood, and a laminar flow hood. The target material would be removed from the carrier by cutting the capsule containing the zinc oxide. The oxide would be dissolved with sulfuric acid. The sulfuric acid solution would be placed in an electrochemical cell and the copper deposited on a platinum electrode (**Figure C–7**). The platinum electrode would be removed from the electrochemical cell and copper-64 or copper-67 dissolved from the surface by immersing the platinum electrode in nitric acid. The zinc could be retrieved by converting it to the oxide form to be used as target material.

Detailed steps in processing the copper isotopes would include removal of the zinc oxide from the metal capsule for dissolution in 1 *M* sulfuric acid and transfer to the electrochemical cell. After 30 minutes, the copper would have been completely deposited on the platinum electrode. The target solution still containing the zinc would be removed and replaced with fresh acid and the copper deposition would be continued for an additional 30 minutes. This step would be repeated for a second time to assure purification. The deposited copper on the platinum electrode then would be removed from the cell and dissolved by immersing the electrode in concentrated nitric acid for 1 to 2 minutes. This solution would be evaporated to near dryness to remove the strong acid. The dried product then would be redissolved in an appropriate acid. The solution would be analyzed for chemical and radionuclide purity, and aliquots of the final product placed in containers and shipped to customers.

The zinc ion contained in the spent electrochemical solution could be retrieved, converted back to zinc oxide, and returned to the reactor for re-irradiation. If other unwanted metal ions are found in this solution, the zinc ion would be purified by ion exchange prior to oxalate precipitation and calcination to the oxide.

All liquid waste would be neutralized and captured on a solid absorbent as solid waste. Solid compacted waste (excluding cladding hardware and radon holdup charcoal) is estimated at 0.28 cubic meters (10 cubic feet) per year (BWHC 1999).



Source: BWHC 1999.

Figure C–7 Electrochemical Separation of Copper-67 Product from Zinc-67 Target Material

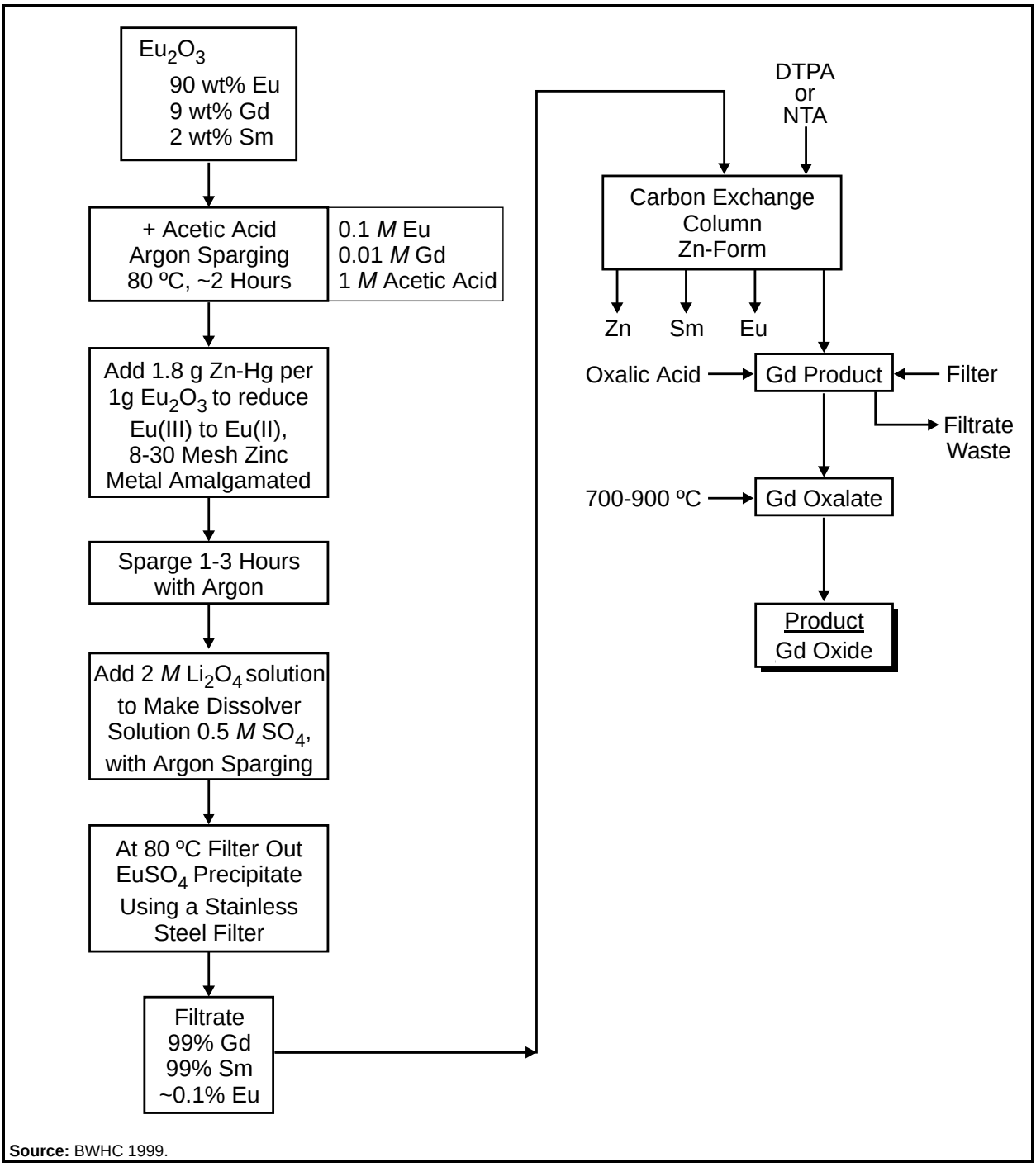
GADOLINIUM-153 FROM EUROPIUM AND SAMARIUM

The capsule containing the irradiated europium oxide target pellets would be transferred to a hot cell for dissolution and europium removal, followed by ion exchange band displacement in a heavily shielded glovebox. The pellets would be removed by cutting open the capsule containing the irradiated europium oxide. The europium oxide pellets would be dissolved with acid, greater than 99.9 percent of the europium isotopes would be removed by sulfate precipitation of europium (II), and ion exchange band displacement would be used to separate 0.1 percent of the remaining europium, along with the samarium and gadolinium, into fractional purified bands (**Figure C–8**). The final gadolinium-153 product would be precipitated, dried, and heated to a high temperature to form gadolinium oxide.

Detailed steps in processing the gadolinium-153 include removal of the oxide pellets from the metal capsule for dissolution in acetic acid. With argon sparging to prevent air oxidation, the solution would be contacted with amalgamated zinc (Jones reductor) to reduce the europium (III) to europium (II). A sulfate salt would be added to precipitate the europium (II), separating it from samarium (III) and gadolinium (III).

The resulting filtrate solution would contain gadolinium, samarium, and less than 0.1 percent of the europium. This solution would be transferred to a shielded glovebox to separate and purify the gadolinium-153 using band-displacement cation exchange chromatography. Ammonium-buffered chelating agents such as nitrilotriacetic acid or diethylenetriamine-pentaacetic acid would be used to fractionate gadolinium/samarium/europium using a zinc-loaded cation exchange column.

The purified gadolinium-153 product solution would be transferred into a second shielded glovebox for oxalate precipitation, filtration, and calcination to the oxide, and then pressed into pellets for shipment to customers. Mixed liquid waste would contain zinc, acetic acid, 45 curies of europium per curie of gadolinium product separated, and ammonium diethylenetriamine-pentaacetic acid organic complexant (0.05 to 0.5 liters [0.013 to 0.13 gallons] of liquid waste per curie of gadolinium recovered). Solid compacted waste (excluding cladding



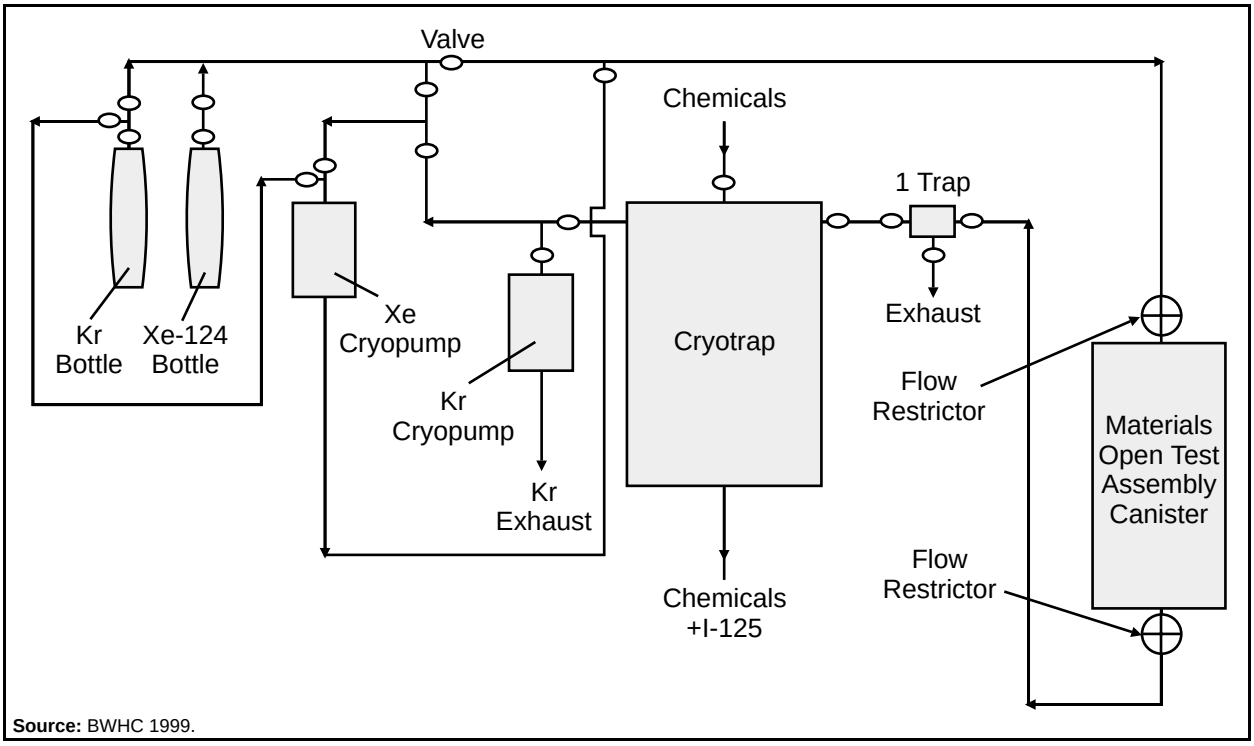
Source: BWHC 1999.

Figure C–8 Procedure for Separation of Gadolinium-153 Product from Europium Target Isotopes

and hardware) is estimated at 0.28 cubic meters (10 cubic feet) per year, including a Jones reductor. Ion exchange columns, sulfate precipitation equipment, and the pellet press would be reused and only become waste at the termination of the project. This waste would be about 0.28 to 2.8 cubic meters (10 to 100 cubic feet) (BWHC 1999).

IODINE-125 FROM XENON-124 GAS TARGET

The method of production and separation would be based on the irradiation of 5 liters (1.3 gallons) of enriched xenon-124 for approximately 2.5 hours, trapping the irradiated gas, letting the xenon-125 gas decay to iodine-125 for approximately 2 days, distilling off the inert gases, chemically reacting to remove iodine-125 from the wall of the cryotrap, followed by final processing, packaging, and shipping of the product. The conceptual system is shown in **Figure C–9**. The system would consist of several cold traps (cryotraps), flow restrictors, a gas canister, and a processing system (glovebox and cryopump).



Source: BWHC 1999.

Figure C–9 Gas-Trapping Procedure for Separation of Iodine-125 Product from Xenon-124 Gas Target

Detailed procedures for producing iodine-125 are at the conceptual stage. Once the system is ready, the xenon-124 gas bottle valve would be opened. The rate of flow would be controlled by the flow restrictors located downstream and upstream of the canister. Once the xenon-124 bottle becomes empty, krypton gas would be valved in and used to push the xenon-124 first through the gas line and next through the canister. Once the gas has been pushed through the downstream flow restrictor, the gas would flow through the iodine trap to the cryotrap, which would be maintained at a low temperature and result in a low pressure (less than 10^{-6} torr). The xenon-124, transmuted xenon-125, and some krypton "pusher" gas would be absorbed on the cryotrap cold surface. The cryotrap inlet valve then would be closed and the krypton in the gas lines would be evacuated to the krypton cryopump.

After about 2 days with the xenon-125 (17-hour half-life) decaying to iodine-125 (60-day half-life), the cryotrap downstream valve would be opened and the cryotrap warmed to first distill the xenon-124 to the xenon cryopump, and then any krypton to the krypton cryopump. The product iodine-125 remaining in the cryotrap then would be chemically reacted (caustic) and the product analyzed, processed, packaged, and shipped.

C.4 REFERENCES

BWHC (B & W Hanford Company), 1999, *Hanford Data Request for FFTF Operational Support Facilities*, Richland, WA, October 6.

Nielsen, D.L., 1999, *Fast Flux Test Facility Data Request in Response to Data Call for Nuclear Infrastructure Programmatic Environmental Impact Statement*, BWHC-9958233, B & W Hanford Company, Richland, WA, December 21.

Appendix D
Fast Flux Test Facility Operations

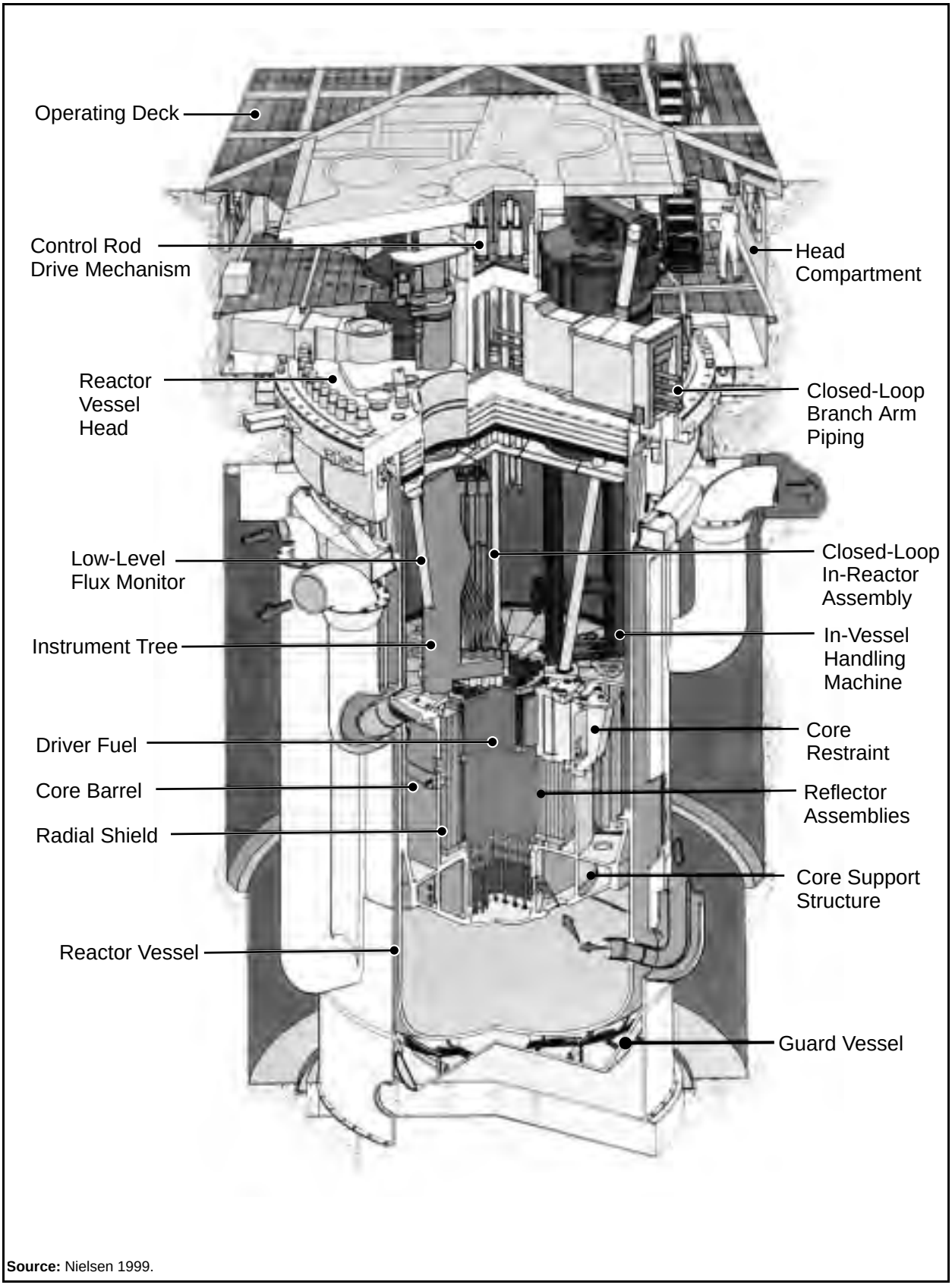
D.1 FAST FLUX TEST FACILITY DESCRIPTION

The Fast Flux Test Facility (FFTF) is an advanced liquid-metal-cooled research reactor located in the 400 Area of the Hanford Site. This appendix provides a description of FFTF operations that is based on the *Fast Flux Test Facility Data Request in Response to Data Call from Nuclear Infrastructure Programmatic Environmental Impact Statement* (Nielsen 1999). The reactor (**Figure D–1**) is located in a shielded cell in the center of the containment building. Heat is removed from the reactor by liquid sodium circulated under low pressure through three separate closed primary piping systems, referred to as loops, which include pumps, piping, and intermediate heat exchangers. These loops are located within inerted cells in containment. **Figure D–2** is a cutaway view of the containment building showing the location of the reactor, primary pumps, and intermediate heat exchangers. Three secondary sodium loops transport the reactor heat from the intermediate heat exchangers to the air-cooled tubes of the dump heat exchangers for dissipation to the atmosphere (FFTF does not generate electricity). **Figure D–3** depicts one of the three cooling loops.

FFTF is a versatile fast flux reactor capable of producing plutonium-238 and a variety of medical and industrial isotopes, as well as supporting materials testing and nuclear research and development activities. Due to the reactor size, the number of available test locations, and the instrumentation capabilities for monitoring individual irradiation and test locations in the core, a wide variety of irradiations and tests can be carried out concurrently.

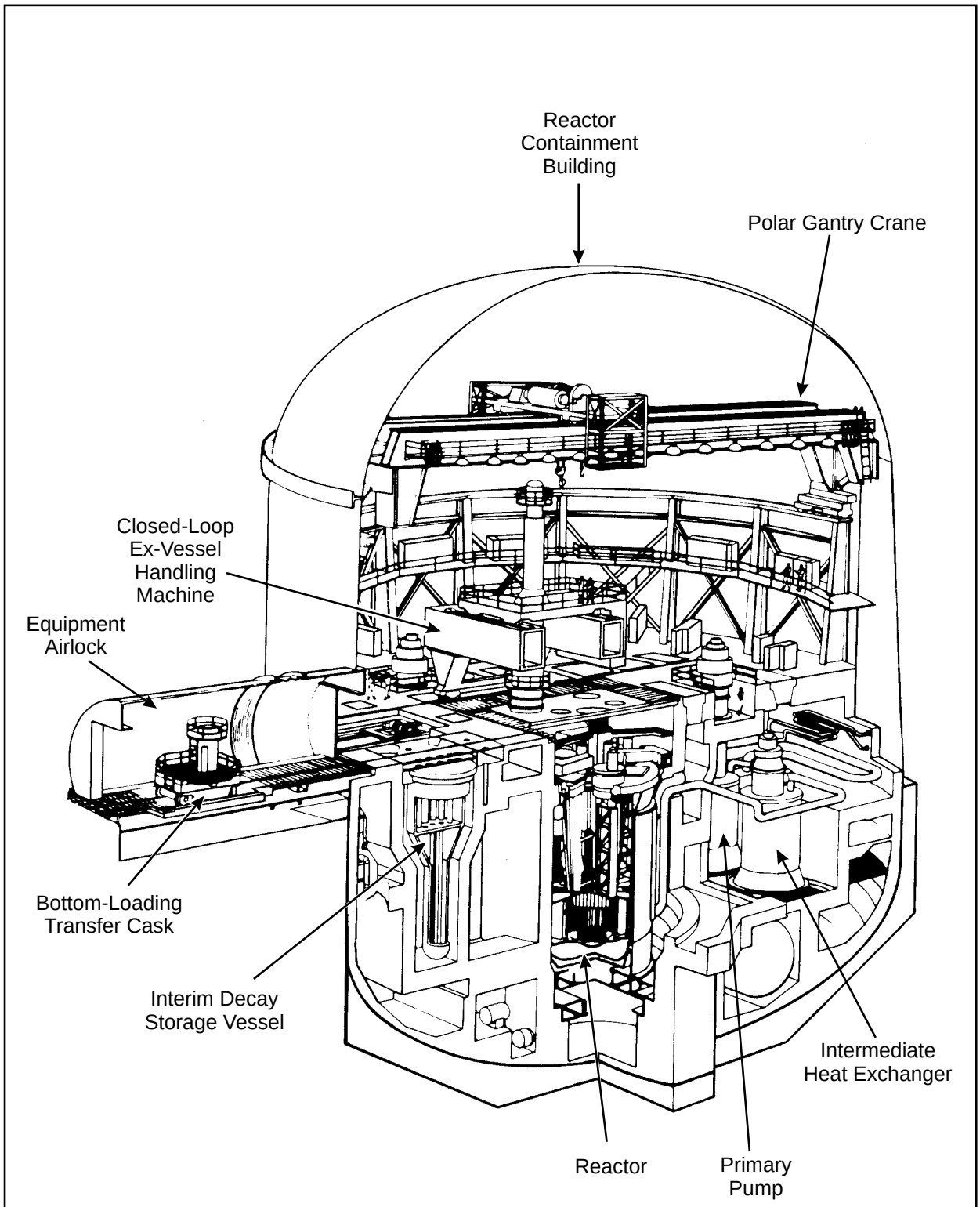
The term "fast flux" is indicative of the high energy (speed) of the neutrons within the reactor core. The total flux density of FFTF (fast plus thermal neutrons) is significantly higher than in a light water reactor. A supply of fast or high-energy neutrons allows FFTF to test a variety of materials and carry out research in an environment where fast neutrons are needed. For producing many isotopes, a thermal neutron environment may be more desirable. In this case, FFTF can slow down, or thermalize, fast neutrons by placing appropriate materials around the irradiation targets. **Figure D–4** shows a possible multitest core configuration for the various missions under consideration. These various reactor locations are discussed in the following paragraphs.

There are eight locations available in the reactor that are called open test assembly positions. These eight locations are distinct from the rest of the reactor in that they allow direct-contact instrumentation for remote monitoring during reactor operations. They are called "open" because they are directly cooled and exposed to the sodium environment within the reactor, as are most of the in-vessel components. What makes these locations distinct are the instrument stalks attached to them that communicate with and extend above the reactor head for routing of various instrumentation packages (**Figure D–5**). They also are positioned so that they allow for inner (core row 2), middle (row 4), and outer (row 6) reactor fluence environments. As many as eight Rapid Radioisotope Retrieval systems could be installed in these positions for the production of short-lived isotopes. However, it is expected that initially only one of these rapid retrieval systems would be installed in open test assembly positions in the reactor core.



Source: Nielsen 1999.

Figure D–1 Cutaway View of the FFTF Reactor Vessel



Source: Nielsen 1999.

Figure D–2 Cutaway View of the FFTF Containment Building

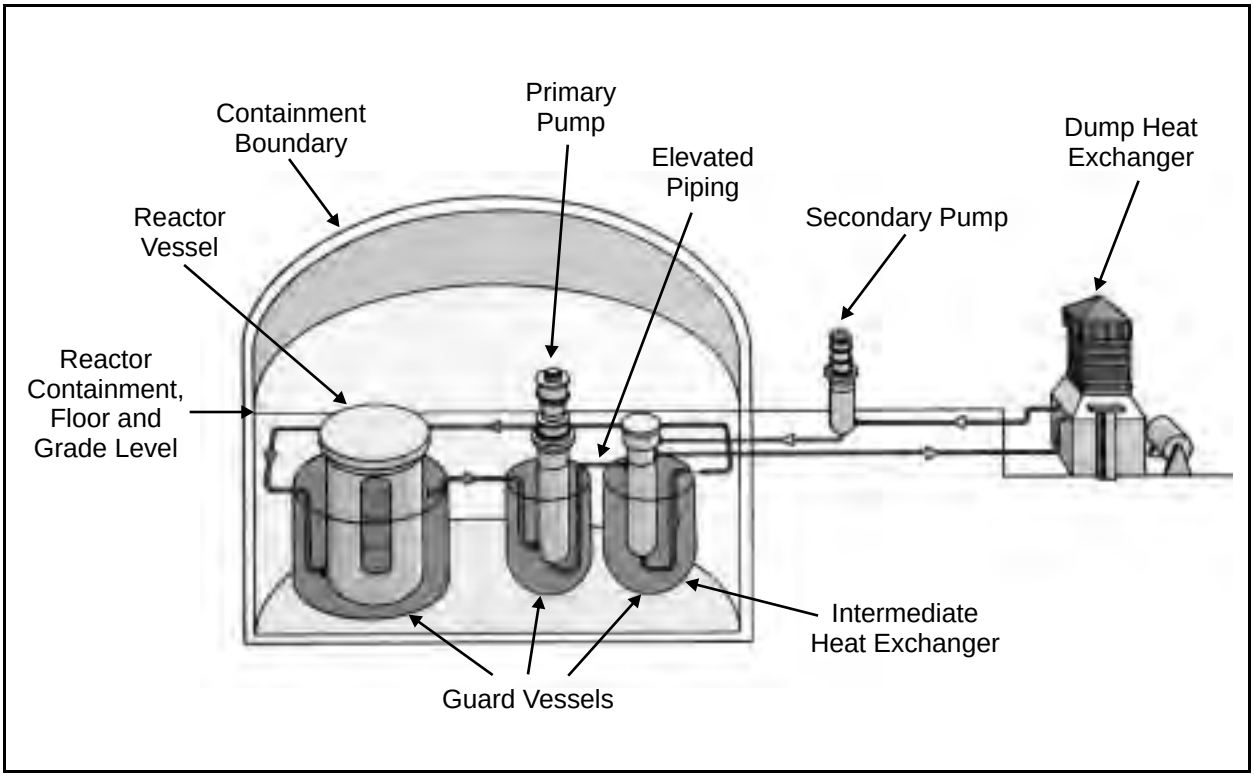
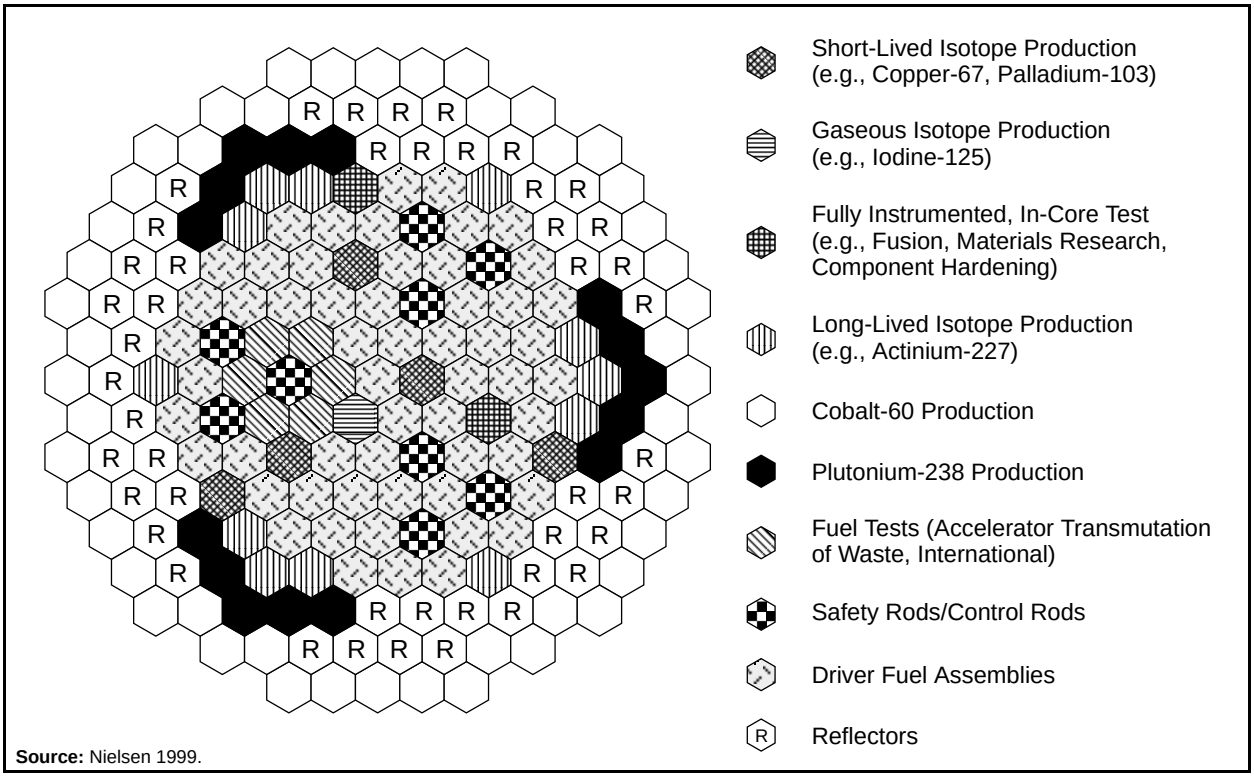
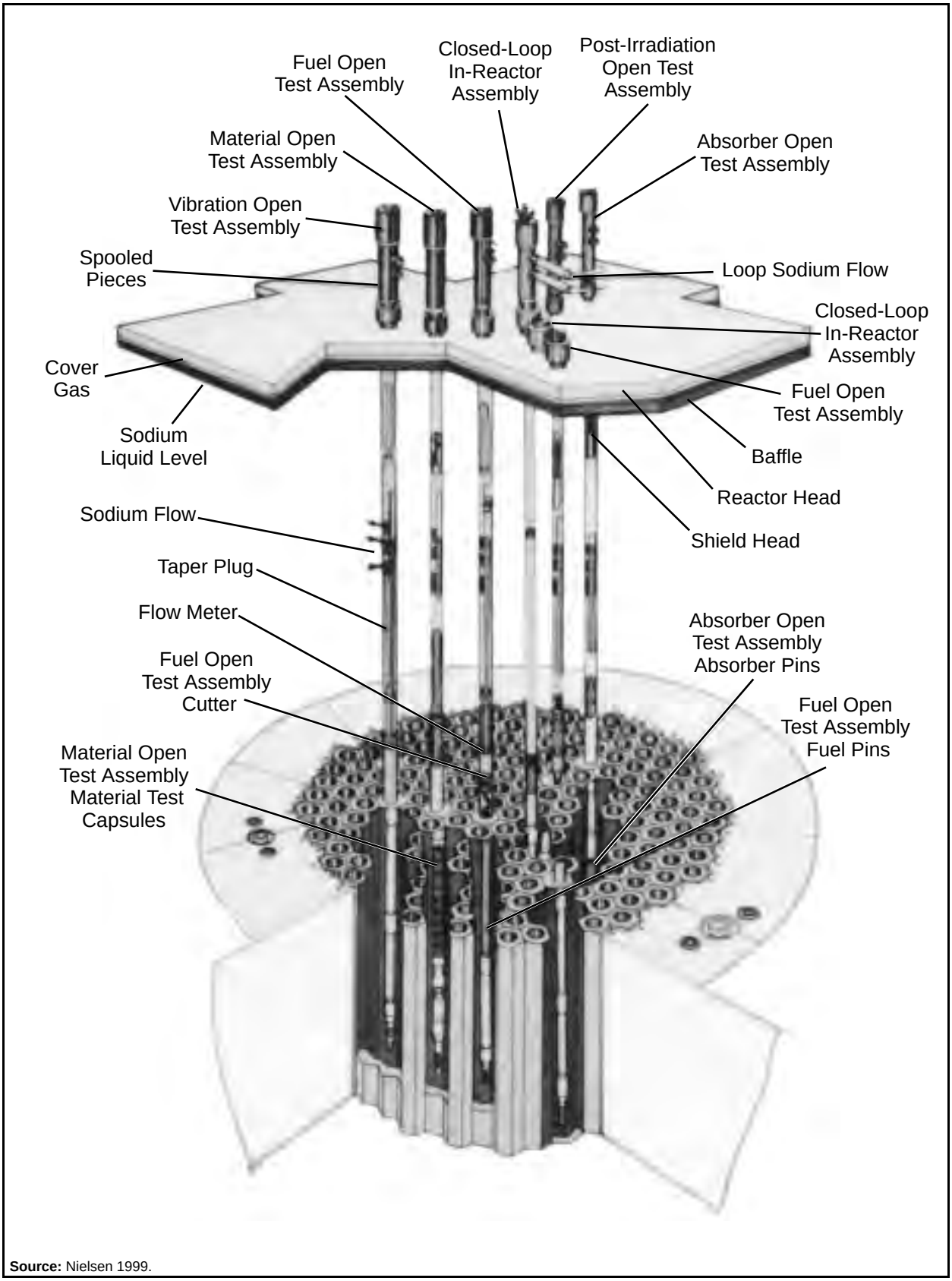


Figure D–3 Schematic View of One FFTF Cooling Loop



Source: Nielsen 1999.

Figure D–4 Multitest Core Example



Source: Nielsen 1999.

Figure D–5 Reactor Core with Various Test Packages Installed

Within the 82 active core locations, there are up to 20 or more additional locations that could contain a standard length (12-foot) irradiation assembly within the active core region. These locations also have specific on-line outlet temperature and flow measurements from installed plant instrumentation in the reactor core Instrument Trees, which are shown in **Figure D–6**. In addition to these test locations within the active fueled region of the core, there are 108 locations available in the surrounding reflector region where other irradiation assemblies could be inserted (e.g., plutonium-238 and cobalt-60 targets). These three basic irradiation configurations enable large-quantity and very diverse testing capabilities. Target designs vary according to the test requirements and the location of the test within the reactor.

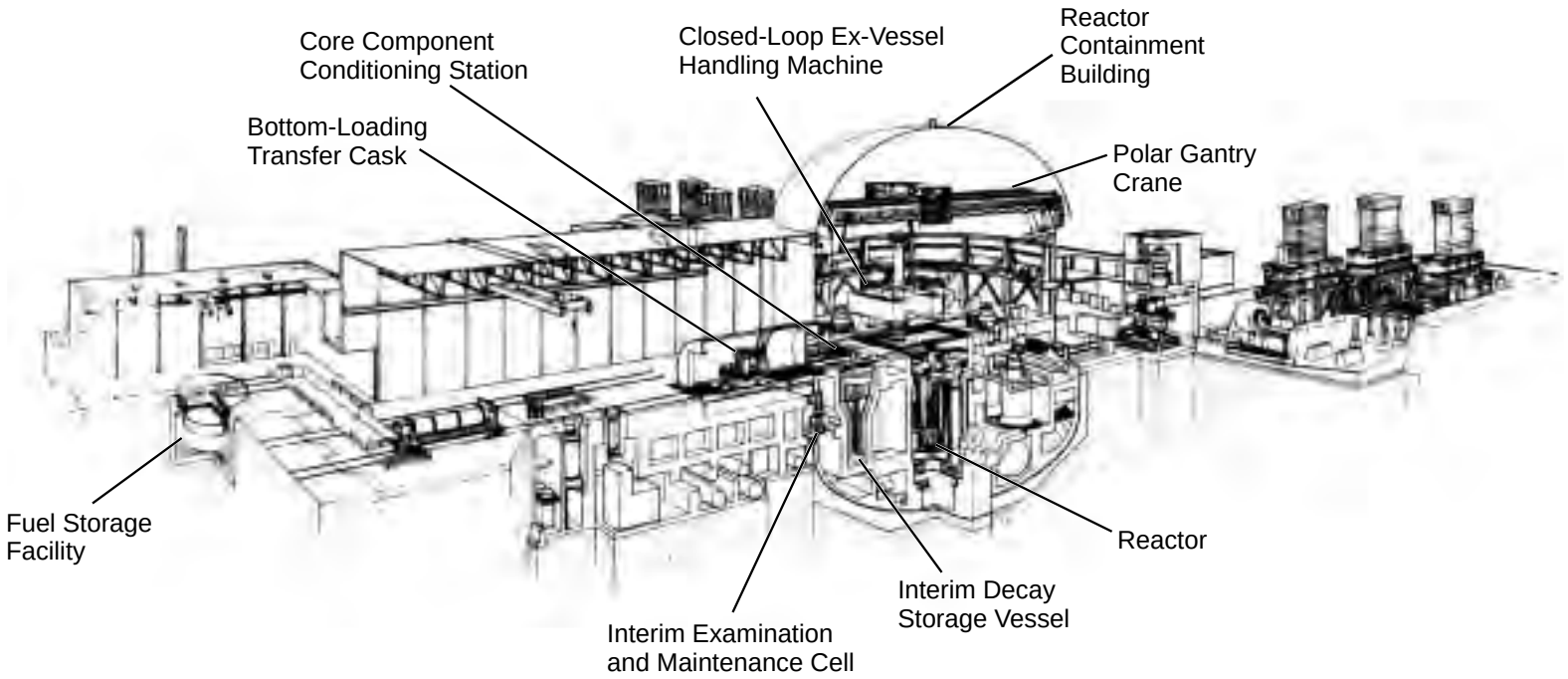
Source: Nielsen 1999.

Figure D–6 Instrument Trees Over the Initial (Unfueled) Core

D.2 LONG-TERM IRRADIATION VEHICLES

The Long-Term Irradiation Vehicles described in Appendix C (e.g., plutonium-238 targets and Long-Term Irradiation Vehicles for isotope production) would be installed in the reactor during normal refueling operations and would be handled using the standard FFTF component handling equipment. FFTF includes areas for receiving, conditioning, storing, installing, and removing from the reactor core all routinely removable core components. There are also areas for washing and storing irradiated fuel and nonfuel reactor components. Test and component examination and packaging capabilities also are provided.

FFTF uses state-of-the-art computer-controlled shielded transfer machines to perform reactor refueling operations as well as component and experiment transfers into and out of the reactor, the Interim Examination and Maintenance Cell, and into shipping/transfer casks. **Figure D–7** is a schematic of the equipment and transfer locations. The transfer machines are designed and operated with safety features and redundant systems to ensure safe transfer of irradiated materials. They are maintained reactor-grade clean; all internal surfaces are made of stainless steel and are maintained inert with argon gas.



Source: Nielsen 1999.

Figure D–7 FFTF Equipment and Transfer Locations

The Closed Loop Ex-Vessel Machine, **Figure D–8**, is used to handle both standard-length reactor components (i.e., 12 feet long) and longer test assemblies such as the open test assemblies described in Section D.3. The Closed Loop Ex-Vessel Machine is used for inserting all components into the reactor vessel. Open test assemblies are inserted directly into the reactor core and standard-length components are placed into In-Vessel Storage inside the reactor vessel before transfer into the reactor core. In-Vessel Storage modules are provided in three sections of the annular region between the core barrel and the reactor vessel thermal liner. Each storage module provides 19 natural-convection sodium-cooled receptacles for core components.

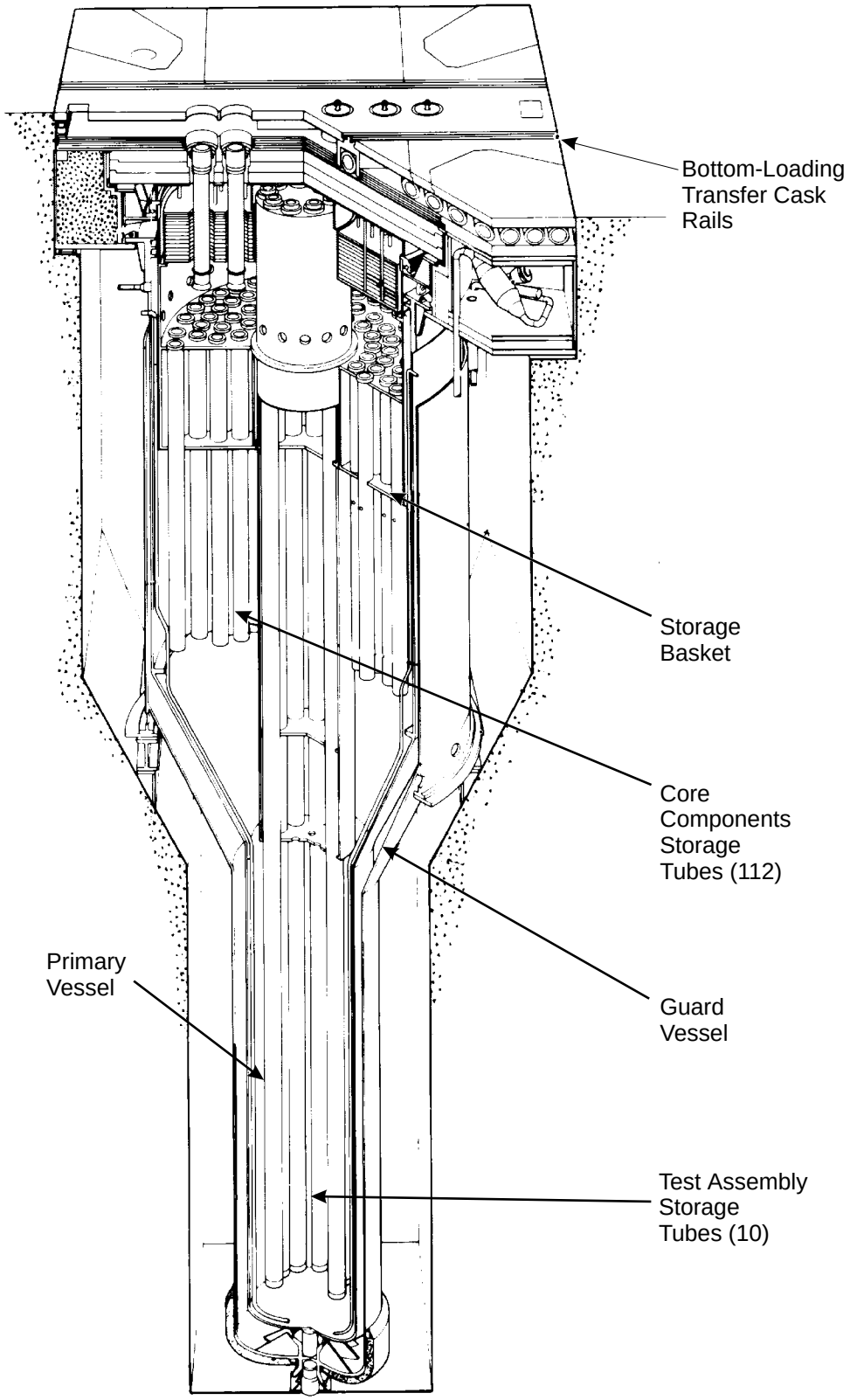
Source: Nielsen 1999.

Figure D–8 Closed Loop Ex-Vessel Machine

Assemblies for material surveillance samples can be installed in the In-Vessel Storage modules. During reactor operations, these samples are exposed to the sodium, thermal, and radiation environment typical of the reactor vessel.

The Closed Loop Ex-Vessel Machine is also used for the transfer of sodium-wetted irradiated components from the reactor vessel to either Interim Decay Storage (a sodium-cooled storage vessel inside containment shown in **Figure D–9**) or to the Interim Examination and Maintenance Cell, depending on whether it will be stored for later disposition or undergo examination. The standard-length assemblies are handled by one of the three In-Vessel Handling Machines for installation into or removal from the core. After irradiation, the Long-Term Irradiation Vehicles would be transferred to the Interim Examination and Maintenance Cell for washing and disassembly prior to shipment of the pins to the processing facility for isotope extraction and purification.

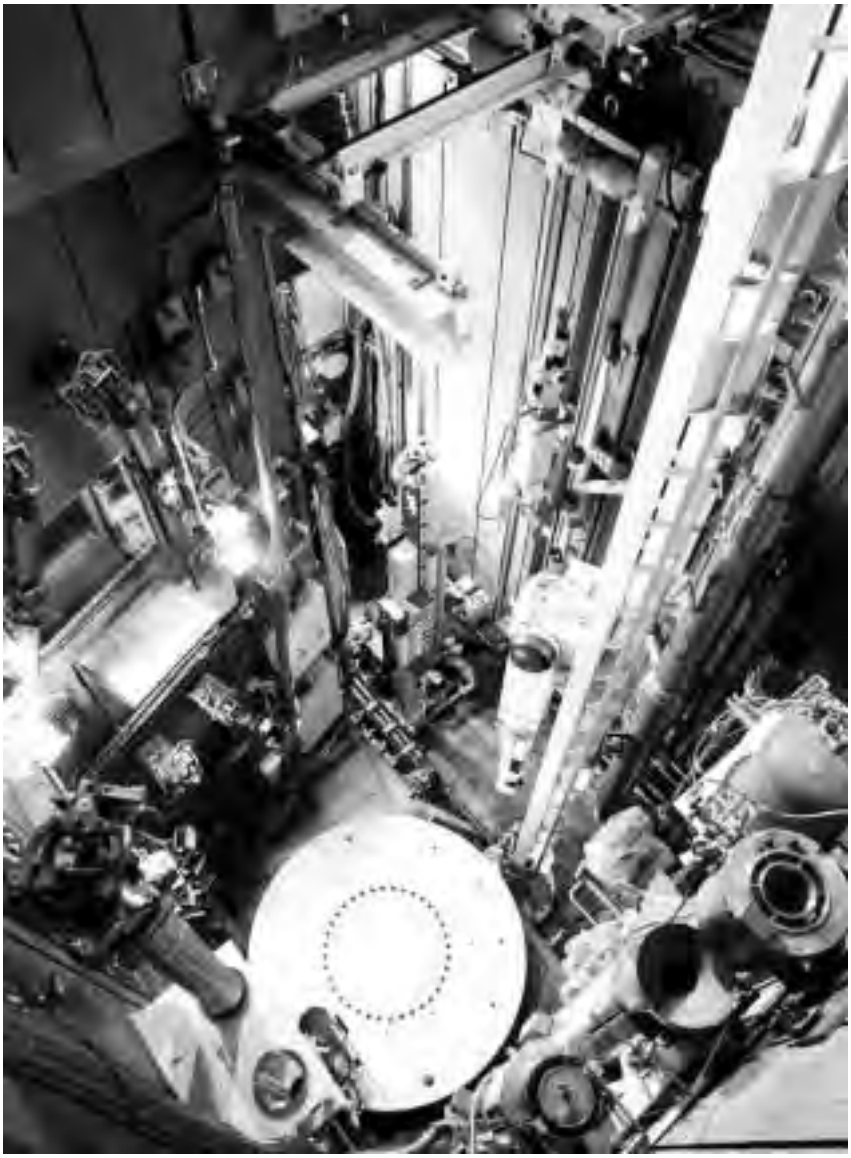
The Interim Examination and Maintenance Cell (**Figure D–10**) is a large, shielded, hot cell complex located inside containment that provides a reliable means of conducting nondestructive examination of test assemblies and core components under controlled argon-atmosphere conditions. Four levels of operating galleries provide visual access for remotely operating the hot cell equipment. This highly shielded hot cell has a



Source: Nielsen 1999.

Figure D–9 Interim Decay Storage Vessel

significant number of remote tools and equipment for diverse examination and disassembly needs. The hot cell, which is over 50 feet deep, contains two cranes and two very large electromechanical manipulators as well as multiple pairs of smaller master-slave-type manipulators at various operating levels for component and equipment handling. A sodium-cleaning station is available to wash irradiated components of all external sodium residues after removal from the reactor's sodium environment. This sodium-removal system has been used extensively to wash all fuel and experimental test assemblies processed in the Interim Examination and Maintenance Cell, as well as many of the FFTF spent fuel assemblies as they were offloaded to interim dry storage. The demineralized water used for washing is recycled through ion beds. The ion beds are periodically changed out and buried as low-level radioactive waste.



Source: Nielsen 1999.

Figure D–10 Interim Examination and Maintenance Cell

Following sodium removal and drying, irradiated components can be remotely disassembled using the manipulators, fixtures, and special tooling located within the Interim Examination and Maintenance Cell. For example, disassembly of plutonium-238 or cobalt-60 targets may be required to accommodate shipments of shorter target pin sections, depending on handling limitations of the selected processing facility. Various equipment is available for postirradiation examinations (e.g., weight, visual exam/photography, disassembly, and packaging for shipment). The hot cell also has been used for interim examination of tests and complex reassembly and qualification to allow a test to be returned to the reactor for further irradiation.

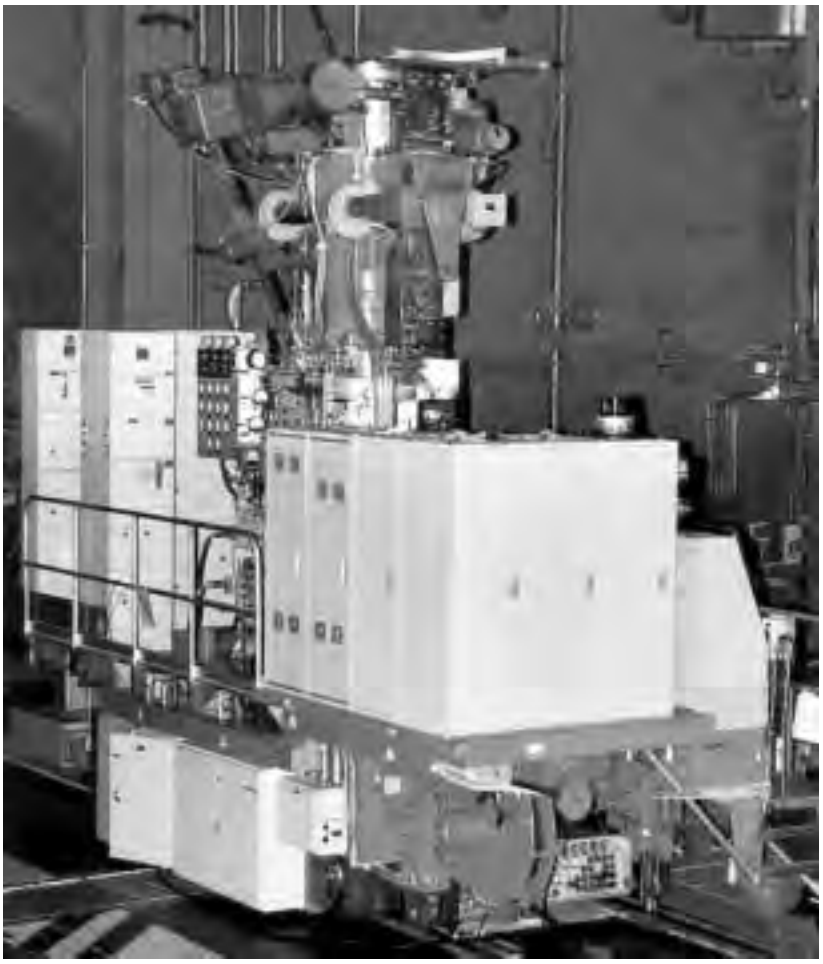
The Bottom-Loading Transfer Cask (**Figure D–11**) is used to transfer test articles, standard-length components, and specimen containers from the Interim Examination and Maintenance Cell to the cask-loading station for transfer to offsite facilities for further examination, or to the Fuel Storage Facility (a sodium-cooled storage vessel located outside of containment) for subsequent storage.

Irradiation assemblies can be transferred from the Interim Examination and Maintenance Cell to the cask-loading station for placement into shipping casks for transport to the processing facility or to offsite locations

for destructive examination, other testing, or processing. The cask-loading facility has the capability to handle large spent fuel casks weighing up to 75 tons for vertical loading and unloading.

D.3 OPEN TEST ASSEMBLIES

During its 10 years of operation, FFTF supported a large and varied test program for industry, nuclear energy (domestic and international), nuclear defense, and medical research and treatment. The testing focused primarily on reactor fuel and different fuel assembly material evaluations, but also provided significant testing for many other programs. Following is a brief description of the major types of tests that were performed at FFTF using the open test assemblies. Similar test vehicles could be used to support the new missions:



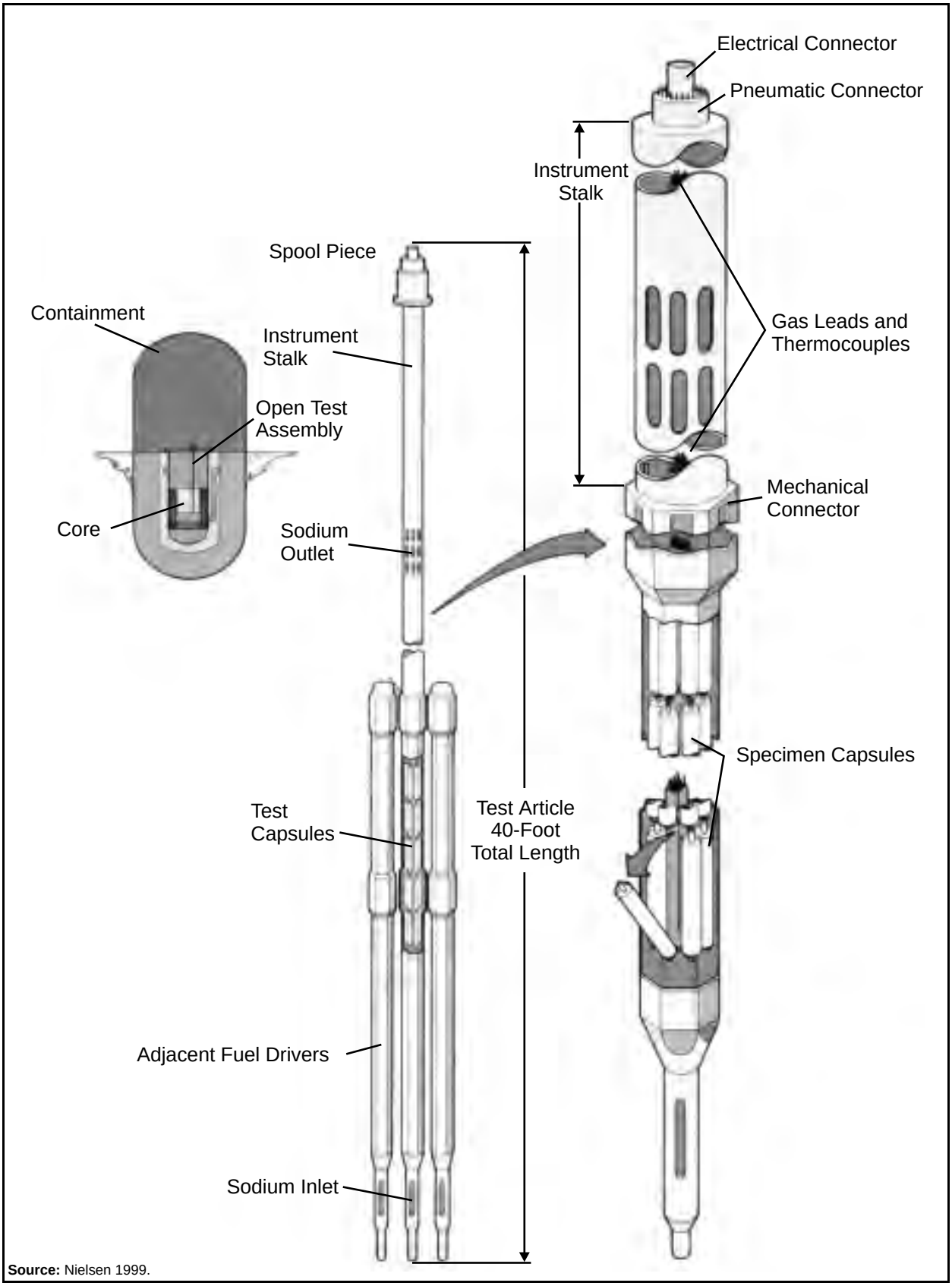
Source: Nielsen 1999.

Figure D–11 Bottom-Loading Transfer Cask

- **Material Open Test Assembly.** The 38-foot-long Material Open Test Assembly (**Figure D–12**) provided multiple containers capable of irradiating many different material specimens. Each container was individually temperature-controlled by the online mixing of argon and helium gases in the container annulus. This provided varying heat transfer from the container to the reactor sodium coolant. The support system for this test vehicle includes multiple gas lines, temperature control loops, and an online control and monitoring system.

- **Fusion Material Open Test Assembly.** The reactor portion of this test vehicle was essentially identical to the Material Open Test Assembly and included many material test specimens as well as the two canisters that were part of the fusion testing program. This test series was a joint venture for the United States, Canada, and Japan to evaluate tritium production by the irradiation of lithium oxide. The purpose of the experiment was to measure tritium release characteristics and thermal stability of lithium oxide as a function of neutron exposure, temperature, gas composition, and sweep gas flow rates. This equipment also included the instrumentation and controls for tritium measurement, analysis, and recovery.

- **Absorber Open Test Assembly.** The Absorber Open Test Assembly provided for online instrumentation (temperature and pressure) of standard boron carbide absorber pins used in reactor control and safety rods.



Source: Nielsen 1999.

Figure D–12 Material Open Test Assembly

- **Fuel Open Test Assembly.** The Fuel Open Test Assembly provided direct measurement during reactor operation of temperatures and pressures of individual fuel pins allowing monitoring of fuel assembly performance during the entire irradiation phase.

D.4 RAPID RADIOISOTOPE RETRIEVAL SYSTEMS

Rapid retrieval systems would be installed in selected open irradiation assembly positions for the production of short-lived isotopes. These systems would allow target materials to be inserted and withdrawn from the reactor core region with the reactor operating at power. Systems for routinely inserting and removing irradiation targets, nuclear instrumentation, and research hardware at an operating reactor have been in use at various research reactors throughout the world for years. Most of these systems use either a pneumatic rabbit-type system or a mechanical cable-type system for insertion and retrieval. **Figure D–13** is a conceptual layout of an FFTF rapid retrieval system, which consists of three major components: a 40-foot-long in-reactor thimble assembly, a replaceable string or chain of isotope target carriers, and a target carrier insertion and retrieval system.

The target carrier insertion and retrieval system(s) would be installed external to the reactor to shuttle a target carrier chain into and out of the core region. This system could use some form of mechanical cable insertion and retrieval mechanism or could be based on a pneumatically operated system. Ideally, the insertion and retrieval system would load irradiated target chains directly into the transportation cask for shipment to the hot cell laboratory facilities for isotope separation and purification.

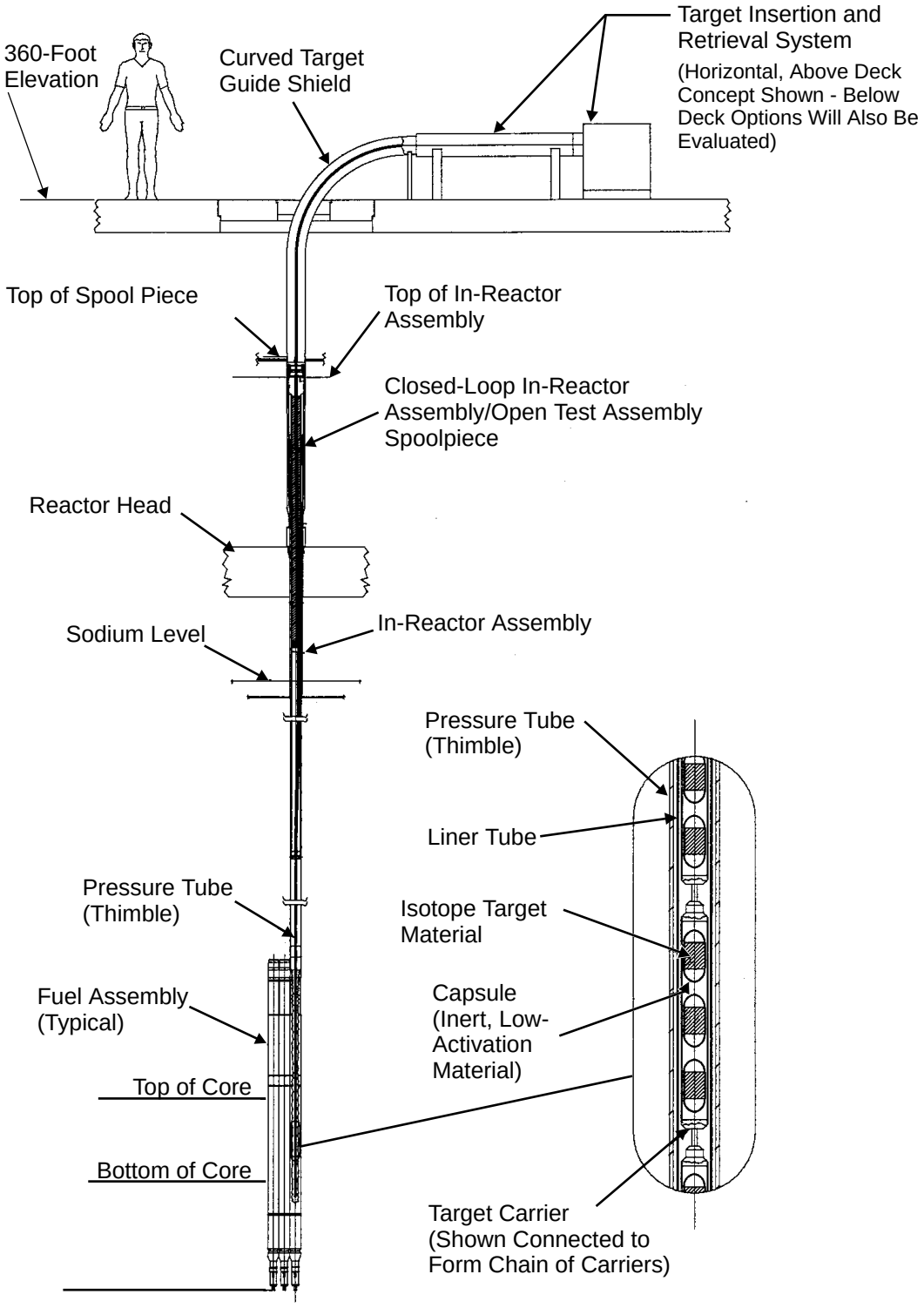
In addition to irradiating solid targets in the rapid retrieval system carrier chains, gas targets also could be irradiated to produce short-lived isotopes. Two options would be evaluated for producing the gas-based isotopes. One option would involve one or more small-diameter, thin-wall tubes routed down through the in-reactor thimble assembly into the active core region. These tubes would be connected via shielded and preheated tubes to a shielded ex-reactor gaseous isotope recovery system. The practice of routing external gas lines into the active core region is not new at FFTF and has been used in several irradiation test assembly designs installed in the reactor (e.g., the Material Open Test Assembly used externally supplied gas mixtures to control material sample temperatures, and the Fusion Material Open Test Assembly had gas lines routed to a glovebox for tritium sampling and gas analysis).

A second option for producing gas-based isotopes would involve irradiating capsules filled with a high-pressure target gas. The gas-filled capsules would be installed in a target carrier and could become part of a target chain.

D.5 REACTOR CORE CONFIGURATION PLANNING

Typical operating cycles for FFTF were approximately 100 days at power followed by shutdown periods that ranged from approximately 20 to 30 days for short outages to 60 to 90 days for extended outages, depending on the extent of maintenance and refueling to be performed. These same operating cycles were assumed for evaluating restart activities. There may be benefits to longer operating cycles (e.g., increased capacity factor, reduced equipment use); therefore, future consideration may be given to such intervals.

Reactor core configuration planning would be completed prior to the start of reactor servicing in preparation for the next operating cycle. This planning would accommodate the user-defined irradiation requirements for each target. Those requirements typically would be defined in terms of neutron energy and flux level, duration of the irradiation (or cumulative exposure), and temperature. Given these requirements, the in-core location for each target assembly would be analyzed and decided based on the neutron energy and flux level at that location; the neutronic characteristics of adjacent assemblies; coolant flow and temperature at that location;



Source: Nielsen 1999.

Figure D–13 Conceptual Layout of the Rapid Radioisotope Retrieval System

reactor power level; target gamma heating and the resultant target temperature; and planned operating cycle duration. The resultant core configuration would have to meet the nuclear safety requirements and limitations defined in the Final Safety Analysis Report and the Technical Specifications.

During prior FFTF operations, the reactor physics and nuclear safety aspects of the planned core configuration were analyzed prior to the start of each operating cycle. If FFTF is directed to restart, the reactor initially would be refueled similar to the configuration in place at shutdown. The data from that operating cycle would be included in analysis of the reloaded core to predict safety margins and control rod positions for initial criticality. Following completion of physics testing of that initial core configuration, it would be used as the "reference core configuration," from which each subsequent core configuration change made to accommodate fuel burnup and irradiation service requirements would be analyzed for control and safety purposes.

D.6 TARGET TEST AND DEVELOPMENT

Testing programs would be conducted for new materials and target designs to be irradiated in the reactor. Brief descriptions of the types of testing that would be associated with the plutonium-238 and medical isotope production missions are given below.

Plutonium-238 Production—As discussed previously, it is expected that the targets used for plutonium-238 production in FFTF would be similar to the concepts developed during the plutonium-238 production core study that was done between 1992 and 1993. Target design would be based on alternating thin pellets or wafers of neptunium dioxide and yttrium hydride. To support the final design of the initial plutonium-238 production target assemblies, at least one lead test assembly would be designed and fabricated to be available after the initial low-power physics testing. Confirmatory irradiation tests would be done in the lead test assembly(ies) at the core periphery prior to the start of fabrication of the plutonium-238 production assemblies. These tests, which would require small amounts of neptunium dioxide, would be done to demonstrate items such as target wafer integrity and dimensional stability, and to confirm target isotopic content from irradiation. Additionally, exploratory materials tests would be irradiated in parallel to investigate any material compatibility or performance issues, and to evaluate options to improve plutonium-238 production rates or optimize/reduce target fabrication costs. These tests, which could be done under accelerated irradiation conditions in a variety of assemblies such as the Long-Term Irradiation Vehicle, could include tests of other hydride materials such as zirconium hydride or calcium hydride as well as various foil wrapper and cladding materials.

Medical and Industrial Isotope Production—A testing program would be performed to support the development and detailed design of the Rapid Radioisotope Retrieval system and the associated initial targets. Feature tests would be conducted as necessary to support development of key portions of target insertion, retrieval, and handling equipment. High-temperature furnace tests would be performed as required to ensure materials compatibility for those materials or material combinations for which high-temperature data do not exist. This would be done both for the Rapid Radioisotope Retrieval system and for the Long-Term Irradiation Vehicle assemblies. A full-scale, heated mockup of the key portions of the Rapid Radioisotope Retrieval system would be built, and insertion and retrieval tests performed using simulated targets and target strings. Ex-reactor modifications needed to support insertion, production, and retrieval of targets in Rapid Radioisotope Retrieval assemblies would be completed, and acceptance tests conducted to the extent possible before initial criticality upon a decision to restart.

Following initial criticality and low-power physics testing, the reactor would be shut down and the lead Long-Term Irradiation Vehicle assemblies and at least one 38-foot-long Rapid Radioisotope Retrieval assembly would be inserted into the reactor. Following additional acceptance testing and low-power physics testing, including Rapid Radioisotope Retrieval characterization, the lead targets would be installed in the Rapid Radioisotope Retrieval assembly. These could consist of relatively small quantities of key target

materials plus dosimeter sets for neutron environment characterization. On removal from the reactor, the initial targets irradiated in the Rapid Radioisotope Retrieval assembly would undergo a variety of special inspections, tests, and radioassays as part of a characterization and target qualification program. Over a period of months, the quantity of target material incorporated into the Rapid Radioisotope Retrieval targets would be increased as feasible to provide beneficial quantities of product isotopes. As the demand for short-lived isotopes grows, additional Rapid Radioisotope Retrieval assemblies and associated support systems would be installed in the reactor as needed to support production.

D.7 NUCLEAR RESEARCH AND DEVELOPMENT

FFTF has demonstrated the capability to produce high-energy, high-fluence neutrons for multiple nuclear science and irradiation services applications. High-energy neutrons are fast neutrons that can be used for the transmutation of elements into useful isotopes (e.g., medical and industrial) and for the investigation and development of materials and components that can be used in harsh, radioactive environments (e.g., fusion reactors). High fluence refers to FFTF's capability to produce a lot of neutrons in a given test volume within the reactor.

The large test volume in FFTF allows the production of larger quantities of isotopes and the ability to test more materials and components when compared to other neutron sources. While other neutron sources may have similar neutron high energies or fluences, FFTF is unique in simultaneously providing all three attributes in a single test facility. FFTF can also produce large quantities of epithermal neutrons by the use of moderating materials that slow down the neutrons in specific areas of the core. These distinctive flux tailoring features, coupled with its large core volume, the ability to vary power from a nominal 100 megawatts up to 400 megawatts, and highly instrumented testing capabilities, enable the reactor to function successfully as a multiple-mission nuclear science and irradiation services facility. Researchers from many different countries have used FFTF for nuclear materials testing and fuel research.

There is particular interest in materials testing associated with extension of commercial nuclear power plant license renewals, cooperative international fusion energy research, space power technology, and transmutation of wastes as a means to destroy long-lived isotopes from commercial spent nuclear fuel. Another area of interest is developing nuclear technologies that advance global nonproliferation. FFTF is ideally suited for the study, research, testing, development, and demonstration of technologies necessary to safely convert plutonium-based materials for disposition and use as proliferation-resistant fuel. Target assemblies to be irradiated in support of these mission areas could be fuel or other materials configured similar to a standard driver fuel assembly (i.e., target material encased in sealed pins and the pins placed in a ducted assembly). Material specimens could also be installed in an open test assembly position within a Material Open Test Assembly, which is described in more detail in Section D.3.

One of the proposed testing activities consists of fuel testing for the Accelerator Transmutation of Waste Program. A fuel development activity for this program could use FFTF for irradiation testing. The tests would be fabricated by the Accelerator Transmutation of Waste program and transported to FFTF for irradiation. Specific test compositions and irradiation parameters for these test fuel assemblies have not been defined. It is anticipated that initial tests would involve a few pins in an assembly, while later tests could involve entire assemblies. The target pins are described as containing a matrix of zirconium and transuranic elements; a composition of 75 percent zirconium and 25 percent transuranic elements, by weight. The transuranic elements would likely be light water reactor discharge fuel at a typical burnup of 33,000 megawatt-days per metric tons of uranium that is stripped of essentially all uranium and fission products. Comparisons of this fuel to the standard FFTF fuel indicate comparable plutonium compositions. Therefore, for purposes of this environmental impact statement evaluation, the Accelerator Transmutation of Waste fuel assemblies were modeled as standard FFTF driver fuel assemblies.

D.8 REFERENCES

Nielsen, D.L., 1999, *Fast Flux Test Facility Data Request in Response to Data Call for Nuclear Infrastructure Programmatic Environmental Impact Statement,* BWHC-9958233, B & W Hanford Company, Richland, WA, December 21.

Appendix E
New Research Reactor Operations

E.1 INTRODUCTION

A preconceptual design of a new research reactor was developed to meet the U.S. Department of Energy's (DOE) missions of (1) producing medical and industrial radioisotopes, (2) producing plutonium-238 (minimum net annual production of 5 kilograms [11 pounds]), and (3) supporting nuclear energy research and development. In accordance with U.S. nuclear nonproliferation policy, a design limitation of this new research reactor is that it can use only low-enriched uranium with an enrichment of less than 20 percent uranium-235. This preconceptual design includes the basic elements of the research reactor facility, which are sufficient to support this programmatic environmental impact statement, but does not include design details (i.e., system and layout drawings, bill of materials, electrical and piping routing, etc.) commensurate with a complete preliminary reactor design. The reactor design uses proven and licensed low-enrichment fuel type in conjunction with a reactor system that features numerous inherent safety features, such as:

- A large low-pressure, low-temperature coolant inventory around the core to keep the core covered under all accident conditions, absorb heat, and filter any released radioisotopes
- A large prompt negative fuel temperature coefficient of reactivity to mitigate any accidental reactivity insertion event
- Extensive research, development, and operational experience in conjunction with a robust nuclear fuel design
- Minimal reliance on the operation of any active system or component for safe shutdown and accident response

Although significant additional work would be required to develop a detailed preliminary design of this research reactor, the preconceptual design provides the basis for evaluating the environmental impacts and cost of this alternative.

E.2 NEW RESEARCH REACTOR GENERAL DESCRIPTION

The design of the new research reactor is based on current research reactor designs, which have been approved by both the U.S. Nuclear Regulatory Commission (NRC) and the International Atomic Energy Agency (IAEA), as well as nuclear regulatory authorities of many nations. Most low-enriched uranium operating research reactors use one of two types of nuclear fuel: (1) uranium-aluminum matrix, or (2) uranium-zirconium-hydride (UZrH) with either a light or heavy water neutron moderator. In addition, most research reactors are based on either a pool or tank enclosure for the reactor core. Research reactor designs are based on several key factors including, but not limited to: (1) mission, (2) enrichment limits, (3) required neutron flux, (4) thermal limits, (5) irradiation volume, (6) safety, (7) operations, and (8) cost. All these factors were considered in the preconceptual design of this new research reactor.

Reactor core physics scoping calculations were performed using the SCALE-4.4 (ORNL 1998) computer code package to evaluate three different low-enriched uranium nuclear fuel designs: (1) ternary uranium-zirconia-calcium oxide clad in stainless steel 304 similar to that used in the Power Burst Facility reactor at the Idaho National Engineering and Environmental Laboratory (INEEL) (ANC 1971), (2) uranium-aluminum-silicide alloy clad in aluminum (NRC 1988), used in many research reactors, and (3) UZrH alloy clad in Incoloy-800 (a nickel-iron-chromium alloy similar to stainless steel), known as TRIGA (training, research, isotopes General

Atomics) fuel (GA 2000). All core physics analyses assumed a light water moderator instead of the alternative heavy water moderator because of the significant additional cost related to the use of heavy water. Based on this scoping analysis, coupled with the desired mission of this reactor, current nuclear fuel manufacturing capability, and safety considerations, a TRIGA fuel design was selected for the new research reactor. The principal distinguishing features of the TRIGA fuel are its proven safety performance during power pulsing and its demonstrated long-term irradiation integrity. The TRIGA fuel core provided the largest irradiation volume and highest thermal neutron flux for low-enriched uranium-235 in a research reactor. The high thermal neutron flux is desirable for plutonium-238 production and for producing most of the medical and industrial radioisotopes. Although the 50-megawatt power level of the new TRIGA research reactor is larger than the largest currently operating TRIGA reactor power of 16 megawatts, the fuel design is almost identical to the current TRIGA 10-megawatt high power design, and the system thermal-hydraulic performance represents a linear extrapolation of existing designs. The power density of the 50-megawatt design is less than or equal to that for existing TRIGA reactor designs.

To produce the desired quantity of plutonium-238 along with medical and industrial radioisotopes concurrently with nuclear research and development support, it was determined that a reactor core power of 50 megawatts-thermal would be adequate. Although not evaluated for environmental impacts, the core and reactor systems were designed to accommodate a power level of up to 100 megawatts. At the 50-megawatts-thermal power level, the core requires an active cooling system with forced coolant flow to maintain the fuel below its material thermal limits. The new research reactor cooling system design uses a tank within a pool, which is connected to primary coolant circulating pumps, heat exchangers, and an ultimate heat sink consisting of two cooling towers. The pool is housed in a reactor building which also encloses the pumps, heat exchangers, secondary systems, and spent nuclear fuel storage pool. The spent nuclear fuel storage pool, sized to store the reactor core's discharged spent nuclear fuel for its entire 35-year lifetime, can be hydraulically connected to the reactor core pool for refueling and emergency reflooding. The ultimate heat sink cooling towers, air exhaust stack, and emergency diesel generators are located outside the reactor building.

E.3 NEW RESEARCH REACTOR FUEL AND CORE DESIGN

As discussed in Section E.2, TRIGA fuel was selected for the new research reactor core. TRIGA fuel has been used in research reactors since 1958 with over 50 TRIGA reactors currently operating worldwide, including 19 operating U.S. TRIGA reactors that are licensed by the NRC (NRC 1999), at licensed steady-state power levels of 0.02 to 16 megawatts-thermal and power pulsing capabilities of up to 22,000 megawatts-thermal (GA 2000; RRSAS 1999; Simnad 1980). This fuel design has demonstrated ability to provide high burnup cladding integrity as well as reliable performance up to actual burnups of 75 percent uranium-235 (Simnad 1980). The TRIGA fuel, because of its unique composition of hydrogen moderator intimately mixed into the fuel itself, has a large negative fuel temperature reactivity coefficient, which shuts the reactor down during any power excursions or reactivity-induced transients.

E.3.1 Nuclear Fuel Design

The new research reactor nuclear fuel design is based on an extension of current licensed low-enriched uranium TRIGA fuel designs for 10- to 16-megawatts-thermal reactors. A comparison of the current high-power low-enriched uranium TRIGA fuel design and the new research reactor low-enriched TRIGA fuel design (IAEA 1992) is presented in **Table E–1**.

As presented in Table E–1, the new research reactor fuel design is identical to current low-enriched TRIGA fuel for higher power cores except that the new reactor fuel has a larger assembly configuration array (i.e., 8 by 8 versus 4 by 4) and a longer active fuel length (153.7 centimeters [60.5 inches] versus 55.88 centimeters

Table E–1 Comparison of New Research Reactor Fuel Design to Current Low-Enriched Uranium TRIGA Fuel Design

Nuclear Fuel Design Parameter	Current Low-Enriched Uranium TRIGA Fuel Value	New Research Reactor TRIGA Fuel Value
Fuel assembly rod configuration	Square 4 rods by 4 rods	Square 8 rods by 8 rods
Fuel assembly shroud outside dimension	7.572 centimeters (2.981 inches) by 7.963 centimeters (3.135 inches)	16.52 centimeters (6.5 inches) by 16.52 centimeters (6.5 inches)
Fuel rod center-to-center pitch	1.634 centimeters (0.643 inch)	2 centimeters (0.787 inch)
Fuel rod cladding outside diameter	1.377 centimeters (0.542 inch)	1.377 centimeters (0.542 inch)
Cladding material	Incoloy-800	Incoloy-800
Cladding thickness	0.041 centimeter (0.016 inch)	0.036 centimeter (0.014 inch)
Fuel-to-cladding radial gap	0.0022 centimeter (0.0009 inch)	0.0025 centimeter (0.001 inch)
Fuel rod gap and gas plenum backfill gas and fill gas pressure	Helium 0.0103 megapascal (1.5 pounds per square inch absolute)	Helium 0.0103 megapascal (1.5 pounds per square inch absolute)
Fuel pellet outer diameter	1.295 centimeters (0.510 inch)	1.295 centimeters (0.510 inch)
Fuel pellet height	13.97 centimeters (5.5 inches)	13.97 centimeters (5.5 inches)
Fuel pellet composition	$UZrH_{1.6}$-Er	$UZrH_{1.6}$-Er
Fuel pellet uranium weight fraction	45 percent	45 percent
Uranium-235 enrichment	19.95 percent	19.7 percent
Hydrogen-to-zirconium ratio	1.6	1.6
Fuel pellet erbium weight fraction	0.8 percent	0.8 percent
Fuel rod active fuel length	55.88 centimeters (22.0 inches)	153.7 centimeters (60.5 inches)
Mass of uranium per fuel rod	274 grams (0.604 pound)	754 grams (1.660 pounds)
Mass of uranium-235 per fuel rod	54.8 grams (0.121 pound)	148.4 grams (0.327 pound)
Mass of uranium per "all fuel rod" fuel assembly	4.38 kilograms (9.64 pounds)	48.3 kilograms (106.2 pounds)
Total fuel rod length	76.2 centimeters (30 inches)	176 centimeters (69.3 inches)

Key: $UZrH_{1.6}$-Er, uranium-zirconium-hydride with erbium.

[22.0 inches]). The larger array and longer length were selected to meet the plutonium-238 production requirements and maintain high safety factors with respect to fuel thermal performance.

E.3.2 Nuclear Core Design

Along with fuel rods, the core is designed to contain a number of plutonium-238, medical radioisotope, and industrial radioisotope production target rods. These target rods would occupy positions in a fuel assembly where a fuel rod would otherwise exist. Each of these positions would have an Incoloy-800 alloy guide tube with the same dimensions as the fuel rod cladding. The target rods, clad in Incoloy-800, would be inserted into these guide tubes for their design irradiation time period. In addition, some fuel rod positions in core fuel assemblies would be replaced with similar guide tubes to accommodate Incoloy-800-clad boron carbide control rods. Boron carbide is a proven, accepted, and widely used neutron absorber for control rods. **Figure E–1** presents a representative illustration of the fuel rod, neptunium-237 target rod, medical and industrial radioisotope target rod, and control rod. **Figure E–2** shows a cross section of each type of fuel assembly in the core.

The new research reactor core design consists of 68 fuel assemblies, each of which is enclosed in a square aluminum shroud for structural support and coolant flow control. Key design features of the core are presented in **Table E–2**.

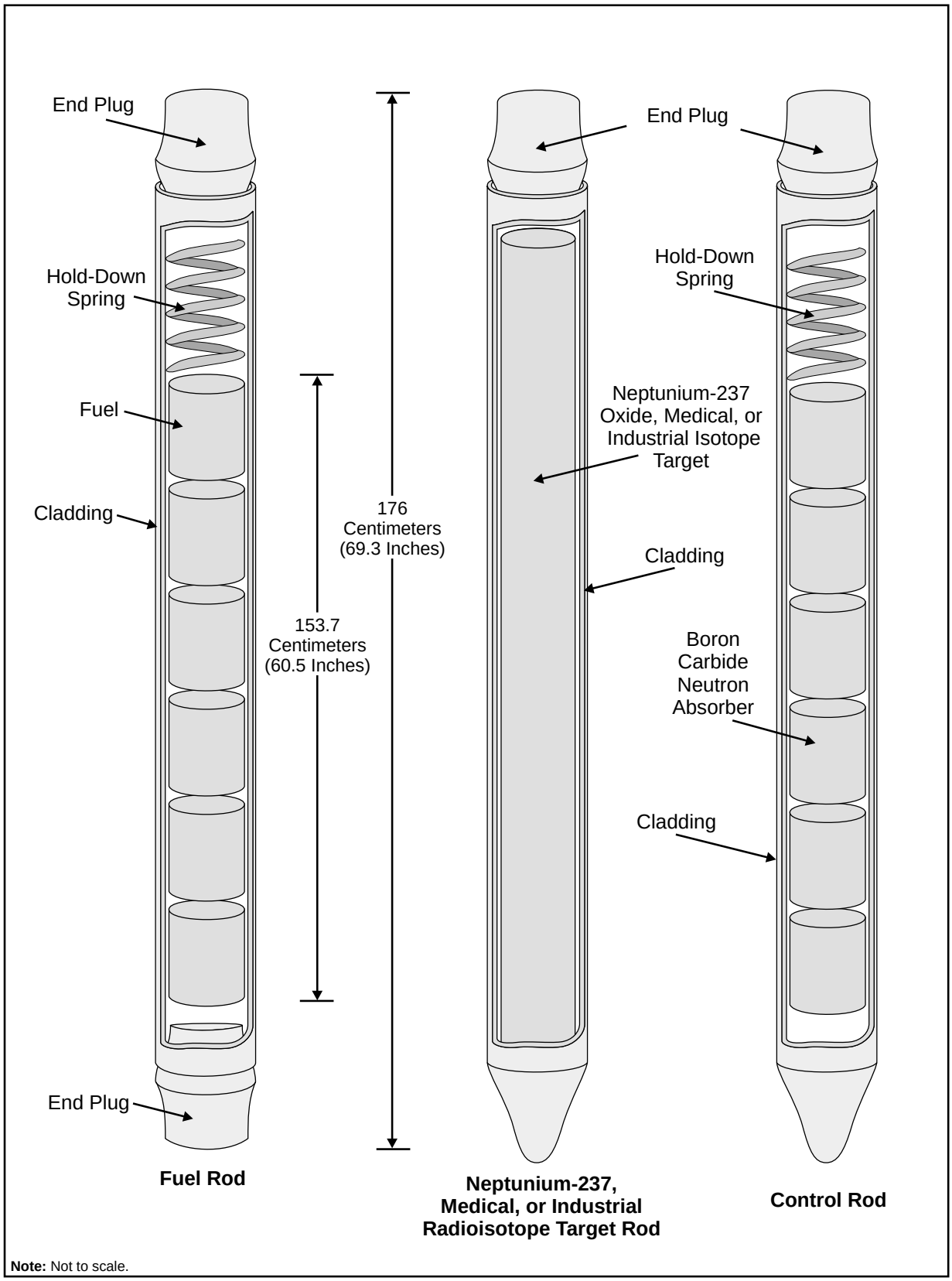


Figure E–1 Representative Illustration of Fuel Rod; Neptunium-237, Medical, or Industrial Radioisotope Target Rod; and Control Rod

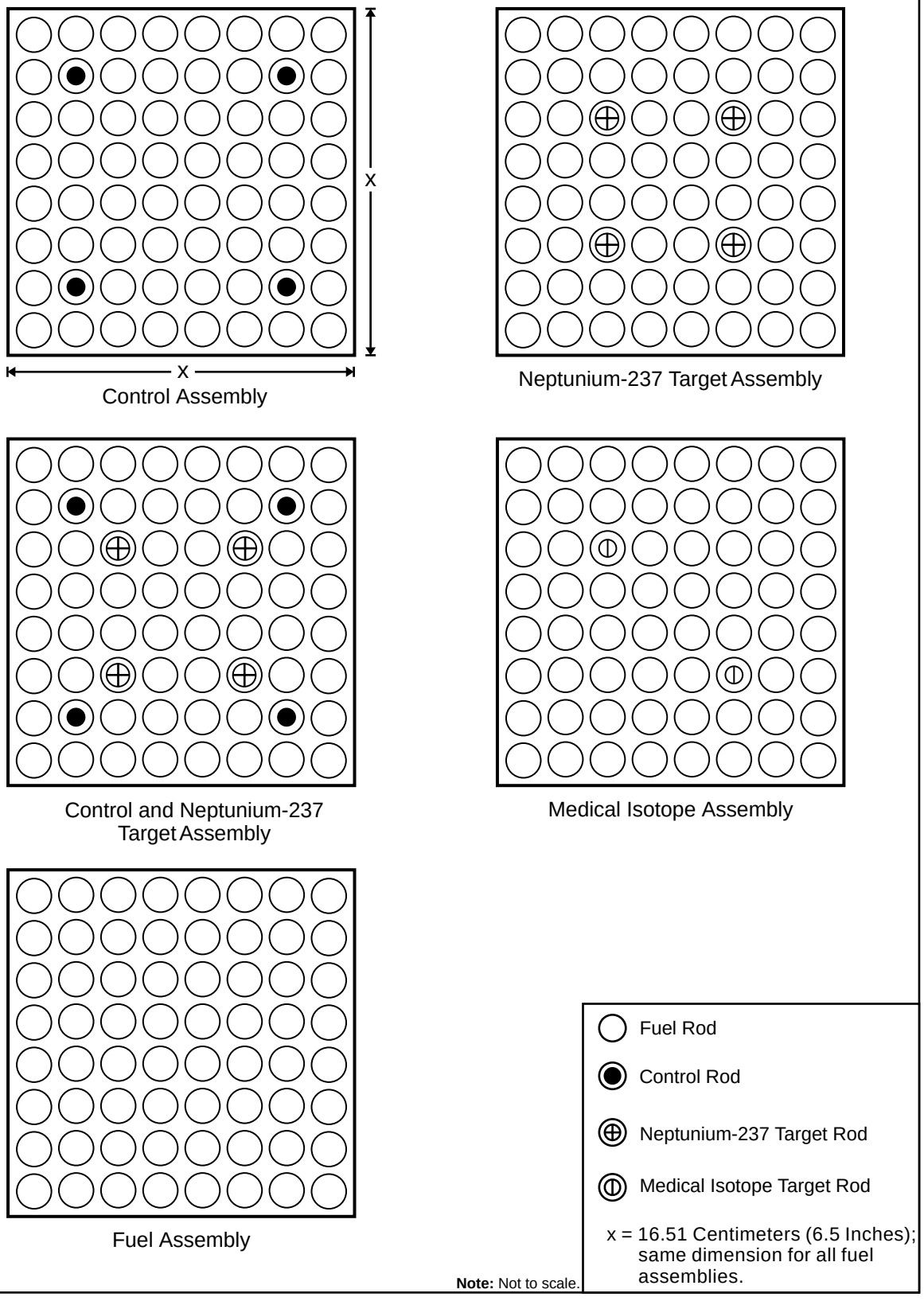


Figure E–2 Cross Section of Fuel Assemblies in the Core

Table E–2 Key Design Features of the New Research Reactor Core

Core Design Parameter	Value
Number of fuel assemblies	68
Core thermal power	50 megawatts
Average fuel assembly power	0.74 megawatt
Number of neptunium-237 target rod assemblies	48
Number of neptunium-237 target rods per assembly and in core	4 per assembly, 192 in core
Number of medical and industrial radioisotope target rod assemblies	8
Number of medical and industrial radioisotope target rods per assembly and in core	2 per assembly, 16 in core
Available radioisotope production volume	42.1 liters (1.5 cubic feet)
Number of control rod assemblies	16
Number of control rods per assembly and in core	4 per assembly, 64 in core
Total number of fuel rods in core	4,080
Core active height	153.7 centimeters (60.5 inches)
Core diameter	166.5 centimeters (65.5 inches)
Core radial reflector material and minimum thickness	Beryllium, 3.0 centimeters (1.18 inches)
Core uranium mass	3.1 metric tons of uranium (6,820 pounds)
Core uranium-235 mass	611 kilograms (1,344 pounds)
Minimum core life at 80 percent capacity factor	10 years

The core design described in Table E–2 also includes eight rabbit tubes for short irradiation-time production of medical or industrial radioisotopes and nuclear research and development. These rabbit tubes are located outside the fuel region of the core, but still within an area with a relatively high neutron flux. A cross-sectional view of the new research reactor core showing the layout of fuel assemblies, target rod assemblies, control rod assemblies, reflector, and rabbit tubes is presented in **Figure E–3**.

E.4 NUCLEAR FUEL THERMAL PERFORMANCE

The nuclear fuel design, based on core physics analyses, also was evaluated to determine if its thermal performance would meet the relevant thermal limits for TRIGA fuel. The steady-state thermal performance of the new research reactor nuclear fuel was analyzed to evaluate three indicators: (1) peak fuel pellet centerline temperature, (2) critical heat flux ratio (sometimes denoted as departure from nucleate boiling ratio), and (3) fuel rod internal volume gas pressure and associated cladding hoop stresses.

Peak fuel pellet temperature is an important variable for TRIGA fuel because, at elevated temperatures, hydrogen within the UZrH fuel matrix is released as a gas and can cause excessive pressure that may result in cladding rupture. A fuel temperature limit of 650 °C (1,202 °F) precludes excessive hydrogen gas release and pressure (Simnad 1980).

The critical heat flux ratio is a measure of the nature of the heat transfer from the fuel rod cladding surface to the coolant water flowing past it. As the cladding surface heat flux increases, heat transfer to the coolant increases until critical heat flux is reached. Beyond this critical heat flux, further increases in surface heat flux will not result in greater heat transfer to the coolant, thus causing the cladding and fuel temperature to rise rapidly. By maintaining the fuel cladding surface conditions below critical heat flux, a safe heat transfer regime exists between the fuel and the coolant. The critical heat flux ratio is the calculated critical heat flux, for a given set of peak core thermal-hydraulic conditions, to the actual cladding maximum surface heat flux.

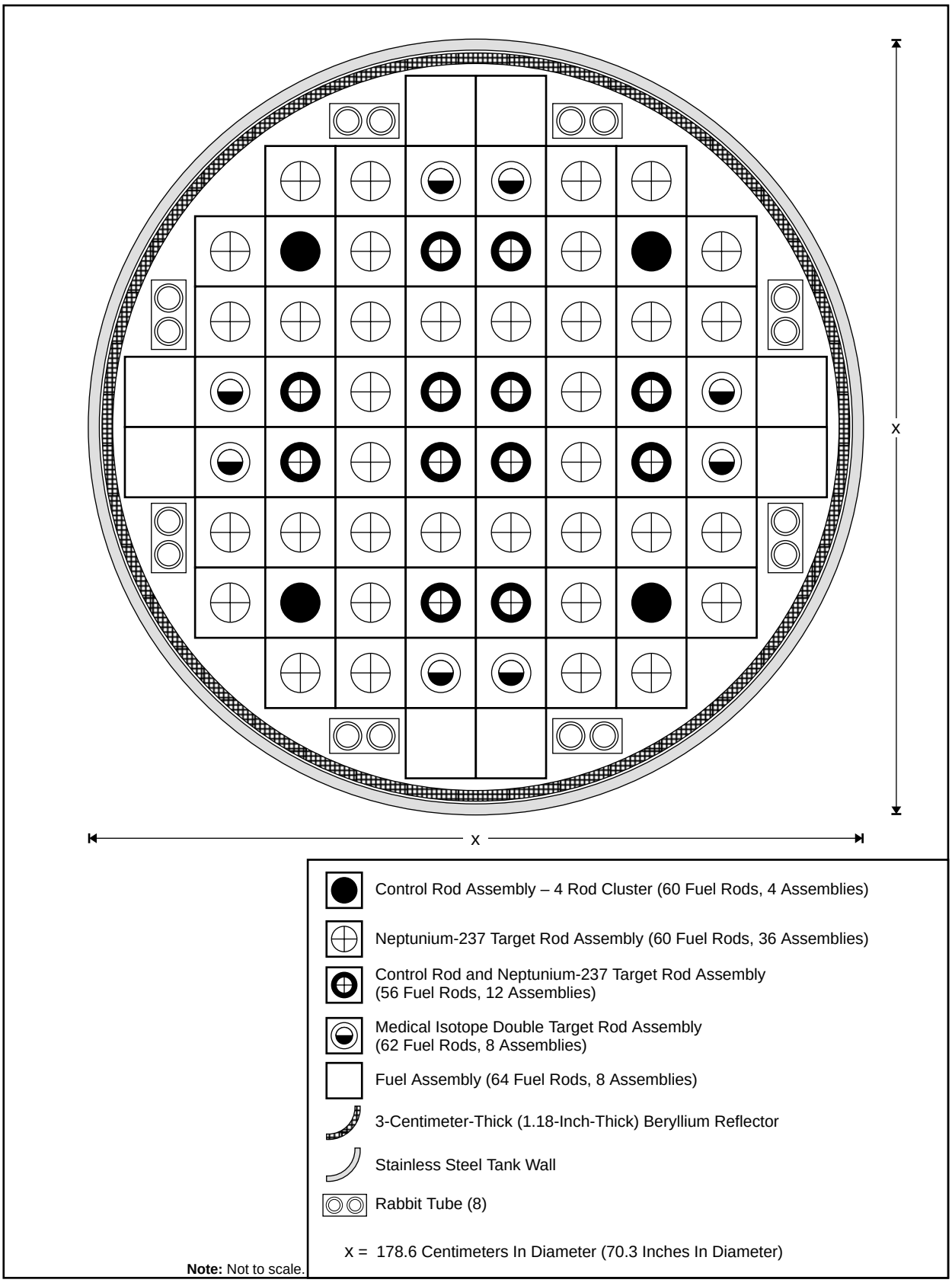


Figure E–3 Cross-Sectional View of Research Reactor Core

A critical heat flux ratio greater than 1 is an indicator of an acceptable thermal design. Typically, due to the statistical uncertainty and experimental conditions associated with maximum core heat flux and critical heat flux correlations, a ratio of greater than 1.2 is desirable.

The calculated peak fuel rod internal gas pressure and associated cladding hoop stress are an indication of the structural integrity of the cladding, which constitutes an important fission product confinement barrier. As the fuel is irradiated in the reactor core, gaseous fission products are produced. Some of these fission products can escape from the fuel pellets and collect in the gap between the fuel and cladding as well as the plenum volume above the fuel pellet stack inside the fuel rod. An increase in the presence of gaseous fission products inside the fuel rod volume results in a higher rod internal pressure and concomitant higher cladding hoop stress over the lifetime of the fuel. It should be noted that the UZrH fuel pellet material has been shown to retain most of its fission products at operating temperatures (Simnad 1980).

The radial temperature distribution of an axial segment in the average and peak fuel rod was calculated using the one-dimensional solution to the conservation of energy equation for a cylindrical geometry. Appropriate values for coolant and fuel rod material thermal-physical properties were used in this analysis. Bounding assumptions were made for boundary conditions and fuel rod peaking factor. **Table E–3** presents the thermal parameters and important results relevant to the analysis of fuel rod temperatures.

Table E–3 Fuel Rod Temperature Distribution Analysis Parameters for Steady-State Operation

Fuel Rod Temperature Parameter	Value
Core power density	14.9 kilowatts per liter
Average fuel rod linear heat generation rate	79.7 watts per centimeter (8,291 Btu/hr/ft)
Average fuel rod surface heat flux	18.43 watts/cm^2 (58,418 Btu/hr/ft^2)
Maximum fuel rod peaking factor	2.25
Average core coolant flow rate	1.1 meters per second (3.6 feet per second)
Core hottest channel inlet coolant temperature	46.1 °C (115 °F)
Average fuel channel axial coolant temperature rise	10.9 °C (19.7 °F)
Hottest fuel channel axial coolant temperature rise	24.4 °C (44 °F)
Cladding surface temperature Average fuel rod Peak fuel rod	 78.9 °C (174 °F) 113.1 °C (235.6 °F)
Fuel pellet centerline temperature Average fuel rod Peak fuel rod	 160 °C (320 °F) 285.7 °C (546.3 °F)
Fuel pellet normal operation temperature limit	650 °C (1,202 °F)
Minimum critical heat flux ratio Average fuel rod Peak fuel rod	 2.91 1.30
End-of-core-life fuel rod internal volume pressure Average fuel rod Peak fuel rod	 0.146 MPa (21.2 psia) 0.170 MPa (24.6 psia)
End-of-core-life fuel rod cladding hoop stress (external pressure = 0.101 Mpa [14.7 psia]) Average fuel rod Peak fuel rod	 0.855 MPa (124 psi) 1.303 MPa (189 psi)
Cladding yield strength	251.5 MPa (36,500 psi)
Cladding ultimate strength	900 MPa (130,500 psi)

Key: Btu/hr/ft^2, British thermal units per hour per square foot; MPa, megapascal; psi, pounds per square inch; psia, pounds per square inch absolute; watts/cm^2, watts per square centimeter.

Table E–3 shows that the maximum calculated fuel pellet centerline temperature, minimum critical heat flux ratio, and maximum fuel rod cladding stress are well within their relevant limits. This analysis demonstrates that the new research reactor fuel and core designs meet basic thermal design criteria with ample safety margins for steady-state operation at a 50 megawatt-thermal core power level.

E.5 NUCLEAR CORE PHYSICS PERFORMANCE

Nuclear core physics calculations were performed with three state-of-the-art digital computer codes and/or code packages: SCALE-4.4 (ORNL 1998), WIMSDB5 (ORNL 1999), and MCNP4B2 (Briesmeister 1998).

SCALE-4.4 uses discrete ordinate deterministic methods, one-dimensional unit cell geometry, transport theory, and point depletion to calculate neutron flux, k-infinity, and fission product inventory. K-infinity is a measure of neutron multiplication factor for a system of infinite size, which is an indication of reactivity to meet the mission during a core cycle of operation. A minimum value of about 1.1 for k-infinity is required to attain a critical state and produce neutrons for radioisotope production. SCALE-4.4 was used with a 44-energy-group cross-section database to calculate fission products and as an independent validation of k-infinity and neutron flux. WIMSD Version 5B uses a one-dimensional transport theory deterministic method with a lattice representation of the fuel assembly that accounts for fuel rod and assembly geometry and a 69-energy-group cross-section database to calculate neutron flux, k-infinity, and plutonium-238 production rate. WIMSD Version 5B was used to calculate plutonium-238 production rate and neutron flux and as an independent validation of the values of k-infinity. MCNP4B2 uses a three-dimensional monte carlo stochastic transport particle simulation of the full core geometry with continuous, point-wise neutron cross-section data to calculate the k-infinity, spatial neutron flux distribution in the core, and the neutron flux energy spectrum. MCNP4B2 was used to calculate peak and average core neutron flux, in terms of both energy and spatial distribution, and k-infinity.

The three computer codes predicted beginning of core life values of k-infinity to within about 1 percent of each other. Comparisons of calculated core neutron flux were also close among the different computer codes. In addition, a review of maximum thermal neutron flux for operating TRIGA research reactors with a core power from 1 to 14 megawatts showed values of 1×10^{13} to 3×10^{14} neutrons per square centimeter per second (IAEA 1989; ANL 2000). This range of neutron flux compares well with the calculated maximum value of 6×10^{13} neutrons per square centimeter per second for this new 50-megawatt TRIGA-based research reactor. Key core physics results are presented in **Table E-4.**

Table E–4 Key Core Physics Performance Parameters

Core Physics Parameter	Value
Beginning-of-life k-infinity	1.5
End-of-life k-infinity	1.1
Plutonium-238 production after 300-day operation	5.3 kilograms (11.7 pounds)
Average core thermal (less than 0.625 electron volt) neutron flux	2.5×10^{13} neutrons/cm^2/sec
Peak core thermal neutron flux	6×10^{13} neutrons/cm^2/sec
End-of-life fuel assembly Average burnup Maximum burnup	 2,292 megawatt-days 5,157 megawatt-days
Core average end-of-life uranium-235 atom burnup	34 percent

Key: neutrons/cm^2/sec, neutrons per square centimeter per second.

Table E–4 shows that the core is designed for a minimum useful life of approximately 10 years and would produce greater than 5 kilograms (11 pounds) of plutonium-238 during a 300-day operating period. It should be noted that limited core physics calculations were performed as part of this new research reactor preconceptual design. Detailed spatial full core depletion calculations were not performed for this reactor. The numerical core physics performance and plutonium-238 production values presented in Table E–4 are based on conservative calculations. The neptunium-237 target design was based on a limitless supply of neptunium-237. A significant reduction (by a factor of four to eight) in the required neptunium-237 target mass for the same annual plutonium-238 production rate can be achieved by: (1) optimizing target design with annular pellets, (2) inclusion of moderator materials mixed in the target, (3) optimizing fuel geometry, and (4) increasing core power.

E.6 PRIMARY COOLANT SYSTEM DESIGN

The major components of the primary coolant system design for the new research reactor are (1) a reactor core pool, (2) two primary coolant pumps, (3) two primary coolant heat exchangers, (4) shutdown pumps, and (5) piping between the pool, pumps, and heat exchangers. This system is designed to remove core thermal power during normal operation and core decay heat after reactor shutdown while maintaining the fuel below its thermal limits. A schematic of the primary coolant system is presented in **Figure E–4.**

Coolant flow would be drawn down through the core by the primary coolant pump suction pipelines, which would be connected to the bottom of the tank enclosing the core inside the pool. The side and bottom of the cylindrical tank would be sealed, but the top of the tank would be open to the pool. The forced coolant downflow through the core, as opposed to upflow, is designed to mitigate radiological consequences from the production of nitrogen-16, which is created from fast neutron reaction with the oxygen in the water molecule. Nitrogen-16 is a radioisotope which emits high-energy gamma (6 million electron volts per disintegration) radiation, but has a decay half-life of approximately 7 seconds. Therefore, the core coolant downflow allows this nitrogen-16, produced in the core coolant region, to decay before it returns to the reactor core pool. The primary coolant system loop (i.e., core to pump to heat exchanger to pool) is designed to delay the coolant removed from the core by at least 21 seconds (i.e., three half-lives) prior to its return to the pool. Appropriate shielding would be designed for all primary coolant system components to protect workers from nitrogen-16 gamma radiation to allow access during power operation.

Another design feature of the primary coolant system intended to reduce nitrogen-16 radiation at the top of the reactor core pool is the submerged location of the return flow from the heat exchanger. This submerged location allows shielding of the water depth above it and further nitrogen-16 decay before it can diffuse to the top of the pool.

The primary coolant system design has several inherently safe design elements that preclude or mitigate postulated accidents. As illustrated in Figure E–4, the suction piping from the bottom of the core tank would be routed, within the pool, to an elevation of 3 meters (10 feet) above the top of the core before it exits the pool through the pool wall and down to the pump in an adjacent compartment. A vacuum breaker (antisyphon device) would be attached to the high point of this suction piping inside the pool to prevent draining the reactor pool during any postulated pipe break accident. The balance of the primary coolant system (i.e., pump discharge piping, heat exchanger, and heat exchanger outlet piping) all would be elevated above the top of the core so that any leak or rupture could not uncover the core. This primary coolant system configuration precludes any leak from draining the pool below the top of the core.

The reactor core pool would be fully lined with 304 stainless steel, which would be attached to reinforced concrete. The reinforced concrete walls and floor of the pool are designed to meet all design-basis earthquake loads in the United States outside of coastal California.

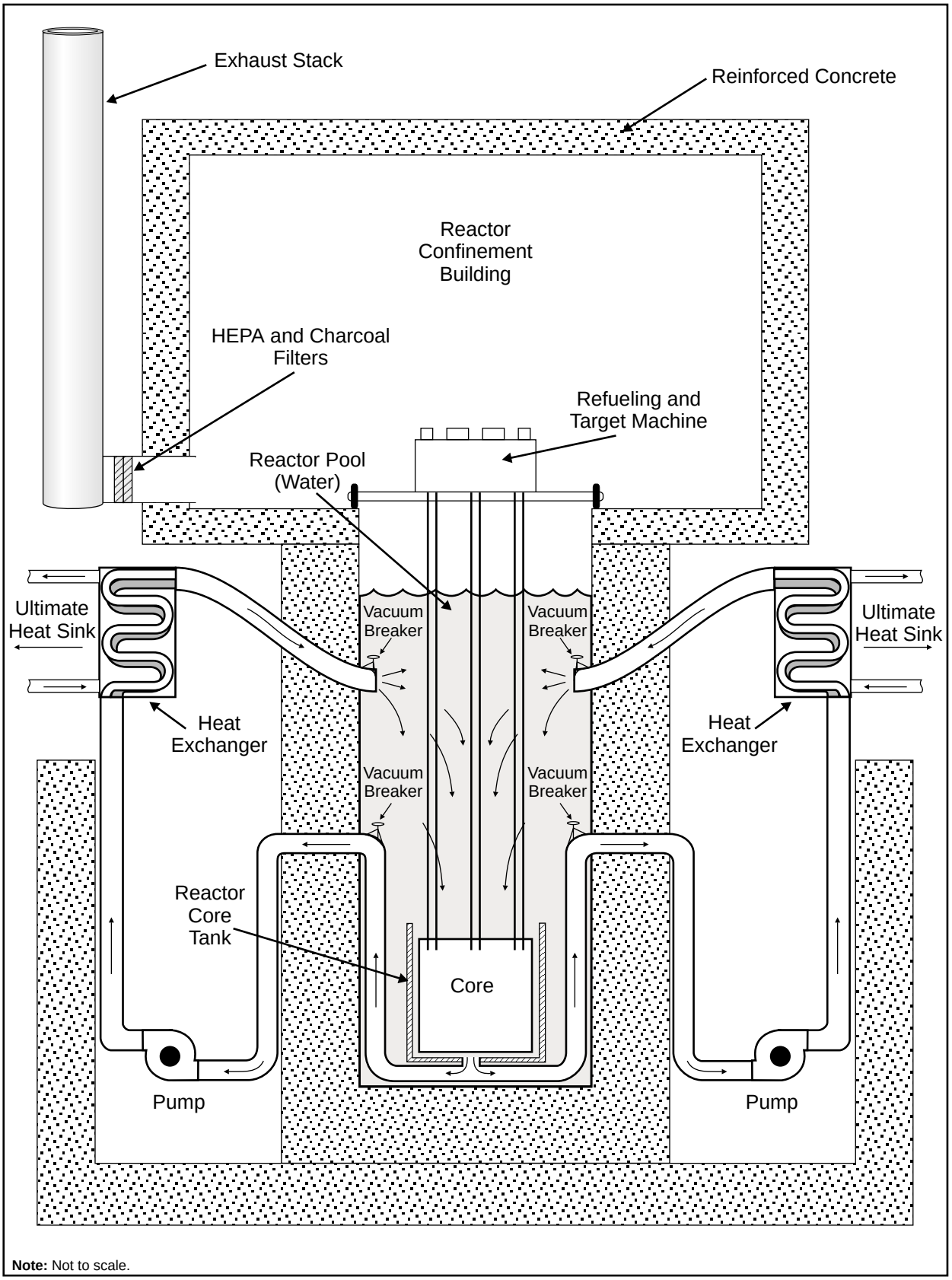


Figure E–4 Schematic of Primary Coolant System

All primary coolant system components would be type 304 stainless steel and would be manufactured to meet current nuclear safety and quality assurance standards. The use of two 100 percent capacity pumps and heat exchangers would allow for normal core power operation in the event of the loss of one component. A low flow-rate shutdown pump would be included in the system to provide sufficient flow for core decay heat removal. Key primary coolant system design parameters are presented in **Table E–5.**

Table E–5 Key Primary Coolant System Design Parameters

Design Parameter	Value
Reactor core pool water dimensions	7.9 meters (26 feet) long 6.7 meters (22 feet) wide 9.1 meters (30 feet) deep
Reactor core pool materials	304 stainless steel, 1.27-centimeter-thick (0.5-inch-thick) liner over reinforced concrete
Primary coolant system pump flow rate (100 percent capacity each)	1.262 cubic meters per second (20,000 gallons per minute)
Primary coolant system pump design head	61 meters (200 feet)
Primary coolant system heat exchanger design heat removal rate (100 percent capacity each)	50 megawatts (170×10^6 British thermal units per hour)
Primary coolant system heat exchanger design inlet temperature	51.7 °C (125 °F)
Primary coolant system heat exchanger design outlet temperature	40.6 °C (105 °F)
Primary coolant system shutdown pump flow rate	0.063 cubic meters per second (1,000 gallons per minute)
Primary coolant system shutdown pump design head	61 meters (200 feet)

The design of the primary coolant system employs accepted and widely used nuclear power plant safety principles such as redundancy, single-failure-proof, nuclear quality assurance, and inherent natural phenomena (e.g., elevated piping).

E.7 BALANCE OF REACTOR PLANT SYSTEMS

Due to the preconceptual design nature of the new research reactor, limited details have been developed for the balance of the reactor plant systems. Details were established where they were judged to significantly affect the determination of environmental impacts or cost.

The secondary cooling system design transfers the heat removed from the core by the primary coolant system to the environment. This system would consist of two 100-percent-capacity pumps and two 50-percent-capacity cooling towers with piping connecting them to each other and to the secondary side of the primary coolant system heat exchangers. Unlike the primary coolant system, the secondary coolant system is not designed or considered to be a nuclear-safety-related system because its failure would not challenge the safety of the reactor. Piping and cooling tower tubes would be constructed of carbon steel. To avoid causing fogging at the reactor building area, the cooling towers would be located about 122 meters (400 feet) from the reactor building.

The spent fuel pool is designed to the same standards and with the same materials as the reactor core pool. The spent fuel pool would be sized to store all the fuel expected to be discharged over the 35-year lifetime of the reactor. The spent fuel pool also is designed to accommodate shipping casks for transport of radioisotope target rods and the spent nuclear fuel after this reactor is shut down, decontaminated, and decommissioned. The reactor core pool can be hydraulically connected to the spent fuel pool using an isolable transfer canal for moving spent nuclear fuel from and reloading the core, as well as transferring isotope target rods from the core

to the spent fuel pool. An important safety feature of this canal is its ability to flood the reactor core pool in the unlikely event of a coolant leak from the reactor core pool. The size of the spent fuel pool allows it to completely reflood the entire volume of the reactor core pool, using the transfer canal, while maintaining a sufficiently high level of water above the stored spent fuel to preclude high dose rates in the spent fuel pool area. Another unique safety feature of the spent fuel pool is the design of its fuel storage racks. These racks are designed to separate each fuel assembly from other fuel assemblies by approximately 23 centimeters (9 inches), which neutronically isolates each assembly, thereby assuring criticality safety without the use of soluble or fixed neutron absorbers. The space between fuel assemblies in the rack is covered with steel bars that prevent any accidental insertion of a fuel assembly. In addition, each storage position is recessed more than 23 centimeters (9 inches) below the top of the rack to avoid a dropped assembly neutronically interacting with another assembly. This simple storage rack design provides inherent criticality safety in a cost-effective design.

A spent fuel pool cooling system is incorporated into the facility design. This nuclear-safety-released system would be capable of maintaining the spent fuel pool temperature within acceptable limits under all modes of plant operation. This system would consist of two redundant small pumps, heat exchangers, and appropriate instrumentation. Spent nuclear fuel decay heat removed from the spent fuel pool would be transferred to the secondary cooling system.

A water makeup and purification system would be included in the reactor design for maintaining the water level in both the reactor core and spent fuel pools and removing any contamination. This system, consisting of a small pump, piping, and appropriate filters, resin beds, and makeup water tank, would also be used to maintain the chemistry of the water to within technical specification limits. Periodic monitoring of pool water quality would be used as an indicator of fuel failure or heat exchanger tube leakage.

The reactor instrumentation and control systems are expected to be similar to other higher-power TRIGA reactors and are nuclear safety related. Instrumentation would monitor important nuclear and thermal-hydraulic parameters with digital displays in the control room. Although TRIGA fuel has an inherently large negative temperature reactivity coefficient, a reactor trip system would scram (loss of power would cause the control rods to drop in the reactor core) the reactor on a number of redundant signals such as high power, low pool level, low coolant flow rate, and high core exit coolant temperature.

In the unlikely event of a loss of all offsite alternating current power, a nuclear safety-related emergency power system consisting of two redundant 1,506-kilowatt emergency diesel generators was included in the design. These emergency diesel generators would be nuclear safety grade and subject to periodic testing to ensure their reliability.

The reactor building heating, ventilating, and air conditioning system is designed to maintain the air temperature within the building within specified limits while removing contaminants and certain radioisotopes that may be present in the building air. This system would consist of an interconnected network of ducts, fans, chillers, heating coils, filters, and a 36.6-meter-high (120-foot-high) exhaust stack. High-efficiency particulate air (HEPA) and charcoal filters would remove a minimum of 99.9 percent of airborne particulates and 99 percent of airborne iodine. The exhaust portion of the heating, ventilating, and air conditioning system, which would perform the contaminant and radioisotope removal function, would be safety-related.

The reactor building complex and associated structures are schematically presented in **Figures E–5 and E–6.** The reactor building would consist of three sections, separated by a radiation shield wall: (1) reactor room, (2) system room, and (3) spent fuel pool. The reactor room would house the reactor core pool and provide confinement. The reactor system section would include: (1) primary coolant system pumps and a shutdown pump; (2) primary coolant system heat exchangers; (3) secondary coolant system pumps; (4) a water makeup

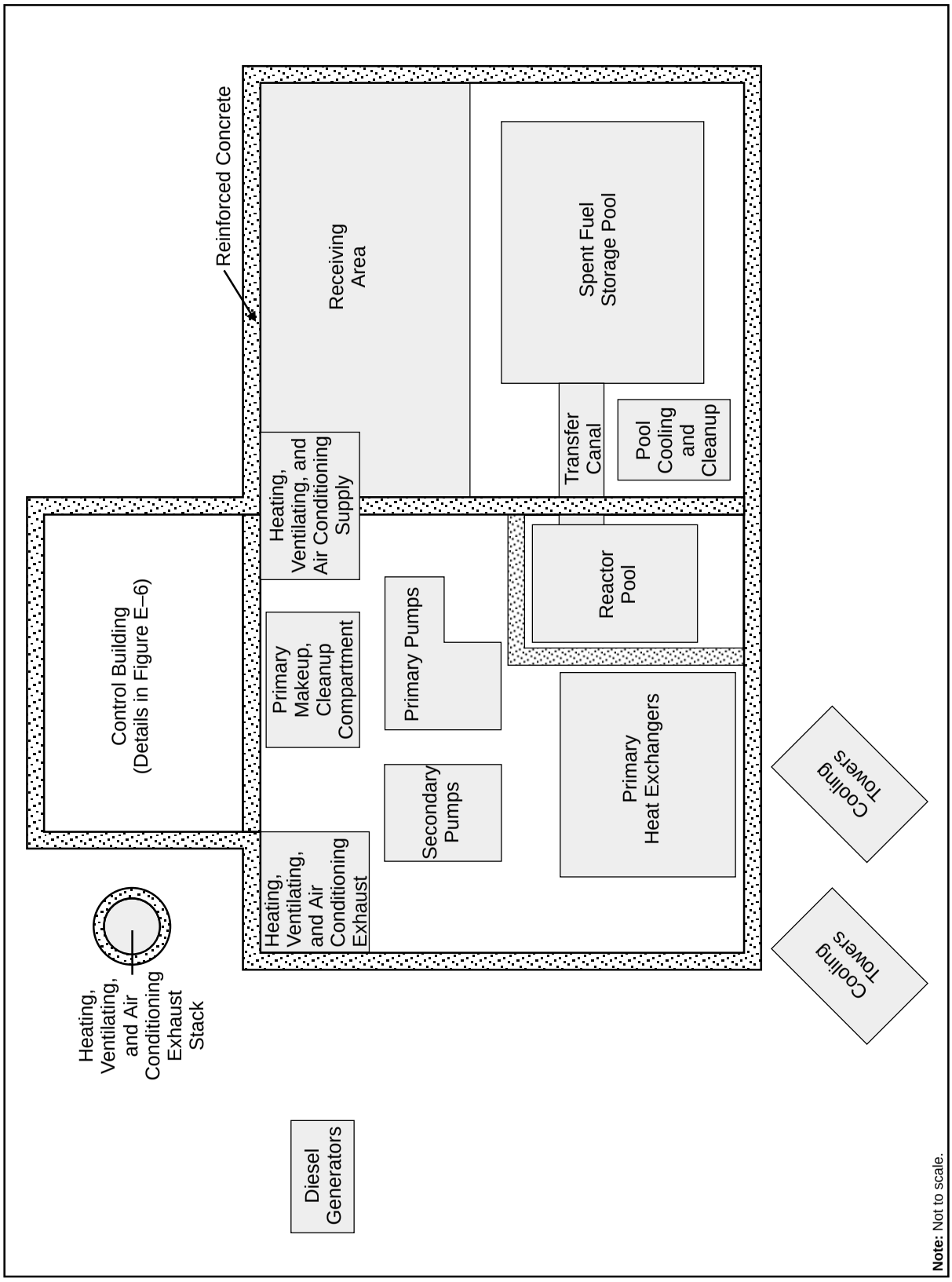


Figure E–5 Schematic of Reactor Building Complex

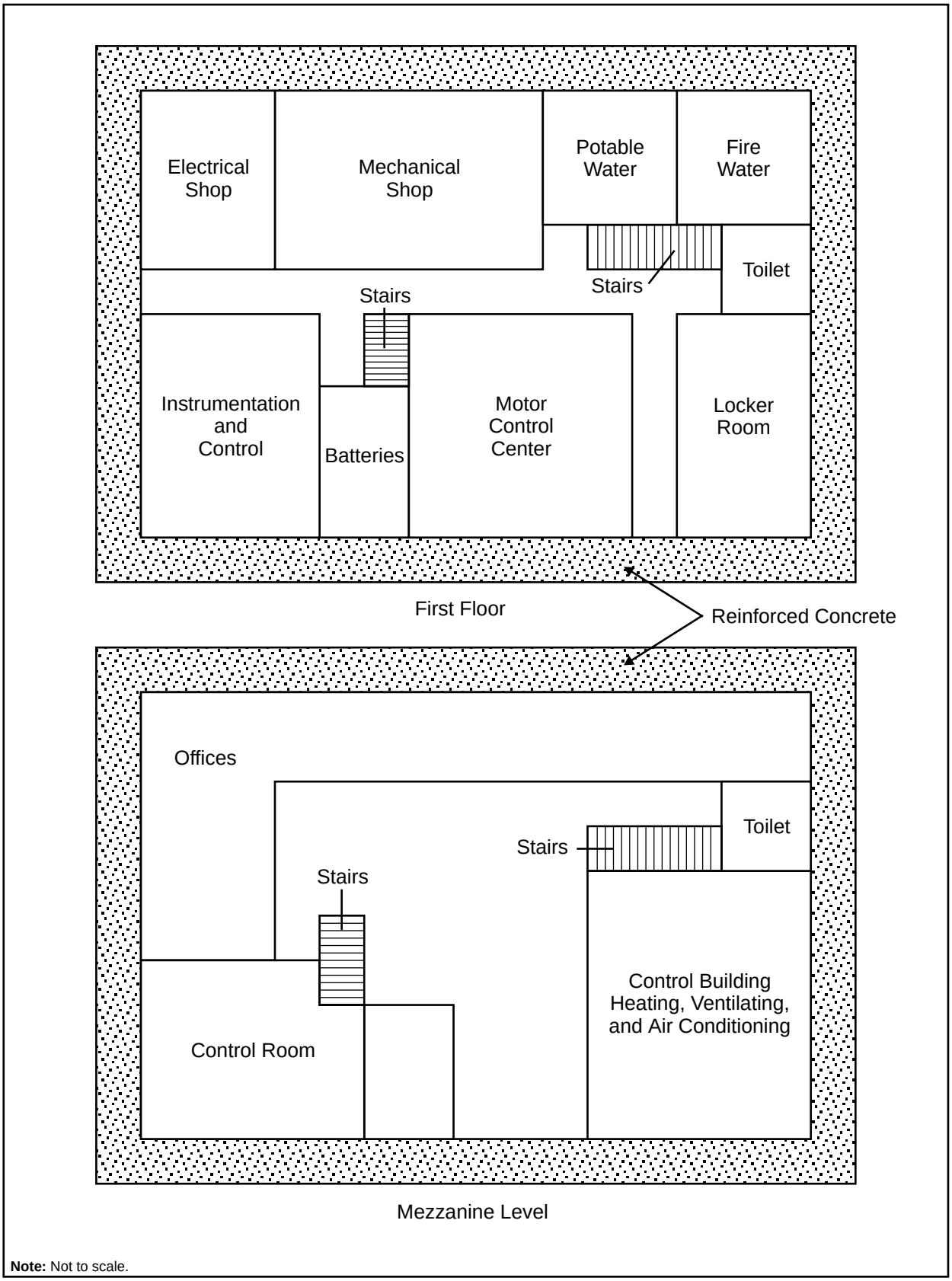


Figure E–6 Schematic of Control Building

and purification system; and (5) a nuclear-safety-related, single-failure-proof overhead crane. The spent fuel pool section would include: (1) a spent fuel pool; (2) a spent fuel pool makeup and purification system; (3) a pool transfer canal; (4) a loading dock; and (5) a nuclear-safety-related, single-failure-proof overhead crane capable of lifting a truck-sized spent nuclear fuel cask. The exterior walls and roof of the reactor building, consisting of reinforced concrete, are designed to withstand design-basis tornado missiles and seismic events for the entire United States outside of coastal California in accordance with current DOE and NRC regulations. The total footprint of the facility, including the reactor and control buildings, cooling towers, emergency diesel generators, exhaust stack, and ancillary structures, is estimated to be approximately 3,623 square meters (39,000 square feet, or about 1 acre). The total land area encompassing the facility is estimated to be approximately 4 acres.

E.8 REACTOR DESIGN SAFETY FEATURES

The new research reactor was designed with numerous inherent and passive safety features which prevent or mitigate the consequences of abnormal operational occurrences, off-normal events, and postulated accidents. The basic design constitutes an extrapolation of existing higher-power TRIGA reactors in the United States and in foreign nations. Over 6,000 TRIGA fuel elements have been fabricated and irradiated in research reactors, some with over 20 years of operation. Low-enriched uranium TRIGA fuel has been subjected to over 25,000 power pulses by the vendor with peak fuel temperatures of up to 1,150 °C (2,102 °F) without fuel damage (Simnad 1980).

The primary radiological source for this reactor is the fission products which are produced in the nuclear fuel pellets. The reactor's unique UZrH TRIGA fuel alloy has been experimentally shown to retain approximately 99.995 percent of all fission products at or below an operating temperature of 400 °C (752 °F). As presented in Table E–3, the peak and average fuel pellet centerline temperatures are less than 300 °C. The TRIGA fuel pellet also has been shown to exhibit no significant corrosion or chemical reactivity if exposed to water, steam, or air at temperatures up to 600 °C (1,112 °F). In addition, TRIGA fuel has been successfully irradiated up to a burnup of 75 percent of the available uranium-235 with no fuel damage (Simnad 1980).

The second confinement barrier for fission products is the fuel rod cladding, Incoloy-800, which can operate at much higher temperatures than aluminum and zircaloy cladding, does not oxidize at higher temperatures like zircaloy, and has a higher tensile and ultimate strength than aluminum, zircaloy, and stainless steel. Thus, the Incoloy-800 cladding provides a high degree of confinement integrity.

A third important fuel safety design feature is the inherently large negative temperature coefficient of reactivity for this fuel because it contains the hydrogen moderator intimately mixed within the solid fuel pellets. An increase in fuel temperature causes the fuel to expand, releases some of its hydrogen into the rod inner gas volume, and reduces the fuel hydrogen density. This reduction in hydrogen density shuts down the fission process and reduces power. As the fuel cools down, most of the hydrogen is reabsorbed into the fuel matrix. TRIGA fuel has always been designed to withstand sudden large power pulses and shut itself down. This pulsing feature is integral in the operation of many TRIGA research reactors worldwide.

The submerged configuration (i.e., under more than 6.1 meters [20 feet] of water) of fuel in both the reactor core and spent fuel storage pool provides another radioisotope removal mechanism if a fuel rod leak should occur. Such large depths of water absorb or retain 100 percent of all released solid fission products and over 99 percent of released halogen fission products.

The primary coolant system is designed with the following safety aspects: (1) a low system pressure (i.e., maximum less than 0.345 megapascal [50 pounds per square inch]); (2) low coolant operating temperatures (i.e., maximum hot leg temperature less than 65.6 °C [150 °F]); (3) 304 stainless steel primary

coolant piping routed inside the pool to an elevation about 3 meters (10 feet) above the core before exiting the pool to the pumps; (4) a 304 stainless-steel-lined, seismically qualified reactor core pool structure; (5) primary coolant downflow through the core; (6) a long (i.e., minimum of 21 seconds) primary loop coolant transit time; and (7) a large reactor pool coolant inventory.

The low pressures and temperatures indicate that a primary coolant system pipe break would not release significant quantities of energy into the reactor building. The expected low pressures in the reactor building, coupled with the tornado missile and seismic design of this reinforced concrete structure, provide a high degree of assurance that it would maintain solid, liquid, and removable non-noble gas radioisotope confinement for all postulated accidents.

The likelihood of a primary coolant system pipe leak or break would be extremely small based on the low pressure and temperature coolant conditions and the use of 304 stainless steel that would be designed, procured, supported, and installed in accordance with all current nuclear safety standards. In addition, the elevated pipe routing above the core inside the reactor pool, in conjunction with high point vacuum breakers, would ensure that any pipe leak or break would not drain the pool below about 3 meters (10 feet) above the core. At this pool level, the remaining water would not reach saturation, with core decay heat, for at least 4 days and the water level would not boil down to the top of the core for more than 40 days. This long period for recovery of a coolant source is indicative of the large thermal heat capacity and safety margins inherent in the pool design.

As discussed earlier, the core coolant downflow direction, along with a minimum transit time of 21 seconds for the coolant to return to the pool, was selected to reduce the nitrogen-16, which is produced by neutron absorption in the oxygen component of water molecules as it passes through the core. Nitrogen-16 is a high-energy gamma-emitting radioisotope, but it has a short half-life of approximately 7 seconds. Therefore, the downflow direction and three-half-life decay delay would reduce the activity of nitrogen-16 before it returns to the pool by about a factor of 10, thereby reducing the dose rates to workers during reactor operation.

Use of the spent fuel storage pool as a source of reactor pool water is another safety enhancement of this design. The spent fuel storage pool can be hydraulically connected to the reactor pool through a transfer canal by automatically or manually opening a valve. In the extremely unlikely event that the reactor pool would lose coolant, the spent fuel storage pool is sized so that it can completely reflood the entire reactor pool volume without compromising the decay heat removal and shielding design of the spent fuel pool. Also, the higher relative elevation of the spent fuel storage pool would allow it to reflood the reactor pool by gravity-driven flow requiring no pumps. The spent fuel storage pool can be used to reflood the reactor pool without any electrical power.

The reactor core and spent fuel storage pool coolant purification system, designed to nuclear safety standards, would remove radioisotopes present in the water. In addition, the nuclear safety reactor building heating, ventilating, and air conditioning exhaust system is designed to remove 99.9 percent of airborne radioisotope particulates and 99 percent of airborne radioisotope halogens. A 36.6-meter-tall (120-foot-tall) heating, ventilating, and air conditioning exhaust stack would provide optimum atmospheric dispersion to reduce the environmental impacts of any released materials after filtration.

Along with the many natural and passive aforementioned safety features, the reactor would be equipped with a reactor instrumentation and protection system that would trip the core under specific monitored conditions including, but not limited to (1) high core power, (2) low primary coolant flow rate, (3) loss of offsite power, (4) high core exit or hot leg temperature, (5) low reactor pool level, and (6) core power distribution beyond technical specification limits. A reactor primary coolant shutdown pump is designed to circulate sufficient coolant through the heat exchanger to remove decay heat. Two 100-percent-capacity redundant emergency

power diesel generators would provide adequate electrical power for all emergency systems in the event of a loss of offsite power.

The new research reactor design provides multiple layers of inherent passive and redundant nuclear-safety-related active systems to preclude, mitigate, and control any radioisotope releases to the environment and to minimize doses to both the public and workers.

E.9 REACTOR OPERATION

Operation of the new research reactor would be similar to other research reactors except that the core would be maintained at full power for a minimum of 80 percent of the year. At the beginning of a cycle of operation, neptunium-237 and long irradiation-time medical radioisotope target rod assemblies would be inserted into their appropriate fuel assembly sleeve locations. The target rods would be mechanically attached to a cluster spider assembly similar to that used for the control rod assembly. The neptunium-237 target rod assemblies would remain in the core for the entire annual fuel cycle. These target rod assemblies would be removed from the host fuel assembly without removing the fuel assembly from the core and then transferred to the spent fuel storage pool using the transfer canal.

Medical and/or industrial radioisotope target rods that require a 100-day irradiation cycle would be removed and replaced with new target rod assemblies during brief reactor shutdown periods. These target rod assemblies would be removed and transferred in a manner similar to that of the neptunium-237 target rod assemblies. Short irradiation-time radioisotopes would be inserted into rabbit tubes for the 10- to 25-day required time period. The eight rabbit tubes would be located outside the core, but inside the reflector region. The insertion and removal of irradiation targets in the rabbit tubes would have no significant effect on core reactivity and would not affect power operation.

After a radioisotope-specific cooling time in the spent fuel pool, neptunium-237, medical, and industrial radioisotope target assemblies would be transferred to a shipping cask in the spent fuel storage pool. Using the overhead crane in the spent fuel pool area, shipping casks would be placed onto a truck in the reactor building bay area adjacent to the fuel storage pool for shipment to the processing facility. New targets would be shipped from the target preparation facility into the reactor building bay by truck, transferred into the spent fuel storage pool, and subsequently moved to the reactor core pool or rabbit tube area for insertion into the core.

The plutonium-238 net annual production mission of 5 kilograms (11 pounds) was calculated to be achieved with a 300-day annual irradiation time, which corresponds to a capacity factor of approximately 80 percent. The redundant heat removal systems, low pressure and temperature conditions, and proven reactor design are expected to ensure this capacity factor. An annual shutdown for maintenance would occur during the remaining time of the year. The 10-year core refueling is not expected to affect the 80 percent annual capacity factor. Key reactor annual resource requirements are delineated in **Table E–6**.

The annual water consumption shown in Table E–6 would be due primarily to water losses from the cooling towers which would constitute over 99 percent of the total water use. Diesel fuel consumption would be due to the monthly and annual testing of the two emergency power diesel generators. Sewer water disposal would be due to the potable and sanitary water use by the research reactor facility staff.

Table E–6 Research Reactor Annual Resource Requirements

Resource Parameter	Value
Staff	120
Electricity	25,000 megawatt-hours
Reactor operating process water	7.95×10^{8} liters (2.10×10^{8} gallons)
Total water	8.07×10^{8} liters (2.13×10^{8} gallons)
Nonhazardous waste	250 cubic meters (327 cubic yards)
Hazardous waste	4 cubic meters (5.2 cubic yards)[a]
Diesel fuel	28,972 liters (7,655 gallons)
Potable and sanitary water	1.16×10^{7} liters (3.06×10^{6} gallons)

a. DOE 2000.

E.9.1 Nonradiological Emissions

During normal operations, the nonradiological emissions from the new research reactor facility would consist primarily of exhaust from testing, assumed to be a total of 72 hours per year, of the two emergency power diesel generators. These emission data were based on the estimated annual consumption of diesel fuel associated with this testing and U.S. Environmental Protection Agency (EPA) guidance on diesel engine emissions (EPA 1996). The estimated annual diesel emissions for the new research reactor are presented in **Tables E–7, E–8, E–9, and E–10**.

Table E–7 Gaseous Emission Factors and Predicted Emissions

Criteria Pollutant	Emission Factor (pounds per million British thermal units)[a]	Predicted Emissions from Subject Diesels (kilograms [pounds] per year)
Nitrogen oxides		
Uncontrolled	3.2	1,493 (3,290)
Controlled	1.9[b]	885 (1,950)
Carbon monoxide	0.85	395 (870)
Carbon dioxide	165	77,163 (170,000)
Sulfur dioxide	$1.01\times S$[c]	472 (1,040)

a. EPA 1996.
b. Controlled by timing ignition retard.
c. S is the percent of sulfur in the fuel, which is assumed to be 1 percent (No. 2 Diesel Fuel Oil) (Avallone and Baumeister 1987).

Table E–8 Emissions Factors and Predicted Emissions for Particulate Matter

Description of Particulates	Emission Factor (pounds per million British thermal units)[a]	Predicted Emissions from Subject Diesels (kilograms [pounds] per year)
Filterable particulates[b]		
Less than 1 micron	0.0478	22.3 (49.1)
Less than 3 microns	0.0479	22.3 (49.2)
Less than 10 microns	0.0496	23.1 (51.0)
Total filterable particulates	0.0620	28.9 (63.7)
Condensable particulates	0.0077	3.6 (7.9)
Total PM_{10}[c]	0.0573	26.7 (58.9)
Total particulates[d]	0.0697	32.5 (71.6)

a. EPA 1996.
b. Particle size is expressed as aerodynamic diameter.
c. Total PM_{10} is the sum of the filterable particulate matter with an aerodynamic diameter less than or equal to 10 microns and the condensable particulate.
d. Total particulates are the sum of the total filterable particulates and the condensable particulates.

Table E–9 Emission Factors and Predicted Emissions for Speciated Organic Compounds

Pollutant	Emission Factor (pounds per million British thermal units)[a]	Predicted Emissions from Subject Diesels (kilograms [pounds] per year)
Benzene	7.76×10^{-4}	0.36 (0.80)
Toluene	2.81×10^{-4}	0.13 (0.29)
Xylenes	1.93×10-4	0.09 (0.20)
Propylene	2.79×10^{-3}	1.30 (2.87)
Formaldehyde	7.89×10^{-5}	0.04 (0.08)
Acetaldehyde	2.52×10^{-5}	0.01 (0.03)
Acrolein	7.88×10^{-6}	0.004 (0.008)

a. EPA 1996.

Table E–10 Emission Factors and Predicted Emissions for Polyaromatic Hydrocarbons

Pollutant	Emission Factor (pounds per million British thermal units)[a]	Predicted Emissions from Subject Diesels (kilograms [pounds] per year)
Naphthalene	1.30×10^{-4}	0.060 (0.133)
Acenaphthylene	9.23×10^{-6}	0.0043 (0.0095)
Acenaphthene	4.68×10^{-6}	0.0022 (0.0048)
Fluorene	1.28×10^{-5}	0.0059 (0.013)
Phenanthrene	4.08×10^{-5}	0.019 (0.042)
Anthracene	1.23×10^{-6}	0.0006 (0.0013)
Fluoranthrene	4.03×10^{-6}	0.0019 (0.0041)
Pyrene	3.71×10^{-6}	0.0017 (0.0038)
Benz(a)anthracene	6.22×10^{-7}	0.00029 (0.00063)
Chrysene	1.53×10^{-6}	0.00073 (0.0016)
Benzo(b)fluoranthene	1.11×10^{-6}	0.00050 (0.0011)
Benzo(k)fluoranthene	Less than 2.18×10^{-7}	Less than 0.00010 (less than 0.00022)
Benzo(a)pyrene	Less than 2.57×10^{-7}	Less than 0.00012 (less than 0.00026)
Indeno (1, 2, 3-cd) pyrene	Less than 4.14×10^{-7}	Less than 0.00020 (less than 0.00043)
Dibenz (a, h) anthracene	Less than 3.46×10^{-7}	Less than 0.00016 (less than 0.00036)
Benzo (g, h, l) perylene	Less than 5.56×10^{-7}	Less than 0.00026 (less than 0.00057)
Total polyaromatic hydrocarbons	2.12×10^{-4}	0.100 (0.220)

a. EPA 1996.

E.9.2 Radiological Emissions

Radiological emissions from the new research reactor during normal operations would be due to the neutron activation of argon gas, which would be dissolved in the reactor pool water, creating argon-41 and neutron capture by oxygen atoms in water molecules, creating tritium. **Table E–11** presents the calculated annual emissions of radioisotopes from the reactor due to normal operations and the estimated maximum annual radioactive waste generation (DOE 2000; AECL 1996).

Table E–11 Normal Operations Annual Radiological Emissions and Waste Generation

Radioisotope	Annual Release
Argon-41	2.8 curies
Tritium (hydrogen-3)	0.1 curies
Low-level liquid radioactive waste volume	<6 cubic meters (212 cubic feet)[a]
Low-level solid radioactive waste volume	50 cubic meters (1,766 cubic feet)[b]
Transuranic waste	0
Mixed low-level radioactive waste	<0.5 cubic meter[a] (17.7 cubic feet)

a. DOE 2000.
b. AECL 1996.
Key: <, less than.

The maximum dose rate to workers at all locations within the reactor building, due to released argon-41, tritium, and direct radiation from the submerged reactor core at power or the spent fuel in the storage pool is estimated to be less than 1 millirem per hour.

E.10 REACTOR CONSTRUCTION

Construction of the new research reactor facility was determined to require 4 years after design and licensing activities have been completed (ANSTO 1999; AECL 1996). Based on the dimensions of the reactor and control buildings, cooling towers, cooling tower separation distance from the reactor and control buildings, and ancillary structures (i.e., emergency power diesel generators, heating, ventilating, and air conditioning exhaust stack, etc.), the total surface area of structures and of the facility restricted area were calculated and are presented in **Table E–12.** This table also presents the total construction workforce; quantities of earth moved; and quantities of concrete, structural, and stainless steel estimated for the facility. It was assumed that the new research reactor would be located at an existing DOE site. Another underlying assumption is that the workforce would be present for 2,080 hours per year at the construction site.

Table E–12 Research Reactor Construction Resources

Construction Parameter	Value
Time period	4 years
Workforce	160[a]
Total reactor facility area	3,623 square meters, or 0.9 acre (39,000 square feet)
Reactor restricted area	15,942 square meters, or 3.9 acres (171,600 square feet)
Volume of earth moved	5,199 cubic meters (6,800 cubic yards)[b]
Concrete volume	5,237 cubic meters (6,850 cubic yards)[b]
Mass of structural steel	50,122 kilograms (110,500 pounds)[b]
Mass of stainless steel	3,468 kilograms (7,645 pounds)[b]
Volume of potable and sanitary water used	4.4×10^7 liters (1.2×10^7 gallons)
Total volume of water used	4.7×10^7 liters (1.24×10^7 gallons)

a. AECL 1996.
b. Tripathi 2000a, 2000b, 2000c.

During construction, pollutant emissions would be generated by vehicle operation, onsite concrete batch plant operation, wind erosion, material handling, bulldozing, scraping, and grading operations. The annual emissions and waste generation for the 4-year construction period were estimated based on the workforce size, concrete production requirements, and facility areas (PFS 1997), and are presented in **Table E–13**.

Table E–13 Annual Reactor Facility Construction Emissions and Waste Generation

Criteria Pollutant	Annual Emissions
PM_{10}[a]	14,279 kilograms (31,414 pounds)
Nitrogen oxides	5,370 kilograms (11,815 pounds)
Carbon monoxide	6,713 kilograms (14,769 pounds)
Volatile organic compounds	1,343 kilograms (2,954 pounds)
Structural steel scrap waste	1,253 kilograms (2,757 pounds)[b]
Stainless steel scrap waste	87 kilograms (191 pounds)[b]
Concrete waste	131 cubic meters (171 cubic yards)[b]
Hazardous liquid waste	0.25 cubic meter (0.3 cubic yard)
Hazardous solid waste	0.75 cubic meter (1.0 cubic yards)
Nonhazardous liquid waste	11,400 cubic meters (14,387 cubic yards)
Nonhazardous solid waste	307,500 kilograms (676,500 pounds)

a. Total PM_{10} is the sum of the filterable particulate matter with an aerodynamic diameter less than or equal to 10 microns and the condensable particulate.
b. Tripathi 2000a, 2000b, 2000c.

E.11 DECONTAMINATION AND DECOMMISSIONING

When the new research reactor ceases its operation, it would be subject to the process of decommissioning. The reactor and its facility would be decontaminated to acceptable levels approved by the regulatory authority such that the land and buildings can be released for unrestricted uses. The research reactor and its facility then would be delicensed. For the decommissioning of a research reactor, decontamination and release for unrestricted use is generally the option chosen, although other options, such as safe storage or entombment are available for consideration.

A conceptual decommissioning plan for the proposed new research reactor emphasizes the major decontamination activities and process for the cleanup of the reactor and its facility, resulting in the final delicensing of the reactor and its facility. Part of the decommissioning plan normally includes the financial assurance requirements for the total cost of decommissioning. This requirement is expected to be exempt from the regulatory agency, since it is a DOE-owned research reactor. If DOE were to select this alternative (i.e., the construction and operation of a new research reactor at an existing DOE site), the formal site-specific decommissioning plan would be submitted for review and approval at the time of decommissioning. The decommissioning action at that time would be under a separate and appropriate National Environmental Policy Act (NEPA) review process.

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Appendix F
New Accelerator(s)

The U.S. Department of Energy (DOE) has determined that accelerators could be used for the production of medical and industrial isotopes, the conversion of neptunium-237 to plutonium-238, and to support nuclear energy research and development initiatives. The production of medical and industrial isotopes that are neutron poor could be effectively accomplished using a low-energy accelerator with energies in the range of 30 million to 70 million electron volts. Isotopes that are neutron rich are made in either reactors or high-energy accelerator spallation neutron sources. The irradiation of neptunium-237 targets for plutonium-238 production can be effectively accomplished using a high-energy accelerator with energies much greater than 100 million electron volts. Both low- and high-energy accelerators can be used to support nuclear energy research and development initiatives.

The accelerator(s) would be constructed and operated at one or two DOE sites where security measures would be in place, including access control and procedures, to ensure the adequate protection of all materials processed and stored. Although each accelerator would be independent of the other for the performance of its mission and can therefore be separately located, there may be important efficiencies to be gained by their colocation at the same DOE site.

F.1 LOW-ENERGY ACCELERATOR

A new 70-million-electron-volt cyclotron can be used for the production of medical and industrial isotopes and to support nuclear energy research and development initiatives. Important uses of the cyclotron would be to:

- Serve as a user facility for radioisotope production research, including excitation function measurements, high-power-density targetry required for isotope production at high beam currents, radiochemical separations, and purification

- Provide research capability for the development and evaluation of next-generation radioisotopes and radiopharmaceuticals for applications to imaging and therapy

- Provide a state-of-the-art, dedicated, multipurpose isotope production facility with simultaneous multiuser capability

- Respond to the national need for a continuous and reliable supply of present and future radioisotopes for biomedical research and other applications

- Provide a training facility for the next generation of nuclear and radiochemists in the areas of: nuclear and radiochemical techniques for radionuclide production, separation, and purification; radiotracer syntheses; radiopharmaceuticals development and evaluation; radiation protection and safety; and application of radiotracer methodology for biomedical investigations

Three low-energy accelerator options would be available for the production of medical and industrial isotopes and to support nuclear energy research and development: (1) a high-current proton linear accelerator (linac), (2) a multiparticle cyclotron, or (3) a proton-only cyclotron. The proton-only cyclotron would have distinct technical advantages over the other two options and is described further in the sections that follow.

F.1.1 Overview

The proton-only cyclotron can be either positive proton or negative ion and is referred to as a proton cyclotron H^+ or proton cyclotron H^-. The positive proton cyclotron alternative would offer lower vacuum requirements and, with the latest technology, high-extraction efficiency can be achieved. But, obtaining variable energy output would be complicated, extraction can be into only a single port, and splitting the beam would require a complicated septum magnet. In comparison, the negative ion cyclotron would offer a continuous beam with a high-current capacity using very simple high-efficiency extraction, a simple method to vary the particle energy, and the possibility of simultaneous irradiation of two different target arrays at different energies. The high-extraction efficiency would be achieved simply by passing the negatively charged beam through a thin foil that strips the electrons from the ion, creating a positive proton. The proton would be ejected directly from the machine by the existing magnetic field with high efficiency (greater than 98 percent). This feature would be important to minimize the activation of the cyclotron structure and thus reduce radiation exposure to the operational staff.

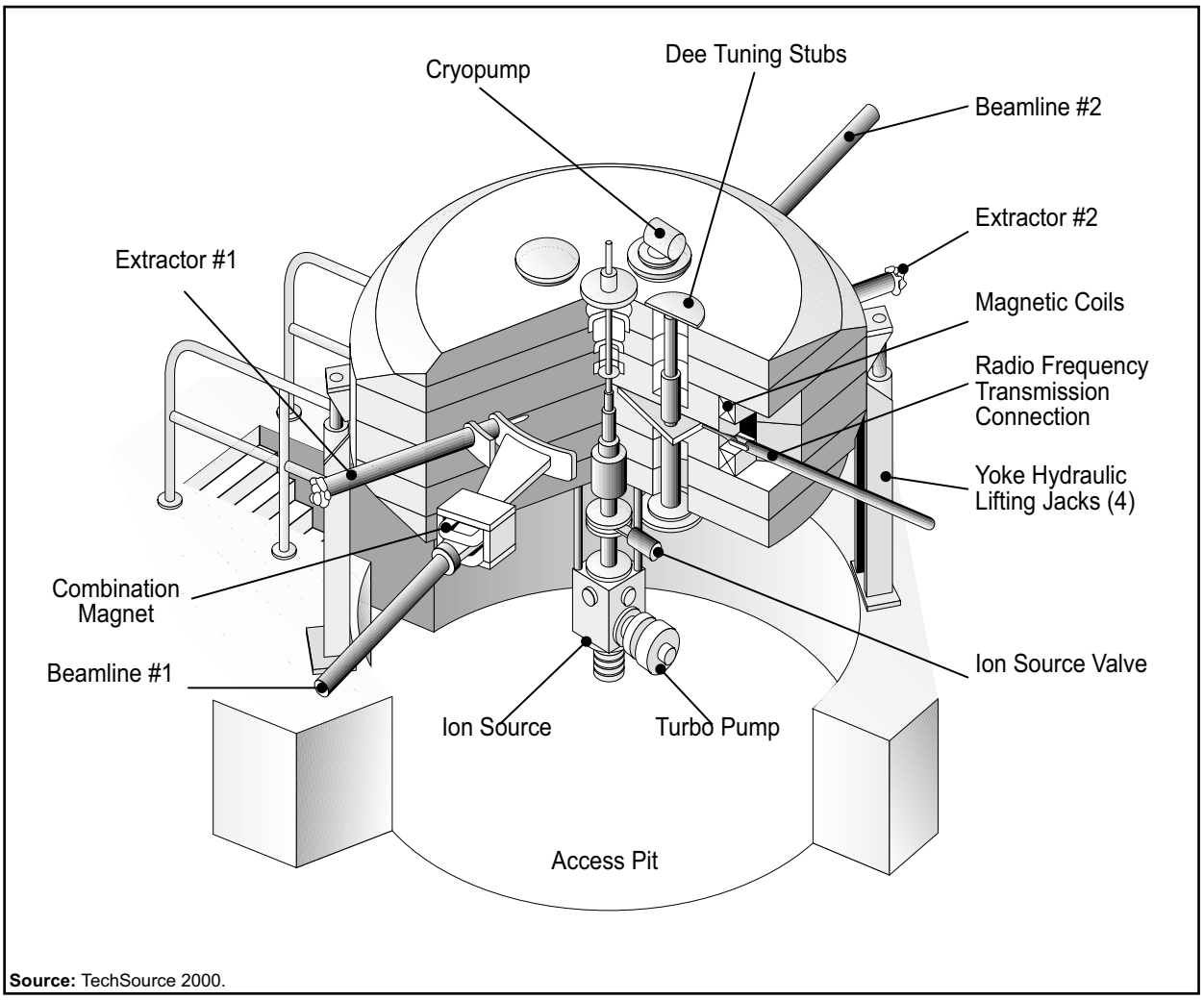
A high-beam current would be advantageous because more products can be prepared in a shorter time. In addition, a much higher specific-activity radioisotope can be prepared at the higher-beam current of the cyclotron. Specific activity is the ratio of radioactive atoms to total atoms of the same atomic number in the sample and is expressed in units of curies per gram. A stable element can enter the process in many ways, most commonly from the target, the reagents, or from the environment during handling of the irradiated target and subsequent processing. These quantities tend to be fairly constant from run to run, and would be independent of irradiation time or beam current. Therefore, a higher-intensity beam generally makes more radioactivity without adding to the amount of stable element in the final product. Specific activity is often a critical parameter in many nuclear medicine applications, including research and clinical use.

The cyclotron can also continuously tune the beam energy, which would be an advantage for research. The ability to tune the energy with precision can also help achieve high-purity isotope production by avoiding energies where impurity isotopes would be readily co-produced. It would be desirable to precisely tune to a low energy to achieve optimal production of certain radioisotopes. Energy variability from 40 million to 70 million electron volts would be easy, 30 million to 70 million electron volts would be possible, and 20 million to 70 million electron volts would need some design effort. It also would be possible to strip only part of the beam in an orbit and have another stripper at 180 degrees to simultaneously extract a second beam. This beam can have the same or different energy and intensity as the first. These are important advantages for flexibility in research isotope production and are within the capabilities of commercially proven technology.

F.1.2 Isotope Production Systems Design

A new building would be constructed to house the cyclotron and the four beam lines. The walls of the facility would be 4.6 meters (15 feet) thick behind the target stations to minimize the neutron flux outside the building. The walls surrounding the cyclotron itself would be 3 meters (10 feet) thick. The mazes throughout the building in general would have walls 1.5 meters (5 feet) thick, so that the total thickness surrounding the cyclotron area would be 3 meters (10 feet). The beam would be diverted to the four target stations by switching magnets located in the cyclotron vault. The beam would be directed through focusing and steering magnets to the target. In the isotope production beam line (northwest cave), the targets would be installed and removed vertically from a hot cell, which would be located on the second floor directly above the target station. The power supplies for the magnets would be housed with the power supplies for the cyclotron. The mechanical equipment for cooling water would be housed in a shielded mechanical room adjacent to the cyclotron vault. Recirculating water for cooling the targets and systems that could contain potentially radioactive material would be separated to prevent cross contamination. These systems would be contained in mechanical equipment rooms near the respective target station. Piping would be contained in waterproof

trenches with leak detection. Isometric views of the new cyclotron and beam lines are shown in **Figures F–1** and **F–2**.



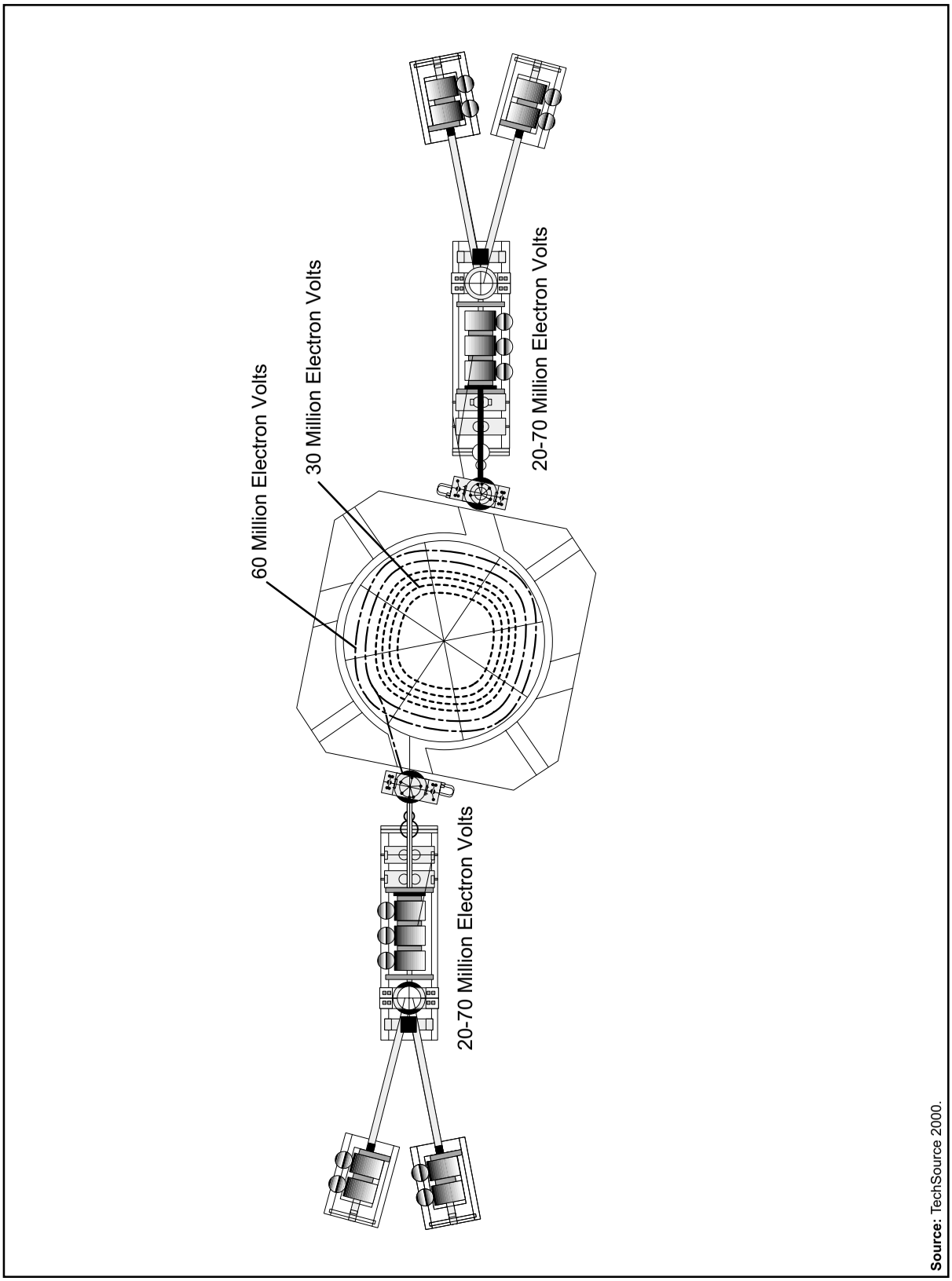
Source: TechSource 2000.

Figure F–1 Isometric View of the New Cyclotron

The isotope production system would be divided into several sections. These are the beam lines out of the cyclotron, the beam lines into each of the target caves, and the target holders and handling system in each of the target caves.

F.1.2.1 Beam Line Design

The beam would be extracted from the cyclotron by means of a thin carbon foil that strips two electrons off the hydrogen minus-one ion (H^-) and would convert the negatively charged hydrogen ions to positively charged ions. These would be bent in the opposite direction in the magnetic field of the cyclotron. Once the beam has been extracted from the cyclotron, it would pass through a beam shutter and into a quadrupole-focusing magnet. After the beam passes through the focusing magnet, it would pass into the switching magnet that would bend the beam into either of the two beam lines located on each end of the facility. After passing through the switching magnet, the beam would pass through another set of focusing and steering magnets inside the cyclotron vault and through the wall into the target cave. The direct current quadrupole and dipole



Source: TechSource 2000.

Figure F–2 Cyclotron and Beam Lines

magnets would be of conventional design and would have pole tip fields that are easily achievable with standard designs. In addition, the beam line for the northwest target cave would have a raster scan system consisting of variable-field horizontal and vertical steering magnets. This would allow rapid sweeping of the beam spot across the target face. Magnet cooling would require high-resistivity deionized water for voltage standoff purposes.

The vacuum system consists of beam pipes, bellows and flanges, vacuum valves and actuators, vacuum pumps, and vacuum instrumentation. The valves for the facility would include gate valves for isolating sections of lines, a fast valve for isolating the cyclotron vacuum volumes from the beam lines in the event of a target window failure, and roughing valves at the cryopumps.

The beam diagnostics system would provide the information needed to monitor and control the beam position throughout the beam lines and on the targets, and the final spot size on the target. Adjusting the fields of the direct current bending magnets would control the coarse beam position. Varying the steering magnet fields would make fine adjustments in the beam position. Adjusting the final quadrupole magnets in the line would control the beam spot size on the target. Each beam line would be equipped with a beam scanner that would give a distribution profile of the beam in both horizontal and vertical directions.

F.1.2.2 Isotope Production Equipment

There would be three separate target systems set up in the low-energy accelerator facility. The first would be the radioisotope production system housed in the northwest target cave. The second would be the positron emission tomography radioisotope production system housed in the southwest target cave. The third would be the research target system housed in the southeast target cave.

F.1.2.2.1 High-Level Radioisotope Production—Northwest Target Cave

SYSTEM FUNCTION

The isotope production equipment would include several components: the target housing (with targets), target transfer mechanisms, a hot cell, and a target radiation shield. All these components, except for the hot cell and portions of the target transfer mechanisms, would be in the lower level of the facility. Collectively, these systems must provide for target irradiation in a safe, cost-effective, and environmentally conscious manner.

SYSTEM DESIGN BASIS

The system would be designed to accept a proton beam with nominal energies of 30 million to 70 million electron volts and a maximum average beam current of 1 milliampere. Normal operations would be expected to use 250 to 500 microamperes of beam current. Target shielding would be sized to accommodate up to 1 milliampere of beam current, since the cyclotron would be capable of delivering currents of up to 2 milliamperes.

Targets would be designed with a circular cross section. Calculations of power deposition would assume a flat beam profile over the entire target surface. A rastering system would be in place on this beam line to ensure an optimum distribution of the beam. More than one target would be irradiated at a time, but the targets would be inserted and retrieved as a single assembly. The thickness of this assembly must be at least 10 centimeters (3.9 inches), not including water gaps between the target faces. This ensures that there would always be enough target material or water to stop 70 million electron-volt protons.

The target housing would become very highly activated during the irradiation process. Therefore, personnel would not have access to the target housing after the initial irradiation. In addition, a concrete radiation shield would be installed downstream of the target assembly inside the cave to provide extra shielding from forward-going prompt neutrons and from residual gamma radiation.

SYSTEM DESIGN DESCRIPTION

The isotope production target facility would be housed in a stainless steel cylinder, which would form a 5.5-meter (18-foot) column of water. The top of the tank would open to a hot cell. The bottom of the tank would connect to the beam vacuum chamber. A beam of protons ranging from 30 million to 70 million electron volts would enter the water column through a thin vacuum window at the end of the beam vacuum chamber. The targets would be transported by a trolley holder, which would use a rectangular tube as a track. The rectangular track would be housed within the 5.5-meter (18-foot) water column. The trolley would transport and locate the targets at the beam centerline just downstream of the beam vacuum chamber exit window. The trolley drive system would be a semiautomatic motor-driven chain and cable assembly. The target would be water-cooled by forced water traveling vertically over the face of the target.

The water column would house the target transport system and contain the cooling water. The diameter of the water column would be sufficient to stop protons with energies up to 70 million electron volts. The 5.5-meter (18-foot) height also would provide neutron shielding in the vertical direction. The target water column would be constructed from nuclear-grade, corrosion-resistant material. The column would have inlet and outlet connections for cooling water and a connection for the beam pipe. All the connections would be as-low-as-is-reasonably-achievable-designed to reduce radiation exposure during maintenance or removal of the system. The bottom of the column would form a plenum where the cooling water inlet would be. On the top of the plenum, there would be alignment holes to align the target drive track system. Above the beam line vacuum chamber connection would be the cooling water outlet. The top of the column would penetrate the hot cell to approximately tabletop height. There would be a lid at the top of the column to prevent objects from falling down the column and to reduce radioactive airborne emissions. All water line connections and seals would be made of nuclear-grade material.

At the downstream end of the beam line vacuum chamber would be a thin window. The window would be designed to permit the beam to irradiate the production targets with minimal beam loss. The window would be made of high-strength, nuclear-grade, corrosion-resistant material. The exit window would be designed for quick replacement in the event of a failure.

The targets would be transported vertically to and from the hot cell to the beam line within the water column. The targets would be transported using a motor-driven chain and cable assembly. The chain and cable assembly would be a continuous loop attached to a trolley. The top half of the loop would be a chain for a slip-free connection to the drive motor via the sprocket. The bottom half of the loop would be a cable. The track system for the target transport would be a rectangular tube that would guide and align the trolley. The track system would be removable from the hot cell for replacement or repair. The track would be aligned to the beam centerline by engagement of alignment holes located at the bottom of the water column. A trolley target holder would be attached to the chain and cable assembly and used to transport the targets in the rectangular track. The target holder would be capable of holding targets of various thicknesses in a horizontal array. The targets would be separated in the array by flowing water coolant to allow cooling of both the front and back faces of the target. To insert and remove the targets one at a time from the target trolley holder, a target extraction system would be needed. The extraction system would be installed in the hot cell and would be motorized to simplify the manipulation of the targets from within the hot cell.

The target assembly would consist of the target material housed in a sealed container. The container would be constructed from material designed to withstand corrosion by either the water coolant or the target material. The sealed container would be a circular disk, fabricated in a variety of thicknesses. The targets would be water-cooled. The water coolant would enter the water column at the bottom into the plenum, pass through a fixed orifice into the target track system, flow vertically across the targets, and then exit the water column above the beam centerline. The targets would be placed in the beam line tilted at 30 degrees. This would reduce the heat flux at the target front and back faces, allow for a larger cooling channel across the target, hence more water flow, and would reduce the thickness of the target, thus reducing the heat path to the water coolant.

F.1.2.2.2 Positron Emission Tomography Radioisotope Production—Southwest Target Cave

SYSTEM FUNCTION

The isotope production equipment would include several components: the automatic target changer (with targets), the water cooling system, the helium cooling system for the front foil of the targets, and the radioisotope removal system and transport line. Collectively, these systems must provide for target irradiation in a safe, cost-effective, and environmentally conscious manner.

SYSTEM DESIGN BASIS

The system would be designed to accept a proton beam with a nominal energy of 30 million to 70 million electron volts and a maximum average beam current of 250 microamperes. Targets would be designed with a circular cross section with a minimum diameter of 2.5 centimeters (1 inch). Calculations of power deposition would assume a Gaussian beam energy density profile, with a minimum full-peak width at half of the maximum of 1 centimeter (0.34 inch). One target would be irradiated at a time, but several targets would be present in the vault at a single time in an automatic target changer. The targets would be inserted and retrieved remotely. The target housing would become activated during the irradiation process.

SYSTEM DESIGN DESCRIPTION

The holders would be cylindrical-shaped with outside diameters of 7.6 centimeters (3 inches) and would occupy the centers of the target holders. The targets themselves would reside inside each of these cylinders. The targets would be solid plates that contain target powder and a small volume of liquid or a larger volume that contains a compressed gas. The beam line would have vacuum isolation foil, which would separate the helium cooling chamber from the cyclotron beam line. The vacuum isolation foil would be 0.0038-centimeter-thick (0.0015-inch-thick) aluminum alloy. The helium chamber would provide the reservoir for the chilled helium that would be passed over the front surface of the target foil. Should a window failure occur, the design of the window and its mounting structures must provide for effective operation and ease of replacement. Capturing the window in a foil holder that would be inserted between the beam line and the helium cooling assembly would satisfy these requirements. The front foil of the target must be thick enough to withstand the pressure generated inside the target during irradiation, which can be more than 600 pounds per square inch. In back of the target chamber would be a water cooling assembly that would help remove heat from the target body.

Radioisotopes can be removed from this target cave in two different ways. The first of these would be used for the solid targets. The target plate containing the irradiated powder would be remotely removed from the target body and placed into a shielded container. This container would be rolled out of the facility and transported to a hot laboratory. Once at a hot laboratory, the target would be processed to extract the desired radioisotope.

The second method of target extraction would be through a processing station in the target vault. In this method, the fluid target contents would be pushed out of the target with a stream of helium and onto a resin or absorbent column that would retain the desired radioisotope. This column then would be transported to a hot laboratory.

F.1.2.2.3 Research Radioisotope Production—Southeast Target Cave

SYSTEM FUNCTION

The isotope production equipment would include several components: the target holder (with target), the water cooling system, and the helium cooling system for the front foil of the targets. Collectively, these systems must provide for target irradiation in a safe, cost-effective, and environmentally conscious manner.

SYSTEM DESIGN BASIS

The system would be designed to accept a proton beam with a nominal energy of 70 million electron volts and a maximum average beam current of 1 milliampere. Cross-section-type experiments would be expected to use only 1 to 10 microamperes of beam current, but target research may use up to 1 milliampere. Target shielding would be sized to accommodate up to 1 milliampere of beam current, since the cyclotron would be capable of delivering currents of up to 2 milliamperes.

The operations sequence breaks down into four major operations: (1) load targets, (2) irradiate targets, (3) extract targets, and (4) transport irradiated targets.

Load Targets

Target irradiation would occur in the southwest target cave in an automated target-changer assembly. Access to the target chamber would be through the maze originating on the west side of the facility. The fluid targets would be loaded remotely from gas or liquid reservoirs residing in the target cave. The solid targets would be prepared outside the facility and brought in and placed in the target holder.

Irradiate Targets

The first step in preparing to start an irradiation run after the target stack has been loaded would be to establish coolant flow. This would be done by starting flow in the water system, then evacuating and pressurizing the helium system, followed by circulation initiation. The beam can be delivered to the target when the cooling system is functioning properly and all interlocks are satisfied. Irradiation would be continuous as long as the systems performance indicators, such as flow indicators and temperature monitors, remain within specified limits. The beam current striking the target would be integrated to determine when the proper number of protons has struck the target and the irradiation is complete.

Extract Targets

A transfer rabbit would be in place inside the target cave, which would hold the resin column or absorbent used to extract the desired radioisotope from the target material. The first step in preparing to extract the irradiated target material would be to stop the coolant flows. The target material would be forced out of the target and onto the resin column with a flow of helium. Once the radioactive isotopes have been transferred to the column, the flow of helium would be stopped and the transfer rabbit dropped into the transfer tube.

Transport Irradiated Targets

The transfer rabbit would be dropped into a transfer tube and pneumatically sent to a hot laboratory.

F.1.3 Facility Systems Design

The physical layout of the new cyclotron building would consist of two levels. The first floor (**Figure F–3**) would contain the vault room that would house the 70-million-electron-volt cyclotron, beam transport systems, cyclotron mechanical room, four target rooms, and storage room. The cyclotron vault room would be centrally located with a high bay equipped with a 15-ton bridge crane. There would be concrete trenches in the floor between the target rooms and the cyclotron vault room and a cross-shaped floor trench beneath the cyclotron. These trenches would be lined with an epoxy coating, which acts as secondary containment for any liquid that may be spilled, and have leak-detection sensors to comply with local environmental regulations. The cyclotron vault room would have a concrete ceiling with a removable roof plug for overhead installation of the cyclotron. The second floor would consist of a transfer room, staging area, and electrical and mechanical room. The transfer room would contain a concrete hot cell connected to the target room located directly below. Materials would be transferred from the hot cell to a shielded cask and transported to the hot side of the building for processing.

F.1.3.1 Architectural and Structural Design

F.1.3.1.1 First-Floor Cyclotron Facility

The interior spaces on the first floor would include the cyclotron vault, the cyclotron mechanical room, four target rooms with individual mechanical and electrical rooms, and a power supply room.

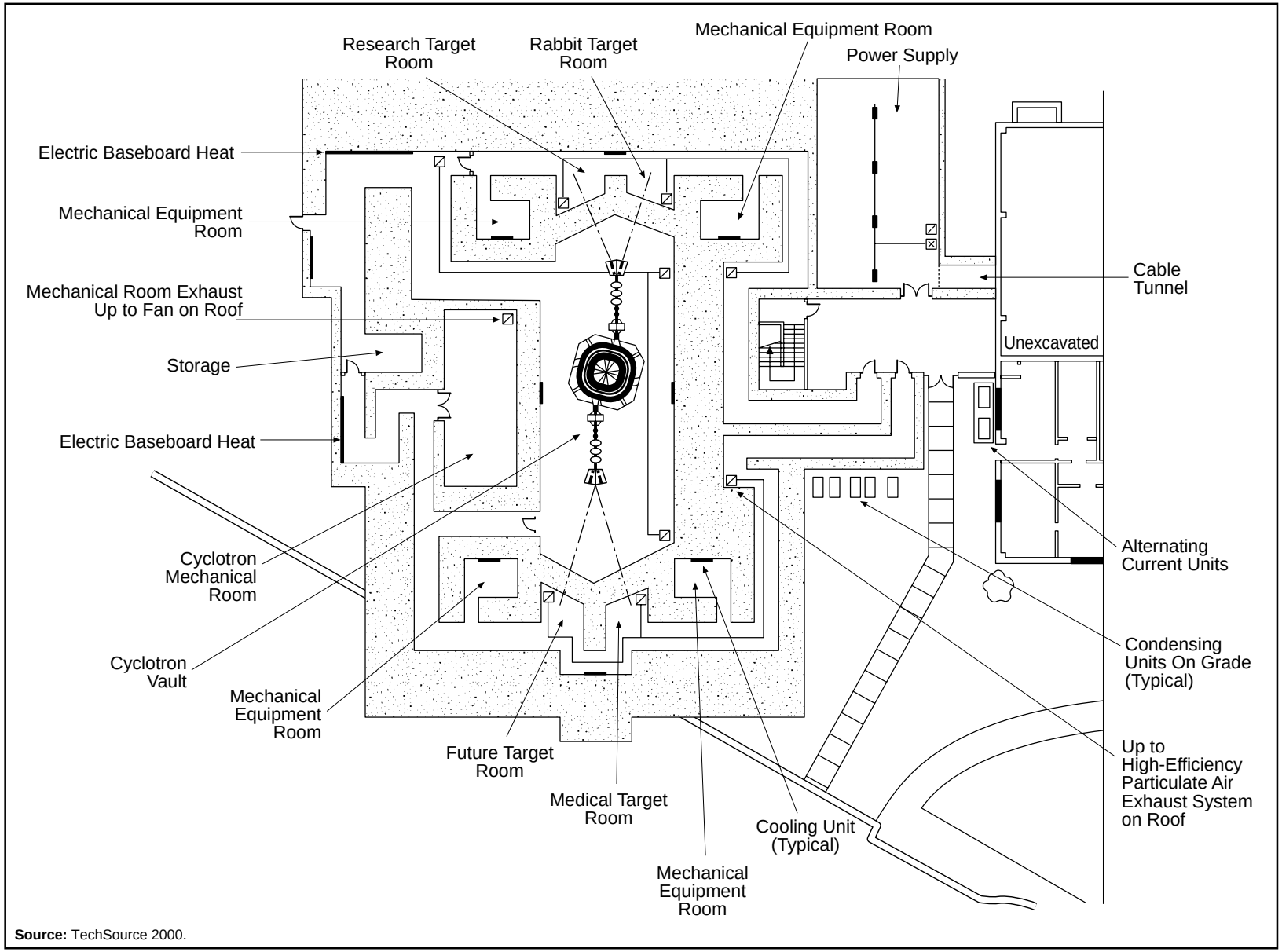
SYSTEM FUNCTION

The cyclotron vault would be surrounded by perimeter shielding walls that are 4.6 meters (15 feet) thick with a 0.91-meter-thick (3-foot-thick) ceiling constructed of reinforced-concrete. All other interior spaces would have a minimum shielding of 1.5 meters (5 feet) of concrete. The 15-ton bridge crane would be used for handling shield components in the cyclotron vault. The lower floor would be approximately 42.7 by 41.1 meters, or 1,755 square meters (140 by 135 feet, or 18,900 square feet), partially below grade. The first floor would house the cyclotron, target rooms, and cyclotron power supply room. The interior ceiling heights would be 5.8 meters (19 feet) for the cyclotron vault, and a minimum of 2.7 meters (9 feet) for all other spaces. The entire first floor facility would be constructed of reinforced cast-in-place concrete faced with brick veneer. The function of the structural systems would be to resist all anticipated loads from the soil, the second-floor facility, and the mechanical and electrical utilities within the facility. Floors would be constructed of epoxy-coated concrete.

SYSTEM DESIGN BASIS

The following is a generalized list of anticipated loads imposed on the first-floor cyclotron facility:

- Roof load: soil pressure, second-floor facility, and construction equipment load
- Floor load: live load of 250 pounds per square foot, 1,000-pound point load, target shield cube of 240 tons
- Wall load: lateral soils pressure of 55 pounds per square foot per foot of depth
- Wind load: not applicable
- Seismic load: design-basis-earthquake peak horizontal ground acceleration of 0.2 g



Source: TechSource 2000.

Figure F–3 Cyclotron Facility Floor Plan

SYSTEM DESIGN DESCRIPTION

The following is a generalized list of descriptions of systems on the first-floor cyclotron facility:

- Roof system: 0.91-meter-thick (3-foot-thick) reinforced cast-in-place concrete
- Wall system: 4.6-meter-thick (15-foot-thick) reinforced cast-in-place concrete
- Foundation system: 0.91-meter-thick (3-foot-thick) reinforced-concrete mat footing
- Lateral resisting system: reinforced-concrete shear walls with a concrete roof diaphragm
- Crane: the cyclotron high-bay facility would include a 15-ton bridge crane for loading and unloading the magnets; the concrete walls would support this crane

F.1.3.1.2 Second-Floor Cyclotron Facility

The interior spaces for this floor include the transfer room containing a concrete hot cell, the electrical mechanical room for the building, and a staging area. This partial second-floor structure would be constructed of a steel frame with concrete block and brick veneer walls and partitions, with a built-up roof system on a metal roof deck. The mechanical and electrical room would contain the main electrical service switchboards, panel boards, and mechanical equipment used for cooling the magnets, the target below, and the building itself. There would be a 5-ton bridge crane located on the second floor of the cyclotron facility to service the hot cell and target components.

SYSTEM FUNCTION

The second floor would consist of one structural area, approximately 30.5 by 40 meters, or 1,220 square meters (100 by 130 feet, or 13,000 square feet), and would house the transfer room, staging area, and major utility space. The entire second-floor facility would be constructed of reinforced cast-in-place concrete floors and a steel frame with masonry walls faced with brick veneer. The function of the structural systems would be to resist all anticipated loads from the roof, wind, and the mechanical and electrical utilities within the facility. Floors would be constructed of epoxy-coated concrete.

SYSTEM DESIGN BASIS

The following is a generalized list of anticipated loads imposed on the second-floor cyclotron facility:

- Roof load: live load of 30 pounds per square foot
- Floor load: live load of 250 pounds per square foot; hot cell live load of 83 tons
- Wind load: the design-basis wind would be a straight wind at 90 miles per hour
- Seismic load: design-basis-earthquake peak horizontal ground acceleration of 0.20 g

SYSTEM DESIGN DESCRIPTION

The following is a generalized list of descriptions of systems on the second-floor cyclotron facility:

- Roof system: a built-up roofing system on a metal roof deck screwed to steel purlins that are welded to steel channel roof purlins
- Wall system: steel columns and girders with masonry in-fill walls running horizontally at the perimeter of the building
- Floor system: reinforced-concrete under the steel columns; a mat footing approximately 30.5 centimeters (12 inches) thick would be incorporated under the hot cell, and 15.2-centimeter-thick

(6-inch-thick) reinforced-concrete slabs on grade would be incorporated throughout the rest of the facility
- Lateral resisting system: a steel-braced frame with a metal roof diaphragm
- Crane: a 5-ton bridge crane installed over the hot cell

F.1.3.1.3 Hot Cell

There would be a hot cell on the second floor of the facility located above the northwest target cave that would contain the top of the target after-shaft and the target drive assembly. The hot cell would provide radiation shielding to safely handle the highly activated targets during target insertion and removal operations. It would also provide prompt neutron shielding above the target cave. Although the hot cell area would be an exclusion zone with the beam on, the presence of the hot cell would help reduce "sky shine" outside the target transfer room to acceptable levels for uncontrolled access.

The hot cell would be made from cast concrete and steel components. The concrete thickness would be determined based on criteria that the maximum personnel exposure rate from the highest likely radiation source term in the interior not exceed 5 millirem per hour. The interior dimensions would be adequate to perform operations such as target insertion and removal, simple repair of the target holder, and installation of an interior lead storage cave. The hot cell would have a viewing window made of sheets of lead glass using standard construction techniques. The effective shielding thickness would be equal to that of the walls. The size would be sufficient to allow operators to see most of the interior and perform all required functions. Two master-slave remote manipulators would be provided for remote target-handling operations. They would be sized to reach the entire working area of the hot cell.

The hot cell would have a shielded port hole mounted in a side wall to enable the removal of radioactive targets and radioactive waste. A shutter plug of lead and steel would be constructed. The location and arrangement of this hot cell opening would prevent direct radiation shine from hot cell contents into the target transfer room. The port hole design would be compatible with an existing target transport cask. The transport cask would be moved from this hot cell to the hot cell in the Target Processing Laboratory of the building by an electrical pallet truck. The port hole also would allow the introduction of small materials and equipment into the hot cell. In order to allow larger equipment to be introduced or removed for repair, a larger lead and steel shielded door also would be provided in a side wall. This door would be large enough for personnel entry for major maintenance or decontamination operations. Suitable locks would be provided to restrict such access unless entry would be permitted.

Other small penetrations would be provided for typical services in a serpentine manner to minimize straight-line radiation paths. The hot cell would be connected to a high-efficiency particulate air-filtered ventilation system that would keep the interior pressure slightly negative with respect to the room. This would assure that no radioactive contamination would spread outward. The hot cell would be equipped with high-intensity lighting to ensure adequate vision through the thick lead-glass window. Standard 120-volt alternating current electrical receptacles, water, and air would be supplied into the hot cell interior.

F.2 HIGH-ENERGY ACCELERATOR

F.2.1 Overview

F.2.1.1 System Description

In accelerator production of plutonium-238, an energetic beam of protons generated by a linac would be transported to a heavy metal target where spallation neutrons would be produced. The beam of protons would

be moderated in a surrounding blanket containing neptunium-237, which would capture the slowed neutrons to produce plutonium-238 through the same nuclear sequence as occurs in a reactor. The accelerator would be housed in a concrete tunnel, buried below ground to provide radiation shielding for operating personnel. A building housing radio frequency power systems and other equipment used to drive, monitor, and control the accelerator would be located above ground close to the accelerator tunnel. The target/blanket assembly would be housed inside a steel and concrete shield located within a multistory building that would contain appropriate service equipment. At the target, the small-diameter proton beam transported magnetically from the accelerator would be converted to a much larger cross section by a beam expander to reduce the power density to acceptable levels for the target cooling systems.

Figure F–4 shows the accelerator production of plutonium plant layout.

F.2.1.2 Plutonium-238 Production Process

An accelerator-driven spallation neutron source can be used to produce plutonium-238 from neptunium-237 feedstock through the capture and decay nuclear processes. A 1,000-million-electron-volt proton beam produced by a radio frequency linac would bombard a heavy metal (uranium-238) target, with each proton producing about 40 neutrons. Surrounding the spallation target would be a blanket containing a mixture of neptunium-237 and water coolant in an aluminum structure, all inside a beryllium reflector. The combination of materials in the target/blanket assembly would moderate the neutron spectrum down to thermal energies, where the capture cross section in neptunium-237 would be about 200 barns. As in a reactor, the nuclear reactions would be:

$$Np237 + n \rightarrow Np\text{-}238$$

$$Np238 \rightarrow Pu238 + \beta$$

Plutonium-238 nuclei, once formed, would have a significant cross section for destruction through neutron capture, which must be taken into account when optimizing the blanket neutron spectrum, the neutron flux at the neptunium-237 locations, and determining the optimum length of the irradiation periods.

The use of medium-energy proton beams for nuclear material production or conversion is a well-established concept of several decades. The technology basis for high-energy proton accelerators and spallation neutron sources has been developing, over the past 10 years, through two U.S. projects, as well as in Europe and Japan. One U.S. project is the DOE Defense Programs' development of a backup tritium production method (the Accelerator Production of Tritium program) and the other is the design of high-power pulsed spallation sources in the United States (the Oak Ridge Spallation Neutron Source facility).

In the spallation process, the high-energy protons smash into nuclei of the target material, initiating an intranuclear cascade followed by an evaporation process in which many neutrons are emitted. The spallation neutron spectrum would be similar to a fission spectrum, peaking at 1.2 million electron volts, but it has a high-energy tail.

An important factor in the selection of proton beam energy and current is that the number of spallation neutrons produced depends on the beam energy. While there is a nearly linear relationship over a large energy range (up to 2 giga electron volts), there is also a finite energy threshold (about 200 million electron volts) below which neutron production is effectively zero. The existence of this threshold energy means that neutron production as a function of beam power rises steeply for the first few hundred million electron volts, but asymptotically reaches a constant at high energies.

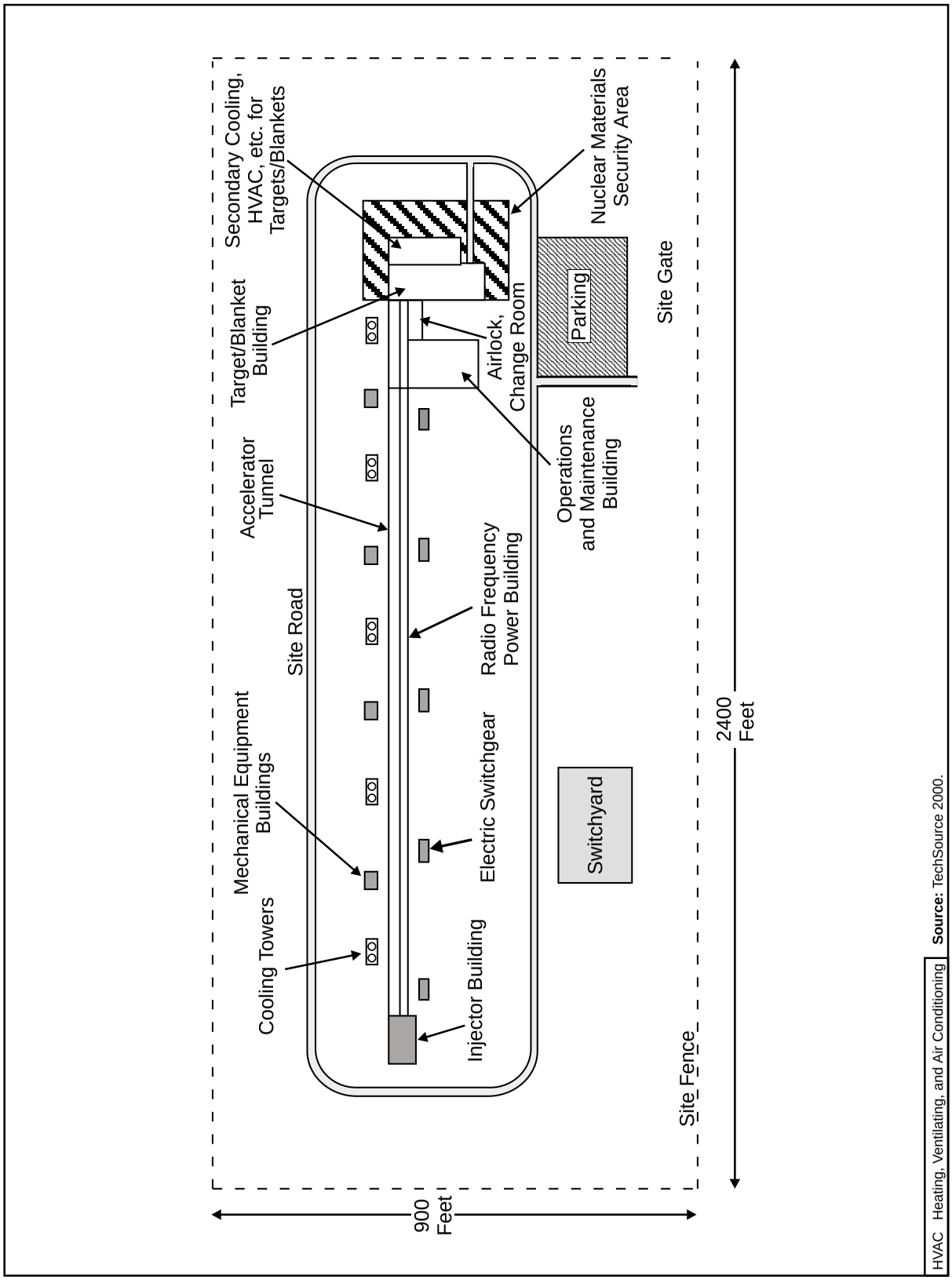


Figure F–4 Plant Layout for the Accelerator Production of Plutonium

Several high-Z materials are potential target material contenders, but depleted uranium would be the obvious choice for plutonium-238 production, since it produces more neutrons (by a factor of two) per incident proton than tungsten (the target material used in the Accelerator Production of Tritium), tantalum, or lead. The additional neutrons are produced by fast fission of uranium-238 nuclei induced by a fraction of the spallation neutrons.

F.2.1.3 Production Requirements

A preliminary target and blanket design has been developed for scoping purposes, based on the architecture employed in the Accelerator Production of Tritium target and blanket design. It would use uranium-238 (cooled by deuterium) as the neutron-production target. The target would be surrounded by a blanket of neptunium-237 in a dilute mixture of aluminum and water coolant. Enclosing the blanket would be a beryllium reflector. Initial code calculations show that, with 72 kilograms (158.7 pounds) of neptunium-237 in the blanket, about 40 neutrons would be produced by each proton, of which about 60 percent would be captured in neptunium-237 to produce plutonium-238. Further optimization may increase both the number of neutrons per proton and the fraction useful for making plutonium-238 nuclei, but improvements greater than a factor of 1.3 in the number of plutonium-238 nuclei made by each proton are unlikely.

Once produced, the plutonium-238 nuclei are subject to destruction processes as long as they remain in the neutron flux. The dominant process is plutonium-238 + n $\rightarrow$ plutonium-239 + γ, which has a 540-barn thermal cross section, significantly greater than the 200-barn cross section for plutonium-238 production. Calculations show that there is an optimum neutron flux and irradiation campaign period that minimizes plutonium-238 destruction without invoking an excessive frequency of blanket reloading cycles. For example, at a flux of 4×10^{13} per square centimeter, the fractional plutonium-238 destruction in 90 days would be 6.7 percent.

The annual production requirement for plutonium-238 for this *Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* has been given as 5 kilograms (11 pounds). Assuming 20 percent losses in chemical processing, 6 kilograms (13.2 pounds) per year would be needed in the material extracted from the production blanket. The year would be divided into three 4-month production campaigns, with a net amount of 2 kilograms (4.4 pounds) of plutonium-238 produced in each, allowing for the fraction destroyed. Each campaign would be divided into 100 days of production and 21 days for recycling the production blanket. A 90 percent plant availability during the scheduled operating periods is assumed, which should be achievable based on operating experience at the Los Alamos National Laboratory linac. Then the average proton current required as a function of beam energy is determined by the production relationship:

$$M/(1 - 0.067) = 6.24 \times 10^{18} \times I(E) \times Y(E) \times 238 \times 1.67 \times 10^{-27} \times 0.90 \times 0.854 \times 10^{7}$$

$$M = 17.75\ I(E)Y(E)$$

where M is the mass of plutonium-238 required per campaign in kilograms, I(E) is the average proton current in amperes, and Y(E) is the yield of plutonium-238 nuclei per incident proton. With M = 2 kilograms, this becomes

$$I(E) = 0.113\ /Y(E)$$

At a beam energy of 1,000 million electron volts, the preliminary target/blanket neutronics show that about 24 plutonium-238 nuclei are produced by each proton, setting 4.7 milliamperes as the nominal average current requirement.

F.2.1.4 Accelerator

As a stand-alone machine, a reasonable design basis for a plutonium-238 production accelerator would be a pulsed normal-conducting linac having an architecture similar to that of the January 2000 (baseline) normal-conducting linac design for the Oak Ridge Spallation Neutron Source facility. The normal-conducting version of the Spallation Neutron Source linac has a 1,000-million-electron-volt beam energy, and an average output current of 2 milliamperes, about one-third the requirement for plutonium-238 production. The peak current would be 36 milliamperes, corresponding to a beam duty factor of 0.056. However, the Spallation Neutron Source linac design was intended to be directly upgradeable to a 4-milliampere average current by doubling the peak current to 72 milliamperes. For the plutonium-238 production linac scoping design, this step also would be taken and the duty factor would be stretched to 0.066 to provide the required average current. Since the radio frequency power systems in the Spallation Neutron Source machine are designed for up to 0.09 duty factor, the same kind of power stations could be used for the plutonium-238 production linac. This modified version of the Spallation Neutron Source linac would satisfy the beam power requirement for plutonium-238 production and seems a reasonable model for carrying out the estimates needed for the mission.

The plutonium-238 linac concept would begin with an H^+ injector supplying beam to a 6-million-electron-volt radio frequency quadrupole operating at 400 megahertz. The Spallation Neutron Source machine uses an H^- beam instead of protons, and has complex chopping arrangements at the front end. However, both of these features pertain to injection into a storage ring following the linac and can be eliminated in the plutonium-238 production application, allowing considerable simplification. The radio frequency quadrupole would be followed by a 400-megahertz drift-tube linac to 80 million electron volts, and an 800-megahertz coupled-cavity linac to full energy of 1,000 million electron volts. Accelerating structure lengths, gradients, focusing periods, and aperture sizes would be nominally the same as in the Spallation Neutron Source design. Beam dynamic simulations for the Spallation Neutron Source linac provide assurance of beam losses low enough (less than 0.1 nanoampere at 1,000 million electron volts) to permit unrestricted hands-on maintenance. The Spallation Neutron Source linac radio frequency power system employs high-peak-power klystrons (2.5 megawatts in the drift-tube linac, 5 megawatts in the coupled-cavity linac). Because the radio frequency efficiency (beam loading) of the Spallation Neutron Source linac would be relatively low (0.25), doubling the peak current as proposed above would increase the number of required radio frequency stations by a factor of only 1.25. However, because of the duty factor increase, each station would operate at a factor 1.18 higher average power. Key parameters for the reference accelerator are given in **Table F–1**.

Other accelerator options are possible, and might result in somewhat lower costs, coupled with superior performance. A reduced operating cost can be obtained by using super conducting accelerating cavities in the high-energy part of the pulsed linac, as in the Spallation Neutron Source final design. However, this introduces extra complications and results in somewhat higher capital costs. Using a linac with a lower beam-output beam energy (500 million electron volts) and higher current (14 milliamperes) would be another choice. Such a machine would employ super conducting cavities above 100 million electron volts, and could be somewhat more compact than the nominal Spallation Neutron Source linac. However, scoping cost estimates suggest that its capital costs would likely be not very different than for the reference 1,000-million-electron-volt pulsed linac.

Optimization of target and blanket performance as a function of beam energy, choice of materials and geometry, etc., has not been done. In concert with an analysis of accelerator costs as a function of proton

Table F–1 Linac Parameters

Parameter	Radio Frequency Quadrupole	Drift-Tube Linac	Coupled-Cavity Linac	Total
Radio frequency (megahertz)	400	400	800	
Output beam energy (million electron volts)	6	80	1,000	1,000
Accelerating gradient (average) (megavolts per meter)	–	–	2.63	–
Average current (milliamperes)	4.7	4.7	4.7	4.7
Peak current (milliamperes)	72	72	72	72
Beam duty factor (milliamperes)	0.066	0.066	0.066	0.066
Radio frequency duty factor (milliamperes)	0.081	0.081	0.081	0.081
Output beam power (average) (megawatts)	0.03	0.38	4.7	4.7
Output beam power (peak) (megawatts)	0.45	5.76	71.2	71.2
Average cavity power loss (megawatts)	0.17	0.45	6.58	7.20
Radio frequency station peak power (megawatts)	2.5	2.5	5.0	–
Number of radio frequency stations (megawatts)	1	9	37	47
Radio frequency power delivered (peak) (megawatts)	2.0	12.9	165.2	180.3
Radio frequency power delivered (average) (megawatts)	0.15	0.95	12.2	13.3
Alternating current power for radio frequency (megawatts)	0.3	2.1	26.5	28.9
Alternating current for linac and high-energy beam transport (megawatts)	–	–	–	34.4
Section length (meters)	7	59	404	470

Key: Linac, linear accelerator.
Source: TechSource 2000.

energy (using models developed for the Accelerator for Production of Tritium and other high-power linac projects), such an optimization might well lead to a plutonium-238 production system design with significantly improved performance and lower costs. In this connection, it should be noted that a multipurpose higher-power accelerator-driven system, in which the costs of producing the protons are shared between several nuclear missions, would result in much lower capital costs for plutonium-238 production, as well as lower operating costs. Such a system almost certainly would involve an accelerator with a beam current of 30 to 50 milliamperes, using a super conducting high-energy section, and suitable beam-sharing arrangements for the different missions.

F.2.2 Isotope Production Systems Design

F.2.2.1 Target Blanket Assembly

The spallation target portion of the reference target blanket assembly would be 28.4 centimeters (11.2 inches) wide by 28.4 centimeters (11.2 inches) high, and 100 centimeters (39.4 inches) long in the beam direction. It would consist of 504 kilograms (1,111 pounds) of depleted uranium packaged in 11 kilograms (24.3 pounds) of aluminum and contain 10 kilograms (22 pounds) of heavy water in its cooling channels. The target would require the removal of 4,360 kilowatts of heat due to fissions and 956 kilowatts due to gamma heating.

The blanket portion of the target blanket assembly would contain 72 kilograms (158.7 pounds) of neptunium-237 packaged in 1 kilogram of aluminum structure and would be cooled with 93 kilograms of light

water. It would occupy a 5-centimeter-thick (2-inch-thick) layer wrapped around the four long sides and the down-beam end of the target. The side layers would extend 30 centimeters (11.8 inches) beyond the uncovered end of the target. The purpose of this extension would be to capture some of the neutrons that would emanate from the beam entrance face of the target. That face would be exposed because placement of feedstock in the proton beam would cause its demise by spallation. The blanket region would require the removal of 361 kilowatts of heat due to fissions and 966 kilowatts due to gamma heating.

Cooling water manifolds 2 centimeters (0.8 inch) thick and containing 25 kilograms of light water would cover the top and bottom surfaces of the target assembly. The total weight of the target assembly would be 716 kilograms (1,578 pounds). The weight of the dry assembly would be 588 kilograms (1,296 pounds).

F.2.2.2 Reflector Assembly

The reflector would consist of 30-centimeter-thick (11.8-inch-thick) beryllium slabs that would cover all but the beam-entrance face of the target. It would weigh 2.5 metric tons. The outer dimensions of the reflector would be 99 centimeters (39 inches) wide by 103 centimeters (40.6 inches) high, and 165 centimeters (65 inches) long in the beam direction.

F.2.2.3 Vacuum Tank and Internal Shielding

The target assembly would be positioned near the center of a vacuum tank that would be 6.1 meters (20 feet) in diameter and 5.5 meters (18 feet) high. The target and reflector assemblies would be hung from the bottom of a shield plug that would be 1 meter (3.3 feet) wide, 1.7 meters (5.6 feet) long, and 2.4 meters (7.9 feet) tall. The top of the plug seals, and would be supported by, a penetration in the top lid of the tank. Cooling water and instrumentation lines would be routed from the target reflector assemblies, up through the plug, to the top of the tank.

The purpose of the large tank would be to allow space to install sufficient steel shielding so that the tank would not become activated. This would allow personnel to work on the plug at the top of the tank when the beam is off. It also would make it possible to use elastomers as seals for the plug and tank lid. It would greatly simplify tank removal when the facility is decommissioned.

The shielding within the tank would be 2.4 meters (7.9 feet) thick above and downstream of the target assembly, 2.0 meters (6.6 feet) below and upstream, and 2 meters (6.6 feet) to the sides.

The shielding to the sides of the plug and surrounding the target and reflector assembly would be movable to allow extraction of the plug and the assemblies hung from it.

F.2.2.4 External Shielding

An additional 3 meters (9.8 feet) of steel shielding, plus 1 meter (3.3 feet) of concrete, are wrapped around and over the tank to permit unlimited personnel access when the beam is on. Access to this region would be required to prepare for target blanket assembly change out and to prepare irradiated target blanket assemblies for disassembly in the hot cells. A portion of the shielding directly above the tank would be mounted on rollers that would allow it to be rolled aside to provide easy access to the plug and its associated piping.

F.2.2.5 Beam Transport

The beam transport system would direct the proton beam from the accelerator onto a straight-ahead beam stop or bend it toward the target. The straight-ahead beam stop would be required for the commissioning and tune-up of the accelerator.

The beam transport system would consist of quadrupole magnets that would maintain the focus of the beam and dipole magnets that would bend the beam. The beam would travel inside a vacuum pipe that would be located within the magnets. Instruments that would diagnose the location of the beam would be positioned at strategic points within the beam transport line. A few very thick plugs would be located within the transport line, but would be held above the beam position. These plugs would be lowered to block passage of the beam as protection to workers in the very unlikely situation that a beam would be directed into the wrong area.

The beam line that would be bent toward the target would contain additional diagnostic and beam-steering components. This equipment would raster the proton beam back and forth across the front face of the target to provide uniform heating of the target. A large-capacity vacuum pumping system would maintain the beam transport line and the target assembly at pressures in the microtor region.

F.2.2.6 Target Building

The building that houses the target would be a massive concrete structure with a 23- by 62-meter (75.5- by 203.6-foot) footprint. The beam stop and the target assembly tank occupy about one-half of the building. The roof height at that location would be 27 meters (23 feet). The beam centerline has a 100-meter (328.4-foot) reference elevation. The roof would be at 121 meters (397.4 feet) and the floor would be at 94 meters (308.7 feet).

The building would be oriented so that the long side would be at right angles to the accelerator. The straight-ahead beam line runs parallel to the short side and would be 6 meters into the building from the end wall. The tune-up beam stop would be located on that line. The target assembly tank would be located 15 meters (49.3 feet) from the short wall (9 meters [29.6 feet] from the beam stop). The beam transport system bends the beam about 30 degrees to enter the tank. The tank rests on 3 meters (9.9 feet) of steel that would be stacked on the floor. Additional steel and concrete shielding surrounds the beam stop and tank. The top of the shielding would be 13 meters (42.7 feet) above the floor (elevation of 107 meters [351.4 feet]).

A remotely operable crane covers the large area of the deck. The crane hook rises 9 meters (29.6 feet) above the deck floor.

Three hot cells with 1-meter-thick (3.3-foot-thick) walls would be lined up in a row down the centerline of the first floor. The space on either side of the row of hot cells forms operating galleys for the remote-handling operations within the cells. The exterior wall of the galley floor would be indented under the operating deck. The indentation forms a truck unloading station. Large hatches over the station allow the crane hook to lift heavy loads from trucks into the operating deck.

The hot cell closest to the tank houses the purification systems for the target assembly and beam stop water systems. The middle cell would be used for demounting the target assembly from its shield plug and for disassembling the irradiated target and packaging it for shipment.

The third cell would be used for general maintenance and repair. It would also be used for preparing a new target for installation, especially if old components were reused.

Hatches in the operating deck provide crane and remote-handling equipment access to several locations below the deck. Access would be provided to each hot cell and to several storage wells located within the massive beam stop and tank shielding. The primary cooling systems, heavy water (deuterium oxide) for the target and light water (hydrogen oxide) for the blanket, would also be located under hatches in pits within the shielding.

The floor, exterior walls, and roof of the operating deck would be at least 1-meter-thick concrete, which would provide both structural strength to the building and shielding from the radiation sources within the room during target transfers. All exterior penetrations for services would be sealed, and all pipes and ducts contain valves, which permit the building to be sealed airtight if necessary in an accident situation.

F.2.2.7 Cooling Systems

The proton beam passes through thin layers of heavy water used to cool the uranium in the target. The neutrons and other particles that would be created in the uranium also pass through the heavy water and scatter outward and interact with the light water cooling the neptunium in the blanket. Particle interactions with the oxygen atoms of water create every isotope lighter than oxygen, and several that would be heavier. Of all the isotopes created, beryllium-7 would be particularly bad because of its energetic 500-kiloelectron-volt gamma ray and relatively long half-life of 53 days. Calculations show that the proton beam alone creates 4.4×10^{13} beryllium-7 nuclei per second. Minute failures of the cladding would allow uranium and neptunium, and their fission fragments, to enter the cooling water.

This water would be a potential source of radioactive emissions; therefore, an intermediate water-cooling loop would be inserted between the water that cools the target and the water that flows through the air in the cooling towers. The three loops are known as primary, secondary, and tower.

The roles of the three cooling loops are:

- There would be two primary loops, one filled with heavy water to cool the target and the other filled with light water to cool the blanket and reflector. The components used in these loops would be shielded and provided with leak detectors. Ion exchange resin tanks used to remove impurities, including beryllium-7, would be located in shields. These two loops reject their heat to two primary heat exchangers. All the components of these two loops would be located in pits beneath the operating deck and would be designed to be remotely maintained and removed.

- The two primary heat exchangers would be cooled by the secondary cooling loop. All the components of this loop (except the primary heat exchangers) would be in a mechanical equipment room outside the shielded target building.

- The secondary loop heat exchanger would be cooled by water that flows through wet heat exchangers that reject the heat in the water to air.

The heat removal requirements for the two primary cooling systems would be heavy water cooling the target (5.1 megawatts) and light water cooling the blanket and reflector (1.3 megawatts). The heat removal requirement for the secondary and tower cooling systems would be 6.4 megawatts.

F.3 REFERENCES

TechSource (TechSource, Inc.), 2000, *Nuclear Infrastructure PEIS Data Submittal for Accelerators*, Santa Fe, NM, July 24.

Appendix G
Methods for Assessing Environmental Impacts

This appendix briefly describes the methods used to assess the potential direct, indirect, and cumulative effects of the alternatives in this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]). Included are impact assessment methods for land resources, noise, air quality, water resources, geology and soils, ecological resources, cultural and paleontological resources, socioeconomics, waste management, and cumulative impacts. Each section includes a description of the affected resource and the impact assessment method. Impact assessment methods are described separately, as appropriate, for alternatives involving existing facilities and for those involving the new accelerator(s) or a new research reactor at a generic U.S. Department of Energy (DOE) site. Descriptions of the methods for the evaluation of human health effects from normal operations, facility accidents, and transportation, and for environmental justice are presented in Appendixes H, I, J, and K, respectively.

Impact analyses vary for each resource area. For air quality, for example, estimated pollutant emissions from the candidate facilities were compared with appropriate regulatory standards or guidelines. Comparison with regulatory standards is a commonly used method for benchmarking environmental impacts and is done here to provide perspective on the magnitude of identified impacts. For waste management, waste generation ratios were compared with the capacities of waste management facilities. Impacts within each resource area were analyzed consistently; that is, the impact values were estimated using a consistent set of input variables and computations. Moreover, efforts were made to ensure that calculations in all areas used accepted protocols and up-to-date models.

Baseline conditions at the three DOE sites (Oak Ridge Reservation [ORR], Idaho National Engineering and Environment Laboratory [INEEL], and the Hanford Site [Hanford]) assessed in this NI PEIS include present actions at each site. Option 1 of the No Action Alternative was used as the basis for the comparison of impacts that would occur under implementation of the other options and alternatives.

G.1 LAND RESOURCES

G.1.1 Land Use

G.1.1.1 Description of Affected Resources

Land use includes the land on and adjacent to each candidate site, the physical features that influence current or proposed uses, pertinent land use plans and regulations, and land ownership and availability. The region of influence for land use varies due to the extent of land ownership, adjacent land use patterns and trends, and other geographic or safety considerations.

G.1.1.2 Description of Impact Assessment

The amount of land disturbed and conformity with existing land use were considered in order to evaluate impacts **(Table G–1)**. Conformity with existing land use was evaluated for each alternative. However, land disturbance was considered only for those alternatives involving new construction. These alternatives include the Fuels and Materials Examination Facility (FMEF) at Hanford, which requires a new stack, and the new accelerator(s) or a research reactor at a generic DOE site. For the new stack at FMEF, the general location and amount of land to be disturbed is known; thus, impacts to land use may be determined. However, the location of the new accelerator(s) or research reactor is unknown, and the acreage required is only an approximation.

Table G–1 Impact Assessment Protocol for Land Resources

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Land area used	Site acreage	Facility acreage requirement	Acreage converted to project use
Compatibility with existing or future facility land use	Existing facility land use configurations	Location of facility on the site; expected modifications of facility activities and missions to accommodate the alternatives	Incompatibility with existing or future facility land use
Visual resources	Current Visual Resource Management classification	Location of facility on the site; facility dimensions and appearance	Change in Visual Resource Management classification

Thus, the evaluation of impacts for these new facilities are addressed in general terms. In order to determine the range of potential effects from new facilities, the analysis considered potential impacts from construction and operation at both a disturbed and an undisturbed location at a generic DOE site.

G.1.2 Visual Resources

G.1.2.1 Description of Affected Resources

Visual resources are the natural and human-created features that give a particular landscape its character and aesthetic quality. Landscape character is determined by the visual elements of form, line, color, and texture. All four elements are present in every landscape; however, they exert varying degrees of influence. The stronger the influence exerted by these elements in a landscape, the more interesting the landscape. The region of influence for visual resources includes the geographic area from which the candidate facilities may be seen.

G.1.2.2 Description of Impact Assessment

Impacts to visual resources may be determined by evaluating whether or not the Bureau of Land Management Visual Resource Management classifications of the candidate sites would change as a result of the proposed action (DOI 1986) (Table G–1). Existing classifications were derived from an inventory of scenic qualities, sensitivity levels, and distance zones for particular areas. For those alternatives involving existing facilities at candidate DOE sites, alterations to visual features may be readily evaluated and the impact on the current Visual Resource Management classification determined. For those alternatives involving construction and operation of the new accelerator(s) or research reactor at a generic DOE site, the visual characteristics of the site are unknown. Thus, impacts are addressed in a general manner. In order to determine the range of potential visual effects from new facilities at a generic DOE site, the analysis considered potential impacts from construction and operation at both a disturbed and an undisturbed location at the generic site. Impacts associated with the use of an existing CLWR are also described in a general manner because its location is not known.

G.2 NOISE

G.2.1 Description of Affected Resources

Sound results from the compression and expansion of air or some other medium when an impulse is transmitted through it. Sound requires a source of energy and a medium for transmitting the sound wave. Propagation of sound is affected by various factors, including meteorology, topography, and barriers. Noise

is undesirable sound that interferes or interacts negatively with the human or natural environment. Noise may disrupt normal activities (e.g., hearing, sleep), damage hearing, or diminish the quality of the environment.

Sound-level measurements used to evaluate the effects of nonimpulsive sound on humans are compensated by an A-weighting scale that accounts for the hearing response characteristics (i.e., frequency) of the human ear. Sound levels are expressed in decibels, or in the case of A-weighted measurements, decibels A-weighted The U.S. Environmental Protection Agency (EPA) has developed noise-level guidelines for different land use classifications. Some states and localities have established noise control regulations or zoning ordinances that specify acceptable noise levels by land use category.

Noise from facility operations and associated traffic could affect human and animal populations. The region of influence for each candidate site includes the site and surrounding area, including transportation corridors, where proposed activities might increase noise levels. Transportation corridors most likely to experience increased noise levels are those roads within a few miles of the site boundary that carry most of the site's employee and shipping traffic.

Sound-level data representative of site environs were obtained from existing reports. The acoustic environment was further described in terms of existing noise sources for each candidate site. Generic sites are described in terms of existing noise characteristics at existing DOE and nuclear power plant sites.

G.2.2 Description of Impact Assessment

Noise impacts associated with the alternatives may result from modification (including construction of a new stack at FMEF) and operation of existing facilities, as well as increased traffic **(Table G–2)**. Impacts from facility modification and operation were assessed according to the types of noise sources and the locations of the candidate facilities relative to the site boundary. Potential noise impacts from traffic were based on the likely increase in traffic volume. Possible impacts to wildlife were evaluated based on the possibility of sudden loud noises occurring during facility modification and operation.

Table G–2 Impact Assessment Protocol for Noise

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Noise	Identification of sensitive offsite receptors (e.g., nearby residences); description of sound levels in the vicinity of the site	Description of major construction, modification, and operational noise sources; shipment and workforce traffic estimates	Increase in day/night average sound level at sensitive receptors

Acoustic impacts from facility construction, modification, and operation at generic sites were assessed according to the types of new noise sources and characteristics identified for a generic site. The potential for traffic noise impacts is discussed, but the change in traffic noise levels at a generic site could not be assessed without site-specific data.

G.3 AIR QUALITY

G.3.1 Description of Affected Resources

Air pollution refers to the introduction, directly or indirectly, of any substance into the air that could endanger human health and harm living resources and ecosystems, as well as material property and impair or interfere with the comfortable enjoyment of life and other legitimate uses of the environment. For the purpose of this

NI PEIS, only outdoor air pollutants were addressed. They may be in the form of solid particles, liquid droplets, gases, or a combination of these forms. Generally, they can be categorized as primary pollutants (those emitted directly from identifiable sources) and secondary pollutants (those produced in the air by interaction between two or more primary pollutants, or by reaction with normal atmospheric constituents that may be influenced by sunlight). Air pollutants are transported, dispersed, or concentrated by meteorological and topographical conditions. Thus, air quality is affected by air pollutant emission characteristics, meteorology, and topography.

Ambient air quality in a given location can be described by comparing the concentrations of various pollutants in the atmosphere with the appropriate standards. Ambient air quality standards have been established by Federal and state agencies, allowing an adequate margin of safety for the protection of public health and welfare from the adverse effects of pollutants in the ambient air. Pollutant concentrations higher than the corresponding standards are considered unhealthy; those below such standards, acceptable.

The pollutants of concern are primarily those for which Federal and state ambient air quality standards have been established, including criteria air pollutants, hazardous air pollutants, and other toxic air compounds. Criteria air pollutants are those listed in 40 CFR Part 50, "National Primary and Secondary Ambient Air Quality Standards." Hazardous air pollutants and other toxic compounds are those listed in Title I of the Clean Air Act as amended, those regulated by the National Emissions Standards for Hazardous Air Pollutants (NESHAPs), and those that have been proposed or adopted for regulation by the respective state, or are listed in state guidelines. NESHAPs (40 CFR Part 61) is also discussed in Section H.2.1.1. States may set ambient standards that are more stringent than the national ambient air quality standards. The more stringent of the state or Federal standards for each site is shown in this document. Also of concern are air pollutant emissions that may contribute to the depletion of stratospheric ozone or global warming.

Areas with air quality better than the National Ambient Air Quality Standards (NAAQS) for criteria air pollutants are designated as being in attainment, while areas with air quality worse than the NAAQS for such pollutants are designated as nonattainment. Areas may be designated as unclassified when sufficient data for attainment status designation are lacking. Attainment status designations are assigned by county, metropolitan statistical area, consolidated metropolitan statistical area, or portions thereof, or air quality control regions. Air Quality Control Regions designated by EPA are listed in 40 CFR Part 81, "Designation of Areas for Air Quality Planning Purposes." ORR, INEEL, and Hanford are all located in attainment areas (40 CFR Sections 81.313, 81.343, and 81.348).

For locations that are in an attainment area for criteria air pollutants, Prevention of Significant Deterioration regulations limit pollutant emissions from new or modified sources and establish allowable increments of pollutant concentrations. Three Prevention of Significant Deterioration classifications are specified with the criteria established in the Clean Air Act. Class I areas include national wilderness areas, memorial parks larger than 2,020 hectares (5,000 acres), national parks larger than 2,430 hectares (6,000 acres), and areas that have been redesignated as Class I. Class II areas are all areas not designated as Class I. No Class III areas have been designated (42 U.S.C. 7472, Title I, Section 162).

ORR, INEEL, and Hanford are all in Class II areas. In addition, ORR is 48.3 kilometers (30 miles) from the Great Smoky Mountains Class I area, and INEEL is 53 kilometers (33 miles) from the Craters of the Moon Wilderness Area Class I area. There are no Prevention of Significant Deterioration Class I areas within 100 kilometers (62 miles) of Hanford (DOE 1996; DOE 1999a). The recent designation of the Hanford Reach as a national monument may eventually lead to the redesignation of this area, which includes part of Hanford and adjoining areas, as a Prevention of Significant Deterioration Class I area.

The region of influence for air quality encompasses an area surrounding a candidate site that is potentially affected by air pollutant emissions caused by the alternatives. The air quality impact area normally evaluated is the area in which concentrations of criteria pollutants would increase more than a significant amount in a Class II area (i.e., on the basis of averaging period: 1 microgram per cubic meter for annual, 5 micrograms per cubic meter for 24 hours, 500 micrograms per cubic meters for 8 hours, 25 micrograms per cubic meters for 3 hours, and 2,000 micrograms for 1 hour [40 CFR Section 51.165]). Generally, this covers a few kilometers downwind from the source. Further, for sources within 100 kilometers (60 miles) of a Class I area, the air quality impact area evaluated would include the Class I area if the increase in concentration were greater than 1 microgram per cubic meter (24-hour average). The area of the region of influence depends on emission source characteristics, pollutant types, emission rates, and meteorological and topographical conditions. For the purpose of this analysis, where most of the candidate sites are large, impacts were evaluated at the site boundary and roads within the sites to which the public has access, plus any additional area in which contributions to pollutant concentrations are expected to exceed significance levels.

Baseline air quality is typically described in terms of pollutant concentrations modeled for existing sources at each candidate site and background air pollutant concentrations measured near the sites. For this analysis, concentrations for existing sources were obtained from existing source documents such as the *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement* (DOE 1999c) and *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE 2000) and from modeling of concentrations using recent emissions inventories and the Industrial Source Complex (ISCST3) model (EPA 1995a, 2000). These concentrations were compared with Federal and state standards or guidelines (**Table G–3**). To determine human health risk, modeling outputs on chemical concentrations in air were weighed against chemical-specific toxicity values.

Table G–3 Impact Assessment Protocol for Air Quality

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Criteria air pollutants and other regulated pollutants[a]	Measured and modeled ambient concentrations (micrograms per cubic meter) from existing sources at site	Emission (kilograms per year) of air pollutants from facility; source characteristics (e.g., stack height and diameter, exit temperature and velocity)	Concentration of alternative and total site concentration of each pollutant at or beyond site boundary, or within boundary on public road compared to applicable standard
Toxic and hazardous air pollutants[b]	Measured and modeled ambient concentrations (micrograms per cubic meter) from existing sources at site	Emission rate (kilograms per year) of pollutants from facility; source characteristics (e.g., stack height and diameter, exit temperature and velocity)	Concentration of alternative and total site concentration of each pollutant at or beyond site boundary, or within boundary on public road compared to applicable standard

a. Carbon monoxide; hydrogen fluoride; lead; nitrogen oxides; ozone; particulate matter with an aerodynamic diameter less than or equal to 10 microns; particulate matter with an aerodynamic diameter less than 2.5 microns; sulfur dioxide; total suspended particulates.

b. Clean Air Act, Section 112, hazardous air pollutant; pollutants regulated under the National Emissions Standard for Hazardous Air Pollutants; and other state-regulated pollutants.

G.3.2 Description of Impact Assessment

Potential air quality impacts of pollutant emissions from facility modification and normal operations were evaluated for those alternatives associated with the Fast Flux Test Facility (FFTF) restart and the use of existing facilities. This assessment included a comparison of pollutant concentrations from each alternative with applicable Federal and state ambient air quality standards. If both Federal and state standards exist for

a given pollutant and averaging period, compliance was evaluated using the more stringent standard. Operational air pollutant emissions data for each alternative were based on conservative engineering analyses.

For each alternative, contributions to offsite air pollutant concentrations were modeled on the basis of guidance presented in EPA's "Guidelines on Air Quality Models" (40 CFR Part 51, Appendix W). The EPA-recommended screening model SCREEN3 (EPA 1995b), was selected as an appropriate model to perform the air dispersion modeling because it is designed to support the EPA regulatory modeling program and predicts conservative worst-case impacts. The SCREEN3 model was used to estimate maximum 1-hour concentrations. Appropriate regulatory scaling factors were used to estimate concentrations for other averaging periods based on the maximum 1-hour concentration (3 hours, 0.9; 8 hours, 0.7; 24 hours, 0.4; annual, 0.05) (Brode 1988).

The modeling analysis incorporated conservative assumptions, which tend to overestimate pollutant concentrations. The maximum modeled concentration for each pollutant and averaging time was selected for comparison with the applicable standard. The concentrations evaluated were the maximum occurring at or beyond the site boundary and a public access road, or other publicly accessible area within the site. Available monitoring data, which reflect both onsite and offsite sources, were also taken into consideration. Concentrations of the criteria air pollutants, hazardous air pollutants, and toxic air compounds were presented for each alternative. A set of worst-case meteorological conditions were used in the air quality modeling.

Ozone is typically formed as a secondary pollutant in the ambient air (troposphere). It is formed in the presence of sunlight from the mixing of primary pollutants, such as nitrogen oxides, and volatile organic compounds that emanate from vehicular (mobile), natural, and other stationary sources. Ozone is not emitted directly as a pollutant from the candidate sites. Although ozone may be regarded as a regional issue, specific ozone precursors, notably nitrogen dioxide and volatile organic compounds, were analyzed as applicable to the alternatives under consideration.

The Clean Air Act, as amended, required that Federal actions conform to the host state's "state implementation plan." A state implementation plan provides for the implementation, maintenance, and enforcement of NAAQS for the six criteria pollutants: sulfur dioxide, particulate matter with an aerodynamic diameter less than or equal to 10 microns, carbon monoxide, ozone, nitrogen dioxide, and lead. Its purpose is to eliminate or reduce the severity and number of violations of NAAQS and to expedite the attainment of these standards. No department, agency, or instrumentality of the Federal Government shall engage in or support in any way (i.e., provide financial assistance for, license or permit, or approve) any activity that does not conform to an applicable implementation plan. The final rule for "Determining Conformity of General Federal Actions to State or Federal Implementation Plans" (58 FR 63214) took effect on January 31, 1994. ORR, INEEL, and Hanford are within areas currently designated as attainment for criteria air pollutants. Therefore, the alternatives being considered at these sites are not affected by the provisions of the conformity rule.

Continued operation of a CLWR at an unknown site would result in a small amount of nonradiological air pollutants being released to the atmosphere, mainly due to the requirement of periodical testing of the emergency diesel generators. Air quality impacts associated with a CLWR were addressed as a contribution from the facility operation.

Air quality impacts from the new accelerator(s) or a research reactor are discussed for construction and operation at a generic DOE site. The potential for an increase in nonradiological air emissions is attributed to the supporting facility equipment and construction activities, such as increased employee vehicles, truck traffic, and diesel generator use.

Emissions of potential stratospheric ozone-depleting compounds such as chlorofluorocarbons were not evaluated, as no emissions of these pollutants were identified in the conceptual engineering design reports.

G.4 WATER RESOURCES

G.4.1 Description of Affected Resources

Water resources are the surface and subsurface waters that are suitable for human consumption, aquatic or wildlife propagation, agricultural purposes, irrigation or industrial/commercial purposes. The region of influence used for water resources encompasses those surface water and groundwater systems which could be impacted by water withdrawals, effluent discharges, and spills or stormwater runoff associated with construction and operation of the candidate facilities.

G.4.2 Description of Impact Assessment

G.4.2.1 Water Use

This analysis involved the review of engineering estimates of expected water use and effluent discharges associated with each alternative, and the impacts on local water availability and quality, including surface water and groundwater. Impacts on water use were assessed by determining changes in the volume of current water usage and effluent discharges as a result of the proposed activities. The determination of the impacts on water usage and effluent discharge are summarized in **Table G–4**.

Table G–4 Impact Assessment Protocol for Water Use and Effluent Discharge

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Surface water availability	Surface waters near the facilities, including average flow and current usage	Volume of withdrawals from, and discharges to, surface waters	Changes in availability to downstream users of water for drinking, irrigation, or animal feeding
Groundwater availability	Groundwater near the facilities, including existing water rights for major water users and current usage	Volume of withdrawals from, and discharges to, groundwater	Changes in availability of groundwater for human consumption, irrigation, or animal feeding

If the determination reflected an increase in water use or effluent discharge, then an evaluation of the design capacity of the water and effluent treatment facilities was made to determine whether the design capacity would be exceeded by the additional flow. If the combined flow (i.e., the existing flow plus those from the proposed activities), was less than the design capacity of the water supply systems and effluent treatment plants, then it was assumed that there would be no impact on water availability for local users, nor on the receiving stream from effluent discharges. Because flows from the candidate facilities were generally found not to exceed the design capacity of existing water supply systems or effluent treatment facilities, no additional analyses were performed.

G.4.2.2 Water Quality

The water quality impact assessment for this NI PEIS analyzed how effluent discharges to surface water, as well as discharges reaching groundwater, from the candidate facilities would affect current water quality. The determination of the impacts of the alternatives is summarized in **Table G–5** and consisted of a comparison of the projected water quality with relevant regulatory standards such as the Clean Water Act, Safe Drinking

Water Act, state regulations, and existing permit conditions. Separate analyses were conducted for surface water and groundwater impacts.

Table G–5 Impact Assessment Protocol for Water Quality

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Surface water quality	Surface waters near the facilities in terms of stream classifications and changes in water quality	Expected contaminants and contaminant concentrations in discharges to surface water	Compliance of discharge to surface water with relevant standards of Clean Water Act or with state regulations and existing National Pollutant Discharge Elimination System permits
Groundwater quality	Groundwater near the facilities in terms of classification, presence of designated sole source aquifers, and changes in quality of groundwater	Expected contaminants and contaminant concentrations in discharges that could reach groundwater	Concentrations of contaminants in groundwater exceeding standards established in accordance with Safe Drinking Water Act or state regulations

Surface Water Quality. The evaluation of the surface water quality impacts focused on the quality and quantity of the effluent to be discharged and the quality of the receiving stream upstream and downstream from the discharge. The evaluation of effluent quality featured review of the expected parameters, such as the design average and maximum flows, as well as the effluent parameters reflected in the existing or expected National Pollutant Discharge Elimination System (NPDES) permit. Those parameters include total suspended solids, metals, organic and inorganic chemicals, radionuclides, and any other parameters that affect the local environment. Water quality management practices were reviewed to ensure that NPDES permit limitations would be met. Factors that currently degrade water quality were also identified.

Groundwater Quality. Potential groundwater quality impacts associated with effluent discharges were examined. Engineering estimates of contaminant concentrations were weighed against Federal and state groundwater quality standards, effluent limitations, and drinking water standards to determine the impacts of each alternative. Also evaluated were the consequences for groundwater use in the area.

The water resources impact assessment for activities involving generic DOE or CLWR sites was generally conducted in the same manner as described above. However, as the exact nature of the sites is not known, it was necessary to make bounding assumptions regarding the range of potential resource conditions that could be present and potentially affected (e.g., surface or groundwater) coupled with using highly conservative estimates of expected impacts (e.g., water withdrawals). This was done to better ensure that the resulting analysis would be applicable to any site and to provide a comparative basis for the impacts assessment.

G.5 GEOLOGY AND SOILS

G.5.1 Description of Affected Resources

Geologic resources include consolidated and unconsolidated earth materials, including mineral assets such as ore and aggregate materials, and fossil fuels such as coal, oil, and natural gas. Geologic conditions include hazards such as earthquakes, faults, volcanoes, landslides, and land subsidence. Soil resources include the loose surface materials of the earth in which plants grow, usually consisting of mineral particles from disintegrating rock, organic matter, and soluble salts. Prime farmland, as defined in 7 CFR Part 657, is land

that contains the best combination of physical and chemical characteristics for producing crops. It includes cropland, pasture land, rangeland, and forest land.

The region of influence for geology and soils includes all areas subject to disturbance by construction and operation or decontamination and decommissioning of the candidate facilities, as applicable, and those areas beneath existing or new facilities that would remain inaccessible for the life of the facilities.

Geology and soil conditions that could affect the integrity and safety of the candidate facilities include large-scale geologic hazards (e.g., earthquakes) and attributes of the soil and bedrock beneath the new facilities. The area within which these conditions exist, and which could impact existing or new facilities, constitute the region of influence for this resource area.

G.5.2 Description of Impact Assessment

The geology and soils impact analysis (**Table G–6**) considered the risks to the existing and new facilities of large-scale geologic hazards such as faulting and earthquakes, lava extrusions and other volcanic activity, landslides, and sinkholes, (i.e., conditions that tend to affect broad expanses of land). While evidence of impacts in facility-specific areas was developed, as appropriate, there was no attempt to revisit the basic conclusions of the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement (Storage and Disposition PEIS)* (DOE 1996) as reviewed in the *Surplus Plutonium Disposition Final Environmental Impact Statement (Surplus Plutonium Disposition EIS)* (DOE 1999a) for INEEL and Hanford in this regard: that the risks of such hazards to storage and disposition facilities at the candidate sites are acceptable. The findings of those analyses, which focused on the presence of the hazard and the distance of the facilities from it, were accepted as generally applicable to the candidate facilities. Because no major construction is associated with any of the alternatives (which involve only existing facilities), geologic resources and soils (including prime farmland) would not be affected.

Table G–6 Impact Assessment Protocol for Geology and Soils

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Geologic hazards	Presence of geologic hazards within the region of influence	Location of facility on the site	Potential for damage to facilities
Prime farmland soils	Presence of prime farmland within the region of influence	Location of facility on the site	Loss of prime farmland

The geology and soils impact assessment for activities involving generic DOE or CLWR sites was generally conducted in the same manner as described above. However, as the exact nature of the sites is not known, it was necessary to make bounding assumptions regarding the range of potential geologic and soils conditions that could be present (e.g., subsurface composition, proximity of faults) coupled with using highly conservative estimates of expected impacts (e.g., land disturbance). This was done to better ensure that the resulting analysis would be applicable to any site and to provide a more comparative basis for the impacts assessment. If a DOE or CLWR site were selected, subsequent National Environmental Policy Act (NEPA) actions would be required.

G.6 ECOLOGICAL RESOURCES

G.6.1 Description of Affected Resources

Ecological resources include terrestrial and aquatic resources (plants and animals), wetlands, and threatened and endangered species. The region of influence used for the ecological resource analysis encompassed the area potentially disturbed by construction and operation of the candidate facilities.

Terrestrial resources are defined as those plant and animal species and communities that are most closely associated with the land; for aquatic resources, a water environment. Wetlands are defined by the U.S. Army Corps of Engineers and EPA as ". . . those areas that are inundated or saturated by surface or groundwater at a frequency and duration sufficient to support, and that under normal circumstances do support, a prevalence of vegetation typically adapted for life in saturated soil conditions. Wetlands generally include swamps, marshes, bogs, and similar areas" (33 CFR Section 328.3).

Endangered species are defined under the Endangered Species Act of 1973 as those in danger of extinction throughout all or a large portion of their range. Threatened species are defined as those species likely to become endangered within the foreseeable future. The U.S. Fish and Wildlife Service and the National Marine Fisheries Service propose species to be added to the lists of threatened and endangered species. They also maintain a list of "candidate" species for which they have evidence that listing may be warranted but for which listing is currently precluded by the need to list species more in need of Endangered Species Act protection. Candidate species do not receive legal protection under the Endangered Species Act, but should be considered in project planning in case they are listed in the future. Critical habitat for threatened and endangered species is designated by the U.S. Fish and Wildlife Service or the National Marine Fisheries Service. Critical habitat is defined as specific areas that contain physical and biological features essential to the conservation of species and that may require special management consideration or protection. States may also designate species as endangered, threatened, sensitive protected, in need of management, of concern, monitored, or species of special concern.

G.6.2 Description of Impact Assessment

Impacts to ecological resources may occur as a result of land disturbance, water use, air and water emissions, human activity, and noise associated with project implementation **(Table G–7)**. Each of these factors was considered when evaluating potential impacts from the proposed action. All alternatives, except those involving construction and operation of the new accelerator(s) or research reactor, involve only internal facility modification or limited new construction (a new stack for FMEF). Thus, direct impacts to ecological resources from land disturbance and human activity would be minimal. For alternatives involving construction and operation of the new accelerator(s) or research reactor at a generic DOE site, potential impacts to terrestrial resources were determined based on the approximate acreage of land disturbed. Because a specific facility location is not known, the analysis generally considered impacts at both a disturbed and an undisturbed location at a generic DOE site. Impacts to terrestrial and aquatic ecosystems and wetlands from water use and air and water emissions were evaluated based on the results of the analysis conducted for air quality and water resources. Consultations were conducted with the U.S. Fish and Wildlife Service and National Marine Fisheries Service to comply with Section 7 of the Endangered Species Act. Consultations were also conducted with appropriate state agencies (see Table 5–3). The determination of impacts to threatened and endangered species was based on similar factors as noted above for terrestrial resources, wetlands, and aquatic resources.

Table G–7 Impact Assessment Protocol for Ecological Resources

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Terrestrial resources	Vegetation and wildlife within vicinity of facilities	Facility location, air and water emissions, and noise	Loss or disturbance to terrestrial habitat; emissions and noise values above levels shown to cause impacts to terrestrial resources
Aquatic resources	Aquatic resources within vicinity of facilities	Facility air and water emissions, water source and quantity, and wastewater discharge location and quantity	Discharges above levels shown to cause impacts to aquatic resources; changes in water withdrawals and discharges
Wetlands	Wetlands within vicinity of facilities	Facility location, air and water emissions, and wastewater discharge quantity and location	Loss or disturbance to wetlands; discharge to wetlands
Threatened and endangered species	Threatened and endangered species within vicinity of facilities	Facility location, air and water emissions, noise, water source and quantity, and wastewater discharge location and quantity	Measures similar to those noted above for terrestrial and aquatic resources

G.7 CULTURAL AND PALEONTOLOGICAL RESOURCES

G.7.1 Description of Affected Resources

Cultural resources are the indications of human occupation and use of the landscape as defined and protected by a series of Federal laws, regulations, and guidelines. For this NI PEIS, potential impacts were assessed separately for each of the three general categories of cultural resources: prehistoric, historic, and Native American. Paleontological resources are the physical remains, impressions, or traces of plants or animals from a former geological age, and may be sources of information on paleoenvironments and the evolutionary development of plants and animals. Although not governed by the same historic preservation laws as cultural resources, they could be affected by the proposed action in much the same manner.

Prehistoric resources are physical remains of human activities that predate written records; they generally consist of artifacts that may alone or collectively yield otherwise inaccessible information about the past. Historic resources consist of physical remains that postdate the emergence of written records; in the United States, they are architectural structures or districts, archaeological objects, and archaeological features dating from 1492 and later. Ordinarily, sites less than 50 years old are not considered historic, but exceptions can be made for such properties if they are of particular importance, such as structures associated with Cold War themes. Native American resources are sites, areas, and materials important to Native Americans for religious or heritage reasons. Such resources may include geographical features, plants, animals, cemeteries, battlefields, trails, and environmental features. The region of influence for the cultural and paleontological resource analysis encompassed the area potentially disturbed by construction and operation of the candidate facilities.

G.7.2 Description of Impact Assessment

The analysis of impacts to cultural and paleontological resources addressed potential direct and indirect impacts at each candidate site, including an unspecified CLWR site and a generic DOE site (**Table G–8**).

Table G–8 Impact Assessment Protocol for Cultural and Paleontological

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Prehistoric resources	Prehistoric resources within the vicinity of facilities	Location of facility on the site	Potential for loss, isolation, or alteration of the character of prehistoric resources; introduction of visual, audible, or atmospheric elements out of character; neglect of resources listed or eligible for listing on the National Register of Historic Places
Historic resources	Historic resources within the vicinity of facilities	Location of facility on the site	Potential for loss, isolation, or alteration of the character of historic resources; introduction of visual, audible, or atmospheric elements out of character; neglect of resources listed or eligible for listing on the National Register of Historic Places
Native American resources	Native American resources within the vicinity of facilities	Location of facility on the site	Potential for loss, isolation, or alteration of the character of Native American resources; introduction of visual, audible or atmospheric elements out of character
Paleontological resources	Paleontological resources within the vicinity of facilities	Location of facility on the site	Potential for loss, isolation or alteration of paleontological resources

Potential indirect impacts include those associated with reduced access to a resource site, as well as impacts associated with increased traffic and visitation to sensitive areas. Direct impacts include those resulting from groundbreaking activities associated with new construction. Direct impacts would be associated with construction of the new accelerator(s) or research reactor at a generic DOE site. Because the specific location is unknown, however, impacts from new construction were addressed in a general manner. In order to determine the range of potential impacts, the analysis considered potential effects at both a disturbed and an undisturbed location at a generic DOE site. Impacts associated with the use of an existing CLWR at an unknown location were also addressed in a general manner. Consultations to comply with Section 106 of the National Historic Preservation Act were conducted with the various State Historic Preservation Officers. Consultations were also conducted with interested Native American tribes (see Table 5–3).

G.8 SOCIOECONOMICS

G.8.1 Description of Affected Resources

Socioeconomic impacts are defined in terms of changes to the demographic and economic characteristics of a region. The number of jobs created by the proposed action would affect regional employment, income, and expenditures. Job creation is characterized by two types: (1) construction-related jobs, transient in nature and short in duration, and thus less likely to impact public services; and (2) jobs related to plant operations, required for a decade or more, and thus possibly creating additional service requirements in the region of influence.

The socioeconomic environment is made up of two geographic regions, the regional economic area and region of influence. Regional economic areas are made up of regional economies and include descriptions of industrial and service sector characteristics and their linkages to the communities within a region. These linkages determine the nature and magnitude of any effect associated with a change in regional economic activity. For example, as work expands within a region, the money spent on accomplishing this work flows into the local economy; it is spent on additional jobs, goods, and services within the regional economic area.

Similarly, potential demographic impacts were assessed for the region of influence. The region of influence could represent a smaller geographic area, one in which only the housing market and local community services would be significantly affected by a given alternative. Site-specific regions of influence were identified as those counties in which approximately 90 percent or more of the site's workforce reside. This distribution reflects an existing residential preference for people currently employed at the sites and was used to estimate the distribution of new workers supporting the alternatives.

G.8.2 Description of Impact Assessment

For each regional economic area, data were compiled on the current socioeconomic conditions, including unemployment rates, economic industrial and service sector activities, and the civilian labor force. The workforce and cost requirements of each alternative were determined in order to measure their possible effect on these socioeconomic conditions. Although workforce requirements may be able to be filled by employees already working at DOE sites, it was assumed these requirements would be filled by new employees to ensure that the maximum impact was assessed. For each region of influence, census statistics were also compiled on population, housing demand, and community services. U.S. Bureau of the Census population forecasts for the regions of influence were combined with overall projected workforce requirements for each of the alternatives being considered at each candidate site to determine the extent of impacts on housing demand and levels of community services (**Table G–9**).

For those alternatives involving construction and operation of the new accelerator(s) or research reactor at a generic DOE site, the socioeconomic characteristics of the site are unknown. Impacts cannot be measured until candidate sites are identified. Therefore, if one of these alternatives were selected, additional NEPA documentation would be required, which would address the socioeconomic impacts.

Impacts associated with the use of an existing CLWR were addressed in a general manner as the location is unknown.

Table G–9 Impact Assessment Protocol for Socioeconomics

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Regional Economic Characteristics			
Workforce requirements	Site workforce projections from DOE sites	Estimated construction and operating staff requirements and timeframes	Workforce requirements added to sites' workforce projections
Regional economic area civilian labor force	Labor force projections based on state population projections	Estimated construction and operating staff requirements and timeframes	Workforce requirements as a percentage of the civilian labor force
Unemployment rate	1996 unemployment rates in counties surrounding sites and in host states	Estimated construction and operating staff requirements	Projected change in unemployment rates
Population and Housing			
Population	Latest available population projection estimates from the U.S. Bureau of the Census	Estimated contribution to projected population	Projected change in population projection
Housing—Percent of occupied housing units	Latest available rates from the U.S. Bureau of the Census	Assess potential need for new housing units to meet workforce requirements	Projected change in workforce
Community Services			
Education			
Percent operating capacity for school districts in the region of influence	Latest available rates from the U.S. Bureau of the Census	Assess potential need for new schools	Projected change in student population
Teacher-to-student ratio	Latest available rates from the U.S. Bureau of the Census	Assess potential need for additional teachers	Projected change to maintain current teacher-to-student ratio
Public safety—Ratio of police and firefighters to 100,000 residents	Latest available rates from the U.S. Bureau of the Census	Assess potential need for additional officers and firefighters	Projected change to maintain the current police officer/firefighter to population ratio
Health care—Number of hospital beds and physicians per 100,000 residents	Latest available rates from the U.S. Bureau of the Census	Assess potential need for additional hospitals and physicians	Projected change in the availability of hospital beds/physicians to population ratio

G.9 WASTE MANAGEMENT

G.9.1 Description of Affected Resources

Depending on the alternative, construction and operation of the candidate facilities, as well as the permanent deactivation of FFTF and decontamination and decommissioning of the new accelerator(s), research reactor, and support facility, would generate several types of waste. Such wastes may include the following:

- **High-level radioactive:** The highly radioactive waste resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from the liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation.

- **Transuranic:** Waste containing more than 100 nanocuries per gram of alpha-emitting transuranic isotopes, with half-lives greater than 20 years, except for (1) high-level radioactive waste; (2) waste that DOE has determined, with the concurrence of EPA, does not need the degree of isolation required by 40 CFR Part 191; and (3) waste that the U.S. Nuclear Regulatory Commission has approved for disposal, case-by-case in accordance with 10 CFR Part 61. Contact-handled transuranic waste is packaged transuranic waste whose external surface dose rate does not exceed 200 millirem per hour. Remote-handled transuranic waste is packaged transuranic waste whose external surface dose rate exceeds 200 millirem per hour. Mixed transuranic waste contains hazardous components regulated under the Resource Conservation and Recovery Act (RCRA).

- **Low-level radioactive:** Waste that contains radioactivity and is not classified as high-level radioactive waste, transuranic waste, or spent nuclear fuel, or the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material. Test specimens of fissionable material irradiated for research and development only, and not for the production of power or plutonium, may be classified as low-level radioactive waste, provided the transuranic concentration is less than 100 nanocuries per gram of waste.

- **Mixed low-level radioactive:** Low-level radioactive waste that also contains hazardous components regulated under RCRA.

- **Hazardous:** Under RCRA, a waste that, because of its characteristics, may (1) cause or significantly contribute to an increase in mortality or an increase in serious irreversible, or incapacitating reversible illness, or (2) pose a substantial present or potential hazard to human health or the environment when improperly treated, stored, transported, disposed of, or otherwise managed. Hazardous wastes appear on special EPA lists or possess at least one of the following characteristics: ignitability, corrosivity, reactivity, or toxicity. This category does not include source, special nuclear, or byproduct material as defined by the Atomic Energy Act.

- **Nonhazardous:** Discarded material including solid, liquid, semisolid, or contained gaseous material resulting from industrial, commercial, mining, and agricultural operations, and from community activities. This category does not include source, special nuclear, or byproduct material as defined by the Atomic Energy Act.

The alternatives could have an impact on existing site facilities devoted to the treatment, storage, and disposal of these categories of waste. Waste management activities in support of the proposed action would be contingent on Records of Decision issued for the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste (Waste Management PEIS)* (DOE 1997). The high-level radioactive waste Record of Decision, issued on August 12, 1999 (64 FR 46661), states that immobilized high-level radioactive waste will be stored at the site of generation until transfer to a geologic repository. The Record of Decision for transuranic waste, issued on January 20, 1998 (63 FR 3629), states that transuranic and transuranic mixed waste would be certified on site and shipped to a suitable geologic repository. According to the Record of Decision for hazardous waste, released on August 5, 1998 (63 FR 41810), DOE sites evaluated in this NI PEIS will continue to use offsite facilities for the treatment and disposal of major portions of their nonwastewater hazardous waste, with ORR continuing to treat some of its nonwastewater hazardous waste in existing facilities where economically feasible. Based on the Record of Decision for low-level radioactive waste and mixed low-level radioactive waste issued on February 18, 2000 (65 FR 10061), minimal treatment of low-level radioactive waste will be performed at all sites, and to the extent practical, onsite disposal of low-level radioactive waste will continue. Hanford and the Nevada Test Site will be made available to all DOE sites for the disposal of low-level radioactive waste. Mixed low-level radioactive waste analyzed in the *Waste Management PEIS* (DOE 1997)

will be treated at Hanford, INEEL, ORR, and the Savannah River Site and will be disposed of at Hanford and the Nevada Test Site.

G.9.2 Description of Impact Assessment

As shown in **Table G–10**, impacts were assessed by comparing the projected waste stream volumes generated from the proposed activities at each candidate site with that site's waste management capacities and generation rates. Only the impacts relative to the capacities of waste management facilities were considered; other environmental impacts of waste management facility operations (e.g., human health effects) are evaluated in other sections of this NI PEIS, or in other facility-specific or sitewide NEPA documents. Projected waste generation rates for the proposed activities were compared with site processing rates and capacities of those treatment, storage, and disposal facilities likely to be involved in managing the additional waste. The waste generation rates were provided by the sites' technical personnel.

Table G–10 Impact Assessment Protocol for Waste Management

Resource	Required Data		Measure of Impact
	Affected Environment	Alternative	
Waste management capacity High-level radioactive waste Transuranic waste Low-level radioactive waste Mixed low-level radioactive waste Hazardous waste Nonhazardous waste	Site generation rates (cubic meters per year) for each waste type Site management capacities (cubic meters) or rates (cubic meters per year) for potentially affected treatment, storage, and disposal facilities for each waste type	Generation rates (cubic meters per year) from facility operations for each waste type	Combination of facility waste generation volumes and other site generation volumes in comparison to the capacities of applicable waste management facilities

For the generic DOE site or CLWR site, projected waste stream volumes could not be compared to site waste management capacities and generation rates because a specific location was not identified.

G.10 CUMULATIVE IMPACTS

Cumulative impacts can result from individually minor but collectively significant actions taking place over a period of time (40 CFR Section 1508.7). The cumulative impact analysis for this NI PEIS involved combining the impacts of the alternatives (including No Action) with the impacts of other present and reasonably foreseeable activities in the region of influence. The regions of influence for different resources can vary widely in extent. For example, the regions of influence for waste management would generally be confined to the areal extent of each site, whereas the region of influence for human health would include the areas extending out to 80 kilometers (50 miles) from each site.

In general, cumulative impacts were calculated by adding the values for the baseline affected environment (i.e., conditions attributable to present actions by DOE and other public and private entities), the proposed action (or no action), and other future actions. This cumulative value was then weighed against the appropriate impact indicators (e.g., standards or number of fatalities) to determine the potential for impact. For this cumulative impact assessment, it was conservatively assumed that all facilities would operate concurrently at the candidate DOE sites. The selected indicators of cumulative impacts evaluated in this NI PEIS are shown in **Table G–11**.

Table G–11 Selected Indicators of Cumulative Impact

Category	Indicator
Resource use	Electricity use compared with site capacity Water use compared with site capacity Workers required compared with existing workforce
Air quality	Criteria pollutant concentrations and comparisons with standards or guidelines
Human health	Public Offsite population dose Maximally exposed individual dose Fatalities Comparison with DOE dose limits Workers Total dose Average worker dose Fatalities Comparison with DOE dose limits
Waste	Transuranic waste/high-level radioactive waste generation rate compared with existing management capacities and generation rate Low-level radioactive generation rate compared with existing management capacities and generation rate Mixed low-level radioactive generation rate compared with existing management capacities and generation rate Hazardous waste generation rate compared with existing management capacities and generation rate Nonhazardous generation rate compared with existing management capacities and generation rate
Spent nuclear fuel	Spent nuclear fuel generation rate and storage capacity
Transportation	Radiation exposures Public Transportation workers Fatalities Traffic fatalities

Public documents prepared by agencies of Federal, state, and local governments were the primary sources of information for non-DOE actions.

The analysis focused on the potential for cumulative impacts at each candidate site from DOE actions under detailed consideration at the time of this NI PEIS, as well as cumulative impacts associated with transportation between the sites, between Savannah River Site and other sites, and between the processing sites and Los Alamos National Laboratory (**Table G–12**). Non-DOE actions were also considered where information was readily available.

It is assumed that construction impacts would not be cumulative because construction is typically short in duration, and construction impacts are generally temporary. Further, except for a stack required for FMEF, construction is limited to internal modifications to existing DOE facilities. Decontamination and decommissioning of the candidate facilities was not addressed in the cumulative impact estimates. Given the uncertainty regarding the timing of decontamination and decommissioning, any impact estimate at this time would be highly speculative. A detailed evaluation of decontamination and decommissioning would be provided in follow-on NEPA documentation closer to the actual time of those actions.

Table G–12 Other Present and Reasonably Foreseeable Actions Considered in the Cumulative Impact Assessment

Activities	ORR	INEEL	Hanford
Disposition of Surplus Plutonium	X		
Storage and Disposition of Weapons-Usable Fissile Materials	X	X	X
Disposition of Surplus Highly Enriched Uranium	X		
Waste Management PEIS	X	X	X
Spent Nuclear Fuel Management and INEL Environmental Restoration and Waste Management	X	X	X
Foreign Research Reactor Spent Nuclear Fuel Management		X	X
Stockpile Stewardship and Management	X		
Tank Waste Remediation			X
Radioactive Releases from WNP Nuclear Power Plant			X
Hanford Reach of the Columbia River Comprehensive River Conservation Study			X
Hanford Comprehensive Land Use Plan			X
K Basins Spent Fuel Management			X
Advanced Mixed Waste Treatment Project		X	
Treatment and Management of Sodium-Bonded Spent Nuclear Fuel		X	
Construction and Operation of the Spallation Neutron Source	X		
Long-Term Management and Use of Depleted Uranium Hexafluoride	X		
Treatment and Shipment of Transuranic Waste	X		
Management of Liquid Low-Level Radioactive Waste	X		
Management of Spent Nuclear Fuel	X		
Transportation of Low-Level Radioactive Waste to Offsite Treatment or Disposal	X		
Transportation of Low-Level Radioactive Mixed Waste to Offsite Treatment or Disposal	X		
Natural and Accelerated Bioremediation Field Research Center Assessment	X		
Idaho High-Level Waste and Facilities Disposition		X	

The *Surplus Plutonium Disposition EIS* (DOE 1999a), the *Final Environmental Impact Statement, Construction and Operation of the Spallation Neutron Source* (DOE 1999b), the *Storage and Disposition PEIS* (DOE 1996), and the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering and Environmental Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995) provide a comprehensive evaluation of cumulative impacts for the DOE sites.

G.11 REFERENCES

Code of Federal Regulations

7 CFR Part 657, "Prime and Unique Farmland," U.S. Department of Agriculture.

10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," U.S. Nuclear Regulatory Commission.

33 CFR Section 328.3, "Definitions of Waters of the United States," Corps of Engineers, Department of the Army.

40 CFR Part 50, "National Primary and Secondary Ambient Air Quality Standards," U.S. Environmental Protection Agency.

40 CFR Part 51, Appendix W, "Guidelines on Air Quality Models," U.S. Environmental Protection Agency.

40 CFR Section 51.165, " Permit Requirements," U.S. Environmental Protection Agency.

40 CFR Part 61, "National Emission Standards for Hazardous Air Pollutants," U.S. Environmental Protection Agency.

40 CFR Section 81.313, "Designation of Areas for Air Quality Planning Purposes, Idaho," U.S. Environmental Protection Agency.

40 CFR Section 81.343, "Designation of Areas for Air Quality Planning Purposes, Tennessee," U.S. Environmental Protection Agency.

40 CFR Section 81.348, "Designation of Areas for Air Quality Planning Purposes, Washington," U.S. Environmental Protection Agency.

40 CFR Part 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," U.S. Environmental Protection Agency.

40 CFR Section 1508.7, "Cumulative Impact," Council on Environmental Quality.

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63 FR 41810, U.S. Department of Energy, 1998, "Record of Decision for the Department of Energy Waste Management Programs: Treatment of Nonwastewater Hazardous Waste," August 5.

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DOE (U.S. Department of Energy), 1999b, *Final Environmental Impact Statement, Construction and Operation of the Spallation Neutron Source,* DOE/EIS-0247, Office of Science, Germantown, MD, April.

DOE (U.S. Department of Energy), 1999c, *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement,* DOE/EIS-0287D, Idaho Operations Office, Idaho Falls, ID, December.

DOE (U.S. Department of Energy), 2000, *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel,* DOE/EIS-0306, Office of Nuclear Energy, Science and Technology, Washington, DC, July.

DOI (U.S. Department of Interior), 1986, *Visual Resource Contrast Rating,* BLM Manual Handbook H-8431-1, Bureau of Land Management, Washington, DC, January 17.

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EPA (U.S. Environmental Protection Agency), 2000, *Addendum - User's Guide for the Industrial Source Complex (ISC3) Dispersion Models, Vol. 1 - User Instructions,* EPS-454/B-95-003a, Office of Air Quality Planning and Standards, April.

Appendix H
Evaluation of Human Health Effects from Normal Facility Operations

H.1 INTRODUCTION

This appendix presents detailed information on the methodology employed for calculating potential impacts and risks to humans associated with releases of radioactivity and hazardous chemicals from the proposed facilities during normal operations and certain accident scenarios. This information is intended to support the public and occupational health and safety assessments described in Chapter 4 of this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]). Section H.2.1 provides general background information on ionizing radiation and associated health effects, Section H.2.2 discusses the methodology used in the assessment of normal radiological impacts, and Section H.2.3 provides a brief overview of data used in the radiological assessments. Hazardous chemical impacts are presented in Section H.3. Further detailed information regarding potential radiological impacts resulting from facility accidents is discussed in Appendix I of this NI PEIS.

This appendix presents numerical information using engineering and/or scientific notation. For example, the number 100,000 can also be expressed as 1×10^{5}. The fraction 0.00001 can also be expressed as 1×10^{-5}. The following chart defines the equivalent numerical notations that may be used in this appendix.

Fractions and Multiples of Units

Multiple	Decimal Equivalent	Prefix	Symbol
1×10^{6}	1,000,000	mega-	M
1×10^{3}	1,000	kilo-	k
1×10^{2}	100	hecto-	h
1×10	10	deka-	da
1×10^{-1}	0.1	deci-	d
1×10^{-2}	0.01	centi-	c
1×10^{-3}	0.001	milli-	m
1×10^{-6}	0.000001	micro-	μ
1×10^{-9}	0.000000001	nano-	n
1×10^{-12}	0.000000000001	pico-	p
1×10^{-15}	0.000000000000001	femto-	f
1×10^{-18}	0.000000000000000001	atto-	a

H.2 RADIOLOGICAL IMPACTS ON HUMAN HEALTH

H.2.1 Background Information

H.2.1.1 Nature of Ionizing Radiation and Its Effects on Humans

What Is Ionizing Radiation? Ionizing radiation (hereafter referred to as "radiation") is energy transferred in the form of particles or waves. Humans are exposed constantly to cosmic radiation and radiation from the earth's rocks and soil. (The term "radiation" encompasses several phenomena, including light, heat waves,

microwaves, radio waves, and ionizing radiation. The discussion of radiation in this section addresses ionizing radiation, and the term "radiation" is used to mean ionizing radiation.) This radiation contributes to the natural background radiation that has always surrounded us. Manmade sources of radiation also exist, including medical and dental x-rays, household smoke detectors, and materials released from nuclear and coal-fired power plants.

Radiation comes from the activity of atoms, which form the substance of all matter in the universe. Atoms are composed of even smaller particles (protons, neutrons, electrons), whose number and arrangement distinguish atoms of one element from another. Elements consist of atoms having the same number of protons. Atoms of the same element with varying numbers of neutrons are known as isotopes of that element. There are more than 100 natural and manmade elements. Some of these isotopes (including isotopes of elements, such as uranium, radium, plutonium, and thorium) share a very important quality: they are unstable (i.e., they decay). As they change into more stable forms, invisible waves of energy or particles, known as ionizing radiation, are released. Radioactivity is the emitting of this radiation.

Ionizing radiation refers to the fact that this energy emitted from unstable atoms can ionize, or electrically charge, atoms by stripping off electrons, leaving them with a positive charge. Ionizing radiation can cause a change in the chemical composition of many materials, including living tissue (organs), which can affect the way they function.

- Alpha particles are one type of ionizing radiation and the heaviest of the types discussed here; despite a speed of approximately 16,000 kilometers per second (9,940 miles per second), they can travel only several centimeters in air. Alpha particles lose their energy almost as soon as they collide with anything. They can be stopped easily by a sheet of paper or by the skin's surface.

Radiation Type	Typical Speed (km/sec)	Typical Travel Distance in Air (meters)	Barrier
Alpha	16,000	Less than 1	Sheet of paper or skin's surface
Beta	160,000	3	Thin sheet of aluminum foil or glass
Gamma	300,000	Very large[a]	Thick wall of concrete, lead, or steel
Neutron	39,000	Very large	Water, paraffin, graphite

a. Would be infinite in a vacuum.

- Beta particles are much lighter than alpha particles. They can travel at a speed of up to 160,000 kilometers per second (99,400 miles per second) and can travel in the air for a distance of approximately 3 meters (9.8 feet). Beta particles can pass through a sheet of paper but may be stopped by a thin sheet of aluminum foil or glass.

- Gamma rays and x-rays, unlike alpha or beta particles, are waves of pure energy. Gamma rays travel at the speed of light (300,000 kilometers per second [186,000 miles per second]). Gamma radiation is very penetrating and requires a thick wall of concrete, lead, or steel to stop it.

- The neutron is another particle that contributes to radiation exposure, both directly and indirectly. The latter is associated with the gamma rays and alpha particles that are emitted following neutron capture in matter. A neutron has about one quarter the weight of an alpha particle and can travel at speeds of up to 39,000 kilometers per second (24,200 miles per second). Neutrons are more penetrating than beta particles but typically less penetrating than gamma rays.

The effects on people of radiation emitted during the disintegration (decay) of a radioactive substance depend on the type of radiation (alpha and beta particles and gamma and x-rays) and the total amount of radiation energy absorbed by the body. The total energy absorbed per unit quantity of tissue is referred to as absorbed dose. The absorbed dose, when multiplied by certain quality factors and factors that take into account different sensitivities of various tissues, is referred to as effective dose equivalent or, where the context is clear, simply dose. The common unit of effective dose equivalent is the roentgen equivalent man (rem); 1 rem equals 1,000 millirem.

The radioactivity of a material decreases with time. The time it takes a material to lose half of its original radioactivity is designated its half-life. For example, a quantity of iodine-131, a material that has a half-life of eight days, will lose one-half of its radioactivity in that amount of time. In eight more days, one-half of the remaining radioactivity will be lost, and so on. Eventually, the radioactivity will essentially disappear. Each radioactive element has a characteristic half-life. The half-lives of various radioactive elements may vary from millionths of a second to millions of years.

When a radioactive element emits a particle or gamma-ray, it often changes to an entirely different element, one that may or may not be radioactive. Eventually, a stable element is formed. This transformation, which may take several steps, is known as a decay chain. Radium, for example, is a naturally occurring radioactive element with a half-life of 1,622 years. It emits an alpha particle and becomes radon, a radioactive gas with a half-life of only 3.8 days. Radon decays first to polonium, then through a series of steps to bismuth, and ultimately to lead.

Units of Radiation Measure. Scientists and engineers use a variety of units to measure radiation. These different units can be used to determine the amount, type, and intensity of radiation. Just as heat can be measured in terms of its intensity or effects using units of calories or degrees, amounts of radiation can be measured in curies, radiation absorbed dose (rad), or rem.

- **Curie.** The curie, named after the French scientists Marie and Pierre Curie, describes the "intensity" of a sample of radioactive material. The rate of decay of 1 gram of radium is the basis of this unit of measure. It is equal to 3.7×10^{10} disintegrations (decays) per second.

- **Rad.** The total energy absorbed per unit quantity of tissue is referred to as absorbed dose. The rad is the unit of measurement for the physical absorption of radiation. As sunlight heats pavement by giving up an amount of energy to it, radiation gives up rads of energy to objects in its path. One rad is equal to the amount of radiation that leads to the deposition of 0.01 joule of energy per kilogram of absorbing material.

Radiation Units and Conversions

1 Ci = 3.7×10^{10} sec^{-1} = 3.7×10^{10} becquerel
1 rad = 100 erg/g = 0.01 gray
1 erg = 10^{-7} joule
1 gray = 1 joule/kg = 100 rad
1 rem = 0.01 sievert

- **Rem.** A rem is a measurement of the dose from radiation based on its biological effects. The rem is used in measuring the effects of radiation on the body. Thus, 1 rem of one type of radiation is presumed to have the same biological effects as 1 rem of any other kind of radiation. This allows comparison of the biological effects of radionuclides that emit different types of radiation.

An individual may be exposed to ionizing radiation externally (from a radioactive source outside the body) or internally (from ingesting or inhaling radioactive material). The external dose is different from the internal dose because an external dose is delivered only during the actual time of exposure to the external radiation

source, but an internal dose continues to be delivered as long as the radioactive source is in the body. For the analyses conducted in this NI PEIS, the dose from internal exposure is calculated over 50 years following the initial exposure; both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time.

The three types of doses calculated in this NI PEIS are external dose, internal dose, and combined external and internal dose. Each type of dose is discussed separately in the following paragraphs.

- **External dose.** The external dose can result from several different pathways, all having in common the fact that the radiation causing the exposure is external to the body. In this NI PEIS, these pathways include exposure to a cloud of radiation passing over the receptor or standing on ground that is contaminated with radioactivity. The appropriate measure of dose is called the effective dose equivalent. If the receptor departs from the source of radiation exposure, the dose rate will be reduced. It is assumed that external exposure occurs uniformly during the year.

- **Internal dose.** The internal dose results from a radiation source entering the human body via any means, such as through ingestion of contaminated food or water or inhalation of contaminated air. In this NI PEIS, pathways for internal exposure include: (1) ingestion of crops contaminated by airborne radiation deposits, (2) ingestion of animal products from animals that ingested contaminated food, and (3) inhalation of contaminated air. In contrast to external exposure, once radioactive material enters the body, it remains there for a period of time that depends on the rate of radiological decay and biological elimination rates. The unit of measure for internal doses is the committed dose equivalent. It is the internal dose that each body organ receives from the ingestion and inhalation of radioactive material. In this analysis of health impacts from normal operations, the committed dose equivalent is calculated for an annual intake period. Normally, a 50-year dose-commitment period is used (i.e., the 1-year intake period plus 49 years). The dose rate increases during the 1-year intake. The dose rate after the first year intake declines slowly as the radioactivity in the body continues to produce a dose. The integral of the dose rate over the 50 years gives the committed dose equivalent.

The various organs of the body have different susceptibilities to harm from radiation. The quantity that takes these different susceptibilities into account to provide a broad indicator of the risk to the health of an individual from radiation is called the committed effective dose equivalent. It is obtained by multiplying the committed dose equivalent in each major organ or tissue by a weighting factor associated with the risk susceptibility of the tissue or organ, then summing the totals. It is possible for the committed dose equivalent to an organ to be larger than the committed effective dose equivalent if that organ has a small weighting factor. The concept of committed effective dose equivalent applies only to internal pathways.

- **Combined external and internal dose.** The sum of the committed effective dose equivalent from internal pathways and the effective dose equivalent from external pathways is called the "total effective dose equivalent." The U.S. Department of Energy (DOE), in DOE Order 5400.5, calls this quantity the "effective dose equivalent."

The units used in this NI PEIS for committed dose equivalent, effective dose equivalent, and committed effective dose equivalent to an individual are the rem and millirem (1/1000 of 1 rem). The corresponding unit for the collective dose to a population (the sum of the doses to members of the population, or the product of the number of exposed individuals and their average dose) is the person-rem.

Sources of Background Radiation. The average American receives a total of approximately 360 millirem per year from all sources of radiation, both natural and manmade. The sources of radiation can be divided into six different categories: (1) cosmic radiation, (2) external terrestrial radiation, (3) internal radiation,

(4) consumer products, (5) medical diagnosis and therapy, and (6) other sources (NCRP 1987). These categories are discussed in the following paragraphs:

- **Cosmic radiation.** Cosmic radiation is ionizing radiation resulting from energetic charged particles from space continuously hitting the earth's atmosphere. These particles, and the secondary particles and photons they create, are cosmic radiation. Because the atmosphere provides some shielding against cosmic radiation, the intensity of this radiation increases with altitude above sea level. The average dose to the people in the United States from this source is approximately 27 millirem per year.

- **External terrestrial radiation.** External terrestrial radiation is the radiation emitted from the radioactive materials in the earth's rocks and soils. The average dose from external terrestrial radiation is approximately 28 millirem per year.

- **Internal radiation.** Internal radiation results from the human body metabolizing natural radioactive material that has entered the body by inhalation or ingestion. Natural radionuclides in the body include isotopes of uranium, thorium, radium, radon, polonium, bismuth, potassium, rubidium, and carbon. The major contributor to the annual dose equivalent for internal radioactivity are the short-lived decay products of radon, which contribute approximately 200 millirem per year. The average dose from other internal radionuclides is approximately 39 millirem per year.

- **Consumer products.** Consumer products also contain sources of ionizing radiation. In some products such as smoke detectors and airport x-ray machines, the radiation source is essential to product operation. In other products, such as televisions and tobacco, the radiation occurs incidentally to the product function. The average dose from consumer products is approximately 10 millirem per year.

- **Medical diagnosis and therapy.** Radiation is an important diagnostic medical tool and cancer treatment. Diagnostic x-rays result in an average exposure of 39 millirem per year. Nuclear medical procedures result in an average exposure of 14 millirem per year.

- **Other sources.** There are a few additional sources of radiation that contribute minor doses to individuals in the United States. The dose from nuclear fuel-cycle facilities (e.g., uranium mines, mills, and fuel processing plants), nuclear power plants, and transportation routes has been estimated to be less than 1 millirem per year. Radioactive fallout from atmospheric atomic bomb tests, emissions of radioactive material from DOE facilities and facilities licensed by the U.S. Nuclear Regulatory Commission (NRC), emissions from certain mineral extraction facilities, and transportation of radioactive materials contribute less than 1 millirem per year to the average dose to an individual. Air travel contributes approximately 1 millirem per year to the average dose.

The collective (or population) dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. This total dose received by the exposed population is measured in person-rem. For example, if 1,000 people each receive a dose of 1 millirem (0.001 rem), the collective dose is 1,000 persons × 0.001 rem = 1.0 person-rem. Alternatively, the same collective dose (1.0 person-rem) results if 500 people each receive a dose of 2 millirem (500 persons × 2 millirem = 1 person-rem).

Limits of Radiation Exposure. The amount of manmade radiation that the public may be exposed to is limited by Federal regulations. Although most scientists believe that radiation absorbed in small doses over several years is not harmful, U.S. Government regulations assume that the effects of all radiation exposures are cumulative.

Under the Clean Air Act, releases of materials to the atmosphere from DOE facilities are limited by the U.S. Environmental Protection Agency (EPA) to quantities that would produce a dose of less than 10 millirem per year to a member of the general public (40 CFR Part 61). DOE also limits to 10 millirem the dose annually received from material released to the atmosphere (DOE Order 5400.5). EPA and DOE also limit the annual dose to a member of the general public from radioactive releases in drinking water to 4 millirem, as required under the Safe Drinking Water Act (40 CFR Part 141, DOE Order 5400.5). The annual dose from all radiation sources from a nuclear-fuel-cycle facility site is limited by EPA to 25 millirem (40 CFR Part 190). The DOE annual limit of radiation dose from all pathways to a member of the general public is 100 millirem (DOE Order 5400.5).

Each of the three sites covered by this NI PEIS operates below all of these limits. The average individual in the United States receives a dose of approximately 0.3 rem (300 millirem) per year from natural sources of radiation. For perspective, a modern chest x-ray results in an approximate dose of 0.006 rem (6 millirem) and a diagnostic pelvis and hip x-ray results in an approximate dose of 0.065 rem (65 millirem) (NCRP 1987). An acute dose (i.e., a dose over a short period of time) of about 450 rem (450,000 millirem) would result in a 50 percent chance of death.

For people working in an occupation that involves radiation, NRC and DOE limit doses to 5 rem per year (5,000 millirem per year) (10 CFR Part 20, 10 CFR Part 835). The Administrative Control Level of 2 rem (2,000 millirem) per year is typically imposed at DOE sites to comply with "as low as is reasonably achievable" initiatives (10 CFR Part 835).

H.2.1.2 Health Effects

Radiation exposure and its consequences are topics of interest to the general public. For this reason, this NI PEIS places much emphasis on the consequences of exposure to radiation, even though the effects of radiation exposure under most circumstances evaluated in this NI PEIS are small. To provide the background for discussions of impacts, this section explains the basic concepts used in the evaluation of radiation effects.

Radiation can cause a variety of adverse health effects in people. The most significant adverse health effect that depicts the consequences of environmental and occupational radiation exposure is induction of cancer fatalities. This effect is referred to as "latent" cancer fatalities because the cancer may take many years to develop. In the discussions that follow, all fatal cancers are considered latent, and therefore the term "latent" is not used.

Health impacts from radiation exposure, whether from sources external or internal to the body, generally are identified as "somatic" (affecting the individual exposed) or "genetic" (affecting descendants of the exposed individual). Radiation is more likely to produce somatic effects than to produce genetic effects. For this NI PEIS, therefore, only the somatic risks are presented. The somatic risks of most importance are the induction of cancers. With the exception of leukemia, which can have an induction period (time between exposure to carcinogen and cancer diagnosis) of as little as 2 to 7 years, most cancers have an induction period of more than 20 years.

For a uniform irradiation of the body, the incidence of cancer varies among organs and tissues; the thyroid and skin demonstrate a greater sensitivity than other organs. Such cancers, however, also produce relatively low mortality rates because they are relatively amenable to medical treatment. Because of the readily available data for cancer mortality rates and the relative scarcity of prospective epidemiologic studies, somatic effects leading to cancer fatalities rather than cancer incidence are presented in this NI PEIS. The numbers of cancer fatalities can be used to compare the risks among the various alternatives.

The National Research Council's Committee on the Biological Effects of Ionizing Radiation (BEIR) has prepared a series of reports to advise the U.S. Government on the health consequences of radiation exposures. The latest of these reports, *Health Effects of Exposure to Low Levels of Ionizing Radiation BEIR V* (NAS 1990), provides the most current estimates for excess mortality from leukemia, and cancers other than leukemia, expected to result from exposure to ionizing radiation. This report updates the models and risk estimates provided in an earlier report of the Committee, *The Effects on Populations of Exposure to Low Levels of Ionizing Radiation*. The BEIR V models were developed for application to the U.S. population.

BEIR V provides estimates that are consistently higher than those in its predecessor BEIR III. This increase is attributed to several factors, including the use of a linear dose response model for cancers other than leukemia, revised dosimetry for the Japanese atomic bomb survivors, and additional follow-up studies of the atomic bomb survivors and other cohorts. BEIR III employs constant relative and absolute risk models, with separate coefficients for each of several sex and age-at-exposure groups; BEIR V develops models in which the excess relative risk is expressed as a function of age at exposure, time after exposure, and sex for each of several cancer categories. The BEIR III models were based on the assumption that absolute risks are comparable between the atomic bomb survivors and the U.S. population; BEIR V models were based on the assumption that the relative risks are comparable. For a disease such as lung cancer, where baseline risks in the United States are much larger than those in Japan, the BEIR V approach leads to larger risk estimates than the BEIR III approach.

The models and risk coefficients in BEIR V were derived through analyses of relevant epidemiologic data that included the Japanese atomic bomb survivors, ankylosis spondylitis patients, Canadian and Massachusetts fluoroscopy patients (breast cancer), New York postpartum mastitis patients (breast cancer), Israel Tinea Capitis patients (thyroid cancer), and Rochester thymus patients (thyroid cancer). Models for leukemia, respiratory cancer, digestive cancer, and other cancers used only the atomic bomb survivor data, although results of analyses of the ankylosis spondylitis patients were considered. Atomic bomb survivor analyses were based on revised dosimetry with an assumed relative biological effectiveness[1] of 20 for neutrons and were restricted to doses less than 400 rads. Estimates of risks of fatal cancers other than leukemia were obtained by totaling the estimates for breast cancer, respiratory cancer, digestive cancer, and other cancers.

Risk Estimates for Doses Equal To or Greater Than 20 Rem (Accident Scenarios). BEIR V includes risk estimates for a single exposure to a high level of radiation to all people in a large population group. The estimates are given in terms of lifetime risks per 1.0×10^6 person-rem. Fatality estimates for leukemia, breast cancer, respiratory cancer, digestive cancer, and other cancers are given for both sexes and nine age-at-exposure groups. These estimates, based on the linear model, are summarized in **Table H–1**. The average risk estimate from all ages and both sexes is 885 excess latent cancer fatalities per million person-rem. This value has been conservatively rounded up to 1,000 excess latent cancer fatalities per million person-rem.

Although values for other health effects are not presented in this NI PEIS, the risk estimators for nonfatal cancers and for genetic disorders to future generations are estimated to be approximately 200 and 260 per million person-rem, respectively. These values are based on information presented in the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991) and are seen to be 20 percent and 26 percent, respectively, of the fatal cancer estimator. Thus, if the number of excess latent fatal cancers is projected to be "X," the number of excess genetic disorders would be 0.26 times "X."

[1] A relative biological effectiveness factor is essentially used to represent a given radiation type's (neutron, gamma, alpha, etc.) ability to transfer energy to a given biological receptor.

Table H–1 Lifetime Risks per Million Person-Rem for Individual Exposures Greater Than 20 Rem

Gender	Type of Fatal Cancer		
	Leukemia[a]	Cancers Other Than Leukemia	Total Cancers
Male	220	660	880
Female	160	730	890
Average	190	695	885[b]

a. These are the linear estimates, which are double the linear-quadratic estimates provided in BEIR V for leukemia at low doses and dose rates.
b. This value has been rounded up to 1,000 excess cancer fatalities per million person-rem.
Source: NAS 1990.

Risk Estimates for Doses Less Than 20 Rem (Normal Operational Scenarios). For doses lower than 20 rem, a linear-quadratic model provides a significantly better fit to the data for leukemia than a linear model, and leukemia risks were based on a linear-quadratic function, which reduces the effects by a factor of two over estimates that are obtained from a linear model. For other cancers, linear models were found to provide an adequate fit to the data and were used for extrapolation to low doses. The BEIR V Committee, however, recommended reducing these linear estimates by a factor between 2 and 10 for doses received at low dose rates. For this NI PEIS, a risk reduction factor of two was adopted for conservatism.

Based on the preceding discussion, the resulting risk estimator would be equal to half the value observed for high-dose situations or approximately 500 excess latent cancer fatalities per million person-rem (0.0005 excess cancer fatality per person-rem). This is the risk value used in this NI PEIS to calculate cancer fatalities to the general public during normal operations and also for accidents in which individual doses are less than 20 rem. For workers, a value of 400 excess latent cancer fatalities per million person-rem (0.0004 excess latent cancer fatality per person-rem) is used in this NI PEIS. This lower value reflects the absence of children (who are more radiosensitive than adults) in the workforce. Again, based on information provided in the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991), the health risk estimators for nonfatal cancer and genetic disorders among the public are 20 percent and 26 percent, respectively, of the fatal cancer risk estimator. For workers, the health risk estimators are both 20 percent of the fatal cancer risk estimator. For this NI PEIS, only fatal cancers are presented.

The risk estimates may be applied to calculate the effects of exposing a population to radiation. For example, in a population of 100,000 people exposed only to natural background radiation (0.3 rem per year), 15 latent cancer fatalities per year would result from this radiation (100,000 persons × 0.3 rem per year × 0.0005 latent cancer fatality per person-rem = 15 latent cancer fatalities per year).

Calculations of the number of excess cancer fatalities associated with radiation exposure do not always yield whole numbers; calculations may yield numbers less than 1.0, especially in environmental applications. For example, if a population of 100,000 were exposed as described in the previous paragraph, but to a total dose of only 0.001 rem, the collective dose would be 100 person-rem, and the corresponding estimated number of latent cancer fatalities would be 0.05 (100,000 persons × 0.001 rem × 0.0005 latent cancer fatality per person-rem = 0.05 latent cancer fatality).

For latent cancer fatalities less than 1.0, the estimated latent cancer fatality is a statistical estimate. The latent cancer fatality of 0.05 associated with a 100 person-rem dose is the average number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, no person (zero people) would incur a latent cancer fatality from the 0.001 rem dose each member would have received. In a small fraction of the groups, one latent cancer fatality would result; in exceptionally few groups, two or more latent cancer fatalities would occur. The average number of deaths over all the groups would be

0.05 latent cancer fatality (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is 0 latent cancer fatality.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to background radiation over a lifetime. The "number of latent cancer fatalities" corresponding to a single individual's exposure over a (presumed) 72-year lifetime to 0.3 rem per year is the following:

1 person × 0.3 rem per year × 72 years × 0.0005 latent cancer fatality/person-rem = 0.011 latent cancer fatality.

Again, this is a statistical estimate; that is, the estimated effect of background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual might incur a latent cancer fatality caused by the exposure over his full lifetime. Presented another way, this method estimates that approximately 1.1 percent of the population might die of cancers induced by background radiation.

H.2.2 Methodology for Estimating Radiological Impacts

The potential radiological impacts associated with normal operating conditions and accidents at the processing facilities were calculated using Version 1.485 of the GENII computer code. Site-specific and technology-specific input data were used, including location, meteorology, population, food production and consumption, and source terms. Section H.2.2.1 briefly describes GENII and outlines the approach used for modeling normal operations and facility accidents.

H.2.2.1 GENII Computer Code

The GENII computer model, developed by DOE at the Pacific Northwest National Laboratory, is an integrated system of various computer modules that analyze environmental contamination resulting from acute or chronic releases to, or initial contamination in, air, water, or soil. The model calculates radiation doses to individuals and populations. The GENII computer model is well documented for assumptions, technical approach, methodology, and quality assurance issues (Napier et al. 1988). The GENII computer model has gone through extensive quality assurance and quality control steps, including comparing results from model computations with those from hand calculations and performing internal and external peer reviews. Recommendations given in these reports were incorporated into the final GENII computer model, as deemed appropriate.

For this NI PEIS, only the ENVIN, ENV, and DOSE computer modules were used. The codes are connected through data transfer files. The output of one code is stored in a file that can be used by the next code in the system.

- **ENVIN.** The ENVIN module of the GENII code controls the reading of input files and organizes the input for optimal use in the environmental transport and exposure module, ENV. The ENVIN code interprets the basic input, reads the basic GENII data libraries and other optional input files, and organizes the input into sequential segments based on radionuclide decay chains.

  A standardized file that contains scenario, control, and inventory parameters is used as input to ENVIN. Radionuclide inventories can be entered as functions of releases to air or water, concentrations in basic environmental media (air, soil, or water), or concentrations in foods. If certain atmospheric dispersion options have been selected, this module can generate tables of atmospheric dispersion parameters that will be used in later calculations. If the finite plume air submersion option is requested in addition to the atmospheric dispersion calculations, preliminary energy-dependent finite plume dose factors are prepared. The ENVIN module prepares the data transfer files that are used as

input by the ENV module; ENVIN generates the first portion of the calculation documentation—the run input parameters report.

- **ENV.** The ENV module calculates the environmental transfer, uptake, and human exposure to radionuclides that result from the chosen scenario for the user-specified source term. The code reads the input files from ENVIN and then, for each radionuclide chain, sequentially performs the precalculations to establish the conditions at the start of the exposure scenario. Environmental concentrations of radionuclides are established at the beginning of the scenario by assuming decay of preexisting sources, considering biotic transport of existing subsurface contamination, and defining soil contamination from continuing atmospheric or irrigation depositions. For each year of postulated exposure, the code then estimates the air, surface soil, deep soil, groundwater, and surface water concentrations of each radionuclide in the chain. Human exposures and intakes of each radionuclide are calculated for (1) pathways of external exposure from finite atmospheric plumes; (2) inhalation; (3) external exposure from contaminated soil, sediments, and water; (4) external exposure from special geometries; and (5) internal exposures from consumption of terrestrial foods, aquatic foods, drinking water, animal products, and inadvertent intake of soil. The intermediate information on annual media concentrations and intake rates are written to data transfer files. Although these may be accessed directly, they are usually used as input to the DOSE module of GENII.

- **DOSE.** The DOSE module reads the intake and exposure rates defined by the ENV module and converts the data to radiation dose.

H.2.2.2 Data and General Assumptions

To perform the dose assessments for this NI PEIS using the GENII code, different types of data were collected and/or generated. In addition, calculational assumptions were made. This section discusses both the data collected and/or generated for use in performing the dose assessments and the assumptions made for this NI PEIS.

- **Meteorological data.** The meteorological data used for all normal operational and accident assessments were in the form of Oak Ridge Reservation (ORR), Idaho National Environmental and Engineering Laboratory (INEEL), and Hanford Site (Hanford) joint frequency data files. A joint frequency data file is a table listing the fractions of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The joint frequency data files were based on measurements taken over a period of several years at different locations and heights. Average annual meteorological conditions (averaged over the measurement period) were used for normal operation and the 50th percentile atmospheric conditions were used for accident scenarios. (Accident analysis results and additional analysis detail are presented in Appendix I.)

- **Population data.** Population distributions were based on the *1990 Census of Population and Housing* data (DOC 1992). Projections were determined for the year 2020 (approximate midlife of operations) for areas within 80 kilometers (50 miles) of ORR, INEEL, and Hanford release locations. The site populations in 2020, assumed to be representative of the populations over the operational period evaluated, were used in the impact assessments. The populations were spatially distributed on circular grids with 16 directions and 10 radial distances up to 80 kilometers (50 miles). The grids were centered at the precise locations from which the radionuclides were assumed to be released.

- **Source term data.** Source term(s) (i.e., quantities of radioactive material released to the environment over a given period) were estimated based on characteristic releases associated with historical data.

The source term used to estimate the incremental impacts of normal operations addresses releases of radioactive material during normal operations and from anticipated events (e.g., powder spills).

- **Food production and consumption data.** Agricultural data from *Health Risk Data for Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (HNUS 1996) were used as a source for food production quantities. Food production was spatially distributed on the same circular grid used for the population distributions. The consumption rates used in GENII were those for the maximum individual and average individual. People living within the 80-kilometer (50-mile) assessment area were assumed to consume only food grown in that area.

- **Calculational assumptions.** For normal operations, impact assessments were performed for both members of the general public and workers associated with processing facility activities. These assessments were made to determine the incremental impacts that would be associated with the action alternatives addressed in this NI PEIS. Incremental doses for members of the public were calculated (via GENII) for two different types of receptors: the maximally exposed offsite individual and the general population living within 80 kilometers (50 miles) of a given facility. The maximally exposed individual associated with the alternatives addressed in this NI PEIS was assumed to be located at a position on the site boundary that would yield the highest impacts during normal operations of a given alternative. For facility workers, incremental doses were cited directly from facility-specific data reports. For doses associated with storage actions (i.e., "No Action" neptunium-237 storage), it was conservatively assumed that 10 percent of the total fabrication and processing doses are attributable to storage impacts exclusively.

To estimate radiological impacts from normal operations, the following additional assumptions and factors were considered in using GENII.

- Ground surfaces were assumed to have no previous deposition of radionuclides.

- The annual external exposure time to the plume and to soil contamination was 0.7 year (16.8 hours per day) for the maximally exposed offsite individual (NRC 1977).

- The annual external exposure time to the plume and to soil contamination was 0.5 year (12 hours per day) for the population (NRC 1977).

- The annual inhalation exposure time to the plume was 1.0 year for the maximally exposed individual and general population (NRC 1977).

- The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

- A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops and animal products contaminated by either deposition of radioactivity from the air or irrigation. No liquid pathways were analyzed because expected releases will only be to the air.

- Reported release heights were used for atmospheric releases and were assumed to be the effective stack heights.

- The calculated doses were 50-year committed doses from 1 year of intake.

The exposure, uptake, and usage parameters used in the GENII model for normal operations are provided in **Tables H–2** through **H–4**.

Table H–2 GENII Exposure Parameters to Plumes and Soil Contamination (Normal Operations)

Maximum Individual				General Population			
External Exposure		Inhalation of Plume		External Exposure		Inhalation of Plume	
Plume (hours)	Soil Contamination (hours)	Exposure Time (hours)	Breathing Rate (cm^3/sec)	Plume (hours)	Soil Contamination (hours)	Exposure Time (hours)	Breathing Rate (cm^3/sec)
6,136	6,136	8,766	270	4,383	4,383	8,766	270

Key: cm^3/sec, cubic centimeters per second.
Source: Napier et al. 1988; NRC 1977.

Table H–3 GENII Usage Parameters for Consumption of Terrestrial Food (Normal Operations)

	Maximum Individual				General Population			
Food Type	Growing Time (days)	Yield (kg/m^2)	Holdup Time (days)	Consumption Rate (kg/yr)	Growing Time (days)	Yield (kg/m^2)	Holdup Time (days)	Consumption Rate (kg/yr)
Leafy vegetables	90.0	1.5	1.0	30.0	90.0	1.5	14.0	15.0
Root vegetables	90.0	4.0	5.0	220.0	90.0	4.0	14.0	140.0
Fruit	90.0	2.0	5.0	330.0	90.0	2.0	14.0	64.0
Grains/cereals	90.0	0.8	180.0	80.0	90.0	0.8	180.0	72.0

Key: kg/m^2, kilograms per square meter; kg/yr, kilograms per year.
Source: Napier et al. 1988.

Table H–4 GENII Usage Parameters for Consumption of Animal Products (Normal Operations)

			Stored Feed				Fresh Forage			
Food Type	Consumption Rate (kg/yr)	Holdup Time (days)	Diet Fraction	Growing Time (days)	Yield (kg/m^2)	Storage Time (days)	Diet Fraction	Growing Time (days)	Yield (kg/m^2)	Storage Time (days)
Maximum individual										
Beef	80.0	15.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	18.0	1.0	1.00	90.0	0.80	180.0	–	–	–	–
Milk	270.0	1.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	30.0	1.0	1.00	90.0	0.80	180.0	–	–	–	–
General population										
Beef	70.0	34.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	8.5	34.0	1.0	90.0	0.80	180.0	–	–	–	–
Milk	230.0	3.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	20.0	18.0	1.0	90.0	0.80	180.0	–	–	–	–

Key: kg/m^2, kilograms per square meter; kg/yr, kilograms per year.
Source: Napier et al. 1988.

H.2.2.3 Health Effects Calculations

In this NI PEIS, the collective combined effective dose equivalent is the sum of the collective committed effective dose equivalent (internal dose) and the collective effective dose equivalent (external dose). Doses calculated by GENII were used to estimate health effects using the risk estimators presented in Section H.2.1.2.

The incremental cancer fatalities in the general population and in groups of workers were, therefore, estimated by multiplying the collective combined effective dose equivalent by 0.0005 and 0.0004 cancer fatality per person-rem, respectively. Although health risk factors are statistical factors and not strictly applicable to individuals, they have been used in the past to estimate the incremental risk to an individual from exposure to radiation. Therefore, the factor of 0.0005 and 0.0004 per rem of individual committed effective dose equivalent for a member of the public and for a worker, respectively, have also been used in this NI PEIS to calculate the individual's incremental fatal cancer risk from exposure to radiation. As stated previously, for doses greater than 20 rem to an individual, these factors are doubled.

Under the realm of normal operations, for the public, the health effects expressed in this NI PEIS are the risk of fatal cancer to the maximally exposed individual and the number of fatal cancers to the 80-kilometer (50-mile) population from exposure to radioactivity released from any of the candidate sites over the full period of operations. For workers, the health effects expressed are the risk of fatal cancer to the average worker at a facility and the number of fatal cancers to all workers at that facility from the full period of operations.

H.2.2.4 Uncertainties

The sequence of analyses performed to generate the radiological impact estimates from normal operation include: (1) selection of normal operational modes, (2) estimation of source terms, (3) estimation of environmental transport and uptake of radionuclides, (4) calculation of radiation doses to exposed individuals, and (5) estimation of health effects. There are uncertainties associated with each of these steps. Uncertainties exist in the way the physical systems being analyzed are represented by the computational models and in the data required to exercise the models (due to measurement, sampling, or natural variability).

In principle, one can estimate the uncertainty associated with each source and predict the remaining uncertainty in the results of each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final results. However, conducting such a full-scale quantitative uncertainty analysis is neither practical nor a standard practice for a study of this type. Instead, the analysis is designed to ensure through judicious selection of release scenarios, models, and parameters, that the results represent the potential risks. This is accomplished by making conservative assumptions in the calculations at each step. The models, parameters, and release scenarios used in the calculations are selected in such a way that most intermediate results and, consequently, the final estimates of impacts are greater than what would be expected. As a result, even though the range of uncertainty in a quantity might be large, the value calculated for the quantity is close to one of the extremes in the range of possible values, so that the chance of the actual quantity being greater than the calculated value is low (or the chance of the quantity being less than the calculated value if the criteria are such that the quantity has to be maximized). This has been the goal of the radiological assessment for normal operation in this study (i.e., to produce results that are conservative).

The degree of conservatism in the calculated results is closely related to the range of possible values the quantity can have. This range is determined by what can be expected to realistically occur. Thus, the only processes considered are those credible for the conditions under which the physical system being modeled operates. This consideration has been employed for normal operation analyses.

Although the radionuclide composition of source terms are reasonable estimates, there are uncertainties in the radionuclide inventory and release reactions that affect estimated impacts.

H.2.3 Radiological Impact Assessment Data and Releases to the Environment

This section discusses the various site-dependent GENII input data required for quantifying the potential radiological impacts associated with the action alternatives in this NI PEIS. Agricultural data, population data, meteorological data, and release quantity data are discussed for the candidate sites.

- **Agricultural data.** Agricultural food production data (wheels) were cited from *Health Risk Data for Storage and Disposition of Weapons-Usable Fissile Materials Final PEIS* (HNUS 1996). The wheels were generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII (leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs). Each county's food production (in kilograms) was assumed to be distributed uniformly over a given county's land area. These categorized food wheels were fed into GENII as an input file and were used in the assessment of doses to a given general population from the ingestion pathway.

- **Population data.** Population data (wheels) were generated based on the 1990 *U.S. Census of Population and Housing* (DOC 1992). For each block in the 1990 census, the population was assigned a distance and direction from the release point; then the block's population was projected based on estimates of county growth in the year 2020. The population in each segment (e.g., south, southwest, north-northeast) was cumulated over all the blocks in the census. These population wheels were fed into GENII as an input file and were used in the assessment of a total dose incurred to a given general population.

- **Meteorological data.** Meteorological data (i.e., joint frequency distributions) were based on measurements of the fractions (given as percentages) of time the wind blows in a certain direction, at a certain speed, and within a certain stability class for ORR, INEEL, and Hanford, as cited in *Health Risk Data for Storage and Disposition of Weapons-Usable Fissile Materials Final PEIS* (HNUS 1996). The joint frequency distribution data is derived from 1 year of data (from X–10 Plant Tower 4 at ORR [1990 data] and the Grid 3 meteorological tower [1986 data] at INEEL), 9 years of data (1983–1991 from Hanford's 400 Area tower), or 13 years of data (1983–1996 from Hanford's 300 Area Tower 11). Data for facilities to be located at a generic site (the new accelerator(s), research reactor, and support facility) are derived from the hourly meteorological data developed for the health impacts from facility accidents presented in Appendix I. These data were fed into GENII as an input file and were used in the evaluation of χ/Q or E/Q values (these values represent radioisotope concentrations divided by the rates at which they are emitted to the environment); these were then used to determine the total dose incurred to a given general population, or an offsite maximally exposed individual.

- **Radiological releases to the environment.** Normal operational radiological releases to the environment (1.7×10^{-7} curies per year plutonium-238) were determined based on the conservative assumption that a 5 kilograms (11 pounds) inventory of plutonium-238 is processed on an annual basis at ORR, INEEL, or Hanford. Employing a processing facility emission factor of 1.98×10^{-12} (Wham 1999), and a specific activity of 17 curies per gram, a resulting annual release quantity of 1.7×10^{-7} curies is calculated as shown below:

  (5,000 grams per year of plutonium-238) × (17 curies of plutonium-238 per gram of plutonium-238) × (1.98×10^{-12}) = 1.7×10^{-7} curies per year of plutonium-238

Normal operational releases associated with the fabrication and processing of medical target material are based on an estimate of the releases that might occur during the normal handling and processing of target materials,

including anticipated off-normal conditions such as powder spills. Ventilation systems in all facilities used for processing of the target material would consist of at least two sets of high-efficiency particulate air filters, providing an emission removal efficiency of 2.4×10^{-6} and a total facility emission factor of 1.5×10^{-9}. (The emission factor for the elements radon and krypton, both gases, is 1.0.) (BWHC 1999). This results in the normal operational releases that are shown in **Table H–5**. These are the normal operational releases used for the facilities that process only the medical targets; the source term for facilities that process both medical targets and the plutonium-238 would be a combination of the plutonium operational release defined above and the medical releases shown in Table H–5.

Table H–5 Annual Normal Operational Releases Associated with Medical Target Processing

Isotope	Quantity Released (curies per year)	Isotope	Quantity Released (curies per year)
Copper-64	4.9×10^{-5}	Xenon-131m	2.0×10^{1}
Zinc-65	5.2×10^{-6}	Europium-152	2.1×10^{-5}
Strontium-85	9.7×10^{-6}	Europium-152m	2.9×10^{-4}
Krypton-85	2.9×10^{-3}	Gadolinium-153	5.0×10^{-6}
Krypton-85m	4.5×10^{-4}	Samarium-153	1.4×10^{-4}
Molybdenum-99	6.3×10^{-5}	Europium-154	7.0×10^{-5}
Palladium-103	2.0×10^{-5}	Europium-155	1.6×10^{-5}
Rhodium-103m	2.0×10^{-5}	Europium-156	1.5×10^{-3}
Technetium-99m	6.9×10^{-5}	Holmium-166	2.2×10^{-6}
Cadmium-109	2.9×10^{-6}	Tungston-187	3.3×10^{-3}
Iodine-125	1.1×10^{-5}	Iridium-192	1.6×10^{-5}
Iodine-131	4.6×10^{-6}	Radon-222	4.3×10^{1}

Source: BWHC 1999.

The Radiochemical Processing Laboratory (RPL) at Hanford would require modification prior to its use in medical isotope target fabrication and processing. Preoperational activities were assumed to result in the same emissions as those associated with operation of the facility in 1998; modification activities are not expected to result in significant quantities of airborne particulate of gaseous materials in excess of those generated during facility operation in prior years (BWHC 1999). These normal operational emissions are provided in **Table H–6**.

Table H–6 Annual Normal Operational Releases from RPL During Preoperational Activities

Isotope	Quantity Released (curies per year)
Tritium	1.6×10^{2}
Strontium-90[a]	1.5×10^{-7}
Plutonium-239[a]	4.4×10^{-8}

a. Strontium and plutonium releases have been increased to include all alpha and beta emissions detected during facility operation but not attributed to any single isotope.

Source: BWHC 1999.

The normal operational releases associated with operation (for target irradiation), preoperational startup, and standby operation of the Fast Flux Test Facility (FFTF) are based on measured releases for the facility in 1990 when the facility was operating at 300 megawatts and in 1998 when the facility was maintained in standby. These measured releases are provided in **Table H–7**. Normal operational releases have been scaled from those associated with 300-megawatt operations in 1990 to 400-megawatt operations. Operation at 400 megawatts was assumed for the analysis of normal operational impacts. Although operation at a lower power level should meet production goals for most of the mission time, operation at this higher level may be required and impacts

Table H–7 Annual Normal Operational Releases from FFTF

FFTF Normal Operations			
Isotope	**Quantity Released (curies per year)**		
	Combined Exhaust Release	**Heat Transport System Release**	**Service Building Release**
H-3 (tritium)	4.0	–	–
Argon-41	40	–	–
Cesium-137	6.4×10^{-7}	7.6×10^{-6}	7.6×10^{-6}
FFTF Standby			
Isotope	**Quantity Released (curies per year)**		
H-3 (tritium)	4.2		
Plutonium-239[a]	4.9×10^{-7}		
Strontium-90[a]	3.9×10^{-6}		

a. Plutonium and strontium are used to represent the total measured alpha and beta release from the FFTF during standby operation.
Source: BWHC 1999; DOE 1995.

from operations at the 400-megawatt power level will bound the normal operational impacts. The standby normal releases are used for both standby and preoperational startup.

The normal operational releases from target irradiation at one or two new accelerators (a low-energy accelerator and a high-energy accelerator) at a generic DOE site used in the analysis of public health impacts are derived from information in Appendix F. The release terms for that accelerator were modified to reflect differences in energy levels to produce the releases provided in **Table H–8**.

Table H–8 Annual Normal Operational Releases from the Low- and High-Energy Accelerators

Accelerator Startup Operations				
	Low-Energy Accelerator		**High-Energy Accelerator**	
Isotope	**Airborne Release (curies per year)**	**Liquid Release (curies per year)**	**Airborne Release (curies per year)**	**Liquid Release (curies per year)**
Nitrogen-13	0.0027	0.33	0.039	4.7
Carbon-14	8.0×10^{-5}	–	0.0011	–
Beryllium-7	2.7×10^{-5}	0.0014	3.9×10^{-4}	0.020
Fluorine-18	–	6.5×10^{-4}	–	0.0096
Argon-41	9.7×10^{-4}	–	0.014	–
Hydrogen-3	3.6×10^{-5}	–	5.1×10^{-4}	–
Accelerator Normal Operations				
Nitrogen-13	0.052	0.33	0.74	4.7
Carbon-14	8.0×10^{-5}	–	0.0011	–
Beryllium-7	2.4×10^{-4}	0.0014	0.0034	0.020
Fluorine-18	1.0×10^{-4}	6.5×10^{-4}	0.0014	0.0096
Argon-41	1	–	22	–
Hydrogen-3	3.6×10^{-5}	–	5.1×10^{-4}	–

The normal operational releases from target irradiation at a new research reactor at a generic DOE site used in the analysis of public health impacts are taken from the analysis in Appendix E. The isotopes and release quantities are provided in **Table H–9**.

Table H–9 Annual Normal Operational Releases from the Research Reactor

Isotope	Quantity Released (curies per year)
Tritium	0.1
Argon-41	2.8

A site has not been selected for either the accelerator(s) or the research reactor. The analysis of public health impacts was performed by assuming a population distribution consisting of a uniform population density of 100 people per square mile within 10 miles of the facility (excluding the area within 2 miles of the facility that was assumed to be within the DOE property) and a density of 200 people per square mile for the area 10 to 50 miles from the facility. Also, a representative "generic" meteorological profile was selected. This weather profile was determined to be representative of the average normal weather conditions for the continental United States. Additional information supporting the selection of this population and weather profile is provided in Appendix I, as the same information has been used for the analyses of both normal operations and accident-related public health impacts.

In the event that DOE selects an alternative that incorporates the use of either the new reactor or the accelerator(s), a specific DOE site would have to be selected for the location of these generic facilities. Selection of a specific site would require additional site- and facility-specific National Environmental Policy Act analysis and documentation, which would address the potential human health impacts associated with operation of the facility at the selected site.

Chapter 3 provides information on the current environmental impacts associated with the operation of reactors at the three candidate DOE sites. Included in these current impacts is an assessment of the radiological impact of operation of HFIR at ORR and ATR at INEEL. **Tables H–10** and **H–11** list the isotopes used in the evaluation of the population doses resulting from normal operations of these two reactors. Population doses calculated for HFIR are based on monitored operational releases during the years 1997 through 1999 (Boyd 2000). For HFIR, the largest annual release for each isotope released through the facility stack during one of these three years was selected. Table H–11 is based on ATR emission data for the year 1999 (Perry 2000).

Table H–10 Maximum HFIR Annual Releases for 1997 Through 1999

Isotope[a]	Quantities Released (curies per year)	Isotope[a]	Quantities Released (curies per year)
Argon-41	13,000	Krypton-87	56
Barium-139	0.27	Krypton-88	66
Barium-140	4.8×10^{-4}	Krypton-89	44
Cesium-137	0.83	Lanthanum-140	1.6×10^{-4}
Cesium-138	4700	Lead-212	0.26
Tritium (hydrogen-3)	96	Ruthenium-105	0.050
Iodine-129	3.8×10^{-4}	Xenon-131m	27
Iodine-131	0.062	Xenon-133	320
Iodine-132	0.50	Xenon-133m	3.1
Iodine-133	0.37	Xenon-135	190
Iodine-134	0.040	Xenon-135m	120
Iodine-135	1.0	Xenon-137	300
Krypton-85	480	Xenon-138	770
Krypton-85m	16		

a. Only isotopes with release quantities of 10^{-4} curies or greater were included in the analysis.

Table H–11 ATR Stack Releases for 1999

Isotope[a]	Quantities Released (curies per year)	Isotope[a]	Quantities Released (curies per year)
Argon-41	1200	Iodine-135	0.0036
Cobalt-60	1.4×10^{-4}	Sodium-24	5.5×10^{-4}
Chromium-51	0.0025	Rubidium-88	0.42
Cesium-138	0.021	Technetium-99m	0.0011
Iodine-131	8.9×10^{-4}	Xenon-133	11
Iodine-132	0.0015	Xenon-135	15
Iodine-133	0.0029	Gross α (plutonium-239)	6.0×10^{-6}
Iodine-134	0.0022	Gross β (strontium-90)	5.0×10^{-4}

a. Only isotopes with release quantities of 10^{-4} curies or greater were included in the analysis.

Table H–12 provides the results of the human health impacts—population dose and maximally exposed individual dose—for each of the scenarios evaluated using GENII. Some of these results are combined to produce the human health effects from normal operations presented in Chapter 4. For example, the impacts associated with the operation of FFTF are the combined impacts of the releases from the three identified release points (the combined exhaust system, the heat transfer system, and the reactor service building). Other health impacts that are reported in Chapter 4 are the result of multiple activities being performed in the same facility. For example, target processing (fabrication and irradiated target processing) and storage may both be done at the Radiochemical Engineering Development Center (REDC), the Fluorinel Dissolution Process Facility (FDPF), and the Fuels and Materials Examination Facility (FMEF). The health impacts from these combined activities are also presented in Table H–12.

Storage of the neptunium-237 is considered for each of the alternatives. (In the No Action Alternative, storage is the only activity assessed.) Releases associated with storage of the neptunium-237 would be expected to be essentially zero. Neptunium-237 would remain in its containment vessels and would be stored in a shielded area (see Section A.1.2). However, it has been assumed that the doses due to storage would be 10 percent of the doses due to processing activities. The analysis of health impacts that could result from processing irradiated targets is based on information that accounts for normal handling of the targets during operation as well as accidents that could occur during handling, such as spills. The determining factor used for annual releases is the amount of material that would be released from an operational spill. During storage, the neptunium-237 would not be subject to activities that could result in a material spill. Therefore, this assumption ensures that the doses associated with neptunium-237 storage have been conservatively addressed.

Occupational (Worker) Health Impacts

Health impacts from radiological exposure due to normal facility operation were determined for the facility worker directly involved in the fabrication, irradiation, processing, and storage of the medical isotope and plutonium-238 targets. Health risks to individual workers and to the total workforce were assessed.

The dose to facility workers was derived from recorded occupational exposures at the candidate facilities, or from recorded exposures at facilities that perform similar operations as those being considered in each of the alternatives.

Typically, either the average annual worker dose or the total workforce dose has been provided. The number of workers has been estimated based on prior experience with similar activities at the facility or on activities at similar facilities with the same type of operations.

Table H–12 Radiological Impacts on Populations from Normal Operations

Facility	Process	Population Dose (person-rem)	Maximally Exposed Individual Dose (millirem)
REDC	Neptunium target processing	8.0×10^{-5}	1.7×10^{-6}
FDPF	Neptunium target processing	3.5×10^{-6}	1.4×10^{-7}
FMEF	Neptunium target processing	4.0×10^{-5}	4.3×10^{-7}
	Target processing - all targets[a]	0.085	3.0×10^{-4}
RPL	Preoperational activities	1	0.043
	Medical and research and development target processing	0.21	0.005
FFTF	Standby	0.028	1.3×10^{-4}
	Standby: gross alpha and beta emissions	4.8×10^{-4}	6.6×10^{-6}
	Normal operations: combined exhaust system release	0.044	4.1×10^{-4}
	Normal operations: heat transfer system release	3.7×10^{-5}	3.3×10^{-7}
	Normal operations: reactor service building release	4.3×10^{-5}	3.9×10^{-7}
Low-energy accelerator	Preoperation/startup	0.0024	1.4×10^{-5}
	Normal operations: target irradiation	0.0043	1.1×10^{-4}
High-energy accelerator	Preoperation/startup	0.035	1.8×10^{-4}
	Normal operations: target irradiation	0.055	8.8×10^{-4}
Generic research reactor	Normal operations: target irradiation	0.0023	6.8×10^{-5}
Generic support facility	Normal operations: medical and research and development target processing	0.01	0.0025
ATR	Current operations[b]	0.013	0.0013
HFIR	Current operations[b]	8.4	0.46
Human Health Impacts Derived from the Above GENII Computer Analysis Results			
REDC	Neptunium-237 storage	8.0×10^{-6}	1.7×10^{-7}
	Target processing and storage	8.8×10^{-5}	1.9×10^{-6}
FDPF	Neptunium-237 storage	3.5×10^{-7}	2.4×10^{-8}
	Target processing and storage	3.9×10^{-6}	2.6×10^{-7}
FMEF	Neptunium-237 storage	4.0×10^{-6}	4.3×10^{-8}
	Neptunium target processing and storage	4.4×10^{-5}	4.7×10^{-7}
	Target processing and storage: all targets	0.085	3.0×10^{-4}
FFTF	Preoperations and standby	0.028	1.4×10^{-4}
	Normal operations: target irradiation	0.044	4.1×10^{-4}
	Deactivation[c]	0.036	0.00026

a. All targets include medical isotope, civilian nuclear energy research and development, and neptunium targets.
b. Impacts are from existing operations at this reactor. Incremental impacts are 0.
c. FFTF deactivation health impacts are derived from the normal and standby calculations and information provided in the *Environmental Assessment, Shutdown of the Fast Flux Test Faciltiy* (DOE 1995).

Note: Target processing includes target fabrication and irradiated target processing.

Table H–13 provides the source data used for the calculation of worker health impacts from radiological exposure associated with normal operations. Additional health impacts (latent cancer fatalities) are derived from these dose parameters, and this information is presented in Chapter 4 for each of the alternatives. Worker doses were converted into the number of projected latent cancer fatalities using the risk estimator of 400 fatal cancers per million person-rem given in the *1990 Recommendation of the International Commission on Radiological Protection* (ICRP 1991). This risk estimator, compared with the estimator of 500 fatal cancers per million person-rem for members of the public, reflects the absence of infants and children (the most radiosensitive age groups) from the workforce.

Table H–13 Radiological Impacts on Workers from Normal Operations

Activity	Number of Workers	Average Annual Individual Dose (millirem)	Total Annual Workforce Dose (person-rem)	Source
FFTF in standby	200	3.5[a]	0.69	Nielsen 1999
FFTF preoperational	200	3.5[a]	0.69	Nielsen 1999
FFTF operational	200	6.6[a]	1.3	Nielsen 1999
FFTF deactivation	10	6	0.06[a]	DOE 1995
FMEF medical target processing	30	160	4.8[a]	BWHC 1999
FMEF plutonium-238 target processing	75	170	12	Wham 2000
FMEF total target processing	105[b]	160	17[b]	–
Hanford RPL preoperational	40	81	3.2[a]	BWHC 1999
Hanford RPL operations	30	160	4.8[a]	BWHC 1999
ORR REDC	75	170[a]	12	Wham 2000
INEEL FDPF	75	170[a]	12	Wham 2000
INEEL ATR[c]	0	0	0	–
ORR HFIR[c]	0	0	0	–
Generic CLWR[c]	0	0	0	–
Low-energy accelerator startup	150	150	23[a]	TechSource 2000[d]
Low-energy accelerator operations	100	150	15[a]	TechSource 2000
High-energy accelerator startup	300	150	45[a]	TechSource 2000[d]
High-energy accelerator operations	200	150	30[a]	TechSource 2000
Research reactor operations	120	100	12[a]	Appendix E
Accelerator and reactor support facility operations	100	102	10[a]	BWHC 1999
Low-energy accelerator decontamination and decommissioning	35	160	5.6[a]	Gallagher 2000
High-energy accelerator decontamination and decommissioning	70	160	11[a]	Gallagher 2000
Research support facility	40	25	1	NRC 1988
Research reactor decontamination and decommissioning	40	275	11	NRC 1988

a. This value is derived from the other two parameters for this facility.
b. These values are the sum of medical isotope target and plutonium target processing at this facility.
c. There are no incremental worker impacts from the use of these currently operating facilities.
d. Number of workers for startup increased by 50 percent from normal operation staff levels.

Support facility worker dose estimations are derived from the dose estimates for the operation of RPL at Hanford for the fabrication, processing, and storage of medical isotope targets. The support facility would meet the same DOE requirements and similar administrative requirements for the radiological protection of workers as at existing facilities. Because similar processes would be performed at the support facility as at RPL, it was assumed that radiological and nonradiological worker doses would be similar.

H.3 IMPACTS OF EXPOSURES TO HAZARDOUS CHEMICALS ON HUMAN HEALTH

The potential impacts of exposure to hazardous chemicals released to the atmosphere were evaluated for routine operations associated with the alternatives analyzed in this NI PEIS.

The receptors considered in these evaluations are the public. Impacts of exposures to hazardous chemicals for workers directly involved in the treatment process were not quantitatively evaluated because workers use

personal protective equipment and engineering process controls which limits their exposure to levels within applicable Occupational Safety and Health Administration Permissible Exposure Limits or American Conference of Governmental Industrial Hygienists Threshold Limit Values.

As a result of releases from routine operations, receptors are expected to be potentially exposed to concentrations of hazardous chemicals that are below those that could cause acutely toxic health effects. Acutely toxic health effects generally result from short-term exposure to relatively high concentrations of contaminants, such as those that may be encountered during facility accidents. Long-term exposure to relatively lower concentrations of hazardous chemicals can produce adverse chronic health effects that include both carcinogenic and noncarcinogenic effects. The health effect endpoints evaluated in this analysis include excess incidences of latent cancers for carcinogenic chemicals, and a spectrum of chemical-specific noncancer health effects such as headache, membrane irritation, neurotoxicity, immunotoxicity, liver toxicity, kidney toxicity, developmental toxicity, reproductive toxicity, and genetic toxicity for noncarcinogens.

METHODOLOGY

Annual airborne concentrations of hazardous chemicals were estimated from the expected chemical usage provided by the sites and a conservative screening dispersion model described in Chapter 4.

This NI PEIS estimates the noncancer health risks by comparing annual air concentrations of contaminants to the EPA Reference Concentrations published in the Integrated Risk Information System. For each noncarcinogenic chemical, potential health risks are estimated by dividing the estimated airborne concentration by the chemical-specific Reference Concentration value to obtain a noncancer hazard quotient:

$$\text{Noncancer Hazard Quotient} = \text{air concentration}/\text{Reference Concentrations}$$

Reference Concentrations are estimates (with an uncertainty spanning perhaps an order of magnitude) of a daily exposure to the human population (including sensitive subgroups) that is likely to be without appreciable risk of harmful effects during a lifetime. Hazard Quotients are calculated for each hazardous chemical to which receptors may be exposed. Hazard Quotients for each chemical are summed to generate a Hazard Index. The Hazard Index is an estimate of the total noncancer toxicity potential from exposure to hazardous chemicals. According to EPA risk assessment guidelines, if the Hazard Index value is less than or equal to 1.0, the exposure is unlikely to produce adverse toxic effects. If the Hazard Index exceeds 1.0, adverse noncancer health effects may result from the exposure.

For carcinogenic chemicals, risk is estimated by the following equation:

$$\text{Risk} = \text{CA} \times \text{URF}$$

where:

Risk	=	a unitless probability of cancer incidence
CA	=	contaminant concentration in air (in micrograms per cubic meters)
URF	=	cancer inhalation unit risk factor (in units of cancers per micrograms per cubic meters)

Cancer unit risk factors are used in risk assessments to estimate an upper-bound lifetime probability of an individual developing cancer as a result of exposure to a particular level of a potential carcinogen.

ASSUMPTIONS

The airborne pathway is assumed to be the principal exposure route by which the offsite population maximally exposed individual is exposed to hazardous chemicals released from processing facilities. No synergistic or antagonistic effects are assumed to occur from exposure to the hazardous chemicals. Synergistic effects among released contaminants may result in adverse health effects that are greater than those estimated, whereas antagonistic effects among released chemicals may result in less severe health effects than those estimated.

Because released contaminants may have either synergistic or antagonistic effects, chemical risk factors are not combined into a single chemical risk factor, either carcinogenic or noncarinogenic. A risk factor, either a cancer unit risk factor or a noncancer hazard quotient, is derived for each released contaminant. (The analyses presented in Chapter 4 provide the risk factors for all chemical contaminants considered. In providing the overview of the chemical hazards in Chapter 2, only the highest chemical risk factor is presented.) These chemical cancer unit risk factors are a measure of the likelihood of an individual developing cancer given an exposure to a particular concentration of a known carcinogen for a specified duration. Population risk factors are not developed for chemical contaminants. Concentrations and exposure periods used in this analysis are the maximum that a member of the public could be expected to encounter. (In contrast, the population risk factors developed for the radiological impacts of normal operation do take into consideration the population distribution and varying isotope concentrations, and resulting dose, depending on an individual's location.) Not all of the population could experience this level of exposure. It would be inaccurate to apply the chemical risk factor to the entire population—an application that would overestimate the impact on the total population.

ANALYSIS

The potential impacts of exposure to hazardous chemicals released to the atmosphere during routine operations of the processing facilities are presented in Chapter 4 for each alternative.

H.4 REFERENCES

Code of Federal Regulations

10 CFR Part 20, "Standards for Protection Against Radiation," U.S. Nuclear Regulatory Commission.

10 CFR Part 835, "Occupational Radiation Protection," U.S. Department of Energy.

40 CFR Part 61, "National Emission Standards for Hazardous Air Pollutants," U.S. Environmental Protection Agency.

40 CFR Part 141, "National Primary Drinking Water Regulations," U.S. Environmental Protection Agency.

40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," U.S. Environmental Protection Agency.

DOE Orders

DOE Order 5400.5, *Radiation Protection of the Public and the Environment*, January 7, 1993.

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Wham, R.M., 1999, Oak Ridge National Laboratory, Oak Ridge, TN, personal communication to G. Waldman, Science Applications International Corporation, Germantown, MD, *Population Dose*, February 18.

Wham, R.M., 2000, Oak Ridge National Laboratory, Oak Ridge, TN, personal communication to J. Schinner, Science Applications International Corporation, Germantown, MD, *Data for Estimating Radworker Dose*, September 18.

Appendix I
Evaluation of Human Health Effects from Facility Accidents

This appendix presents the method and assumptions used for estimating potential impacts on, and risks to, individuals and the general public from exposure to releases of radioactive and hazardous chemical materials during hypothetical accidents at irradiation and processing facilities cited under the production alternatives described in this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]). The impacts of accidental radioactive material releases are given in Section I.1; the impacts of releases of hazardous chemicals, in Section I.2; and industrial accident impacts in Section I.3.

I.1 Radiological Accident Impacts on Human Health

The accidents considered in this NI PEIS for both the irradiation facilities and the processing facilities were based on a complete spectrum of accidents ranging from high-probability low-consequence events to extremely unlikely and incredible events. For this NI PEIS, a design-basis accident and a beyond-design-basis accident were specifically evaluated for each facility. More frequent events were specifically evaluated at the processing facilities because of the contribution to risk. These higher frequency events were not specifically evaluated for the irradiation facilities because they do not contribute to the risk (i.e., the risks of the design-basis accident and beyond-design-basis accidents are orders of magnitude higher than any more frequent event).

An extensive review of facility safety documentation (safety analysis reports, process hazard reviews, hazard analysis documents, and probabilistic risk analyses) was conducted. The review identified several accidents and their causes (initiating events). The initiating events reviewed included external events (e.g., airplane crashes, nearby explosions, fires), internal events (e.g., equipment failures, human error), natural phenomena (e.g., floods, tornadoes, earthquakes), and sabotage and terrorist activities. The review also determined that the only significant common-cause initiating event would be a catastrophic earthquake. In a common-cause event, the consequences from colocated facilities are summed. However, because of the low frequency of a catastrophic earthquake, the accidents evaluated in this NI PEIS bound the risks of a common-cause summation.

The accidents were grouped into one of four categories—anticipated occurrences, unlikely events, extremely unlikely events, or incredible events—based on the estimated frequency of occurrence. The accidents within each frequency category were examined to determine which accident(s) would result in the highest consequences (i.e., dose) and the highest risks (frequency × consequence). As a result, all other accident scenarios were screened from further consideration in this NI PEIS because the consequences and risks associated with those accidents would be lower than—or bounded by—the consequences and risks of the selected accidents.

The accident evaluation methodology ensures that all the facilities are treated on an equal basis. The analysis also considered facility-specific differences in design and mitigation features (e.g., filtration systems). Filtration efficiencies were obtained from facility safety reports, facility descriptions, and appropriate U.S. Department of Energy (DOE) and U.S. Nuclear Regulatory Commission (NRC) guidance.

I.1.1 Irradiation Facility Accident Scenario Selection and Description

A spectrum of potential accident scenarios was considered in this accident analysis assessment for the High Flux Isotope Reactor (HFIR) at the Oak Ridge National Laboratory (ORNL) in Oak Ridge, Tennessee; the

Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL) near Idaho Falls, Idaho; a generic commercial light water reactor (CLWR); the Fast Flux Test Facility (FFTF) at the Hanford Site near Richland, Washington; low-energy and high-energy accelerators at a generic site; and a new research reactor at a generic site.

For each irradiation facility, a spectrum of accidents encompassing the full range of probabilities and consequences was considered for evaluation and inclusion in this NI PEIS. From the reactor final safety analysis reports, it was determined that only a few low-probability design-basis accidents or very low probability beyond-design-basis accidents contributed significantly to risk. Hence, only these events were specifically evaluated in this NI PEIS. In addition, handling accidents involving irradiated targets were also analyzed for HFIR, ATR, and FFTF. For the generic CLWR and the new research reactor, NRC guidance and published studies were used to determine appropriate design-basis and beyond-design-basis accidents. The specific guidance and studies used are presented in each of the following reactor analysis sections. The irradiation facilities analyses include the development of accident scenarios, transport of radioisotopes, and evaluation of health consequences, in addition to discussions of the methodologies used in these evaluations.

Accident	Frequency Range
Anticipated occurrences	1.0 – 0.01
Unlikely events	1×10^{-2} – 1×10^{-4}
Extremely unlikely events	1×10^{-4} – 1×10^{-6}
Incredible events	$<1\times10^{-6}$

Irradiation facility accident source terms include postulated neptunium-237 targets with a common spectrum of isotopes at the end of the plutonium-238 production cycle. The accident consequences were analyzed with end-of-cycle irradiated targets. Because of the radioisotope content, the end-of-cycle irradiated targets contribute most significantly to offsite consequences. **Table I–1** presents the inventory of target radioisotopes per gram of plutonium-238 produced.

Table I–1 Neptunium-237 Irradiated Target End-of-Cycle Nuclide Inventory (All Values Normalized to 1 Gram of Plutonium-238)

Isotope	Curies	Isotope	Curies
Cobalt-58	0.0	Tellurium-132	47.1
Cobalt-60	0.0	Iodine-131	32.5
Krypton-85	0.0202	Iodine-132	48.7
Krypton-85m	5.30	Iodine-133	65.0
Krypton-87	8.83	Iodine-134	69.0
Krypton-88	12.4	Iodine-135	60.8
Rubidium-86	0.00762	Xenon-133	61.3
Strontium-89	9.63	Xenon-135	7.69
Strontium-90	0.127	Cesium-134	0.159
Strontium-91	23.4	Cesium-136	0.92
Strontium-92	28.4	Cesium-137	0.375
Yttrium-90	0.128	Barium-139	54.1
Yttrium-91	13.2	Barium-140	45.1
Yttrium-92	28.7	Lanthanum-140	44.5
Yttrium-93	37.2	Lanthanum-141	51.3
Zirconium-95	24.7	Lanthanum-142	47.6
Zirconium-97	51.0	Cerium-141	35.0
Niobium-95	16.8	Cerium-143	42.5
Molybdenum-99	56.3	Cerium-144	7.13
Technetium-99m	50.1	Praseodymium-143	35.6
Ruthenium-103	42.5	Neodymium-147	17.1
Ruthenium-105	51.7	Neptunium-237	0.0036
Ruthenium-106	6.41	Neptunium-239	16.8
Rhodium-105	41.1	Plutonium-238	17
Antimony-127	4.44	Plutonium-239	0.00921
Antimony-129	13.5	Plutonium-240	0.00393
Tellurium-127	4.18	Plutonium-241	0.853
Tellurium-127m	0.243	Americium-241	0.0
Tellurium-129	12.9	Curium-242	0.0122
Tellurium-129m	1.39	Curium-244	0.0
Tellurium-131m	5.96	**Total**	**1,358.6**

Source: Schnitzler 1999.

The FFTF reactor, low-energy accelerator, and new research reactor accident source terms include medical, industrial, and research and development isotope targets. Projected radioisotope inventories for the target systems most likely to be considered for medical, industrial, and research and development isotope production are presented in **Table I–2**. These are maximum irradiated target inventories. The radium-226 target for actinium-227 production is the only target with a significantly radioactive target material. However, the dose due to the radium-226 target is insignificant compared with the dose due to the product isotopes. Therefore, the accident consequences were analyzed with the irradiated target products. Several of the isotope production targets generate substantial amounts of radioactive byproduct isotope in addition to the desired product. In these cases (gadolinium-153, actinium-227, and plutonium-238 production targets), the additional target inventory was included when calculating consequences.

Table I–2 Medical, Industrial, and Research and Development Isotope Irradiated Target Product Inventories

Product Isotope	Radioisotope	Target Inventory (curies)
Rapid Radioisotope Retrieval System		
Gold-198	Gold-198	132
Copper-64	Copper-64	1,300
Copper-67	Copper-67	6.26
Holmium-166	Holmium-166	58.9
Iodine-125	Iodine-125	2,530
Iodine-131	Iodine-131	307
Lutecium-177	Lutecium-177	0.519
Molybdenum-99	Molybdenum-99	1,680
Phosphorus-32	Phosphorus-32	39.1
Palladium-103	Palladium-103	1,340
Platinum-195m	Platinum-195m	168
Rhenium-186	Rhenium-186	4,350
Scandium-47	Scandium-47	29.6
Samarium-153	Samarium-153	70.7
Tin-117m	Tin-117m	48.5
Long-Term Irradiation Vehicle		
Cadmium-109	Cadmium-109	656
Gadolinium-153[a]	Gadolinium-153	1,100
Gadolinium-153	Europium-152	4,660
Gadolinium-153	Europium-152m	6.41×10^{4}
Gadolinium-153	Europium-154	1.55×10^{4}
Gadolinium-153	Europium-154m	2.20×10^{-4}
Gadolinium-153	Europium-155	3,540
Gadolinium-153	Europium-156	3.39×10^{5}
Gadolinium-153	Samarium-153	3.16×10^{4}
Iridium192	Iridium-192	3,570
Osmium-194	Osmium-194	2.20
Phosphorus-33	Phosphorus-33	76.2
Selenium-75	Selenium-75	17.9
Samarium-145	Samarium-145	11.8
Strontium-85	Strontium-85	2,160
Strontium-89	Strontium-89	156
Tungsten-188	Tungsten-188	5,810
Xenon-127	Xenon-127	7.26
Yttrium-91	Yttrium-91	17.8
Actinium-227[a]	Actinium-227	34.0
Actinium-227	Actinium-228	56.1
Actinium-227	Actinium-229	6.04×10^{-9}
Actinium-227	Radium-226	14.3
Actinium-227	Radium-227	4.23×10^{-7}
Actinium-227	Radium-228	0.00101
Actinium-227	Radium-229	5.00×10^{-14}
Actinium-227	Thorium-227	24.8
Actinium-227	Thorium-228	42.1

Product Isotope	Radioisotope	Target Inventory (curies)
Actinium-227	Thorium-229	8.63×10^{-4}
Actinium-227	Actinium-225	3.72×10^{-4}
Actinium-227	Astatine-217	3.72×10^{-4}
Actinium-227	Bismuth-210	0.109
Actinium-227	Bismuth-211	19.6
Actinium-227	Bismuth-212	24.6
Actinium-227	Bismuth-213	3.71×10^{-4}
Actinium-227	Bismuth-214	14.3
Actinium-227	Francium-221	3.72×10^{-4}
Actinium-227	Francium-223	1.40×10^{-5}
Actinium-227	Lead-209	3.69×10^{-4}
Actinium-227	Lead-210	0.118
Actinium-227	Lead-211	19.6
Actinium-227	Lead-212	38.4
Actinium-227	Lead-214	14.3
Actinium-227	Polonium-210	0.106
Actinium-227	Polonium-211	0.0535
Actinium-227	Polonium-212	24.6
Actinium-227	Polonium-213	3.63×10^{-4}
Actinium-227	Polonium-214	14.3
Actinium-227	Polonium-215	19.6
Actinium-227	Polonium-216	38.8
Actinium-227	Polonium-218	14.3
Actinium-227	Radium-223	19.6
Actinium-227	Radium-224	38.8
Actinium-227	Radium-225	5.46×10^{-4}
Actinium-227	Radon-217	4.46×10^{-8}
Actinium 227	Radon-219	19.6
Actinium-227	Radon-220	38.8
Actinium-227	Radon-222	14.3
Actinium-227	Thallium-207	19.6
Actinium-227	Thallium-208	8.83
Actinium-227	Thallium-209	8.16×10^{-6}

a. The gadolinium-153 and actinium-227 production targets include radioactive byproducts.
Source: BWHC 1999.

I.1.1.1 Advanced Test Reactor

ATR would generate 5 kilograms (11 pounds) per year of plutonium-238 in support of Alternative 2, Options 1 through 3, and 3 kilograms (6.6 pounds) per year of plutonium-238 in support of Alternative 2, Options 7 through 9. On average, ATR has seven refueling outages per year. ATR accident analyses assumed that one-seventh of the annual plutonium-238 production would be harvested at each refueling outage and an equal amount of plutonium-238 would remain in the core in targets that were not ready to be harvested. The accident analyses postulated that the plutonium-238 at risk in targets during ATR accidents is 857 grams (1.89 pounds) for the annual production rate of 3 kilograms (6.6 pounds) per year and 1,429 grams (3.144 pounds) for the annual production rate of 5 kilograms (11 pounds) per year.

I.1.1.1.1 Design-Basis Accident

The *ATR Upgraded Final Safety Analysis Report* (LMIT 1998) stated that seven design-basis accidents would provide the greatest challenge to the engineered safety features of ATR. These accidents and the affected engineered safety systems are summarized in **Table I–3**.

Table I–3 ATR Engineered Safety Feature Design-Basis Accidents

Accident Sequence	Engineered Safety System
3-inch diameter opening in the primary coolant system due to an opening of a drain valve, relief valve, or vent valve	Emergency firewater injection system
Experiment loop piping failure	Radiation monitoring and seal system
Long-term complete loss of flow or complete loss of heat sink	Primary coolant overpressure relief and vent systems and emergency firewater injection system
Opening of flow control butterfly valve to full open	Primary pump shutoff system
Loss of primary coolant system pressure control (loss of instrument air)	Pressurizing pumps and gland seal pumps shutoff system
Loss of pressure control of primary coolant system and failure of the pressurizing pumps and gland seal pumps shutoff system	Primary coolant overpressure relief system
Loss of primary coolant system inventory during depressurized and outage operations when irradiated fuel elements are in the reactor vessel	Vessel level alarm system

Source: LMIT 1998.

The accident sequences listed in the table do not lead to core damage and do not have the potential to damage appropriately designed neptunium-237 targets being irradiated in the core.

I.1.1.1.2 Severe Reactor Accident

The large-break loss-of-coolant accident postulated for ATR is a severe reactor accident. This event would result in a decrease in the primary coolant inventory of ATR. As treated in the *ATR Upgraded Final Safety Analysis Report*, the large-break loss-of-coolant accident is a limiting accident compared with other initiating events because 100 percent core damage is estimated to occur. The probability for the occurrence of an ATR large-break loss-of-coolant accident is on the order of 1×10^{-4} per year.

The radiological analysis of the large-break loss-of-coolant accident shows that an ATR core inventory of 1.11 gigacuries at reactor scram conditions releases an available source term of 175 megacuries (LMIT 1998). The emergency firewater injection system is assumed to pump water through the break into confinement, until shutoff level is reached, about 33 hours after the break. Within that period, about 65 percent of the available source term, or 113 megacuries, will have been released as the early release source term. Following the termination of emergency firewater injection system flow at 33 hours, the confinement leak rate is assumed to drop to the design value of 10 percent per day, resulting in a release of the remaining 62 megacuries as the late-release source term, ending about 85 hours after the loss-of-coolant accident. Consequently, the total release duration for the large-break loss-of-coolant accident is 118.5 hours, or the sum of 33.3 hours for the early-release source term and 85.2 hours for the late-release source term.

The core inventories and environmental releases for the three possible plutonium-238 production rates (0, 3, or 5 kilograms per year) are presented in **Table I–4**. The core inventory was based on a maximum design power level of 250 megawatts.

Table I–4 ATR Large-Break Loss-of-Coolant Accident Source Terms

Isotope	Core Inventory (curies) Versus Plutonium-238 Production Rate			Environmental Release (curies) Versus Plutonium-238 Production Rate		
	0 kilograms per year	3 kilograms per year	5 kilograms per year	0 kilograms per year	3 kilograms per year	5 kilograms per year
Krypton-85	5,900	6,000	6,000	5,900	6,000	6,000
Krypton-85m	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}
Krypton-87	5.2×10^{6}	5.2×10^{6}	5.2×10^{6}	5.2×10^{6}	5.2×10^{6}	5.2×10^{6}
Krypton-88	7.3×10^{6}	7.3×10^{6}	7.3×10^{6}	7.3×10^{6}	7.3×10^{6}	7.3×10^{6}
Rubidium-86	3,900	3,900	3,900	1,900	1,900	1,900
Strontium-89	5.6×10^{6}	5.6×10^{6}	5.6×10^{6}	3.4×10^{5}	3.4×10^{5}	3.4×10^{5}
Strontium-90	4.7×10^{4}	4.7×10^{4}	4.7×10^{4}	2,800	2,800	2,800
Strontium-91	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	7.1×10^{5}	7.1×10^{5}	7.1×10^{5}
Strontium-92	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	7.2×10^{5}	7.3×10^{5}	7.3×10^{5}
Yttrium-90	4.9×10^{4}	4.9×10^{4}	4.9×10^{4}	200	200	200
Yttrium-91	6.2×10^{6}	6.2×10^{6}	6.2×10^{6}	2.5×10^{4}	2.5×10^{4}	2.5×10^{4}
Yttrium-92	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.8×10^{4}	4.9×10^{4}	4.9×10^{4}
Yttrium-93	1.3×10^{7}	1.3×10^{7}	1.3×10^{7}	5.2×10^{4}	5.2×10^{4}	5.2×10^{4}
Zirconium-95	6.4×10^{6}	6.4×10^{6}	6.4×10^{6}	2.6×10^{4}	2.6×10^{4}	2.6×10^{4}
Zirconium-97	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}
Niobium-95	2.9×10^{6}	2.9×10^{6}	2.9×10^{6}	1.1×10^{4}	1.2×10^{4}	1.2×10^{4}
Molybdenum-99	1.2×10^{7}	1.3×10^{7}	1.3×10^{7}	2.5×10^{5}	2.5×10^{5}	2.5×10^{5}
Technetium-99m	1.1×10^{7}	1.1×10^{7}	1.1×10^{7}	2.2×10^{5}	2.3×10^{5}	2.3×10^{5}
Ruthenium-103	4.3×10^{6}	4.3×10^{6}	4.3×10^{6}	8.6×10^{4}	8.6×10^{4}	8.7×10^{4}
Ruthenium-105	2.2×10^{6}	2.3×10^{6}	2.3×10^{6}	4.4×10^{4}	4.5×10^{4}	4.6×10^{4}
Ruthenium-106	9.8×10^{4}	1.0×10^{5}	1.1×10^{5}	2,000	2,100	2,100
Rhodium-105	1.5×10^{6}	1.5×10^{6}	1.5×10^{6}	3.0×10^{4}	3.0×10^{4}	3.1×10^{4}
Antimony-127	3.4×10^{5}	3.4×10^{5}	3.4×10^{5}	1.0×10^{5}	1.0×10^{5}	1.0×10^{5}
Antimony-129	1.4×10^{6}	1.4×10^{6}	1.4×10^{6}	4.2×10^{5}	4.3×10^{5}	4.3×10^{5}
Tellurium-127	3.2×10^{5}	3.2×10^{5}	3.2×10^{5}	0	0	0
Tellurium-127m	1.4×10^{4}	1.4×10^{4}	1.4×10^{4}	0	0	0
Tellurium-129	1.4×10^{6}	1.4×10^{6}	1.4×10^{6}	0	0	0
Tellurium-129m	1.5×10^{5}	1.5×10^{5}	1.5×10^{5}	0	0	0
Tellurium-131	5.3×10^{6}	5.3×10^{6}	5.3×10^{6}	0	0	0
Tellurium-131m	7.6×10^{5}	7.6×10^{5}	7.7×10^{5}	0	0	0
Tellurium-132	8.9×10^{6}	9.0×10^{6}	9.0×10^{6}	0	0	0
Iodine-131	6.0×10^{6}	6.0×10^{6}	6.0×10^{6}	3.2×10^{5}	3.2×10^{5}	3.2×10^{5}
Iodine-132	9.1×10^{6}	9.1×10^{6}	9.1×10^{6}	4.8×10^{5}	4.8×10^{5}	4.8×10^{5}
Iodine-133	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}	7.3×10^{5}	7.3×10^{5}	7.3×10^{5}
Iodine-134	1.5×10^{7}	1.5×10^{7}	1.6×10^{7}	8.2×10^{5}	8.2×10^{5}	8.2×10^{5}
Iodine-135	1.3×10^{7}	1.3×10^{7}	1.3×10^{7}	6.8×10^{5}	6.8×10^{5}	6.8×10^{5}
Xenon-133	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}	1.4×10^{7}
Xenon-135	4.7×10^{5}	4.8×10^{5}	4.8×10^{5}	4.7×10^{5}	4.8×10^{5}	4.8×10^{5}
Cesium-134	3.8×10^{4}	3.8×10^{4}	3.8×10^{4}	0	0	0
Cesium-136	2.7×10^{4}	2.8×10^{4}	2.8×10^{4}	0	0	0
Cesium-137	4.8×10^{4}	4.9×10^{4}	4.9×10^{4}	0	0	0
Barium-139	1.3×10^{7}	1.3×10^{7}	1.3×10^{7}	7.8×10^{5}	7.9×10^{5}	7.9×10^{5}
Barium-140	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	7.5×10^{5}	7.5×10^{5}	7.5×10^{5}

Isotope	Core Inventory (curies) Versus Plutonium-238 Production Rate			Environmental Release (curies) Versus Plutonium-238 Production Rate		
	0 kilograms per year	3 kilograms per year	5 kilograms per year	0 kilograms per year	3 kilograms per year	5 kilograms per year
Lanthanum-140	1.3×10^{7}	1.3×10^{7}	1.3×10^{7}	5.0×10^{4}	5.0×10^{4}	5.0×10^{4}
Lanthanum-141	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}
Lanthanum-142	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.8×10^{4}	4.8×10^{4}	4.8×10^{4}
Cerium-141	8.8×10^{6}	8.8×10^{6}	8.8×10^{6}	3.5×10^{4}	3.5×10^{4}	3.5×10^{4}
Cerium-143	1.2×10^{7}	1.2×10^{7}	1.2×10^{7}	4.9×10^{4}	4.9×10^{4}	4.9×10^{4}
Cerium-144	1.5×10^{6}	1.5×10^{6}	1.5×10^{6}	6,200	6,200	6,200
Praseodymium-143	1.1×10^{7}	1.1×10^{7}	1.1×10^{7}	4.5×10^{4}	4.5×10^{4}	4.5×10^{4}
Neodymium-147	4.4×10^{6}	4.4×10^{6}	4.4×10^{6}	1.8×10^{4}	1.8×10^{4}	1.8×10^{4}
Neptunium-237	8.5×10^{-2}	3.2	5.2	3.4×10^{-4}	1.3×10^{-2}	2.1×10^{-2}
Neptunium-239	3.7×10^{6}	3.8×10^{6}	3.8×10^{6}	1.5×10^{4}	1.5×10^{4}	1.5×10^{4}
Plutonium-238	170	1.5×10^{4}	2.4×10^{4}	0.69	59	97
Plutonium-239	6.5	14	20	0.026	0.058	0.079
Plutonium-240	4.1	7.5	9.7	0.016	0.030	0.039
Plutonium-241	1,500	2,300	2,800	6.1	9.1	11
Americium-241	0.088	0.088	0.088	3.5×10^{-4}	3.5×10^{-4}	3.5×10^{-4}
Curium-242	15	25	32	0.059	0.10	0.13
Curium-244	1.3	1.3	1.3	0.0052	0.0052	0.0052

Source: LMIT 1998 and Schnitzler 1999.

The ATR core inventory and release fractions were obtained from the *ATR Upgraded Final Safety Analysis Report* which provides the end-of-cycle core inventory for several hundred isotopes (LMIT 1998). These isotopes were screened and reduced to those that contribute to human health effects.

I.1.1.1.3 Neptunium-237 Target-Handling Accident

The neptunium-237 target-handling accident scenario postulates the maximum amount of targets in the storage pool. A drop sufficient to damage the entire neptunium-237 target inventory is assumed. This accident is assumed to have a likelihood of occurrence of 0.001 per year.

For the purposes of this analysis, the following assumptions are made for the target-handling accident. The fuel-clad gap contains 10 percent of the fission product gases and iodine (NRC 1978). One-hundred percent of the noble gases and tritium gas in the fuel-clad gap is released to the environment through the reactor building exhaust system. This results in an overall release fraction of 0.1 for the noble gases and tritium. Twenty-five percent of the iodine in the fuel-clad gap is released from the fuel assembly, and 90 percent of the released iodine is absorbed in the reactor pool. The remaining iodine is released to the environment through the reactor building exhaust system. The exhaust system charcoal filter is assumed to remove 99 percent of the iodine (NRC 1978). This results in an overall release fraction of 2.5×10^{-5} ($0.1 \times 0.25 \times 0.1 \times 0.01 = 2.5\times10^{-5}$) for the iodine. These assumptions result in the source terms shown in **Table I–5** for the 3- and 5-kilogram-per-year (6.6- and 11-pounds-per-year) production rates.

Table I–5 ATR Neptunium-237 Target-Handling Accident Source Terms

Isotope	Environmental Release (curies) Versus Plutonium-238 Production Rate	
	3 kilograms per year	5 kilograms per year
Hydrogen-3	0.207	0.344
Krypton-85	1.73	2.89
Krypton-85m	454	757
Krypton-87	757	1,260
Krypton-88	1,060	1,770
Iodine-131	0.698	1.16
Iodine-132	1.04	1.74
Iodine-133	1.39	2.07
Iodine-134	1.48	2.47
Iodine-135	1.30	2.17
Xenon-133	5,250	8,760
Xenon-135	659	1,100

Source: Calculated results.

I.1.1.1.4 Meteorological Data

Meteorological characteristics of the ATR site are described by 1 year of hourly windspeed, atmospheric stability, and rainfall recorded at INEEL.

I.1.1.1.5 Population Data

The population distribution surrounding ATR is based on the 1990 Census of Population and Housing (DOC 1992). State and county population estimates were examined to extrapolate the 1990 data to the year 2020.

I.1.1.1.6 Evacuation Information

In the event of an accident, DOE would implement site emergency plans and procedures that include restricting site access, patrolling onsite roads, and relocating members of the public. These actions would significantly reduce the consequences to onsite individuals. DOE sites also coordinate with offsite agencies in the event of an emergency. However, no relocation or evacuation of the offsite population was assumed for ATR accident analyses. It was assumed that interdiction and condemnation of contaminated crops and foods were implemented based on U.S. Environmental Protection Agency (EPA) Protective Action Guides.

I.1.1.2 High Flux Isotope Reactor Accident Analyses

HFIR would generate 2 kilograms (4.4 pounds) per year of plutonium-238 in support of Alternative 2, Options 7 through 9. On average, HFIR has 11 refueling outages per year. HFIR accident analyses assumed that one-eleventh of the annual plutonium-238 production would be harvested at each refueling outage and an equal amount of plutonium-238 would remain in the core in targets that were not ready to be harvested. The accident analyses postulated that the plutonium-238 at risk in targets during HFIR accidents is 364 grams (0.80 pound).

I.1.1.2.1 Design-Basis Accident

The *HFIR Safety Analysis Report* (LMER 1998) detailed numerous small-break loss-of-coolant accidents. The worst-case scenario is a 2-inch (5-centimeter) break at the reactor vessel. The primary flow drops sharply in the first few seconds after the break before recovering at about one-fourth of its normal value. However, the primary coolant system fluid remains subcooled throughout the event, and there is considerable margin to critical heat flux. This is the maximum tolerable break short of fuel damage and nonrecoverable flow. It also represents the largest break size that still has a frequency of occurrence greater than 1×10^{-4} per year.

No reactor fuel or target rods fail as a result of the worst-case small-break loss-of-coolant accident.

I.1.1.2.2 Severe Reactor Accident

The large-break loss-of-coolant accident is the limiting severe reactor accident at HFIR. Two large-break loss-of-coolant accidents were evaluated in the *HFIR Safety Analysis Report* (LMER 1998). Both accidents involve breaks in the primary coolant system piping. The first is a double-ended guillotine break of the cold leg in the reactor pool, in which the reactor coolant is retained inside confinement. The second is a double-ended guillotine break of a primary coolant pump discharge line in a heat exchanger cell. The consequences of a large-break loss-of-coolant accident in the heat exchanger cell are bounded by those resulting from a large-break loss-of-coolant accident in the reactor pool. Therefore, the large-break loss-of-coolant accident in the reactor pool was chosen for analysis in this NI PEIS.

The large-break loss-of-coolant accident in the reactor pool assumes that 100 percent of the core melts. Equipment in service at the beginning of the accident is assumed to operate for the duration of the accident. This equipment includes the special building or confinement hot-exhaust system, which is designed to filter out airborne particulate activity from the HFIR building.

The *HFIR Safety Analysis Report* (LMER 1998) states that 100 percent of noble gases and 1 percent of iodines are released to the environment. The accident scenario presented in the facility safety analysis report assumes that the primary coolant piping breaks in the reactor pool. Therefore, even though the primary coolant piping inventory is lost, the core remains covered with water. Because of this assumption, only noble gases and iodine are assumed to be released to the environment. This differs from other reactors in the assumption that no other radioisotopes are released. For most reactors, a severe loss-of-coolant accident results in an uncovered core, leading to a fractional release of all isotopes.

The accident source term is presented in **Table I–6** for the two possible HFIR core configurations.

Table I–6 HFIR Large-Break Loss-of-Coolant Accident Source Term

Isotope	Core Inventory (curies) Versus Plutonium-238 Production Rate		Environmental Release (curies) Versus Plutonium-238 Production Rate	
	0 kilograms per year	2 kilograms per year	0 kilograms per year	2 kilograms per year
Krypton-85	800	810	800	810
Krypton-85m	8.8×10^{5}	8.8×10^{5}	8.8×10^{5}	8.8×10^{5}
Krypton-87	1.8×10^{6}	1.8×10^{6}	1.8×10^{6}	1.8×10^{6}
Krypton-88	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}	2.6×10^{6}
Rubidium-86	130	130	0	0
Strontium-89	9.5×10^{5}	9.6×10^{5}	0	0
Strontium-90	6,500	6,600	0	0
Strontium-91	4.1×10^{6}	4.1×10^{6}	0	0
Yttrium-90	5,600	5,600	0	0
Yttrium-91	1.0×10^{6}	1.0×10^{6}	0	0
Zirconium-95	1.0×10^{6}	1.0×10^{6}	0	0
Zirconium-97	4.1×10^{6}	4.1×10^{6}	0	0
Niobium-95	2.2×10^{5}	2.3×10^{5}	0	0
Molybdenum-99	4.2×10^{6}	4.3×10^{6}	0	0
Technetium-99m	3.9×10^{6}	3.9×10^{6}	0	0
Ruthenium-103	7.7×10^{5}	7.8×10^{5}	0	0
Ruthenium-105	7.2×10^{5}	7.4×10^{5}	0	0
Ruthenium-106	1.3×10^{4}	1.5×10^{4}	0	0
Rhodium-105	5.8×10^{5}	5.9×10^{5}	0	0
Antimony-127	1.1×10^{5}	1.1×10^{5}	0	0
Antimony-129	4.8×10^{5}	4.9×10^{5}	0	0
Tellurium-127	9.6×10^{4}	9.8×10^{4}	0	0
Tellurium-127m	1,700	1,800	0	0
Tellurium-129	4.5×10^{5}	4.5×10^{5}	0	0
Tellurium-129m	2.8×10^{4}	2.8×10^{4}	0	0
Tellurium-131m	2.6×10^{5}	2.6×10^{5}	0	0
Tellurium-132	3.0×10^{6}	3.0×10^{6}	0	0
Iodine-131	1.7×10^{6}	1.7×10^{6}	1.7×10^{4}	1.7×10^{4}
Iodine-132	3.0×10^{6}	3.0×10^{6}	3.0×10^{4}	3.0×10^{4}
Iodine-133	4.6×10^{6}	4.6×10^{6}	4.6×10^{4}	4.6×10^{4}
Iodine-134	5.4×10^{6}	5.4×10^{6}	5.4×10^{4}	5.4×10^{4}
Iodine-135	4.4×10^{6}	4.4×10^{6}	4.4×10^{4}	4.4×10^{4}
Xenon-133	4.6×10^{6}	4.6×10^{6}	9.2×10^{6}	9.2×10^{6}
Xenon-135	1.5×10^{5}	1.5×10^{5}	3.5×10^{6}	3.5×10^{6}
Cesium-134	440	500	0	0
Cesium-136	4,000	4,300	0	0
Cesium-137	6,600	6,700	0	0
Cerium-141	1.6×10^{6}	1.6×10^{6}	0	0
Cerium-143	4.1×10^{6}	4.2×10^{6}	0	0
Cerium-144	2.2×10^{5}	2.2×10^{5}	0	0
Barium-140	3.2×10^{6}	3.2×10^{6}	0	0
Lanthanum-140	3.1×10^{6}	3.1×10^{6}	0	0
Praseodymium-143	2.8×10^{6}	2.8×10^{6}	0	0
Neodymium-147	1.3×10^{6}	1.3×10^{6}	0	0
Neptunium-237	0	1.3	0	0
Neptunium-239	2.9×10^{5}	3.0×10^{5}	0	0

Isotope	Core Inventory (curies) Versus Plutonium-238 Production Rate		Environmental Release (curies) Versus Plutonium-238 Production Rate	
	0 kilograms per year	2 kilograms per year	0 kilograms per year	2 kilograms per year
Plutonium-238	0.32	6,200	0	0
Plutonium-239	0.38	3.7	0	0
Plutonium-240	0.055	1.5	0	0
Plutonium-241	1.1	310	0	0
Americium-241	2.4×10^{-5}	2.4×10^{-5}	0	0
Curium-242	4.6×10^{-4}	4.4	0	0
Curium-244	9.9×10^{-7}	9.9×10^{-7}	0	0

Source: Rothrock 1999; Schnitzler 1999; Wham 1999.

I.1.1.2.3 Neptunium-237 Target-Handling Accident

The neptunium-237 target-handling accident scenario postulates the maximum number of targets in the storage pool. A drop sufficient to damage the entire neptunium-237 target inventory is assumed. This accident is assumed to have a likelihood of occurrence of 0.001 per year. The accident assumptions are described in Section I.1.1.1.3. These assumptions result in the source terms, shown in **Table I–7**, for a 2-kilograms-per-year (4.4-pounds-per-year) production rate.

Table I–7 HFIR Neptunium-237 Target-Handling Accident Source Term

Isotope	Environmental Release[a] (curies)
Hydrogen-3	0.0877
Krypton-85	0.735
Krypton-85m	193
Krypton-87	321
Krypton-88	451
Iodine-131	0.295
Iodine-132	0.443
Iodine-133	0.593
Iodine-134	0.628
Iodine-135	0.553
Xenon-133	2,230
Xenon-135	280

a. Based on a 2-kilogram-per-year plutonium-238 production rate.
Source: Calculated results.

I.1.1.2.4 Meteorological Data

Meteorological characteristics of the HFIR site are described by 1 year of hourly windspeed, atmospheric stability, and rainfall recorded at ORNL.

I.1.1.2.5 Population Data

The population distribution surrounding HFIR is based on the 1990 census (DOC 1992). State and county population estimates were examined to extrapolate the 1990 data to the year 2020.

I.1.1.2.6 Evacuation Information

In the event of an accident, DOE would implement site emergency plans and procedures that include restricting site access, patrolling onsite roads, and relocating members of the public. These actions would significantly reduce the consequences to onsite individuals. DOE sites also coordinate with offsite agencies in the event of an emergency. However, no relocation or evacuation of the offsite population was assumed for HFIR accident analyses. It was assumed that interdiction and condemnation of contaminated crops and foods were implemented based on EPA Protective Action Guides.

I.1.1.3 Commercial Light Water Reactor

The CLWR would generate 5 kilograms (11 pounds) per year of plutonium-238 in support of Option 4, 5, or 6 of Alternative 2. On average, CLWR has one refueling outage every 18 months. The accident analysis assumes that 100 percent of the targets in the reactor core would be harvested at each refueling outage. The analysis postulates that the plutonium-238 at risk in targets during CLWR operation is 7.5 kilograms (16.5 pounds).

The analysis is based primarily on NUREG/CR-6295 (Davis 1997). NUREG/CR-6295 provides simplified design-basis and severe-accident source terms and generic site parameters based on the risk insights of NUREG-1150 (NRC 1990). These simplified source terms and generic parameters are used to analyze accidents for the current core for a baseline impact and with the proposed neptunium-237 targets to determine the incremental impact of plutonium-238 production. Core damage and containment failure frequencies were updated using more recent risk insights from the Individual Plant Examination (IPE) database (NRC 1997).

I.1.1.3.1 Core Inventories

After a review of NUREG/CR-6295, the 3,800 megawatts-thermal pressurized-water-reactor accident release fractions were chosen for this analysis. This reactor has the highest energy level and the consequences result in the highest risk of the reactors analyzed in NUREG/CR-6295. The MELCOR Accident Consequence Code System (MACCS2) documentation provides a typical end-of-cycle core inventory for a 3,412 megawatts-thermal pressurized-water-reactor. This power level was selected for the analysis because only 5 of the 73 currently operating pressurized water reactors have higher power levels, and 19 have a power level of 3,411 megawatts-thermal.

Table I–8 provides inventories for the current core configuration, the target inventory, and the core-containing targets. The end-of-cycle inventories provide bounding source terms which lead to maximum consequences. The calculation conservatively assumes that the targets are additions to the core and not replacements for some fuel rods. Replacing some burned fuel rods with targets would lower the core activity, perhaps below that without targets. As noted in the total activities line of the table, there is very little difference (approximately 0.16 percent) between the current core at 6.37×10^{9} curies versus 6.38×10^{9} curies for the current core plus the targets.

Table I–8 Core Inventories Based on a Target Maximum Core Loading of 7.5 Kilograms

Isotope	Core Inventory (curies)[a]	Target Inventory (curies)[b]	Core + Target Inventory (curies)
Cobalt-58	8.71×10^{5}	0.00	8.71×10^{5}
Cobalt-60	6.66×10^{5}	0.00	6.66×10^{5}
Krypton-85	6.69×10^{5}	152	6.69×10^{5}
Krypton-85m	3.13×10^{7}	3.98×10^{4}	3.14×10^{7}
Krypton-87	5.72×10^{7}	6.62×10^{4}	5.73×10^{7}
Krypton-88	7.74×10^{7}	9.30×10^{4}	7.75×10^{7}
Rubidium-86	5.10×10^{4}	57.2	5.11×10^{4}
Strontium-89	9.70×10^{7}	7.22×10^{4}	9.71×10^{7}
Strontium-90	5.24×10^{6}	953	5.24×10^{6}
Strontium-91	1.25×10^{8}	1.76×10^{5}	1.25×10^{8}
Strontium-92	1.30×10^{8}	2.13×10^{5}	1.30×10^{8}
Yttrium-90	5.62×10^{6}	960	5.62×10^{6}
Yttrium-91	1.18×10^{8}	9.90×10^{4}	1.18×10^{8}
Yttrium-92	1.30×10^{8}	2.15×10^{5}	1.31×10^{8}
Yttrium-93	1.47×10^{8}	2.79×10^{5}	1.48×10^{8}
Zirconium-95	1.49×10^{8}	1.85×10^{5}	1.50×10^{8}
Zirconium-97	1.56×10^{8}	3.83×10^{5}	1.56×10^{8}
Niobium-95	1.41×10^{8}	1.26×10^{5}	1.41×10^{8}
Molybdenum-99	1.65×10^{8}	4.22×10^{5}	1.65×10^{8}
Technetium-99m	1.42×10^{8}	3.76×10^{5}	1.43×10^{8}
Ruthenium-103	1.23×10^{8}	3.19×10^{5}	1.23×10^{8}
Ruthenium-105	7.98×10^{7}	3.88×10^{5}	8.02×10^{7}
Ruthenium-106	2.79×10^{7}	4.81×10^{4}	2.79×10^{7}
Rhodium-105	5.53×10^{7}	3.08×10^{5}	5.56×10^{7}
Antimony-127	7.53×10^{6}	3.33×10^{4}	7.57×10^{6}
Antimony-129	2.67×10^{7}	1.01×10^{5}	2.68×10^{7}
Tellurium-127	7.28×10^{6}	3.14×10^{4}	7.31×10^{6}
Tellurium-127m	9.63×10^{5}	1,820	9.65×10^{5}
Tellurium-129	2.50×10^{7}	9.68×10^{4}	2.51×10^{7}
Tellurium-129m	6.60×10^{6}	1.04×10^{4}	6.61×10^{6}
Tellurium-131m	1.26×10^{7}	4.47×10^{4}	1.27×10^{7}
Tellurium-132	1.26×10^{8}	3.53×10^{5}	1.26×10^{8}
Iodine-131	8.66×10^{7}	2.44×10^{5}	8.69×10^{7}
Iodine-132	1.28×10^{8}	3.65×10^{5}	1.28×10^{8}
Iodine-133	1.83×10^{8}	4.88×10^{5}	1.84×10^{8}
Iodine-134	2.01×10^{8}	5.18×10^{5}	2.02×10^{8}
Iodine-135	1.73×10^{8}	4.56×10^{5}	1.73×10^{8}
Xenon-133	1.83×10^{8}	4.60×10^{5}	1.84×10^{8}
Xenon-135	3.44×10^{7}	5.77×10^{4}	3.45×10^{7}
Cesium-134	1.17×10^{7}	1,190	1.17×10^{7}
Cesium-136	3.56×10^{6}	6,900	3.56×10^{6}
Cesium-137	6.53×10^{6}	2,810	6.54×10^{6}
Barium-139	1.70×10^{8}	4.06×10^{5}	1.70×10^{8}
Barium-140	1.68×10^{8}	3.38×10^{5}	1.68×10^{8}
Lanthanum-140	1.72×10^{8}	3.34×10^{5}	1.72×10^{8}
Lanthanum-141	1.57×10^{8}	3.85×10^{5}	1.58×10^{8}
Lanthanum-142	1.52×10^{8}	3.57×10^{5}	1.52×10^{8}
Cerium-141	1.53×10^{8}	2.63×10^{5}	1.53×10^{8}
Cerium-143	1.48×10^{8}	3.19×10^{5}	1.49×10^{8}
Cerium-144	9.20×10^{7}	5.35×10^{4}	9.21×10^{7}

Isotope	Core Inventory (curies)[a]	Target Inventory (curies)[b]	Core + Target Inventory (curies)
Praseodymium-143	1.46×10^{8}	2.67×10^{5}	1.46×10^{8}
Neodymium-147	6.52×10^{7}	1.28×10^{5}	6.53×10^{7}
Neptunium-239	1.75×10^{9}	1.26×10^{5}	1.75×10^{9}
Plutonium-238	9.90×10^{4}	1.27×10^{5}	2.26×10^{5}
Plutonium-239	2.23×10^{4}	69.1	2.24×10^{4}
Plutonium-240	2.82×10^{4}	29.5	2.82×10^{4}
Plutonium-241	4.74×10^{6}	6,400	4.75×10^{6}
Americium-241	3,130	0.00	3,130
Curium-242	1.20×10^{6}	91.5	1.20×10^{6}
Curium-244	7.02×10^{4}	0.00	7.02×10^{4}
Totals	6.37×10^{9}	1.02×10^{7}	6.38×10^{9}

a. Chanin et al. 1990; inventory converted from becquerels (Bq) to curies (Ci); 3.7×10^{10} Bq = 1 Ci.
b. Schnitzler 1999.

I.1.1.3.2 Meteorological Data

According to NUREG/CR-6295 (Davis 1997), the Sandia Siting Study evaluated data from 29 National Weather Service sites representing the nation's meteorological conditions. The 29 sites were compared to determine which site best represents the nation's meteorological conditions. It was determined that the site with the least deviation from the mean is the one at Omaha, Nebraska. Another comparison of the 29 sites indicated that the mean mixing height is 1.5 kilometers (0.93 mile). The mean meteorological data used in the NI PEIS analysis are a composite of the Omaha meteorological conditions and the mean mixing height.

I.1.1.3.3 Population Data

To be as generic as possible, the population around the plant was assumed to be uniformly distributed. The analysis was performed for a population density of 100 persons per square mile (38.6 persons per square kilometer) from 0 to 10 miles (representing the median population density for all pressurized water reactors) and 200 persons per square mile (77.2 persons per square kilometer) from 10 to 50 miles (representing an average population density beyond 10 miles). The exclusion area boundary was assumed to be 640 meters (0.4 mile) from the reactor.

I.1.1.3.4 Evacuation Information

Consistent with NUREG-1150, this analysis assumes that 99.5 percent of the population within the 16.1-kilometer (10-mile) emergency planning zone participates in an evacuation. It was also assumed that the 0.5 percent of the population that did not participate in the initial evacuation was relocated within 12 to 24 hours after plume passage, based on the measured concentrations of radioactivity in the surrounding area and the comparison of projected doses with EPA guidelines. Mean evacuation time and speed were based on the average of the five NUREG-1150 plants. This results in an evacuation delay time of 1.9 hours and an evacuation speed of 9.3 kilometers (5.8 miles) per hour.

I.1.1.3.5 Design-Basis Accident

Design-basis events are defined by the American Nuclear Society as Condition IV occurrences or limiting faults. Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of substantial radioactive material. These are the most serious events which must be designed against and represent limiting design cases.

A realistic design-basis large-break loss-of-coolant accident was chosen for evaluation because it is the limiting design-basis accident at pressurized water reactor plants. The large-break loss-of-coolant accident is defined as a break equivalent in size to a double-ended rupture of the largest pipe of the reactor coolant system. Following a postulated double-ended rupture of a reactor coolant pipe, the emergency core cooling system keeps cladding temperatures well below melting, ensuring that the core remains intact and in a coolable geometry. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the containment. Although no core melting would occur for the design-basis loss-of-coolant accident, a postulated gross release of fission products is evaluated in accordance with NRC accident analysis guidelines (AEC 1974). The only postulated mechanism for such a release would be a number of simultaneous and extended failures in the engineered safety feature systems, producing severe physical degradation of core geometry and partial melting of the fuel.

The realistic large-break loss-of-coolant accident release characteristics, obtained from NUREG/CR-6295, are described by the release height, timing, duration, and heat content of the plume; the fraction of each isotope group released; and the warning time (time when offsite officials are warned that an emergency response should be initiated.) **Tables I–9 and I–10** provide the release parameters for the realistic large-break loss-of-coolant accident.

Table I–9 Design-Basis Accident Release Characteristics

Accident Scenario	Scenario Frequency	Elevation of Release (m)	Energy of Release (W)	Warning Time (hr)	Time of Release (hr)	Duration of Release (hr)
Large-break loss-of-coolant accident[a]	4.65×10^{-5}	0	0.0	5.0	6.0	10.0
					16.0	

a. The accident is represented by two separate releases.
Key: hr, hour; m, meter; W, watts.
Source: Davis 1997.

Table I–10 Design-Basis Accident Release Fractions

Release Category	Release Fractions by Isotope								
	Kr, Xe	I	Cs, Rb	Sb, Te	Sr	Co, Mo, Rh, Ru, Tc	Am, Cm, La, Nb, Nd, Pr, Y, Zr	Ce, Np, Pu	Ba
Large-break loss-of-coolant accident[a]	2.5×10^{-3}	1.5×10^{-5}	1.2×10^{-8}	7.5×10^{-9}	2.5×10^{-9}	2.0×10^{-10}	3.0×10^{-10}	4.0×10^{-10}	2.5×10^{-9}
	2.5×10^{-3}	1.5×10^{-5}	1.2×10^{-8}	7.5×10^{-9}	2.5×10^{-9}	2.0×10^{-10}	3.0×10^{-10}	4.0×10^{-10}	2.5×10^{-9}

a. The accident is represented by two separate releases.
Key: Am, americium; Ba, barium; Ce, cerium; Cm, curium; Co, cobalt; Cs, cesium; I, iodine; Kr, krypton; La, lanthanum; Mo, molybdenum; Nb, niobium; Nd, neodymium; Np, neptunium; Pu, plutonium; Pr, praseodymium; Rb, rubidium; Rh, rhodium; Ru, ruthenium; Sb, antimony; Sr, strontium; Tc, technetium; Te, tellurium; Xe, xenon; Y, yttrium; Zr, zirconium.
Source: Davis 1997.

NUREG/CR-6295 (Davis 1997) provides frequencies for each accident category. However, these frequencies are based solely on the NUREG-1150 (NRC 1990) plant data. To apply more recent accident frequencies, data from commercial pressurized water reactor IPEs were reviewed. For each of the accident categories (loss-of-coolant accident, early containment failure, late containment failure, and containment bypass) the failure probability medians were calculated. These data represent significant additional risk studies more recent than NUREG-1150.

The frequency of occurrence for the design-basis large-break loss-of-coolant accident is 4.65×10^{-5} per year. This frequency is based on internal initiators (i.e., plant upsets) and does not include external initiators (e.g., earthquakes). External initiators were not included because the frequencies depend solely on site location.

I.1.1.3.6 Beyond-Design-Basis Events

Beyond-design-basis accidents (severe reactor accidents) are less likely to occur than reactor design-basis accidents. In reactor design-basis accidents, the mitigating systems are assumed to be available. In severe reactor accidents, even though the initiating event could be a design-basis event (e.g., large-break loss-of-coolant accident), additional failures of mitigating systems would cause some degree of physical deterioration of the fuel in the reactor core and a possible breach of the containment structure leading to the direct release of radioactive materials to the environment.

In NUREG/CR-6295, representative source terms were developed which represent the full spectrum of severe accidents. A small set of source terms was developed by considering release categories which account for a spectrum of possible times and modes of containment failure. For each containment failure mode the source terms were selected based on the dominant accident progression characteristics leading to the containment failure. The magnitudes of releases for each release category were obtained by using the mean values of the probability distributions of source term parameters used in NUREG-1150.

In NUREG/CR-6295, a total of four release categories was selected to represent the spectrum of containment failure modes of the 3,800 megawatts-thermal pressurized water reactor: a containment bypass event, an early containment failure coincident with reactor core vessel breach, a late containment failure, and a no-containment-failure event. The no-containment-failure event is initiated by a large-break loss-of-coolant accident and was used to represent a realistic design-basis large-break loss-of-coolant accident. The containment bypass and failure scenarios are considered beyond-design-basis events and are evaluated in this section.

Containment Bypass. A containment bypass involves failure of the pressure boundary between the high-pressure reactor coolant and low-pressure auxiliary system. For pressurized water reactors, steam generator tube rupture, either as an initiating event or as a result of severe accident conditions, will lead to containment bypass. In these scenarios, if core damage occurs, a direct path to the environment can exist.

Early Containment Failure. This accident is defined as the failure of containment prior to, or very soon (within a few hours) after, breach of the reactor vessel. A variety of mechanisms (e.g., direct contact of core debris with the containment, rapid pressure and temperature loads, hydrogen combustion, fuel-coolant interactions) can cause structural failure of the containment. Failure to isolate the containment and early containment venting after core damage are also classified as early containment failures.

Late Containment Failure. A late containment failure involves structural failure of the containment several hours after breach of the reactor vessel. A variety of mechanisms (e.g., gradual pressure and temperature increase, hydrogen combustion, basemat melt-through by core debris) can cause late containment failure. Venting the containment late in the accident is also classified as a late containment failure.

The release characteristics for each accident, obtained from NUREG/CR-6295, are described by the release height, timing, duration, and heat content of the plume, the fraction of each isotope group released, and the warning time (time when offsite officials are warned that an emergency response should be initiated). **Tables I–11 and I–12** provide the release parameters for the beyond-design-basis accidents.

Table I–11 Beyond-Design-Basis Accident Release Characteristics

Accident Scenario	Scenario Frequency	Elevation of Release (m)	Energy of Release (W)	Warning Time	Time of Release (hr)	Duration of Release
Containment bypass[a]	1.53×10^{-6}	10	5.5×10^{6}	20 min	1.0	30 min
			9.9×10^{5}		1.5	2 hr
Early containment failure[a]	7.92×10^{-8}	10	8.6×10^{5}	5.0 hr	6.0	10 min
			1.5×10^{6}		6.167	2 hr
Late containment failure	1.07×10^{-5}	10	1.9×10^{5}	5.0 hr	12.0	3 hr

a. The accident is represented by two separate releases.
Key: hr, hour; m, meters; min, minute; W, watts.
Source: Davis 1997.

Table I–12 Beyond-Design-Basis Accident Release Fractions

Accident Scenario	Release Fractions by Isotope								
	Kr, Xe	I	Cs, Rb	Sb, Te	Sr	Co, Mo, Rh, Ru, Tc	Am, Cm, La, Nb, Nd, Pr, Y, Zr	Ce, Np, Pu	Ba
Containment bypass[a]	1.0	7.5×10^{-2}	6.0×10^{-2}	2.0×10^{-2}	5.0×10^{-3}	1.0×10^{-3}	3.0×10^{-4}	1.0×10^{-3}	5.0×10^{-3}
	0.0	4.0×10^{-2}	6.0×10^{-2}	5.0×10^{-2}	2.0×10^{-2}	6.0×10^{-4}	3.0×10^{-3}	3.0×10^{-3}	2.0×10^{-2}
Early containment failure[a]	1.0	2.5×10^{-1}	1.8×10^{-1}	8.0×10^{-2}	2.0×10^{-2}	5.0×10^{-3}	1.0×10^{-3}	5.0×10^{-3}	2.0×10^{-2}
	0.0	2.0×10^{-2}	3.0×10^{-2}	2.0×10^{-2}	1.0×10^{-2}	2.0×10^{-4}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-2}
Late containment failure	1.0	3.0×10^{-2}	6.0×10^{-6}	7.0×10^{-6}	1.0×10^{-6}	2.0×10^{-8}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-6}

a. The accident is represented by two separate releases.
Key: Am, americium; Ba, barium; Ce, cerium; Cm, curium; Co, cobalt; Cs, cesium; I, iodine; Kr, krypton; La, lanthanum; Mo, molybdenum; Nb, niobium; Nd, neodymium; Np, neptunium; Pu, plutonium; Pr, praseodymium; Rb, rubidium; Rh, rhodium; Ru, ruthenium; Sb, antimony; Sr, strontium; Tc, technetium; Te, tellurium; Xe, xenon; Y, yttrium; Zr, zirconium.
Source: Davis 1997.

As in the design-basis-accident analysis, the frequency of occurrence is based on internal initiators and does not include external initiators.

I.1.1.4 Fast Flux Test Facility (FFTF)

A spectrum of postulated accidents was evaluated for three separate FFTF conditions: operation, standby, and deactivation. Conservative assumptions were made on core configuration and isotopic inventory in order to provide conservative estimates of impacts.

I.1.1.4.1 FFTF Operation

For operation, the FFTF core would be modified to include an array of target assemblies and Rapid Radioisotope Retrieval systems to produce cobalt-60, a number of long- and short-lived isotopes for medical applications, and 5 kilograms (11 pounds) per year of plutonium-238 for space power applications. In addition, space is to be provided for research and development test articles such as Accelerator Transmutation of Waste test assemblies.

It is expected that the characteristics of the new mission core will be similar to previous cores, and that the existing facility safety analysis report analyses will be comparable to the new core accidents. A wide range of postulated reactor accidents was analyzed in the existing *FFTF Final Safety Analysis Report* (Dautel 2000).

These include design-basis and beyond-design-basis accidents. A spectrum of postulated accidents was evaluated to provide bounding scenarios for determining potential environmental and health impacts of the new missions. The accident scenarios were selected from the existing *FFTF Final Safety Analysis Report* and represent design-basis and beyond-design-basis events, including reactor, target-handling, and fuel storage accidents. Source terms and frequencies were selected to provide conservative estimates of the potential impacts.

The accident analysis included a review of external events (e.g., nearby explosions, fires), internal events (e.g., equipment failures, human errors), natural phenomena (e.g., floods, earthquakes), as well as sabotage and terrorist acts. A recent external event of concern is the threat of wildfires. Several features of FFTF make it well equipped to deal with an event like a large range fire. First, the layout and construction of the facility make it very unlikely that an external fire would spread to the plant structures (e.g., there is a large clear gravel and asphalt buffer zone, and much of the facility is constructed of fire-resistant materials). Furthermore, most of the critical plant systems, including the reactor and its heat transport system, are housed inside of the steel and concrete containment building, which is completely closed during reactor operation. As appropriate, the balance of the facility is protected by automatic fire detection/suppression systems. Although FFTF has several sources of both offsite and onsite electrical power, another significant safety factor is that, except for a few batteries, FFTF requires no electricity to accomplish any required safety function (i.e., reactor shutdown, isolation of the containment building, and emergency core cooling). Finally, the FFTF control building includes systems such as a filtered air supply for ensuring habitability during a variety of offnormal conditions, and emergency respirators are available to the operators. A wildfire-initiated accident would be bounded by the accidents evaluated and therefore not considered further.

Two large range fires at Hanford burned very close to FFTF. In 1984, a very large fire occurred while the reactor was in operation at 100 percent power. The plant continued to operate normally and safely throughout this event, although a reduction in power was initiated as a precautionary measure. The second fire occurred in June 2000 while the reactor was in standby. In neither case, did the range fire cause any damage or operational difficulties at FFTF. Both fires reached the gravel and asphalt buffer zone around FFTF, but never posed any significant threat to plant structures. Precautionary measures were taken by essential plant personnel to perform continuous monitoring and to reduce or eliminate the intake of smoke passing over the facility.

The reactor power will be 100 megawatts, which is one-fourth of the design power, for most of the mission operation. However, periodic increases in power level between 100 and 400 megawatts may be required to support civilian nuclear energy research and development activities. The accident analyses provided are based on the FFTF design power level of 400 megawatts and will provide conservative estimates of operation at 400 megawatts-thermal and lower power levels.

CORE INVENTORIES

Mixed Oxide Driver Fuel

The current FFTF fuel contains mixed oxide driver fuel assemblies. The plutonium fuel enrichment is assumed to be the same as during previous reactor operations and as currently authorized by the facility safety analysis report. A total of 76 driver fuel assemblies were assumed in the facility safety analysis report. Although it is expected that some of the driver fuel positions will be taken up by test articles and isotope production targets, the same number of driver fuel assemblies are to be assumed for conservatism for purposes of this analysis. A total of six fueled test articles were included in the assumed core loading for this analysis, but were treated as part of the complement of 76 driver fuel assemblies.

An ORIGEN2 (Wootan 1999) calculation for a reference driver fuel assembly was used to generate the radioisotope inventories used in the accident analyses. Evaluation focused on a typical end-of-irradiation inner-row driver fuel assembly with a plutonium enrichment of about 22 percent—specifically, assembly 16439 irradiated to 445.8 effective full-power days through cycle four in core location 1201. Previous studies have determined that 60 isotopes are important for offsite impact analysis. These 60 isotopes are provided in NUREG/CR-4691, *MELCOR Accident Consequence Code System*, Volume 1, Table B.4–2 (Chanin et al. 1990). The resulting driver fuel inventory is shown in **Table I–13**.

Highly Enriched Uranium Driver Fuel

A future core loading may require use of highly enriched uranium. The highly enriched uranium fuel would be in an oxide form. Radioisotope inventories were calculated for a highly enriched uranium fuel assembly that is directly comparable to the reference mixed oxide fuel assembly. To generate comparable values for a highly enriched uranium fueled core, a highly enriched uranium fuel assembly with a uranium-235 enrichment of 25 percent was used to replace the reference mixed oxide assembly in the ORIGEN2 calculation of radioisotope inventories. This enrichment provides about 25 percent more uranium-235 in the highly enriched uranium assembly than plutonium-239 in the mixed oxide assembly, so that the highly enriched uranium assembly would have comparable power and burnup at a lower flux level than the reference mixed oxide assembly. This enrichment is lower than the enrichments expected in a full highly enriched uranium core (likely in the range of 35 percent), but the dose rates for this assembly should bound the higher enrichments, since the fission products would be nearly identical and the plutonium contribution would be less with higher enrichments. The resulting highly enriched uranium driver fuel inventory is shown in **Table I–14**.

Although accidents were evaluated for both the mixed oxide and highly enriched uranium core configurations, it is important to point out that the radiological consequences of the mixed oxide fueled core assumed in this analysis will bound those of the highly enriched uranium core.

Table I–13 FFTF Core Inventory with Mixed Oxide Driver Fuel

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	Core Activity (Ci)
	Driver Activity (Ci)	76 Drivers (6 ATWs)[a] Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[b] (Ci)	12 Ac-227[c] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	
Hydrogen-3	57.56	4,370	0.00241	8.82	–	–	4,380
Cobalt-60	0.000	0.00	0.00	0.00	–	1.48×10^{7}	1.48×10^{7}
Krypton-85	517.6	3.93×10^{4}	0.0202	73.9	–	–	3.94×10^{4}
Krypton-85m	2.850×10^{4}	2.17×10^{6}	5.30	1.94×10^{4}	–	–	2.19×10^{6}
Krypton-87	4.716×10^{4}	3.58×10^{6}	8.83	3.23×10^{4}	–	–	3.62×10^{6}
Krypton-88	6.567×10^{4}	4.99×10^{6}	12.4	4.54×10^{4}	–	–	5.04×10^{6}
Rubidium-86	907.6	6.90×10^{4}	0.00762	27.9	–	–	6.90×10^{4}
Strontium-89	7.598×10^{4}	5.77×10^{6}	9.63	3.52×10^{4}	–	–	5.81×10^{6}
Strontium-90	3,181	2.42×10^{5}	0.127	465	–	–	2.42×10^{5}
Strontium-91	1.205×10^{5}	9.16×10^{6}	23.4	8.56×10^{4}	–	–	9.24×10^{6}
Strontium-92	1.418×10^{5}	1.08×10^{7}	28.4	1.04×10^{5}	–	–	1.09×10^{7}
Yttrium-90	3,561	2.71×10^{5}	0.128	468	–	–	2.71×10^{5}
Yttrium-91	9.984×10^{4}	7.59×10^{6}	13.2	4.83×10^{4}	–	–	7.64×10^{6}
Yttrium-92	1.434×10^{5}	1.09×10^{7}	28.7	1.05×10^{5}	–	–	1.10×10^{7}
Yttrium-93	1.785×10^{5}	1.36×10^{7}	37.2	1.36×10^{5}	–	–	1.37×10^{7}
Zirconium-95	1.776×10^{5}	1.35×10^{7}	24.7	9.04×10^{4}	–	–	1.36×10^{7}
Zirconium-97	2.357×10^{5}	1.79×10^{7}	51.0	1.87×10^{5}	–	–	1.81×10^{7}
Niobium-95	1.492×10^{5}	1.13×10^{7}	16.8	6.15×10^{4}	–	–	1.14×10^{7}
Molybdenum-99	2.690×10^{5}	2.04×10^{7}	56.3	2.06×10^{5}	–	–	2.07×10^{7}
Technetium-99m	2.355×10^{5}	1.79×10^{7}	50.1	1.83×10^{5}	–	–	1.81×10^{7}
Ruthenium-103	2.718×10^{5}	2.07×10^{7}	42.5	1.56×10^{5}	–	–	2.08×10^{7}
Ruthenium-105	2.261×10^{5}	1.72×10^{7}	51.7	1.89×10^{5}	–	–	1.74×10^{7}
Ruthenium-106	9.408×10^{4}	7.15×10^{6}	6.41	2.35×10^{4}	–	–	7.17×10^{6}
Rhodium-105	2.246×10^{5}	1.71×10^{7}	41.1	1.50×10^{5}	–	–	1.72×10^{7}
Antimony-127	2.579×10^{4}	1.96×10^{6}	4.44	1.63×10^{4}	–	–	1.98×10^{6}
Antimony-129	6.280×10^{4}	4.77×10^{6}	13.5	4.94×10^{4}	–	–	4.82×10^{6}
Tellurium-127m	2,535	1.93×10^{5}	0.243	889	–	–	1.94×10^{5}
Tellurium-127	2.471×10^{4}	1.88×10^{6}	4.18	1.53×10^{4}	–	–	1.89×10^{6}
Tellurium-129	6.211×10^{4}	4.72×10^{6}	12.9	4.72×10^{4}	–	–	4.77×10^{6}
Tellurium-129m	8,626	6.56×10^{5}	1.39	5,090	–	–	6.61×10^{5}
Tellurium-131	1.546×10^{5}	1.17×10^{7}	30.9	1.13×10^{5}	–	–	1.19×10^{7}
Tellurium-131m	2.684×10^{4}	2.04×10^{6}	5.96	2.18×10^{4}	–	–	2.06×10^{6}
Tellurium-132	2.314×10^{5}	1.76×10^{7}	47.1	1.72×10^{5}	–	–	1.78×10^{7}
Iodine-125	0.000	0.00	0.00	0.00	2,530	–	2,530
Iodine-131	1.759×10^{5}	1.34×10^{7}	32.5	1.19×10^{5}	–	–	1.35×10^{7}
Iodine-132	2.367×10^{5}	1.80×10^{7}	48.7	1.78×10^{5}	–	–	1.82×10^{7}
Iodine-133	2.996×10^{5}	2.28×10^{7}	65.0	2.38×10^{5}	–	–	2.30×10^{7}
Iodine-134	3.173×10^{5}	2.41×10^{7}	69.0	2.53×10^{5}	–	–	2.44×10^{7}
Iodine-135	2.883×10^{5}	2.19×10^{7}	60.8	2.23×10^{5}	–	–	2.21×10^{7}
Xenon-133	3.051×10^{5}	2.32×10^{7}	61.3	2.24×10^{5}	–	–	2.34×10^{7}
Xenon-135	3.254×10^{5}	2.47×10^{7}	7.69	2.81×10^{4}	–	–	2.48×10^{7}
Cesium-134	4,980	3.78×10^{5}	0.159	582	–	–	3.79×10^{5}

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	Core Activity (Ci)
	Driver Activity (Ci)	76 Drivers (6 ATWs)[a] Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[b] (Ci)	12 Ac-227[c] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	
Cesium-136	9,451	7.18×10^{5}	0.920	3,370	–	–	7.22×10^{5}
Cesium-137	8,361	6.35×10^{5}	0.375	1,370	–	–	6.37×10^{5}
Barium-139	2.594×10^{5}	1.97×10^{7}	54.1	1.98×10^{5}	–	–	1.99×10^{7}
Barium-140	2.397×10^{5}	1.82×10^{7}	45.1	1.65×10^{5}	–	–	1.84×10^{7}
Lanthanum-140	2.421×10^{5}	1.84×10^{7}	44.5	1.63×10^{5}	–	–	1.86×10^{7}
Lanthanum-141	2.460×10^{5}	1.87×10^{7}	51.3	1.88×10^{5}	–	–	1.89×10^{7}
Lanthanum-142	2.179×10^{5}	1.66×10^{7}	47.6	1.74×10^{5}	–	–	1.67×10^{7}
Cerium-141	2.294×10^{5}	1.74×10^{7}	35.0	1.28×10^{5}	–	–	1.76×10^{7}
Cerium-143	1.998×10^{5}	1.52×10^{7}	42.5	1.56×10^{5}	–	–	1.53×10^{7}
Cerium-144	9.360×10^{4}	7.11×10^{6}	7.13	2.61×10^{4}	–	–	7.14×10^{6}
Praseodymium-143	1.966×10^{5}	1.49×10^{7}	35.6	1.30×10^{5}	–	–	1.51×10^{7}
Neodymium-147	9.847×10^{4}	7.48×10^{6}	17.1	6.26×10^{4}	–	–	7.55×10^{6}
Rhenium-186	0.000	0.00	0.00	0.00	3.05×10^{4}	–	3.05×10^{4}
Radium-223	2.644×10^{-9}	2.01×10^{-7}	0.00	0.00	235	–	235
Radium-224	1.245×10^{-4}	0.00946	0.00	0.00	466	–	466
Radium-226	0.000	0.00	0.00	0.00	172	–	172
Actinium-227	1.780×10^{-9}	1.35×10^{-7}	0.00	0.00	408	–	408
Thorium-227	2.714×10^{-9}	2.06×10^{-7}	0.00	0.00	298	–	298
Thorium-228	1.239×10^{-4}	0.00942	0.00	0.00	505	–	505
Neptunium-237	0.009117	0.693	3.60×10^{-3}	13.2	–	–	13.9
Neptunium-239	2.723×10^{6}	2.07×10^{8}	16.8	6.15×10^{4}	–	–	2.07×10^{8}
Plutonium-238	123.6	9,390	17	6.19×10^{4}	–	–	7.12×10^{4}
Plutonium-239	320.1	2.43×10^{4}	9.21×10^{-3}	33.7	–	–	2.44×10^{4}
Plutonium-240	259.3	1.97×10^{4}	3.93×10^{-3}	14.4	–	–	1.97×10^{4}
Plutonium-241	1.213×10^{4}	9.22×10^{5}	0.853	3,120	–	–	9.25×10^{5}
Americium-241	141.1	1.07×10^{4}	0.00	0.00	–	–	1.07×10^{4}
Curium-242	9,829	7.47×10^{5}	0.0122	44.7	–	–	7.47×10^{5}
Curium-244	8.305	631	0.00	0.00	–	–	631

a. Six Accelerator Transmutation of Waste test assemblies included as driver fuel assemblies.
b. Based on a 5-kilogram-per-year plutonium-238 production rate.
c. For the actinium-227 target, over 99.9 percent of the consequences are attributable to six isotopes (actinium-227; radium-223, 224, 226; thorium-227, 228). Therefore, the other actinium-227 target byproducts are not included.

Key: Ac-227, actinium-227; ATW, Accelerator Transmutation of Waste; Ci, curies; Co-60, cobalt-60; I-125, iodine-125; Pu-238, plutonium-238; Re-186, rhenium-186.

Source: BWHC 1999; Schnitzler 1999; Wootan 1999.

Table I–14 FFTF Core Inventory with Highly Enriched Uranium Driver Fuel

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	ATWs	
	HEU Driver Activity (Ci)	70 HEU Drivers Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[a] (Ci)	12 Ac-227[b] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	6 MOX Driver Assemblies (6 ATWs)[c] Activity (Ci)	Core Activity (Ci)
Hydrogen-3	66.04	4,620	0.00241	8.82	–	–	345	4,980
Cobalt-60	0.000	0.00	0.00	0.00	–	1.48×10^{7}	0.00	1.48×10^{7}
Krypton-85	962.1	6.73×10^{4}	0.0202	73.9	–	–	3,110	7.05×10^{4}
Krypton-85m	5.236×10^{4}	3.67×10^{6}	5.30	1.94×10^{4}	–	–	1.71×10^{5}	3.86×10^{6}
Krypton-87	9.952×10^{4}	6.97×10^{6}	8.83	3.23×10^{4}	–	–	2.83×10^{5}	7.28×10^{6}
Krypton-88	1.433×10^{5}	1.00×10^{7}	12.4	4.54×10^{4}	–	–	3.94×10^{5}	1.05×10^{7}
Rubidium-86	1.571×10^{3}	1.10×10^{5}	0.00762	27.9	–	–	5,450	1.15×10^{5}
Strontium-89	1.586×10^{5}	1.11×10^{7}	9.63	3.52×10^{4}	–	–	4.56×10^{5}	1.16×10^{7}
Strontium-90	7.107×10^{3}	4.97×10^{5}	0.127	465	–	–	1.91×10^{4}	5.17×10^{5}
Strontium-91	2.279×10^{5}	1.60×10^{7}	23.4	8.56×10^{4}	–	–	7.23×10^{5}	1.68×10^{7}
Strontium-92	2.385×10^{5}	1.67×10^{7}	28.4	1.04×10^{5}	–	–	8.51×10^{5}	1.76×10^{7}
Yttrium-90	7.885×10^{3}	5.52×10^{5}	0.128	468	–	–	2.14×10^{4}	5.74×10^{5}
Yttrium-91	1.922×10^{5}	1.35×10^{7}	13.2	4.83×10^{4}	–	–	5.99×10^{5}	1.41×10^{7}
Yttrium-92	2.397×10^{5}	1.68×10^{7}	28.7	1.05×10^{5}	–	–	8.60×10^{5}	1.77×10^{7}
Yttrium-93	2.655×10^{5}	1.86×10^{7}	37.2	1.36×10^{5}	–	–	1.07×10^{6}	1.98×10^{7}
Zirconium-95	2.274×10^{5}	1.59×10^{7}	24.7	9.04×10^{4}	–	–	1.07×10^{6}	1.71×10^{7}
Zirconium-97	2.636×10^{5}	1.85×10^{7}	51.0	1.87×10^{5}	–	–	1.41×10^{6}	2.01×10^{7}
Niobium-95	1.921×10^{5}	1.34×10^{7}	16.8	6.15×10^{4}	–	–	8.95×10^{5}	1.44×10^{7}
Molybdenum-99	2.693×10^{5}	1.89×10^{7}	56.3	2.06×10^{5}	–	–	1.61×10^{6}	2.07×10^{7}
Technetium-99m	2.358×10^{5}	1.65×10^{7}	50.1	1.83×10^{5}	–	–	1.41×10^{6}	1.81×10^{7}
Ruthenium-103	1.727×10^{5}	1.21×10^{7}	42.5	1.56×10^{5}	–	–	1.63×10^{6}	1.39×10^{7}
Ruthenium-105	9.789×10^{4}	6.85×10^{6}	51.7	1.89×10^{5}	–	–	1.36×10^{6}	8.40×10^{6}
Ruthenium-106	2.729×10^{4}	1.91×10^{6}	6.41	2.35×10^{4}	–	–	5.82×10^{5}	2.52×10^{6}
Rhodium-105	9.844×10^{4}	6.89×10^{6}	41.1	1.50×10^{5}	–	–	1.35×10^{6}	8.39×10^{6}
Antimony-127	2.172×10^{4}	1.52×10^{6}	4.44	1.63×10^{4}	–	–	1.55×10^{5}	1.69×10^{6}
Antimony-129	5.288×10^{4}	3.70×10^{6}	13.5	4.94×10^{4}	–	–	3.77×10^{5}	4.13×10^{6}
Tellurium-127	2.082×10^{4}	1.46×10^{6}	4.18	1.53×10^{4}	–	–	1.48×10^{5}	1.62×10^{6}

Table I–14 FFTF Core Inventory with Highly Enriched Uranium Driver Fuel (Continued)

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	ATWs	
	HEU Driver Activity (Ci)	70 HEU Drivers Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[a] (Ci)	12 Ac-227[b] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	6 MOX Driver Assemblies (6 ATWs)[c] Activity (Ci)	Core Activity (Ci)
Tellurium-127m	2.134×10^{3}	1.49×10^{5}	0.243	4.72×10^{4}	–	–	3.73×10^{5}	4.18×10^{6}
Tellurium-129	5.366×10^{4}	3.76×10^{6}	12.9	4.72×10^{4}	–	–	3.73×10^{5}	4.18×10^{6}
Tellurium-129m	7,338	5.14×10^{5}	1.39	5,090	–	–	5.18×10^{4}	5.71×10^{5}
Tellurium-131	1.413×10^{5}	9.89×10^{6}	30.9	1.13×10^{5}	–	–	9.28×10^{5}	1.09×10^{7}
Tellurium-131m	2.028×10^{4}	1.42×10^{6}	5.96	2.18×10^{4}	–	–	1.61×10^{5}	1.60×10^{6}
Tellurium-132	2.174×10^{5}	1.52×10^{7}	47.1	1.72×10^{5}	–	–	1.39×10^{6}	1.68×10^{7}
Iodine-125	0.000	0.00	0.00	0.00	2,530	–	0.00	2,530
Iodine-131	1.574×10^{5}	1.10×10^{7}	32.5	1.19×10^{5}	–	–	1.06×10^{6}	1.22×10^{7}
Iodine-132	2.209×10^{5}	1.55×10^{7}	48.7	1.78×10^{5}	–	–	1.42×10^{6}	1.71×10^{7}
Iodine-133	2.996×10^{5}	2.10×10^{7}	65.0	2.38×10^{5}	–	–	1.80×10^{6}	2.30×10^{7}
Iodine-134	3.412×10^{5}	2.39×10^{7}	69.0	2.53×10^{5}	–	–	1.90×10^{6}	2.60×10^{7}
Iodine-135	2.763×10^{5}	1.93×10^{7}	60.8	2.23×10^{5}	–	–	1.73×10^{6}	2.13×10^{7}
Xenon-133	3.034×10^{5}	2.12×10^{7}	61.3	2.24×10^{5}	–	–	1.83×10^{6}	2.33×10^{7}
Xenon-135	2.995×10^{5}	2.10×10^{7}	7.69	2.81×10^{4}	–	–	1.95×10^{6}	2.29×10^{7}
Cesium-134	4,676	3.27×10^{5}	0.159	582	–	–	2.99×10^{4}	3.58×10^{5}
Cesium-136	5,314	3.72×10^{5}	0.920	3,370	–	–	5.67×10^{4}	4.32×10^{5}
Cesium-137	8,319	5.82×10^{5}	0.375	1,370	–	–	5.02×10^{4}	6.34×10^{5}
Barium-139	2.833×10^{5}	1.98×10^{7}	54.1	1.98×10^{5}	–	–	1.56×10^{6}	2.16×10^{7}
Barium-140	2.674×10^{5}	1.87×10^{7}	45.1	1.65×10^{5}	–	–	1.44×10^{6}	2.03×10^{7}
Lanthanum-140	2.698×10^{5}	1.89×10^{7}	44.5	1.63×10^{5}	–	–	1.45×10^{6}	2.05×10^{7}
Lanthanum-141	2.658×10^{5}	1.86×10^{7}	51.3	1.88×10^{5}	–	–	1.48×10^{6}	2.03×10^{7}
Lanthanum-142	2.461×10^{5}	1.72×10^{7}	47.6	1.74×10^{5}	–	–	1.31×10^{6}	1.87×10^{7}
Cerium-141	2.492×10^{5}	1.74×10^{7}	35.0	1.28×10^{5}	–	–	1.38×10^{6}	1.89×10^{7}
Cerium-143	2.467×10^{5}	1.73×10^{7}	42.5	1.56×10^{5}	–	–	1.20×10^{6}	1.86×10^{7}
Cerium-144	1.277×10^{5}	8.94×10^{6}	7.13	2.61×10^{4}	–	–	5.62×10^{5}	9.53×10^{6}
Praseodymium-143	2.437×10^{5}	1.71×10^{7}	35.6	1.30×10^{5}	–	–	1.18×10^{6}	1.84×10^{7}

Table I–14 FFTF Core Inventory with Highly Enriched Uranium Driver Fuel (Continued)

Core Isotope	Driver Assemblies		Pu-238 Production		Medical	Co-60	ATWs	
	HEU Driver Activity (Ci)	70 HEU Drivers Activity (Ci)	Per Gram Pu-238 Activity (Ci)	Max. Core Activity[a] (Ci)	12 Ac-227[b] 7 Re-186 I-125 Activity (Ci)	48 Co-60 Assemblies Activity (Ci)	6 MOX Driver Assemblies (6 ATWs)[c] Activity (Ci)	Core Activity (Ci)
Neodymium-147	1.098×10^{5}	7.69×10^{6}	17.1	6.26×10^{4}	–	–	5.91×10^{5}	8.34×10^{6}
Rhenium-186	0.000	0.00	0.00	0.00	3.05×10^{4}	–	0.00	3.64×10^{4}
Radium-223	8.279×10^{-8}	5.80×10^{-6}	0.00	0.00	235	–	1.59×10^{-8}	235
Radium-224	2.260×10^{-4}	0.0158	0.00	0.00	466	–	7.47×10^{-4}	466
Radium-226	3.80×10^{-11}	2.66×10^{-9}	0.00	0.00	172	–	0.00	172
Actinium-227	8.421×10^{-8}	5.89×10^{-6}	0.00	0.00	408	–	1.07×10^{-8}	408
Thorium-227	8.293×10^{-8}	5.81×10^{-6}	0.00	0.00	298	–	1.63×10^{-8}	298
Thorium-228	2.251×10^{-4}	0.0158	0.00	0.00	505	–	7.43×10^{-4}	505
Neptunium-237	0.02577	1.80	0.00360	13.2	–	–	0.0547	15.0
Neptunium-239	2.406×10^{6}	1.68×10^{8}	16.8	6.15×10^{4}	–	–	1.63×10^{7}	1.85×10^{8}
Plutonium-238	57.38	4,020	17	6.19×10^{4}	–	–	742	6.66×10^{4}
Plutonium-239	68.66	4,810	0.00921	33.7	–	–	1,920	6,760
Plutonium-240	10.45	732	0.00393	14.4	–	–	1,560	2,300
Plutonium-241	132.2	9,250	0.853	3,120	–	–	7.28×10^{4}	8.52×10^{4}
Americium-241	0.08854	6.20	0.00	0.00	–	–	847	853
Curium-242	2.844	199	0.0122	44.7	–	–	6.06×10^{4}	6.08×10^{4}
Curium-244	9.215×10^{-4}	0.0645	0.00	0.00	–	–	51.2	51.3

a. Based on a 5-kilogram-per-year plutonium-238 production rate.
b. For the actinium-227 target, over 99.9 percent of the consequences are attributable to six isotopes (actinium-227; radium-223, 224, 226; thorium-227, 228). Therefore, the other actinium-227 target byproducts are not included.
c. Six Accelerator Transmutation of Waste test assemblies included as mixed oxide driver fuel assemblies.

Key: Ac-227, actinium-227; ATW, Accelerator Transmutation of Waste; Ci, curies; Co-60, cobalt-60; HEU, highly enriched uranium; I-125, iodine-125; Pu-238, plutonium-238; Re-186, rhenium-186.

Source: BWHC 1999; Schnitzler 1999; Wootan 1999.

Targets

The proposed core modifications include an array of target assemblies and Rapid Radioisotope Retrieval systems to produce plutonium-238 for space power applications, cobalt-60, and a number of long- and short-lived isotopes for medical applications. In addition, space is to be provided for research and development test articles such as Accelerator Transmutation of Waste test assemblies. As stated previously, a total of six Accelerator Transmutation of Waste test assemblies were conservatively modeled as mixed oxide driver fuel assemblies and included as part of the complement of 76 driver fuel assemblies.

To determine which Rapid Radioisotope Retrieval system and Long-Term Irradiation Vehicle irradiated targets would result in the maximum consequences, the radioisotope inventories for each of the irradiated targets were multiplied by the same release fractions as were assumed for the fuel and fission products (1 percent for solids and 100 percent for noble gases). The resulting inventories were then multiplied by dose conversion factors resulting in a dose for each isotope. The isotope doses within each target were totaled for a target dose, and the target doses were compared to determine which target would result in the maximum consequence for each target type.

Rapid Radioisotope Retrieval Systems

There is to be a maximum of eight Rapid Radioisotope Retrieval systems in the core. One of the Rapid Radioisotope Retrieval systems is to be configured as a gas target to produce iodine-125 from xenon-124. The other seven will be used for production of solid, short-lived medical isotopes. These seven targets are all modeled as the worst-case type (other than gas) to maximize the resulting dose contribution of an accident. The worst-case target planned for insertion in a Rapid Radioisotope Retrieval system is the xenon-124 gas tube, which is assumed to release 100 percent of its iodine-125 inventory along with the xenon-124 gas into containment in the event of any break in the system. As the next worst is the rhenium-186 production target, the other seven Rapid Radioisotope Retrieval systems were assumed to be rhenium-186 production targets. The Rapid Radioisotope Retrieval system inventory is shown in Tables I–13 and I–14.

Long-Term Irradiation Vehicle

Twelve Long-Term Irradiation Vehicle assemblies for production of long-lived medical isotopes are assumed. These assemblies are all modeled as the worst-case type to maximize the dose contribution of an accident. The worst-case Long-Term Irradiation Vehicle target is the actinium-227 production target. All 12 Long-Term Irradiation Vehicle targets are therefore assumed to be actinium-227 production targets. The Long-Term Irradiation Vehicle inventory is shown in Tables I–13 and I–14.

Cobalt-60 Production Target Assemblies

Forty-eight cobalt-60 production targets are to be included in row 9 (outside the reflector assemblies) with a currently assumed annual production rate of 2.016×10^7 curies. The residence time for these targets is to be three 100-day cycles with 16 assemblies being harvested at the end of each cycle. Assuming 2.73 cycles per year for FFTF, this leads to a maximum end-of-cycle core inventory of $(1/3 + 2/3 + 1)$ $(2.016\times10^7 \text{ curies})/2.73 = 1.48\times10^7$ curies. This inventory is included in Tables I–13 and I–14.

Plutonium-238 Production Target Assemblies

Fifteen plutonium-238 production targets are to be included in the reflector region with a currently assumed annual production rate of 5 kilograms (11 pounds). The residence time for these targets is to be three 100-day cycles with five assemblies being harvested at the end of each cycle. Assuming 2.73 cycles per year for FFTF,

this leads to a maximum end-of-cycle core inventory of (1/3 + 2/3 + 1) (5 kilograms)/2.73 = 3.66 kilograms (8.07 pounds). The end-of-cycle target inventory per gram of plutonium-238 and the associated maximum end-of-cycle inventory (3.66 kilograms [8.07 pounds]) are shown in Tables I–13 and I–14.

DESIGN-BASIS ACCIDENT

A wide range of design-basis accidents is analyzed in Chapter 15 of the *FFTF Final Safety Analysis Report* (Dautel 2000). The reactor accidents include various reactivity increase and heat removal reduction transients as well as local fuel failure and natural phenomena (e.g., seismic) events. It should be noted that the *FFTF Final Safety Analysis Report* does not specifically identify a probability of occurrence for each event, but does classify each as being in the anticipated, unlikely, or extremely unlikely category.

For the design-basis transients, the reactor shutdown system was shown to initiate automatic reactor shutdown (scram) in sufficient time to maintain calculated cladding temperatures/strains within limits that ensured that the integrity of the fuel cladding was maintained. Postulated local fuel failure events were shown to remain local (not propagate) and thus have minimal radiological consequences. The core characteristics (physical, nuclear, and thermal) used in the accident analyses of the final safety analysis report were selected to bound those for any anticipated core design. Also, the characteristics of the new-missions core are expected to be similar to those of previous cores (at the same power level). Therefore, the design-basis transients for the new missions core are expected to be essentially the same as those documented in the final safety analysis report. However, the isotopic inventory of the new-missions core will be different from that is used in the final safety analysis report.

The accidental release of primary sodium from the main heat transport system resulted in a larger radiological release than any other reactor related design-basis event (radiological releases can also occur due to non-reactor-related events, such as fuel-handling accidents). The analyses given in the final safety analysis report include some large spills of primary sodium but the spills involving primary sodium are in cells which are inerted and located within containment. The amount of radioactivity released to the environment is bounded by the main heat transport system spill. Secondary-loop sodium does not contain significant radioactive materials, so the radiological consequences of secondary sodium spills are negligible.

Primary sodium is radioactive and may also contain small amounts of fission products. Sodium temperatures are maintained at less than 566 °C (1,050 °F), much below the sodium boiling point (881 °C [1,618 °F] at atmospheric pressure). Sodium at this temperature will retain practically all of the fission products dissolved in it (except the noble gases). Therefore, sodium itself provides the first barrier to the release of any radioactive species.

Primary sodium is contained in high-integrity stainless steel piping and vessels, which provide the second barrier to release. An additional safety margin is provided by the low system operating pressure (less than 200 pounds per square inch gage). The primary sodium systems are located in inert-gas-filled cells (nitrogen plus 0.8–1.2 volume-percent oxygen) to preclude sustained burning in the event of a spill. Therefore, the cell temperature and pressure rise due to primary sodium spills are minimized. An additional protective feature is the sensitive primary sodium leak detection system, including detectors which annunciate on low oxygen level in the cell atmosphere. These cells are constructed of reinforced concrete several feet thick and completely lined with welded steel plate. They provide the third barrier to radioactivity release in case of a primary sodium spill. These subgrade primary cells are located within the Containment Building, the fourth barrier to the release of radioactivity. Some low-pressure, low inventory auxiliary sodium systems, which are connected to the main primary sodium coolant system, are located in similar concrete, steel-lined subgrade cells in a building adjacent to the Containment Building.

The limiting accident is a spill of primary sodium in the inerted sodium and argon sampling pipeway located outside of containment in the Heat Transport System Building–South. The spill is assumed to occur with the reactor at full power (400 megawatts) and very conservatively assumed to be operating with 1 percent failed fuel. In other words, it is assumed that 1 percent of the radioactive inventory (fission products and actinides plus daughters) of 76 fuel assemblies is dispersed uniformly in the primary sodium. Radioactive gases would not be dispersed in the sodium and therefore would not be available for release in this accident. However, for conservatism, 1 percent of the radioactive gas inventory was included in the analysis.

The mass of primary sodium in the main heat transport system is 421,940 kilograms (930,220 pounds). In addition to fission products and fuel, sodium activation products (sodium-22 and sodium-24) will be created during irradiation. The equilibrium sodium activity is assumed to be 4.1×10^{-4} curie per pound for sodium-22 and 5.38 curies per pound for sodium-24. The final safety analysis report assumes that 393 kilograms (867 pounds) of primary sodium is spilled in the inerted sodium and argon sampling pipeway and that 24.4 kilograms (53.7 pounds) of the total spill burns to form an airborne oxide. The fraction of radioactive inventory (sodium activation, fission products and actinides plus daughters) available for release from the sodium and argon sampling pipeway is $(0.01 \times 53.7)/930{,}220 = 5.77\times10^{-7}$. The leak rate of the sodium and argon sampling pipeway is assumed to be 25 percent per day. The probability of this event is judged to be extremely unlikely (1×10^{-4} to 1×10^{-6} per year). For this NI PEIS, the probability is conservatively chosen to be 1×10^{-4} per year.

The source terms for the design-basis sodium spill with mixed oxide fuel and highly enriched uranium fuel are presented in **Tables I–15 and I–16**, respectively.

It should be noted that the reactor power will be 100 megawatts or one-fourth of the design power, for most of the new-missions operation. The fission production rate will be less for this lower-power operation. Therefore, the actual inventory of radioisotopes will likely be less than the conservative bounding inventory assumed for this analysis.

Table I–15 Design-Basis-Accident Source Term—Mixed Oxide Fuel

Radioisotope	Primary Sodium Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	4,380	5.77×10^{-7}	0.00253
Sodium-22	381	5.77×10^{-5}	0.0220
Sodium-24	5.00×10^{6}	5.77×10^{-5}	289
Cobalt-60	1.48×10^{7}	5.77×10^{-7}	8.54
Krypton-85	3.94×10^{4}	5.77×10^{-7}	0.0227
Krypton-85m	2.19×10^{6}	5.77×10^{-7}	1.26
Krypton-87	3.62×10^{6}	5.77×10^{-7}	2.09
Krypton-88	5.04×10^{6}	5.77×10^{-7}	2.91
Rubidium-86	6.90×10^{4}	5.77×10^{-7}	0.0398
Strontium-89	5.81×10^{6}	5.77×10^{-7}	3.35
Strontium-90	2.42×10^{5}	5.77×10^{-7}	0.140
Strontium-91	9.24×10^{6}	5.77×10^{-7}	5.33
Strontium-92	1.09×10^{7}	5.77×10^{-7}	6.28
Yttrium-90	2.71×10^{5}	5.77×10^{-7}	0.156
Yttrium-91	7.64×10^{6}	5.77×10^{-7}	4.41
Yttrium-92	1.10×10^{7}	5.77×10^{-7}	6.35
Yttrium-93	1.37×10^{7}	5.77×10^{-7}	7.91
Zirconium-95	1.36×10^{7}	5.77×10^{-7}	7.84
Zirconium-97	1.81×10^{7}	5.77×10^{-7}	10.4
Niobium-95	1.14×10^{7}	5.77×10^{-7}	6.58
Molybdenum-99	2.07×10^{7}	5.77×10^{-7}	11.9
Technetium-99m	1.81×10^{7}	5.77×10^{-7}	10.4
Ruthenium-103	2.08×10^{7}	5.77×10^{-7}	12.0
Ruthenium-105	1.74×10^{7}	5.77×10^{-7}	10.0
Ruthenium-106	7.17×10^{6}	5.77×10^{-7}	4.14
Rhodium-105	1.72×10^{7}	5.77×10^{-7}	9.94
Antimony-127	1.98×10^{6}	5.77×10^{-7}	1.14
Antimony-129	4.82×10^{6}	5.77×10^{-7}	2.78
Iodine-125	2,530	5.77×10^{-7}	0.00146
Tellurium-127	1.89×10^{6}	5.77×10^{-7}	1.09
Tellurium-127m	1.94×10^{5}	5.77×10^{-7}	0.112
Tellurium-129	4.77×10^{6}	5.77×10^{-7}	2.75
Tellurium-129m	6.61×10^{5}	5.77×10^{-7}	0.381
Tellurium-131	1.19×10^{7}	5.77×10^{-7}	6.84
Tellurium-131m	2.06×10^{6}	5.77×10^{-7}	1.19
Tellurium-132	1.78×10^{7}	5.77×10^{-7}	10.2
Iodine-131	1.35×10^{7}	5.77×10^{-7}	7.78
Iodine-132	1.82×10^{7}	5.77×10^{-7}	10.5
Iodine-133	2.30×10^{7}	5.77×10^{-7}	13.3
Iodine-134	2.44×10^{7}	5.77×10^{-7}	14.1
Iodine-135	2.21×10^{7}	5.77×10^{-7}	12.8
Xenon-133	2.34×10^{7}	5.77×10^{-7}	13.5
Xenon-135	2.48×10^{7}	5.77×10^{-7}	14.3
Cesium-134	3.79×10^{5}	5.77×10^{-7}	0.219
Cesium-136	7.22×10^{5}	5.77×10^{-7}	0.416
Cesium-137	6.37×10^{5}	5.77×10^{-7}	0.367
Barium-139	1.99×10^{7}	5.77×10^{-7}	11.5
Barium-140	1.84×10^{7}	5.77×10^{-7}	10.6
Lanthanum-140	1.86×10^{7}	5.77×10^{-7}	10.7

Radioisotope	Primary Sodium Activity (curies)	Release Fraction	Environmental Release (curies)
Lanthanum-141	1.89×10^{7}	5.77×10^{-7}	10.9
Lanthanum-142	1.67×10^{7}	5.77×10^{-7}	9.66
Cerium-141	1.76×10^{7}	5.77×10^{-7}	10.1
Cerium-143	1.53×10^{7}	5.77×10^{-7}	8.85
Cerium-144	7.14×10^{6}	5.77×10^{-7}	4.12
Praseodymium-143	1.51×10^{7}	5.77×10^{-7}	8.70
Neodymium-147	7.55×10^{6}	5.77×10^{-7}	4.35
Rhenium-186	3.64×10^{4}	5.77×10^{-7}	0.0210
Radium-223	235	5.77×10^{-7}	1.36×10^{-4}
Radium-224	466	5.77×10^{-7}	2.69×10^{-4}
Radium-226	172	5.77×10^{-7}	9.92×10^{-5}
Actinium-227	408	5.77×10^{-7}	2.35×10^{-4}
Thorium-227	298	5.77×10^{-7}	1.72×10^{-4}
Thorium-228	505	5.77×10^{-7}	2.91×10^{-4}
Neptunium-237	13.9	5.77×10^{-7}	8.00×10^{-6}
Neptunium-239	2.07×10^{8}	5.77×10^{-7}	119
Plutonium-238	7.12×10^{4}	5.77×10^{-7}	0.0411
Plutonium-239	2.44×10^{4}	5.77×10^{-7}	0.0141
Plutonium-240	1.97×10^{4}	5.77×10^{-7}	0.0114
Plutonium-241	9.25×10^{5}	5.77×10^{-7}	0.534
Americium-241	1.07×10^{4}	5.77×10^{-7}	0.00619
Curium-242	7.47×10^{5}	5.77×10^{-7}	0.431
Curium-244	631	5.77×10^{-7}	3.64×10^{-4}

Source: Nielsen 1999.

Table I–16 Design-Basis-Accident Source Term—Highly Enriched Uranium Fuel

Radioisotope	Primary Sodium Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	4,980	5.77×10^{-7}	0.00287
Sodium-22	381	5.77×10^{-5}	0.0220
Sodium-24	5.00×10^{6}	5.77×10^{-5}	289
Cobalt-60	1.48×10^{7}	5.77×10^{-7}	8.54
Krypton-85	7.05×10^{4}	5.77×10^{-7}	0.0407
Krypton-85m	3.86×10^{6}	5.77×10^{-7}	2.22
Rubidium-86	1.15×10^{5}	5.77×10^{-7}	0.0666
Krypton-87	7.28×10^{6}	5.77×10^{-7}	4.20
Krypton-88	1.05×10^{7}	5.77×10^{-7}	6.04
Strontium-89	1.16×10^{7}	5.77×10^{-7}	6.69
Strontium-90	5.17×10^{5}	5.77×10^{-7}	0.298
Strontium-91	1.68×10^{7}	5.77×10^{-7}	9.67
Strontium-92	1.76×10^{7}	5.77×10^{-7}	10.2
Yttrium-90	5.74×10^{5}	5.77×10^{-7}	0.331
Yttrium-91	1.41×10^{7}	5.77×10^{-7}	8.14
Yttrium-92	1.77×10^{7}	5.77×10^{-7}	10.2
Yttrium-93	1.98×10^{7}	5.77×10^{-7}	11.4
Zirconium-95	1.71×10^{7}	5.77×10^{-7}	9.85
Zirconium-97	2.01×10^{7}	5.77×10^{-7}	11.6
Niobium-95	1.44×10^{7}	5.77×10^{-7}	8.31
Molybdenum-99	2.07×10^{7}	5.77×10^{-7}	11.9
Technetium-99m	1.81×10^{7}	5.77×10^{-7}	10.4
Ruthenium-103	1.39×10^{7}	5.77×10^{-7}	8.01
Ruthenium-105	8.40×10^{6}	5.77×10^{-7}	4.85
Ruthenium-106	2.52×10^{6}	5.77×10^{-7}	1.45
Rhodium-105	8.39×10^{6}	5.77×10^{-7}	4.84
Antimony-127	1.69×10^{6}	5.77×10^{-7}	0.976
Antimony-129	4.13×10^{6}	5.77×10^{-7}	2.38
Tellurium-127m	1.65×10^{5}	5.77×10^{-7}	0.0955
Tellurium-127	1.62×10^{6}	5.77×10^{-7}	0.935
Tellurium-129	4.18×10^{6}	5.77×10^{-7}	2.41
Tellurium-129m	5.71×10^{5}	5.77×10^{-7}	0.329
Tellurium-131	1.09×10^{7}	5.77×10^{-7}	6.31
Tellurium-131m	1.60×10^{6}	5.77×10^{-7}	0.925
Tellurium-132	1.68×10^{7}	5.77×10^{-7}	9.68
Iodine-125	2,530	5.77×10^{-7}	0.00146
Iodine-131	1.22×10^{7}	5.77×10^{-7}	7.03
Iodine-132	1.71×10^{7}	5.77×10^{-7}	9.84
Iodine-133	2.30×10^{7}	5.77×10^{-7}	13.3
Iodine-134	2.60×10^{7}	5.77×10^{-7}	15.0
Iodine-135	2.13×10^{7}	5.77×10^{-7}	12.3
Xenon-133	2.33×10^{7}	5.77×10^{-7}	13.4
Xenon-135	2.29×10^{7}	5.77×10^{-7}	13.2
Cesium-134	3.58×10^{5}	5.77×10^{-7}	0.206
Cesium-136	4.32×10^{5}	5.77×10^{-7}	0.249
Cesium-137	6.34×10^{5}	5.77×10^{-7}	0.366
Barium-139	2.16×10^{7}	5.77×10^{-7}	12.5
Barium-140	2.03×10^{7}	5.77×10^{-7}	11.7

Radioisotope	Primary Sodium Activity (curies)	Release Fraction	Environmental Release (curies)
Lanthanum-140	2.05×10^{7}	5.77×10^{-7}	11.8
Lanthanum-141	2.03×10^{7}	5.77×10^{-7}	11.7
Lanthanum-142	1.87×10^{7}	5.77×10^{-7}	10.8
Cerium-141	1.89×10^{7}	5.77×10^{-7}	10.9
Cerium-143	1.86×10^{7}	5.77×10^{-7}	10.7
Cerium-144	9.53×10^{6}	5.77×10^{-7}	5.50
Praseodymium-143	1.84×10^{7}	5.77×10^{-7}	10.6
Neodymium-147	8.34×10^{6}	5.77×10^{-7}	4.81
Rhenium-186	3.64×10^{4}	5.77×10^{-7}	0.0210
Radium-223	235	5.77×10^{-7}	1.36×10^{-4}
Radium-224	466	5.77×10^{-7}	2.69×10^{-4}
Radium-226	172	5.77×10^{-7}	9.92×10^{-5}
Actinium-227	408	5.77×10^{-7}	2.35×10^{-4}
Thorium-227	298	5.77×10^{-7}	1.72×10^{-4}
Thorium-228	505	5.77×10^{-7}	2.91×10^{-4}
Neptunium-237	15.0	5.77×10^{-7}	8.67×10^{-6}
Neptunium-239	1.85×10^{8}	5.77×10^{-7}	107
Plutonium-238	6.66×10^{4}	5.77×10^{-7}	0.0384
Plutonium-239	6,760	5.77×10^{-7}	0.00390
Plutonium-240	2,300	5.77×10^{-7}	0.00133
Plutonium-241	8.52×10^{4}	5.77×10^{-7}	0.0491
Americium-241	853	5.77×10^{-7}	4.92×10^{-4}
Curium-242	6.08×10^{4}	5.77×10^{-7}	0.0351
Curium-244	51.3	5.77×10^{-7}	2.96×10^{-5}

Source: Nielsen 1999.

SEVERE REACTOR ACCIDENT

In addition to the design-basis accidents analyzed in Chapter 15 of the *FFTF Final Safety Analysis Report* (Dautel 2000), Appendix A of the facility safety analysis report documents the analysis of two beyond-design-basis events: unprotected transient overpower and unprotected loss of primary sodium flow (unprotected refers to the assumption that the reactor shutdown system fails to shut down the reactor). These two unprotected events are considered to bound the consequences of other potential beyond-design-basis events such as loss of decay heat removal capability.

The unprotected transient overpower event was found to be relatively benign (i.e., no substantial release of radioactive material is expected). The final safety analysis report results indicated that the event would be terminated by fuel melting and sweepout from a few fuel assemblies and in-place cooling of the remainder of the core. There was no identified source of substantial energetics that would challenge the integrity of the reactor vessel, primary heat transport system, or containment boundaries.

In the case of the unprotected loss-of-flow event, meltdown of the entire core could not be precluded, and release of the entire core contents to the primary heat transport system could occur. Extensive analysis showed, however, that a core meltdown does not threaten the integrity of the reactor vessel or primary heat transport system. The core contents are released and severely contaminate the primary system, but are not expected to leak from the primary boundary. Although a relatively benign scenario of fuel melting/boilout was predicted, the possibility of energetics from either large-reactivity insertion events or hot-core interaction with outlet plenum sodium (rapid generation/expansion of sodium vapor) could not be precluded. Conservative estimates of the energy releases from these scenarios were made, and it was shown that the reactor vessel, primary heat

transport system, and containment boundaries would remain intact (although some primary sodium was calculated to be expelled through reactor head seals into the Containment Building due to sodium slug impact on the underside of the reactor head).

Since the neutronic and thermal-hydraulic conditions for the proposed new-missions core are expected to be similar to those for the previous FFTF cores (at 400 megawatts), the severe-accident scenarios are also expected to be similar. It should be noted that lower-power (100-megawatt) operation would reduce the severity of severe accidents. It is assumed that all the core fuel assemblies and reflector target assemblies eventually melt during the unprotected loss-of-flow accident. It is further assumed that no mitigating actions are taken to restore core cooling during the event, and that, as a bounding case, all fuel assemblies melt immediately after reactor shutdown with no decay time prior to release from containment, and that an energetic sodium release into containment occurs consistent with the final safety analysis report–stipulated unprotected loss-of-flow accident.

In the final safety analysis report analysis, 136 kilograms (300 pounds) of sodium was assumed to spray into the containment and burn, thereby heating and pressurizing the containment atmosphere. This provides the driving force for leakage from the containment into the environment. The inclusion of up to eight Rapid Radioisotope Retrieval systems provides additional potential leakage paths for sodium ejection into the Containment Building during sodium slug impact on the bottom of the reactor head. An increase in the quantity of sodium ejected from the primary system would cause increased leakage into the environment.

According to the current conceptual design for the Rapid Radioisotope Retrieval system, each system uses a target tube with an inside diameter of 1.89 centimeters (0.745 inch). This leads to a leakage area for eight systems (including the gas target) of 22.5 square centimeters (3.49 square inches). Assuming that the leak rate is proportional to the leakage area leads to an estimated total leakage of approximately 336 kilograms (740 pounds) of sodium from the eight Rapid Radioisotope Retrieval positions during a postulated unprotected loss-of-flow accident. To account for uncertainty in the calculation and to add conservatism to the evaluation, a leakage of 363 kilograms (800 pounds) of sodium was specified as the contribution from the Rapid Radioisotope Retrieval systems. This increases the total sodium leakage into containment from 136 kilograms (300 pounds) to 499 kilograms (1,100 pounds). In addition to the sodium, 100 percent of the noble gases and one percent of the core fuel and fission product inventory were assumed to be released to the containment.

One of the effects of an additional amount of sodium being sprayed into the containment and burned is to increase the pressurization of the containment and hence the amount and rate of release from the containment. Heating of the containment atmosphere due to the 136-kilograms (300-pound) sodium spill resulted in a peak containment pressure of 1.84 pounds per square inch gage. The revised analysis assumes that 499 kilograms (1,100 pounds) of sodium is ejected into the containment, increasing the peak containment pressure to 4.99 pounds per square inch gage.

The sodium is assumed to mix uniformly with the air in the containment and burn completely. In addition, all the heat conducted from the sodium due to its elevated temperature and all of the heat of combustion are used to heat the air in the containment, with no transmission to the walls or structure. The resulting peak containment pressure of 4.99 pounds per square inch gage is well below the containment design pressure of 10 pounds per square inch gage. The release from the containment building is based on the design release rate of 0.5 percent per day for the duration of the pressure buildup (approximately 225 hours).

Assuming 100 percent of the noble gases and tritium in the core is released to the containment, and the containment leaks at 0.5 percent per day for 24 hours, the release fraction for noble gases and tritium is 0.05 ($.005/24 \times 225 = 0.047 \approx 0.05$). Assuming 1 percent of the fission products, fuel, and target inventory is released to the containment, the release fraction for these isotopes is 5×10^{-4} (0.05×0.01). Assuming

499 kilograms (1,100 pounds) of the 421,940 kilograms (930,220 pounds) of primary sodium is released to the containment, the release fraction for sodium is 5.92×10^{-5} (0.05 × 1,100/930,220).

The unprotected loss-of-flow event, resulting in a complete core melt, represents the most severe accident analyzed for FFTF. The frequency of this event was estimated to be 1×10^{-9} per year, based on internally initiated events (Dautel 2000). For this analysis, the frequency was increased to 1×10^{-6} to incorporate the spectrum of externally initiated events that could contribute to the severe core melt scenario. The main contributor to the increased frequency is a catastrophic earthquake. The magnitude of potential earthquakes with return periods greater than 10,000 years is highly uncertain. For the purposes of this NI PEIS, it was assumed that an earthquake with a return period of 1 million years would result in sufficient ground motion to cause major damage to FFTF resulting in a core melt scenario. An earthquake of this magnitude could result in severe effects to the entire region, including building collapses, power outages, and road hazards.

The source terms for the beyond-design-basis core melt accident with mixed oxide fuel and highly enriched uranium fuel are presented in **Tables I–17** and **I–18**, respectively.

Table I–17 Beyond-Design-Basis Accident Source Term—Mixed Oxide Fuel

Radioisotope	Core Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	4,380	0.05	219
Sodium-22	381	5.92×10^{-5}	0.0226
Sodium-24	5.00×10^{6}	5.92×10^{-5}	296
Cobalt-60	1.48×10^{7}	5×10^{-4}	7,400
Krypton-85	3.94×10^{4}	0.05	1,971
Krypton-85m	2.19×10^{6}	0.05	1.09×10^{5}
Krypton-87	3.62×10^{6}	0.05	1.81×10^{5}
Krypton-88	5.04×10^{6}	0.05	2.52×10^{5}
Rubidium-86	6.90×10^{4}	5×10^{-4}	34.5
Strontium-89	5.81×10^{6}	5×10^{-4}	2,905
Strontium-90	2.42×10^{5}	5×10^{-4}	121
Strontium-91	9.24×10^{6}	5×10^{-4}	4,622
Strontium-92	1.09×10^{7}	5×10^{-4}	5,440
Yttrium-90	2.71×10^{5}	5×10^{-4}	136
Yttrium-91	7.64×10^{6}	5×10^{-4}	3,818
Yttrium-92	1.10×10^{7}	5×10^{-4}	5,502
Yttrium-93	1.37×10^{7}	5×10^{-4}	6,851
Zirconium-95	1.36×10^{7}	5×10^{-4}	6,794
Zirconium-97	1.81×10^{7}	5×10^{-4}	9,050
Niobium-95	1.14×10^{7}	5×10^{-4}	5,700
Molybdenum-99	2.07×10^{7}	5×10^{-4}	1.03×10^{4}
Technetium-99m	1.81×10^{7}	5×10^{-4}	9,041
Ruthenium-103	2.08×10^{7}	5×10^{-4}	1.04×10^{4}
Ruthenium-105	1.74×10^{7}	5×10^{-4}	8,686
Ruthenium-106	7.17×10^{6}	5×10^{-4}	3,698
Rhodium-105	1.72×10^{7}	5×10^{-4}	8,610
Antimony-127	1.98×10^{6}	5×10^{-4}	988
Antimony-129	4.82×10^{6}	5×10^{-4}	2,411
Tellurium-127	1.89×10^{6}	5×10^{-4}	947
Tellurium-127m	1.94×10^{5}	5×10^{-4}	96.8
Tellurium-129	4.77×10^{6}	5×10^{-4}	2,383
Tellurium-129m	6.61×10^{5}	5×10^{-4}	330
Tellurium-131	1.19×10^{7}	5×10^{-4}	5,931
Tellurium-131m	2.06×10^{6}	5×10^{-4}	1,031
Tellurium-132	1.78×10^{7}	5×10^{-4}	8,879
Iodine-125	2,530	5×10^{-4}	1.30
Iodine-131	1.35×10^{7}	5×10^{-4}	6,744
Iodine-132	1.82×10^{7}	5×10^{-4}	9,084
Iodine-133	2.30×10^{7}	5×10^{-4}	1.15×10^{4}
Iodine-134	2.44×10^{7}	5×10^{-4}	1.22×10^{4}
Iodine-135	2.21×10^{7}	5×10^{-4}	1.11×10^{4}
Xenon-133	2.34×10^{7}	0.05	1.17×10^{6}
Xenon-135	2.48×10^{7}	0.05	1.24×10^{6}
Cesium-134	3.79×10^{5}	5×10^{-4}	190

Radioisotope	Core Activity (curies)	Release Fraction	Environmental Release (curies)
Cesium-136	7.22×10^{5}	5×10^{-4}	361
Cesium-137	6.37×10^{5}	5×10^{-4}	318
Barium-139	1.99×10^{7}	5×10^{-4}	9,956
Barium-140	1.84×10^{7}	5×10^{-4}	9,191
Lanthanum-140	1.86×10^{7}	5×10^{-4}	9,281
Lanthanum-141	1.89×10^{7}	5×10^{-4}	9,442
Lanthanum-142	1.67×10^{7}	5×10^{-4}	8,367
Cerium-141	1.76×10^{7}	5×10^{-4}	8,781
Cerium-143	1.53×10^{7}	5×10^{-4}	7,670
Cerium-144	7.14×10^{6}	5×10^{-4}	3,570
Praseodymium-143	1.51×10^{7}	5×10^{-4}	7,536
Neodymium-147	7.55×10^{6}	5×10^{-4}	3,773
Rhenium-186	3.05×10^{4}	5×10^{-4}	18.2
Radium-223	235	5×10^{-4}	0.00
Radium-224	466	5×10^{-4}	0.00
Radium-226	172	5×10^{-4}	0.00
Actinium-227	408	5×10^{-4}	0.204
Thorium-227	298	5×10^{-4}	0.00
Thorium-228	505	5×10^{-4}	0.00
Neptunium-237	13.9	5×10^{-4}	0.00693
Neptunium-239	2.07×10^{8}	5×10^{-4}	1.04×10^{5}
Plutonium-238	7.12×10^{4}	5×10^{-4}	35.6
Plutonium-239	2.44×10^{4}	5×10^{-4}	12.2
Plutonium-240	1.97×10^{4}	5×10^{-4}	9.86
Plutonium-241	9.25×10^{5}	5×10^{-4}	463
Americium-241	1.07×10^{4}	5×10^{-4}	5.36
Curium-242	7.47×10^{5}	5×10^{-4}	384
Curium-244	631	5×10^{-4}	0.325

Source: Calculated results.

Table I–18 Beyond-Design-Basis Accident Source Term—Highly Enriched Uranium Fuel

Radioisotope	Core Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	4,980	0.05	249
Sodium-22	381	5.92×10^{-5}	0.0226
Sodium-24	5.00×10^{6}	5.92×10^{-5}	296
Cobalt-60	1.48×10^{7}	5×10^{-4}	7,400
Krypton-85	7.05×10^{4}	0.05	3,526
Krypton-85m	3.86×10^{6}	0.05	1.93×10^{5}
Krypton-87	7.28×10^{6}	0.05	3.64×10^{5}
Krypton-88	1.05×10^{7}	0.05	5.24×10^{5}
Rubidium-86	1.15×10^{5}	5×10^{-4}	57.7
Strontium-89	1.16×10^{7}	5×10^{-4}	5,797
Strontium-90	5.17×10^{5}	5×10^{-4}	259
Strontium-91	1.68×10^{7}	5×10^{-4}	8,381
Strontium-92	1.76×10^{7}	5×10^{-4}	8,825
Yttrium-90	5.74×10^{5}	5×10^{-4}	287
Yttrium-91	1.41×10^{7}	5×10^{-4}	7,051
Yttrium-92	1.77×10^{7}	5×10^{-4}	8,872
Yttrium-93	1.98×10^{7}	5×10^{-4}	9,896
Zirconium-95	1.71×10^{7}	5×10^{-4}	8,537
Zirconium-97	2.01×10^{7}	5×10^{-4}	1.00×10^{4}
Niobium-95	1.44×10^{7}	5×10^{-4}	7,202
Molybdenum-99	2.07×10^{7}	5×10^{-4}	1.03×10^{4}
Technetium-99m	1.81×10^{7}	5×10^{-4}	9,051
Ruthenium-103	1.39×10^{7}	5×10^{-4}	6,938
Ruthenium-105	8.40×10^{6}	5×10^{-4}	4,199
Ruthenium-106	2.52×10^{6}	5×10^{-4}	1,260
Rhodium-105	8.39×10^{6}	5×10^{-4}	4,194
Antimony-127	1.69×10^{6}	5×10^{-4}	846
Antimony-129	4.13×10^{6}	5×10^{-4}	2,064
Tellurium-127	1.62×10^{6}	5×10^{-4}	810
Tellurium-127m	1.65×10^{5}	5×10^{-4}	82.7
Tellurium-129	4.18×10^{6}	5×10^{-4}	2,088
Tellurium-129m	5.71×10^{5}	5×10^{-4}	285
Tellurium-131	1.09×10^{7}	5×10^{-4}	5,466
Tellurium-131m	1.60×10^{6}	5×10^{-4}	801
Tellurium-132	1.68×10^{7}	5×10^{-4}	8,389
Iodine-125	2,530	5×10^{-4}	1.30
Iodine-131	1.22×10^{7}	5×10^{-4}	6,096
Iodine-132	1.71×10^{7}	5×10^{-4}	8,531
Iodine-133	2.30×10^{7}	5×10^{-4}	1.15×10^{4}
Iodine-134	2.60×10^{7}	5×10^{-4}	1.30×10^{4}
Iodine-135	2.13×10^{7}	5×10^{-4}	1.06×10^{4}
Xenon-133	2.33×10^{7}	0.05	1.16×10^{6}
Xenon-135	2.29×10^{7}	0.05	1.15×10^{6}
Cesium-134	3.58×10^{5}	5×10^{-4}	179

Radioisotope	Core Activity (curies)	Release Fraction	Environmental Release (curies)
Cesium-136	4.32×10^{5}	5×10^{-4}	216
Cesium-137	6.34×10^{5}	5×10^{-4}	317
Barium-139	2.16×10^{7}	5×10^{-4}	1.08×10^{4}
Barium-140	2.03×10^{7}	5×10^{-4}	1.02×10^{4}
Lanthanum-140	2.05×10^{7}	5×10^{-4}	1.03×10^{4}
Lanthanum-141	2.03×10^{7}	5×10^{-4}	1.01×10^{4}
Lanthanum-142	1.87×10^{7}	5×10^{-4}	9,354
Cerium-141	1.89×10^{7}	5×10^{-4}	9,474
Cerium-143	1.86×10^{7}	5×10^{-4}	9,312
Cerium-144	9.53×10^{6}	5×10^{-4}	4,763
Praseodymium-143	1.84×10^{7}	5×10^{-4}	9,184
Neodymium-147	8.34×10^{6}	5×10^{-4}	4,170
Rhenium-186	3.64×10^{4}	5×10^{-4}	18.2
Radium-223	235	5×10^{-4}	0.00
Radium-224	466	5×10^{-4}	0.00
Radium-226	172	5×10^{-4}	0.00
Actinium-227	408	5×10^{-4}	0.204
Thorium-227	298	5×10^{-4}	0.00
Thorium-228	505	5×10^{-4}	0.00
Neptunium-237	15.0	5×10^{-4}	0.00752
Neptunium-239	1.85×10^{8}	5×10^{-4}	9.24×10^{4}
Plutonium-238	6.66×10^{4}	5×10^{-4}	33.3
Plutonium-239	6,760	5×10^{-4}	3.38
Plutonium-240	2,300	5×10^{-4}	1.15
Plutonium-241	8.52×10^{4}	5×10^{-4}	42.6
Americium-241	853	5×10^{-4}	0.426
Curium-242	6.08×10^{4}	5×10^{-4}	30.4
Curium-244	51.3	5×10^{-4}	0.026

Source: Calculated results.

FUEL- AND TARGET-HANDLING ACCIDENTS

A range of accidents related to ex-reactor irradiated fuel- and target-handling were postulated to occur outside of the reactor vessel (i.e., nonreactor accidents). The accident scenarios were selected from the *FFTF Final Safety Analysis Report* (Dautel 2000) and evaluated using the existing FFTF irradiated-fuel source term and new source terms for the neptunium-237 and medical isotope targets. The consequences of ex-reactor accidents involving industrial and civilian nuclear energy research and development targets are expected to be bounded by the accident selected in this NI PEIS.

The accident that would lead to the maximum radiological consequences is a seismic event during fuel assembly transfer. The bottom-loading transfer cask is used to transfer single core components from the containment building to the sodium storage vessel located in the Fuel Storage Facility or to a cask at the cask-loading station in the Reactor Service Building. The bottom-loading transfer cask is qualified to protect a fuel element from breach of cladding during a design-basis earthquake. However, if an element is being transferred into or out of another vessel when a design-basis earthquake occurs, a potential for damage to the component exists. This event is much less likely than the design-basis earthquake because of the small fraction of process time spent in the transfer of an assembly from one vessel to another.

Although the bottom-loading transfer cask is designed to remain upright during a design-basis earthquake at all transfer locations, it could move along the supporting rails during such an event. The probability of a design-basis earthquake is about 1×10^{-4} per year. If an assembly were being transferred through the interface between the bottom-loading transfer cask and the top of the other vessel or a floor valve at the exact moment of a design-basis earthquake, then the movement of the bottom-loading transfer cask could produce bending stresses on the assembly. The likelihood of such an occurrence is on the order of 0.001 per year for the proposed mission, resulting in a combined frequency of 1×10^{-7} for this scenario. Failure of fuel pin cladding as a result of assembly bending is not predicted by analysis. However, for the purpose of showing the depth of protection provided by FFTF against any undue risk to the public health and safety, the conservative assumptions listed below for an extreme beyond-design-basis fuel-handling accident were made in the final safety analysis report and are specified for this reevaluation of a fuel assembly.

- The fuel region of the assembly is in the transfer interface, such that the fuel could be damaged.
- Five percent of the fuel pins are assumed to lose cladding integrity.
- Release fractions are 1.0 for tritium and noble gases, 0.5 for halogens, and 0.05 for volatile solids.
- The release fraction for transuranics and nonvolatile solids is determined as follows: 5 percent of the fuel in the column is crushed and 5 percent of the crushed fuel is of respirable size (equal to or less than 10 microns). A suspension and release fraction of 1 percent is assumed for the respirable particles, i.e., 1 percent is released from the bottom-loading transfer cask and from containment or the Reactor Service Building.
- A 50 percent plateout fraction is assumed for halogens.
- No containment isolation is assumed, and the release is assumed to occur at ground level.

In addition to the mixed oxide and highly enriched uranium fuel assemblies, this accident was analyzed for maximum releases from the neptunium-237 and worst-case medical, industrial, and research and development isotope targets. Because the medical, industrial, and research and development isotope target assemblies have not been structurally analyzed for this type of impact event, all the target assembly rods are assumed to breach. No credible scenario has been identified that could produce temperatures high enough to vaporize target material. Because only one assembly can be accommodated by the bottom-loading transfer cask, the maximum release for this accident is from one fuel or target assembly only.

The radioisotope inventory, release fractions, and resulting environmental release for the mixed oxide and highly enriched uranium assemblies are provided in **Tables I–19 and I–20**.

Table I–19 Mixed Oxide Driver Fuel Assembly Source Term

Radioisotope	MOX Driver Fuel Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	57.56	0.05	2.88
Krypton-85	517.6	0.05	25.9
Krypton-85m	2.850×10^{4}	0.05	1,430
Krypton-87	4.716×10^{4}	0.05	2,360
Krypton-88	6.567×10^{4}	0.05	3,280
Rubidium-86	907.6	1.25×10^{-6}	0.00113
Strontium-89	7.598×10^{4}	1.25×10^{-6}	0.0950
Strontium-90	3.181×10^{3}	1.25×10^{-6}	0.00398
Strontium-91	1.205×10^{5}	1.25×10^{-6}	0.151
Strontium-92	1.418×10^{5}	1.25×10^{-6}	0.177
Yttrium-90	3,561	1.25×10^{-6}	0.00445
Yttrium-91	9.984×10^{4}	1.25×10^{-6}	0.125
Yttrium-92	1.434×10^{5}	1.25×10^{-6}	0.179
Yttrium-93	1.785×10^{5}	1.25×10^{-6}	0.223
Zirconium-95	1.776×10^{5}	1.25×10^{-6}	0.222
Zirconium-97	2.357×10^{5}	1.25×10^{-6}	0.295
Niobium-95	1.492×10^{5}	1.25×10^{-6}	0.187
Molybdenum-99	2.690×10^{5}	1.25×10^{-6}	0.336
Technetium-99m	2.355×10^{5}	1.25×10^{-6}	0.294
Ruthenium-103	2.718×10^{5}	1.25×10^{-6}	0.340
Ruthenium-105	2.261×10^{5}	1.25×10^{-6}	0.283
Ruthenium-106	9.408×10^{4}	1.25×10^{-6}	0.118
Rhodium-105	2.246×10^{5}	1.25×10^{-6}	0.281
Antimony-127	2.579×10^{4}	1.25×10^{-6}	0.0322
Antimony-129	6.280×10^{4}	1.25×10^{-6}	0.0785
Tellurium-127	2.471×10^{4}	1.25×10^{-6}	0.0309
Tellurium-127m	2,535	1.25×10^{-6}	0.00317
Tellurium-129	6.211×10^{4}	1.25×10^{-6}	0.0776
Tellurium-129m	8,626	1.25×10^{-6}	0.0108
Tellurium-131	1.546×10^{5}	1.25×10^{-6}	0.193
Tellurium-131m	2.684×10^{4}	1.25×10^{-6}	0.0336
Tellurium-132	2.314×10^{5}	1.25×10^{-6}	0.289
Iodine-131	1.759×10^{5}	0.0125	2,200
Iodine-132	2.367×10^{5}	0.0125	2,960
Iodine-133	2.996×10^{5}	0.0125	3,750
Iodine-134	3.173×10^{5}	0.0125	3,970
Iodine-135	2.883×10^{5}	0.0125	3,600
Xenon-133	3.051×10^{5}	0.05	1.53×10^{4}
Xenon-135	3.254×10^{5}	0.05	1.63×10^{4}
Cesium-134	4,980	0.00250	12.5
Cesium-136	9,451	0.00250	23.6
Cesium-137	8,361	0.00250	20.9
Barium-139	2.594×10^{5}	1.25×10^{-6}	0.324
Barium-140	2.397×10^{5}	1.25×10^{-6}	0.300
Lanthanum-140	2.421×10^{5}	1.25×10^{-6}	0.303
Lanthanum-141	2.460×10^{5}	1.25×10^{-6}	0.308
Lanthanum-142	2.179×10^{5}	1.25×10^{-6}	0.272
Cerium-141	2.294×10^{5}	1.25×10^{-6}	0.287

Radioisotope	MOX Driver Fuel Activity (curies)	Release Fraction	Environmental Release (curies)
Cerium-143	1.998×10^{5}	1.25×10^{-6}	0.250
Cerium-144	9.360×10^{4}	1.25×10^{-6}	0.117
Praseodymium-143	1.966×10^{5}	1.25×10^{-6}	0.246
Neodymium-147	9.847×10^{4}	1.25×10^{-6}	0.123
Neptunium-237	9.117×10^{-3}	1.25×10^{-6}	1.14×10^{-8}
Neptunium-239	2.723×10^{6}	1.25×10^{-6}	3.40
Plutonium-238	123.6	1.25×10^{-6}	1.55×10^{-4}
Plutonium-239	320.1	1.25×10^{-6}	4.00×10^{-4}
Plutonium-240	259.3	1.25×10^{-6}	3.24×10^{-4}
Plutonium-241	1.213×10^{4}	1.25×10^{-6}	0.0152
Americium-241	141.1	1.25×10^{-6}	1.76×10^{-4}
Curium-242	9,829	1.25×10^{-6}	0.0123
Curium-244	8.305	1.25×10^{-6}	1.04×10^{-5}

Key: MOX, mixed oxide.
Source: Nielsen 1999; Wootan 1999.

Table I–20 Highly Enriched Uranium Driver Fuel Assembly Source Term

Radioisotope	HEU Driver Fuel Activity (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	66.04	0.0500	3.302
Krypton-85	962.1	0.0500	48.11
Krypton-85m	5.236×10^{4}	0.0500	2,618
Krypton-87	9.952×10^{4}	0.0500	4,976
Krypton-88	1.433×10^{5}	0.0500	7,165
Rubidium-86	1,571	1.25×10^{-6}	0.001964
Strontium-89	1.586×10^{5}	1.25×10^{-6}	0.1983
Strontium-90	7,107	1.25×10^{-6}	0.008884
Strontium-91	2.279×10^{5}	1.25×10^{-6}	0.2849
Strontium-92	2.385×10^{5}	1.25×10^{-6}	0.2981
Yttrium-90	7,885	1.25×10^{-6}	0.009856
Yttrium-91	1.922×10^{5}	1.25×10^{-6}	0.2403
Yttrium-92	2.397×10^{5}	1.25×10^{-6}	0.2996
Yttrium-93	2.655×10^{5}	1.25×10^{-6}	0.3319
Zirconium-95	2.274×10^{5}	1.25×10^{-6}	0.2843
Zirconium-97	2.636×10^{5}	1.25×10^{-6}	0.3295
Niobium-95	1.921×10^{5}	1.25×10^{-6}	0.2401
Molybdenum-99	2.693×10^{5}	1.25×10^{-6}	0.3366
Technetium-99m	2.358×10^{5}	1.25×10^{-6}	0.2948
Ruthenium-103	1.727×10^{5}	1.25×10^{-6}	0.2159
Ruthenium-105	9.789×10^{4}	1.25×10^{-6}	0.1224
Ruthenium-106	2,729	1.25×10^{-6}	0.03411
Rhodium-105	9.844×10^{4}	1.25×10^{-6}	0.1231
Antimony-127	2.172×10^{4}	1.25×10^{-6}	0.02715
Antimony-129	5.288×10^{4}	1.25×10^{-6}	0.06610
Tellurium-127	2.082×10^{4}	1.25×10^{-6}	0.02603
Tellurium-127m	2,134	1.25×10^{-6}	0.002668
Tellurium-129	5.366×10^{4}	1.25×10^{-6}	0.06708
Tellurium-129m	7,338	1.25×10^{-6}	0.009173
Tellurium-131	1.413×10^{5}	1.25×10^{-6}	0.1766
Tellurium-131m	2.028×10^{4}	1.25×10^{-6}	0.02535
Tellurium-132	2.174×10^{5}	1.25×10^{-6}	0.2718
Iodine-131	1.574×10^{5}	0.0125	1,968
Iodine-132	2.209×10^{5}	0.0125	2,761
Iodine-133	2.996×10^{5}	0.0125	3,745
Iodine-134	3.412×10^{5}	0.0125	4,265
Iodine-135	2.763×10^{5}	0.0125	3,454
Xenon-133	3.034×10^{5}	0.0500	1.517×10^{4}
Xenon-135	2.995×10^{5}	0.0500	1.498×10^{4}
Cesium-134	4,676	0.00250	11.69
Cesium-136	5,314	0.00250	13.29
Cesium-137	8,319	0.00250	20.80
Barium-139	2.833×10^{5}	1.25×10^{-6}	0.3541
Barium-140	2.674×10^{5}	1.25×10^{-6}	0.3343
Lanthanum-140	2.698×10^{5}	1.25×10^{-6}	0.3373
Lanthanum-141	2.658×10^{5}	1.25×10^{-6}	0.3323
Lanthanum-142	2.461×10^{5}	1.25×10^{-6}	0.3076
Cerium-141	2.492×10^{5}	1.25×10^{-6}	0.3115

Radioisotope	HEU Driver Fuel Activity (curies)	Release Fraction	Environmental Release (curies)
Cerium-143	2.467×10^{5}	1.25×10^{-6}	0.3084
Cerium-144	1.277×10^{5}	1.25×10^{-6}	0.1596
Praseodymium-143	2.437×10^{5}	1.25×10^{-6}	0.3046
Neodymium-147	1.098×10^{5}	1.25×10^{-6}	0.1373
Neodymium-237	0.02577	1.25×10^{-6}	3.221×10^{-8}
Neptunium-239	2.406×10^{6}	1.25×10^{-6}	3.008
Plutonium-238	57.38	1.25×10^{-6}	7.173×10^{-5}
Plutonium-239	68.66	1.25×10^{-6}	8.583×10^{-5}
Plutonium-240	10.45	1.25×10^{-6}	1.306×10^{-5}
Plutonium-241	132.2	1.25×10^{-6}	1.653×10^{-4}
Americium-241	0.08854	1.25×10^{-6}	1.107×10^{-7}
Curium-242	2.844	1.25×10^{-6}	3.555×10^{-6}
Curium-244	9.215×10^{-4}	1.25×10^{-6}	1.152×10^{-9}

Key: HEU, highly enriched uranium.
Source: Nielsen 1999; Wootan 1999.

Each neptunium-237 target will contain 333 grams (11.7 ounces) of plutonium-238 (5,000 grams [176 ounces] per year divided by 15 targets per year). The release fractions are assumed to be the same as were used for the driver fuel assemblies. The radioisotope inventory, release fractions, and resulting environmental release for the neptunium-237 target assembly are provided in **Table I–21**.

Table I–21 Neptunium-237 Target Assembly Source Term

Radioisotope	Neptunium-237 Target Normalized to 1 Gram of Plutonium-238 Activity (curies)	Maximum Target Activity of 333 Grams of Plutonium-238 (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3	0.00241	0.803	0.0500	0.0401
Krypton-85	0.0202	6.73	0.0500	0.336
Krypton-85m	5.30	1,760	0.0500	88.2
Krypton-87	8.83	2,940	0.0500	147
Krypton-88	12.4	4,130	0.0500	206
Rubidium-86	0.00762	2.54	1.25×10^{-6}	3.17×10^{-6}
Strontium-89	9.63	3,210	1.25×10^{-6}	0.00401
Strontium-90	0.127	42.3	1.25×10^{-6}	5.29×10^{-5}
Strontium-91	23.4	7,790	1.25×10^{-6}	0.00974
Strontium-92	28.4	9,460	1.25×10^{-6}	0.0118
Yttrium-90	0.128	42.6	1.25×10^{-6}	5.33×10^{-5}
Yttrium-91	13.2	4,400	1.25×10^{-6}	0.00549
Yttrium-92	28.7	9,560	1.25×10^{-6}	0.0119
Yttrium-93	37.2	1.24×10^{4}	1.25×10^{-6}	0.0155
Zirconium-95	24.7	8,230	1.25×10^{-6}	0.0103
Zirconium-97	51.0	1.70×10^{4}	1.25×10^{-6}	0.0212
Niobium-95	16.8	5,590	1.25×10^{-6}	0.00699
Molybdenum-99	56.3	1.87×10^{4}	1.25×10^{-6}	0.0234
Technetium-99m	50.1	1.67×10^{4}	1.25×10^{-6}	0.0209
Ruthenium-103	42.5	1.42×10^{4}	1.25×10^{-6}	0.0177
Ruthenium-105	51.7	1.72×10^{4}	1.25×10^{-6}	0.0215
Ruthenium-106	6.41	2,130	1.25×10^{-6}	0.00267
Rhodium-105	41.1	1.37×10^{4}	1.25×10^{-6}	0.0171
Antimony-127	4.44	1,480	1.25×10^{-6}	0.00185
Antimony-129	13.5	4,500	1.25×10^{-6}	0.00562
Tellurium-127	4.18	1,390	1.25×10^{-6}	0.00174
Tellurium-127m	0.243	80.9	1.25×10^{-6}	1.01×10^{-4}
Tellurium-129	12.9	4,300	1.25×10^{-6}	0.00537
Tellurium-129m	1.39	463	1.25×10^{-6}	5.79×10^{-4}
Tellurium-131	30.9	1.03×10^{4}	1.25×10^{-6}	0.0129
Tellurium-131m	5.96	1,980	1.25×10^{-6}	0.00248
Tellurium-132	47.1	1.57×10^{4}	1.25×10^{-6}	0.0196
Iodine-131	32.5	1.08×10^{4}	0.0125	135
Iodine-132	48.7	1.62×10^{4}	0.0125	203
Iodine-133	65.0	2.16×10^{4}	0.0125	271
Iodine-134	69.0	2.30×10^{4}	0.0125	287
Iodine-135	60.8	2.02×10^{4}	0.0125	253
Xenon-133	61.3	2.04×10^{4}	0.0500	1,020
Xenon-135	7.69	2,560	0.0500	128
Cesium-134	0.159	52.9	0.00250	0.132
Cesium-136	0.920	306	0.00250	0.766
Cesium-137	0.375	125	0.00250	0.312
Barium-139	54.1	1.80×10^{4}	1.25×10^{-6}	0.0225
Barium-140	45.1	1.50×10^{4}	1.25×10^{-6}	0.0188
Lanthanum-140	44.5	1.48×10^{4}	1.25×10^{-6}	0.0185
Lanthanum-141	51.3	1.71×10^{4}	1.25×10^{-6}	0.0214

Radioisotope	Neptunium-237 Target Normalized to 1 Gram of Plutonium-238 Activity (curies)	Maximum Target Activity of 333 Grams of Plutonium-238 (curies)	Release Fraction	Environmental Release (curies)
Lanthanum-142	47.6	1.59×10^{4}	1.25×10^{-6}	0.0198
Praseodymium-143	35.6	1.19×10^{4}	1.25×10^{-6}	0.0148
Cerium-141	35.0	1.17×10^{4}	1.25×10^{-6}	0.0146
Cerium-143	42.5	1.42×10^{4}	1.25×10^{-6}	0.0177
Cerium-144	7.13	2,370	1.25×10^{-6}	0.00297
Neodymium-147	17.1	5,690	1.25×10^{-6}	0.00712
Neptunium-237	0.00360	1.20	1.25×10^{-6}	1.50×10^{-6}
Neptunium-239	16.8	5,590	1.25×10^{-6}	0.00699
Plutonium-238	16.9	5,630	1.25×10^{-6}	0.00703
Plutonium-239	0.00921	3.07	1.25×10^{-6}	3.83×10^{-6}
Plutonium-240	0.00393	1.31	1.25×10^{-6}	1.64×10^{-6}
Plutonium-241	0.853	284	1.25×10^{-6}	3.55×10^{-4}
Americium-241	0.00	0.00	1.25×10^{-6}	0.00
Curium-244	0.00	0.00	1.25×10^{-6}	0.00

Source: Nielsen 1999; Schnitzler 1999.

The bottom-loading transfer cask would be used to transfer the Long-Term Irradiation Vehicle medical isotope targets. Except for the xenon-127 product target, which has a gaseous target material (xenon-126), the chemical and physical forms of the target material have not been decided upon. The release mechanism is assumed to be a breaking or tearing of the cladding tube due to the impact of a heavy object. The recommended bounding airborne release fraction for powder in a can which is broken or torn open due to the impact of a heavy object is 0.001; the respirable fraction, 0.1 (DOE 1994a). This gives a net release fraction of 1.0×10^{-4} for nongases. The release fraction of gases is assumed to be 1.0. The Long-Term Irradiation Vehicle targets were screened using these release fractions, and it was determined that the actinium-227 product target would result in the maximum consequences. The complete radioisotope inventory, release fraction, and resulting environmental release are presented in **Table I–22**. Although the entire radioisotope content of the actinium-227 product target is presented, 98.6 percent of the consequences are attributable to actinium-227 and thorium-228. Over 99.9 percent of the consequences are attributable to six radioisotopes (actinium-227, radium-223, radium-224, radium-226, thorium-227, and thorium-228).

Table I–22 Actinium-227 Product Target Assembly Source Term

Radioisotope	Target Activity (curies)	Release Fraction	Environmental Release (curies)
Actinium-227	34.0	1.00×10^{-4}	0.00340
Actinium-228	56.1	1.00×10^{-4}	0.00561
Actinium-229	6.04×10^{-9}	1.00×10^{-4}	6.04×10^{-13}
Radium-226	14.3	1.00×10^{-4}	0.00143
Radium-227	4.23×10^{-7}	1.00×10^{-4}	4.23×10^{-11}
Radium-228	0.00101	1.00×10^{-4}	1.01×10^{-7}
Radium-229	5.00×10^{-14}	1.00×10^{-4}	5.00×10^{-18}
Thorium-227	24.8	1.00×10^{-4}	0.00248
Thorium-228	42.1	1.00×10^{-4}	0.00421
Thorium-229	8.63×10^{-4}	1.00×10^{-4}	8.63×10^{-8}
Actinium-225	3.72×10^{-4}	1.00×10^{-4}	3.72×10^{-8}
Astatine-217	3.72×10^{-4}	1.00×10^{-4}	3.72×10^{-8}
Bismuth-210	0.109	1.00×10^{-4}	1.09×10^{-5}
Bismuth-211	19.6	1.00×10^{-4}	0.00196
Bismuth-212	24.6	1.00×10^{-4}	0.00246
Bismuth-213	3.71×10^{-4}	1.00×10^{-4}	3.71×10^{-8}
Bismuth-214	14.3	1.00×10^{-4}	0.00143
Francium-221	3.72×10^{-4}	1.00×10^{-4}	3.72×10^{-8}
Francium-223	1.40×10^{-5}	1.00×10^{-4}	1.40×10^{-9}
Lead-209	3.69×10^{-4}	1.00×10^{-4}	3.69×10^{-8}
Lead-210	0.118	1.00×10^{-4}	1.18×10^{-5}
Lead-211	19.6	1.00×10^{-4}	0.00196
Lead-212	38.4	1.00×10^{-4}	0.00384
Lead-214	14.3	1.00×10^{-4}	0.00143
Polonium-210	0.106	1.00×10^{-4}	1.06×10^{-5}
Polonium-211	0.0535	1.00×10^{-4}	5.35×10^{-6}
Polonium-212	24.6	1.00×10^{-4}	0.00246
Polonium-213	3.63×10^{-4}	1.00×10^{-4}	3.63×10^{-8}
Polonium-214	14.3	1.00×10^{-4}	0.00143
Polonium-215	19.6	1.00×10^{-4}	0.00196
Polonium-216	38.8	1.00×10^{-4}	0.00388
Polonium-218	14.3	1.00×10^{-4}	0.00143
Radium-223	19.6	1.00×10^{-4}	0.00196
Radium-224	38.8	1.00×10^{-4}	0.00388
Radium-225	5.46×10^{-4}	1.00×10^{-4}	5.46×10^{-8}
Radon-217	4.46×10^{-8}	1.00	4.46×10^{-8}
Radon-219	19.6	1.00	19.6
Radon-220	38.8	1.00	38.8
Radon-222	14.3	1.00	14.3
Thallium-207	19.6	1.00×10^{-4}	0.00196
Thallium-208	8.83	1.00×10^{-4}	8.83×10^{-4}
Thallium-209	8.16×10^{-6}	1.00×10^{-4}	8.16×10^{-10}

Source: Nielsen 1999; BWHC 1999.

I.1.1.4.2 FFTF Standby

The limiting accident for FFTF in its current standby condition is a primary heat transport system sodium spill. This accident has a frequency of about 1×10^{-4} per year. In its standby condition, the FFTF primary sodium is far less radioactive than under the proposed operating conditions. This is mainly because the fuel has been previously removed, but also because the radioactive sodium has had time to decay.

The current radioactive inventory in the primary heat transport sodium is provided **Table I–23**.

Table I–23 Current FFTF Primary Sodium Activity

Isotope	Activity (curies)
Hydrogen-3	54.9
Sodium-22	76.0
Cesium-137	0.0384
Plutonium-239	5.07×10^{-4}

Source: Nielsen 2000.

The size of the sodium spill is equivalent to that of the design-basis accident (393 kilograms [867 pounds], of which 24.4 kilograms [53.7 pounds] burn). Since the reactor is in a standby condition, no credit is taken for containment holdup of releases. Therefore, the release fraction is simply the ratio of the sodium burned to the total sodium inventory (i.e., $53.7/930{,}220 = 5.77\times10^{-5}$). The FFTF standby accident source term is provided in **Table I–24**.

Table I–24 FFTF Standby Accident Source Term

Isotope	Environmental Release (curies)
Hydrogen-3	0.00317
Sodium-22	0.00439
Cesium-137	2.22×10^{-6}
Plutonium-239	2.93×10^{-8}

Source: Calculated results.

It should be noted that the radioactive isotopes are continuously reduced by radioactive decay. Examination of the current inventories and dose conversion factors for these isotopes reveals that almost the entire dose would be attributable to plutonium-239 and sodium-22. Plutonium-239 has an extremely long half-life (24,000 years) and therefore its rather small decay would have little effect on consequences for quite some time. Sodium-22, however, has a fairly short half-life (2.6 years), and its decay would have a significant effect on the dose. For instance, after 35 years, only 21 percent of the original dose level would remain.

I.1.1.4.3 FFTF Deactivation

The limiting deactivation accident was determined from a review of the *Environmental Assessment, Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington* (DOE 1995). The bounding accident is a sodium spill during the transfer of the primary sodium to a treatment tank. A 9.1-kilogram (20-pound) spill of primary sodium outside the containment is assumed. The release fractions are 100 percent for noble gases and 1 percent for nongases.

The resulting source term, based on current primary sodium radioactivity is presented in **Table I–25**. As noted in the standby accident, the primary sodium radioactivity is continuously being reduced by radioactive decay.

Table I–25 FFTF Deactivation Accident Source Term

Isotope	Environmental Release (curies)
Hydrogen-3	0.00118
Sodium-22	1.63×10^{-5}
Cesium-137	8.26×10^{-9}
Plutonium-239	1.09×10^{-10}

Source: Calculated results.

The environmental analysis states that the accident frequency is greater than 0.01. For this NI PEIS, the accident frequency is conservatively chosen to be 0.10. This frequency is the probability of a sodium spill during the sodium transfer process. It is a frequency per event rather than per year.

I.1.1.4.4 Meteorological Data

Meteorological characteristics of the FFTF site are described by 1 year of hourly windspeed, atmospheric stability, and rainfall recorded at the Hanford 400 Area.

I.1.1.4.5 Population Data

The population distribution surrounding FFTF is based on the 1990 census (DOC 1992). State and county population estimates were examined to extrapolate the 1990 data to the year 2020.

I.1.1.4.6 Evacuation Information

In the event of an accident, DOE would implement site emergency plans and procedures that include restricting site access, patrolling onsite roads, and relocating members of the public. These actions would significantly reduce the consequences to onsite individuals. DOE sites also coordinate with offsite agencies in the event of an emergency. However, no relocation or evacuation of the offsite population was assumed for FFTF accident analyses. It was assumed that interdiction and condemnation of contaminated crops and foods were implemented based on EPA Protective Action Guides.

I.1.1.5 Low-Energy Accelerator

A spectrum of potential accidents at a low-energy accelerator used for the production of medical, industrial, and research and development isotopes was investigated. The accidents with the greatest potential for onsite and offsite consequences were evaluated in detail.

I.1.1.5.1 Design-Basis Accident

The limiting design-basis accident at the low-energy accelerator was determined to be a target assembly handling accident with an estimated probability of 1.0×10^{-4} per year (TechSource 2000).

The accident is assumed to occur one day after the beam is shut off. The medical, industrial, or research and development target is assumed to be damaged from mishandling. One hundred percent of the volatile fission products is assumed to be released from the target into the building. One percent of all the nonvolatile radioisotopes are released into the building. Fifty percent of the released radioisotopes, except noble gases, are assumed to plateout in the building. The radioisotopes which do not plateout are released to the environment through two stages of high-efficiency particulate air (HEPA) filters with a 99.95 percent efficiency for each stage and an activated charcoal filter with an assumed 99 percent iodine removal efficiency.

These assumptions result in a release fraction of 1.25×10^{-9} ($0.01 \times 0.5 \times 0.0005 \times 0.0005$) for the nonvolatile radioisotopes, 0.005 ($1 \times 0.5 \times 0.01$) for iodine, and 1.0 for noble gases. The likely medical, industrial, and research and development targets were screened with these release fractions to determine which target would result in the highest consequences from the target-handling accident. The target with the highest consequence is the iodine-125 product target with an environmental release of 12.7 curies. The likely medical, industrial, and research and development target product inventories are provided in Section I.1.4.2.

I.1.1.5.2 Beyond-Design-Basis Accident

The beyond-design-basis accident for the low-energy accelerator is a severe earthquake with an estimated frequency of 1.0×10^{-5} per year (TechSource 2000).

The medical, industrial, or research and development target is assumed to be crushed. One hundred percent of the volatile fission products are assumed to be released from the target into the building. One percent of the nonvolatile radioisotopes are assumed to be released into the building. None of the noble gases, 50 percent of the iodine, and 90 percent of the other radioisotopes are assumed to plateout in the building. The HEPA and charcoal filters are assumed to be destroyed and ineffective.

These assumptions result in a release fraction of 0.001 (0.01×0.1) for the nonvolatile radioisotopes, 0.5 (1×0.5) for iodine, and 1.0 for noble gases. The likely medical, industrial, and research and development targets were screened with these release fractions to determine which target would result in the highest consequences from the severe earthquake accident. The target with the highest consequence is the actinium-227 product target with the source term presented in **Table I–26**.

Table I–26 Low-Energy Accelerator Beyond-Design-Basis Accident Source Term

Isotope	Target Product Inventory[a] (curies)	Release Fraction	Environmental Release (curies)
Actinium-227	3.40×10^{1}	1.0×10^{-3}	3.40×10^{-2}
Radium-223	1.96×10^{1}	1.0×10^{-3}	1.96×10^{-2}
Radium-224	3.88×10^{1}	1.0×10^{-3}	3.88×10^{-2}
Radium-226	1.43×10^{1}	1.0×10^{-3}	1.43×10^{-2}
Thorium-227	2.48×10^{1}	1.0×10^{-3}	2.48×10^{-2}
Thorium-228	4.21×10^{1}	1.0×10^{-3}	4.21×10^{-2}

a. Although the product target contains several other radioisotopes, these six radioisotopes contribute over 99.9 percent of the dose consequences.

I.1.1.5.3 Meteorological Data

The meteorological characteristics of the generic accelerator site are assumed to be the same as those for the generic CLWR site and are described in Section I.1.1.3.2.

I.1.1.5.4 Population Data

The population distribution surrounding the generic accelerator site is assumed to be the same as that for the generic CLWR site and is described in Section I.1.1.3.3.

I.1.1.5.5 Evacuation Information

In the event of an accident, DOE would implement site emergency plans and procedures that include restricting site access, patrolling onsite roads, and relocating members of the public. These actions would significantly reduce the consequences to onsite individuals. DOE sites also coordinate with offsite agencies in the event

of an emergency. However, no relocation or evacuation of the offsite population was assumed for FFTF accident analyses. It was assumed that interdiction and condemnation of contaminated crops and foods were implemented based on EPA Protective Action Guides.

I.1.1.6 High-Energy Accelerator

A spectrum of potential accidents at a high-energy accelerator used for the production of plutonium-238 was investigated. The accidents with the greatest potential for onsite and offsite consequences were evaluated in detail. The meteorological data, population data, and evacuation information for the high-energy accelerator analysis are the same as those used for the low-energy accelerator analysis.

I.1.1.6.1 Design-Basis Accident

The limiting design-basis accident for the high-energy proton accelerator was determined to be a target assembly handling accident with an estimated probability of 1.0×10^{-4} per year (TechSource 2000).

The target is exposed to the beam for an estimated 99 days. The accident is assumed to occur after the target has been exposed to the beam for the full 99 days and 1 day after the beam is shut off. The target assembly is postulated to be dropped and partially crushed as it is being moved from the beam location to a cooled storage well. Without cooling, the assembly is estimated to begin to melt in 70 minutes. It would take about 1 to 4 hours to retrieve a target assembly and place it in a cooled storage well. It is assumed that 2 hours pass before retrieving the assembly and that the target has begun to melt.

Secondary neutrons produce a number of fissions within the uranium-238 target and the neptunium-237 blanket. The target and blanket generate 2.464 and 0.164 megawatts, respectively, from fission. The quantity of fission products produced in the target and blanket assembly is equivalent to that of a low-power reactor operating at a power level of 2.63 megawatts for 99 days.

In addition to fission products, high-energy protons striking the target also produce spallation products that results in isotopes from several to many mass units lower than the original target nucleus. The heat generated by spallation products is estimated to be 3.32 megawatts (2.604 and 0.719 megawatts in the target and blanket, respectively). The resulting spallation product isotopes were not directly calculated. An estimate was made by noting that approximately one isotope chain results per spall, whereas two result per fission. Therefore, it was assumed that there are one-half as many isotopic chains for the spallation process and that these chains are also comparable to those found in nuclear reactors. Hence, the radionuclides generated by spallation were characterized as having the same composition and quantities as fission products from a reactor operating at a power level of one-half of 3.32, or 1.66 megawatts, for a period of 99 days. This is considered to be conservative since the fraction of spallation products having masses comparable to the more volatile and hazardous fission products such as iodine are estimated to be smaller than the fission product yield of these same isotopes.

The total activity in the target and blanket assembly consists of the target and blanket materials, plutonium-238 and other isotopes produced (e.g., beryllium-7), fission products, and spallation products.

As a result of the accident, 1 percent of all the radioisotopes is assumed to be released into the building. Fifty percent of the released radioisotopes, except noble gases, is assumed to plate out or deposit within the building. The radioisotopes that do not plate out are released to the environment after passing through two stages of HEPA filters with a 99.95 percent efficiency for each stage, and a charcoal filter with an iodine removal efficiency of 99 percent. This results in release fractions of 0.01 for the noble gases, 5×10^{-5} ($0.01 \times 0.5 \times 0.01$) for the iodines, and 1.25×10^{-9} ($0.01 \times 0.5 \times 0.0005 \times 0.0005$) for the nonvolatile radioisotopes. The release

is elevated from a 30-meter-high (98-foot-high) stack. The source term for the design-basis accident is presented in **Table I–27**.

Table I–27 Accelerator Design-Basis Accident Source Term

Isotope	Activity in Target and Blanket (curies)	Activity from Fission Products (curies)	Activity from Spallation Products (curies)	Total Activity (curies)	Release Fraction	Environmental Release (curies)
Beryllium-7	3.60×10^{2}	-	-	3.60×10^{2}	1.25×10^{-9}	4.50×10^{-7}
Cobalt-58	–	4.15×10^{2}	2.62×10^{2}	6.77×10^{2}	1.25×10^{-9}	8.47×10^{-7}
Cobalt-60	–	7.78×10^{1}	4.91×10^{1}	1.27×10^{2}	1.25×10^{-9}	1.59×10^{-7}
Krypton-85	–	7.39×10^{1}	4.66×10^{1}	1.21×10^{2}	1.00×10^{-2}	1.21
Krypton-85m	–	2.41×10^{4}	1.52×10^{4}	3.94×10^{4}	1.00×10^{-2}	3.94×10^{2}
Krypton-87	–	4.41×10^{4}	2.78×10^{4}	7.19×10^{4}	1.00×10^{-2}	7.19×10^{2}
Krypton-88	–	5.97×10^{4}	3.77×10^{4}	9.73×10^{4}	1.00×10^{-2}	9.73×10^{2}
Rubidium-86	–	3.83×10^{1}	2.42×10^{1}	6.25×10^{1}	1.25×10^{-9}	7.81×10^{-8}
Strontium-89	–	5.48×10^{4}	3.46×10^{4}	8.94×10^{4}	1.25×10^{-9}	1.12×10^{-4}
Strontium-90	–	5.60×10^{2}	3.53×10^{2}	9.13×10^{2}	1.25×10^{-9}	1.14×10^{-6}
Strontium-91	–	9.64×10^{4}	6.08×10^{4}	1.57×10^{5}	1.25×10^{-9}	1.96×10^{-4}
Strontium-92	–	1.00×10^{5}	6.32×10^{4}	1.63×10^{5}	1.25×10^{-9}	2.04×10^{-4}
Yttrium-90	–	4.33×10^{3}	2.73×10^{3}	7.07×10^{3}	1.25×10^{-9}	8.83×10^{-6}
Yttrium-91	–	6.26×10^{4}	3.95×10^{4}	1.02×10^{5}	1.25×10^{-9}	1.28×10^{-4}
Yttrium-92	–	1.00×10^{5}	6.32×10^{4}	1.63×10^{5}	1.25×10^{-9}	2.04×10^{-4}
Yttrium-93	–	1.13×10^{5}	7.15×10^{4}	1.85×10^{5}	1.25×10^{-9}	2.31×10^{-4}
Zirconium-95	–	7.46×10^{4}	4.71×10^{4}	1.22×10^{5}	1.25×10^{-9}	1.52×10^{-4}
Zirconium-97	–	1.20×10^{5}	7.59×10^{4}	1.96×10^{5}	1.25×10^{-9}	2.45×10^{-4}
Niobium-95	–	9.33×10^{4}	5.89×10^{4}	1.52×10^{5}	1.25×10^{-9}	1.90×10^{-4}
Molybdenum-99	–	1.27×10^{5}	8.03×10^{4}	2.07×10^{5}	1.25×10^{-9}	2.59×10^{-4}
Technetium-99m	–	1.09×10^{5}	6.91×10^{4}	1.79×10^{5}	1.25×10^{-9}	2.23×10^{-4}
Ruthenium-103	–	7.80×10^{4}	4.93×10^{4}	1.27×10^{5}	1.25×10^{-9}	1.59×10^{-4}
Ruthenium-105	–	6.15×10^{4}	3.88×10^{4}	1.00×10^{5}	1.25×10^{-9}	1.25×10^{-4}
Ruthenium-106	–	4.89×10^{3}	3.09×10^{3}	7.98×10^{3}	1.25×10^{-9}	9.97×10^{-6}
Rhodium-105	–	4.26×10^{4}	2.69×10^{4}	6.95×10^{4}	1.25×10^{-9}	8.69×10^{-5}
Antimony-127	–	5.80×10^{3}	3.66×10^{3}	9.47×10^{3}	1.25×10^{-9}	1.18×10^{-5}
Antimony-129	–	2.06×10^{4}	1.30×10^{4}	3.36×10^{4}	1.25×10^{-9}	4.20×10^{-5}
Tellurium-127	–	5.61×10^{3}	3.54×10^{3}	9.15×10^{3}	1.25×10^{-9}	1.14×10^{-5}
Tellurium-127m	–	3.50×10^{2}	2.21×10^{2}	5.71×10^{2}	1.25×10^{-9}	7.14×10^{-7}
Tellurium-129	–	1.93×10^{4}	1.22×10^{4}	3.14×10^{4}	1.25×10^{-9}	3.93×10^{-5}
Tellurium-129m	–	4.43×10^{3}	2.80×10^{3}	7.23×10^{3}	1.25×10^{-9}	9.04×10^{-6}
Tellurium-131m	–	9.71×10^{3}	6.13×10^{3}	1.58×10^{4}	1.25×10^{-9}	1.98×10^{-5}
Tellurium-132	–	9.71×10^{4}	6.13×10^{4}	1.58×10^{5}	1.25×10^{-9}	1.98×10^{-4}
Iodine-131	–	6.67×10^{4}	4.21×10^{4}	1.09×10^{5}	5.00×10^{-5}	5.44×10^{1}
Iodine-132	–	9.87×10^{4}	6.23×10^{4}	1.61×10^{5}	5.00×10^{-5}	8.05×10^{1}
Iodine-133	–	1.41×10^{5}	8.90×10^{4}	2.30×10^{5}	5.00×10^{-5}	1.15×10^{2}
Iodine-134	–	1.55×10^{5}	9.78×10^{4}	2.53×10^{5}	5.00×10^{-5}	1.26×10^{2}
Iodine-135	–	1.33×10^{5}	8.42×10^{4}	2.18×10^{5}	5.00×10^{-5}	1.09×10^{2}
Xenon-133	–	1.41×10^{5}	8.90×10^{4}	2.30×10^{5}	1.00×10^{-2}	2.30×10^{3}
Xenon-135	–	2.65×10^{4}	1.67×10^{4}	4.33×10^{4}	1.00×10^{-2}	4.33×10^{2}
Cesium-134	–	1.61×10^{3}	1.01×10^{3}	2.62×10^{3}	1.25×10^{-9}	3.27×10^{-6}
Cesium-136	–	2.73×10^{3}	1.72×10^{3}	4.45×10^{3}	1.25×10^{-9}	5.57×10^{-6}
Cesium-137	–	6.96×10^{2}	4.39×10^{2}	1.14×10^{3}	1.25×10^{-9}	1.42×10^{-6}
Barium-139	–	1.31×10^{5}	8.27×10^{4}	2.14×10^{5}	1.25×10^{-9}	2.67×10^{-4}

Isotope	Activity in Target and Blanket (curies)	Activity from Fission Products (curies)	Activity from Spallation Products (curies)	Total Activity (curies)	Release Fraction	Environmental Release (curies)
Barium-140	–	1.29×10^{5}	8.14×10^{4}	2.10×10^{5}	1.25×10^{-9}	2.63×10^{-4}
Lanthanum-140	–	1.33×10^{5}	8.37×10^{4}	2.16×10^{5}	1.25×10^{-9}	2.70×10^{-4}
Lanthanum-141	–	1.21×10^{5}	7.64×10^{4}	1.97×10^{5}	1.25×10^{-9}	2.47×10^{-4}
Lanthanum-142	–	1.17×10^{5}	7.40×10^{4}	1.91×10^{5}	1.25×10^{-9}	2.39×10^{-4}
Cerium-141	–	1.04×10^{5}	6.54×10^{4}	1.69×10^{5}	1.25×10^{-9}	2.11×10^{-4}
Cerium-143	–	1.14×10^{5}	7.20×10^{4}	1.86×10^{5}	1.25×10^{-9}	2.33×10^{-4}
Cerium-144	–	1.83×10^{4}	1.15×10^{4}	2.98×10^{4}	1.25×10^{-9}	3.73×10^{-5}
Praseodymium-143	–	1.12×10^{5}	7.06×10^{4}	1.82×10^{5}	1.25×10^{-9}	2.28×10^{-4}
Neodymium-147	–	5.02×10^{4}	3.17×10^{4}	8.18×10^{4}	1.25×10^{-9}	1.02×10^{-4}
Neptunium-237	4.93×10^{1}	–	–	4.93×10^{1}	1.25×10^{-9}	6.16×10^{-8}
Neptunium-238	1.41×10^{7}	–	–	1.41×10^{7}	1.25×10^{-9}	1.76×10^{-2}
Neptunium-239	3.01×10^{6}	1.35×10^{6}	8.51×10^{5}	5.21×10^{6}	1.25×10^{-9}	6.51×10^{-3}
Plutonium-238	3.40×10^{4}	1.04×10^{1}	6.58	3.40×10^{4}	1.25×10^{-9}	4.25×10^{-5}
Plutonium-239	2.69×10^{1}	2.33	1.47	3.07×10^{1}	1.25×10^{-9}	3.84×10^{-8}
Plutonium-240	–	2.95	1.86	4.81	1.25×10^{-9}	6.01×10^{-9}
Plutonium-241	–	5.16×10^{2}	3.26×10^{2}	8.42×10^{2}	1.25×10^{-9}	1.05×10^{-6}
Americium-241	–	3.28×10^{-1}	2.07×10^{-1}	5.34×10^{-1}	1.25×10^{-9}	6.68×10^{-10}
Curium-242	–	3.33×10^{2}	2.10×10^{2}	5.43×10^{2}	1.25×10^{-9}	6.78×10^{-7}
Curium-244	–	7.58	4.79	1.24×10^{1}	1.25×10^{-9}	1.55×10^{-8}

I.1.1.6.2 Beyond-Design-Basis Accident

The beyond-design-basis accident for the high-energy accelerator is a severe earthquake with an estimated frequency of 1.0×10^{-5} per year (TechSource 2000). The earthquake is postulated to occur on the 99th day of beam irradiation, rupturing all cooling pipes serving the target assembly. Without cooling, the target assembly is estimated to begin to melt within about 30 minutes.

Five percent of the radioisotopes are assumed to be released into the building. The ventilation systems fail, resulting in a slow radioisotope transport from the building. Because of the slow transport, 90 percent of the radioisotopes (except for the noble gases) are assumed to plate out in the building. It is assumed that the earthquake has also destroyed the HEPA and charcoal filters as well as the stack. These assumptions result in a ground level release with release fractions of 0.05 (0.05 × 1) for the noble gases, and 0.005 (0.05 × 0.1) for the others. The source term for the beyond-design-basis accident is presented in **Table I–28**.

Table I–28 Accelerator Beyond-Design-Basis Accident Source Term

Isotope	Activity in Target and Blanket (curies)	Activity from Fission Products (curies)	Activity from Spallation Products (curies)	Total Activity (curies)	Release Fraction	Environmental Release (curies)
Beryllium-7	3.60×10^{2}	-	-	3.60×10^{2}	5.00×10^{-3}	1.80
Cobalt-58	-	4.15×10^{2}	2.62×10^{2}	6.77×10^{2}	5.00×10^{-3}	3.39
Cobalt-60	-	7.78×10^{1}	4.91×10^{1}	1.27×10^{2}	5.00×10^{-3}	6.34×10^{-1}
Krypton-85	-	7.39×10^{1}	4.66×10^{1}	1.21×10^{2}	5.00×10^{-2}	6.03
Krypton-85m	-	2.41×10^{4}	1.52×10^{4}	3.94×10^{4}	5.00×10^{-2}	1.97×10^{3}
Krypton-87	-	4.41×10^{4}	2.78×10^{4}	7.19×10^{4}	5.00×10^{-2}	3.60×10^{3}
Krypton-88	-	5.97×10^{4}	3.77×10^{4}	9.73×10^{4}	5.00×10^{-2}	4.87×10^{3}
Rubidium-86	-	3.83×10^{1}	2.42×10^{1}	6.25×10^{1}	5.00×10^{-3}	3.13×10^{-1}
Strontium-89	-	5.48×10^{4}	3.46×10^{4}	8.94×10^{4}	5.00×10^{-3}	4.47×10^{2}
Strontium-90	-	5.60×10^{2}	3.53×10^{2}	9.13×10^{2}	5.00×10^{-3}	4.56
Strontium-91	-	9.64×10^{4}	6.08×10^{4}	1.57×10^{5}	5.00×10^{-3}	7.86×10^{2}
Strontium-92	-	1.00×10^{5}	6.32×10^{4}	1.63×10^{5}	5.00×10^{-3}	8.17×10^{2}
Yttrium-90	-	4.33×10^{3}	2.73×10^{3}	7.07×10^{3}	5.00×10^{-3}	3.53×10^{1}
Yttrium-91	-	6.26×10^{4}	3.95×10^{4}	1.02×10^{5}	5.00×10^{-3}	5.11×10^{2}
Yttrium-92	-	1.00×10^{5}	6.32×10^{4}	1.63×10^{5}	5.00×10^{-3}	8.17×10^{2}
Yttrium-93	-	1.13×10^{5}	7.15×10^{4}	1.85×10^{5}	5.00×10^{-3}	9.24×10^{2}
Zirconium-95	-	7.46×10^{4}	4.71×10^{4}	1.22×10^{5}	5.00×10^{-3}	6.08×10^{2}
Zirconium-97	-	1.20×10^{5}	7.59×10^{4}	1.96×10^{5}	5.00×10^{-3}	9.81×10^{2}
Niobium-95	-	9.33×10^{4}	5.89×10^{4}	1.52×10^{5}	5.00×10^{-3}	7.61×10^{2}
Molybdenum-99	-	1.27×10^{5}	8.03×10^{4}	2.07×10^{5}	5.00×10^{-3}	1.04×10^{3}
Technetium-99m	-	1.09×10^{5}	6.91×10^{4}	1.79×10^{5}	5.00×10^{-3}	8.93×10^{2}
Ruthenium-103	-	7.80×10^{4}	4.93×10^{4}	1.27×10^{5}	5.00×10^{-3}	6.37×10^{2}
Ruthenium-105	-	6.15×10^{4}	3.88×10^{4}	1.00×10^{5}	5.00×10^{-3}	5.02×10^{2}
Ruthenium-106	-	4.89×10^{3}	3.09×10^{3}	7.98×10^{3}	5.00×10^{-3}	3.99×10^{1}
Rhodium-105	-	4.26×10^{4}	2.69×10^{4}	6.95×10^{4}	5.00×10^{-3}	3.48×10^{2}
Antimony-127	-	5.80×10^{3}	3.66×10^{3}	9.47×10^{3}	5.00×10^{-3}	4.73×10^{1}
Antimony-129	-	2.06×10^{4}	1.30×10^{4}	3.36×10^{4}	5.00×10^{-3}	1.68×10^{2}
Tellurium-127	-	5.61×10^{3}	3.54×10^{3}	9.15×10^{3}	5.00×10^{-3}	4.58×10^{1}
Tellurium-127m	-	3.50×10^{2}	2.21×10^{2}	5.71×10^{2}	5.00×10^{-3}	2.86
Tellurium-129	-	1.93×10^{4}	1.22×10^{4}	3.14×10^{4}	5.00×10^{-3}	1.57×10^{2}
Tellurium-129m	-	4.43×10^{3}	2.80×10^{3}	7.23×10^{3}	5.00×10^{-3}	3.62×10^{1}
Tellurium-131m	-	9.71×10^{3}	6.13×10^{3}	1.58×10^{4}	5.00×10^{-3}	7.92×10^{1}
Tellurium-132	-	9.71×10^{4}	6.13×10^{4}	1.58×10^{5}	5.00×10^{-3}	7.92×10^{2}
Iodine-131	-	6.67×10^{4}	4.21×10^{4}	1.09×10^{5}	5.00×10^{-3}	5.44×10^{2}
Iodine-132	-	9.87×10^{4}	6.23×10^{4}	1.61×10^{5}	5.00×10^{-3}	8.05×10^{2}
Iodine-133	-	1.41×10^{5}	8.90×10^{4}	2.30×10^{5}	5.00×10^{-3}	1.15×10^{3}
Iodine-134	-	1.55×10^{5}	9.78×10^{4}	2.53×10^{5}	5.00×10^{-3}	1.26×10^{3}
Iodine-135	-	1.33×10^{5}	8.42×10^{4}	2.18×10^{5}	5.00×10^{-3}	1.09×10^{3}
Xenon-133	-	1.41×10^{5}	8.90×10^{4}	2.30×10^{5}	5.00×10^{-2}	1.15×10^{4}
Xenon-135	-	2.65×10^{4}	1.67×10^{4}	4.33×10^{4}	5.00×10^{-2}	2.16×10^{3}
Cesium-134	-	1.61×10^{3}	1.01×10^{3}	2.62×10^{3}	5.00×10^{-3}	1.31×10^{1}
Cesium-136	-	2.73×10^{3}	1.72×10^{3}	4.45×10^{3}	5.00×10^{-3}	2.23×10^{1}
Cesium-137	-	6.96×10^{2}	4.39×10^{2}	1.14×10^{3}	5.00×10^{-3}	5.68
Barium-139	-	1.31×10^{5}	8.27×10^{4}	2.14×10^{5}	5.00×10^{-3}	1.07×10^{3}
Barium-140	-	1.29×10^{5}	8.14×10^{4}	2.10×10^{5}	5.00×10^{-3}	1.05×10^{3}

Isotope	Activity in Target and Blanket (curies)	Activity from Fission Products (curies)	Activity from Spallation Products (curies)	Total Activity (curies)	Release Fraction	Environmental Release (curies)
Lanthanum-140	-	1.33×10^{5}	8.37×10^{4}	2.16×10^{5}	5.00×10^{-3}	1.08×10^{3}
Lanthanum-141	-	1.21×10^{5}	7.64×10^{4}	1.97×10^{5}	5.00×10^{-3}	9.87×10^{2}
Lanthanum-142	-	1.17×10^{5}	7.40×10^{4}	1.91×10^{5}	5.00×10^{-3}	9.56×10^{2}
Cerium-141	-	1.04×10^{5}	6.54×10^{4}	1.69×10^{5}	5.00×10^{-3}	8.45×10^{2}
Cerium-143	-	1.14×10^{5}	7.20×10^{4}	1.86×10^{5}	5.00×10^{-3}	9.30×10^{2}
Cerium-144	-	1.83×10^{4}	1.15×10^{4}	2.98×10^{4}	5.00×10^{-3}	1.49×10^{2}
Praseodymium-143	-	1.12×10^{5}	7.06×10^{4}	1.82×10^{5}	5.00×10^{-3}	9.12×10^{2}
Neodymium-147	-	5.02×10^{4}	3.17×10^{4}	8.18×10^{4}	5.00×10^{-3}	4.09×10^{2}
Neptunium-237	4.93×10^{1}	-	-	4.93×10^{1}	5.00×10^{-3}	2.47×10^{-1}
Neptunium-238	1.41×10^{7}	-	-	1.41×10^{7}	5.00×10^{-3}	7.05×10^{4}
Neptunium-239	3.01×10^{6}	1.35×10^{6}	8.51×10^{5}	5.21×10^{6}	5.00×10^{-3}	2.61×10^{4}
Plutonium-238	3.40×10^{4}	1.04×10^{1}	6.58	3.40×10^{4}	5.00×10^{-3}	1.70×10^{2}
Plutonium-239	2.69×10^{1}	2.33	1.47	3.07×10^{1}	5.00×10^{-3}	1.54×10^{-1}
Plutonium-240	-	2.95	1.86	4.81	5.00×10^{-3}	2.40×10^{-2}
Plutonium-241	-	5.16×10^{2}	3.26×10^{2}	8.42×10^{2}	5.00×10^{-3}	4.21
Americium-241	-	3.28×10^{-1}	2.07×10^{-1}	5.34×10^{-1}	5.00×10^{-3}	2.67×10^{3}
Curium-242	-	3.33×10^{2}	2.10×10^{2}	5.43×10^{2}	5.00×10^{-3}	2.71
Curium-244	-	7.58	4.79	1.24×10^{1}	5.00×10^{-3}	6.18×10^{-2}

I.1.1.7 New Research Reactor

The new research reactor would produce a number of long- and short-lived isotopes for medical and industrial applications and 5 kilograms (11 pounds) of plutonium-238 per year for space power applications. The new research reactor would contain 48 neptunium-237 target assemblies, each assembly consisting of four neptunium-237 target rods. The maximum plutonium-238 produced in each target rod is 27.6 grams (0.97 ounce). The reactor would also contain eight medical and industrial target assemblies, each assembly consisting of two medical and industrial target rods. The reactor would also contain eight rabbit tubes for short-irradiation-time production of medical or industrial isotopes and civilian nuclear energy research and development. The rabbit tubes are outside the fuel region of the core, but still within an area with a rather high flux. Detailed descriptions of the new research reactor are provided in Appendix E. The meteorological data, population data, and evacuation information for the new research reactor analysis are the same as those used for the low-energy accelerator analysis.

I.1.1.7.1 Design-Basis Accident

A spectrum of accidents was reviewed according to the guidance provided in NUREG-1537 (NRC 1996). It was concluded that the maximum hypothetical accident is an accident whose potential consequences would exceed and bound all credible accidents. The accident scenario was assumed to represent the design-basis accident for the new research reactor.

Operational incidents leading to loss of coolant, loss of flow, loss of normal electrical power, and reactivity insertion would not result in any fuel damage. The built-in safety features of the new research reactor, such as elevation of the spent fuel pool system above the core, elevation of primary piping above the core, and antisyphon devices, would preclude loss of core cooling capability. The inherent large prompt negative fuel temperature coefficient of reactivity, would minimize the effect of accidental reactivity insertion. The reactivity insertion would cause a sudden increase in reactor power, leading to a higher fuel temperature which,

in turn, because of its large negative temperature coefficient of reactivity, would shut down the reactor. The design of the control rods would limit reactivity insertion below that which could cause any fuel failure.

The design-basis accident for the reactor is cladding failure of a single irradiated element (NRC 1996). The single fuel element could fail due to material deficiencies at any time during normal operation or while the reactor is shut down. Judging from experience with TRIGA (training, research, isotopes General Atomics) fuels, this type of failure is considered infrequent (a likelihood of 1 in 100 years) (UC-Davis 1999). For the new research reactor, it was assumed that the cladding of all fuel rods in an assembly (a maximum of 64 rods) would fail at any time during normal reactor operation. Further, it was assumed that this event would occur in an irradiated fuel assembly with high burnup. This accident was assumed without any consideration of mechanisms that could cause the failure of all cladding. The accident would cause the gaseous fission products and halogens collected in the fuel-clad gap to be released to the reactor pool. The likelihood of such an event was assumed to be 1 in 10,000 years.

The failed fuel assembly was assumed to have been operated at a power density 2.25 times that of the average. It was assumed that the assembly had a burnup of 5,157 megawatt days, which would occur at the end of a 10-year fuel cycle.

The fraction of fission gases released to the fuel-clad gap depends on the operating temperature of the fuel. Based on the calculated fuel temperature for the peak rod of less than 300 °C (572 °F), the fraction of volatile fission products that would escape the fuel material would be 1.5×10^{-5} (Simnad 1980; West, Simnad, and Copeland 1986). For this analysis, the fractional release was conservatively assumed to be 1×10^{-4}, which corresponds to an average operating fuel material temperature of 490 °C (914 °F).

One hundred percent of the noble gases and tritium gas collected in the fuel-clad gap would be released from the fuel assembly and subsequently enter the reactor room. Twenty-five percent of the halogens in the fuel-clad gap would be released from the fuel assembly, and 90 percent of the released halogens would be absorbed in the 9.1-meter-deep (30-foot deep) reactor pool before entering the reactor room. All the radioactive noble gases and halogens that were released to the reactor room are assumed to enter the environment through the reactor building exhaust stack after passing through an activated charcoal filter. The charcoal filter is assumed to remove 99 percent of the halogens (NRC 1978). These assumptions result in an overall release fraction of 1×10^{-4} for the noble gases and tritium gas and 2.5×10^{-8} ($10^{-4} \times 0.25 \times 0.1 \times 0.01$) for the halogens.

A neptunium-237 target assembly is assumed to be damaged along with the fuel assembly. The same release fractions are assumed for the neptunium-237 target as the fuel.

The release to the environment is assumed to occur over 1 hour without decay. This assumption is conservative, because the concentration of the fission products in the reactor room would activate the emergency ventilation system, thereby reducing the room air exchange rate and extending release duration, thus resulting in further decay of the short-lived isotopes.

The radioactive noble, tritium, and halogen gases that would be released to the environment from the design-basis accident scenario are provided in **Table I–29**.

I.1.1.7.2 Fuel- and Target-Handling Accidents

Fuel movements would occur once every 10 years when the whole core (68 fuel assemblies) would be replaced with fresh fuel assemblies. Neptunium-237 target movements would occur once a year. Each year, the irradiated target rods would be removed from the fuel assemblies, packaged in cans, and transferred to the

Table I–29 Design-Basis Accident Source Term

Isotope	Fuel Assembly Inventory (curies)	Neptunium-237 Target Inventory per Gram of Plutonium-238 (curies)	Neptunium-237 Target Assembly Inventory[a] (curies)	Environmental Release (curies)
Hydrogen-3	60.8	0.00241	0.266	0.00611
Krypton-83m	6,530	2.86	316	0.685
Krypton-85	1,440	0.0202	2.23	0.144
Krypton-85m	1.43×10^{4}	5.30	585	1.49
Krypton-87	2.90×10^{4}	8.83	975	3.00
Krypton-88	4.08×10^{4}	0.124	1,370	4.22
Xenon-131m	554	0.303	33.5	0.0587
Xenon-133	9.30×10^{4}	61.3	6,770	9.98
Xenon-133m	2,810	2.14	236	0.305
Xenon-135	5.16×10^{4}	7.69	849	5.24
Xenon-135m	1.76×10^{4}	15.4	1,700	1.93
Xenon-138	8.40×10^{4}	46.7	5,160	8.92
Bromine-82	122	0.0422	4.66	3.17×10^{-6}
Bromine-83	6,480	2.86	316	1.70×10^{-4}
Bromine-84	1.23×10^{4}	4.24	468	3.19×10^{-4}
Iodine-128	278	0.0782	8.63	7.17×10^{-6}
Iodine-130	247	0.273	30.1	6.93×10^{-6}
Iodine-131	4.22×10^{4}	32.5	3,590	0.00114
Iodine-132	6.22×10^{4}	48.7	5,380	0.00169
Iodine-133	9.27×10^{4}	65.0	7,180	0.00250
Iodine-134	1.05×10^{5}	69.0	7,620	0.00282
Iodine-135	8.76×10^{4}	60.8	6,710	0.00236

a. Contains 110.4 grams of plutonium-238 (four target rods of 27.6 grams of plutonium-238).
Source: Calculated results.

spent fuel pool for temporary cooling and storage. The medical and industrial isotope movements would occur more frequently depending on the isotope. The likelihood of a fuel assembly or target drop is estimated to be in the range of 0.01 to 0.0001 per year, or an unlikely event. For this analysis, the likelihood is estimated to be 0.01 per year.

The drop of a fuel assembly could lead to releases of radioactive fission gases. Since the fuel rods are protected by the assembly shroud, fuel damage would be minimal. It is assumed that the drop would damage one fuel rod, releasing the gaseous fission products and halogens to the reactor pool. It is also assumed that the earliest fuel movement would start about 24 hours after the reactor was shut down. Since handling activities would be performed under 3 meters (10 feet) of water, the halogens and gaseous fission products release fractions are assumed to be the same as those for the design-basis accident. The estimated radioactive material release from this accident is provided in **Table I–30**.

A neptunium-237 target assembly consists of four target rods, each containing approximately 27.6 grams (0.97 ounces) of plutonium-238. As these rods are not protected, a drop could lead to a breach of all four.

The target rods are made from neptunium oxides. The fission gas release fraction from the target material to the gap would be similar to that from uranium oxides. Fractional fission gas release was estimated using American National Standards Institute 5.4 (ANSI 1982) and the low-temperature release calculation method. Target rod temperature is not expected to be greater than that of the cladding temperature of an average-power-density fuel rod (approximately 80 °C [176 °F]). For an estimated target rod burnup of

Table I–30 Fuel-Handling Accident Source Term

Isotope	Fuel Rod Inventory (curies)	Environmental Release
Hydrogen-3	0.950	9.50×10^{-5}
Krypton-83m	0.413	4.13×10^{-5}
Krypton-85	22.5	0.00225
Krypton-85m	550	0.0550
Krypton-87	9.55×10^{-4}	9.55×10^{-8}
Krypton-88	1.81	1.81×10^{-4}
Xenon-131m	8.56	8.56×10^{-4}
Xenon-133	1,400	0.140
Xenon-133m	39.8	0.00398
Xenon-135	420	0.0420
Xenon-135m	17.8	0.00178
Bromine-82	1.19	2.98×10^{-8}
Bromine-83	0.108	2.70×10^{-9}
Iodine-130	3.06	7.65×10^{-8}
Iodine-131	614	1.54×10^{-5}
Iodine-132	802	2.01×10^{-5}
Iodine-133	670	1.68×10^{-5}
Iodine-134	4.00×10^{-5}	1.00×10^{-12}
Iodine-135	109	2.73×10^{-6}

706 megawatt days per metric ton of heavy metal, about 0.01 percent of both the long- and short-lived noble gases and halogen gases would be available for release.

As in the fuel-handling accident, 100 percent of the noble and tritium gases in the fuel-clad gap would be released to the environment through the reactor building exhaust system. This results in an overall release fraction of 1×10^{-4} for the noble gases and tritium. Twenty-five percent of the iodine in the fuel-clad gap would be released from the fuel assembly, and 90 percent of the released iodine would be absorbed in the reactor pool. The remaining iodine would be released to the environment through the Reactor Building exhaust system. The exhaust system charcoal filter is assumed to remove 99 percent of the iodine (NRC 1978). This results in an overall release fraction of 2.5×10^{-8} ($10^{-4} \times 0.25 \times 0.1 \times 0.01$) for the iodine.

These assumptions result in the source term shown in **Table I–31**.

Medical, industrial, and research and development isotope targets could also be damaged from a drop accident. Only targets which produce noble gases and halogens either as products or byproducts (including decay) need be considered for analysis. Since the fuel-handling activities are performed under 3 meters (10 feet) of water, these will be the isotopes that have releases to the environment. The iodine-125 product target consequences bound those of the other possible medical and industrial isotope targets.

The iodine-125 product target is assumed to release 100 percent of its inventory to the water. Interaction with the water removes 90 percent of the iodine. The building exhaust system charcoal filters then removes 99 percent of the iodine released from the water. This results in a release fraction of 0.001. The iodine-125 product target would contain approximately 2,530 curies of iodine-125. The estimated radioactive material release from this accident is 2.53 curies of iodine-125.

Table I–31 Neptunium-237 Target-Handling Accident Source Term

Isotope	Neptunium-237 Target Inventory per Gram of Plutonium-238 (curies)	Neptunium-237 Target Assembly Inventory[a] (curies)	Environmental Release (curies)
Hydrogen-3	0.00241	0.266	2.66×10^{-5}
Krypton-83m	2.86	316	0.0316
Krypton-85	0.0202	2.23	2.23×10^{-4}
Krypton-85m	5.30	585	0.0585
Krypton-87	8.83	975	0.0975
Krypton-88	12.4	1,370	0.137
Xenon-131m	0.303	33.5	0.00335
Xenon-133	61.3	6,770	0.677
Xenon-133m	2.14	236	0.0236
Xenon-135	7.69	849	0.0849
Xenon-135m	15.4	1,700	0.170
Xenon-138	46.7	5,160	0.516
Bromine-82	0.0422	4.66	1.16×10^{-7}
Bromine-83	2.86	316	7.89×10^{-6}
Bromine-84	4.24	468	1.17×10^{-5}
Iodine-128	0.0782	8.63	2.16×10^{-7}
Iodine-130	0.273	30.1	7.53×10^{-7}
Iodine-131	32.5	3,590	8.97×10^{-5}
Iodine-132	48.7	5,380	1.34×10^{-4}
Iodine-133	65.0	7,180	1.79×10^{-4}
Iodine-134	69.0	7,620	1.90×10^{-4}
Iodine-135	60.8	6,710	1.68×10^{-4}

a. Contains 110.4 grams of plutonium-238 (four target rods of 27.6 grams of plutonium-238).

I.1.1.7.3 Beyond-Design-Basis Accident

The beyond-design-basis accident assumes an earthquake with sufficient energy to cause structural and equipment failure. The likelihood of such an event was assumed to be the equal to the Reactor Building performance goal for a Performance Category 4 structure. The performance goal for the Reactor Building is 1×10^{-5}, a safety factor of 10 over the return period of 1 in 10,000 years for a Performance Category 4 structure (DOE 1994b). Performance Category 4 is the highest deterministic seismic design criteria for structures, systems, and components in accordance with DOE standards (DOE 1993, 1994b). A performance goal of 1×10^{-5} refers to the annual probability that a seismic event would cause damage to a component so that it could not perform its function. Therefore, an earthquake level with a return period of 1 in 100,000 years was assumed to initiate the beyond-design-basis accident. Since both the reactor pool and the spent fuel pool would be designed to withstand a higher-level earthquake than that for Performance Category 4, no failure of these pools was assumed. However, it was assumed that the equipment and systems that support these pools would fail. Further, it is assumed that the earthquake would initiate reactor scram (loss of power would cause the control rods to drop in the reactor core), damage the cooling pipe outside of the reactor pool, and possibly breach the reactor room confinement.

Since the accident would not result in a loss of reactor pool coolant below the level at which primary piping leaves the pool and the reactor shuts down, sufficient coolant would be available to keep the core covered for

at least 40 days after the accident. For this analysis, it was assumed that the fuel-handling crane above the pool would fall into the pool and damage the core tank and fuel rods inside the core. This assumption is conservative since the top of the core tank would be 0.61 meters (2 feet) above the top of the fuel assemblies acting as a chimney to enhance natural-convection core cooling during reactor shutdowns. In addition, the top of the active fuel is another foot below the top of the fuel assembly. Therefore, the crane would have to damage both the upper core barrel and the top of the fuel assemblies before it could damage the fuel. Nevertheless, it was assumed that the drop would cause fuel damage.

The drop was assumed to cause releases of all gaseous fission products and halogens through the pool water directly to the environment, bypassing the charcoal filter and the building exhaust stack. For the fuel and neptunium-237 targets, the release fractions are 1×10^{-4} for the noble gases and tritium and 2.5×10^{-6} for the halogens. For the medical isotope targets, the release fractions are 1.0 for noble gases and 0.1 for halogens.

The new research reactor core consists of 68 fuel assemblies with a total of 4,080 fuel rods; 48 neptunium-237 assemblies with 4 target rods each; 8 medical, industrial, and research and development target assemblies with 2 target rods each; and 8 rabbit tubes. For this analysis, the 8 medical, industrial, and research and development target assemblies are assumed to contain the xenon-127 product target. The rabbit tubes would contain 7 iodine-131 product targets and 1 iodine-125 product target. This core configuration results in the highest consequences from accidental releases.

This core configuration and these release fraction assumptions result in the source term presented in **Table I–32**.

Table I–32 Beyond-Design-Basis Earthquake Accident Source Term

Isotope	Fuel Core Inventory[a] (curies)	Medical, Industrial, and Research and Development Isotope Core Inventory[b] (curies)	Neptunium-237 Target Inventory per Gram of Plutonium-238[c] (curies)	Neptunium-237 Target Core Inventory[d] (curies)	Environmental Release (curies)
Hydrogen-3	1,790		0.00241	12.8	0.180
Krypton-85	4.60×10^{4}		0.0202	107	4.61
Krypton-85m	5.30×10^{5}		5.30	2.81×10^{4}	55.8
Krypton-87	7.27×10^{5}		8.83	4.68×10^{4}	77.4
Krypton-88	1.38×10^{6}		12.4	6.57×10^{4}	145
Iodine-125		2,530	0.0	0.0	253
Iodine-131	1.59×10^{6}	2,150	32.5	1.72×10^{5}	219
Iodine-132	2.35×10^{6}		48.7	2.58×10^{5}	6.52
Iodine-133	3.53×10^{6}		65.0	3.44×10^{5}	9.69
Iodine-134	3.03×10^{6}		69.0	3.66×10^{5}	8.49
Iodine-135	3.04×10^{6}		60.8	3.22×10^{5}	8.41
Xenon-127		116			116
Xenon-133	3.60×10^{6}		61.3	3.05×10^{5}	392
Xenon-135	2.72×10^{6}		7.69	4.07×10^{4}	276

a. Fuel inventory lists only those isotopes with an environmental release.
b. Medical, industrial, and research and development inventory lists only the isotopes with the highest environmental release.
c. Neptunium-237 inventory lists only those isotopes with an environmental release.
d. Based on a 5-kilogram-per-year plutonium-238 production rate.

I.1.1.7.4 Decontamination and Decommissioning Accidents

The decontamination and decommissioning activities would be performed according to a preestablished plan, known as the decommissioning plan. Activities would include decontamination and dismantling of reactor components, removal of spent nuclear fuel, cleaning and removal of the reactor pool and spent fuel pool water, decontamination and dismantling of equipment and structures, and preparation of the site for unrestricted use. These activities could potentially result in an accidental release of radioactive material. Radioactive releases could occur from improper cutting of activated components and equipment, dropping of a radioactively contaminated component, and from spills of contaminated liquids. The potential on- and offsite impacts of accidents would be expected to be less than, or within, the values estimated for occurrences during normal operations.

At this preconceptual research reactor design stage, the major areas with the greatest inventory of radioactivity would be the spent fuel pool and the components within the reactor pool and primary coolant system. The spent fuel pool would contain about 272 spent fuel assemblies, that is, four full core loads. The minimum decay times for each assembly would range between 5 and 30 years. The assumption is that spent fuel removal would begin 5 years after the last core was removed from the reactor. Once the fuel assemblies were removed from the core, the beryllium reflector and the reactor core tank would contain the highest radioactive inventory of tritium and cobalt-60 in the reactor pool area.

A spectrum of accidents was evaluated considering activities that would occur during decontamination and decommissioning of the research reactor and support facility. It was determined that two accidents had the greatest potential for onsite and offsite impacts: a drop of a spent nuclear fuel cask during fuel removal and an accidental vaporization of a small segment of the reactor core tank during dismantlement.

SPENT FUEL CASK DROP ACCIDENT

The lifting capability of the spent fuel pool crane would be limited to truck-sized spent nuclear fuel transportation casks that would be used to move the spent fuel to a central storage location. The cask would be loaded under water, the cask cover would be installed but not tightly sealed, and the cask raised above the water where it would be sprayed with demineralized water before it was put on the ground for decontamination and draining of pool water. The cask then would be sealed, backfilled with inert gas, and moved to be loaded onto the truck trailer bed. The maximum lift would be less than 9.1 meters (30 feet) above the pool floor, or less than 30 centimeters (1 foot) above the spent fuel pool building floor level.

A spent nuclear fuel cask was assumed to drop while it was stopped to be rinsed. The drop would not damage the cask or the spent fuel pool liner. The cask is designed to withstand a drop from 9.1 meters (30 feet) onto an unyielding surface without failure. The cask would not be lifted above 9.1 meters (30 feet) above the ground, and the drop over the spent fuel pool would hit the pool surface which provides 7.62 meters (25 feet) of water acting as a damper, reducing the impact velocity. Therefore, no damage to the spent fuel pool liner would be expected.

The fuel rods in the cask would be protected from damage not only by the cask, but also by the assembly shroud. However, for this analysis, it was assumed that one row of fuel in one assembly would fail and release the gaseous fission products from the fuel-clad gap. The fraction of fission gases released to the fuel-clad gap was conservatively assumed to be 1×10^{-4}. One hundred percent of the noble gases and tritium gas in the fuel-clad gap would be released through the pool to the reactor room. Twenty-five percent of the halogens in the fuel-clad gap would be released and 90 percent of the released halogens would be absorbed in the reactor pool before entering the reactor room. All the noble gases, tritium, and halogens that enter the reactor room would be released to the environment through the reactor building exhaust system after passing through an

activated charcoal filter. The charcoal filter was assumed to remove 99 percent of the halogens (NRC 1978). These assumptions result in a release fraction of 1×10^{-4} for noble gases and tritium and 2.5×10^{-8} for halogens.

The likelihood of such an accident was estimated to be less than 5 in 1 million, or 5×10^{-6} per year. This estimate was derived from a recent NRC technical study of spent fuel accident risk at decommissioning nuclear power plants (NRC 2000). Based on an assumption of 100 heavy-load cask lifts per year, the NRC estimated a cask drop mean frequency of 9.6×10^{-6} per year. Considering that the total number of spent fuel cask lifts at this facility would be less than 40, assuming that all the fuel would be shipped offsite in a year, the cask drop frequency would be less than 5×10^{-6} per year for that year.

For analysis, the frequency of this accident was assumed to be 5.0×10^{-6} per year. The source term for the spent fuel cask drop accident is presented in **Table I–33**.

Table I–33 Spent Fuel Cask Drop Accident Source Term

Isotopes	Fuel Assembly - Fission Gas and Halogen Inventory (curies)	Eight Fuel Rods - Fission Gas and Halogen Inventory (curies)	Release Fraction	Environmental Release (curies)
Hydrogen-3 (tritium)	2.12×10^{1}	2.65	1.00×10^{-4}	2.65×10^{-4}
Krypton-85	1.44×10^{3}	1.80×10^{2}	1.00×10^{-4}	1.80×10^{-2}
Iodine-129	1.86×10^{-3}	2.33×10^{-4}	2.50×10^{-8}	5.81×10^{-12}

REACTOR CORE TANK VAPORIZATION ACCIDENT

An accidental vaporization of a small segment of the reactor core tank during size reduction was assumed. The stainless steel reactor core tank would need to be cut into pieces in order to be transported offsite. The major activation product in the tank would be cobalt-60, with an inventory of 359 curies after a 5-year decay time. Plasma torches would most likely be used for the process. The cutting process would occur with strict radiological controls under a tent with proper ventilation to collect any vaporized particulates. The vaporized particulates would be passed through HEPA filters before exhausting to the environment.

For this analysis, it was assumed that the exhaust system would fail and that the torch would not shut down and would vaporize a small segment of the tank. It was assumed that the torch would burn through the wall of the tank creating a 6.25-square centimeter (1-square inch) hole in the wall. The frequency of this accident was assumed to be 1×10^{-4}. The source term for the reactor core tank vaporization accident is estimated to be 0.026 curies of cobalt-60 released directly to the environment.

I.1.2 Methodology for Estimating Irradiation Facility Accident Radiological Impacts

The MACCS2 computer code (Version 1.12) was used to estimate the consequences of the postulated accidents. A detailed description of the MACCS model is provided in NUREG/CR-4691 (Chanin et al. 1990). The enhancements incorporated in MACCS2 are described in the MACCS2 User's Guide (Chanin and Young 1997). Originally developed to model the radiological consequences of nuclear reactor accidents, this code has been used for the analysis of accidents for many EISs and other safety documentation, and is considered applicable to the analysis of accidents associated with the production of plutonium-238 and other proposed isotopes.

MACCS2 models the offsite consequences of an accident that releases a plume of radioactive materials to the atmosphere. Should such an accidental release occur, the radioactive gases and aerosols in the plume would be transported by the prevailing wind and dispersed into the atmosphere, and the population would be exposed

to radiation. The atmospheric dispersion is modeled on a polar-coordinate spatial grid centered on the facility and extending out to 80 kilometers (50 miles). The user specifies the number of radial divisions and their endpoint distances. The angular divisions used to define the spatial grid correspond to the 16 directions of the compass. MACCS2 generates the distribution of downwind doses at specified distances, as well as the distribution of population doses.

Radiological consequences may vary somewhat as a result of variations in the duration of release. For longer releases, there is a greater chance of plume meander (i.e., changes in flow attributable to variations in wind direction over the duration of release). MACCS2 models plume meander by increasing the lateral dispersion coefficient of the plume for longer release durations, thus lowering the dose. The other effect of longer release durations is involvement of a greater variety of meteorological conditions in a given release, which reduces the variance of the resulting dose distributions. This would tend to lower high-percentile doses, raise low-percentile doses, and have no effect on the mean dose.

The MACCS2 code was applied in a probabilistic manner using a weather bin–sampling technique. The weather bin–sampling method sorts weather sequences into categories and assigns a probability to each category according to the initial conditions (wind speed and stability class) and the occurrence of rain. Each of the sampled meteorological sequences was applied to each of the 16 sectors accounting for the frequency of occurrence of the wind blowing in that direction (i.e., site compass sector wind rose frequencies). Individual doses, as a function of distance and direction, were calculated for each of the meteorological sequence samples. The mean dose values of the sequences were generated for each of the 16 sectors. The highest of these dose values was used as the dose delivered to the maximally exposed offsite individual and the noninvolved worker. Population doses within 80 kilometers (50 miles) of each facility were also calculated.

In addition to short-term health effects of exposure to the plume passage, long-term effects were also modeled. The long-term health effects include direct exposure to contaminated ground and inhalation of resuspended materials, as well as indirect health effects of the consumption of contaminated food and water. Long-term protective measures such as decontamination, temporary relocation, contaminated crops, milk condemnation, and prohibition of farmland production are based on EPA Protection Action Guides.

For each potential accident, information is provided on accident consequences and frequencies to three types of receptors: (1) a noninvolved worker, (2) the maximally exposed individual, and (3) the offsite population. The first receptor, a noninvolved worker, is a hypothetical individual working on site but not involved in the proposed activity. The worker is assumed to be downwind at a point 640 meters (0.4 miles) from the accident. Although other distances closer to the accident could have been assumed, the calculations break down at distances of about 200 meters (656 feet) or less due to limitations in modeling of the effects of building wake and local terrain on dispersion of the released radioactive substances. A worker closer than 640 meters (2,100 feet) to the accident would generally receive a higher dose; a worker farther away, a lower dose. The second receptor, the maximally exposed individual, is a hypothetical individual assumed to be downwind at the site boundary or on a highway within the site boundary, whichever results in a higher dose. Exposures received by this individual are intended to represent the highest doses to a member of the public. The third receptor, the offsite population, is all members of the public within 80 kilometers (50 miles) of the accident location.

It is possible that an individual member of the public could be closer to a facility than either the site boundary or the nearest onsite highway. However, such individuals would be present only occasionally and for brief periods (a few hours or more). Hence, the annual probability that an individual would be close is relatively low, and the associated risk to that individual would be bounded by the site boundary or onsite highway maximally exposed individual risk.

For the CLWR analysis, a noninvolved worker was not evaluated for two reasons. First, the noninvolved worker was originally developed for large DOE sites, where several different facilities are under facility-specific control. The noninvolved worker is an individual not under specific facility control, but also not outside the site boundary. At a CLWR, however, the entire site is within the exclusion area and under the same control.

Second, each accident scenario has a warning time and a subsequent release time. The warning time is the time at which notification is given to offsite emergency response officials to initiate protective measures for the surrounding population. The release time is the time when the release to the environment begins. The minimum time between the warning time and the release time for this analysis is 20 minutes. Twenty minutes is enough time to evacuate onsite personnel. It is also conservatively assumed that an onsite emergency has not been declared prior to initiating offsite notification.

Consequences to involved workers are addressed in Section I.1.7.

All radiological impacts are calculated in terms of committed dose and associated health effects for an individual or exposed population. The radiation dose calculated is the total effective dose equivalent, which is the sum of the effective dose equivalent from the external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of rem for individuals and person-rem for a population. The impacts are further expressed as health risks, specifically in terms of latent cancer fatalities.

The health risks for a noninvolved worker and the maximally exposed offsite individual are expressed as the additional potential or likelihood of a latent cancer fatality. The health risk to the population is expressed as the increased number of latent cancer fatalities.

The probability coefficients for determining the likelihood of latent cancer fatality, given a dose, are taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991). For low doses or dose rates, respective probability coefficients of 0.0004 and 0.0005 fatal cancer per rem are applied for workers and the general public. For high doses received at a high rate, respective probability coefficients of 0.0008 and 0.001 fatal cancer per rem are applied for noninvolved workers and the public. These higher probability coefficients apply where doses are above 20 rad or dose rates are above 10 rad per hour.

Tritium releases were modeled as tritiated water vapor rather than elemental tritium. Tritiated water is more effectively absorbed by humans and therefore results in a much greater health hazard.

I.1.2.1 Uncertainties

The analyses of accidents are based on calculations relevant to hypothetical sequences of events and models of their effects. The models provide estimates of the frequencies, source terms, pathways for dispersion, exposures, and the effects on human health and the environment that are as realistic as possible within the scope of the analysis. In many cases, the scarcity of experience with the accidents postulated leads to uncertainty in the calculation of their consequences and frequencies. This fact has prompted the use of models or input values that yield conservative estimates of consequence and frequency. All alternatives have been evaluated using uniform methods and data, allowing for a fair comparison of all alternatives.

Of particular interest are the uncertainties in the estimate of cancer deaths from exposure to radioactive materials. The numerical values of the health risk estimates used in this NI PEIS were obtained by the practice of linear extrapolation from the nominal risk estimate for lifetime total cancer mortality resulting from

exposures at 10 rad. Other methods of extrapolation to the low-dose region could yield higher or lower estimates of cancer deaths. Studies of human populations exposed at low doses are inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiological observation, and the possibility of no risk or even health benefits (hormesis effects) cannot be excluded. Because the health risk estimators are multiplied by conservatively calculated radiological doses to predict fatal cancer risks, the fatal cancer values presented in this NI PEIS are expected to be overestimates.

For the purposes of presentation in this NI PEIS, the impacts calculated from the linear model are treated as an upper-bound case, consistent with the widely used methodologies for quantifying radiogenic health impacts. This does not imply that health effects are expected. Moreover, in cases where the upper-bound estimators predict a number of latent cancer deaths that is greater than 1, this does not imply that the latent cancer death can be determined for a specific individual.

I.1.3 Irradiation Facility Accident Consequences and Risks

The irradiation facility accident source term data presented in Sections I.1.1.2–I.1.1.7 were evaluated using the MACCS2 accident analysis computer code. **Tables I–34** through **I–41** summarize the consequences and risks of the accidents, with mean meteorological conditions, for the maximally exposed individual, the offsite population within an 80-kilometer (50-mile) radius of the facility, and a noninvolved worker 640 meters (2,100 feet) from the release point. As explained in Section I.1.2, noninvolved worker consequences were not evaluated for the CLWR accidents.

Table I–34 presents ATR accident consequences and risks for three possible plutonium-238 production rates: 0, 3, and 5 kilograms (0, 6.6, and 11 pounds) per year.

Table I–35 presents HFIR accident consequences and risks for two possible plutonium-238 production rates: 0 and 2 kilograms (0 and 4.4 pounds) per year.

Table I–36 presents CLWR accident consequences and risks for two possible plutonium-238 production rates: 0 and 5 kilograms (0 and11 pounds) per year.

Severe-accident scenarios that postulate large, abrupt releases could result in early fatalities if the radiation dose were sufficiently high. For the irradiation facilities analyzed, early fatalities are postulated only for the early containment failure and containment bypass event at the generic CLWR.

Table I–37 presents CLWR-estimated early fatalities and associated risks for two possible plutonium-238 production rates: 0 and 5 kilograms (0 and 11 pounds) per year.

Table I–38 presents FFTF accident consequences and risks for simultaneous medical, industrial, research and development, and plutonium-238 production for both mixed oxide and highly enriched uranium fuels.

Table I–39 presents accelerator accident consequences and risks for medical, industrial, research and development, and plutonium-238 isotope production.

Table I–40 presents new research reactor accident consequences and risks for the simultaneous medical, industrial, research and development, and plutonium-238 isotope production.

Table I–41 presents new research reactor decontamination and decommissioning accident consequences and risks.

Table I–34 ATR Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	0.465	2.33×10^{-4}	2.33×10^{-8}	5.11×10^{4}	25.5	0.00255	5.15	0.00206	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production (1×10^{-4})	0.549	2.75×10^{-4}	2.75×10^{-8}	5.15×10^{4}	25.7	0.00257	6.52	0.00261	2.61×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production (1×10^{-4})	0.604	3.02×10^{-4}	3.02×10^{-8}	5.17×10^{4}	25.9	0.00259	7.61	0.00304	3.04×10^{-7}
Target-handling with 0 kg/yr plutonium-238 production (0.001)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Target-handling with 3 kg/yr plutonium-238 production (0.001)	1.23×10^{-4}	6.15×10^{-8}	6.15×10^{-11}	0.0786	3.93×10^{-5}	3.93×10^{-8}	0.00195	7.80×10^{-7}	7.80×10^{-10}
Target-handling with 5 kg/yr plutonium-238 production (0.001)	2.05×10^{-4}	1.03×10^{-7}	1.03×10^{-10}	0.128	6.41×10^{-5}	6.41×10^{-8}	0.00324	1.30×10^{-6}	1.30×10^{-9}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Key: ATR, Advanced Test Reactor; kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–35 HFIR Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Large-break LOCA with 0 kg/yr plutonium-238 production (1×10^{-4})	2.41	0.00121	1.21×10^{-7}	2,990	1.49	1.49×10^{-4}	17.2	0.00688	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production (1×10^{-4})	2.41	0.00121	1.21×10^{-7}	3,000	1.50	1.50×10^{-4}	17.2	0.00688	6.88×10^{-7}
Target-handling with 0 kg/yr plutonium-238 production (0.001)	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Target-handling with 2 kg/yr plutonium-238 production (0.001)	4.96×10^{-4}	2.48×10^{-7}	2.48×10^{-10}	0.335	1.68×10^{-4}	1.68×10^{-7}	0.00245	9.80×10^{-7}	9.80×10^{-10}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Key: HFIR, High Flux Isotope Reactor; kg/yr, kilograms per year; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–36 Generic CLWR Accident Consequences and Risks

Accident (Frequency)	Annual Plutonium-238 Production (kilograms per year)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)		
		Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]
Containment early failure (7.92×10^{-8})	0	3350	1.00[e]	7.92×10^{-8}[f]	1.80×10^{6}	1,250	9.89×10^{-5}
	5	3670	1.00[e]	7.92×10^{-8}[f]	1.90×10^{6}	1,340	1.06×10^{-4}
Containment late failure (1.07×10^{-5})	0	1.11	5.55×10^{-4}	5.94×10^{-9}	1.06×10^{5}	53.6	5.74×10^{-4}
	5	1.12	5.60×10^{-4}	5.99×10^{-9}	1.06×10^{5}	53.6	5.74×10^{-4}
LOCA (4.65×10^{-5})	0	0.0312	1.56×10^{-5}	7.25×10^{-10}	186	0.0931	4.33×10^{-6}
	5	0.0313	1.57×10^{-5}	7.30×10^{-10}	187	0.0935	4.35×10^{-6}
Containment bypass (1.53×10^{-6})	0	1540	1.00[e]	1.53×10^{-6}[f]	1.45×10^{6}	922	1.41×10^{-3}
	5	1680	1.00[e]	1.53×10^{-6}[f]	1.52×10^{6}	978	1.49×10^{-3}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred. MACCS2 calculates the dose to each exposed individual in the population, applies the appropriate cancer risk factor, and then sums the individual probabilities to determine the number of latent cancer fatalities.
d. Increased number of latent cancer fatalities per year.
e. Early fatality due to radiation dose assuming the accident occurred. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.
f. Increased likelihood of an early fatality per year.

Key: CLWR, commercial light water reactor; LOCA, loss-of-coolant accident.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–37 Generic CLWR Early Fatality Consequences and Risks

Accident (Frequency)	Reactor Annual Plutonium-238 Production (kilograms per year)	Population to 80 Kilometers (50 Miles)	
		Early Fatalities[a]	Annual Risk[b]
Early containment failure (7.92×10^{-8})	0	8.65	6.85×10^{-7}
	5	8.76	6.94×10^{-7}
Containment bypass (1.53×10^{-6})	0	3.48	5.32×10^{-6}
	5	3.51	5.37×10^{-6}

a. Number of early fatalities assuming the accident occurred.
b. Increased number of early fatalities per year.
Key: CLWR, commercial light water reactor.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–38 FFTF Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	LCF[a]	Annual Risk[b]	Dose (person-rem)	LCF[c]	Annual Risk[d]	Dose (rem)	LCF[a]	Annual Risk[b]
Design-basis accident primary sodium spill (MOX) (1×10^{-4})	0.00113	5.65×10^{-7}	5.65×10^{-11}	78.6	0.0393	3.93×10^{-6}	0.00313	1.25×10^{-6}	1.25×10^{-10}
Design-basis accident primary sodium spill (HEU) (1×10^{-4})	8.63×10^{-4}	4.32×10^{-7}	4.32×10^{-11}	72.6	0.0363	3.63×10^{-6}	0.00181	7.24×10^{-7}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX) (1×10^{-6})[e]	0.679	3.40×10^{-4}	3.40×10^{-10}	6.68×10^{4}	33.4	3.34×10^{-5}	0.679	2.72×10^{-4}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU) (1×10^{-6})[e]	0.481	2.41×10^{-4}	2.41×10^{-10}	6.16×10^{4}	30.8	3.08×10^{-5}	0.375	1.50×10^{-4}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX) (1×10^{-7})	0.00383	1.92×10^{-6}	1.92×10^{-13}	1,280	0.639	6.39×10^{-8}	0.357	1.43×10^{-4}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU) (1×10^{-7})	0.00384	1.92×10^{-6}	1.92×10^{-13}	1,230	0.617	6.17×10^{-8}	0.340	1.36×10^{-4}	1.36×10^{-11}
BLTC plutonium-238 target-handling accident (1×10^{-7})	2.61×10^{-4}	1.31×10^{-7}	1.31×10^{-14}	25.8	0.0129	1.29×10^{-9}	0.0279	1.12×10^{-5}	1.12×10^{-12}
BLTC isotope target-handling accident (1×10^{-7})	1.22×10^{-4}	6.10×10^{-8}	6.10×10^{-15}	2.74	0.00137	1.37×10^{-10}	0.0143	5.72×10^{-6}	5.72×10^{-13}
Standby accident (1×10^{-4})	1.34×10^{-7}	6.70×10^{-11}	6.70×10^{-15}	0.00999	4.99×10^{-6}	4.99×10^{-10}	1.62×10^{-8}	6.48×10^{-12}	6.48×10^{-16}
Deactivation accident (0.10)[f]	4.75×10^{-10}	2.38×10^{-13}	2.38×10^{-14}	3.64×10^{-5}	1.82×10^{-8}	1.82×10^{-9}	3.88×10^{-9}	1.55×10^{-12}	1.55×10^{-13}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Frequency incorporates external initiators.
f. Frequency is per event (deactivation) rather than per year.

Key: BLTC, bottom-loading transfer cask; FFTF, Fast Flux Test Facility; HEU, highly enriched uranium fuel; LCF, latent cancer fatalities; MOX, mixed oxide fuel.
Source: Model results, using the MACCS2 computer code (Chanin and Young 1997).

Table I–39 Accelerator Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
High-energy accelerator									
Design-basis target-handling accident (1×10^{-4})	2.93×10^{-4}	1.47×10^{-7}	1.47×10^{-11}	9.80×10^{-1}	4.90×10^{-4}	4.90×10^{-8}	9.35×10^{-4}	3.74×10^{-7}	3.74×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	11.7	5.85×10^{-3}	5.85×10^{-8}	3.01×10^{4}	18.0	1.80×10^{-4}	184	1.47×10^{-1}	1.47×10^{-6}
Low-energy accelerator									
Design-basis target-handling accident (1×10^{-4})	8.05×10^{-5}	4.03×10^{-8}	4.03×10^{-12}	17.7	8.85×10^{-3}	8.85×10^{-7}	1.12×10^{-3}	4.48×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake (1×10^{-5})	1.32×10^{-2}	6.60×10^{-6}	6.60×10^{-11}	32.4	1.62×10^{-2}	1.62×10^{-7}	2.08×10^{-1}	8.32×10^{-5}	8.32×10^{-10}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Source: Model results using MACCS2 (Chanin and Young 1997).

Table I–40 New Research Reactor Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Design-basis accident (1×10^{-4})	1.33×10^{-6}	6.65×10^{-10}	6.65×10^{-14}	2.41×10^{-3}	1.20×10^{-6}	1.20×10^{-10}	5.49×10^{-6}	2.20×10^{-9}	2.20×10^{-13}
Beyond-design-basis earthquake (1×10^{-5})	0.00373	1.87×10^{-6}	1.87×10^{-11}	27.6	1.38×10^{-2}	1.38×10^{-7}	0.0531	2.12×10^{-5}	2.12×10^{-10}
Fuel-handling accident (0.01)	1.90×10^{-9}	9.50×10^{-13}	9.50×10^{-15}	6.79×10^{-6}	3.40×10^{-9}	3.40×10^{-11}	5.83×10^{-9}	2.33×10^{-12}	2.33×10^{-14}
Neptunium-237 target-handling accident (0.01)	5.42×10^{-8}	2.71×10^{-11}	2.71×10^{-13}	8.95×10^{-5}	4.47×10^{-8}	4.47×10^{-10}	2.43×10^{-7}	9.72×10^{-11}	9.72×10^{-13}
Medical isotope target-handling accident (0.01)	1.04×10^{-5}	5.20×10^{-9}	5.20×10^{-11}	0.101	5.06×10^{-5}	5.06×10^{-7}	6.76×10^{-6}	2.70×10^{-9}	2.70×10^{-11}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Source: Model results, using the GENII (Napier et al. 1988) and MACCS2 (Chanin and Young 1997) computer codes.

Table I–41 New Research Reactor Decontamination and Decommissioning Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Spent fuel cask drop (5.0×10^{-6})	7.01×10^{-12}	3.51×10^{-15}	1.75×10^{-20}	2.78×10^{-8}	1.39×10^{-11}	6.95×10^{-17}	1.30×10^{-11}	5.20×10^{-15}	2.60×10^{-20}
Reactor core tank vaporization (1.0×10^{-4})[e]	1.55×10^{-5}	7.75×10^{-9}	7.75×10^{-13}	3.46×10^{-1}	1.73×10^{-4}	1.73×10^{-8}	5.23×10^{-5}	2.09×10^{-8}	2.09×10^{-12}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Frequency per event.

Source: Model results using MACCS2 (Chanin and Young 1997).

I.1.4 Processing Facility Accident Scenario Selection and Description

I.1.4.1 Plutonium-238 Processing

For the processing facilities, a spectrum of accidents was developed that considered a full range of accidents associated with such facilities. The scenarios evaluated, however, represent bounding cases that are considered to envelop the risk profile.

The processing facility accidents presented in the ORNL Radiochemical Engineering Development Center (REDC) Safety Analysis Report for Building 7920 (ORNL 1999) were reviewed for evaluation in this NI PEIS. Process and facility details were based on the preconceptual design study to support plutonium-238 production (Wham et al. 1998). Since process details at the Fluorinel Dissolution Processing Facility (FDPF) and the Fuel and Materials Examination Facility (FMEF) are essentially the same as those at REDC, the same spectrum of accidents was evaluated for all the processing facilities. However, facility differences were accounted for in evaluating the consequences of these accidents.

Several design-basis accidents were selected for inclusion in this NI PEIS. These include:

1. A postulated explosion in a glovebox during neptunium-237 target fabrication, representing the glovebox-handling accident having the largest potential consequences

2. A postulated failure of the target dissolver tank containing both neptunium-237 and plutonium-238, representing the accidental spill having the greatest consequences

3. A postulated explosion of an ion exchange column during plutonium-238 purification, which has the potential to release more plutonium-238 than any other processing facility design-basis accident.

A fire in a hot cell was judged to have lower consequences than an explosion, and was not included in this NI PEIS. This is based on an extensive experimental investigation (Hasegawa et al. 1992), which concluded that a fire in a hot cell would not represent a threat to the effectiveness of the facility roughing or HEPA filters and would be self-extinguishing within a short time from lack of oxygen.

Criticality accidents were not evaluated in this NI PEIS because the risk of accidental criticality, given appropriate administrative and process controls, is considered to be very low. Both neptunium-237 and plutonium-238 would be stored in shielded containers in quantities and configurations that preclude criticality. Target preparation and postirradiation processing will be carried out in batches involving quantities well below those at which criticality could occur. As a result, a criticality accident could occur only as a result of a series of gross, deliberate violations of established controls.

The postulated beyond-design-basis processing facility accident selected for use in this NI PEIS is a catastrophic earthquake resulting in a collapse of the nearby stack and failure of the HEPA filter system intended to mitigate the consequences of releases. Two cases of this accident were evaluated. Case 1 assumed that the facility was only being used to store neptunium-237. Case 2 assumed that the facility was an integrated storage, target fabrication, and irradiated-target-processing facility.

The waste stream from the irradiated targets would be processed in the same facilities as the irradiated targets. Accidents occurring during the processing of the waste stream were not evaluated in this NI PEIS because their consequences are bounded by the irradiated target accidents that have been evaluated.

I.1.4.1.1 Design-Basis Accidents

ION EXCHANGE EXPLOSION DURING NEPTUNIUM-237 TARGET FABRICATION

An accident can occur during fabrication of the neptunium-237 targets. As part of the target preparation, 1-kilogram (2.2-pound) quantities of neptunium-237 solution are processed (Wham et al. 1998) to yield neptunium in an oxide form for use as a target material. This operation takes place in a shielded glovebox and involves use of an ion exchange column. This accident scenario postulates an explosion of the ion exchange column in the glovebox. Judging from historical occurrences of this type of accident at radiochemical laboratories and processing facilities, the frequency of this event is "unlikely" (between 1×10^{-2} and 1×10^{-4} per year) (ORNL 1999). For the purpose of this NI PEIS, the accident frequency is assumed to be 1×10^{-2} per year.

The glovebox is maintained at a slight negative pressure with respect to that portion of the building outside the hot cells, and is continually exhausted to the atmosphere through roughing filters and HEPA filters, and via a stack.

An explosion is estimated to release essentially all of the neptunium-237 into the glovebox. Additional data to calculate releases were taken from relevant facility data (ORNL 1999; Green 1998, 1999) and other accepted sources (DOE 1994a). Since an explosion involves small quantities of materials, any increase in pressure is expected to be small and is not expected to result in transitory leakage of radioactive material from the glovebox into the operating area.

The glovebox is exhausted through roughing filters and then through two banks of HEPA filters arranged in series outside the building and then to the environs via a stack. Each bank of the HEPA filters is assumed to remove 99.95 percent of all particulates at or above a size of 0.3 micron (Burchsted et al. 1976). (Note: This assumes two HEPA filters are in series and each is 99.95 percent efficient, yielding a 2.5×10^{-7} reduction factor.)

Airborne releases can be divided into respirable (smaller than about 10 microns) and nonrespirable fractions. Nonrespirable airborne particles can cause localized onsite contamination, but they do not contribute significantly to offsite doses for several reasons. For design-basis accidents, the filter efficiency for the larger, nonrespirable particles is greater than that for all particles of the respirable fractions, and significantly greater than the minimum value of 99.95 percent for 0.3-micron particles. For the beyond-design-basis earthquake, where filters are postulated to be ineffective, leakage from the hot cells is at a low rate, allowing for increased deposition and settling of the larger particles prior to release. Even where large, nonrespirable particles are released to the environment, their atmospheric transport is limited and they will "fall out" within a short distance from the release point.

Table I–42 shows the release fractions and source terms for this accident.

Table I–42 Neptunium-237 Target Preparation Accident Source Terms

Analysis Parameters	Units
Neptunium-237 inventory in glovebox	1,000 grams
Neptunium-237 released into glovebox from explosion	1,000 grams
Airborne release fraction times respirable particle fraction	7×10^{-2}
Leak path factor	0.50
Neptunium-237 reaching HEPA filters	35.0 grams
Neptunium-237 released from stack to environs	8.75×10^{-6} grams

Source: Calculated results.

TARGET DISSOLVER TANK FAILURE DURING PLUTONIUM-238 SEPARATION

A hypothetical accident considered was the failure of a tank in which irradiated neptunium-237 targets are to be dissolved. The irradiated neptunium-237 target processing is planned to be carried out in approximately five batches per year. Each batch of irradiated targets is expected to contain approximately 1 kilogram (2.2 pounds) of plutonium-238 and 8 to 10 kilograms (17.6 to 22 pounds) of neptunium-237. A complete failure of the dissolver tank envelops a spectrum of accidental spills involving plutonium-238 in the hot cells. The complete failure of this tank is judged to be unlikely (between 1×10^{-2} and 1×10^{-4} per year) (ORNL 1999). For the purpose of this NI PEIS, the accident frequency is assumed to be 1×10^{-2} per year.

This scenario postulates the sudden, complete failure of the dissolver tank and the spilling of its contents onto the floor of the hot cell. The product of the airborne release fraction and the respirable fraction is the sum of that for a free-fall spill, plus evaporation of a shallow pool and are estimated (DOE 1994a) to be 0.00013. A leak path factor of 0.75, applicable for a hot cell (Green 1998), was used.

The cell is exhausted first to roughing filters, then through two stages of HEPA filters in series, and then to the environs via a stack. (Note: This assumes two HEPA filters are in series, and each is 99.95 percent efficient, yielding a 2.5×10^{-7} reduction factor.)

Table I–43 shows the release fractions and source terms for this accident.

Table I–43 Target Dissolver Tank Failure Source Terms

Analysis Parameters	Neptunium-237	Plutonium-238
Inventory in dissolver tank	9,000 grams	1,000 grams
Spilled onto hot cell floor	9,000 grams	1,000 grams
Airborne release fraction times respirable fraction	0.00013	0.00013
Leak path factor	0.75	0.75
Amount entering HEPA filters	0.88 gram	0.098 gram
Amount released from stack to environs	2.19×10^{-7} gram	2.44×10^{-8}gram

Source: Calculated results.

ION EXCHANGE EXPLOSION DURING PLUTONIUM-238 SEPARATION

A hypothetical accident considered is the postulated explosion of an ion exchange column during plutonium-238 purification in a hot cell. Although plans for plutonium purification call for a solvent extraction process, an alternative method involves the use of an ion exchange process (Wham et al. 1998). In this alternative procedure, 495 grams (1.1 pounds) of plutonium-238 are loaded onto an ion exchange column. This postulated accident scenario involves an explosion of this ion exchange column. Judging from historical occurrences of this type of accident at radiochemical laboratories and processing facilities, the frequency of

this event is unlikely (between 1×10^{-2} and 1×10^{-4} per year) (ORNL 1999). For the purpose of this NI PEIS, the accident frequency is assumed to be 1×10^{-2} per year.

Most of the plutonium will be deposited on the cell walls and floor along with other explosion debris. The fraction of plutonium estimated to be released in airborne form and respirable size particles is 0.07 (DOE 1994a).

The hot cell is maintained at a slight negative pressure with respect to the rest of the building. After effluents are exhausted from the hot cell, they pass first through roughing filters, then through two banks of HEPA filters outside the building. On exiting the HEPA filters, effluents are released to the environs through a stack. At the REDC, the explosion could also result in the generation of a weak shock wave and a momentary pressure increase of up to several pounds per square inch gage in the hot cell (ORNL 1999). This accident would not be expected to generate dynamic pressures sufficient to damage the hot cell confinement structure, but could result in some leakage of radioactive materials into the operating areas of the building due to the brief pressurization of the hot cell cubicle (ORNL 1999). Because of the larger volume of the FDPF and FMEF facilities, the magnitude of a shock wave would be much lower.

For REDC, the shock wave may impact the HEPA filters, possibly degrading their performance. Although the HEPA filters are tested to retain 99.97 percent efficiency, tornado conditions are estimated (DOE 1994a) to reduce their efficiency to approximately 99 percent. This scenario assumes that the efficiency of the first-stage HEPA filters at REDC is partially degraded to 99.5 percent while the second-stage efficiency is 99.95 percent. This yields a reduction factor of 2.5×10^{-6} at REDC. Both HEPA stages are 99.95 percent efficient, yielding a reduction factor of 2.5×10^{-7} at FDPF and FMEF. For FDPF and FMEF, the HEPA filters were assumed not to be degraded, because the magnitude of any shock wave generated would be much less. The release to the environment was conservatively assumed to consist of a single "puff" associated with the immediate explosion.

Table I–44 shows the release fractions and source terms for this accident.

Table I–44 Plutonium-238 Ion Exchange Explosion Accident Source Terms

Analysis Parameters	Units
Plutonium-238 material at risk	495 grams
Plutonium-238 released into Hot Cell E from explosion	495 grams
Airborne release fraction times respirable particle fraction	7×10^{-2}
Leak path factor	0.75
Plutonium-238 reaching HEPA filters	26.0 grams
Plutonium-238 released to environs	6.50×10^{-5} gram REDC
	6.50×10^{-6} gram FDPF, FMEF

Source: Calculated results.

I.1.4.1.2 Beyond-Design-Basis Accident

The postulated beyond-design-basis processing facility accident selected for use in this NI PEIS is a catastrophic earthquake. Such an event is less likely than the design-basis processing facility accidents, although its consequences could be severe. Its frequency is assumed to be 1×10^{-5} per year.

CASE 1—STORAGE FACILITY

The earthquake is postulated to collapse the stack, severely damaging the HEPA filter system located nearby. Although the building is expected to collapse, the hot cells are expected to remain intact, but with cracked walls. In addition, one or more of the shielded viewing windows may be cracked or broken. The ventilation systems exhausting from the hot cells are expected to fail. Neptunium-237 is stored in double steel cans, with both the inner and outer cans sealed. The double cans are stacked in an array of robust, seismically supported steel storage tubes inside the hot cell. The analysis postulated the storage tube array would maintain geometry and not be damaged by equipment dislodged within the hot cell during the event. It was postulated that none of the storage cans in the storage tubes would be damaged. The storage cans would not be stressed to a level that would breach the double containment of the can design. No neptunium was postulated to be released from the storage cans during the event.

At INEEL, neptunium-237 may be stored in the CPP–651 vault, which is within 91 meters (100 yards) of FDPF. The CPP–651 vault has 100 in-ground concrete storage silo positions sealed with 5-centimeter (2-inch) stainless steel shielding plugs. The neptunium-237 storage cans would be placed in a rack inside the silo. While the postulated beyond-design-basis earthquake could cause portions of the facility to collapse, none of the storage cans in the in-ground storage silos would be breached. The storage cans would not be stressed to a level that would breach the double containment of the can design.

CASE 2—PROCESSING FACILITY

The earthquake is postulated to collapse the stack, severely damaging the HEPA filter system located nearby. Although the building is expected to collapse, the hot cells are expected to remain intact, but with cracked walls. In addition, one or more of the shielded viewing windows may be cracked or broken. The ventilation systems exhausting from the hot cells are expected to fail. Radioactive materials in the hot cells will be released as a result of cracks in cell walls and shielded windows, but the rate of leakage is expected to be low, since the hot cells are not pressurized and there is no forced ventilation. The leak path factor (i.e., the mass fraction of airborne particulates in an enclosure that is released to the environment) under these conditions has been conservatively estimated to be 0.1 (Green 1997).

The plutonium-238 inventory in the facility will be in several different chemical and physical forms. Since processing is carried on in batches that overlap one another (Wham et al. 1998), the total quantity of plutonium-238 considered available for release from the facility is the sum of the amounts in the dissolver tank, in the ion exchange column during purification, and in powder form and not yet been placed into a sealed canister. Any plutonium-238 in irradiated targets awaiting processing is unlikely to be mechanically damaged by the earthquake because of their rather small size and thus resistance to mechanical breakage. Even if some targets were broken, the plutonium-238 is intimately mixed with the neptunium-237 oxide and an aluminum matrix, rendering it essentially immobile. The earthquake is postulated to result in a massive spill and/or failure of the dissolver tank, an explosion in an ion exchange column, and a spill of any plutonium-238 powder not in a sealed container.

Table I–45 shows the release fractions and the ground-level release source terms for this accident.

Table I–45 Beyond-Design-Basis Earthquake Accident Source Terms

Analysis Parameters	Plutonium-238 Form and Location			
	Solution—Dissolver Tank	Solution—Ion Exchange Column	Powder—Hot Cell Cubicle	Total
Material at risk	1,000 grams	495 grams	186 grams	1,681 grams
Released into hot cell	1,000 grams	495 grams	186 grams	1,681 grams
Airborne release fraction times respirable fraction	0.00013	0.07	0.0033	–
Leak path factor	0.1	0.1	0.1	–
Released to environs	0.013 gram	3.47 grams	0.061gram	3.54 grams

Source: Calculated results.

I.1.4.2 Medical, Industrial, and Research and Development Isotope Processing

The accident analyses included identification of a set of potential accidents that could occur during medical, industrial, and research and development isotope processing at the Radiochemical Processing Laboratory (RPL) (Building 325), FMEF, and the support facility. The analyses are based on scenarios evaluated in the *Building 325 Safety Analysis Report* (the safety analysis report for RPL) for similar types of processes (Battelle 2000). Since process details at the FMEF and the support facility are essentially the same as those at RPL, the same spectrum of accidents was evaluated for all the processing facilities.

The set of accidents evaluated for the Safety Analysis Report was selected using a standard Preliminary Hazards Assessment to identify the potential hazardous conditions in facility operations and to determine the significance of potential accidents. The types of events considered in the Safety Analysis Report included operator errors and handling accidents, fires and explosions, natural phenomena such as seismic events, criticality, and external events such as loss of support services.

For this analysis one bounding event was identified in each of the frequency categories evaluated in the Safety Analysis Report (anticipated, unlikely, or extremely unlikely) in order to identify the events that result in both maximum consequence and maximum risk to onsite and offsite individuals. All types of events that could apply to medical isotope processing were evaluated to determine which scenarios could result in the maximum radionuclide release fraction for each frequency category. The analysis is intended to provide a conservative estimate for the potential consequences of the proposed activities.

Potential accidental releases of radioactive materials during medical, industrial, and research and development isotope processing were estimated using projected radionuclide inventories for the target systems most likely to be considered for production of medical, industrial, and research and development isotopes. **Table I–46** presents the radioactive inventories for the most likely target products. The irradiated targets have a much greater radioactive inventory than the unirradiated targets. Only the radium-226 target is initially radioactive, and its products, actinium and thorium, have greater health consequences than the initial radium-226. Therefore, the accident consequences were analyzed for the irradiated target products.

Processing was assumed to occur 1 day after removal of the targets from the reactor, and only the isotopes associated with a single product target are assumed to be at risk for release in any given event. Release fractions for the radionuclides associated with each product were calculated using the same assumptions as those used for similar types of materials in the Safety Analysis Report scenarios.

Table I–46 Medical, Industrial, and Research and Development Target Product Inventories

Product Isotope	Radionuclide	Target Inventory[a] (curies)
Gold-198	Gold-198	132
R3	Gold-198m	0.00
	Gold-199	150
	Gold-200	0.00816
	Gold-200m	0.0434
	Mercury-203	1.79×10^{-8}
	Platinum-197	9.83×10^{-6}
Cadmium-109	Cadmium-109	654
LTIV	Sodium-108	3.53×10^{-5}
	Sodium-108m	3.92×10^{-4}
	Sodium-110	0.00878
	Sodium-110m	0.627
	Sodium-111	0.119
	Sodium-111m	4.11×10^{-12}
	Palladium-109	2.73×10^{-4}
Copper-64	Copper-64	1,300
R3	Copper-66	2.50×10^{-14}
	Copper-67	4.36×10^{-7}
	Nickel-65	6.42×10^{-9}
	Nickel-66	8.80×10^{-4}
	Zinc-65	138
Copper-67	Copper-67	6.26
R3	Copper-66	0.00
	Copper-68	1.98×10^{-13}
	Copper-69	8.88×10^{-31}
	Zinc-65	0.00
	Zinc-69	0.00268
	Zinc-69m	0.00268
Gadolinium-153	Gadolinium-153	1,100
LTIV	Europium-152	4,660
	Europium-152m	6.41×10^{4}
	Europium-154	1.55×10^{4}
	Europium-154m	2.20×10^{-4}
	Europium-155	3,540
	Europium-156	3.39×10^{5}
	Samarium-153	3.16×10^{4}
Holmium-166	Holmium-166	58.9
R3	Dysprosium-166	2.07×10^{-6}
	Dysprosium-167	8.80×10^{-20}
	Erbium-167m	1.91×10^{-4}
	Erbium-169	2.13×10^{-6}
	Holmium-166m	9.92×10^{-5}

Product Isotope	Radionuclide	Target Inventory[a] (curies)
	Holmium-167	1.91×10^{-4}
	Holmium-168	3.12×10^{-19}
	Holmium-169	8.29×10^{-36}
	Holmium-170	0.00
	Holmium-170m	0.00
Samarium-145	Samarium-145	11.8
LTIV	Neodymium-147	1.54×10^{-5}
	Neodymium-149	3.72×10^{-20}
	Neodymium-151	3.36×10^{-25}
	Promethium-145	1.21
	Promethium-146	3.50×10^{-4}
	Promethium-147	0.00162
	Promethium-148	0.00365
	Promethium-148m	8.70×10^{-4}
	Promethium-149	0.00218
	Promethium-150	2.56×10^{-7}
	Promethium-151	1.89×10^{-9}
	Samarium-151	2.38×10^{-7}
Samarium-153	Samarium-153	70.7
R3	Europium-152	0.00
	Europium-152m	0.00
	Europium-154	0.00348
	Europium-154m	8.75×10^{-10}
	Europium-155	2.32×10^{-4}
	Europium-156	0.00446
	Gadolinium-153	0.00
	Samarium-151	0.00
	Samarium-155	8.63×10^{-14}
	Samarium-156	1.11×10^{-9}
Tin-117m	Tin-117m	48.5
R3	Antimony-122	0.00118
	Antimony-122m	4.72×10^{-8}
	Tin-119m	3.92×10^{-8}
	Tin-121	0.00
	Tin-121m	4.76×10^{-11}
Strontium-85	Strontium-85	2,160
LTIV	Krypton-83m	0.00
	Krypton-85	9.51×10^{-4}
	Krypton-85m	1.50×10^{-4}
	Rubidium-83	0.00
	Rubidium-84	8.48
	Rubidium-86	4.18
	Strontium-83	0.00
	Strontium-85m	0.00101
	Strontium-89	9.40×10^{-9}

Product Isotope	Radionuclide	Target Inventory[a] (curies)
	Strontium-90	6.09×10^{-15}
Strontium-89	Strontium-89	156
LTIV	Krypton-85	0.00
	Krypton-85m	0.00
	Krypton-87	0.00
	Rubidium-84	0.00
	Rubidium-86	0.00
	Rubidium-86m	0.00
	Rubidium-88	2.09×10^{-12}
	Strontium-85	0.00
	Strontium-87m	0.00
	Strontium-90	1.69×10^{-4}
	Yttrium-88	2.71×10^{-6}
	Yttrium-90	0.0218
	Yttrium-90m	2.08×10^{-7}
	Zirconium-89	8.15×10^{-9}
Iodine-125	Iodine-125	2,530
Gas Line	Iodine-124	0.00
	Iodine-126	0.00
	Xenon-125	0.00
Iodine-131	Iodine-131	307
R3	Iodine-132	0.00867
	Iodine-132m	3.52×10^{-7}
	Tellurium-131	3.46
	Tellurium-131m	15.5
	Tellurium-132	0.00830
	Xenon-131m	2.02
Iridium-192	Iridium-192	3,570
LTIV	Iridium-192m	8.36×10^{-8}
	Iridium-193m	27.6
	Iridium-194	0.0317
	Iridium-194m	0.00991
	Iridium-195	1.10×10^{-6}
	Iridium-195m	3.55×10^{-6}
	Platinum-193	0.0886
	Platinum-193m	13.2
	Platinum-195m	3.82×10^{-4}
Lutecium-177	Lutecium-177	0.519
R3	Hafnium-177m	0.00111
	Hafnium-178m	9.88×10^{-13}
	Hafnium-179m	3.10×10^{-9}
	Lutecium-176m	0.00
	Lutecium-177m	0.00143
	Lutecium-178	9.87×10^{-13}

Product Isotope	Radionuclide	Target Inventory[a] (curies)
	Lutecium-179	3.09×10^{-9}
Molybdenum-99	Molybdenum-99	1,680
R3	Molybdenum-101	2.08×10^{-5}
	Molybdenum-102	3.79×10^{-18}
	Molybdenum-103	3.71×10^{-34}
	Ruthenium-103	2.09×10^{-6}
	Technetium-99m	1,830
	Technetium-100	8.94×10^{-10}
	Technetium-101	2.08×10^{-5}
	Technetium-102	3.79×10^{-18}
	Technetium-103	2.84×10^{-34}
Osmium-194	Osmium-194	2.20
LTIV	Iridium-192	0.00
	Iridium-192m	0.00
	Iridium-193m	0.00
	Iridium-194	2.19
	Iridium-194m	0.00
	Osmium-190m	0.00
	Osmium-191	0.00
	Osmium-191m	0.00
	Osmium-192m	0.00
	Osmium-193	9.02×10^{4}
	Rhenium-189	0.00
	Rhenium-190	0.00
	Rhenium-190m	0.00
	Rhenium-191	0.00
Tungsten-188	Tungsten-188	5,810
LTIV	Hafnium-181	0.00
	Hafnium-182m	0.00
	Osmium-189m	140
	Rhenium-186	0.00
	Rhenium-188	4.45×10^{4}
	Rhenium-188m	1.86×10^{-7}
	Rhenium-189	46.4
	Tantalum-182	0.00
	Tantalum-182m	0.00
	Tantalum-183	0.00
	Tungsten-181	0.00
	Tungsten-185	0.00
	Tungsten-185m	0.00
	Tungsten-187	7.24×10^{5}
	Tungsten-189	3.69×10^{-9}
Xenon-127	Xenon-127	7.26
LTIV		

Product Isotope	Radionuclide	Target Inventory[a] (curies)
Yttrium-91	Yttrium-91	17.8
LTIV	Yttrium-90	0.00
	Zirconium-89	0.00
	Zirconium-95	2.88×10^{-19}
Phosphorus-32	Phosphorus-32	39.1
R3	Phosphorus-33	0.0518
	Phosphorus-34	1.63×10^{-14}
	Phosphorus-35	2.92×10^{-32}
	Phosphorus-36	0.00
	Sulfur-35	1.99×10^{-8}
Phosphorus-33	Phosphorus-33	76.2
LTIV	Argon-37	1.88×10^{-25}
	Phosphorus-32	0.00
	Sulfur-35	7.29×10^{-6}
Palladium-103	Palladium-103	1,340
R3	Silver-107m	1.58×10^{-26}
	Palladium-107m	2.00×10^{-20}
	Rhodium-103m	1,350
	Rhodium-104	2.74×10^{-9}
	Rhodium-104m	1.89×10^{-10}
	Rhodium-105	9.07×10^{-6}
	Rhodium-105m	2.13×10^{-26}
	Rhodium-106	2.45×10^{-18}
	Rhodium-106m	3.86×10^{-11}
	Rhodium-107	5.24×10^{-26}
Platinum-195m	Platinum-195m	168
R3		
Rhenium-186	Rhenium-186	4,350
R3	Osmium-189m	1.48×10^{-4}
	Osmium-190m	1.23×10^{-11}
	Rhenium-188	0.0550
	Rhenium-188m	2.60×10^{-13}
	Rhenium-189	1.16×10^{-5}
	Rhenium-190	5.39×10^{-12}
	Rhenium-190m	1.15×10^{-11}
	Tungsten-187	6.41
	Tungsten-188	0.0113
	Tungsten-189	9.74×10^{-15}
Scandium-47	Scandium-47	29.6
R3	Calcium-45	0.00
	Calcium-47	1.82×10^{-5}
	Scandium-46	0.00
	Scandium-48	0.0202

Product Isotope	Radionuclide	Target Inventory[a] (curies)
Selenium-75	Selenium-75	17.9
LTIV	Arsenic-76	0.114
	Arsenic-77	3.78×10^{-5}
	Arsenic-78	3.94×10^{-13}
	Bromine-80	0.00
	Bromine-80m	2.92×10^{-36}
	Selenium-77m	1.82×10^{-13}
	Selenium-79m	9.30×10^{-18}
Actinium-227	Actinium-227	34.0
LTIV	Actinium-228	56.1
	Actinium-229	6.04×10^{-9}
	Radium-226	14.3
	Radium-227	4.23×10^{-7}
	Radium-228	0.00101
	Radium-229	5.00×10^{-14}
	Thorium-227	24.8
	Thorium-228	42.1
	Thorium-229	8.63×10^{-4}
	Actinium-225	3.72×10^{-4}
	Astatine-217	3.72×10^{-4}
	Bismuth-210	0.109
	Bismuth-211	19.6
	Bismuth-212	24.6
	Bismuth-213	3.71×10^{-4}
	Bismuth-214	14.3
	Francium-221	3.72×10^{-4}
	Francium-223	1.40×10^{-5}
	Lead-209	3.69×10^{-4}
	Lead-210	0.118
	Lead-211	19.6
	Lead-212	38.4
	Lead-214	14.3
	Polonium-210	0.106
	Polonium-211	0.0535
	Polonium-212	24.6
	Polonium-213	3.63×10^{-4}
	Polonium-214	14.3
	Polonium-215	19.6
	Polonium-216	38.8
	Polonium-218	14.3
	Radium-223	19.6
	Radium-224	38.8
	Radium-225	5.46×10^{-4}
	Radon-217	4.46×10^{-8}
	Radon-219	19.6
	Radon-220	38.8
	Radon-222	14.3

Product Isotope	Radionuclide	Target Inventory[a] (curies)
	Thallium-207	19.6
	Thallium-208	8.83
	Thallium-209	8.16×10^{-6}

a. Assumes a 1-day cooling time after irradiation.
Key: LTIV, Long-Term Irradiation Vehicle; R^3, Rapid Radioisotope Retrieval system.
Source: BWHC 1999.

I.1.4.2.1 Localized Solvent Fire

The safety analysis report for RPL identified a number of accident scenarios with an anticipated frequency greater than 0.01 per year. The types of accidents that fell into this category included the following:

1. Localized solvent fire
2. Localized solid fire
3. Spill in a hot cell
4. Spill in a laboratory

Of these events, the scenario with the highest radionuclide release was the solvent fire. A localized fire of sufficient severity to produce radionuclide releases was estimated to occur no more than once in 10 years. The upper-bound frequency of such an event was supported by the fire loss history at Hanford over a 45-year period. During that time, the site experienced 10 fires that resulted in significant property loss. Of those fires, 6 potentially involved radioactive materials, and 2 of the 6 occurred in laboratory facilities. No fires of that magnitude have occurred in RPL since it was occupied in 1953 and would not be expected to occur routinely in that facility because of the facility design, administrative controls on conduct of operations, and the fire protection program. Since only 2 events potentially involving radioactive materials occurred in laboratory facilities over a 45-year period, a frequency of 4.44×10^{-2} per year was assumed for this accident.

The heating, ventilating, and air conditioning system was assumed to be operating during and after the fire. Combustibles (e.g., solvent-soaked rags) were assumed to be present in sufficient quantity to support combustion. The source terms used for this accident scenario were based on radioactive materials representative of anticipated medical-isotope-processing activities in the hot cells and other laboratories in the facility. Manual fire suppression was assumed not to occur or to be ineffective.

The final HEPA filters were assumed to be unaffected by the fire because they are located in a facility separate from RPL. This assumption was based on the observations that the primary filters would stop most smoke particles and that air dilution would cool the hot gases leaving the laboratory or hot cell so the final HEPA filter bank would not be subjected to extreme temperatures. Therefore, the final stage of HEPA filters was assumed to remain intact. For conservatism, particle deposition along the release path was assumed not to occur. The radon holdup system was assumed to be ineffective, and it was also assumed that no deposition or filtration of noble gases would occur.

It was assumed that charcoal filters will be included in the emergency ventilation system. The activated-charcoal filters will comply with current industry standards. Filter efficiency was conservatively assumed to be 99 percent, consistent with NRC Regulatory Guide 1.52 (NRC 1978). For conservatism, iodine deposition prior to filtration was assumed not to occur.

The radionuclide releases for this event, as estimated in the safety analysis report, were assumed to be the same as those for a solvent fire involving radioactive solutions. Although many of the processes for preparing medical and industrial isotopes would involve only dissolution in aqueous acid solution, some of the chemical

separations could require solvent extraction or ion exchange apparatus. Therefore, the solvent fire was assumed to be a bounding case for this scenario. Separate release fractions were calculated for nonvolatile materials, volatile materials (iodine, sulfur), and noble gases.

The release fractions were calculated by the following generic formula: Airborne Release Fraction × Respirable Fraction × Leak Path Factor × Filter Removal Factor. Calculations for the three releases fractions are:

Nonvolatiles: $0.01 \times 1.0 \times 1.0 \times 0.0005 = 5\times10^{-6}$

Volatiles: $1.0 \times 1.0 \times 1.0 \times 0.01 = 0.01$

Gases: $1.0 \times 1.0 \times 1.0 \times 1.0 = 1.0$

To determine which irradiated target would result in the maximum consequences, the radionuclide inventories for each of the irradiated targets were multiplied by the appropriate release fractions. The resulting inventories were then multiplied by dose conversion factors. This final multiplication resulted in a dose for each isotope. The isotope doses within each target were totaled for a target dose. The target doses were compared to determine the target with the maximum dose consequences. The iodine-125 gas line product target resulted in the highest target dose. Therefore, the iodine-125 product target was used to determine the bounding consequences for the localized fire event.

The resulting source term for the localized fire accident is an elevated release of 25.3 curies of iodine-125.

I.1.4.2.2 Unlikely Seismic Event

Events in the unlikely frequency category (between 1×10^{-4} and 1×10^{-2} per year) in the RPL safety analysis report included:

1. Liquid waste cask failure and spill
2. Unlikely seismic event

Of these scenarios, the unlikely seismic event resulted in a higher radionuclide release fraction. This event was assigned to the unlikely frequency category due to the return period of the initiating earthquake. Earthquake hazard curves have been developed for the 300 Area that define ground acceleration at RPL for a given frequency. The seismic event analyzed in this section has a peak horizontal ground acceleration of 0.139 g for a frequency range of 1×10^{-4} to 1×10^{-2} per year. For earthquakes in the unlikely category, a single potential process upset was assumed, but it was estimated that multiple major upsets would not occur. The facility's superstructure was assumed to remain intact, but the heating, ventilating, and air conditioning system was assumed to fail because it has not been seismically qualified. For the purpose of this NI PEIS, the accident frequency is assumed to be 1×10^{-2} per year.

Spilling of the powdered contents of one in-process medical, industrial, or research and development isotope target was conservatively assumed to occur (i.e., the probability of the spill given that the seismic event occurs was assumed to be 1.0). The release from the spill and holdup release were reduced by 50 percent to account for deposition of the powder within the facility. This 50 percent building removal factor could be applied to this scenario because of essentially static conditions that result from failure of the ventilation system. Because this event does not involve a heat source to mobilize volatile materials, the release fraction was assumed to be the same for all materials except noble gases. The radon holdup system was assumed to be ineffective for this scenario.

Releases from this event were estimated as follows: Airborne Release Fraction × Respirable Fraction × Leak Path Factor × Filter Removal Factor. The specific release fractions are:

Nongases: $0.002 \times 0.3 \times 0.5 \times 1.0 = 3 \times 10^{-4}$

Gases: $1.0 \times 1.0 \times 1.0 \times 1.0 = 1.0$

As in the anticipated category source term analysis, the radionuclide inventories for each of the irradiated targets were multiplied by the appropriate release fractions and dose conversion factors. A comparison of the target doses indicated that the actinium-227 product target results in maximum dose consequences for this accident. In addition to the actinium-227 product, the irradiated target contains 9 impurity isotopes and 32 decay products. The dose screening, however, determined that at least 99.9 percent of the total dose is attributable to six isotopes (actinium-227, radium-223, radium-224, radium-226, thorium-227, thorium-228).

The ground-level release source term for the unlikely seismic event is as follows (releases in curies):

Isotope	Release
Actinium-227	0.0102
Radium-223	0.00588
Radium-224	0.0116
Radium-226	0.00429
Thorium-227	0.00744
Thorium-228	0.0126

I.1.4.2.3 Loss of Electric Power and Explosion

The safety analysis report for RPL identified the following events in the extremely unlikely category (between 1×10^{-6} and 1×10^{-4} per year):

1. Loss of electric power and explosion
2. Large uncontrolled fire
3. Extremely unlikely seismic event

Of the extremely unlikely events identified in the safety analysis report for RPL, the highest radionuclide release was associated with the loss of electrical services followed by an explosion. This scenario assumes loss of power to RPL, which inactivates the ventilation system. On failure of the ventilation systems, airflow through the hot cells, gloveboxes, hoods, and tanks would also cease. Without ventilation, the potential exists for a buildup of flammable or combustible vapors in those areas with volatile chemicals. A deflagration in a glovebox from the buildup of a flammable solvent or volatile chemical is assumed to occur, potentially breaching the primary confinement barriers. The walls and ceilings of the glovebox or fume hood would mitigate the impact of an explosion. Most of the airborne material within a glovebox or hood would be carried out through the exhaust system, even if the explosion were to cause material to be released from the glovebox to an adjoining area. Judging from actual glovebox explosions, the front panel of a glovebox could fail. In most cases, these events have not resulted in offsite releases because the explosions did not cause malfunctions of ventilation systems or the failure of other barriers, including room walls and ceilings. However, for the extremely unlikely scenario in this analysis, the explosion is assumed to be sufficiently forceful to breach the building or ventilation system barriers, rendering the HEPA filters and radon holdup system ineffective.

Releases from this event were estimated as follows: Airborne Release Fraction × Respirable Fraction × Leak Path Factor × Filter Removal Factor. The specific release fractions are:

Nonvolatiles: $0.05 \times 0.4 \times 0.5 \times 1.0 = 0.01$

Volatiles and gases: $1.0 \times 1.0 \times 1.0 \times 1.0 = 1.0$

As in the previous source term analyses, the radionuclide inventories for each of the irradiated targets were multiplied by the appropriate release fractions and dose conversion factors. A comparison of the target doses indicated that the actinium-227 product target results in maximum dose consequences for this accident.

The ground-level release source term for the extremely unlikely loss of electrical services followed by an explosion is:

Isotope	Release
Actinium-227	0.340
Radium-223	0.196
Radium-224	0.388
Radium-226	0.143
Thorium-227	0.248
Thorium-228	0.421

I.1.5 Methodology for Estimating Processing Facility Accident Radiological Impacts

The exposure, uptake, and usage parameters used in the GENII model for assessing processing facility accident impacts are provided in **Tables I–47** through **I–49**. The GENII computer code was used to estimate the radiological consequences of the postulated accidents at the processing facilities. A discussion of the GENII computer code is presented in Appendix H. Doses to a noninvolved worker, the maximally exposed individual, and the population within 80 kilometers (50 miles) of each plant were calculated. To determine the consequences for the maximally exposed individual, doses were calculated at the site boundary and at the nearest highway within the site boundary. The population and boundary maximally exposed individual doses included doses via ingestion. The nearest highway maximally exposed individual is assumed to be exposed for a period of 2 hours. The consequences (doses) were then multiplied by the frequencies of the accidents to determine the risk.

Table I–47 GENII Exposure Parameters to Plumes and Soil Contamination (Accidents)

Maximum Individual				General Population			
External Exposure		Inhalation of Plume		External Exposure		Inhalation of Plume	
Plume (hours)	Soil Contamination (hours)	Exposure Time (hours)	Breathing Rate (cubic centimeters per second)	Plume (hours)	Soil Contamination (hours)	Exposure Time (hours)	Breathing Rate (cubic centimeters per second)
0.00	6,136	100% of release time	330	0.00	4,383	100% of release time	330

Source: Napier et al. 1988; NRC 1977.

Table I–48 GENII Usage Parameters for Consumption of Terrestrial Food (Accidents)

Food Type	Maximum Individual				General Population			
	Growing Time (days)	Yield (kg/m^2)	Holdup Time (days)	Consumption Rate (kg/yr)	Growing Time (days)	Yield (kg/m^2)	Holdup Time (days)	Consumption Rate (kg/yr)
Leafy vegetables	90.0	1.5	1.0	30.0	90.0	1.5	14.0	15.0
Root vegetables	90.0	4.0	5.0	220.0	90.0	4.0	14.0	140.0
Fruit	90.0	2.0	5.0	330.0	90.0	2.0	14.0	64.0
Grains/cereals	90.0	0.8	180.0	80.0	90.0	0.8	180.0	72.0

Key: kg/m^2, kilograms per square meter; kg/yr, kilograms per year.
Source: Napier et al. 1988.

Table I–49 GENII Usage Parameters for Consumption of Animal Products (Accidents)

Food Type	Consumption Rate (kg/yr)	Holdup Time (days)	Stored Feed				Fresh Forage			
			Diet Fraction	Growing Time (days)	Yield (kg/m^2)	Storage Time (days)	Diet Fraction	Growing Time (days)	Yield (kg/m^2)	Storage Time (days)
Maximum individual										
Beef	80.0	15.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	18.0	1.0	1.00	90.0	0.80	180.0	--	--	--	--
Milk	270.0	1.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	30.0	1.0	1.00	90.0	0.80	180.0	--	--	--	--
General population										
Beef	70.0	34.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	8.5	34.0	1.0	90.0	0.80	180.0	--	--	--	--
Milk	230.0	3.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	20.0	18.0	1.0	90.0	0.80	180.0	--	--	--	--

Key: kg/yr, kilograms per year; kgs/m^2, kilograms per square meter.
Source: Napier et al. 1988.

It is possible that an individual member of the public could be closer to a facility than either the site boundary or the nearest onsite highway. However, such individuals would be present only occasionally and for brief periods (a few hours or more). Hence, the annual probability that an individual would be close is relatively low, and the associated risk to that individual would be bounded by the site boundary or onsite highway maximally exposed individual risk.

For all of the processing facilities, accidents were evaluated using 50 percent meteorology. The meteorological data used were in the form of joint frequency data files at each site location. The joint frequency data files were based on measurements taken over a period of several years at different locations and heights. Population distributions were based on the 1990 Census of Population and Housing (DOC 1992). Projections were determined for the year 2020 (approximate midlife of operations) for areas within 80 kilometers (50 miles) of each facility.

Consequences to involved workers are addressed in Section I.1.7.

In addition to the GENII computer code, the MACCS2 computer code was used for the localized solvent fire accident analysis. GENII does not permit interdiction or the disposal of food and therefore is inappropriate for the ingestion pathway analysis for the solvent fire accident. The iodine release postulated during the fire accident is sufficient to prompt interdiction. Therefore, GENII was used for the inhalation and external exposure pathways, and MACCS2 was used for the ingestion pathway. The doses via the two pathways were then summed for the total dose.

I.1.6 Processing Facility Accident Consequences and Risks

The processing facility accident source term data presented in Sections I.1.4.1 and I.1.4.2 were evaluated using the GENII accident analysis computer code. The MACCS2 computer code was used in addition to GENII for the medical, industrial, and research and development isotope localized solvent fire accident. **Tables I–50** through **I–54** summarize the consequences and risks of the accidents, with mean meteorological conditions, for the maximally exposed individual, the offsite population within an 80-kilometer (50-mile) radius of the facility, and a noninvolved worker 640 meters (2,100 feet) from the release point.

Table I–50 presents REDC accident consequences and risks for plutonium-238 fabrication and processing.

Table I–51 presents FDPF accident consequences and risks for plutonium-238 fabrication and processing.

Table I–52 presents FMEF accident consequences and risks for plutonium-238 fabrication and processing and for simultaneous plutonium-238, medical, industrial, and research and development isotope fabrication and processing.

Table 1–53 presents RPL accident consequences and risks for medical, industrial, and research and development isotope fabrication and processing.

Table I–54 presents generic support facility accident consequences and risks for medical, industrial, and research and development isotope fabrication and processing.

Table I–50 REDC Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Ion exchange explosion during neptunium-237 target fabrication (0.01)	6.13×10^{-9}	3.06×10^{-12}	3.06×10^{-14}	8.58×10^{-5}	4.29×10^{-8}	4.29×10^{-10}	5.60×10^{-10}	2.24×10^{-13}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	1.76×10^{-7}	8.79×10^{-11}	8.79×10^{-13}	0.00196	9.82×10^{-7}	9.82×10^{-9}	1.69×10^{-8}	6.74×10^{-12}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	4.68×10^{-4}	2.34×10^{-7}	2.34×10^{-9}	5.23	0.00261	2.61×10^{-5}	4.49×10^{-5}	1.79×10^{-8}	1.79×10^{-10}
Plutonium-238 processing facility beyond design-basis earthquake (1×10^{-5})	163	0.163	1.63×10^{-6}	8.91×10^{5}	445	0.00445	1,310	1.00[e]	1.00×10^{-5}[f]

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Early fatality due to radiation dose assuming the accident occurred. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.
f. Increased likelihood of an early fatality per year.

Key: REDC, Radiochemical Engineering Development Center.

Source: Model results, using the GENII computer code (Napier et al. 1988).

Table I–51 FDPF Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Ion exchange explosion during neptunium-237 target fabrication (0.01)	2.01×10^{-9}	1.01×10^{-12}	1.01×10^{-14}	2.49×10^{-5}	1.24×10^{-8}	1.24×10^{-10}	7.26×10^{-9}	2.91×10^{-12}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation (0.01)	6.11×10^{-8}	3.05×10^{-11}	3.05×10^{-13}	5.65×10^{-4}	2.82×10^{-7}	2.82×10^{-9}	2.17×10^{-7}	8.69×10^{-11}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation (0.01)	1.63×10^{-5}	8.13×10^{-9}	8.13×10^{-11}	0.150	7.51×10^{-5}	7.51×10^{-7}	5.79×10^{-5}	2.31×10^{-8}	2.31×10^{-10}
Plutonium-238 processing facility beyond design-basis earthquake (1×10^{-5})	42.5	0.0425	4.25×10^{-7}	1.64×10^{5}	82.0	8.20×10^{-4}	1,200	1.00[e]	1.00×10^{-5}[f]

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Early fatality due to radiation dose assuming the accident occurred. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those exposed. Early fatalities are expected for exposures greater than 600 rem.
f. Increased likelihood of an early fatality per year.

Key: FDPF, Fluorinel Dissolution Process Facility.

Source: Model results, using the GENII computer code (Napier et al. 1988).

Table I–52 FMEF Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Plutonium-238 processing									
Ion Exchange explosion during neptunium-237 target fabrication (0.01)	2.02×10^{-9}	1.01×10^{-12}	1.01×10^{-14}	7.26×10^{-5}	3.63×10^{-8}	3.63×10^{-10}	6.65×10^{-10}	2.66×10^{-13}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation (0.01)	4.64×10^{-8}	2.32×10^{-11}	2.32×10^{-13}	0.00169	8.47×10^{-7}	8.47×10^{-9}	1.95×10^{-8}	7.81×10^{-12}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation (0.01)	1.24×10^{-5}	6.18×10^{-9}	6.18×10^{-11}	0.451	2.25×10^{-4}	2.25×10^{-6}	5.20×10^{-6}	2.08×10^{-9}	2.08×10^{-11}
Plutonium-238 processing only									
Plutonium-238 processing facility beyond-design-basis earthquake (1×10^{-5})	16.5	0.00823	8.23×10^{-8}	6.41×10^{5}	321	0.00321	921	1.00[e]	1.00×10^{-5}[(f)]
Medical and industrial isotope processing									
Medical and industrial isotopes localized solvent fire (0.044)	0.00276	1.38×10^{-6}	6.13×10^{-8}	56.2	0.0281	0.00125	9.51×10^{-5}	3.80×10^{-8}	1.69×10^{-9}
Medical and industrial isotopes glovebox explosion (1×10^{-4})	1.00	5.00×10^{-4}	5.00×10^{-8}	2.95×10^{4}	14.8	0.00148	24.0	0.0192	1.92×10^{-6}
Medical and industrial isotope and plutonium-238 processing									
Processing facility beyond-design-basis earthquake (1×10^{-5})	16.5	0.00825	8.25×10^{-8}	6.42×10^{5}	321	0.00321	922	1.00[e]	1.00×10^{-5}[(f)]

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.
e. Early fatality due to radiation dose assuming the accident occurred. A radiation dose of 450 to 500 rem causes fatalities in 50 percent of those e×posed. Early fatalities are expected for exposures greater than 600 rem.
f. Increased likelihood of an early fatality per year.

Key: FMEF, Fuels and Materials Examination Facility.

Source: Model results, using the GENII computer code (Napier et al. 1998) and the MACCS2 computer code (Chanin and Young 1997).

Table I–53 RPL Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Medical and industrial isotopes localized solvent fire (0.044)	0.0135	6.74×10^{-6}	2.99×10^{-7}	77.8	0.0389	0.00173	0.00470	1.88×10^{-6}	8.35×10^{-8}
Medical and industrial isotopes unlikely seismic event (0.01)	1.52	7.60×10^{-4}	7.60×10^{-6}	1,350	0.675	0.00675	1.50	6.00×10^{-4}	6.00×10^{-6}
Medical and industrial isotopes glovebox explosion (1.00×10^{-4})	50.0	0.050	5.00×10^{-6}	4.60×10^{4}	23.0	0.00230	49.0	0.0392	3.92×10^{-6}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Key: RPL, Radiochemical Processing Laboratory.

Source: Model results, using the GENII computer code (Napier et al. 1988).

Table I–54 Generic Support Facility Accident Consequences and Risks

Accident (Frequency)	Maximally Exposed Individual			Population to 80 Kilometers (50 Miles)			Noninvolved Worker		
	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Annual Risk[d]	Dose (rem)	Latent Cancer Fatality[a]	Annual Risk[b]
Medical and industrial localized solvent fire (0.044)	0.0194	9.72×10^{-6}	4.32×10^{-7}	31.1	0.0156	6.91×10^{-4}	0.00530	2.12×10^{-6}	9.41×10^{-8}
Medical and industrial unlikely seismic event (0.01)	0.0750	3.75×10^{-5}	3.75×10^{-7}	136	0.0680	6.80×10^{-4}	0.510	2.04×10^{-4}	2.04×10^{-6}
Medical and industrial glovebox explosion (1.00×10^{-4})	2.50	0.00125	1.25×10^{-7}	4,600	2.30	2.30×10^{-4}	17.0	0.00680	6.80×10^{-7}

a. Likelihood of a latent cancer fatality assuming the accident occurred.
b. Increased likelihood of a latent cancer fatality per year.
c. Number of latent cancer fatalities assuming the accident occurred.
d. Increased number of latent cancer fatalities per year.

Source: Model results, using the GENII (Napier et al. 1988) and MACCS2 (Chanin and Young 1997) computer codes.

I.1.7 Involved Worker Accident Consequences and Risks

The estimated number of involved workers at each of the proposed irradiation and processing facilities is shown in **Table I–55**.

Table I–55 Estimated Number of Involved Workers at Each Facility

Facility	Type	Involved Workers
FFTF	Irradiation	242
ATR	Irradiation	100
HFIR	Irradiation	100
CLWR	Irradiation	300
New research reactor	Irradiation	120
High-energy accelerator	Irradiation	225
Low-energy accelerator	Irradiation	13
REDC	Processing	116
FDPF	Processing	75
FMEF	Processing	105
RPL/306–E	Processing	30
New support facility	Processing	100

I.1.7.1 Irradiation Facility Consequences and Risks

I.1.7.1.1 Design-Basis Accident

Each of the proposed irradiation facilities would have an approved onsite emergency plan. The likelihood of a design-basis accident is estimated to be once in 10,000 years. Since an accident could occur at any given time, the number of workers on site at the time of an accident would be unlikely to exceed one-third the total number of involved workers shown above (assuming a three-shift operation). The workers at the facility at the time of a design-basis accident can be grouped into three major categories, as follows:

1. Those workers not having duties associated with accident management or recovery. These would be promptly notified and evacuated from the site. Individuals in this group would be expected to receive low doses significantly below the EPA Protective Action Guides (PAGs) (EPA 1992) of 1-5 rem. Most involved workers would be in this group.

2. Those workers located in shielded areas such as the control room or other designated plant emergency operation areas having duties associated with accident management and recovery. These workers, because of the radiation protection afforded by their locations, would be unlikely to receive doses in excess of the EPA PAGs. For the irradiation facilities, this group is estimated to range from about 6 to 20 individuals.

3. Those few workers in areas of the plant who may be directly affected or impacted by the accident, (e.g., performing maintenance in the immediate area where an accident initiating event occurs). This very small group of involved workers could receive significant doses in excess of the EPA PAGs. With appropriate radiation instrumentation, alarms and administrative controls, it is unlikely that individuals in this group would receive doses high enough to result in acute radiation effects (doses greater than about 100 rem).

I.1.7.1.2 Beyond-Design-Basis Accident

The likelihood of a beyond-design-basis accident is estimated to be once in 100,000 years. A beyond-design-basis accident may begin as a design-basis accident, but would involve additional equipment failures that lead to more serious reactor or facility damage than in a design basis event. For this reason, the previous discussion on design-basis events is largely applicable. Most involved workers would likely be evacuated prior to receiving any significant dose. A small group of workers, including operators and personnel directly involved in accident management and recovery could receive significant doses, however. One or two individuals could conceivably receive high doses if emergency actions were to be taken (e.g., entering a high radiation area for a short time to actuate a valve or pump).

I.1.7.2 Processing Facility Consequences and Risks

I.1.7.2.1 Design-Basis Accident

GLOVEBOX EXPLOSION DURING NEPTUNIUM-237 TARGET FABRICATION

For the purposes of this NI PEIS, this accident frequency is estimated to be 1×10^{-2} per year. Assuming this accident occurs, the involved worker at the affected glovebox may be seriously injured as a result of the explosion as well as likely to be contaminated with the explosion debris. This worker could receive a significant radiation dose. The extent of the contamination and the radiation doses are likely to be highly localized, however. Neighboring workers in nearby gloveboxes will be exposed to significantly lower doses and effects from the explosion, while workers in other locations in the processing facility will be only minimally affected.

TARGET DISSOLVER TANK FAILURE DURING PLUTONIUM-238 SEPARATION

For the purposes of this NI PEIS, this accident frequency is estimated to be 1×10^{-2} per year. This accident is postulated to occur in a shielded hot cell, whose integrity is not challenged by this accident. Consequently, those workers outside the hot cell carrying out this operation will not be affected. Some plutonium-238 and neptunium-237 will be released from an elevated stack after passing through two stages of HEPA filters, which removes all but a very small fraction of the spilled tank contents. Workers at the processing facility will be exposed to very low concentrations of plutonium-238 and neptunium-237 as a result of this release. It is estimated that since an elevated release results in very low concentrations at ground level close to the release point, that worker doses will be generally similar to those received by the maximally exposed offsite individual for this accident.

ION EXCHANGE EXPLOSION DURING PLUTONIUM-238 SEPARATION

For the purposes of this NI PEIS, this accident frequency is estimated to be 1×10^{-2} per year. As for the postulated dissolver tank failure discussed above, this accident is postulated to occur in a shielded hot cell, whose integrity is unlikely to be challenged by this accident. Consequently, those workers outside the hot cell carrying out this operation will not be affected. Some plutonium-238 will be released from an elevated stack after passing through two stages of HEPA filters, which removes all but a very small fraction of the explosion debris. Workers at the processing facility will be exposed to very low concentrations of plutonium-238 as a result of this release. It is estimated that since an elevated release results in very low concentrations at ground level close to the release point, that worker doses will be generally similar to those received by the maximally exposed offsite individual for this accident.

I.1.7.2.2 Beyond-Design-Basis Accident

The beyond-design-basis accident postulated for the processing facilities is a catastrophic earthquake whose likelihood is taken to be once in 100,000 years for this NI PEIS. The earthquake is postulated to collapse the stack, severely damaging the HEPA filters. Although the building is expected to collapse, the hot cells are expected to remain intact, but with cracked walls. In addition, one or more of the shielded viewing windows may be cracked and broken. The ventilation systems exhausting from the hot cells are expected to fail. Radioactive materials in the hot cells will be released as a result of cracks in cell walls and shielded windows, but the rate of leakage is expected to be low, since the hot cells are not pressurized and there is no forced ventilation.

Many of the workers in the processing facility are expected to be injured as a direct result of the earthquake. Those workers who are mobile are expected to leave the facility, and this group is not likely to receive any significant radiation dose. Workers who are trapped in the rubble and debris of the earthquake and unable to leave the immediate vicinity could receive significant additional radiation doses, however.

I.1.7.3 Medical and Industrial Isotopes Processing Facility Consequences and Risks

I.1.7.3.1 Localized Solvent Fire

This event postulates a localized solvent fire in a hot cell. Because only two events potentially involving radioactive materials have occurred in laboratory facilities over a 45-year period, a frequency of 4.44×10^{-2} per year was assumed for this accident. The integrity of the hot cell would not be challenged by the localized nature of the fire, and the final HEPA and iodine-removing charcoal filters were assumed to be unaffected by the fire because they are located away from the hot cells. Some radioactive materials would be released after passing through the filters, which would remove all but a small fraction of the materials reaching them. Workers at the processing facility would be unaffected by the fire, but would be exposed to low concentrations of released radioactive isotopes. It is estimated that worker doses would generally be similar to those received by the maximally exposed individual (0.0135 rem for RPL and 0.0194 rem for the generic support facility) for this accident.

I.1.7.3.2 Unlikely Seismic Event

For the purposes of this NI PEIS, this accident frequency is assumed to be 1×10^{-2} per year. The facility's superstructure was assumed to remain intact, but the heating, ventilating, and air-conditioning system was assumed to fail because it has not been seismically qualified. Some injuries may occur because of the seismic event itself. Releases of radioactive materials would be at ground level. It is estimated that worker doses would generally be somewhat greater than those received by the maximally exposed individual (1.52 rem for RPL and 0.075 rem for the generic support facility) for this accident.

I.1.7.3.3 Loss of Electrical Power and Explosion

For the purposes of this NI PEIS, this accident frequency is assumed to be 1×10^{-4} per year. This scenario assumes a loss of electrical power that inactivates the ventilation system. This is postulated to lead to an explosion in a glovebox as a result of a buildup of flammable vapors. Most of the airborne material within a glovebox would be carried out through the exhaust system. For this scenario, the explosion is assumed to be sufficiently forceful to render the HEPA filters and radon holdup system ineffective. Assuming this accident occurs, the involved worker at the affected glovebox may be seriously injured as a result of the explosion and may be contaminated with explosion debris. This worker could receive a significant radiation dose. The extent of the contamination and the radiation dose is likely to be highly localized, however. Neighboring

workers in nearby gloveboxes, if any, would be exposed to significantly lower doses and effects from the explosion. Releases of radioactive materials would be at ground level. It is estimated that worker doses at the facility (except for the worker directly affected by the explosion) would generally be somewhat greater than those received by the maximally exposed individual (50.0 rem for RPL and 2.5 rem for the generic support facility) for this accident.

I.1.8 Risk Summation

To provide a better indication of risks of the postulated accidents, the risks are summed for each facility and also for each option. The summed risks for each alternative and option are presented in **Tables I–56** through **I–76**. Although the summation provides the combined risk for the spectrum of accidents analyzed, it does not indicate total risk. To determine total risk of accidents, a full-scope (Level 3) probabilistic risk analysis is required for each facility. However, since full-scope probabilistic risk analyses are not available to incorporate into this NI PEIS, summation of the spectrum of accident risks was considered appropriate.

As explained previously, a full spectrum of accidents was considered at the irradiation and fabrication and processing facilities. The accidents evaluated represent bounding cases that are considered to envelop the risk profile.

For each option, the highest risks are presented for the maximally exposed individual and the noninvolved worker. The highest risk to an individual may result from either a single facility or a combination of facilities. A combination of facilities can occur only if the facilities are colocated. In this case, the individual risks are summed. For each option, all facility population risks are summed.

For the currently operating reactors (ATR, HFIR, and CLWR), the incremental risk of target irradiation is determined by subtracting the risk without target irradiation from the risk with target irradiation. For example, in Alternative 2, Option 1, there is an incremental risk to the maximally exposed individual from a large-break loss-of-coolant accident. The incremental risk is determined by subtracting the maximally exposed individual risk from a large-break loss-of-coolant accident without target irradiation (i.e., 0 kilograms per year plutonium-238 production) from the maximally exposed individual risk with target irradiation (i.e., 5 kilograms per year plutonium-238 production). The incremental risk is therefore 3.02×10^{-8} - 2.33×10^{-8} = 6.09×10^{-9} as presented in the table. The incremental risks are used to determine the summed risks. Therefore, summing every risk presented in the tables will not directly produce the summed risk.

Table I–56 Risk Summation for Alternative 1—Option 1

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (21 years with MOX, 14 years with HEU)	1.23×10^{-8}	1.27×10^{-3}	1.20×10^{-8}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
RPL			
Medical and industrial isotope localized solvent fire	2.99×10^{-7}	1.73×10^{-3}	8.35×10^{-8}
Unlikely seismic event	7.60×10^{-6}	6.75×10^{-3}	6.00×10^{-6}
Medical and industrial isotope glovebox explosion	5.00×10^{-6}	2.30×10^{-3}	3.92×10^{-6}
RPL risk summation	1.29×10^{-5}	1.08×10^{-2}	1.00×10^{-5}
35-year RPL risk summation	4.51×10^{-4}	3.77×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 1	$\mathbf{4.51\times10^{-4}}$	$\mathbf{5.35\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–57 Risk Summation for Alternative 1—Option 2

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (21 years with MOX, 14 years with HEU)	1.23×10^{-8}	1.27×10^{-3}	1.20×10^{-8}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
RPL			
Medical and industrial isotope localized solvent fire	2.99×10^{-7}	1.73×10^{-3}	8.35×10^{-8}
Unlikely seismic event	7.60×10^{-6}	6.75×10^{-3}	6.00×10^{-6}
Medical and industrial isotope glovebox explosion	5.00×10^{-6}	2.30×10^{-3}	3.92×10^{-6}
RPL risk summation	1.29×10^{-5}	1.08×10^{-2}	1.00×10^{-5}
35-year RPL risk summation	4.51×10^{-4}	3.77×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 2	$\mathbf{4.51\times10^{-4}}$	$\mathbf{4.07\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–58 Risk Summation for Alternative 1—Option 3

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (21 years with MOX, 14 years with HEU)	1.23×10^{-8}	1.27×10^{-3}	1.20×10^{-8}
FMEF (full processing)			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Medical and industrial isotope localized solvent fire	6.13×10^{-8}	1.25×10^{-3}	1.69×10^{-9}
Medical and industrial isotope glovebox explosion	5.00×10^{-8}	1.48×10^{-3}	1.92×10^{-6}
Beyond-design-basis earthquake	8.25×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	1.94×10^{-7}	5.94×10^{-3}	1.19×10^{-5}
35-year FMEF risk summation	6.79×10^{-6}	2.08×10^{-1}	4.17×10^{-4}
35-year risk summation for Option 3	**6.80×10^{-6}**	**2.09×10^{-1}**	**4.17×10^{-4}**

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–59 Risk Summation for Alternative 1—Option 4

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (6 years with MOX, 29 years with HEU)	1.06×10^{-8}	1.22×10^{-3}	9.37×10^{-9}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
RPL			
Medical and industrial isotope localized solvent fire	2.99×10^{-7}	1.73×10^{-3}	8.35×10^{-8}
Unlikely seismic event	7.60×10^{-6}	6.75×10^{-3}	6.00×10^{-6}
Medical and industrial isotope glovebox explosion	5.00×10^{-6}	2.30×10^{-3}	3.92×10^{-6}
RPL risk summation	1.29×10^{-5}	1.08×10^{-2}	1.00×10^{-5}
35-year RPL risk summation	4.51×10^{-4}	3.77×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 4	**4.51×10^{-4}**	**5.35×10^{-1}**	**3.50×10^{-4}**

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–60 Risk Summation for Alternative 1—Option 5

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (6 years with MOX, 29 years with HEU)	1.06×10^{-8}	1.22×10^{-3}	9.37×10^{-9}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
RPL			
Medical and industrial isotope localized solvent fire	2.99×10^{-7}	1.73×10^{-3}	8.35×10^{-8}
Unlikely seismic event	7.60×10^{-6}	6.75×10^{-3}	6.00×10^{-6}
Medical and Industrial isotope glovebox explosion	5.00×10^{-6}	2.30×10^{-3}	3.92×10^{-6}
RPL risk summation	1.29×10^{-5}	1.08×10^{-2}	1.00×10^{-5}
35-year RPL risk summation	4.51×10^{-4}	3.77×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 5	**4.51×10^{-4}**	**4.07×10^{-1}**	**3.50×10^{-4}**

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–61 Risk Summation for Alternative 1—Option 6

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
FFTF			
Design-basis primary sodium spill (MOX)	5.65×10^{-11}	3.93×10^{-6}	1.25×10^{-10}
Design-basis primary sodium spill (HEU)	4.32×10^{-11}	3.63×10^{-6}	7.24×10^{-11}
Beyond-design-basis core melt accident (MOX)	3.40×10^{-10}	3.34×10^{-5}	2.72×10^{-10}
Beyond-design-basis core melt accident (HEU)	2.41×10^{-10}	3.08×10^{-5}	1.50×10^{-10}
BLTC driver fuel-handling accident (MOX)	1.92×10^{-13}	6.39×10^{-8}	1.43×10^{-11}
BLTC driver fuel-handling accident (HEU)	1.92×10^{-13}	6.17×10^{-8}	1.36×10^{-11}
BLTC neptunium-237 target-handling accident	1.31×10^{-14}	1.29×10^{-9}	1.12×10^{-12}
BLTC isotope target-handling accident	6.10×10^{-15}	1.37×10^{-10}	5.72×10^{-13}
FFTF risk summation (MOX)	3.97×10^{-10}	3.74×10^{-5}	4.13×10^{-10}
FFTF risk summation (HEU)	2.84×10^{-10}	3.45×10^{-5}	2.38×10^{-10}
35-year FFTF risk summation (6 years with MOX, 29 years with HEU)	1.06×10^{-8}	1.22×10^{-3}	9.37×10^{-9}
FMEF (full processing)			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Medical and industrial isotope localized solvent fire	6.13×10^{-8}	1.25×10^{-3}	1.69×10^{-9}
Medical and industrial isotope glovebox explosion	5.00×10^{-8}	1.48×10^{-3}	1.92×10^{-6}
Beyond-design-basis earthquake	8.25×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	1.94×10^{-7}	5.94×10^{-3}	1.19×10^{-5}
35-year FMEF risk summation	6.79×10^{-6}	2.08×10^{-1}	4.17×10^{-4}
35-year risk summation for Option 6	$\mathbf{6.80\times10^{-6}}$	$\mathbf{2.09\times10^{-1}}$	$\mathbf{4.17\times10^{-4}}$

Key: BLTC, bottom-loading transfer cask; HEU, highly enriched uranium fuel; MOX, mixed oxide fuel.

Table I–62 Risk Summation for Alternative 2—Option 1

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production	3.02×10^{-8}	2.59×10^{-3}	3.04×10^{-7}
Large-break LOCA incremental risk	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target-handling accident incremental risk	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
ATR risk summation	7.00×10^{-9}	4.01×10^{-5}	9.93×10^{-8}
35-year ATR risk summation	2.45×10^{-7}	1.40×10^{-3}	3.48×10^{-6}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target Dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 1	$\mathbf{5.71\times10^{-5}}$	$\mathbf{1.58\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–63 Risk Summation for Alternative 2—Option 2

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production	3.02×10^{-8}	2.59×10^{-3}	3.04×10^{-7}
Large-break LOCA incremental risk	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target-handling accident incremental risk	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
ATR risk summation	7.00×10^{-9}	4.01×10^{-5}	9.93×10^{-8}
35-year ATR risk summation	2.45×10^{-7}	1.40×10^{-3}	3.48×10^{-6}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 2	$\mathbf{1.51\times10^{-5}}$	$\mathbf{3.01\times10^{-2}}$	$\mathbf{3.53\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–64 Risk Summation for Alternative 2—Option 3

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 5 kg/yr plutonium-238 production	3.02×10^{-8}	2.59×10^{-3}	3.04×10^{-7}
Large-break LOCA incremental risk	6.90×10^{-9}	4.00×10^{-5}	9.80×10^{-8}
Neptunium-237 target-handling accident incremental risk	1.03×10^{-10}	6.41×10^{-8}	1.30×10^{-9}
ATR risk summation	7.00×10^{-9}	4.01×10^{-5}	9.93×10^{-8}
35-year ATR risk summation	2.45×10^{-7}	1.40×10^{-3}	3.48×10^{-6}
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 3	$\mathbf{2.88\times10^{-6}}$	$\mathbf{1.14\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–65 Risk Summation for Alternative 2—Option 4

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
CLWR			
Design-basis large-break with 0 kg/yr plutonium-238 production	7.25×10^{-10}	4.33×10^{-6}	NA
Design-basis large-break with 5 kg/yr plutonium-238 production	7.30×10^{-10}	4.35×10^{-6}	NA
Early containment failure with 0 kg/yr plutonium-238 production	7.92×10^{-8}	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production	7.92×10^{-8}	1.06×10^{-4}	NA
Late containment failure with 0 kg/yr plutonium-238 production	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production	5.99×10^{-9}	5.74×10^{-4}	NA
Containment bypass with 0 kg/yr plutonium-238 production	1.53×10^{-6}	1.41×10^{-3}	NA
Containment bypass with 5 kg/yr plutonium-238 production	1.53×10^{-6}	1.49×10^{-3}	NA
Design-basis large-break LOCA incremental risk	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure incremental risk	0.00	7.10×10^{-6}	NA
Late containment failure incremental risk	5.00×10^{-11}	0.00	NA
Containment bypass incremental risk	0.00	8.00×10^{-5}	NA
CLWR risk summation	5.50×10^{-11}	8.71×10^{-5}	NA
35-year CLWR risk summation	1.93×10^{-9}	3.05×10^{-3}	NA
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 4	$\mathbf{5.71\times10^{-5}}$	$\mathbf{1.60\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–66 Risk Summation for Alternative 2—Option 5

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
CLWR			
Design-basis large-break with 0 kg/yr plutonium-238 production	7.25×10^{-10}	4.33×10^{-6}	NA
Design-basis large-break with 5 kg/yr plutonium-238 production	7.30×10^{-10}	4.35×10^{-6}	NA
Early containment failure with 0 kg/yr plutonium-238 production	7.92×10^{-8}	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production	7.92×10^{-8}	1.06×10^{-4}	NA
Late containment failure with 0 kg/yr plutonium-238 production	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production	5.99×10^{-9}	5.74×10^{-4}	NA
Containment bypass with 0 kg/yr plutonium-238 production	1.53×10^{-6}	1.41×10^{-3}	NA
Containment bypass with 5 kg/yr plutonium-238 production	1.53×10^{-6}	1.49×10^{-3}	NA
Design-basis large-break LOCA incremental risk	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure incremental risk	0.00	7.10×10^{-6}	NA
Late containment failure incremental risk	5.00×10^{-11}	0.00	NA
Containment bypass incremental risk	0.00	8.00×10^{-5}	NA
CLWR risk summation	5.50×10^{-11}	8.71×10^{-5}	NA
35-year CLWR risk summation	1.93×10^{-9}	3.05×10^{-3}	NA
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 5	$\mathbf{1.49\times10^{-5}}$	$\mathbf{3.18\times10^{-2}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–67 Risk Summation for Alternative 2—Option 6

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
CLWR			
Design-basis large-break with 0 kg/yr plutonium-238 production	7.25×10^{-10}	4.33×10^{-6}	NA
Design-basis large-break with 5 kg/yr plutonium-238 production	7.30×10^{-10}	4.35×10^{-6}	NA
Early containment failure with 0 kg/yr plutonium-238 production	7.92×10^{-8}	9.89×10^{-5}	NA
Early containment failure with 5 kg/yr plutonium-238 production	7.92×10^{-8}	1.06×10^{-4}	NA
Late containment failure with 0 kg/yr plutonium-238 production	5.94×10^{-9}	5.74×10^{-4}	NA
Late containment failure with 5 kg/yr plutonium-238 production	5.99×10^{-9}	5.74×10^{-4}	NA
Containment bypass with 0 kg/yr plutonium-238 production	1.53×10^{-6}	1.41×10^{-3}	NA
Containment bypass with 5 kg/yr plutonium-238 production	1.53×10^{-6}	1.49×10^{-3}	NA
Design-basis large-break LOCA incremental risk	5.00×10^{-12}	2.00×10^{-8}	NA
Early containment failure incremental risk	0.00	7.10×10^{-6}	NA
Late containment failure incremental risk	5.00×10^{-11}	0.00	NA
Containment bypass incremental risk	0.00	8.00×10^{-5}	NA
CLWR risk summation	5.50×10^{-11}	8.71×10^{-5}	NA
35-year CLWR risk summation	1.93×10^{-9}	3.05×10^{-3}	NA
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 6	**2.88×10^{-6}**	**1.15×10^{-1}**	**3.50×10^{-4}**

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–68 Risk Summation for Alternative 2—Option 7

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production	2.75×10^{-8}	2.57×10^{-3}	2.61×10^{-7}
Large-break LOCA incremental risk	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target-handling accident incremental risk	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
ATR risk summation	4.26×10^{-9}	2.00×10^{-5}	5.58×10^{-8}
35-year ATR risk summation	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
HFIR			
Large-break LOCA with 0 kg/yr plutonium-238 production	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risk	0.00	1.00×10^{-6}	0.00
Neptunium-237 target-handling accident incremental risk	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
HFIR risk summation	2.48×10^{-10}	1.17×10^{-6}	9.80×10^{-10}
35-year HFIR risk summation	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 7	$\mathbf{5.71\times10^{-5}}$	$\mathbf{1.57\times10^{-1}}$	$\mathbf{3.50\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–69 Risk Summation for Alternative 2—Option 8

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production	2.75×10^{-8}	2.57×10^{-3}	2.61×10^{-7}
Large-break LOCA incremental risk	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target-handling accident incremental risk	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
ATR risk summation	4.26×10^{-9}	2.00×10^{-5}	5.58×10^{-8}
35-year ATR risk summation	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
HFIR			
Large-break LOCA with 0 kg/yr plutonium-238 production	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risk	0.00	1.00×10^{-6}	0.00
Neptunium-237 target-handling accident incremental risk	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
HFIR risk summation	2.48×10^{-10}	1.17×10^{-6}	9.80×10^{-10}
35-year HFIR risk summation	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 8	$\mathbf{1.50\times10^{-5}}$	$\mathbf{2.95\times10^{-2}}$	$\mathbf{3.52\times10^{-4}}$

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–70 Risk Summation for Alternative 2—Option 9

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
ATR			
Large-break LOCA with 0 kg/yr plutonium-238 production	2.33×10^{-8}	2.55×10^{-3}	2.06×10^{-7}
Large-break LOCA with 3 kg/yr plutonium-238 production	2.75×10^{-8}	2.57×10^{-3}	2.61×10^{-7}
Large-break LOCA incremental risk	4.20×10^{-9}	2.00×10^{-5}	5.50×10^{-8}
Neptunium-237 target-handling accident incremental risk	6.15×10^{-11}	3.93×10^{-8}	7.80×10^{-10}
ATR risk summation	4.26×10^{-9}	2.00×10^{-5}	5.58×10^{-8}
35-year ATR risk summation	1.49×10^{-7}	7.01×10^{-4}	1.95×10^{-6}
HFIR			
Large-break LOCA with 0 kg/yr plutonium-238 production	1.21×10^{-7}	1.49×10^{-4}	6.88×10^{-7}
Large-break LOCA with 2 kg/yr plutonium-238 production	1.21×10^{-7}	1.50×10^{-4}	6.88×10^{-7}
Large-break LOCA incremental risk	0.00	1.00×10^{-6}	0.00
Neptunium-237 target-handling accident incremental risk	2.48×10^{-10}	1.68×10^{-7}	9.80×10^{-10}
HFIR risk summation	2.48×10^{-10}	1.17×10^{-6}	9.80×10^{-10}
35-year HFIR risk summation	8.68×10^{-9}	4.09×10^{-5}	3.43×10^{-8}
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 9	**2.88×10^{-6}**	**1.13×10^{-1}**	**3.50×10^{-4}**

Key: kg/yr, kilograms per year; LOCA, loss-of-coolant accident.

Table I–71 Risk Summation for Alternative 3—Option 1

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Low-energy accelerator			
Design-basis target-handling accident	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
Low-energy accelerator risk summation	7.00×10^{-11}	1.05×10^{-6}	8.77×10^{-10}
35-year low-energy accelerator risk summation	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
High-energy accelerator			
Design-basis target-handling accident	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
High-energy accelerator risk summation	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk summation	2.05×10^{-6}	6.30×10^{-3}	5.15×10^{-5}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 1	$\mathbf{9.18\times10^{-5}}$	$\mathbf{2.19\times10^{-1}}$	$\mathbf{5.00\times10^{-4}}$

Table I–72 Risk Summation for Alternative 3—Option 2

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Low-energy accelerator			
Design-basis target-handling accident	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
Low-energy accelerator risk summation	7.00×10^{-11}	1.05×10^{-6}	8.77×10^{-10}
35-year low-energy accelerator risk summation	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
High-energy accelerator			
Design-basis target-handling accident	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
High-energy accelerator risk summation	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk summation	2.05×10^{-6}	6.30×10^{-3}	5.15×10^{-5}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 2	**4.95×10^{-5}**	**9.11×10^{-2}**	**5.00×10^{-4}**

Table I–73 Risk Summation for Alternative 3—Option 3

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Low-energy accelerator			
Design-basis target-handling accident	4.03×10^{-12}	8.85×10^{-7}	4.48×10^{-11}
Beyond-design-basis earthquake	6.60×10^{-11}	1.62×10^{-7}	8.32×10^{-10}
Low-energy accelerator risk summation	7.00×10^{-11}	1.05×10^{-6}	8.77×10^{-10}
35-year low-energy accelerator risk summation	2.45×10^{-9}	3.66×10^{-5}	3.07×10^{-8}
High-energy accelerator			
Design-basis target-handling accident	1.47×10^{-11}	4.90×10^{-8}	3.74×10^{-11}
Beyond-design-basis earthquake	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
High-energy accelerator risk summation	5.85×10^{-8}	1.80×10^{-4}	1.47×10^{-6}
35-year high-energy accelerator risk summation	2.05×10^{-6}	6.30×10^{-3}	5.15×10^{-5}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 3	**3.76×10^{-5}**	**1.75×10^{-1}**	**5.00×10^{-4}**

Table I–74 Risk Summation for Alternative 4—Option 1

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Research reactor			
Design-basis accident	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis accident	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical, industrial, research and development isotope target-handling accident	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
Research reactor risk summation	7.10×10^{-11}	6.45×10^{-7}	2.40×10^{-10}
35-year research reactor risk summation	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
REDC			
Ion exchange explosion during neptunium-237 target fabrication	3.06×10^{-14}	4.29×10^{-10}	2.24×10^{-15}
Target dissolver tank failure during plutonium-238 separation	8.79×10^{-13}	9.82×10^{-9}	6.74×10^{-14}
Ion exchange explosion during plutonium-238 separation	2.34×10^{-9}	2.61×10^{-5}	1.79×10^{-10}
Beyond-design-basis earthquake	1.63×10^{-6}	4.45×10^{-3}	1.00×10^{-5}
REDC risk summation	1.63×10^{-6}	4.48×10^{-3}	1.00×10^{-5}
35-year REDC risk summation	5.71×10^{-5}	1.57×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 1	$\mathbf{8.98\times10^{-5}}$	$\mathbf{2.13\times10^{-1}}$	$\mathbf{4.49\times10^{-4}}$

Table I–75 Risk Summation for Alternative 4—Option 2

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Research reactor			
Design-basis accident	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis accident	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical, industrial, research and development isotope target-handling accident	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
Research reactor risk summation	7.10×10^{-11}	6.45×10^{-7}	2.40×10^{-10}
35-year research reactor risk summation	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
FDPF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	1.24×10^{-10}	2.91×10^{-14}
Target dissolver tank failure during plutonium-238 separation	3.05×10^{-13}	2.82×10^{-9}	8.69×10^{-13}
Ion exchange explosion during plutonium-238 separation	8.13×10^{-11}	7.51×10^{-7}	2.31×10^{-10}
Beyond-design-basis earthquake	4.25×10^{-7}	8.20×10^{-4}	1.00×10^{-5}
FDPF risk summation	4.25×10^{-7}	8.21×10^{-4}	1.00×10^{-5}
35-year FDPF risk summation	1.49×10^{-5}	2.87×10^{-2}	3.50×10^{-4}
35-year risk summation for Option 2	**4.75×10^{-5}**	**8.48×10^{-2}**	**4.49×10^{-4}**

Table I–76 Risk Summation for Alternative 4—Option 3

Accident	Maximally Exposed Individual	Population to 80 Kilometers (50 Miles)	Noninvolved Worker (640 Meters)
Research reactor			
Design-basis accident	6.65×10^{-14}	1.20×10^{-10}	2.20×10^{-13}
Beyond-design-basis accident	1.87×10^{-11}	1.38×10^{-7}	2.12×10^{-10}
Fuel-handling accident	9.50×10^{-15}	3.40×10^{-11}	2.33×10^{-14}
Neptunium-237 target-handling accident	2.71×10^{-13}	4.47×10^{-10}	9.72×10^{-13}
Medical, industrial, research and development isotope target-handling accident	5.20×10^{-11}	5.06×10^{-7}	2.70×10^{-11}
Research reactor risk summation	7.10×10^{-11}	6.45×10^{-7}	2.40×10^{-10}
35-year research reactor risk summation	2.49×10^{-9}	2.26×10^{-5}	8.41×10^{-9}
Support facility			
Medical, industrial, and research and development isotope localized solvent fire	4.32×10^{-7}	6.91×10^{-4}	9.41×10^{-8}
Medical, industrial, and research and development isotope unlikely seismic event	3.75×10^{-7}	6.80×10^{-4}	2.04×10^{-6}
Medical, industrial, and research and development isotope glovebox explosion	1.25×10^{-7}	2.30×10^{-4}	6.80×10^{-7}
Support facility risk summation	9.32×10^{-7}	1.60×10^{-3}	2.81×10^{-6}
35-year support facility risk summation	3.26×10^{-5}	5.60×10^{-2}	9.85×10^{-5}
FMEF			
Ion exchange explosion during neptunium-237 target fabrication	1.01×10^{-14}	3.63×10^{-10}	2.66×10^{-15}
Target dissolver tank failure during plutonium-238 separation	2.32×10^{-13}	8.47×10^{-9}	7.81×10^{-14}
Ion exchange explosion during plutonium-238 separation	6.18×10^{-11}	2.25×10^{-6}	2.08×10^{-11}
Beyond-design-basis earthquake	8.23×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
FMEF risk summation	8.24×10^{-8}	3.21×10^{-3}	1.00×10^{-5}
35-year FMEF risk summation	2.88×10^{-6}	1.12×10^{-1}	3.50×10^{-4}
35-year risk summation for Option 3	$\mathbf{3.55\times10^{-5}}$	$\mathbf{1.68\times10^{-1}}$	$\mathbf{4.49\times10^{-4}}$

I.2 HAZARDOUS CHEMICAL ACCIDENT IMPACTS ON HUMAN HEALTH

I.2.1 Irradiation Facility

I.2.1.1 Advanced Test Reactor

Irradiation of neptunium-237 targets to produce plutonium-238 at ATR would not introduce any additional operations that require the use of hazardous chemicals. No hazardous chemical accidents attributable to the irradiation of neptunium-237 targets for plutonium-238 production are postulated at ATR.

I.2.1.2 High Flux Isotope Reactor

Irradiation of neptunium-237 targets to produce plutonium-238 at HFIR would not introduce any additional operations that require the use of hazardous chemicals. No hazardous chemical accidents attributable to the irradiation of neptunium-237 targets for plutonium-238 production are postulated at HFIR.

I.2.1.3 Commercial Light Water Reactor

Irradiation of neptunium-237 targets to produce plutonium-238 at the generic CLWR would not introduce any additional operations that require the use of hazardous chemicals. No hazardous chemical accidents attributable to the irradiation of neptunium-237 for plutonium-238 production are postulated at the generic CLWR.

I.2.1.4 Fast Flux Test Facility

The FFTF primary heat transport system contains a substantial quantity of liquid sodium. In the event of an accident, sodium could be released from the primary system. Sodium mainly forms sodium oxide upon burning; however, in the present of moisture, it can form the more hazardous sodium hydroxide. The beyond-design-basis core melt accident would result in the greatest sodium release to the environment. Previous analyses have shown that the concentration of sodium hydroxide released in this accident scenario would be below the Emergency Response Planning Guidelines-1 (ERPG-1) limit of 2 milligrams per cubic meter at 100 meters (DOE 1997). Since this is the bounding sodium release and all the sodium released was assumed to be in the form of sodium hydroxide, no further analysis was performed for this NI PEIS.

I.2.2 Processing Facility

Processing associated with the plutonium-238 production program at REDC, including storage of neptunium-237 and plutonium-238, neptunium-237 target fabrication, and postirradiation processing to extract plutonium-238 and to recycle the unconverted neptunium-237 into new targets, does not require the introduction of hazardous chemicals that are not in current use in the facility. The quantities of in-process hazardous chemicals for the plutonium-238 production program are bounded by the quantities of the material currently stored in the facility. The impacts of in-process hazardous chemical accidents associated with plutonium-238 production are bounded by the impacts of hazardous chemical accidents for existing storage facilities at REDC.

No chemical processing activities are currently performed at FDPF and FMEF, and no chemicals are stored in these facilities. If either of these facilities is selected to support the plutonium-238 production program, a hazardous chemical accident analysis will be required. The analysis for FDPF and FMEF assumes that the chemical inventory required for 1 year of operation is stored in the facility and that each chemical is stored in a single tank or container with no mitigating design features (e.g., dikes to limit the spill area).

I.2.2.1 Accident Scenario Selection

This section describes the process used to identify the chemicals for the accident analysis, the methodology used in analyzing potential accidents involving hazardous chemicals, the baseline accident scenarios, and the potential health risks associated with a release from identified scenarios. The anticipated chemical inventory for 1 year of plutonium-238 processing at FMEF and FDPF is given in **Table I–77**.

Table I–77 Anticipated Annual Inventory for Plutonium-238 Processing

Chemical	Inventory (pounds)	TPQ (pounds)	Chemical	Inventory (pounds)	TPQ (pounds)
Aluminum nitrate	5.2	Not in list	Acetone	69	–
Aluminum powder	303	–	Acetylene	16	–
Aluminum stearate	0.6	Not in list	Adogen 364	459	Not in list
Anion exchange resin	97	Not in list	Argon	1,333	Not in list
Ascorbic acid	157	Not in list	Compressed air	135	Not in list
Diethyl benzene	485	Not in list	Devcon 5-minute epoxy resin	7.4	Not in list
Dodecanol	93	Not in list	Helium	62	Not in list
Ferrous sulfamate	7	Not in list	Hydrochloric acid	321	–
Hydrazine nitrate	28	Not in list	Hydrogen (2-5 percent) in argon	24	Not in list
Hydroxylamine nitrate	60	Not in list	Hydrogen peroxide solution (< 52 percent)	8.8	1,000 lb for >52 percent
Methanol	17	Not in list	Nitric oxide	156	100
Nitric acid	2,170	1,000	Nitrogen	833	Not in list
Normal paraffin hydrocarbons	157	Not in list	Oxygen	29	Not in list
Oxalic acid	56	Not in list	P-10 nuclear counter mixture	1,184	Not in list
Polystyrene resin	783	Not in list	Potassium carbonate	5.5	Not in list
Sodium fluoride	0.6	–	Propane	450	–
Sodium hydroxide	1,078	–	Sodium carbonate	8.8	Not in list
Sodium nitrate	1,146	Not in list	Sodium hydroxide (40-50 percent)	7,422	–
Sodium nitrite	1.7	–	Sodium hydroxide (>10 percent solution)	1,068	–
Tributyl phosphate	849	Not in list	Sodium hypochlorite solution	27	–

Key: TPQ, Extremely Hazardous Substances List Threshold Planning Quantity Value; <, less than; >, greater than; –, no value in the list.
Source: EPA 1998.

Only the anticipated annual usage of nitric acid and nitric oxide for plutonium-238 processing exceeds the Threshold Planning Quantities for these substances as stipulated on the Extremely Hazardous Substances List provided in Section 3.02 of the Emergency Planning and Community Right-to-Know Act (EPA 1998). The respective Threshold Planning Quantities for nitric acid and nitric oxide are 454 kilograms (1,000 pounds) and 45.4 kilograms (100 pounds). Since inventories of these chemicals exceed the Threshold Planning Quantities, an evaluation of potential accident scenarios is presented in this NI PEIS.

I.2.2.2 Accident Scenario Descriptions

Two accidental chemical scenarios are postulated for this NI PEIS: the accidental uncontrolled release of nitric acid, and the accidental uncontrolled release of nitric oxide.

I.2.2.2.1 Nitric Acid Release

The nitric acid release scenario was developed on the basis of the following assumptions: the nitric acid released from the tank is red fuming nitric acid (100 percent). A catastrophic tank failure is the initiating event. There are no engineered safety features for the tank. The tank is in an unsheltered building in an open rural area. The release fraction is 100 percent. However, the actual amount of nitric acid that volatilizes to the atmosphere was determined by the method described in the *Technical Guidance for Hazards Analysis* (EPA 1987).

The consequences of the postulated nitric acid release scenario are overstated because of the conservatism of two assumptions: 100 percent red fuming nitric acid and a lack of engineered safety features to restrict releases from spills. Facilities under consideration for this program do not permit storage of red fuming nitric acid, which has a high vapor pressure, and they have engineered safety features (e.g., sloped floors and dikes) to restrict releases from spills.

I.2.2.2.2 Nitric Oxide Release

The analysis postulated a storage cylinder failure. A release fraction (percentage of material released) of 100 percent was used. An aggregated release of 71.8 kilograms (158 pounds) for nitric oxide gas was postulated. The rate of release of nitric oxide was calculated by the method described in the *Technical Guidance for Hazards Analysis* (EPA 1987).

I.2.2.3 Hazardous Chemical Accident Analysis Methodology

The potential health impacts of accidental releases of hazardous chemicals were assessed by comparing estimated airborne concentrations of the chemicals with the ERPG developed by the American Industrial Hygiene Association. The ERPG values are not regulatory exposure guidelines and do not incorporate the safety factors normally included in healthy worker exposure guidelines. ERPG-1 values are maximum airborne concentrations below which nearly all individuals could be exposed for up to 1 hour, resulting in only mild, transient, and reversible adverse health impacts. ERPG-2 values are indicative of irreversible or serious health effects or impairment of an individual's ability to take protective action. ERPG-3 values are indicative of potentially life-threatening health effects.

No approved ERPG levels are available for nitrous oxide (Kelly 1999). The ERPG values for nitric acid are presented in **Table I–78**. The ERPG values referenced by Kelly (1999) were used in this NI PEIS.

Table I–78 Emergency Response Planning Guideline Values for Nitric Acid

ERPG Level	DOE 1997	Kelly 1999
ERPG-1	2 parts per million	0.5 parts per million
ERPG-2	15 parts per million	10 parts per million
ERPG-3	30 parts per million	25 parts per million

Key: ERPG, Emergency Response Planning Guideline.

There are also no ERPG values for nitric oxide. For these cases, the "level of concern" has been estimated by using one-tenth of the "immediately dangerous to life and health" level for that substance—i.e., 100 parts per million—as published by the National Institute for Occupational Safety and Health (NIOSH 1997). For this NI PEIS, therefore, the level of concern for nitric oxide is 10 parts per million. Level of concern is defined as the concentration of an extremely hazardous substance in air above which there may be serious irreversible health effects or death as a result of a single exposure for a rather short period of time (EPA 1987).

I.2.2.3.1 Receptor Description

The potential health impacts of the accidental release of nitric acid and nitric oxide were assessed for three types of receptors:

1. A noninvolved worker—a worker assumed to be 640 meters (0.4 mile) from the point of release.

2. Site boundary maximally exposed individual—a hypothetical member of the public off site at the nearest point of access (7,210 meters [4.5 miles] for FMEF and 13,952 meters [8.7 miles] for FDPF) to the point of release.

3. Nearest highway maximally exposed individual—a hypothetical member of the public at the nearest point on an onsite highway (7,100 meters [4.4 miles] for FMEF and 5,800 meters [3.6 miles] for FDPF) to the point of release.

Facility workers (i.e., those individuals in the building at the time of the accident) are assumed to be killed by the release.

I.2.2.3.2 Analysis Computer Code

The computer code used for estimation of airborne concentrations was the Computer Aided Management of Emergency Operations Areal Locations of Hazardous Atmospheres (ALOHA), developed by the National Safety Council, EPA, and the National Oceanic and Atmospheric Administration.

DESCRIPTION OF THE MODEL

The atmospheric dispersion modeling for the above scenarios was conducted using the ALOHA 5.05 computer code (NSC 1990). The ALOHA code was designed for use by first responders. The model is most useful for estimating plume extent and concentration downwind from the release source for short-duration chemical accidents. It uses a Gaussian dispersion model to describe the movement and spreading of a gas that is neutrally buoyant. For heavier-than-air vapor releases, the model uses the same calculations as those used in EPA's Dense Gas Dispersion Model DEGADIS 2.1 (EPA-450/4-89-019). There are a number of limitations to the model:

1. ALOHA is not intended for modeling accidents involving radioactive chemicals.

2. It is not intended for use in modeling the permitting of stack gas or chronic, low-level (fugitive) emissions.

3. The ALOHA-DEGADIS heavy-gas module is more conservative than the DEGADIS model, which could result in a larger footprint than would actually be expected.

4. ALOHA does not consider the effects of thermal energy from fire scenarios or the byproducts of chemical reactions.

5. ALOHA does not include the process needed to model particulate dispersion.

6. ALOHA does not consider the shape of the ground under the spill or in the area affected by the plume.

7. ALOHA does not estimate concentrations under very low wind speeds (less than 1 meter [3.3 feet] per second), as the wind direction may become inconsistent under these conditions.

8. For very stable atmospheric conditions (usually late night or early morning), there will be uncertainties in the model estimates due to shifting wind directions and virtually no mixing of the plume into the surrounding air. The estimates may in fact, reflect to high airborne concentrations for long periods of time or at great distances from the release source.

9. ALOHA does not accurately represent variations associated with near-field (close to the release source) patchiness. In the case of a neutrally buoyant gas, the plume will move downwind; very near the source, however, it can be oriented in a different direction (e.g., going backward) due to the effect of drifting eddies in the wind.

WEATHER CONDITION ASSUMPTIONS

The model results are presented for atmospheric Stability Classes D and F, with wind speeds of 5.3 meters (17 feet) per second and 1.5 meters (4.9 feet) per second, respectively. Atmospheric Stability Class D is considered to be representative of average weather conditions; Stability Class F is considered to be representative of worst-case weather conditions. These weather conditions were selected because they are recommended in *Technical Guidance for Hazards Analysis* (EPA 1987). **Table I–79** presents the model parameter values for these weather conditions.

Table I–79 Analysis Weather Conditions

Parameter	Average Condition Stability Class D	Worst-Case Condition Stability Class F
Ambient air temperature	75 °F	60 °F
Relative humidity	50 percent	25 percent
Cloud cover	50 percent	20 percent
Average wind speed	5.3 meters per second	1.5 meters per second

Source: EPA 1987.

I.2.3 Human Health Impacts

The potential health impacts of the accidental releases were assessed by comparing the modeled ambient concentrations of nitric acid and nitrous oxide at each of the previously identified receptor locations with the ERPGs. The estimated airborne concentrations of nitric acid and nitric oxide are presented in **Table I–80** and **Table I–81**, respectively. **Table I–82** and **Table I–83** present of the impacts data for nitric acid and nitric oxide.

Table I–80 Airborne Concentration Estimates for Nitric Acid Release Scenarios

Downwind Distance from Source (meters)	Nitric Acid Concentration Under Stability Class D (parts per million)		Nitric Acid Concentration Under Stability Class F (parts per million)	
Facility	**FMEF**	**FDPF**	**FMEF**	**FDPF**
30	1,200	1,130	1,070	1,040
640	3.3	3.3	8.6	8.4
1,000	1.4	1.4	3.9	3.9
3,000	0.17	0.17	0.5	0.5
5,000	0.06	0.06	0.2	0.2
Nearest highway[a]	0.03	0.05	0.1	0.15
Site boundary [b]	0.03	(c)	0.1	(c)

a. FMEF = 7,100 meters (4.4 miles); FDPF = 5,800 meters (3.6 miles).
b. FMEF = 7,210 meters (4.5 miles); FDPF = 13,952 meters (8.7 miles).
c. Not calculated; the distance to the site boundary exceeds the analysis code 10-kilometer maximum distance limit for calculations.
Source: Calculated results.

Table I–81 Airborne Concentration Estimates for Nitric Oxide Scenarios

Downwind Distance from Source (meters)	Nitric Oxide Concentration Under Stability Class D (parts per million)		Nitric Oxide Concentration Under Stability Class F (parts per million)	
Facility	**FMEF**	**FDPF**	**FMEF**	**FDPF**
30	1,370	1,370	9,990	9,480
640	4.2	4.2	66	67.5
1,000	2	2	29.2	29.6
3,000	0.36	0.36	3.6	3.6
5,000	0.17	0.17	1.2	1.2
Nearest highway[a]	0.09	0.09	0.55	0.87
Site boundary [b]	0.09	(c)	0.53	(c)

a. FMEF = 7,100 meters (4.4 miles); FDPF = 5,800 meters (3.6 miles).
b. FMEF = 7,210 meters (4.5 miles); FDPF = 13,952 meters (8.7 miles).
c. Not calculated; the distance to the site boundary exceeds the analysis code 10-kilometer maximum distance limit for calculations.
Source: Calculated results.

Table I–82 Summary of Impacts Data for Release Scenarios (Nitric Acid)

Evaluation Parameter		FMEF (Stability Class D)	FDPF (Stability Class D)	FMEF (Stability Class F)	FDPF (Stability Class F)
Maximum distance to (meters)	ERPG-1 ERPG-2 ERPG-3	2,000 500 375	2,000 500 375	3,000 600 450	3,000 600 450
Noninvolved worker (640 meters)	Parts per million (ppm) Level of concern Potential health effects	3.3 < ERPG-2 Mild, transient	3.3 < ERPG-2 Mild, transient	8.6 < ERPG-2 Mild, transient	8.4 < ERPG-2 Mild, transient
Nearest highway maximally exposed individual	Parts per million (ppm) Level of concern	0.03 < ERPG-1	0.05 < ERPG-1	0.1 ERPG-1	0.15 ERPG-1
Site boundary maximally exposed individual	Parts per million (ppm) Level of concern Potential health effects	0.03 < ERPG-1 None	(a) < ERPG-1 None	0.1 ERPG-1 Mild, transient	(a) ERPG-1 Mild, transient

a. Not calculated; the distance to the site boundary exceeds the analysis code 10-kilometer maximum distance limit for calculations.
Key: <, less than; ERPG, Emergency Response Planning Guideline.

Table I–83 Summary of Impacts Data for Release Scenarios (Nitric Oxide)

Evaluation Parameter		FMEF (Stability Class D)	FDPF (Stability Class D)	FMEF (Stability Class F)	FDPF (Stability Class F)
Maximum distance (meters)	To concentrations of level of concern	500	500	1,900	2,000
Noninvolved worker (640 meters)	Parts per million (ppm) Level of concern Potential health effects	4.2 < LOC Mild, transient	4.2 < LOC Mild, transient	66 > LOC Serious	67.5 > LOC Serious
Nearest highway maximally exposed individual	Parts per million (ppm) Level of concern Potential health effects	0.09 < LOC None	0.09 < LOC None	0.55 < LOC None	0.87 < LOC None
Site boundary maximally exposed individual	Parts per million (ppm) Level of concern Potential health effects	0.09 < LOC None	(a) < LOC None	0.53 < LOC None	(a) < LOC None

a. Not calculated; the distance to the site boundary exceeds the analysis code 10-kilometer maximum distance limit for calculations.
Key: <, less than; >, greater than.

I.2.3.1 Impacts to Noninvolved Workers

Nitric Acid. A noninvolved worker is assumed to be located 640 meters (0.4 mile) from the point of release. The concentrations of nitric at that distance range from 3.3 to 8.6 parts per million for FMEF and 3.3 to 8.4 parts per million for FDPF, given assumed meteorological conditions. The maximum estimated airborne concentration at 640 meters (0.4 mile) Stability Class F exceeds the ERPG-1 value of 0.5 part per million for nitric acid, which suggests the potential for only mild, transient, and reversible health impacts on a noninvolved workers at that distance from the release.

Nitric Oxide. For the nitric oxide release scenarios, the concentrations at 640 meters (0.4 miles) range from 4.2 to 66 parts per million for FMEF and 4.2 to 67.5 parts per million for FDPF, given assumed meteorological conditions. As a result, the maximum estimated airborne concentration at 640 meters (0.4 miles) exceeds the level-of-concern value of 10 parts per million for nitric oxide, which suggests that a noninvolved worker may experience irreversible or serious, but not life-threatening, health impacts if the exposures are not mitigated.

I.2.3.2 Impacts on Access Roads

Nitric Acid. The receptor at the nearest highway is assumed to be located 7,100 meters (4.4 miles) and 5,800 meters (3.6 miles) for FDPF from the points of release at FMEF and FDPF respectively. For the nitric acid release scenarios, the receptor on the nearest highway could be exposed to a nitric acid concentration of 0.03 to 0.05 part per million under Stability Class D conditions, which is below the ERPG-1 value for nitric acid of 0.5 part per million. Exposures to concentrations below the ERPG-1 value are not expected to have any adverse health impacts on the receptor. Under Stability Class F conditions, the offsite receptor may be exposed to a nitric acid concentration of about 0.1 to 0.15 part per million, which is below the ERPG-1 value for nitric acid of 10 parts per million. Exposure of the receptor to concentrations greater than the ERPG-1 value may have only mild, transient and reversible health impacts.

Nitric Oxide. For the nitric oxide release scenarios, the receptor on the nearest highway could be exposed to concentrations of 0.09 part per million under Stability Class D conditions, which is below the level-of-concern value for nitric oxide of 10 parts per million. Exposures to concentrations below the level-of-concern value are not expected to produce any adverse health effects for the receptor. Under Stability Class F conditions, the offsite receptor may be exposed to a nitric oxide concentration of about 0.55 to 0.87 parts per million, which is below the level-of-concern value for nitric oxide of 10 parts per million.

I.2.3.3 Offsite Impacts

The site boundary receptor is assumed to be located at a distance of 7,210 meters (4.5 miles) and 13,952 meters (8.7 miles) from the points of release at FMEF and FDPF, respectively. ALOHA does not draw any plume larger than 10 kilometers (6.2 miles) (NSC 1990). The FDPF site boundary is 13.3 kilometers (8.3 miles) from the point of release. Therefore, no impacts to site boundary receptor at FDPF were performed. Health impacts on the nearest highway at a distance of 5.8 kilometers (3.6 miles) from the point of release were only mild, transient, and reversible. Exposure of the receptor at concentrations below the level-of-concern value may have only mild, transient, and reversible health impacts. At a distance of 14.0 kilometers (8.7 miles) from the point of release, adverse health effects are not expected.

Nitric Acid. For the nitric acid release scenarios, the site boundary receptor at Hanford could be exposed to a nitric acid concentration of 0.03 part per million under Stability Class D conditions, which is below the ERPG-1 value for nitric acid of 0.5 part per million. Exposures to concentrations below the ERPG-1 value are not expected to have any adverse health impacts on the receptor. Under Stability Class F conditions, the site boundary receptor may be exposed to a nitric acid concentration of about 0.1 part per million, which is below the ERPG-1 value for nitric acid of 10 parts per million. Exposure of the receptor at concentrations greater than the ERPG-1 value may have only mild, transient, and reversible health impacts.

Nitric Oxide. For the nitric oxide release scenarios, the site boundary receptor exposures were 0.09 part per million under Stability Class D conditions, which is below the level-of-concern value for nitric oxide of 10 parts per million. Exposures to concentrations below the level of concern value are not expected to have any adverse health impacts on the receptor. Under Stability Class F conditions, the site boundary receptor may be exposed to a nitric oxide concentration of about 0.53 part per million, which is below the level-of-concern value for nitric oxide of 10 parts per million. Exposure of the receptor at concentrations below the level-of-concern value may produce only mild, transient, and reversible health impacts.

I.2.3.4 Uncertainties

This screening-level analysis is subject to a number of uncertainties relative to the atmospheric dispersion modeling. Among those uncertainties are the following:

1. On the day of an accident, it will undoubtedly be very difficult to establish exactly the rate or magnitude of the release.

2. The weather conditions and wind speed may well be different from those used in the analysis.

3. The dispersion modeling does not take into account the deposition of highly reactive vapors onto surfaces, including equipment, groundwater, and vegetation. This means that the model overestimates airborne concentrations at longer distances.

4. Overall, the uncertainties in predicted airborne concentrations may be as large as a factor of plus or minus two times the estimated concentration.

5. In view of these uncertainties, the results of this analysis should be considered only as screening-level estimations.

I.3 INDUSTRIAL SAFETY

Estimates of potential industrial impacts to workers during construction, irradiation, fabrication, and processing were evaluated based on DOE and Bureau of Labor Statistics data. Impacts are classified into two groups: total recordable cases and fatalities. A recordable case includes work-related death, illness, or injury that resulted in loss of consciousness, restriction of work or motion, transfer to another job, or required medical treatment beyond first aid.

DOE and contractor total recordable cases and fatality incidence rates were obtained from the CAIRS database (DOE 2000a, 2000b). The CAIRS database is used to collect and analyze DOE and DOE contractor reports of injuries, illnesses, and other accidents that occur during DOE operations. The 5-year average (1995–1999) rates were determined for average construction total recordable cases, average operation total recordable cases, and average operation fatalities. The average construction fatality rate was obtained from the Bureau of Labor Statistics (Toscano and Windau 1998).

Table I–84 presents the average occupational total recordable cases and fatality rates for construction and operation activities.

Table I–84 Average Occupational Total Recordable Cases and Fatality Rates (per worker-year)

Labor Category	Total Recordable Cases	Fatalities
Construction	0.053	0.000139
Operation	0.033	0.000013

Expected impacts (both annual and for the duration of the activity) to workers at each facility for construction and operation are presented in **Table I–85**.

Table I–85 Industrial Safety Impacts from Construction and Operation

Facility	Estimated Number of Workers	Construction or Operation Duration (years)	Annual Total Recordable Cases	Activity Duration Total Recordable Cases	Annual Fatalities	Activity Duration Fatalities
Construction						
Low-energy accelerator	75	3	4.0	12	0.010	0.03
High-energy accelerator	410	5	22	110	0.057	0.285
New research reactor	160	7	8.5	59.5	0.022	0.154
Operation						
ATR[a]	0	35	NA	NA	NA	NA
HFIR[a]	0	35	NA	NA	NA	NA
CLWR[a]	0	35	NA	NA	NA	NA
FFTF	242	35	8.0	280	0.0031	0.109
Low-energy accelerator	13	35	0.4	14	0.00017	0.00595
High-energy accelerator	225	35	7.4	259	0.0029	0.102
New research reactor	120	35	4.0	140	0.0016	0.056
REDC	116	35	3.8	133	0.0015	0.0525
FDPF	75	35	2.5	87.5	0.00098	0.0343
FMEF	105	35	3.5	123	0.0014	0.049
RPL/306–E	30	35	1.0	35	0.00039	0.0137
New support facility	100	35	3.3	116	0.0013	0.0455

a. Not applicable. No additional workers would be required for the proposed activities evaluated in this NI PEIS.

Computation of total recordable cases and fatalities expected during construction or modification of target fabrication and processing facilities, prior to operation, have been neglected because of the relatively short duration of these activities.

As expected, the incidence of impacts, above and beyond those requiring first aid, do indeed exceed impacts from radiation and hazardous chemical accidents evaluated in this NI PEIS. No fatalities would be expected from either construction or operation of any facility.

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Appendix J
Evaluation of Human Health Effects of Transportation

J.1 INTRODUCTION

The overland transportation of any commodity involve risk to both transportation crew members and members of the public. This risk can result directly from transportation-related accidents and indirectly from the increased levels of pollution from vehicle emissions, regardless of the cargo. The transportation of certain materials, such as hazardous or radioactive substances, can pose an additional risk because of the unique nature of the material itself. To permit a complete appraisal of the environmental impacts of the proposed action and alternatives, the human health risks associated with the transportation of radiological material are analyzed in this appendix.

This appendix provides an overview of the approach used to assess the human health risks that may result from overland transportation. The appendix includes discussion of the scope of the assessment, analytical methods used for the risk assessment (i.e., computer models), assessment assumptions, potential transportation routes, and presents the results of the assessment. In addition, to assist in understanding and interpreting the results, specific areas of uncertainty are described with an emphasis on how the uncertainties may affect comparisons of the alternatives.

The risk assessment results are presented in this appendix in terms of "per-shipment" risk factors, as well as for the total risks for a given alternative. Per-shipment risk factors provide an estimate of the risk from a single shipment. The total risks for a given alternative are found by multiplying the expected number of shipments by the appropriate per-shipment risk factors.

J.2 SCOPE OF ASSESSMENT

The scope of the transportation human health risk assessment, including the alternatives and options, transportation activities, potential radiological and nonradiological impacts, and transportation modes considered, is described below. Additional details of the assessment are provided in the remaining sections of the appendix.

PROPOSED ACTION AND ALTERNATIVES

The transportation risk assessment conducted for this *Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]) estimates the human health risks associated with the transportation of radioactive materials for a number of alternatives.

TRANSPORTATION-RELATED ACTIVITIES

The transportation risk assessment is limited to estimating the human health risks incurred during transportation and handling when away from U.S. Department of Energy (DOE) facilities (i.e., at a port) for each alternative. The transportation risk assessment does not address possible impacts from increased transportation levels on local traffic flow, noise levels, or infrastructure.

RADIOLOGICAL IMPACTS

For each alternative, radiological risks (those risks that result from the radioactive nature of the materials) are assessed for both incident-free (normal) and accident transportation conditions. The radiological risk associated with incident-free transportation conditions would result from the potential exposure of people to external radiation in the vicinity of a loaded shipment. The radiological risk from transportation accidents would come from the potential release and dispersal of radioactive material into the environment during an accident and the subsequent exposure to people.

All radiological impacts are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent (10 CFR Part 20), which is the sum of the effective dose equivalent from external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of roentgen equivalent man (rem) for individuals and person-rem for collective populations. The impacts are further expressed as health risks in terms of latent cancer fatalities and cancer incidence in exposed populations, using the dose-to-risk conversion factors published by the National Council on Radiation Protection and Measurement (NCRP 1993), and by the International Council on Radiation Protection (ICRP 1991).

NONRADIOLOGICAL IMPACTS

In addition to the radiological risks posed by overland transportation activities, vehicle-related risks are also assessed for nonradiological causes (i.e., causes related to the transport vehicles and not the radioactive cargo) for the same transportation routes. The nonradiological transportation risks, which would be incurred for similar shipments of any commodity, are assessed for both incident-free and accident conditions. The nonradiological risks during incident-free transportation conditions would be caused by potential exposure to increased vehicle exhaust emissions. The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities unrelated to the shipment of cargo. National transportation fatality rates are used in the assessment. Nonradiological risks are presented in terms of estimated fatalities.

TRANSPORTATION MODES

All overland shipments are assumed to use trucks, those requiring secure shipment will use safe, secure trailer/SafeGuards Transport (SST/SGT). Transatlantic shipments of mixed oxide fuel would use purpose-built vessels[1]. Medical isotopes would be shipped via aircraft as well as trucks.

RECEPTORS

Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck, ship, and aircraft crew members involved in the actual transportation. The general public includes all persons who could be exposed to a shipment while it is moving or stopped during transit. Potential risks are estimated for the collective populations of exposed people and for the hypothetical maximally exposed individual. The collective population risk is a measure of the radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing various alternatives. Persons handling casks at DOE facilities are included in site dose assessments. Workers handling packages at military ports are included in this appendix.

[1] Purpose-built vessels are specifically designed to transport casks containing radioactive material.

J.3 PACKAGING AND REPRESENTATIVE SHIPMENT CONFIGURATIONS

Shipment Configurations Regulations that govern the transportation of radioactive materials are designed to protect the public from the potential loss or dispersal of radioactive materials, as well as from routine radiation doses during transit. The primary regulatory approach to promote safety is through the specification of standards for the packaging of radioactive materials. Because packaging represents the primary barrier between the radioactive material being transported and radiation exposure to the public and the environment, packaging requirements are an important consideration for transportation risk assessment. Regulatory packaging requirements are discussed briefly below. The representative packaging and shipment configurations assumed for this NI PEIS also are described below.

J.3.1 Packaging Overview

Although several Federal and state organizations are involved in the regulation of radioactive material transportation, primary regulatory responsibility resides with the U.S. Department of Transportation (DOT) and the U.S. Nuclear Regulatory Commission (NRC). All transportation activities must take place in accordance with the applicable regulations of these agencies as specified in 49 CFR and 10 CFR Part 71. Packaging for radioactive materials may be either DOE-specification packagings, Type A packagings designed and tested commercially, Type B certified (DOE or NRC) packagings, or DOE-designed and -tested Type A packagings. Transatlantic shipments would also be in accordance with the International Atomic Energy Agency (IAEA) regulations. DOT and NRC work to ensure that U.S. regulations are consistent with IAEA regulations.

Transportation packaging for small quantities of radioactive materials must be designed, constructed, and maintained to contain and shield their contents during normal transport conditions. For large quantities and for more highly radioactive material, such as spent nuclear fuel, packaging must contain and shield their contents in the event of severe accident conditions. The type of packaging used is determined by the total radioactive hazard presented by the material within the packaging. Four basic types of packaging are used. Excepted, Industrial, Type A, and Type B. Another packaging option, “Strong, Tight,” is available for some domestic shipments.

Excepted packages are limited to transporting materials with extremely low-levels of radioactivity. Industrial packages are used to transport materials that, because of their low concentration of radioactive materials, present a limited hazard to the public and the environment. Type A packages are designed to protect and retain their contents under normal transport conditions and must maintain sufficient shielding to limit radiation exposure to handling personnel. These packages are used to transport radioactive materials with higher concentrations or amounts of radioactivity than Excepted, or Industrial packages. Strong, Tight packages are used in the United States for shipment of certain materials with low-levels of radioactivity, such as natural uranium and rubble from the decommissioning of nuclear reactors. Type B packages used to transport material with the highest radioactivity levels, are designed to protect and retain their contents under transportation accident conditions and are described in more detail in the following sections.

J.3.2 Regulations Applicable to Type B Casks

Regulations for the transport of radioactive materials in the United States are issued by DOT and are codified in 49 CFR Part 173. The regulation authority for radioactive materials transport is jointly shared by DOT and NRC. As outlined in a 1979 Memorandum of Understanding with the NRC, DOT specifically regulates the carriers of spent nuclear fuel and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements (44 FR 38690). DOT also regulates the labeling, classification, and marking of all packages. NRC regulates the packaging and transport of spent nuclear fuel for its licensees, which include

commercial shippers of spent nuclear fuel. In addition, NRC sets the standards for packages containing fissile materials and spent nuclear fuel.

DOE policy requires compliance with applicable Federal regulations regarding domestic shipments of spent nuclear fuel. Accordingly, DOE has adopted the requirements of 10 CFR Part 71, "Packaging and Transportation of Radioactive Materials," and 49 CFR Part 173, "Shippers–General Requirements for Shipments and Packagings." DOE Headquarters can issue a certificate of compliance for a package to be used only by DOE and its contractors.

J.3.2.1 Cask Design Regulations

Neptunium-237, neptunium-237 targets, mixed oxide fuel, plutonium-238 and many isotopes are transported in robust "Type B" transportation casks that are certified for transporting radioactive materials. Casks designed and certified for spent nuclear fuel transportation within the United States must meet the applicable requirements promulgated by the NRC for design, fabrication, operation, and maintenance, as contained in 10 CFR Part 71.

Cask design and fabrication can only be done by approved vendors with established quality assurance programs (10 CFR Section 71.101). Cask and component suppliers or vendors are required to obtain and maintain documents that prove the materials, processes, tests, instrumentation, measurements, final dimensions, and cask operating characteristics meet the design-basis established in the Safety Analysis Report for Packaging for the cask and that the cask will function as designed.

Regardless of where a transportation cask is designed, fabricated, or certified for use, it must meet certain minimum performance requirements (10 CFR Sections 71.71, 71.73–71.75, 71.77). The primary function of a transportation cask is to provide containment and shielding. Casks similar to the designs being considered for targets have been used to transport spent nuclear fuel for many years. Regulations require that casks must be operated, inspected, and maintained to high standards to ensure their ability to contain their contents in the event of a transportation accident (10 CFR Section 71.87). There are no cases of a major release of radioactive materials from a Type B package, even though thousands of shipments have been made by road, rail, and water transport. Further, a number of obsolete casks have been tested under severe accident conditions to demonstrate their adherence to design criteria without failure. Such tests have demonstrated that transportation casks are not only fabricated to a very high factor of safety, they are even sturdier than required.

Transportation casks are built out of heavy, durable structural materials such as stainless steel. These materials must ensure cask performance under a wide range of temperatures (10 CFR Section 71.43). In addition to the structural materials, shielding is provided to limit radiation levels at the surface and at prescribed distances from the surface of transportation casks (10 CFR Section 71.47). Shielding typically consists of dense material such as lead or depleted uranium. However, heavily shielded casks are needed for targets because irradiated targets have gamma radiation levels similar to those of spent fuel. The cask cavity can be configured to hold various contents, including irradiated or unirradiated targets. The assemblies are supported by internal structures, called baskets, that provide shock and vibration resistance and establish minimum spacing and heat transfer to maintain the temperature of the contents within the limits specified in the Safety Analysis Report for Packaging.

Finally, to limit impact forces and minimize damage to the structural components of a cask in the event of a transportation accident, impact-absorbing structures may be attached to the exterior of the cask. These are usually composed of foam, or aluminum honeycomb that is designed to readily deform upon impact to absorb impact energy. All of these components are designed to work together in order to satisfy the regulatory

requirements for a cask to operate under normal conditions of transportation and maintain its integrity in an accident.

J.3.2.2 Design Certification

For certification, Type B transportation casks must be shown by analysis and/or testing to withstand a series of hypothetical accident conditions. These conditions have been internationally accepted as simulating damage to transportation casks that could occur in most reasonably foreseeable accidents. The impact, puncture, fire, and water-immersion tests are considered in sequence to determine their cumulative effects on one package. These accident conditions are described in **Figure J–1**. NRC issues regulations, 10 CFR Part 71, governing the transportation of radioactive materials. In addition to the tests shown in Figure J–1, the regulations affecting Type B casks require that a transportation cask with activity greater than 10^6 curies (which is applicable to irradiated targets) be designed and constructed so that its undamaged containment system would withstand an external water pressure of 20 kilograms per square centimeter (290 pounds per square inch) or immersion in 200 meters (656 feet) of water, for a period of not less than 1 hour without collapsing, buckling, or allowing water to leak into the cask.

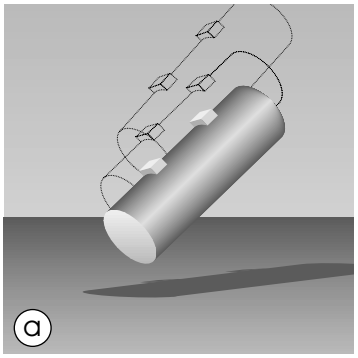
Under the Federal certification program, a Type B packaging design must be supported by a Safety Analysis Report for Packaging, which demonstrates that the design meets Federal packaging standards. The Safety Analysis Report for Packaging must include a description of the proposed packaging in sufficient detail to identify the packaging accurately and provide the basis for evaluating its design. The Safety Analysis Report for Packaging must provide the evaluation of the structural design, materials properties, containment boundary, shielding capabilities, and criticality control, and present the operating procedures, acceptance testing, maintenance program, and the quality assurance program to be used for design and fabrication. Upon completion of a satisfactory review of the Safety Analysis Report for Packaging to verify compliance to the regulations, a Certificate of Compliance is issued. DOE is permitted to certify Type B and fissile material packages for its own use by 49 CFR Section 173.7, and guidance for that certification is provided by DOE Order 461.1.

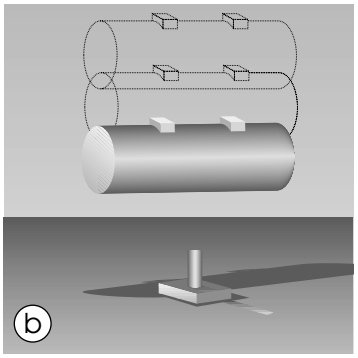
J.3.2.3 Transportation Regulations

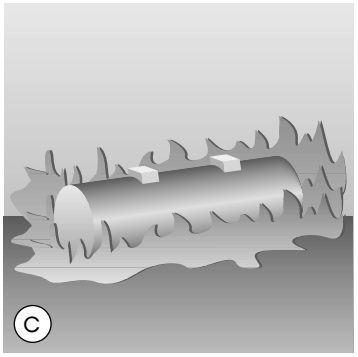
To ensure that the transportation cask is properly prepared for transportation, trained technicians perform numerous inspections and tests (10 CFR Section 71.87). These tests are designed to ensure that the cask components are properly assembled and meet leak-tightness, thermal, radiation, and contamination limits before shipping radioactive material. The tests and inspections are clearly identified in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks can only be operated by registered users who conduct operations in accordance with documented and approved quality assurance programs meeting the requirements of the regulatory authorities. Records must be maintained that document proper cask operations in accordance with the quality requirements of 10 CFR Section 71.91. Reports of defects or accidental mishandling must be submitted to the NRC. DOE would be the Shipper-of-Record for the shipments.

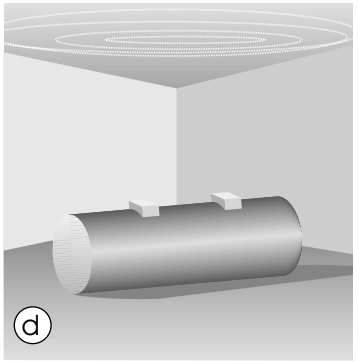
External radiation from a package must be below specified limits that minimize the exposure of handling personnel and the general public. For an exclusive-use shipment (i.e., carrying no other cargo) in a closed transport vehicle, the external radiation dose rate during normal transportation conditions must be maintained below the following limits of 49 CFR Part 173:

- 10 millirems per hour at any point 2 meters (6.6 feet) from the vertical planes projected by the outer lateral surfaces of the transport vehicle (referred to as the regulatory limit throughout this document)









STANDARDS FOR TYPE B CASKS

For certification by NRC, a cask must maintain its integrity under routine transportation conditions and be shown by test or analysis to withstand a series of accident conditions without releasing its contents. These conditions have been internationally accepted as simulating damage to casks that could occur in most severe credible accidents. The impact, fire, and water-immersion tests are considered in sequence to determine their cumulative effects on one package. A separate cask is subjected to a deep water-immersion test.

ⓐ **Free Drop (Impact):** A 30-foot drop onto a flat, unyielding surface so that the package's weakest point is struck.

ⓑ **Puncture:** A 40-inch free drop onto a 6-inch diameter steel rod at least 8 inches long, striking the package at its most vulnerable spot.

Crush: For some low-density, light-weight packages, a drop of a 1,100-pound mass from 30 feet onto the package resting on an unyielding surface is required.

ⓒ **Heat:** Exposure of the entire package to 1,475 °F for 30 minutes.

ⓓ **Immersion (fissile materials):** Package immersed under 3 feet of water in a position where maximum leakage is expected.

Immersion (all packages): A separate, undamaged package is submerged under 50 feet of water.

Additionally, 10 CFR 71.61 requires irradiated nuclear fuel packages (separate and undamaged) to be subjected to pressure that relates to a water depth of greater than 600 feet.

Source: Adapted from DOE 1999.

Note: 1 foot = 0.3048 meters; 1 inch = 2.54 centimeters; °F = (1.8 x °C) + 32.

Figure J–1 Standards for Transportation Casks

- 2 millirems per hour in any normally occupied position in the transport vehicle

Additional restrictions apply to package surface contamination levels, but these restrictions are not important for the transportation radiological risk assessment. For risk assessment purposes, it is important to note that all packaging of a given type is designed to meet the same performance criteria. Therefore, two different Type B designs would be expected to perform similarly during incident-free and accident transportation conditions. The specific containers selected or designed, however, will determine the total number of shipments necessary to transport a given quantity of material.

J.3.2.4 Communications

Proper communication assists in ensuring safe preparation and handling of transportation casks. Communication is provided by labels, markings, placarding, shipping papers, or other documents. Labels applied to the cask, document the contents and the amount of radiation emanating from the cask exterior, known as the transport index (49 CFR Section 172.403). The transport index lists the ionizing radiation level in millirems per year at a distance of 1 meter (3.3 feet) from the cask surface.

In addition to the label requirements, markings (49 CFR Section 173.471) should be placed on the exterior of the cask to show the proper shipping name and the consignor and consignee in case the cask is separated from its original shipping documents (49 CFR Section 172.203). Transportation casks are required to be permanently marked with the designation "Type B," name and address of the owner or fabricator, Certificate of Compliance number, and the gross weight (10 CFR Section 71.83).

Placards are applied to the transport vehicle or freight container holding the transportation cask (49 CFR Section 172.500). The placards indicate the radioactive nature of the contents. Neptunium-237, neptunium-237 targets, and plutonium-238 shipments which constitute a highway route-controlled quantity or "HRCQ," must be placarded according to 49 CFR Section 172.507. Placards provide first responders to a traffic or transportation accident with initial information about the nature of the contents.

Shipping papers should contain the notation "HRCQ" and have entries identifying the following: name of the shipper, emergency response telephone number, description of contents, and the shipper's certificate, as described in 49 CFR Part 172, Subpart C.

In addition, drivers of motor vehicles transporting radioactive material must have training in accordance with the requirements of 49 CFR Section 172.700. The training requirements include familiarization with the regulations, emergency response information, and the hazard communication programs required by the Occupational Health and Safety Administration in 29 CFR Section 1910.1200. Drivers are also required to have training on the procedures necessary for safe operation of the vehicle.

J.3.3 Packages Used in the Nuclear Infrastructure Program

This section describes currently available packaging systems that have been used for similar materials and could be used to implement the activities described in this NI PEIS. DOE could choose to design new or procure similar packaging. This similar packaging would be designed to the same level of safety and would be expected to have similar features. These packages have been used for the purpose of estimating input parameters, such as number of shipments and mass of contents, for the purpose of impact analysis.

J.3.3.1 Neptunium-237 Packaging

The Type B 9975 container could be used to transport neptunium-237 from the Savannah River Site (SRS) to the storage or processing facility the Radiochemical Engineering Development Center (REDC), Fluorinel Dissolution Process Facility (FDPF), or Fuels and Materials Examination Facility (FMEF). The 9975 package includes a 132-liter (35-gallon) drum, insulation, bearing plates, primary containment vessel, secondary containment vessel, lead shielding and aluminum honeycomb spacers (**Figure J–2**). The weight of the package is 163 kilograms (360 pounds), the overall height is 0.9 meter (35 inches) and the diameter is 0.5 meter (20 inches) (WSRC 1996).

In the spring of 2000, the 9975 packaging failed the recertification test and the Certificate of Compliance has been canceled. During the sequential 30-foot drop and puncture bar test, which are part of the hypothetical accident condition testing, the package lid buckled and partially opened. DOE could either redesign the 9975 package, design a new package or modify an existing package. In any case, the new design would be evaluated for compliance with current regulatory requirements by the Package Approval and Safety Program. DOE needs a package of similar size and capability to the 9975 for several programs, including the Rocky Flats cleanup (Scott 2000). The size and general characteristics of the replacement package (i.e., 132-liter [35-gallon] or 206-liter [55-gallon] drum, can-in-can construction, insulation, approximately 5-kilogram [11-pound] capacity) would be similar to the 9975. Therefore, for the purpose of risk analysis, the capacity and general characteristics of the 9975 package will be used.

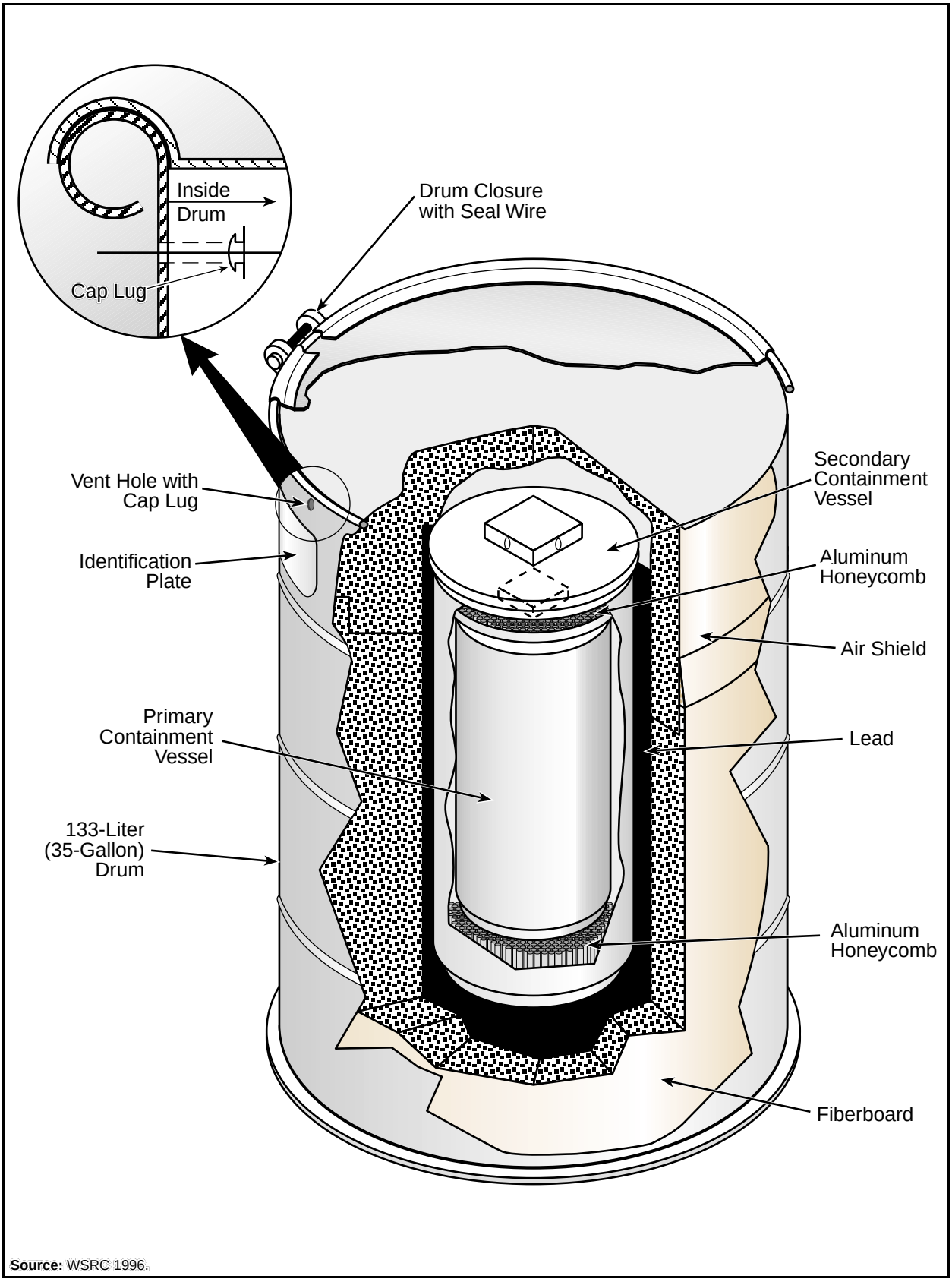
The neptunium-237 would be sealed into a convenience can and placed on a honeycomb spacer inside the stainless steel primary containment vessel. The primary containment vessel would be bolted closed, and placed into the similarly constructed, but larger, secondary containment vessel. The secondary containment vessel would be bolted closed and loaded into the drum. The drum is equipped with lead shielding that reduces radiation levels, and fiberboard insulation that protects the containment vessels in the unlikely event of a severe impact. The drum and cover are made of 18-gage carbon steel and are galvanized and coated with zinc chromate. A locking ring with drop-forged lugs secures the cover to the drum.

J.3.3.2 Neptunium Targets

After targets are fabricated at the processing facility, located at the REDC at ORR, FDPF at the Idaho National Engineering and Environmental Laboratory (INEEL), or FMEF at the Hanford Site (Hanford), they will be transported to either the High Flux Isotope Reactor (HFIR) at ORR, Advanced Test Reactor (ATR) at INEEL, Fast Flux Test Facility (FFTF) at Hanford, a new reactor or a new accelerator, or a commercial light water reactor (CLWR) for irradiation. After they are irradiated, they will be returned to the processing facility for extraction of the plutonium-238. The same casks and number of shipments will apply to both the unirradiated and irradiated targets.

INTRASITE SHIPMENT—REDC TO HFIR OR FDPF TO ATR

If HFIR is selected to irradiate and REDC to process the targets, targets would be transported the short distance between REDC and HFIR in a cask that was formerly certified to Type B standards. These formerly certified packages are verified to be equivalent to Type B packages by site procedures. Since the move is only about 90 meters (100 yards), on closed roads, and entirely at ORR, DOE procedures and NRC regulations do not require the use of a certified Type B cask. Similar procedures and equipment would be used at INEEL for transfers between FDPF and ATR.



Source: WSRC 1996.

Figure J–2 Typical Assembly of Type 9975 Package

INTERSITE SHIPMENT

The transportation of irradiated targets would involve shipments of Type B quantities (based on activation levels) using NRC- or DOE-certified shipping casks. The amount of material that can be loaded in a shipping cask is controlled by the thermal load, the fission product inventory, the neutron dose rate, and the physical size. Although a new cask could be designed for this application, the most likely approach would be to design a new basket to fit inside an existing cask. Since this design effort is not yet underway, the exact number of shipments cannot be determined. As an example case, if the GE–2000 container (**Figure J–3**) is used, the thermal load limit would determine the total number of shipments. Transportation of the irradiated targets is likely to require updating the Certificate of Compliance for the casks. Preliminary calculations to determine the decay heat were done for several irradiation positions and cycles to estimate the thermal load of the irradiated targets. The preliminary analysis indicates that irradiated targets that have cooled for at least 100 days will generate about 0.58 watt (0.033 British thermal unit per minute) of heat for each gram (0.035 ounce) of plutonium-238. If the targets are classified as spent nuclear fuel, then the thermal load limit for the GE–2000 cask is 600 watts (34 British thermal units per minute) per shipment. In this case, a total of nine shipments would be required to move the targets. If the targets are classified as a by-product or special nuclear material, then the thermal load limit is 2,000 watts (114 British thermal units per minute) per shipment for a total of three shipments. Thus, the range is roughly three to nine shipments per year, and the risk analysis is based on nine shipments per year. This range is representative of other commercially available casks. Note that the GE–2000 is too large and heavy to transport on an SST/SGT.

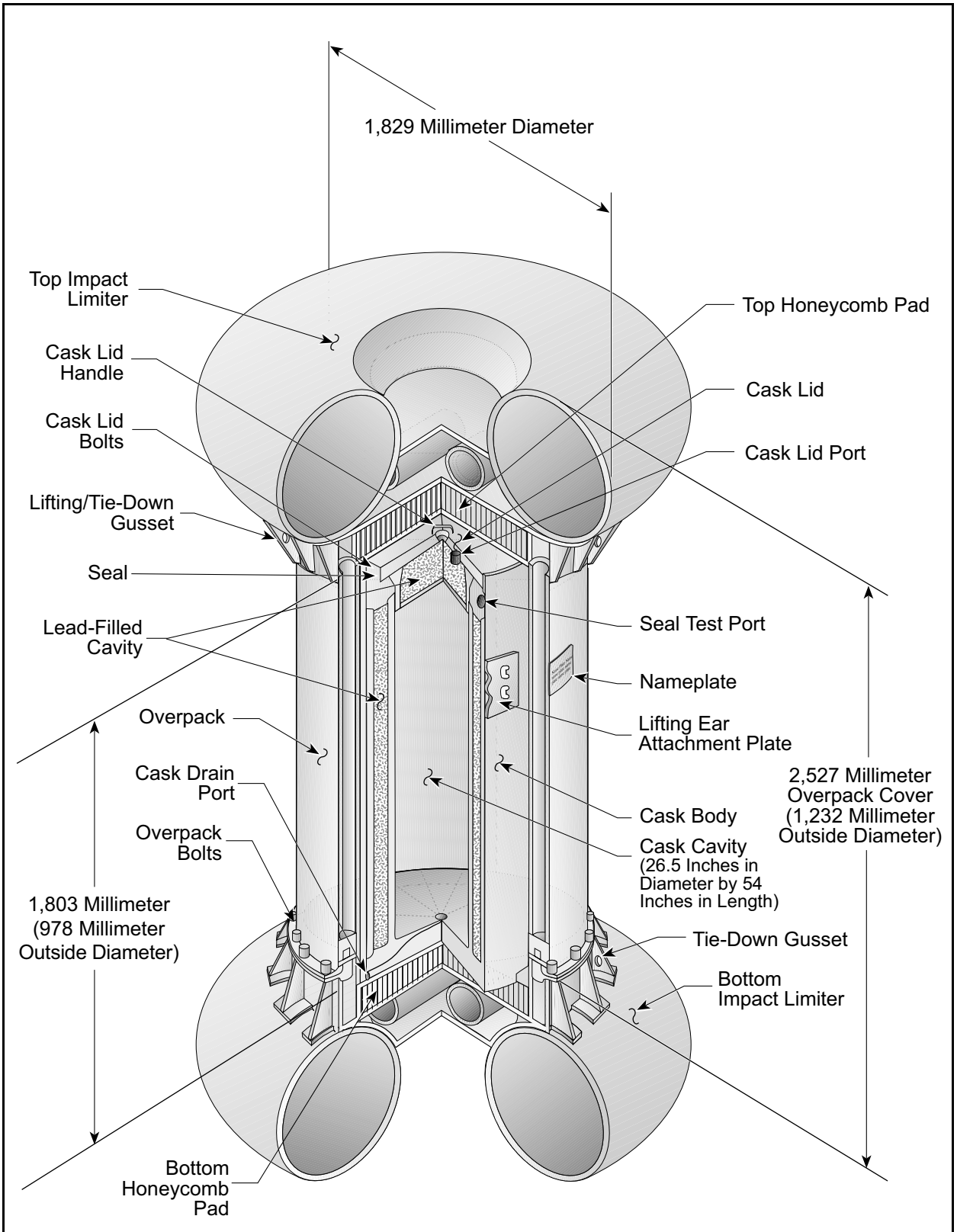
DOE realizes that a CLWR, new reactor, accelerator, or FFTF would use larger targets than HFIR or ATR. A GE–2000 container would not be long enough for these targets. However, about the same thermal parameters would apply for all but the accelerator targets, so the same shipping estimates are used. Based on the preconceptual design, accelerator targets would be much larger, but would require fewer shipments.

J.3.3.3 Plutonium-238

The 5320 cask, designed for shipment of americium or plutonium by surface transportation modes could be used to carry plutonium-238 oxide that would be produced at the processing facility. Several versions of the 5320 (the 5320 B(U) and the 5320 B(M)) comply with the regulatory safety requirements of 10 CFR Part 71, as well as DOE and IAEA requirements. The 5320 package was evaluated for transport of plutonium-238 oxide in any solid form, in excess of Type A quantities as Fissile Class I. The radioactive content is limited to 357 grams (12.6 ounces) of plutonium-238. When the only plutonium isotope is plutonium-238, the 5320 packages may be shipped Fissile Exempt, subject to the provisions of 49 CFR Section 173.453(f). The plutonium-238 oxide may be any density up to 11 grams per cubic centimeter (6.4 ounces per cubic inch), contain a maximum of 1 gram of volatile constituents, and not exceed a decay heat load of 203 watts (11.6 British thermal unit per minute). The time that the plutonium-238 can be sealed within the primary containment vessel prior to and including shipment is limited to 2 years.

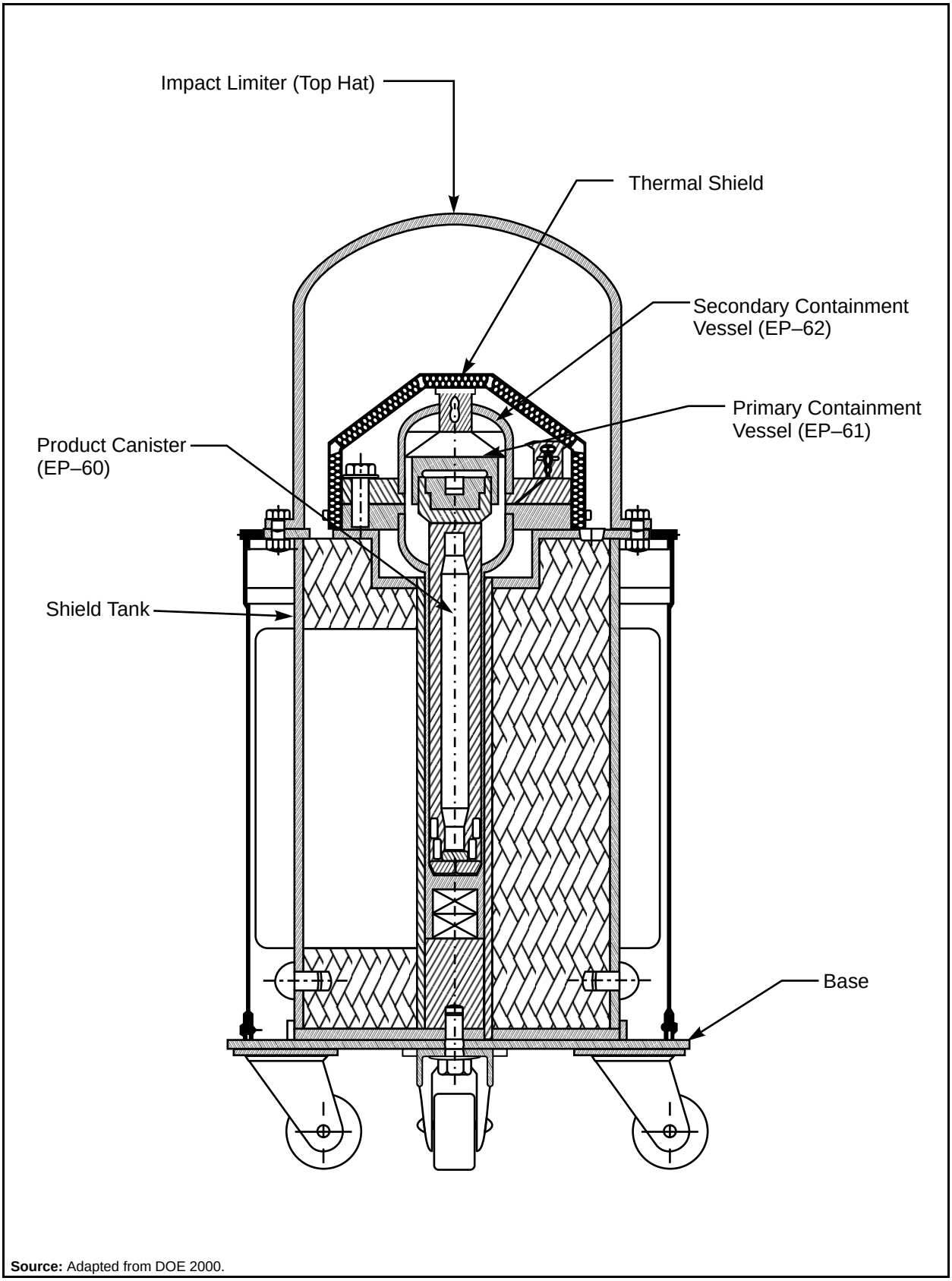
The 5320 packaging is a dome topped, upright circular cylinder mounted on a baseplate supported by casters, as shown in **Figure J–4**. The weight of the packaging is about 148 kilograms (327 pounds), the overall height is 81 centimeters (32 inches) and the diameter is 42.5 centimeters (16.75 inches).

The plutonium-238 would be loaded into an EP–60 product canister. The EP–60 is not credited in the safety analysis as part of the packaging. It is a stainless steel shell confinement vessel which is used to load the product into the package safely and conveniently. The EP–60 would be seal welded into the removable stainless shell primary containment vessel, the EP–61. The EP–61 is placed into the secondary containment vessel, the EP–62. The stainless steel EP–62 has a removable bolted closure lid. The gasketed flange of the EP–62 satisfies the containment requirements of normal conditions of transport and hypothetical accident



Source: DOE 1996.

Figure J–3 GE–2000 Container



Source: Adapted from DOE 2000.

Figure J–4 Cross Section of 5320 Packaging Assembly

conditions. Shipments must be completed within 2 years of sealing the EP–61 because of the possibility of undetected gas buildup.

The nested EP–61 and EP–62 are surrounded by a tinned aluminum-shield tank filled with water-extend polyester neutron shielding material. The EP–62 is retained with the inner shell of the shield tank by a bolt which fastens the bottom of the vessel to the baseplate. Heat from the package contents is conducted to the outer shell of the shield tank by radial aluminum webs that connect the inner shell to the outer shell. Axial fins on the outer shell dissipate the heat to the environment.

DOE estimates that one shipment per year would support the operational requirements for 5 kilograms (11 pounds) of plutonium per year. Based on the heat limit of the cask, 203 watts (11.6 British thermal units per minute), the casks would be loaded with up to 350 grams (12 ounces) of plutonium-238. Therefore, each SST/SGT shipment would carry about fifteen 5320 casks.

The *Environmental Assessment of the Import of Russian Plutonium-238* (DOE 1993) analyzed the smaller Mound 1-kilowatt packages. These packages were actually used for two successful shipments. The risk analysis results from that environmental assessment have been used in this NI PEIS.

J.3.3.4 Irradiated Target Assembly Packages for Medical and Industrial Isotopes

Although there are two different target vehicles, the long irradiated target vehicle (up to 1 meter [3 feet] in length) and the rapid retrieval target vehicle (up to 20 centimeters [8 inches] in length), both irradiation vehicles would be shipped from FFTF at Hanford to the Radiochemical Processing Laboratory (RPL) at Hanford using the T–2 shipping cask. The elements or pins from the long irradiation target vehicles would be inserted directly into the shipping cask, whereas the rapid retrieval targets would be inserted into a smaller package, which would be inserted into the shipping cask. For the purposes of this analysis, it is assumed that the elements or pins from one assembly (i.e., carrying a single target isotope and its associated impurities), would be shipped at a time. Similar packages would be used at the new reactor or accelerator facility.

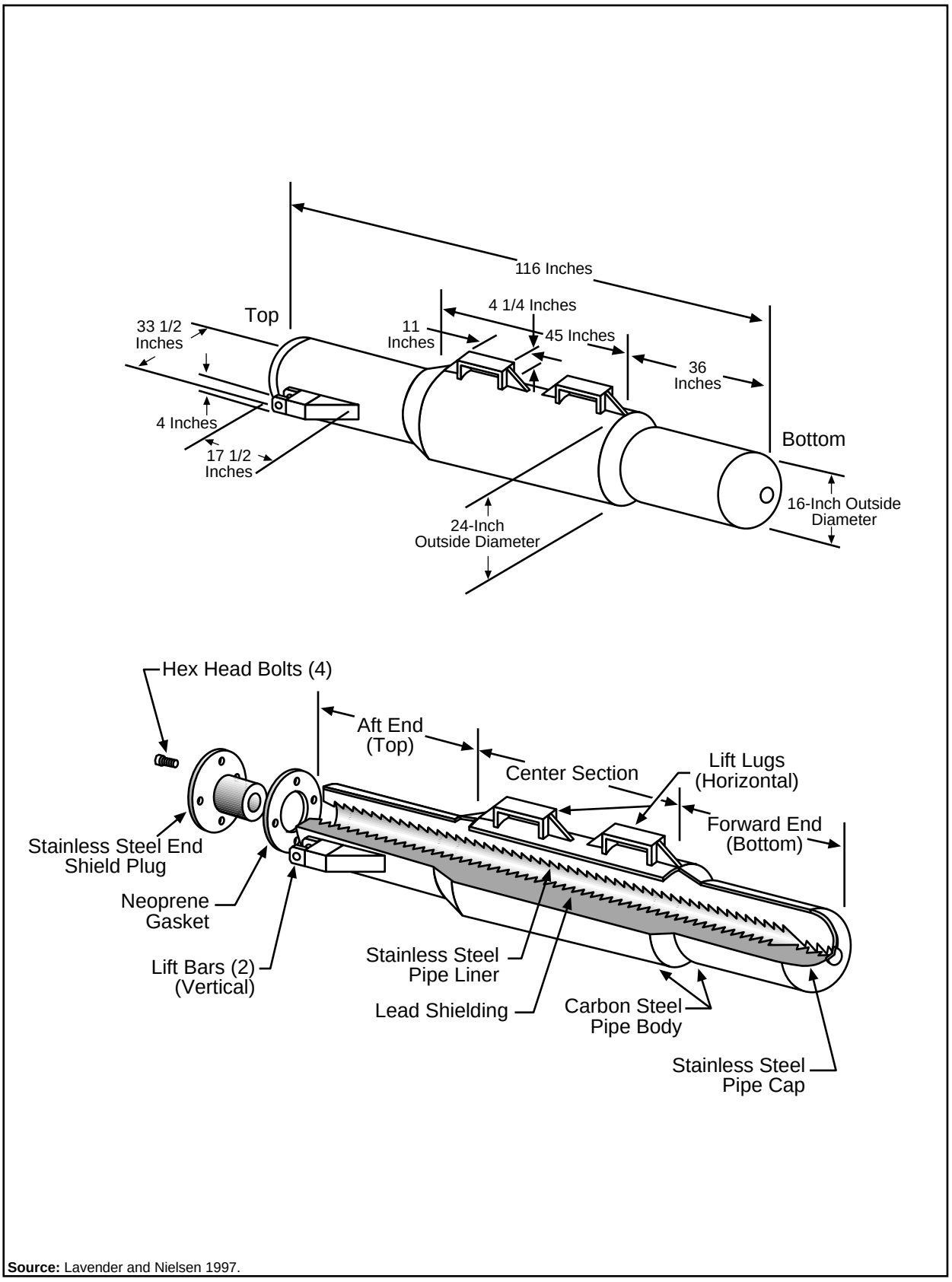
J.3.3.4.1 Long-Length Irradiated Target Vehicle Shipping Cask

The long-length irradiated target shipping cask would be used for transporting irradiated targets. A typical example for this type of casks is the T–2 shipping cask (**Figure J–5**). This cask has been used at Hanford in the past and is certified to carry sodium-bonded metal fuel pins. The T–2 meets the requirements for a Type B shipping package.

The T–2 cask is 295 centimeters (116 inches) long with a 254-centimeter (100-inch) long by 15-centimeter (6-inch) inner diameter liner made of schedule 40 stainless steel pipe. The outside shell of the cask is made in three sections. The center section is a 61-centimeter (24-inch) outer diameter schedule 40 carbon steel pipe. Each end section is made of 41 centimeter (16 inch) outer diameter schedule 40 carbon steel pipe. The space between the liner and shell is filled with lead for shielding. At the top, there is a 19-centimeter (7-5/8-inch) inner diameter opening which is closed by a 20-centimeter (8-inch) thick stainless steel shield plug. Figure J–5 shows the cross section of a T–2 cask and its dimensions. The cask is enclosed in a steel shipping case during transport (Lavender and Nielsen 1997).

J.3.3.4.2 Rapid Retrieval Target Vehicle Package

The rapid retrieval targets would be packaged in a container and inserted in the T–2 shipping cask for transport to RPL from FFTF. A “shielded pig” or sample pig would be used for packaging the rapid retrieval targets. For the purposes of this analysis, the proposed “shielded pig” is assumed to be 122 centimeters (48 inches)



Source: Lavender and Nielsen 1997.

Figure J–5 T–2 Shipping Cask: Long-Length Irradiated Target

tall by 15 centimeters (6 inches) in diameter. Based on existing sample pig design information, the "shielding pig" inner and outer walls will be constructed with schedule 40 carbon steel pipe. Lead shielding would be provided between the inner and outer carbon steel pipes in a "sandwich" configuration. The T–2 would be equipped with spacers to prevent the movement of the sample pig within the T–2 cask cavity. Smaller (i.e., shorter) shielded pigs have been approved for use at Hanford in the past. It is anticipated that the longer shielded pig proposed for the transport of the rapid retrieval targets would also be approved for use at Hanford (Lavender and Nielsen 1997).

J.3.3.5 Packages for Separated Medical Isotopes

DOE has been producing and shipping medical and industrial isotopes for several decades. This NI PEIS proposes alternatives that expand the amount and number of isotopes that DOE can supply for its customers (ultimately to hospitals, research laboratories, and other private and government users of isotopes). Alternatives proposed in this NI PEIS pose no new shipping issues or requirements for package development. Various Type A and Type B containers could be used for shipping the separated isotopes from Pacific Northwest National Laboratory (PNNL) to the pharmaceutical distributor. Following target processing, the separated isotopes, either as liquids, gases or solids, would be placed in glass vials and inserted into the protective container. For this analysis, it is assumed the CI-20WC-2A would be used for the separated isotope shipments. The CI-20WC-2A is a specification package, constructed and used in accordance with 49 CFR Section 178.362.

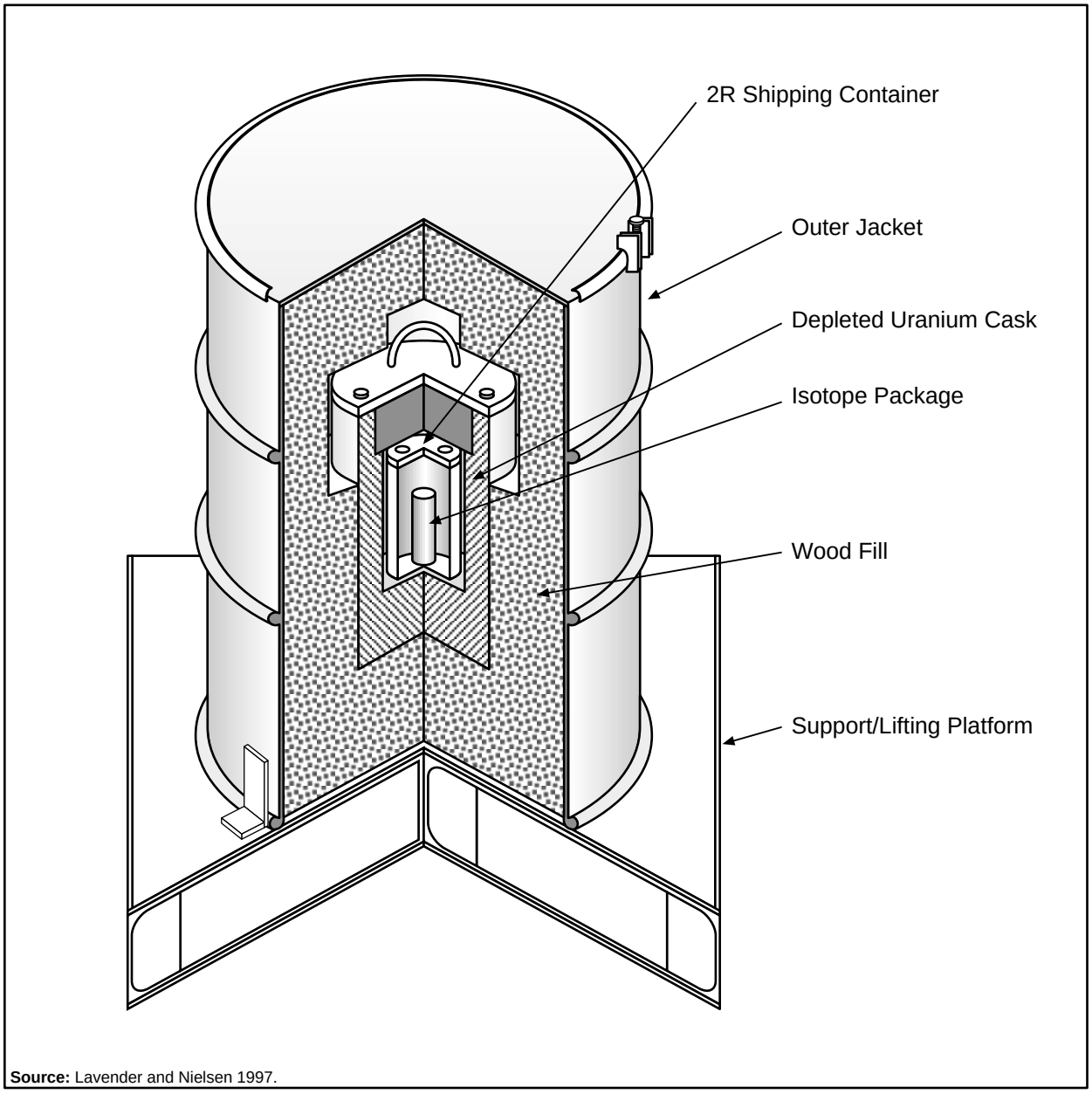
The CI-20WC-2A container, shown in **Figure J–6**, consists of an outer steel jacket, 62 centimeters (24.25 inches) high by 46 centimeters (18 inches) in diameter and an inner depleted uranium shipping cask, 22.9 centimeters (9 inches) high by 11.4 centimeters (4.5 inches) in diameter and 4.6 centimeters (1.8 inches) thick. Between the outer steel jack and the depleted uranium cask is a wooden impact limiter 14.0 centimeters (5.5 inches) thick on the sides, top, and bottom. The inner and outer walls of the depleted uranium cask are constructed of schedule 40 carbon steel pipe with a gasketed and bolted flange closure. The inner cavity of the depleted uranium cask, which is designed to accept a 2R shipping container, is 15.2 centimeters (6 inches) high and 7.9 centimeters (3.1 inches) in diameter.

The 2R shipping container is a 6.9-centimeter (2.7-inch) outer diameter by 14.1-centimeter (5.6-inch) long stainless steel, gasketed and threaded container. Spacers would be placed within the 2R shipping container to limit the movement of the glass vial containing the separated isotope (Lavender and Nielsen 1997).

J.3.3.6 Mixed Oxide Fuel Package

Two European casks could be considered for shipping the SNR–300 mixed oxide fuel to FFTF. The major characteristics are:

1. The British cask GB/136 (owned by AEA Technologies)
   - Capacity is four assemblies per cask
   - 16 casks are available
   - Activity is less than 400 kilocuries
   - Heat load is less than 400 watts
   - Head load/assembly is less than 150 watts
   - Heat load may limit to only two to three assemblies in some casks
   - Quantity of casks makes this cask system desirable



Source: Lavender and Nielsen 1997.

Figure J–6 CI-20WC-2A Shipping Casks: Separated Isotopes

2. The SNR–300 fuel assembly cask
   - Capacity is nine assemblies per cask
   - Two casks are available
   - Activity is less than 57 kilocuries
   - Heat load/assembly is less than 115 watts
   - Capacity, and with this cask being built for this fuel, makes this cask system desirable

Neither of the casks is currently certified by DOE, NRC, or DOT for use in the United States. SBK of Germany, owner of the SNR–300 fuel, could obtain a DOT Certificate of Competent Authority for either cask (Hiller 2000). Alternatively, SBK could select an existing cask for modification or design a new cask. For the purpose of conservative analysis, DOE assumes that two assemblies are shipped in each Type B package.

The shielding and accident performance parameters used in the impact analysis are typical for Type B packages.

J.3.3.7 Highly Enriched Uranium Package

DOE has several Type B packages that could be used to ship highly enriched uranium oxide to the fuel fabricator. The two identified packages are DT-22 and DC-1. Several other packages are used in normal commercial shipment of highly enriched uranium and could be used in this application.

J.3.3.8 Highly Enriched Uranium Fuel Packages

Alternatives 1 (Restart FFTF) and Alternative 4 (Construct New Research Reactor) would require packages for delivering fresh uranium fuel to the sites. Package selection cannot be made at this time, so this section will describe the general characteristics of packages that would be used. The driving requirement comes from 10 CFR Section 71.55 (e), stating "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR Section 71.73 ("Hypothetical accident conditions"), the package would be subcritical." A criticality analysis would have to be performed to design the packages.

Highly Enriched Uranium Fuel for FFTF

FFTF can use highly enriched uranium or mixed oxide fuel. For all options in Alternative 1, FFTF would require highly enriched uranium fuel for part of it's mission. DOE estimates that FFTF would use 12 to 15 fuel assemblies per year containing about 26.5 kilograms (16.5 pounds) of 35 percent enriched uranium. These quantities of uranium would require Type B packages to withstand the hypothetical accident conditions defined in 10 CFR Part 71. A reasonably large Type B package would hold four assemblies, and would be carried alone on a truck. Alternatively, two smaller Type B packages, each holding two assemblies, could be used. In either case, the same number of shipments would be required.

J.3.3.9 Nuclear Research and Development Materials Test Transport

The T–3, an existing licensed DOT irradiated fuel shipping cask is available for offsite transportation (e.g., shipment of fueled tests to other DOE facilities for diagnostic examinations). This cask can accommodate shipments of pins or a single FFTF fuel assembly, as well as non-fuel experiments and materials. There are three of these casks available for shipping material within the fuel descriptions of approved packages shown in the T–3 Certificate of Compliance, or for anyone willing to pay for the addition of new packages through an addendum to the NRC Certificate of Compliance. Equipment is available to use the T–3 cask in either the horizontal or the vertical position. Following are some facts about the T–3 cask:

- NRC License, docket no. 71–9132
- DOT Certificate (IAEA Certificate of Compliance)
- DOE License from 1998–1999
- Designed to meet all licensing requirements of 10 CFR Part 71 for a Fissile Class III, Type B shipping container
- 17,300 kilograms (38,200 pounds) gross weight for a loaded container; overall package weight, 19,296 kilograms (42,595 pounds)
- Overall cask dimensions: 67 centimeters (26 inches) in diameter by 450 centimeters (177 inches) long
- Internal cavity dimensions: 20 centimeters (8 inches) in diameter by 373 centimeters (147 inches) long
- Cavity volume: 0.1 cubic meters (4.3 cubic feet)
- Decay heat for contents: 1,400 watts (80 British thermal units per minute) maximum

- Weight of contents: 320 kilograms (700 pounds) maximum
- Designed and fabricated to meet ASME Section III (1977 ed.)
- Design allows shipping intact FFTF standard core components (control rods, reflector, experiments, etc.)
- Design envelope includes use of fuel pin containers (irradiated fuel pins)
- Design allows for handling of hardware shipping containers (irradiated hardware)

No specific shipments or mission has been identified for this cask, however it is included as a general purpose package available on Hanford.

J.3.4 Safeguarded Transportation

DOE anticipates that any transportation of neptunium, plutonium dioxide, mixed oxide fuel, or highly enriched uranium would be required to be made through use of the Transportation Safeguards System and shipped using SST/SGTs. Transportation safeguards are required for (1) nuclear explosives; (2) components moved in a single shipment that could comprise a complete nuclear explosive; (3) any form of uranium-235 enriched 20 percent or greater in quantities of 5 kilograms or more, or uranium-233 or plutonium in quantities of 2 kilograms or more; (4) classified forms of plutonium and uranium-235 regardless of quantity as requested by Heads of Field Elements; (5) DOE-owned plutonium in any quantity to be transported by air; or (6) any form of plutonium-238 in excess of 5 grams (DOE Order Supplemental Directive AL 5610.14). The SST/SGT is a fundamental component of the Transportation Safeguards System. The Transportation Safeguards System is operated by the DOE Transportation Safeguards Division of the Albuquerque Operations Office for the DOE Headquarters Office of Defense Programs. Based on operational experience between fiscal year 1984 and fiscal year 1998, the mean probability of an accident requiring the tow-away of the SST/SGT was 0.058 accident per million kilometers (0.096 accident per million miles) (Claus and Shyr 1999). By contrast, the rate for commercial trucking in 1989 was about 0.3 accident per million kilometers (0.5 accident per million miles) (Saricks and Tompkins 1999). Commercial trucking accident rates and SST/SGTs were used in the human health effects analysis. Since its establishment in 1975, the Transportation Safeguards Division has accumulated more than 151 million kilometers (94 million miles) of over-the-road experience transporting DOE-owned cargo with no accidents resulting in a fatality or release of radioactive material.

Neptunium must be handled under the safeguards applicable to special nuclear materials, in accordance with DOE Office of Safeguards and Security guidance (McCallum 1999). Pure neptunium is a form of neptunium that would be desirable as a potential weapons material, so this NI PEIS assumes that the neptunium shipped from SRS to the storage locations would need to be shipped under the Transportation Safeguards System. The unirradiated and irradiated targets would carry much less neptunium per shipment, and the form of the neptunium would be less desirable for diversion, so this NI PEIS assumes that the neptunium shipped from SRS to the storage locations might be shipped under the Transportation Safeguards System program. The unirradiated and irradiated targets carry much less neptunium per shipment, and the neptunium is in a less desirable form, so the safeguards requirements would be lower. DOE's policy is to ship DOE-owned Safeguard Categories I and II quantities of special nuclear material and other forms and quantities of strategic materials under the safeguards protection of the Transportation Safeguards System program (DOE Order 5610.14). DOE Order 474.1, *Control and Accountability of Nuclear Materials* contains the methodology for determining the Safeguards Categories of the various nuclear materials that DOE handles. The highly enriched uranium, the highly enriched uranium fuel and mixed oxide fuel required for operation of FFTF will be transported under safeguards protection.

Although DOE may choose to use the Transportation Safeguards System program for unirradiated and irradiated target shipments, for the purposes of conservative safety analysis and flexibility in package selection, this NI PEIS assumes that commercial vehicles are used for target shipments. Under DOE Order 474.1,

plutonium-238 would be in a safeguard category less than Categories I and II, which require the use of a safe, secure trailer. However, DOE Order Supplemental Directive AL 5610.14 directs the use of the Transportation Safeguards System for shipments of plutonium-238.

The SST/SGT is a specially designed component of an 18-wheel tractor-trailer vehicle. While 49 CFR Section 173.7(b) exempts SST/SGT shipments from DOT regulations, DOE operates and maintains these vehicles in a way that exceeds DOT requirements. Although details of vehicle enhancements and some operational aspects are classified, key characteristics of the SST/SGT system include the following:

- Enhanced structural characteristics and a highly-reliable tie-down system to protect cargo from impact
- Heightened thermal resistance to protect the cargo in case of fire (newer SST/SGT models)
- Established operational and emergency plans and procedures governing the shipment of nuclear materials
- Various deterrents to prevent unauthorized removal of cargo
- An armored tractor component that provides courier protection against attack and contains advanced communications equipment
- Specially designed escort vehicles containing advanced communications and additional couriers
- 24-hour-a-day real-time communications to monitor the location and status of all SST/SGT shipments via DOE's Security Communication system
- Couriers, who are armed Federal officers, receive rigorous specialized training and are closely monitored through DOE's Personnel Assurance Program
- Significantly more stringent maintenance standards than those for commercial transport equipment
- Conduct of periodic appraisals of the Transportation Safeguards System operations by the DOE Office of Defense Programs to ensure compliance with DOE orders and management directives, and continuous improvement in transportation and emergency management programs

Loading and unloading of SST/SGTs at DOE sites is routinely done in accordance with site facility and Transportation Safeguards Division procedures. However, special attention is required at commercial facilities and military ports. The DOE SST/SGT operations team will direct and approve loading and securing of packages within SST/SGT vehicles and will be solely responsible for closing and securing SST/SGT vehicles and cargo areas prior to transport. DOE will take custody of packaged mixed oxide nuclear reactor fuel loaded on SST/SGT vehicles for transport at the military port and of the packaged highly enriched uranium or highly enriched uranium nuclear fuel at the commercial site. DOE will require that the commercial German and/or Scottish entities involved in shipping the material fully comply with the Certificate of Compliance for the package and applicable NRC and DOT regulations in preparing and offering packaged mixed oxide fuel for transportation, including proper shipping papers and nuclear material transfer forms. DOE anticipates that, if applicable, approved IAEA safeguard seals will be placed on packages in accordance with established protocols and procedures by the shippers, DOE, and other cognizant authorities prior to release of loaded packages for transport. IAEA safeguard seals may also be applied to transport vehicles.

Task interactions between Transportation Safeguards Division operations teams, the SST/SGT operations center, the shipping company, and military port operations and security personnel involved in loading, securing, and dispatching SST/SGT shipments will be conducted in accordance with the requirements of DOE Orders 461.1, 5632.1C, and 474.1 and SST/SGT operations procedures. The military port and ship will provide necessary labor, loading areas and docks, and package-handling equipment that is necessary for loading mixed oxide transportation packages into SST/SGTs. Personnel involved in fuel-handling operations will be required to have a "need to know" and possess either appropriate NRC [10 CFR 73.50(c)(1)] or DOE Level 3 (DOE M 474.1) access authorization. In dispatching shipments of mixed oxide fuel to FFTF, DOE's SST operations team and operations center will also coordinate with the security operations center at a DOE site. Estimated time of arrival, shipment, and material accountability information will be transmitted to designated persons at the FFTF in accordance with prearranged protocols. DOE anticipates the time necessary to prepare, load, secure, and dispatch SST/SGTs to be on the order of less than 1 day (per convoy).

DOE realizes that the use of SST/SGT vehicles complicates package handling. ORNL/TM-13427 (Ludwig 1997) provides the following general dimensions for an SST:

Gross vehicle weight rating	36,288 kilograms (80,000 pounds)
Maximum payload	6,169 kilograms (13,600 pounds)
Trailer overall length	18.3 meters (60 feet)
Trailer overall width	259 centimeters (102 inches)
Trailer overall height	410 meters (13 feet)
Trailer rear door width	179.1 to 215.9 centimeters (70.5 to 85 inches)
Trailer rear door height	229 centimeters (990 inches)
Trailer floor height above roadway	144 centimeters (56.5 inches)
Tractor trailer minimum turning radius	11.4 meters (37.5 feet)

SGT dimensions are similar. The payload and physical dimensions of the trailer will constrain the selection of a cask for the mixed oxide fuel and for plutonium-238 targets. Additionally, a gurney or similar device will be necessary to place the cask into the SST/SGT. The ship, port, or facility crane would place the cask on the gurney, and the gurney would load the cask into the SST/SGT. The cask would be tied down in accordance with normal SST/SGT operational procedures.

J.3.5 Ground Transportation Route Selection Process

According to DOE guidelines, radioactive material shipments must comply with both NRC and DOT regulatory requirements. NRC regulations cover the packaging and transport of neptunium, plutonium and waste, whereas DOT specifically regulates the carriers and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements. The highway routing of nuclear material is systematically determined according to DOT regulation 49 CFR Part 397 for commercial shipments. Specific routes cannot be publicly identified in advance for DOE's Transportation Safeguards Division's shipments because they are classified to protect national security interests.

DOT routing regulations require that shipment of a highway route controlled quantity of radioactive material be transported over a preferred highway network, including interstate highways, with preference toward interstate system bypasses and beltways around cities and state-designated preferred routes. A state may designate a preferred route to replace or supplement the interstate highway system in accordance with DOT requirements (49 CFR Section 397.103).

Carriers of highway route-controlled quantities are required to use the preferred network unless they are moving from their origin to the nearest interstate highway or from the interstate highway to their destination,

they are making necessary repair or rest stops, or emergency conditions render the interstate highway unsafe or impassable. The primary criterion for selecting the preferred route for a shipment is travel time. Preferred routing takes into consideration accident rate, transit time population density, activities, time of day, and day of the week.

The HIGHWAY computer code (Johnson et al. 1993) is used for selecting highway routes in the United States. The HIGHWAY database is a computerized road atlas that describes about 386,400 kilometers (240,000 miles) of roads. The Interstate System and all U.S. (U.S.-designated) highways are completely described in the database. Most of the principal state highways and many local and community roads are also identified. The code is updated periodically to reflect current road conditions and has been benchmarked against reported mileages and observations of commercial truck firms. Features in the HIGHWAY code allow the user to select routes that conform to DOT regulations. Additionally, the HIGHWAY code contains data on population densities along the routes. The distances and populations from the HIGHWAY code are part of the information used for the transportation impact analysis in this NI PEIS.

J.3.6 Shipment of SNR–300 Fuel

The SNR–300 reactor, located at Kalkar in the northwest of Germany, was designed and constructed as a 327 megawatt (electric) fast breeder power reactor. However, in the 1980s, startup and operation fell into disfavor and the German government decided not to operate the SNR–300 reactor.

In parallel with the construction of the SNR–300 reactor, and in anticipation of its ultimate operation, approximately 205 mixed oxide fuel assemblies were fabricated in Europe. This unused inventory of reactor fuel is now stored at Dounreay, Scotland, and has raised international concerns over its possible proliferation as weapons-usable nuclear material. Currently, there is interest in having these surplus fuel assemblies transferred to the United States for disposition to reduce and eventually eliminate the proliferation potential of this material.

The SNR–300 fuel is very similar in both composition and construction to the fuel used in the FFTF at Hanford. The SNR–300 fuel assemblies, if reconfigured for the FFTF, could make about 150 to 160 FFTF assemblies. This could supply two FFTF core loads for approximately 15 years of FFTF operation at 100 megawatts.

Conversion of SNR–300 fuel into FFTF fuel has been previously studied. Reconfiguration to FFTF fuel would include disassembly of SNR–300 fuel assemblies, removal of fuel pin end caps in a glovebox, addition of tag gas and retainer, re-closing fuel pins (weld) and leak testing, addition of wire wrap to fuel pins, inverting the pins and reinsertion into an FFTF fuel assembly duct, completion of FFTF bundle assembly, and quality assurance verification. This would be done in Europe prior to packaging the fuel for shipment to the United States.

The risk of possible use of nuclear material for non-peaceful purposes underlines the need for its special protection. Therefore, effective systems are required to protect this material from theft, sabotage or other malicious acts. The elaborate measures to be taken to ensure the safety and security of transatlantic mixed oxide shipments are provided in:

- *Convention on the Physical Protection of Nuclear Material*, IAEA publication INFCIRC 274
- *Recommendations on the Physical Protection of Nuclear Material*, IAEA publication INFCIRC 225

- *Code for the Safe Carriage of Irradiated Nuclear Fuel, Plutonium and High-Level Radioactive Wastes in Flasks on Board Ships* (IMO 1993)

- DOE orders and 10 CFR Part 73

J.3.6.1 Port Selection

Physically, any seaport could receive mixed oxide fuel. Legally, the mixed oxide fuel could be brought into many commercial and military ports. In the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (*Foreign Research Reactor Spent Nuclear Fuel EIS*) (DOE 1996), DOE developed and implemented a systematic process for selecting ports of entry for spent nuclear fuel. Information needed to evaluate ports and port activities, and the potential environmental impacts (incident-free and accidents) associated with the receipt of foreign research reactor spent nuclear fuel from vessels were collected and evaluated. In the *Foreign Research Reactor Spent Nuclear Fuel EIS*, 153 commercial ports and 13 military ports were evaluated (DOE 1996).

The criteria used for screening ports in the *Foreign Research Reactor Spent Nuclear Fuel EIS* were 1) appropriate port experience, 2) safe port transit to open ocean; 3) appropriate port facilities for safe receipt, handling and transshipment; 4) ready intermodal access; and 5) low human population of the ports and along transportation routes (DOE 1996). These same criteria can be used to identify ports for receiving mixed oxide fuel from Europe. **Table J–1** shows the military ports (the analysis of Hampton Roads, Virginia, covers several military and commercial facilities in that area) considered in the *Foreign Research Reactor Spent Nuclear Fuel EIS*, the distance to Hanford and the number of persons along the route. The following discusses the application of these criteria for mixed oxide fuel.

Table J–1 Overland Distances from Military Ports to Hanford and Affected Persons Along the Routes

Port	Distance (kilometers [miles])	Number of Affected Persons
Eastern ports		
Charleston Naval Weapons Station, South Carolina	4,677 (2,894)	609,000
Military Ocean Terminal Sunny Point, North Carolina	5,157 (3,194)	679,000
Mayport, Florida	4,754 (2,964)	624,000
Kings Bay, Georgia	4,685 (2,926)	555,000
Pensacola, Florida	4,430 (2,767)	549,000
Yorktown, Virginia	4,717 (2,946)	569,000
Hampton Roads, Virginia	4,748 (2,949)	694,000
Western ports		
Military Ocean Terminal Bay Area, California	1,531 (951)	263,000
Bremerton, Washington	451 (282)	143,000
Everett, Washington	397 (248)	135,000
Port Hueneme, California	2,030 (1268)	386,000
Port Townsend, Washington	666 (416)	159,000

Note: All except the Charleston Naval Weapons Station are from DOE 1996. Charleston Naval Weapons Station is from Table J–2.

In addition to the military ports listed in Table J–1, Naval facilities at the following locations could be used: San Diego, California; Seal Beach, California; Ingleside, Texas; Pascagoula, Mississippi; Pensacola, Florida; Earle, New Jersey; and Groton, Connecticut. The distances and affected persons fall within the range of those

listed in Table J–1. If DOE decides to import mixed oxide fuel, additional analysis will be required to select the appropriate port(s).

Criterion 1, Appropriate Port Experience. DOE believes that all military ports could establish a secure area for loading mixed oxide fuel packages into SST/SGTs. Therefore, any of the named ports could safely and securely handle these packages. Charleston Naval Weapons Station has been the primary port for receipt of foreign research reactor spent nuclear fuel for the last 5 years. Dozens of casks have been safely and securely received and transported to DOE facilities. Therefore, it is clearly the most experienced port. Military Ocean Terminal Sunny Point received two foreign research reactor spent nuclear fuel casks in 1994, so it has some experience. Military Ocean Terminal Bay Area has received several packages of foreign research reactor spent nuclear fuel from training, research, and isotope reactors (TRIGA reactors built by General Atomics) and shipped these packages to INEEL. The port at Hampton Roads, Virginia, has experience handling spent fuel casks and has recent experience handling Russian plutonium-238 for DOE (DOE 1993).

Criterion 2, Safe Port Transit. All of the ports listed have demonstrated safe port transit based on their continuous and routine usage by seagoing military vessels. DOE could choose to provide enhanced safety and security in the immediate vicinity of ports using necessary harbor patrol, Coast Guard and Naval assets. However, western ports require transit through the Panama Canal. While traveling through the Panama Canal, the ship would be in Panamanian waters, where DOE could not directly request assistance from other assets.

Criterion 3, Appropriate Port Facilities. The *Foreign Research Reactor Spent Nuclear Fuel EIS* evaluated Charleston Naval Weapons Station, Military Ocean Terminal Sunny Point and Military Ocean Terminal Bay Area in detail and determined that they had appropriate facilities. Essentially all working seaports have port cranes capable of lifting mixed oxide casks from the ships. At least a 30-metric ton (33-ton) capacity crane would be adequate. For seaports without a port crane, portable cranes are available in most areas. Purpose-built ships are moderate sized oceangoing vessels, and all of the identified ports have berthing facilities with adequate water depth and length to allow safe access. While Charleston Naval Weapons Station, Military Ocean Terminal Sunny Point and Military Ocean Terminal Bay Area have actually used their facilities and procedures to unload Type B casks for DOE, DOE considers the other military ports to have appropriate facilities and could establish procedures for the security necessary around the ship carrying the mixed oxide fuel.

Criterion 4, Ready Intermodal Access. Access to other modes of transportation, such as rail and barge routes, was relevant for the *Foreign Research Reactor Spent Nuclear Fuel EIS* but are not relevant to this NI PEIS since the mixed oxide fuel will travel over roads in SST/SGTs.

Criterion 5, Low Human Population. The distance to Hanford and the population along the routes are similar for each of the eastern ports. All are within 15 percent of the average distance and number of affected persons. The western ports vary significantly, but are all lower than the eastern ports. The *Foreign Research Reactor Spent Nuclear Fuel EIS* did not calculate the populations around all of the military ports listed in Table J–1. However, it conducted detailed risk analyses for delivery of foreign research reactor spent nuclear fuel to ports located near heavily populated metropolitan areas such as New York, New York (port of Elizabeth, New Jersey) and Los Angeles, California (port of Long Beach, California). The accident risk for direct shipment to these ports is less than 1×10^{-5} latent cancer fatality per shipment. The risk of foreign research reactor spent nuclear fuel material bounds the risk of fresh mixed oxide fuel because of the fission products in spent nuclear fuel. The population of the New York and Los Angeles metropolitan areas bounds the population in the area of any military ports. Therefore, all of the ports listed meet the low human population criteria for the area around the port, and the populations along the routes are as shown in Table J–1.

The voyage distance from Europe to the eastern United States is about 4,000 nautical miles, and the distance to the western United States is about 8,000 nautical miles, through the Panama Canal. There are no known restrictions for passing mixed oxide fuel through the Panama Canal. Using the *Foreign Research Reactor Spent Nuclear Fuel EIS* methodology, the voyage duration for the east coast is about 12 days, about 24 days for the west coast. Traveling to the west coast along a route south of South America or Africa is more than double the distance of the route through the Panama Canal and considered to be prohibitive for practical reasons.

DOE is currently in negotiation with Germany on the details for receiving this fuel. The Germans would be responsible for the fuel until it is delivered to DOE at a U.S. port. The Germans may not be willing to ship the fuel to a western port because of the cost associated with the longer voyage, and possible safety and security issues associated with using the Panama Canal. On the open seas, purpose-built ships are considered to be safe and secure based on their design, the distance from threats, and their constant communication with authorities. However, they are more vulnerable in the constrained waters of the Panama Canal, since these waters are not controlled by German or American authorities.

In the *Foreign Research Reactor Spent Nuclear Fuel EIS* Record of Decision, DOE decided to use military ports to take advantage of their characteristics to increase the safety and security of the spent fuel transportation process. DOE concluded that the use of military ports provides additional confidence in the safety of shipments due to the increased security. This could also require much of the spent nuclear fuel to be shipped by chartered ships because commercial ships do not schedule stops at military ports. Since the security issues are far greater for fresh mixed oxide fuel than for spent nuclear fuel because of the potential for proliferation, DOE would use a military port to bring the SNR–300 into the country. This conclusion is consistent with Criterion 3, Appropriate Port Facilities. The impact analysis is described in Section J.6.2. The Charleston Naval Weapons Station is used for the purpose of impact analysis.

J.3.6.2 Purpose-Built Vessels

As used here, purpose-built vessels are those vessels specifically designed to transport spent nuclear fuel casks (**Figure J–7**). These vessels are not used for the transport of any other cargo and they operate as dedicated vessels. Casks are loaded directly into the holds of the vessel because the cargo compartments contain the hardware needed to mate with the tiedown fixtures of the cask. If the vessel has no crane, dockside cranes are used for loading and unloading. The cargo compartments are typically intended to handle a specific cask; other casks cannot be used without modification to the tiedown mechanisms.

At present, purpose-built vessels are operated by Pacific Nuclear Transport Services of Japan, by the Swedish Nuclear Fuel and Waste Management Company, and by British Nuclear Fuels, Limited. They are used to move spent nuclear fuel from operating nuclear power plants to spent nuclear fuel processing facilities operated by COGEMA and British Nuclear Fuels, Limited; or, in the case of Sweden, to the repository in Forsmark. Since 1998, they have been used to transport spent nuclear fuel from foreign research reactors to the Charleston Naval Weapons Station. There are no U.S.-owned purpose-built vessels for spent nuclear fuel transport.

Pacific Nuclear Transport Services operates a fleet of purpose-built vessels that carried mixed oxide fuel from Europe to Japan in the summer of 1999 (COGEMA, BNFL, ORC 2000). Pacific Nuclear Transport Services' vessels are representative of, but not identical to the other fleets. All vessels in the fleet are certified to INF3—the highest safety category of the International Maritime Organization for nuclear voyages. The vessels have been designed and built specifically to carry these nuclear materials. They employ a range of safety features far in excess of those found on conventional cargo vessels. The vessels are constructed with double hulls, effectively making them able to withstand a severe collision with a much larger vessel without penetrating the inner hull. Each vessel has two sets of navigation, communications, cargo monitoring,

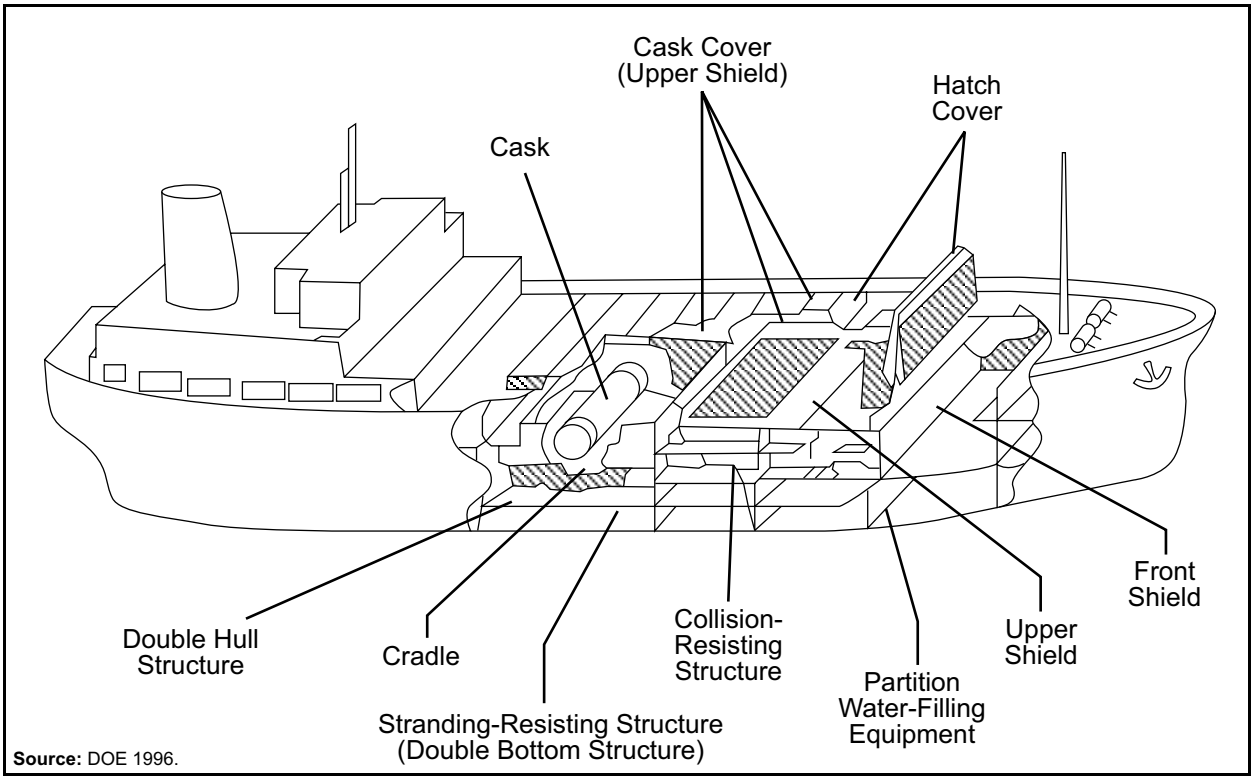


Figure J–7 Purpose-Built Vessel

electrical and cooling systems, so there is always a back-up in the event of failure or damage. The navigation system includes automatic radar plotting and collision avoidance equipment. This redundancy extends to the vessel's propulsion system. Every part of the Pacific Nuclear Transport Services' vessels are covered by a fire detection system. And every vessel has sophisticated firefighting equipment on board. In the highly unlikely event of fire, a vessel's hold, engine room, or any other on-board space, may be flooded with fire-suppressant gases. Individual holds can even be deliberately flooded, and if all the holds were flooded in this way, the vessel would still remain afloat. The vessels carry the most modern satellite and navigation, weather routing, and tracking equipment, enabling them to automatically transmit their position. While at sea, each vessel's crew can maintain permanent communication with a report center that is operated 24 hours a day (COGEMA, BNFL, and ORC 2000).

J.4 METHODS FOR CALCULATING TRANSPORTATION RISKS

The overland transportation risk assessment method is summarized in **Figure J–8**. After this NI PEIS alternatives were identified and the goals of the shipping campaign were understood, data were collected on material characteristics and accident parameters. Accident parameters were largely based on NRC studies of transportation accidents undertaken for NUREG-0170, the *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes* (NRC 1977) and the Modal Study (NUREG/CR-4829) (Fischer et al. 1987).

Representative routes that may be used for the shipments were selected for risk assessment purposes using the HIGHWAY code. They do not necessarily represent the actual routes that would be used to transport nuclear materials. Specific routes cannot be identified in advance because the routes cannot be finalized until they have been reviewed and approved by the NRC. The selection of the actual route would be responsive to

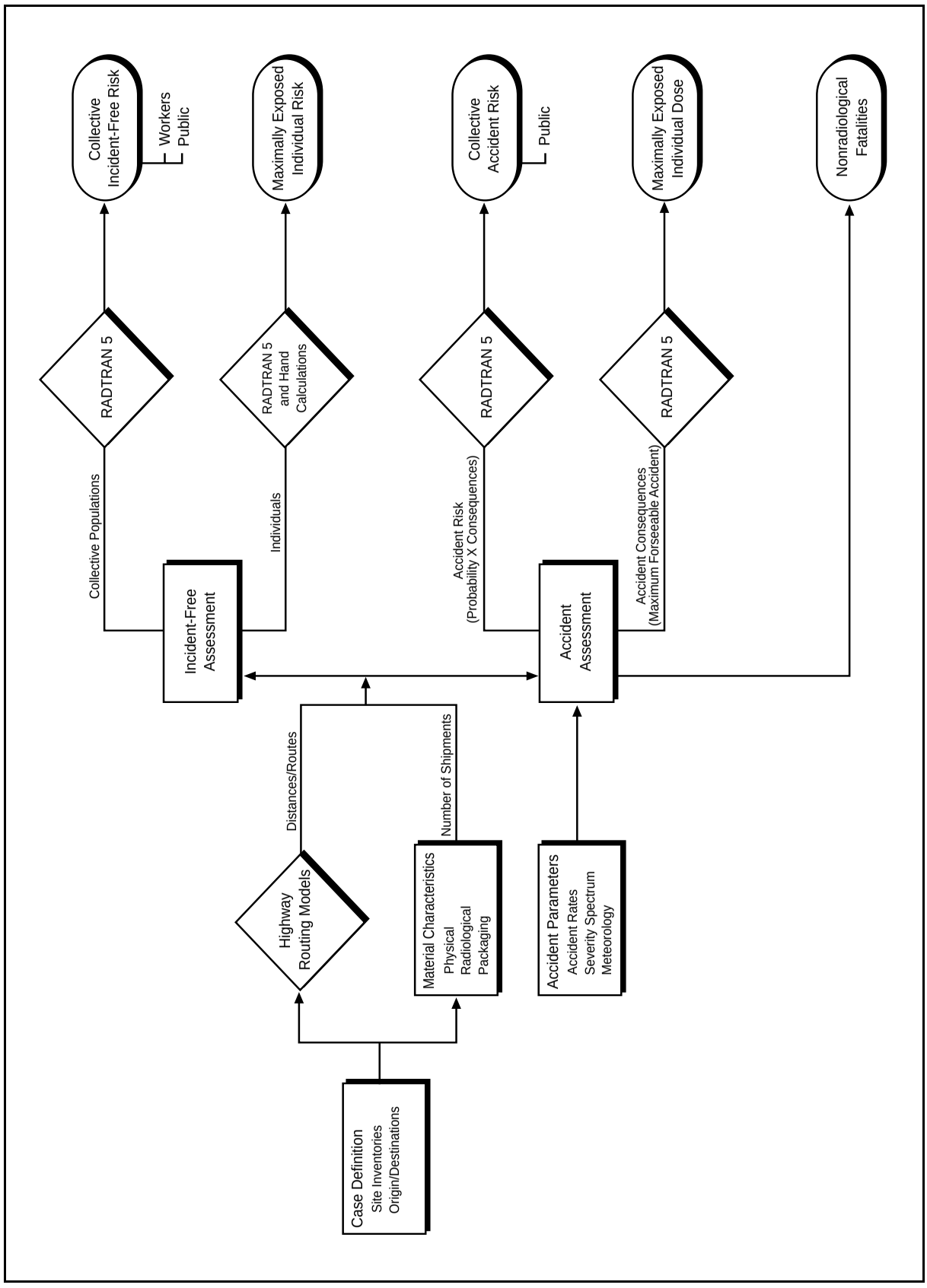


Figure J–8 Overland Transportation Risk Assessment

environmental and other conditions that would be in effect or could be predicted at the time of shipment. Such conditions could include adverse weather conditions, road conditions, bridge closures, and local traffic problems. For security reasons, details about a route would not be publicized before the shipment. Air transport of shipping casks was modeled on procedures for radiopharmaceutical suppliers using commercial air transport at the Tri-Cities Airport in Pasco, Washington. As a bounding assumption, air transport of shipping casks is assumed to be on commercial passenger flights. The shipping cask would be unloaded from their truck shipments and shipped to the destination airport with a stopover and plane change, for the purpose of impact analysis, in Salt Lake City, Utah.

The first analytic step in the transportation analysis was to determine the incident-free and accident risk factors on a per-shipment basis. Risk factors, as with any risk estimate, are the product of the probability of exposure and the magnitude of the exposure. Accident risk factors were calculated for radiological and nonradiological traffic accidents. The probabilities, which are much lower than one, and the magnitudes of exposure were multiplied, yielding very low risk numbers. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the shipping container (cask) and public exposure to the chemical toxicity of the truck exhaust. The probability of incident-free exposure is unity (one).

For each alternative, risks were assessed for both incident-free transportation and accident conditions. For the incident-free assessment, risks are calculated for both collective populations of potentially exposed individuals and for maximally exposed individuals. Handling doses are included in the transportation risk for airport and seaport handling. Truck unloading at DOE sites is included in facility dose estimates. The accident assessment consists of two components: (1) a probabilistic accident risk assessment that considers the probabilities and consequences of a range of possible transportation accident environments, including low-probability accidents that have high consequences, and (2) an accident consequence assessment that considers only the consequences of maximum foreseeable transportation accidents.

As a practical matter, the maximum foreseeable transportation accident is defined as an accident with a frequency of greater than 1×10^{-7} per year (once in 10 million years). This hypothetical accident is well beyond the "design basis" of a transportation cask. The "design basis" of a transportation cask is to survive the tests shown in Figure J–1 without releasing its contents. The risk of accidents that are less likely than the maximum foreseeable accident are included in the analysis, but specific accident sequences and consequences are not analyzed. The RADTRAN 5 computer code (Neuhauser and Kanipe 2000) uses a probability and consequence binning approach to include the impacts of these extremely rare and severe accidents. It would not be practical to analyze all potential accident forces that could affect a transportation cask because there are such a large number of potential scenarios and locations.

The RADTRAN 5 computer code (Neuhauser and Kanipe 2000) is used for incident-free and accident risk assessments to estimate the impacts on population. RADTRAN 5 was developed by Sandia National Laboratories to calculate population risks associated with the transportation of radioactive materials by a variety of modes, including truck, rail, air, ship, and barge. RADTRAN 5 was used to calculate the doses to the maximally exposed individuals.

The RADTRAN 5 population risk calculations take into account both the consequences and probabilities of potential exposure events. The RADTRAN 5 code consequence analyses include cloud shine, ground shine, inhalation, and resuspension exposures. The collective population risk is a measure of the total radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing the various alternatives.

J.5 ALTERNATIVES, PARAMETERS, AND ASSUMPTIONS

J.5.1 Description of Alternatives

Five alternatives with numerous options and a No Action Alternative with options have been identified for this NI PEIS. **Table J–2** depicts these alternatives and the following describes them from a transportation perspective.

J.5.1.1 No Action Alternative

Option 1. Plutonium-238, needed in power systems for future space missions, would continue to be purchased from Russia. The transportation analysis performed in the *Environmental Assessment of the Import of Russian Plutonium-238* (DOE 1993) would be expanded to cover the transportation from Russia, through the ports of entry, and to the Los Alamos National Laboratory (LANL) site for the time period covered by this NI PEIS, as described in Chapter 4. Medical, industrial, and research and development isotope production would continue as per the current conditions. Neptunium-237 would not be converted to oxide or shipped.

Options 2, 3, and 4. These options are like the No Action Alternative, Option 1 in all respects except for the disposition of the neptunium-237. The neptunium-237 would be converted to oxide and transported from SRS to a storage site at ORR, INEEL, or Hanford. The neptunium transportation would be the only transportation leg quantitatively analyzed in this NI PEIS.

J.5.1.2 Alternative 1—Restart FFTF

FFTF Production (Options 1 through 3). FFTF at Hanford would be used to produce up to 5 kilograms (11 pounds) per year of plutonium-238 and medical, industrial, and research and development isotopes. When the mixed oxide fuel already at Hanford is depleted, FFTF would use SNR–300 mixed oxide fuel imported from Europe, and after 21 years of operation, would switch to highly enriched uranium fuel. For the purpose of analysis, it is assumed that the mixed oxide fuel enters the United States at the Charleston Naval Weapons Station and is shipped in SST/SGTs to FFTF. The analysis includes shipment of a single mixed oxide fuel assembly for testing in the FFTF reactor and the shipping campaign during mixed oxide fuel operations. The highly enriched uranium fuel would be fabricated at a commercial fuel fabrication facility located in the eastern United States. Medical and industrial isotope target fabrication and processing would occur at Hanford, using purified materials from ORR, and the products would be shipped to commercial vendors as described in Section J.5.3. Plutonium-238 production would require the transportation of neptunium-237 from SRS to a target fabrication facility at ORR, INEEL, or Hanford; transportation of unirradiated targets from the fabrication facility to Hanford; transportation of irradiated targets from Hanford to a target processing facility at the same locations as the fabrication facility; and transportation of plutonium-238 from the fabrication facility to LANL.

FFTF Production (Options 4 through 6). Options 4 through 6 are similar to Options 1 through 3, with differences in timing and fuel source for FFTF. When the mixed oxide fuel already at Hanford is depleted, FFTF would immediately switch to highly enriched uranium fuel as described for Options 1 through 3.

J.5.1.3 Alternative 2—Use Only Existing Operational Facilities

One-Reactor Production (Options 1 through 6). The ATR at INEEL or a CLWR would be used to produce up to 5 kilograms (11 pounds) per year of plutonium-238. Therefore, production would require the transportation of neptunium-237 from SRS to a target fabrication facility at ORR, INEEL, or Hanford;

Table J–2 NI PEIS Alternatives and Options

Alternative Option Number	Irradiation Options						Storage, Target Fabrication, and Processing Options for Plutonium-238 Production		
	FFTF (Hanford)	ATR (INEEL)	HFIR (ORR)	CLWR	New Accelerator(s)	New Reactor	REDC (ORR)	FDPF (INEEL)	FMEF[a] (Hanford)
No Action Alternative									
Option 1									
Option 2							●		
Option 3								●	
Option 4									●
Alternative 1—Restart FFTF									
Option 1	●						●		
Option 2	●							●	
Option 3	●								●
Option 4	●						●		
Option 5	●							●	
Option 6	●								●
Alternative 2—Use Only Existing Operational Facilities									
Option 1		●					●		
Option 2		●						●	
Option 3		●							●
Option 4				●			●		
Option 5				●				●	
Option 6				●					●
Option 7		●	●				●		
Option 8		●	●					●	
Option 9		●	●						●
Alternative 3—Construct New Accelerator(s)									
Option 1					●		●		
Option 2					●			●	
Option 3					●				●
Alternative 4—Construct New Research Reactor									
Option 1						●	●		
Option 2						●		●	
Option 3						●			●
Alternative 5—Permanently Deactivate FFTF (with No New Missions)									

a. FMEF is being considered along with several other facilities. See Chapter 2 for details.

Key: ATR, Advanced Test Reactor; CLWR, commercial light water reactor, no defined location; FDPF, Fluorinel Dissolution Process Facility; FFTF, Fast Flux Test Facility; FMEF, Fuels and Materials Examination Facility; HFIR, High Flux Isotope Reactor; INEEL, Idaho National Engineering and Environmental Laboratory; ORR, Oak Ridge Reservation; REDC, Radiochemical Engineering Development Center.

transportation of unirradiated targets from the fabrication facility to INEEL or a CLWR; transportation of irradiated targets from INEEL or a CLWR to a target processing facility at the same location as the fabrication facility; and transportation of plutonium-238 from the fabrication facility to LANL. Medical, industrial, and research and development isotope production would continue per current conditions.

Two-Reactor Production (Options 7, 8, and 9). The ATR at INEEL and HFIR at ORR would be used to produce plutonium-238. HFIR could only produce between 1 and 2 kilograms (2.2 and 4.4 pounds) of plutonium-238 per year. Therefore, between 3 and 4 kilograms (6.6 and 8.8 pounds) per year would be produced at ATR. Production would require the transportation of neptunium-237 from SRS to a target fabrication facility at ORR, INEEL, or Hanford; transportation of unirradiated targets from the fabrication facility to ORR and INEEL; transportation of irradiated targets from ORR and INEEL to a target processing facility at the same location as the fabrication facility; and transportation of plutonium-238 from the fabrication facility to LANL. It is assumed that production rates at ATR and HFIR would require the maximum amount of transportation. For example, in Option 7 (target fabrication and processing at ORR), it is assumed that HFIR produces its minimum rate of 1 kilogram (2.2 pounds) per year of plutonium-238, which maximizes the transportation to INEEL. Medical, industrial, and research and development isotope production would continue per the current conditions.

J.5.1.4 Alternative 3—Construct New Accelerator(s)

One or two new accelerators at a generic DOE site would be used to produce up to 5 kilograms (11 pounds) per year of plutonium-238 and medical and industrial isotopes, as well as conducting nuclear research and development. The accelerator(s) would be on a DOE site to be identified later. Shipping distances, route characteristics, and material inventories are assumed to be the same as those modeled in HNF-1844 (Lavender and Nielsen 1997). Medical and industrial isotope target fabrication and processing would occur on the accelerator site, and the products would be shipped to commercial vendors. Plutonium-238 production would require the transportation of neptunium-237 from SRS to a target fabrication facility at ORR, INEEL, or Hanford; transportation of unirradiated targets from the fabrication facility to the accelerator(s); transportation of irradiated targets from the accelerator(s) to a target processing facility at the same locations as the fabrication facility; and transportation of plutonium-238 from the fabrication facility to LANL.

J.5.1.5 Alternative 4—Construct New Research Reactor

A new research reactor at a generic DOE site would be used to produce up to 5 kilograms (11 pounds) per year of plutonium-238 and medical and industrial isotopes, as well as conducting nuclear research and development. The reactor would be on a DOE site to be identified later. Shipping distances, route characteristics, and material inventories are assumed to be the same as those modeled in HNF-1844 (Lavender and Nielsen 1997). Medical and industrial isotope target fabrication and processing would occur on the reactor site, and the products would be shipped to commercial vendors. Plutonium-238 production would require the transportation of neptunium-237 from SRS to a target fabrication facility at ORR, INEEL, or Hanford; transportation of unirradiated targets from the fabrication facility to the reactor; transportation of irradiated targets from the reactor to a target processing facility at the same location as the fabrication facility; and transportation of plutonium-238 from the fabrication facility to LANL.

J.5.1.6 Permanently Deactivate FFTF (with No New Missions)

No offsite transportation of radioactive material would occur as a result of selecting this alternative. The sodium coolant would be removed from FFTF and processed at Hanford. Any transportation impacts would be negligible compared with the impacts of other alternatives. Medical, industrial, and research and development isotope production would continue per the current conditions.

J.5.2 Material Inventory

The amount of neptunium-237 that must be shipped is determined from the basic mission requirement to remove all the 466 kilograms (1,025 pounds) of neptunium-237 (Gibson 1999) from SRS to a new storage

facility. The stated mission is to produce 5 kilograms (11 pounds) of plutonium-238 per year for 35 years. **Table J–3** summarizes the masses of material and the number of shipments required to implement the various alternatives, and can be used in conjunction with Table J–1 to determine the origins and destinations of the shipments for the various alternatives. The material masses listed are those of heavy metal. The word "shipment" means a transportation leg. For example, a package that is loaded onto a truck, driven to an airport, loaded onto an aircraft, flown to another airport, loaded onto a truck, and driven to a final destination would count as three shipments (two by truck and one by air). Neptunium and plutonium shipments also contain small amounts of radioactive decay products, and irradiated targets contain fission products.

DOE estimates that about 50 kilograms (110 pounds) of neptunium-237 will have to be exposed to reactor flux to make 5 kilograms of plutonium-238. This neptunium-237 would be shipped in about 9 shipments per year, each carrying about 6 kilograms (13.2 pounds) of neptunium in the unirradiated targets. The targets would be returned to the fabrication and processing facility with about 5 kilograms (11 pounds) of neptunium-237 and less than a kilogram of plutonium. Again, about 9 shipments per year would be needed to return the irradiated targets. For options in which some or all of the irradiation is done at the same DOE site as the fabrication and processing, less transportation is required. These transportation assumptions are used for irradiation at FFTF (Alternative 1) existing reactors (Alternative 2), the new accelerators (Alternative 3), and the new research reactor (Alternative 4).

The highly enriched uranium transportation assumptions are based on the assumption of 16 fuel assemblies per year made from 35 percent enriched uranium (Nielsen 1999). The fuel assemblies would contain about 27 kilograms of heavy metal. The SNR–300 mixed oxide fuel transportation requirements come from information provided by SBK of Germany (Hiller 2000).

These assumptions are considered preliminary. Since they provide for conservative amounts of material shipped and numbers of shipments, they are considered to be adequate for impact analysis. No specific estimates of transportation requirements for other research and development isotopes have been included. However, DOE believes that they are small compared to the transportation requirements assumed for medical isotopes. If large research and development projects are scheduled in the future, they would displace some of the medical isotope or plutonium-238 production and their transportation impacts would be similar or less.

J.5.3 Transportation of Medical Isotopes

DOE isotope program sales projections are made in the context of a worldwide market for radioactive isotopes. Isotope programs market share is a small fraction of the total, but is very significant for some products, and is particularly important for a large number of isotopes that are used in relatively small quantities for research. There is uncertainty in future growth trends, and recent studies have indicated a large potential for growth if promising research developments in the medical use of radioisotopes can be brought to commercialization. DOE's production rate could increase significantly as world demand changes.

Through the duration of the period covered by this NI PEIS, the transportation impacts of the current DOE isotope programs are and would remain very low compared to the proposed new missions analyzed in this NI PEIS. These isotopes are being produced at reactor and accelerator facilities throughout the country. Selection of Alternative 1, 3, or 4 would significantly increase the production capabilities with a corresponding increase in transportation impacts. Selection of the No Action Alternative or Alternative 2 would not increase production capabilities, nor would their selection significantly affect the baseline production rate. The transportation impacts from isotopes currently produced by DOE are small compared with the impacts of NI PEIS alternatives and are neglected for the purpose of transportation risk analysis. The following describes

Table J–3 Summary of Material Shipments

Hazardous Material	Container	Applicable Alternatives	SST/ SGT[a]	Number of Shipments	Amount of Heavy Metal per Package	Packages per Shipment	Total Heavy Metal Shipped
Neptunium-237	9975[b]	NA-2 through NA-4, 2, 3, and 4	Yes	24	3 kilograms of neptunium	14	446 kilograms
Unirradiated targets (neptunium-237)	To be determined; similar to GE–2000	1, 2-1 through 2-6, and 4	No	315	6 kilograms of neptunium	1	1,750 kilograms of neptunium
		2-7		252			1,400 kilograms of neptunium
		2-8 and 2-9		126			700 kilograms of neptunium
	To be determined	3		105	500 kilograms of uranium 72 kilograms of neptunium	1	52,500 kilograms of uranium 7,560 kilograms of neptunium
Irradiated targets (plutonium-238)	To be determined; similar to GE–2000	1, 2-1 through 2-6, and 4	No	315	0.6 kilograms of plutonium 5 kilograms of neptunium	1	175 kilograms of plutonium 1, 500 kilograms of neptunium[c]
		2-7		252			140 kilograms of plutonium 1,200 kilograms of neptunium[c]
		2-8 and 2-9		126			70 kilograms of plutonium 600 kilograms of neptunium[c]
	To be determined	3		105	500 kilograms of uranium 70 kilograms of neptunium 2 kilograms of plutonium	1	52,500 kilograms of uranium 7,350 kilograms of neptunium 210 kilograms of plutonium
Plutonium-238	5320	1, 2, 3, and 4	Yes	35	0.35 kilograms of plutonium	15	175 kilograms of plutonium
Highly enriched uranium	DT-22 or DC-1	1-1 through 1-3	Yes	5	10 kilograms of highly enriched uranium	12	520 kilograms of highly enriched uranium
		1-4 through 1-6		9			1,080 kilograms of highly enriched uranium
Highly enriched uranium fuel	To be determined	1-1 through 1-3	Yes	56	26.5 kilograms of uranium	4	1,500 kilograms of uranium
		1-4 through 1-6		116			3,100 kilograms of uranium
Fresh SNR–300 mixed oxide fuel	GB/1356 or SNR-300	1-1 through 1-3	Yes	79	70 kilograms of plutonium, uranium, and americium	1	5,500 kilograms of plutonium, uranium, and americium
Irradiated targets (medical isotopes)	T-2	1, 3, and 4	No	8,610	Various	1	NA
Separated isotopes	To be determined	1,3, and 4	No	8,610	Various	1	NA

a. For purposes of analysis.
b. Either a redesigned 9975 or suitable replacement.
c. Much of the neptunium-237 is recycled into new targets during fabrication.

Note: 1 kilogram = 2.2 pounds.

Key: NA, not applicable; SST/SGT, safe, secure trailer/SafeGuards Transport.

the transportation analyzed in this NI PEIS. The isotopes produced and transported are listed in Appendix C. Over 8,000 shipments, each with two truck and one aircraft leg, would be required to deliver these isotopes to commercial vendors. The transportation impacts of these representative isotopes are expected to bound the impacts of transportation associated with unspecified future research and development activities.

The transportation evaluation addressed the shipment of enriched target materials from Oak Ridge National Laboratory (ORNL) to Hanford for target fabrication, shipping the fabricated targets to FFTF, shipping the irradiated targets from FFTF to RPL for target processing, and shipping isotope products to commercial pharmaceutical distributors. HNF-1844 (Lavender and Nielsen 1997) analyzed distributors in Boston, Massachusetts; Chicago, Illinois; and St. Louis, Missouri. Only the results of shipping the isotopes to Boston are shown in this document to ensure that the risk analysis is bounding. The transportation impacts of medical isotopes were analyzed for FFTF and are considered to be the best estimate for FFTF or the hypothetical accelerator(s) or research reactor. The impact analysis is bounding for the accelerator(s) or reactor because either would be located on a major DOE site. The following paragraphs describe the material transportation for production at FFTF, with onsite target fabrication and processing, as analyzed in HNF-1844.

With the exception of the production of actinium-227, thorium-228, and thorium-229, this evaluation assumes that the same transportation scenario, from the target material supplier to the pharmaceutical distributor, is applicable to each isotope. That is, target materials are shipped from ORNL to PNNL, fabrication targets are shipped from PNNL to FFTF, irradiated targets are shipped from FFTF to PNNL, and the separated isotopes are shipped from PNNL to the three isotope distributors. Actinium-227, thorium-228, and thorium-229 are produced by irradiating a radium-226 target. Sufficient quantities of radium-226 would be stored at the target fabrication facility; therefore radium-226 target material is not shipped from ORNL to PNNL. The rest of the transport scenario from the target fabrication facility to the pharmaceutical distributor is the same as for other isotopes.

The target materials (with the exception of radium-226) required to produce the medical isotopes are assumed to be obtained from ORNL. The target materials would be shipped on an as-needed basis from ORNL to PNNL for target fabrication. Target fabrication is assumed to occur in the 300 Area at Hanford. For this analysis, it is assumed that target material would be shipped by truck one at a time from ORNL to PNNL. This is a bounding assumption that maximizes the number of shipments, because the trucks are capable of transporting loads containing multiple types of target materials. All of the target materials receive from ORNL are nonradioactive.

The target materials would be fabricated into specially-designed targets for irradiation at FFTF. The fabricated targets would be shipped by truck from RPL/306–E to FFTF. As with the target materials shipments, it was assumed that only one unirradiated target would be shipped at a time to FFTF. Following irradiation in FFTF, the irradiated targets would be shipped to RPL for required processing. Irradiated targets were assumed to be shipped by truck one at a time from FFTF.

Following required processing and packaging, an isotope product would be shipped by truck from RPL to the Tri-Cities Airport located in Pasco, Washington. From the Tri-Cities Airport, the isotopes are transported by air, using commercial passenger flights, to an intermediate airport or hub (i.e., Salt Lake City, Utah). At Salt Lake City, the isotopes are transferred to another airplane for transport to the airport nearest the pharmaceutical distributor in Chicago, Illinois (Amersham Mediphysics), Boston, Massachusetts (Dupont-Merck), and St. Louis, Missouri (Mallinckrodt). The isotope product is transported from the destination airport to the pharmaceutical distributor by truck, using public roadways. Shipments of waste (i.e., liquid processing waste and solid waste, including spent target hulls and miscellaneous wastes), would go to the 200 East and West Areas for subsequent storage and disposal (Lavender and Nielsen 1997).

J.5.4 Representative Routes

Representative overland truck routes were selected for the shipments to ORR, INEEL, Hanford, and SRS. The routes were selected consistent with current routing practices and all applicable routing regulations and guidelines (40 CFR Section 397.103). However, the routes were determined for risk assessment purposes. They do not necessarily represent the actual routes that would be used to transport materials in the future. Specific routes cannot be identified in advance. The planning process for actual shipments may identify similar routes, which would have similar public risks, but could be determined to be preferable. The representative truck routes are shown in **Figure J–9**.

Route characteristics that are important to the radiological risk assessment include the total shipment distance and the population distribution along the route. The specific route selected determines both the total potentially exposed population and the expected frequency of transportation-related accidents. Route characteristics are summarized in **Table J–4**. The population densities along each route are derived from 1990 data from the U.S. Bureau of the Census (DOC 1992). Rural, suburban, and urban areas are characterized according to the following breakdown: rural population densities range from 0 to 54 persons per square kilometer (0 to 139 person per square mile); the suburban range is from 55 to 1,284 persons per square kilometer (140 to 3,326 persons per square mile); and the urban range includes all population densities greater than 1,284 persons per square kilometer (3,326 persons per square mile). The exposed population, for the purpose of route characterization and incident-free dose calculation, includes all persons living within 800 meters (0.5 mile) of each side of the road.

Alternative 2, Options 4, 5, and 6 include irradiation of the neptunium targets at a CLWR. Determining which CLWR will actually provide the irradiation services is beyond the scope of this NI PEIS. For the purpose of impact analysis, a distance and population distribution that bounds all CLWRs is given for each DOE site evaluated in this NI PEIS in Table J–3. This distance and population would also bound shipments to and from Canada, if DOE should consider the use of a CANDU reactor in the future. Alternatives 3 and 4 include irradiation at a hypothetical facility located on an unspecified DOE site. For shipments originating at or destined for the unspecified reactor or accelerator site, a distance that bounds the furthest major DOE site is used.

J.5.5 External Dose Rates

In absence of analytical information, all shipments of neptunium-237, irradiated targets and plutonium-238 are conservatively assumed to be at the regulatory limit of 10 millirems per hour at a distance of 2 meters (6.6 feet) from the outer surface of the vehicle. The unirradiated targets, shipped in the same shielded cask as the irradiated targets, are assumed to be at one-tenth the regulatory limit. Other dose rates are estimated from cask contents.

J.5.6 Health Risk Conversion Factors

The health risk conversion factors used to estimate expected latent cancer fatalities were 0.0005 and 0.0004 fatal cancer cases per person-rem for members of the public and workers, respectively (ICRP 1991).

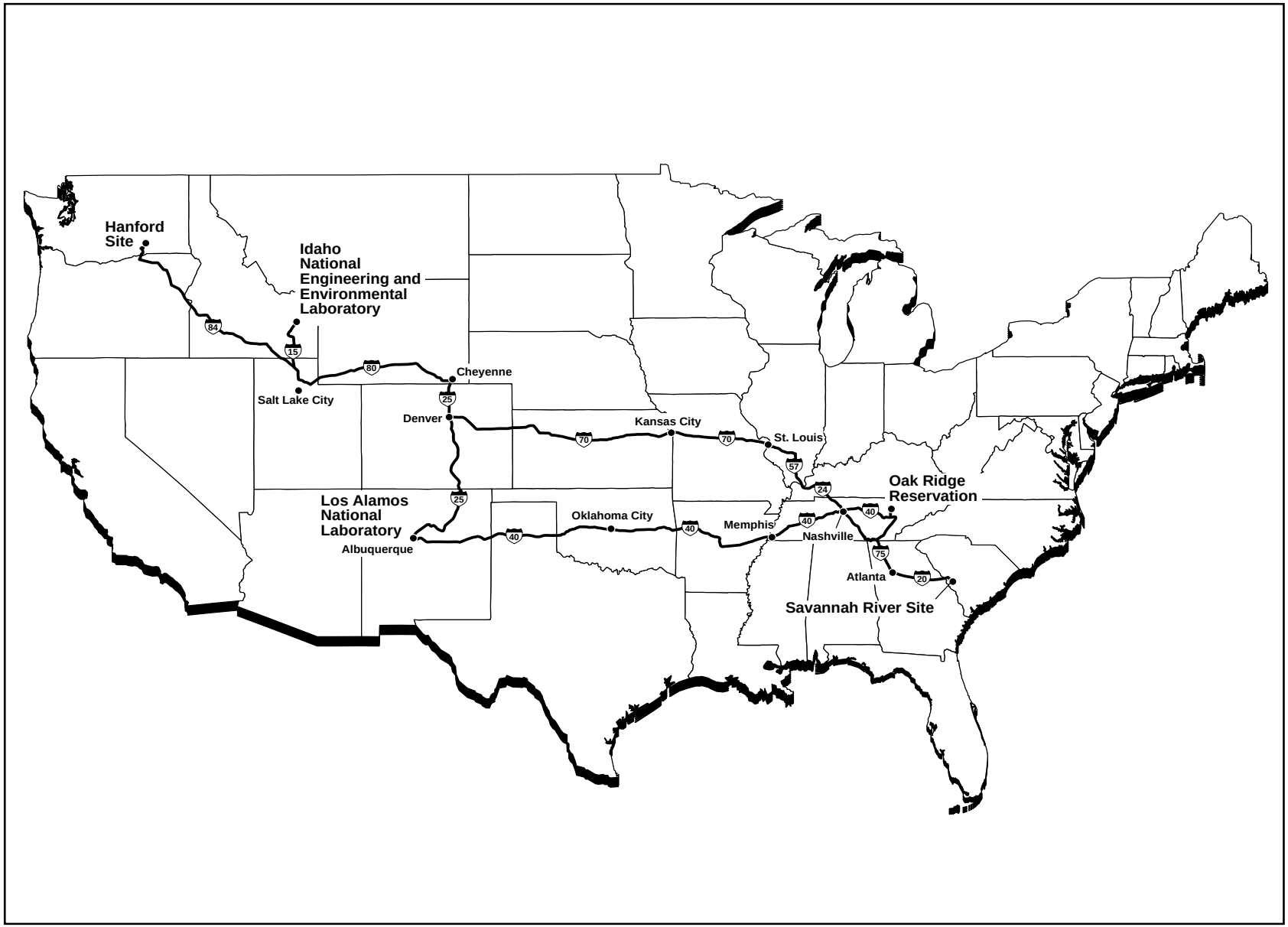


Figure J–9 Representative Overland Truck Routes

Table J–4 Potential Shipping Routes Evaluated for This NI PEIS

From	To	Distance (km)	Percentages in Zones			Population Density in Zone (1/km)			Number of Affected Persons
			Rural	Suburban	Urban	Rural	Suburban	Urban	
SRS	ORR (REDC)	604	60.8	36.0	3.2	18	334	2,195	194,424
SRS	INEEL (FDPF)	3,729	82.8	15.4	1.8	7	352	2,178	593,236
SRS	Hanford (FMEF)	4,429	84.3	14.0	1.6	7	359	2,169	642,594
ORR (REDC or HFIR)	INEEL (ATR or FDPF)	3,320	86.7	11.9	1.4	6	344	2,188	409,700
ORR (HFIR)	Hanford (FMEF)	4,020	87.7	11.0	1.3	6	355	2,175	466,713
INEEL (FDPF)	INEEL (ATR)	5	100	0.0	0.0	1	0	0	8
INEEL (ATR)	Hanford (FMEF)	1,007	92.0	7.4	0.5	6	384	1,984	70,108
ORR (REDC)	CLWR[a]	4,000	84.0	15.0	1.0	6	719	3,861	969,600
INEEL (FDPF)	CLWR[a]	4,700	84.0	15.0	1.0	6	719	3,861	1,139,280
Hanford (FMEF)	CLWR[a]	5,400	84.0	15.0	1.0	6	719	3,861	1,308,960
ORR (REDC)	Generic accelerator or reactor site[b]	4,000	84.0	15.0	1.0	6	719	3,861	969,600
INEEL (FDPF)	Generic accelerator or reactor site[b]	4,000	84.0	15.0	1.0	6	719	3,861	969,600
Hanford (FMEF)	Generic accelerator or reactor site[b]	4,500	84.0	15.0	1.0	6	719	3,861	1,090,800
Fuel fabricator	Generic reactor site	4,000	84.0	15.0	1.0	6	719	3,861	969,600
ORR (REDC)	LANL	2,383	85.6	12.5	1.9	8	340	2,171	346,554
INEEL (FDPF)	LANL	1,846	89.2	9.4	1.4	4	383	2,093	204,112
Hanford (FMEF)	LANL	2,546	90.2	8.7	1.2	4	396	2,085	258,327
ORNL	Hanford 300 Area	3,834[c]	88.2	10.7	1.1	6	342	2,088	401,048
Hanford RPL	FFTF	14	71.4	28.6		2	89		614
Hanford RPL	Pasco Airport	32	68.8	28.1	1.0	6	342	2,088	6,218
Boston Airport	Dupont-Merck	35	14.3	51.4	34.3	15	479	2,564	61,336
Chicago Airport	Amersham Mediphysics	32	21.9	25.0	53.1	8	670	2,829	85,580
St. Louis Airport	Mallinckrodt	13	7.7	46.2	46.2	2	778	2,611	32,574
Hanford RPL	Hanford 200-East Area	35	97.2	2.9		2	90		276
Hanford RPL	Hanford 200-West Area	43	90.7	9.3		2	90		724
Charleston Naval Weapons Station	FFTF	4,677	84.8	13.8	1.3	7.2	342	2,157	609,025
ORR (Y–12)	B&W Lynchburg	550	66.5	32.6	0.9	19.5	283	2,029	108,804
B&W Lynchburg	FFTF	4,516	86.1	12.6	1.3	7.4	354	2,182	573,596
Generic fuel fabricator	Hypothetical site	4,500	84.0	15.0	1.0	6.0	719	3,861	1,090,800

a. CLWR site is assumed to be the furthest operating pressurized water reactor from the processing facility.
b. Bounding distance for a new reactor or accelerator constructed on an existing DOE site.
c. Using routes other than those designated for a vehicle carrying a Highway Route Controlled Quantity of a hazardous material; used by a vehicle carrying unirradiated medical isotope targets. All other routes selected are for Highway Route Controlled Quantities (49 CFR Part 397, Subpart D).

Note: 1 kilometer = 0.62 mile; 1 square kilometer = 0.39 square mile.
Key: km, kilometer.

J.5.7 Truck Accident Rates

For the calculation of accident risks, vehicle accident and fatality rates are taken from data provided in ANL-ESD/TM-150 (Saricks and Tompkins 1999). Accident rates are generically defined as the number of accident involvements (or fatalities) in a given year per unit of travel in that same year. Therefore, the rate is a fractional value, with accident-involvement count as the numerator of the fraction and vehicular activity (total travel distance in truck-kilometers) as its denominator. Accident rates are generally determined for a multiyear period. For assessment purposes, the total number of expected accidents or fatalities is calculated by multiplying the total shipment distance for a specific case by the appropriate accident or fatality rate.

For truck transportation, the rates presented are specifically for heavy combination trucks involved in interstate commerce (Saricks and Tompkins 1999). Heavy combination trucks are rigs composed of a separable tractor unit containing the engine and one to three freight trailers connected to each other. Heavy combination trucks are typically used for radioactive waste shipments. The truck accident rates are computed for each state based on statistics compiled by the Federal Highway Administration, Office of Motor Carriers from 1994 to 1996. A fatality caused by an accident is the death of a member of the public who is killed instantly or dies within 30 days due to the injuries sustained in the accident.

The HIGHWAY code classifies highways as rural, suburban or urban, and provides the distance and population information for use in RADTRAN. These codes require accident frequency data calculated for rural, urban and suburban zones. An older report, TM-68 (Saricks and Kvitek 1994), reports accident rates for Federally Aided Interstates in urban and rural areas, and a composite accident rate for all Federally Aided Interstates. TM-150 does not provide data that can be directly used to estimate frequencies for rural, urban and suburban zones. The ratio's of accident frequencies for the zones was calculated from TM-68 data, and used with the newer TM-150 data to establish up-to-date accident frequency estimates. Since the distance traveled on non-interstate highways was very small compared to the distance traveled on interstates, and the accident rates are similar, interstate accident rates were used for all roads. TM-68 and TM-150 information is used for both the accident rate estimate for the radiological risk, and the fatal accident rate estimate for the nonradiological risk.

For SST/SGT transportation, the rates presented are specifically adjusted for the experience of the DOE Transportation Safeguards Division. Between fiscal year 1984 and fiscal year 1998, the Transportation Safeguards Division reports 0.058 accident per million kilometers (0.096 accident per million miles) (Claus and Shyr 1999). Using influence factors from SAND93-0111 (Phillips, Clauss, and Blower 1994), accident frequencies for rural, urban, and suburban driving can be estimated.

J.5.8 Container Accident Response Characteristics and Release Fractions

NUREG-0170 (NRC 1977) was used to estimate the conditional probabilities and release fractions associated with the neptunium-237, plutonium-238, and highly enriched uranium shipments. The Modal Study, an initiative taken by NRC (Fischer et al. 1987) to refine more precisely the analysis presented in NUREG-0170 (NRC 1977) for spent nuclear fuel shipping casks, was used to estimate the conditional probabilities and release fractions for target and nuclear fuel shipments. The release fractions used for the analysis of medical and industrial isotopes are based on the severity of the accident, the shipping container, and the material being shipped (Lavender and Nielsen 1997).

Whereas the NUREG-0170 analysis was primarily performed using best engineering judgments and presumptions concerning cask response, the Modal Study relies on sophisticated structural and thermal engineering analysis and a probabilistic assessment of the conditions that could be experienced in severe transportation accidents. The Modal Study results are based on representative spent nuclear fuel casks that

were assumed to have been designed, manufactured, operated, and maintained in accordance with national codes and standards. Design parameters of the representative casks were chosen to meet the minimum test criteria specified in 10 CFR Part 71. The study is believed to provide realistic, yet conservative, results for radiological releases under transport accident conditions.

In both NUREG-0170 and the Modal Study, potential accident damage to a cask is categorized according to the magnitude of the mechanical forces (impact) and thermal forces (fire) to which a cask may be subjected during an accident. Because all accidents can be described in these terms, severity is independent of the specific accident sequence. In other words, any sequence of events that results in an accident in which a cask is subjected to forces within a certain range of values, it is assigned to the accident severity region associated with that range. The accident severity scheme is designed to take into account all potential foreseeable transportation accidents, including accidents with low probability but high consequences and those with high probability but low consequences.

As discussed above, the accident consequence assessment only considers the potential impacts from the most severe transportation accidents. In terms of risk, the severity of an accident must be viewed in terms of potential radiological consequences, which are directly proportional to the fraction of the radioactive material within a cask that is released to the environment during the accident. Although regions span the entire range of mechanical and thermal accident loads, they are grouped into accident categories that can be characterized by a single set of release fractions and are, therefore, considered together in the accident consequence assessment. The accident category severity fraction is the sum of all conditional probabilities in that accident category. NUREG-0170 (NRC 1977) provides eight accident severity categories for the neptunium-237 and plutonium-238. The Modal Study (Fischer et al. 1987) provides six accident severity categories for the targets.

J.5.9 Nonradiological Risk (Vehicle-Related)

Vehicle-related health risks resulting from incident-free transport may be associated with the generation of air pollutants by transport vehicles during shipment and are independent of the radioactive nature of the shipment. The health end-point assessed under incident-free transport conditions is the excess latent mortality due to inhalation of vehicle exhaust emissions. Risk factors for pollutant inhalation in terms of latent mortality have been generated (Neuhauser and Kanipe 2000). These risks are 1×10^{-7} mortality per kilometer (1.6×10^{-7} per mile) of truck travel in urban areas. The risk factors are based on regression analyses of the effects of sulfur dioxide and particulate releases from diesel exhaust on mortality rates. Excess latent mortalities are assumed to be equivalent to latent cancer fatalities. Vehicle-related risks from incident-free transportation are calculated for each case by multiplying the total distance traveled in urban areas by the appropriate risk factor. Similar data are not available for rural and suburban areas.

Risks are summed over the entire route and over all shipments for each case. This method has been used in several environmental impact statements to calculate risks from incident-free transport. Lack of information for rural and suburban areas is an apparent data gap, although the risk factor would be much lower than for urban areas because of lower total emissions from all sources and lower population densities in rural and suburban areas.

J.5.10 Intrasite Shipment

If HFIR were selected to irradiate and REDC to process the targets, targets would be transported the short distance between REDC and HFIR in a cask that was formerly certified to Type B standards. Since the move is only about 90 meters (100 yards) on closed roads, and entirely within ORR, DOE procedures and NRC regulations do not require the use of a certified Type B cask. No incident-free risk analysis is necessary because the public would receive no measurable exposure. Similar procedures and equipment would be used

at INEEL for transfers between FDPF and ATR. Worker dose would be included in the handling analysis. No accident analysis is necessary because potential accidents during transportation are bounded in frequency and consequence by handling accidents. Once the cask is closed for the low speed transportation to the nearby building, the likelihood of any foreseeable accident that could expose the cask to conditions severe enough to fail the cask are very small.

At Hanford, the distances between facilities are somewhat larger and the roads could remain open for traffic. Therefore, DOE plans to use certified packaging. Risk analysis for unirradiated and irradiated plutonium-238 and medical and industrial isotope targets have been included. The facility locations for the new accelerator(s) and reactor have not been established, so the risk analysis used the same parameters as for Hanford.

J.6 RISK ANALYSIS RESULTS

J.6.1 Transportation Risk Analysis

Per-shipment risk factors have been calculated for the collective populations of exposed persons and for the crew for all anticipated routes and shipment configurations. The radiological doses are presented in doses per shipment for each unique route, material, and container combination. The radiological dose risk factors per shipment for incident-free transportation are presented in **Table J–5**. The per-shipment doses from medical and industrial isotopes come from HNF-1844 (Lavender and Nielsen 1997). The impacts from importing plutonium-238 were scaled from DOE/EA-0841 the *Environmental Assessment of the Impact of Russian Plutonium-238* (DOE 1993) as described in Section 4.2.1.1. Doses are calculated for the crew, off-link public (i.e., people living along the route), on-link public (i.e., pedestrians and drivers along the route), and public at rest and fueling stops (i.e., stopped cars, buses and trucks, workers, and other bystanders).

The radiological dose risk factors for transportation accidents are also presented in Table J–5. The accident risk factors are called "dose risk" because the values incorporate the spectrum of accident severity probabilities and associated consequences. Commercial vehicles have higher nonradiological and radiological accident risks because of the lower accident frequency calculated for SST/SGTs. The SST/SGTs have lower public risk estimates because they only stop in secure locations. The commercial vehicles have lower emission risk estimates because the SST/SGTs travel with escort vehicles. Crew risks are about the same.

The nonradiological risks of transporting each of the hazardous materials on the various routes are given in **Table J–6**. The risks are calculated by multiplying the previously given per-shipment factors by the number of shipments over the 35-year duration of the program. The risk estimates include the highest conceivable impacts of shipping. The total exhaust emission risks are higher if SST/SGTs are used because of the additional emission of escort vehicles. The accident risk of escort vehicles was considered in the analysis of traffic accident risk.

Table J–7 shows the risks of transportation for each alternative for the production of plutonium-238. The risks are calculated by multiplying the previously given per-shipment factors by the number of shipments over the duration of the program, and for the radiological doses, by the health risk conversion factors. The risks are summed for all material transported under each alternative. The risks shown in Table J–7 conservatively assume that the neptunium-237, mixed oxide fuel, highly enriched uranium fuel, and plutonium-238 would be shipped in an SST/SGT and that all other intersite transportation would be done in commercial vehicles. Use of SST/SGTs for other shipments would lower the radiological risk estimates. They include the risk from overland, sea, and air transportation to the vessel crews and the public. **Table J–8** was created by adding the risk associated with the production and distribution of medical and industrial isotopes (Alternatives 1, 3, and 4) to the values in Table J–7. Table J–8 shows the risks for the missions analyzed in this NI PEIS.

The accident consequence assessment is intended to provide an estimate of the maximum potential impacts posed by the most severe hypothetical transportation accidents involving a shipment of materials covered by this NI PEIS. As a practical matter, the maximum foreseeable transportation accident is defined as an accident with a frequency greater than 1×10^{-7} per year (once in 10 million years). The previously described risk assessment (RADTRAN analysis) takes into account the risk of accidents not considered in the consequence assessment. The risk of accidents with frequencies lower than 1×10^{-7} per year is included in the risk estimates shown in Table J–7.

Table J–5 Radiological Dose for Incident-Free Transportation and Accident Dose-Risk Factors (per Shipment)

From	To	Hazardous Material	Vehicle	Incident-Free Dose (person-rem)					Accident Dose-Risk[a] (person-rem)
				Crew	Public				
					Off-link	On-link	Stops	Total	
SRS	ORR	Neptunium-237	SST/SGT	0.0049	0.0087	0.0024	0.028	0.061	8.3×10^{-7}
			Truck	0.0049	0.0087	0.0024	0.10	0.14	4.0×10^{-6}
SRS	INEEL		SST/SGT	0.031	0.025	0.13	0.17	0.33	2.6×10^{-6}
			Truck	0.031	0.025	0.13	0.64	0.79	0.000012
SRS	Hanford		SST/SGT	0.036	0.027	0.15	0.21	0.38	2.9×10^{-6}
			Truck	0.036	0.027	0.15	0.76	0.94	0.00014
ORR	INEEL	Unirradiated neptunium-237 targets	SST/SGT	0.0021	0.0015	0.0097	0.013	0.025	7.5×10^{-10}
			Truck	0.0021	0.0015	0.0098	0.049	0.061	3.6×10^{-9}
ORR	Hanford		SST/SGT	0.0025	0.0017	0.012	0.016	0.030	8.1×10^{-10}
			Truck	0.0025	0.0017	0.012	0.060	0.073	4.1×10^{-9}
ORR	CLWR		SST/SGT	0.0025	0.0043	0.012	0.016	0.032	1.8×10^{-9}
			Truck	0.0025	0.0043	0.012	0.060	0.076	8.8×10^{-9}
INEEL	CLWR		SST/SGT	0.0030	0.0050	0.014	0.019	0.038	2.1×10^{-9}
			Truck	0.0030	0.0050	0.014	0.070	0.089	1.0×10^{-8}
Hanford	INEEL		SST/SGT	0.00064	0.00033	0.0028	0.004	0.007	1.3×10^{-10}
			Truck	0.00064	0.00033	0.0028	0.015	0.018	6.4×10^{-10}
Hanford	CLWR		SST/SGT	0.0034	0.0058	0.016	0.022	0.044	2.4×10^{-9}
			Truck	0.0034	0.0058	0.016	0.080	0.10	1.2×10^{-8}
ORR	Reactor		SST/SGT	0.0026	0.0043	0.012	0.016	0.032	1.8×10^{-9}
			Truck	0.0026	0.0043	0.012	0.060	0.076	8.8×10^{-9}
INEEL	Reactor		SST/SGT	0.0026	0.0043	0.012	0.016	0.032	1.8×10^{-9}
			Truck	0.0026	0.0043	0.012	0.060	0.076	8.8×10^{-9}
Hanford	Reactor		SST/SGT	0.0028	0.0048	0.013	0.018	0.036	2.0×10^{-9}
			Truck	0.0028	0.0048	0.013	0.067	0.085	9.9×10^{-9}

Table J–5 Radiological Dose for Incident-Free Transportation and Accident Dose-Risk Factors (per Shipment) (Continued)

From	To	Hazardous Material	Vehicle	Incident-Free Dose (person-rem)					Accident Dose-Risk[a] (person-rem)
				Crew	Public				
					Off-link	On-link	Stops	Total	
ORR	Accelerator	Unirradiated neptunium-237 targets	Truck	0.0026	0.0043	0.012	0.060	0.076	1.1×10^{-6}
INEEL	Accelerator		Truck	0.0026	0.0043	0.012	0.060	0.076	1.1×10^{-6}
Hanford	Accelerator		Truck	0.0043	0.0056	0.015	0.077	0.098	1.3×10^{-6}
CLWR	ORR	Irradiated neptunium-237 targets	Truck	0.041	0.049	0.14	0.68	0.87	4.3×10^{-8}
CLWR	INEEL		Truck	0.048	0.058	0.16	0.80	1.02	5.1×10^{-8}
INEEL	Hanford		Truck	0.010	0.0037	0.032	0.17	0.21	3.1×10^{-9}
CLWR	Hanford		Truck	0.055	0.066	0.18	0.02	0.27	5.9×10^{-8}
Accelerator or reactor	Hanford		Truck	0.046	0.055	0.15	0.77	0.98	1.6×10^{-7}
Accelerator or reactor	ORR		Truck	0.041	0.049	0.14	0.68	0.87	1.4×10^{-7}
Accelerator or reactor	INEEL		Truck	0.041	0.049	0.14	0.68	0.87	1.4×10^{-7}
Accelerator	ORR		Truck	0.038	0.049	0.14	0.68	0.87	7.2×10^{-4}
Accelerator	INEEL		Truck	0.038	0.049	0.14	0.68	0.87	7.2×10^{-4}
			Truck	0.043	0.056	0.15	0.77	0.98	8.1×10^{-4}
ORR	LANL	Plutonium-238	SST/SGT	0.020	0.00090	0.0055	0.0024	0.0088	0.0025
			Truck	0.020	0.00090	0.0055	0.0094	0.016	0.0012
INEEL	LANL		SST/SGT	0.015	0.00056	0.0042	0.0020	0.0067	0.0012
			Truck	0.015	0.00056	0.0042	0.0073	0.012	0.0058
Hanford	LANL		SST/SGT	0.021	0.00074	0.0057	0.0027	0.0092	0.0017
			Truck	0.021	0.00074	0.0057	0.010	0.017	0.0080
Europe	CNWS	SNR–300 Mixed oxide fuel C1	Ship	0.0027					1.3×10^{-10}
CNWS	Hanford		SST/SGT	0.012	0.0012	0.0080	0.0030	0.012	2.5×10^{-8}
Europe	CNWS	SNR–300 Mixed oxide fuel C2	Ship	0.0027					1.8×10^{-10}
CNWS	Hanford		SST/SGT	0.012	0.0012	0.0080	0.0030	0.012	3.4×10^{-8}

Table J–5 Radiological Dose for Incident-Free Transportation and Accident Dose-Risk Factors (per Shipment) (Continued)

From	To	Hazardous Material	Vehicle	Incident-Free Dose (person-rem)					Accident Dose-Risk[a] (person-rem)
				Crew	Public				
					Off-link	On-link	Stops	Total	
ORR	Fuel fabricator	Highly enriched uranium fuel	SST/SGT	0.00095	0.000035	0.00026	0.000039	0.00033	3.3×10^{-10}
Fuel fabricator	Hanford	Highly enriched uranium fuel	SST/SGT	0.012	0.0011	0.0077	0.0030	0.012	1.6×10^{-13}
Fuel fabricator	Reactor site	Low-enriched uranium fuel	Truck	0.0029	0.0049	0.00039	0.067	0.072	4.5×10^{-10}

a. Dose-risk factor = dose (due to accident) × accident rate per distance traveled × distance traveled.

Key: CNWS, Charleston Naval Weapons Station; SST/SGT, safe, secure trailer/SafeGuards Transport.

Table J–6 Nonradiological Risk Factors per Shipment

Routes		Exhaust Emissions (latent cancer fatalities)		Accidents (fatalities)	
From	**To**	**Truck**	**SST/SGT**	**Truck**	**SST/SGT**
SRS	ORR (REDC)	3.9×10^{-6}	5.0×10^{-6}	0.00023	3.6×10^{-6}
SRS	INEEL (FDPF)	0.000013	0.000017	0.00011	0.000015
SRS	Hanford (FMEF)	0.000014	0.000018	0.00013	0.000018
ORR (REDC or HFIR)	INEEL (ATR)	9.3×10^{-6}	0.000012	0.000094	0.000013
ORR (HFIR)	Hanford (FMEF)	0.00001	0.000014	0.00011	0.000015
INEEL (FDPF)	INEEL (ATR)	0	0	1.1×10^{-7}	1.3×10^{-8}
INEEL (ATR)	Hanford (FMEF)	1.0×10^{-6}	1.3×10^{-6}	0.000026	3.5×10^{-6}
ORR (REDC)	CLWR	8.0×10^{-6}	0.00001	0.00012	0.000016
INEEL (FDPF)	CLWR	9.4×10^{-6}	0.000012	0.00014	0.000019
Hanford (FMEF)	CLWR	0.000011	0.000014	0.00016	0.000022
ORR (REDC)	Generic accelerator or reactor site[a]	8.0×10^{-6}	1.0×10^{-5}	1.2×10^{-4}	1.6×10^{-5}
INEEL (FDPF)	Generic accelerator or reactor site[a]	8.0×10^{-6}	1.0×10^{-5}	1.2×10^{-4}	1.6×10^{-5}
Hanford (FMEF)	Generic accelerator or reactor site[a]	9.0×10^{-6}	1.2×10^{-5}	1.3×10^{-4}	1.8×10^{-5}
Fuel fabricator	Generic reactor site	9.0×10^{-6}	1.2×10^{-5}	1.3×10^{-4}	1.8×10^{-5}
ORR REDC	LANL	9.1×10^{-6}	0.000012	0.000068	9.3×10^{-6}
INEEL FDPF	LANL	5.2×10^{-6}	6.7×10^{-6}	5.0×10^{-6}	6.7×10^{-6}
Hanford FMEF	LANL	6.1×10^{-6}	7.9×10^{-6}	6.8×10^{-6}	9.1×10^{-6}
ORNL	Hanford 300 Area	8.5×10^{-6}	1.1×10^{-5}	1.06×10^{-4}	1.4×10^{-5}
Hanford RPL	FFTF			4.9×10^{-7}	7.4×10^{-8}
Hanford RPL	Pasco Airport	6.4×10^{-8}	NA	1.1×10^{-6}	NA
Boston Airport	Dupont-Merck	2.3×10^{-6}	NA	1.6×10^{-6}	NA
Chicago Airport	Amersham Mediphysics	3.4×10^{-6}	NA	1.2×10^{-6}	NA
St. Louis Airport	Mallinckrodt	1.2×10^{-6}	NA	6.0×10^{-7}	NA
Charleston Naval Weapons Station	FFTF	1.2×10^{-5}	1.6×10^{-5}	1.4×10^{-4}	1.9×10^{-5}
Y–12	B&W fuel	9.9×10^{-7}	1.3×10^{-6}	2.0×10^{-5}	3.1×10^{-6}
B&W fuel	FFTF	1.2×10^{-5}	1.5×10^{-5}	1.3×10^{-4}	1.8×10^{-5}
Fuel fabrication	Site	9.0×10^{-6}	1.2×10^{-5}	1.3×10^{-4}	1.8×10^{-5}

a. Bounding distance for a new accelerator or reactor constructed on an existing DOE site.

Key: SST/SGT, safe, secure trailer/SafeGuards Transport.

Table J–7 Risks of Transporting the Hazardous Materials for the Production of Plutonium-238

Alternative		Combinations (Target Fabrication and Processing/Reactors)	Shipments[a]	Distance Traveled[b] (km)	Incident-Free Risk[c]			Accident Risk[c]	
					Radiological		Nonradiological		
#	Option				Crew	Public	Emission	Traffic	Radiological
	No Action Option 1	—	35	113,750	0.0046	0.0099	0.00047	0.014	4.4×10^{-4}
	No Action Option 2	Storage of neptunium at ORR	59	128,234	0.0047	0.011	0.00059	0.014	4.4×10^{-4}
	No Action Option 3	Storage of neptunium at INEEL	59	203,239	0.0049	0.014	0.0009	0.014	4.4×10^{-4}
	No Action Option 4	Storage of neptunium at Hanford	59	220,041	0.0050	0.014	0.0009	0.014	4.4×10^{-4}
1	Restart FFTF Option 1	Production[d] at ORR and irradiation at FFTF with MOX and HEU	829	3,255,586	0.0064	0.15	0.009	0.073	4.4×10^{-5}
	Restart FFTF Option 2	Production at INEEL and irradiation at FFTF with MOX and HEU	829	1,413,840	0.0025	0.041	0.003	0.020	2.1×10^{-5}
	Restart FFTF Option 3	Production at Hanford and irradiation at FFTF with MOX and HEU	829	829,300	0.0013	0.006	0.003	0.0032	3.0×10^{-5}
	Restart FFTF Option 4	Production at ORR and irradiation at FFTF with HEU	814	3,159,278	0.0063	0.15	0.009	0.073	4.4×10^{-5}
	Restart FFTF Option 5	Production at INEEL and irradiation at FFTF with HEU	814	1,317,532	0.0024	0.04	0.0031	0.019	2.1×10^{-5}
	Restart FFTF Option 6	Production at Hanford and irradiation at FFTF with HEU	814	732,993	0.0012	0.005	0.0025	0.0028	3.0×10^{-5}
2	Existing Facility Option 1	Production at ORR and irradiation at INEEL	689	2,189,503	0.0048	0.12	0.0064	0.059	4.4×10^{-5}
	Existing Facility Option 2	Production and irradiation at INEEL	59	154,095	0.00050	0.0040	0.00065	0.00060	2.1×10^{-5}
	Existing Facility Option 3	Production at Hanford and irradiation at INEEL	689	830,060	0.0020	0.040	0.0014	0.017	3.0×10^{-5}
	Existing Facility Option 4	Production at ORR and irradiation at a CLWR	689	2,617,903	0.0057	0.15	0.0056	0.074	4.4×10^{-5}
	Existing Facility Option 5	Production at INEEL and irradiation at a CLWR	689	3,115,095	0.0069	0.18	0.0066	0.088	2.1×10^{-5}
	Existing Facility Option 6	Production at Hanford and irradiation at a CLWR	689	3,597,398	0.0080	0.21	0.0075	0.10	3.0×10^{-5}

Table J–7 Risks of Transporting the Hazardous Materials for the Production of Plutonium-238 (Continued)

Alternative		Combinations (Target Fabrication and Processing/Reactors)	Shipments[a]	Distance Traveled[b] (km)	Incident-Free Risk[c]			Accident Risk[c]	
					Radiological		Nonradiological		
#	Option				Crew	Public	Emission	Traffic	Radiological
	Existing Facility Option 7	Production at ORR and irradiation at ORR and INEEL	563	1,771,183	0.0039	0.096	0.0052	0.048	4.4×10^{-5}
	Existing Facility Option 8	Production at INEEL and irradiation at ORR and INEEL	311	990,735	0.0024	0.052	0.0030	0.024	4.4×10^{-5}
	Existing Facility Option 9	Production at Hanford and irradiation at ORR and INEEL	689	1,589,235	0.0036	0.084	0.0037	0.039	3.0×10^{-5}
3	New Accelerator Option 1	Production at ORR and irradiation in accelerator[e]	269	937,903	0.0020	0.050	0.0022	0.025	8.1×10^{-5}
	New Accelerator Option 2	Production at INEEL and irradiation in accelerator[e]	269	994,095	0.0023	0.054	0.0023	0.025	8.1×10^{-5}
	New Accelerator Option 3	Production at Hanford and irradiation in accelerator[e]	269	1,140,398	0.0026	0.061	0.0026	0.029	7.2×10^{-5}
4	New Reactor Option 1	Production at ORR and irradiation in a new research reactor	709	2,707,903	0.0055	0.15	0.0058	0.077	4.8×10^{-5}
	New Reactor Option 2	Production at INEEL and irradiation in a new research reactor	709	2,764,095	0.0057	0.15	0.0059	0.077	4.8×10^{-5}
	New Reactor Option 3	Production at Hanford and irradiation in a new research reactor	709	3,120,398	0.0064	0.17	0.0066	0.082	3.5×10^{-5}
5	Deactivate FFTF	—	—	~0	~0	~0	~0	~0	~0

a. "Shipments" means the number of transportation legs. For example, a package that is loaded onto a truck, driven to an airport, loaded onto an aircraft, flown to another airport, loaded onto a truck, and shipped to a final destination would count as three shipments (two by truck, one by air).
b. Distance traveled by trucks carrying radiological materials. Nonradiological impacts used two-way transportation.
c. All risks are expressed as number of latent cancer fatalities, except for the Accident-Traffic column, which lists number of accident fatalities.
d. Production means storage, target fabrication, and processing.
e. These are the transportation impacts for the high-energy accelerator.

Key: HEU, highly enriched uranium fuel; km, kilometers; MOX, mixed oxide fuel.
Source: Calculated results.

Table J–8 Risks of Transporting the Hazardous Materials for All Research, Development, and Isotope Production Missions

Alternative		Combinations (Target Fabrication and Processing/Reactors)	Shipments[a]	Distance Traveled[b] (km)	Incident-Free Risk[c]			Accident Risk[c]	
					Radiological		Nonradiological		
#	Option				Crew	Public	Emission	Traffic	Radiological
	No Action Option 1	—	35	113,750	0.0046	0.0099	0.00047	0.014	4.4×10^{-4}
	No Action Option 2	Storage of neptunium at ORR	59	128,234	0.0047	0.011	0.00059	0.014	4.4×10^{-4}
	No Action Option 3	Storage of neptunium at INEEL	59	203,239	0.0049	0.014	0.0009	0.014	4.4×10^{-4}
	No Action Option 4	Storage of neptunium at Hanford	59	220,041	0.0050	0.014	0.0009	0.014	4.4×10^{-4}
1	Restart FFTF Option 1	Production[d] at ORR and irradiation at FFTF with MOX and HEU	37,579	8,020,696	0.012	0.15	0.0030	0.19	0.53
	Restart FFTF Option 2	Production at INEEL and irradiation at FFTF with MOX and HEU	37,579	6,178,950	0.008	0.044	0.024	0.13	0.53
	Restart FFTF Option 3	Production at Hanford and irradiation at FFTF with MOX and HEU	37,579	5,594,410	0.0072	0.009	0.023	0.12	0.53
	Restart FFTF Option 4	Production at ORR and irradiation at FFTF with HEU	37,564	7,924,388	0.012	0.15	0.029	0.18	0.53
	Restart FFTF Option 5	Production at INEEL and irradiation at FFTF with HEU	37,564	6,082,642	0.008	0.044	0.023	0.13	0.53
	Restart FFTF Option 6	Production at Hanford and irradiation at FFTF with HEU	37,564	5,498,103	0.0071	0.009	0.023	0.11	0.53
2	Existing Facility Option 1	Production at ORR and irradiation at INEEL	689	2,189,503	0.0048	0.12	0.0064	0.059	4.4×10^{-5}
	Existing Facility Option 2	Production and irradiation at INEEL	59	154,095	0.0005	0.004	0.0007	0.0006	2.1×10^{-5}
	Existing Facility Option 3	Production at Hanford and irradiation at INEEL	689	830,060	0.0020	0.040	0.0014	0.017	3.0×10^{-5}
	Existing Facility Option 4	Production at ORR and irradiation at a CLWR	689	2,617,903	0.0057	0.15	0.0056	0.074	4.4×10^{-5}
	Existing Facility Option 5	Production at INEEL and irradiation at a CLWR	689	3,115,095	0.0069	0.18	0.0066	0.088	2.1×10^{-5}
	Existing Facility Option 6	Production at Hanford and irradiation at a CLWR	689	3,597,398	0.0080	0.21	0.0075	0.101	3.0×10^{-5}

Table J–8 Risks of Transporting the Hazardous Materials for All Research, Development, and Isotope Production Missions (Continued)

Alternative		Combinations (Target Fabrication and Processing/Reactors)	Shipments[a]	Distance Traveled[b] (km)	Incident-Free Risk[c]			Accident Risk[c]	
					Radiological		Nonradiological		
#	Option				Crew	Public	Emission	Traffic	Radiological
	Existing Facility Option 7	Production at ORR and irradiation at ORR and INEEL	563	1,771,183	0.0039	0.096	0.0052	0.048	4.4×10^{-5}
	Existing Facility Option 8	Production at INEEL and irradiation at ORR and INEEL	311	990,735	0.0024	0.052	0.0030	0.024	4.4×10^{-5}
	Existing Facility Option 9	Production at Hanford and irradiation at ORR and INEEL	689	1,589,235	0.0036	0.084	0.0037	0.039	3.0×10^{-5}
3	New Accelerator Option 1	Production at ORR and irradiation in accelerator	37,019	5,703,013	0.0080	0.054	0.023	0.14	0.53
	New Accelerator Option 2	Production at INEEL and irradiation in accelerator	37,019	5,759,205	0.0082	0.057	0.023	0.14	0.53
	New Accelerator Option 3	Production at Hanford and irradiation in accelerator	37,019	5,905,508	0.009	0.065	0.023	0.14	0.53
4	New Reactor Option 1	Production at ORR and irradiation in a new research reactor	37,459	7,473,013	0.011	0.15	0.026	0.19	0.53
	New Reactor Option 2	Production at INEEL and irradiation in a new research reactor	37,459	7,529,205	0.012	0.16	0.026	0.19	0.53
	New Reactor Option 3	Production at Hanford and irradiation in a new research reactor	37,459	7,885,508	0.012	0.18	0.027	0.19	0.53
5	Deactivate FFTF	—	—	~0	~0	~0	~0	~0	~0

a. "Shipments" means the number of transportation legs. For example, a package that is loaded onto a truck, driven to an airport, flown to another airport, loaded onto a truck and shipped to a final destination would count as three shipments (two by truck, one by air).
b. Distance traveled by trucks carrying radiological materials. Nonradiological impacts used two-way transportation.
c. All risks are expressed as number of latent cancer fatalities, except for the Accident-Traffic column, which lists number of accident fatalities.
d. Production means storage, target fabrication, and processing.

Key: HEU, highly enriched uranium fuel; km, kilometers; MOX, mixed oxide fuel.

Source: Calculated results.

Accidents involving neptunium-237, mixed oxide fuel, irradiated targets, and plutonium-238 were evaluated in the consequence assessment. Accidents involving unirradiated targets were not evaluated because they occur at the same frequency, but clearly have lower consequences than accidents involving irradiated targets. SST/SGT accidents with higher frequencies than 1×10^{-7} per year did not release any neptunium-237 or plutonium-238 to the environment because the temperature and mechanical stresses predicted for accidents in this frequency range are within the design basis of the packages. The maximum foreseeable offsite transportation accident involves a shipment of irradiated plutonium-238 targets under neutral (average) weather conditions. The accident has a probability of occurring about once every 10 million years for the alternatives that involve shipping radioactive targets from one DOE facility to another. The accident could result in a dose of 0.61 person-rem to the public with an associated 3.1×10^{-4} latent cancer fatalities and 2.6 millirem to a hypothetical maximally exposed individual 30 meters (about 100 feet) from the vehicle. This results in a latent fatal cancer risk of 1.3×10^{-6}. No immediate fatalities from radiation would be expected. This accident would fall into Severity Category V (Fischer et al. 1987). In this hypothetical accident, the impact would cause the cask to fail, and the deformation of the cask would be assumed to fail a portion of the target material. In the event of a fire, it would not be hot enough or would not last long enough to damage the targets. To incur this level of damage, the cask would have to collide with an immovable object at a speed of greater than 88.5 kilometers (55 miles) per hour. The probability of an accident with a more energetic collision or fire and higher consequences is lower.

For alternatives and options in which irradiated targets are not shipped offsite, but mixed fuel is received at an east coast port, the maximum forseeable offsite transportation accident is a shipment of mixed oxide fuel. This Category V accident in a suburban population zone could result in a dose of 0.40 person-rem to the public with an associated 2.0×10^{-4} latent cancer fatality, and 3.3 millirem to the hypothetical maximally exposed individual. No fatalities would be expected as a result of the radiation exposure.

The risks to various exposed individuals under incident-free transportation conditions have been estimated for hypothetical exposure scenarios. The estimated doses to inspectors and the public are presented in **Table J–9** on a per-event basis (person-rem per event). Note that the potential exists for larger individual exposures if multiple exposure events occur. For example, the dose to a person stuck in traffic next to a shipment for 30 minutes is calculated to be 11 millirem. If the exposure duration were longer, the dose would rise proportionally. In addition, a person working at a truck service station could receive a dose if trucks were to use the same stops repeatedly. The dose to a person fueling a truck could be as much as 1 millirem. Administrative controls could be instituted to control the location and duration of truck stops if multiple exposures were to happen routinely.

Table J–9 Estimated Dose to Exposed Individuals During Incident-Free Transportation Conditions

Receptor		Dose to Maximally Exposed Individual[a]
Workers	Crew member (truck driver)	0.1 rem per year[b]
	Inspector	0.0029 rem per event
Public	Resident	4.0×10^{-7} rem per event
	Person in traffic congestion	0.011 rem per event
	Person at service station	0.001 rem per event

a. Doses are calculated assuming that the shipment external dose rate is equal to the maximum expected dose of 10 millirem per hour at 2 meters (6.6 feet) from the package.
b. This is a dose limit for a nonradiation worker (10 CFR Part 20). The dose to the truck driver could exceed this limit in the absence of administrative controls.

The cumulative dose to a resident was calculated assuming all shipments passed the resident's home. The cumulative doses assume that the resident is present for every shipment and there is no shielding between the

package and the receptor at a distance of 30 meters (98 feet) from the route. If all the material were to be shipped via this route, the maximum dose to this resident would be less than 0.1 millirem.

The estimated dose to transportation crew members is presented for a commercial crew who would be limited to 0.1 rem per year by 10 CFR Part 20. Drivers of SST/SGTs and some commercial trucks are trained as radiological workers. Allowed exposure limits vary. The exposure is limited to 2 millirem per hour in a "normally occupied space," in accordance with 10 CFR Section 71.47.

J.6.2 Marine Transport Risk Analysis for Mixed Oxide Fuel

The potential impacts of marine transport of mixed oxide fuel were considered in two ways, incident-free and accident impacts. Impact analysis includes the impacts on the global commons (i.e., portions of the ocean not within the territorial boundary of any nation) in accordance with Executive Order 12114 (44 FR 1957), the impacts approaching and docking at the port, and the impacts of unloading the mixed oxide package at the port.

The incident-free impacts would be those that occur simply due to the marine shipping of the mixed oxide fuel, assuming there are no accidents. The ships crew and dock crew would be affected in this case. The previously described RADTRAN 5 code was used to analyze the dose to the ships crew for transportation from Europe to the U.S. east coast. The accident impacts for the egress into the Charleston Naval Weapons Station were also modeled using the RADTRAN 5 code and are displayed in Table J–5.

The dose to the ships crew and the dockside personnel that would result from off loading the mixed oxide fuel packages was taken directly from the *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996). The results are included in Table J–5 and are also in the by-alternative risk calculations shown in Table J–7. Exposure to handlers, inspectors, crane operators, and observers are included.

The *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996) analyzed the shipment of spent nuclear fuel, and much of the analysis of shipping mixed oxide fuel can be taken from that document. The *Foreign Research Reactor Spent Nuclear Fuel EIS* is useful for both an absolute assessment of impacts and a relative assessment of impacts of using various ports. This NI PEIS analysis will show that the risk of shipping mixed oxide fuel is significantly less than the risk of shipping spent nuclear fuel by comparing the overland risk assessment of the two fuels. These risk assessments were both carried out using the same systematic approach to the analysis and using the RADTRAN series of codes.

Table J–10 shows the per-shipment risk estimates performed for a mixed oxide fuel shipment from Charleston Naval Weapons Station to Hanford and the comparable estimates for a shipment of spent nuclear fuel from Charleston Naval Weapons Station and 10 other ports to Hanford. The dock and channel accident risk is from Appendix D and Attachment 2 to the *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996). It is based on direct shipment of BR–2 spent nuclear fuel to the ports and includes hypothetical accidents at a point in the channel near population centers and at the dock. For example, the channel accident for the Charleston, South Carolina, area was performed for an accident at commercial anchorage area D, which is near the city of Charleston. The remaining columns in Table J–10 are from RADTRAN analysis of overland transportation and is the same information provided in Table J–5.

Comparing the overland transportation risks of shipping mixed oxide fuel and spent nuclear fuel along the same route from Charleston, South Carolina, to Hanford indicates the relative risks of the two materials. The crew risk is about a factor of 20 higher for the spent nuclear fuel than for the mixed oxide fuel because the dose rate from the spent fuel package is estimated to be at least 20 times higher. Public risk is about 50 times higher because of this dose rate difference. Also, the mixed oxide fuel would be carried in SST/SGTs, which would

Table J–10 Per-Shipment Risk Estimates from Military Seaports to the Hanford Site

	Dock and Channel Accident Risk (LCF)	Incident-Free Risk			Accident Risk	
		Radiological (Person - Rem)		Nonradiological (Fatalities)		Radiological (person-rem)
		Crew	Public	Emission	Traffic	
Eastern ports						
Charleston Naval Weapons Station, South Carolina (MOX)	$<1\times10^{-12}$	0.012	0.012	1.6×10^{-5}	1.9×10^{-5}	3.4×10^{-8}
Charleston Naval Weapons Station, South Carolina	1.3×10^{-9}	0.25	0.64	1.1×10^{-5}	0.00020	0.00015
Military Ocean Terminal Sunny Point, North Carolina	6.2×10^{-10}	0.25	0.64	1.2×10^{-5}	0.00018	0.00014
Mayport, Florida	1.5×10^{-9}	0.26	0.67	1.3×10^{-5}	0.00017	0.00017
Kings Bay, Georgia	1.2×10^{-9}	0.25	0.65	1.2×10^{-5}	0.00017	0.00015
Pensacola, Florida	1.2×10^{-9}	0.24	0.62	1.2×10^{-5}	0.00016	9.7×10^{-5}
Yorktown, Virginia	1.6×10^{-9}	0.25	0.65	1.2×10^{-5}	0.00016	0.00013
Hampton Roads, Virginia	2.1×10^{-9}	0.26	0.67	1.6×10^{-5}	0.00018	0.00014
Western ports						
Military Ocean Terminal Bay Area, California	7.1×10^{-9}	0.081	0.20	7.8×10^{-6}	4.8×10^{-5}	4.5×10^{-5}
Bremerton, Washington	3.1×10^{-9}	0.030	0.068	4.2×10^{-6}	1.1×10^{-5}	1.2×10^{-5}
Everett, Washington	3.4×10^{-9}	0.026	0.060	3.6×10^{-6}	1.0×10^{-5}	1.2×10^{-5}
Port Hueneme, California	6.0×10^{-9}	0.11	0.28	1.2×10^{-5}	0.00077	5.6×10^{-5}
Port Townsend, Washington	Not analyzed	0.035	0.080	3.3×10^{-5}	6.2×10^{-6}	1.4×10^{-5}

Note: All except the Charleston Naval Weapons Station, South Carolina (MOX) are for spent nuclear fuel shipments from DOE 1996. Charleston Naval Weapons Station, South Carolina (MOX) is for a shipment of mixed oxide fuel from the transportation analysis of this NI PEIS, and the dock and channel accident risk for mixed oxide fuel is estimated from DOE 1996 results.
Key: <, less than; LCF, latent cancer fatality, MOX, mixed oxide fuel.

not be expected to expose the public to as much radiation as commercial trucks. In the risk analysis, commercial trucks carrying spent nuclear fuel are assumed to stop for food, fuel, and rest in the same manner as typical long distance trucking practices. However, SST/SGTs have specific procedures to ensure fueling is performed in a safe and secure manner, and routine rest and inspection stops are done in secure areas. Truck emissions are estimated to be the same for SST/SGTs and commercial trucks, but the emission risk for the mixed oxide fuel is higher because the SST/SGTs travel with escorts. SST/SGT accident frequencies are about a factor of 20 lower than truck accident frequencies, but the risk estimate for nonradiological accidents is only a factor of 10 lower because of the increased accident risk associated with SST/SGT escort vehicles.

The overland transportation radiological risk is a factor of one million lower for mixed oxide fuel than for spent nuclear fuel. Part of this difference is because the accident frequency is about 20 times lower for SST/SGTs. The risk estimated in Table J–10 were calculated by multiplying frequencies time consequences. Since the risk is about one million times lower and the frequency is about 20 times lower, the consequences of the same spectrum of accidents for the mixed oxide fuel packages is 50,000 times lower than for spent nuclear fuel packages (20×50,000 = 1,000,000). Based on the risk estimate of 1.3×10^{-9} latent cancer fatality per-shipment risk estimate for the foreign research reactor spent nuclear fuel package, the risk of shipping a mixed oxide fuel package into the port of Charleston Naval Weapons Station would be about 2.6×10^{-14} (1.3×10^{-9} divided by 50,000 = 2.6×10^{-14}).

Table J–10 shows the range of impacts calculated for the use of military ports for spent nuclear fuel, which can be used to bound the range of impacts for mixed oxide fuel. The overland transporation impacts of using

the Charleston Naval Weapons Station are representative (i.e., about the same) as for any of the east coast ports and were used in the presentation of transportation impacts in Tables J–7 and J–8. Based on the estimate for the Charleston Naval Weapons Station, the dock and channel accident risk of bringing a mixed oxide fuel package into any of the east coast military ports listed in Table J–10 would be less than 10^{-13} latent cancer fatality. The incident-free and accident risks would be about the same as Charleston Naval Weapons Station for any eastern port. The dock and channel accident risk for western ports would be higher than those for eastern ports because of the higher local populations, but all would be less than 10^{-12} latent cancer fatality per shipment. The overland transportation incident-free and accident risk for western ports ranges from about one-tenth to about one-half of those for eastern ports. The overland transportation accident risk for any east coast port is estimated to be less than 2×10^{-11} latent cancer fatality per shipment. The lower risk for western ports is caused by the reduced distance to Hanford.

The *Foreign Research Reactor Spent Nuclear Fuel EIS* (DOE 1996) evaluated the risks of damaged and undamaged casks sinking into coastal and deep ocean waters. The analysis included probabilities of recovery, and conservatively assumed failure of the cask in all accidents in greater than 200 meters of water depth. All program risks were less than 10^{-7} rem per year to the peak individual. Since mixed oxide fuel accident consequences are much lower and this NI PEIS is proposing fewer shipments than the *Foreign Research Reactor Spent Nuclear Fuel EIS*, all risks would be less than 10^{-10} rem per year for mixed oxide fuel alternatives to the maximally exposed individual. The *Foreign Research Reactor Spent Nuclear Fuel EIS* concluded that following a hypothetical severe accident, radioactive particles dispersed over the ocean would not be in large enough amounts to have a measurable impact on the environment. The same conclusion would be appropriate for this NI PEIS since the casks would contain considerably fewer curies of radioactive material.

J.7 CONCLUSIONS

The transportation requirements for the alternatives of this NI PEIS have been analyzed, and the following conclusions have been reached:

- It is unlikely that the transportation of radioactive materials will cause an additional fatality as a result of either incident-free transportation or associated with postulated transportation accidents.

- The highest risk estimate for any transportation activity is for the air transport of medical and industrial isotopes. Since the amount and nature of isotopes to be produced is uncertain, this analysis is necessarily conservative. However, in order for an isotope production facility to be successful, it is likely that large amounts of isotopes would be transported to locations throughout the country. All isotopes were assumed to be transported by air for the purpose of analysis.

- Options in which the processing and fabrication facility and the irradiation facility are colocated, such as Alternative 1, Options 3 and 6 and Alternative 2, Options 2, 7, and 8, have lower transportation risks than other alternatives.

- Options in which the irradiation facility is at an unspecified site, such as Alternative 2, Options 4, 5, and 6, and Alternatives 3 and 4, appear to have higher risks than other options. However, if an actual radiation facility were sited nearby the processing and fabrication facility, the risk estimates would be considerably lower.

- The overland transportation impacts are somewhat higher for mixed oxide fuel acceptance at east coast ports rather than a west coast port. The sea transportation impacts are much lower than overland transportation risks for shipment to either coast. The sea transportation impacts for western ports are

double those for eastern ports. Use of the Panama Canal for shipment to western ports poses safeguards and security concerns.

J.8 UNCERTAINTY AND CONSERVATISM IN ESTIMATED IMPACTS

The sequence of analyses performed to generate the estimates of radiological risk for transportation includes: (1) determining the inventory and characteristics, (2) estimating shipment requirements, (3) determining route characteristics, (4) calculating radiation doses to exposed individuals (including estimation of environmental transport and radionuclides uptake), and (5) estimating health effects. Uncertainties are associated with each of these steps. Uncertainties exist in the way that the physical systems being analyzed are represented by the computational models; in the data required to exercise the models due to measurement errors, sampling errors, natural variability, or unknown simply caused by the future nature of the action being analyzed; and in the calculations themselves (e.g., approximate algorithms used by the computer). In principle, one can estimate the uncertainty associated with each input or computational source and predict the resultant uncertainty in each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final, or absolute, result. However, conducting such a full-scale quantitative uncertainty analysis is often impractical and sometimes impossible, especially for actions to be initiated at an unspecified time in the future. Instead, the risk analysis is designed to ensure, through uniform and judicious selection of scenarios, models, and input parameters, that relative comparisons of risk among the various alternatives are meaningful. In the transportation risk assessment, this design is accomplished by uniformly applying common input parameters and assumptions to each alternative. Although considerable uncertainty is inherent in the absolute magnitude of the transportation risk for each alternative, much less uncertainty is associated with the relative differences among the alternatives in a given measure of risk.

In the following sections, areas of uncertainty are discussed for the assessment steps listed above. Special emphasis is placed on identifying whether the uncertainties affect relative or absolute measures of risk. The degree of reality conservatism of the assumption is addressed. Where practical, the parameters that most significantly affect the risk assessment results are identified.

J.8.1 Uncertainties and Conservatism in Neptunium-237 and Plutonium-238 Inventory and Characterization

The inventories and the physical and radiological characteristics are important input parameters to the transportation risk assessment. The potential amount of transportation for any alternative is determined primarily by the projected dimensions of package contents, strength of the radiation field, heat that must be dissipated, and assumptions concerning shipment capacities. The physical and radiological characteristics are important in determining the amount of material released during accidents and the subsequent doses to exposed individuals through multiple environmental exposure pathways.

Uncertainties in the inventory and characterization will be reflected to some degree in the transportation risk results. If the inventory is overestimated (or underestimated), the resulting transportation risk estimates also will be overestimated (or underestimated) by roughly the same factor. However, the same inventory estimates are used to analyze the transportation impacts of each alternative of this NI PEIS. Therefore, for comparative purposes, the observed differences in transportation risks among the alternatives, as given in Table J–6, are believed to represent unbiased, reasonably accurate estimates from current information in terms of relative risk comparisons.

If DOE should enter into the final design and implementation phase of the project, the amount of neptunium and plutonium in the targets could change. The incident-free risk estimate would not change, unless the number of shipments changes, because the maximum regulatory limit dose rate was used.

J.8.2 Uncertainties in Containers, Shipment Capacities, and Number of Shipments

The amount of transportation required for each alternative is based in part on assumptions concerning the packaging characteristics and shipment capacities for commercial trucks and SST/SGTs. Representative shipment capacities have been defined for assessment purposes based on probable future shipment capacities. In reality, the actual shipment capacities may differ from the predicted capacities such that the projected number of shipments and, consequently, the total transportation risk would change. However, although the predicted transportation risks would increase or decrease accordingly, the relative differences in risks among alternatives would remain about the same.

J.8.3 Uncertainties in Route Determination

Representative routes have been determined between all origin and destination sites considered in this NI PEIS. The routes have been determined consistent with current guidelines, regulations, and practices, but may not be the actual routes that would be used in the future. In reality, the actual routes could differ from the representative ones in terms of distances and total population along the routes. Moreover, since materials could be transported over an extended period of time starting at some time in the future, the highway infrastructures and the demographics along routes could change. These effects have not been accounted for in the transportation assessment; however, it is not anticipated that these changes would significantly affect relative comparisons of risk among the alternatives considered in this NI PEIS. Specific routes cannot be identified in advance because the routes are classified to protect national security interests.

J.8.4 Uncertainties and Conservatism in the Calculation of Radiation Doses

The models used to calculate radiation doses from transportation activities introduce a further uncertainty in the risk assessment process. It is generally difficult to estimate the accuracy or absolute uncertainty of the risk assessment results. The accuracy of the calculated results is closely related to the limitations of the computational models and to the uncertainties in each of the input parameters that the model requires. The single greatest limitation facing users of RADTRAN 5, or any computer code of this type, is the scarcity of data for certain input parameters.

Uncertainties associated with the computational models are minimized by using state-of-the-art computer codes that have undergone extensive review. Because there are numerous uncertainties that are recognized, but difficult to quantify, assumptions are made at each step of the risk assessment process that are intended to produce conservative results (i.e., overestimate the calculated dose and radiological risk). Because parameters and assumptions are applied to all alternatives, this model bias is not expected to affect the meaningfulness of relative comparisons of risk; however, the results may not represent risks in an absolute sense.

To understand the most important uncertainties and conservatism in the transportation risk assessment, the results for all cases were examined to identify the largest contributors to the collective population risk.

Postaccident mitigative actions are not considered for dispersal accidents. For severe accidents involving the release and dispersal of radioactive materials in the environment, no postaccident mitigative actions, such as interdiction of crops or evacuation of the accident vicinity, have been considered in this risk assessment. In reality, mitigative actions would take place following an accident in accordance with EPA radiation protection guides for nuclear incidents (EPA 1992). The effects of mitigative actions on population accident doses are

highly dependent upon the severity, location, and timing of the accident. For this risk assessment, ingestion doses are only calculated for accidents occurring in rural areas (the calculated ingestion doses, however, assume all food grown on contaminated ground is consumed and is not limited to the rural population). Examination of the severe accident consequence assessment results has shown that ingestion of contaminated foodstuffs contributes on the order of 50 percent of the total population dose for rural accidents. Interdiction of foodstuffs would act to reduce, but not eliminate, this contribution.

J.9 REFERENCES

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10 CFR Part 71, "Packaging and Transportation of Radioactive Materials," U.S. Nuclear Regulatory Commission.

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10 CFR Section 71.73, "Hypothetical Accident Conditions," U.S. Nuclear Regulatory Commission.

10 CFR Section 71.74, "Accident Conditions for Air Transport of Plutonium," U.S. Nuclear Regulatory Commission.

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Appendix K
Environmental Justice Analysis

K.1 INTRODUCTION

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations* (59 FR 7629), directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse health or environmental effects of their programs, policies, and activities on minority populations and low-income populations.

The Council on Environmental Quality has oversight responsibility for documentation prepared in compliance with the National Environmental Policy Act (NEPA). In December 1997, the Council released its guidance on environmental justice under NEPA (CEQ 1997). The Council's guidance was adopted as the basis for the analysis of environmental justice contained in this *Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]).

This appendix provides an assessment of the potential for disproportionately high and adverse human health or environmental effects on minority and low-income populations from the implementation of each alternative.

K.2 DEFINITIONS

MINORITY INDIVIDUALS AND POPULATION

The following definitions of minority individuals and population were used in this analysis of environmental justice:

- **Minority individuals—**Individuals who are members of the following populations groups: American Indian or Alaskan Native; Asian or Pacific Islander; Black, not of Hispanic origin; or Hispanic.

- **Minority population—**Minority populations should be identified where either: (a) the minority population of the affected area exceeds 50 percent or (b) the minority population percentage of the affected area is meaningfully greater than the minority population percentage in the general population or other appropriate unit of geographic analysis. In identifying minority communities, agencies may consider as a community either a group of individuals living in geographic proximity to one another, or a geographically dispersed and transient set of individuals (such as migrant workers or Native American), where either type of group experiences common conditions of environmental exposure or effect. The selection of the appropriate unit of geographic analysis may be a governing body's jurisdiction, a neighborhood, census tract, or other similar unit that is to be chosen so as to not artificially dilute or inflate the affected minority population. A minority population also exists if there is more than one minority group present and the minority percentage, as calculated by aggregating all minority persons, meets one of the above-stated thresholds.

In the discussions of environmental justice in this NI PEIS, persons self-designated as Hispanic are included in the Hispanic population, regardless of race. For example, the Asian or Pacific Islander population is composed of persons self-designated as Asian or Pacific Islander and not of Hispanic origin. Asian or Pacific Islanders who designated themselves as having Hispanic origins are included in the Hispanic population. Data for the analysis of minority populations in 1990 were extracted from Table P012 of Summary Tape File 3 (DOC 1992). Data for the analysis of minorities and racial populations were extracted for year 2020 from the Census Bureau's World Wide Web site (DOC 1999).

LOW-INCOME INDIVIDUALS AND POPULATION

Executive Order 12898 specifically addresses "disproportionately high and adverse effects" on "low-income" populations. The Council on Environmental Quality recommends that poverty thresholds be used to identify "low-income" individuals.

The following definition of low-income population was used in this analysis:

- **Low-income population—**Low-income populations in an affected area should be identified with the annual statistical poverty thresholds from the U.S. Bureau of the Census' Current Population Reports, Series P–60 on Income and Poverty. In identifying low-income populations, agencies may consider as a community either a group of individuals living in geographic proximity to one another, or a set of individuals (such as migrant workers or Native Americans), where either type of group experiences common conditions of environmental exposure or effect.

Data for the analysis of low-income populations were extracted from Table P121 of Summary Tape File 3 (DOC 1992).

DISPROPORTIONATELY HIGH AND ADVERSE HUMAN HEALTH EFFECTS

Adverse health effects are measured in risks and rates that could result in latent cancer fatalities, as well as other fatal or nonfatal adverse impacts to human health. Disproportionately high and adverse human health effects occur when the risk or rate of exposure to an environmental hazard for a minority population or low-income population is significant and exceeds the risk of exposure rate for the general population or, where available, for another appropriate comparison group (CEQ 1997).

DISPROPORTIONATELY HIGH AND ADVERSE ENVIRONMENTAL IMPACTS

A disproportionately high environmental impact refers to an impact (or risk of an impact) in a low-income or minority community that is significant and exceeds the environmental impact on the larger community. An adverse environmental impact is an impact that is determined to be both harmful and significant. In assessing cultural and aesthetic environmental impacts, impacts that uniquely affect geographically dislocated or dispersed or minority low-income populations are considered (CEQ 1997).

Potentially affected areas examined in this NI PEIS include areas defined by an 80-kilometer (50-mile) radius centered on candidate facilities for plutonium-238 production, radioisotope production, or processing activities located at the Idaho National Engineering and Environmental Laboratory (INEEL), the Oak Ridge Reservation (ORR), and the Hanford Site (Hanford). Potentially affected areas used in the analysis of environmental justice are the same as those used in the analysis of radiological health effects described in Chapter 4.

K.3 METHODOLOGY

K.3.1 Spatial Resolution

For the purposes of enumeration and analysis, the Census Bureau has defined a variety of areal units (DOC 1992). Areal units of concern in this document include (in order of increasing spatial resolution) states, counties, census tracts, block groups, and blocks. The "block" is the smallest of these entities and offers the finest spatial resolution. This term refers to a relatively small geographical area bounded on all sides by visible features such as streets and streams or by invisible boundaries such as city limits and property lines. During the 1990 census, the Census Bureau subdivided the United States and its territories into 7,017,425 blocks. For

comparison, the number of counties, census tracts, and block groups used in the 1990 census were 3,248; 62,276; and 229,192; respectively. While blocks offer the finest spatial resolution, economic data required for the identification of low-income populations are not available at the block-level of spatial resolution. In the analysis below, block groups are used throughout as the areal unit. Block groups generally contain between 250 and 500 housing units (DOC 1992).

During the decennial census, the Census Bureau collects data from individuals and aggregates the data according to residence in a geographical area, such as a county or block group. This NI PEIS uses data from the 1990 census as a baseline. The Census Bureau has not yet published results of the year 2000 census. Boundaries of the areal units are selected to coincide with features such as streams and roads or political boundaries such as county and city borders. Boundaries used for aggregation of the census data usually do not coincide with boundaries used in the calculation of health effects. As discussed in Chapter 4, radiological health effects due to an accident at each of the sites considered for the proposed actions are evaluated for persons residing within a distance of 80 kilometers (50 miles) of an accident site. In general, the boundary of the circle with an 80-kilometer (50-mile) radius centered at the accident site will not coincide with boundaries used by the Census Bureau for enumeration of the population in the potentially affected area. Some block groups lie completely inside or outside of the radius for health effects calculation. However, other block groups are only partially included. As a result of these partial inclusions, uncertainties are introduced into the estimate of the population at risk from the accident.

To estimate the populations at risk in partially included block groups, it was assumed that populations are uniformly distributed throughout the area of each block group. For example, if 30 percent of the area of a block group lies within 80 kilometers (50 miles) of the accident site, it was assumed that 30 percent of the population residing in that block group would be at risk.

K.3.2 Population Projections

Health effects were calculated for populations projected to reside in potentially affected areas during the year 2020. Extrapolations of the total population for individual states are available from both the Census Bureau and various state agencies (Campbell 1996). The Census Bureau also projects populations by ethnic and racial classification in 1-year intervals for the years from 1995 to 2025 at the state level. State agencies project total populations for individual counties. No Federal or state agency projects block group or low-income populations. Data used to project minority populations were extracted from the Census Bureau's World Wide Web site (DOC 1999). To project minority populations in potentially affected areas, minority populations determined from the 1990 census data were taken as a baseline for each block group. Then it was assumed that percentage changes in the minority population of each block group for a given year (compared to the 1990 baseline data) will be the same as percentage changes in the state minority population projected for the same year. An advantage to this assumption is that the projected populations are obtained using a consistent method, regardless of the state and associated block group involved in the calculation. A disadvantage is that the method is insensitive to localized demographic changes that could alter the projection in a specific area.

The Census Bureau uses the cohort-component method to estimate future populations for each state (Campbell 1996). The set of cohorts is comprised of: (1) age groups from 1 year or less to 85 years or more, (2) male and female populations in each age group, and (3) the following racial and ethnic groups in each age group: Hispanic, non-Hispanic Asian, non-Hispanic Black, non-Hispanic Native American, and non-Hispanic White. Components of the population change used in the demographic accounting system are births, deaths,

net state-to-state migration, and net international migration. If P(t) denotes the number of individuals in a given cohort at time "t," then:

$$P(t) = P(t_0) + B - D + DIM - DOM + IIM - IOM$$

where:

$P(t_0)$	=	Cohort population at time $t_0 \leq t$. For this analysis, t_0 denotes the year 1990.
B	=	Births expected during the period from t_0 to t.
D	=	Deaths expected during the period from t_0 to t.
DIM	=	Domestic migration into the state expected during the period from t_0 to t.
DOM	=	Domestic migration out of the state expected during the period from t_0 to t.
IIM	=	International migration into the state expected during the period from t_0 to t.
IOM	=	International migration out of the state expected during the period from t_0 to t.

Estimated values for the components shown on the right side of the equation are based on past data and various assumptions regarding changes in the rates for birth, mortality, and migration (Campbell 1996). Persons of Hispanic origin are included in the Hispanic population regardless of race. It should be noted that the Census Bureau does not project populations of individuals who identified themselves as "other race" during the 1990 census. This population group is less than 2 percent of the total population in each of the states. However, to project total populations in the environmental justice analysis, population projections for the "other race" group were made under the assumption that the growth rate for the "other race" population will be identical to the growth rate for the combined minority and white populations.

K.4 ENVIRONMENTAL JUSTICE ASSESSMENT

The analysis of environmental justice concerns was based on an assessment of the impacts reported in Chapter 4. This analysis was performed to identify any disproportionately high and adverse human health or environmental impacts on minority or low-income populations surrounding the candidate sites. Demographic information obtained from the Census Bureau was used to identify the minority populations and low-income communities in the zone of potential impact surrounding the sites (DOC 1992).

K.5 RESULTS FOR THE SITES FOR THE NO ACTION ALTERNATIVE, ALTERNATIVE 1, ALTERNATIVE 2, AND ALTERNATIVE 5

As discussed in Chapter 2, the Advanced Test Reactor (ATR) at INEEL, the High Flux Isotope Reactor (HFIR) at ORR, and the Fast Flux Test Facility (FFTF) at Hanford are candidate reactors for DOE's nuclear infrastructure programs. Candidate processing facilities include the Fluorinel Dissolution Process Facility (FDPF) at INEEL, the Radiochemical Engineering Development Center (REDC) at ORR, and the Fuels and Materials Examination Facility (FMEF) at Hanford. This section describes the analysis of potentially affected minority and low-income populations residing near those facilities. Projections of the total population provided in this appendix differ from the projected total populations used in the health effects calculations described in Chapter 4. This is because the projections used in the analysis of environmental justice are based on projections for the states provided by the Census Bureau (Campbell 1996). Projections used in the analysis of health effects are based on county-wide projections provided by state agencies. As discussed above, the county projections are more sensitive to localized demographic changes. However, the states do not provide projections for minority populations. Therefore, the Census Bureau's projections were used in the analysis of environmental justice. Population projections obtained with the two approaches differ by 14 percent or less and have essentially no effect on the results of the analyses.

K.5.1 Results for INEEL

As discussed in Chapter 2, three of the candidate facilities for participation in DOE's program for the supply of isotope products and services are located at INEEL: ATR, FDPF, and CPP–651. **Figure K–1** shows the racial and Hispanic composition of the minority population residing within 80 kilometers (50 miles) of ATR in 1990 (DOC 1992) and those projected to reside in the potentially affected area in the year 2020. In the interval between 1990 and 2020, the percentage of the total population composed of minorities is projected to increase from 10 percent to 18 percent. ATR and FDPF are within approximately 3 kilometers (1.9 miles) of one another, and the populations in potentially affected areas for the two facilities are similar.

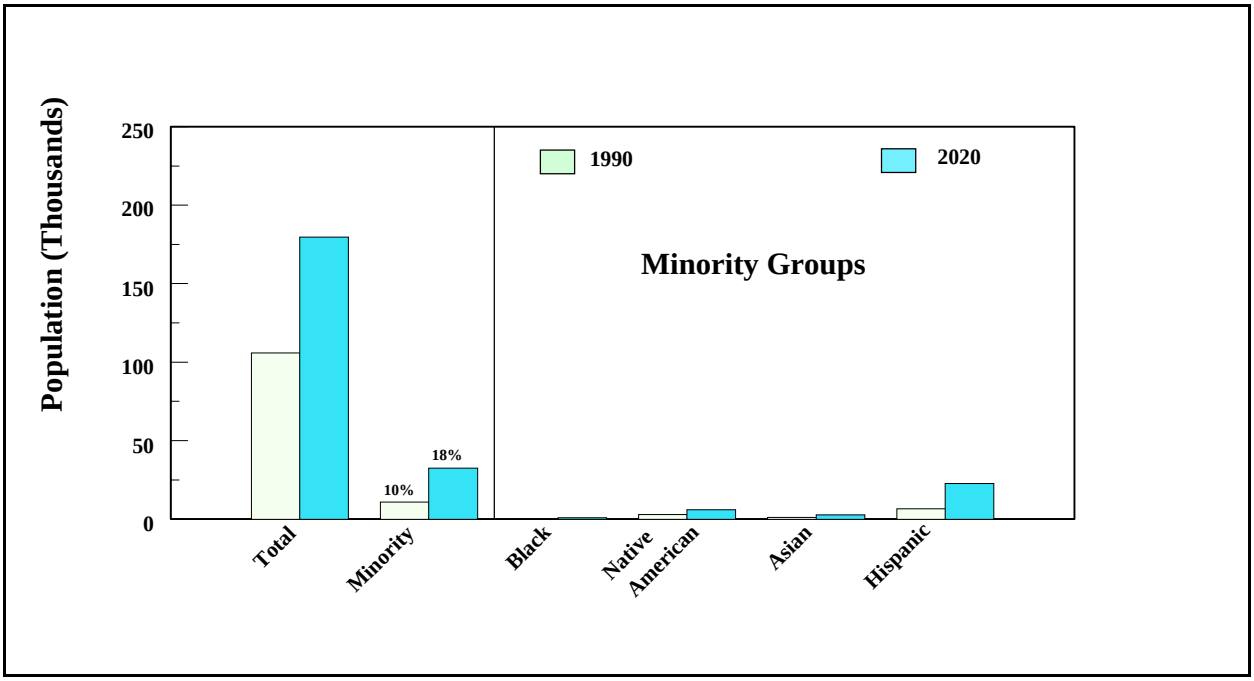

Figure K–1 Racial and Hispanic Composition of the Population Residing Within 80 Kilometers (50 Miles) of ATR at INEEL

For comparison, during the 1990 census, minorities were found to comprise approximately one-quarter of the total national population. By the year 2020, minorities are projected to comprise approximately one-third of the total national population. The percentage of the minority population residing in the potentially affected areas surrounding the INEEL site was less than the corresponding national percentage in 1990, and is expected to remain so through the year 2020. Hispanics are the largest minority group residing in the potentially affected area, and both Asian and Hispanic populations are projected to show the largest growth rates.

During the 1990 census, approximately 13 percent of the residents within the potentially affected area surrounding the INEEL site reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and approximately 13 percent of the residents of Idaho reported incomes below the poverty threshold during the same year. Thus, the percentage of low-income population residing within the potentially affected area in 1990 was equal to that for the Nation and the State of Idaho.

Figures K–2 and **K–3** show the geographical distribution of minority and low-income populations residing near INEEL in 1990. As indicated in Figure K–2, block groups for which the percentage of minority residents exceeds the corresponding national percentage (approximately 24 percent in 1990) are concentrated in the area

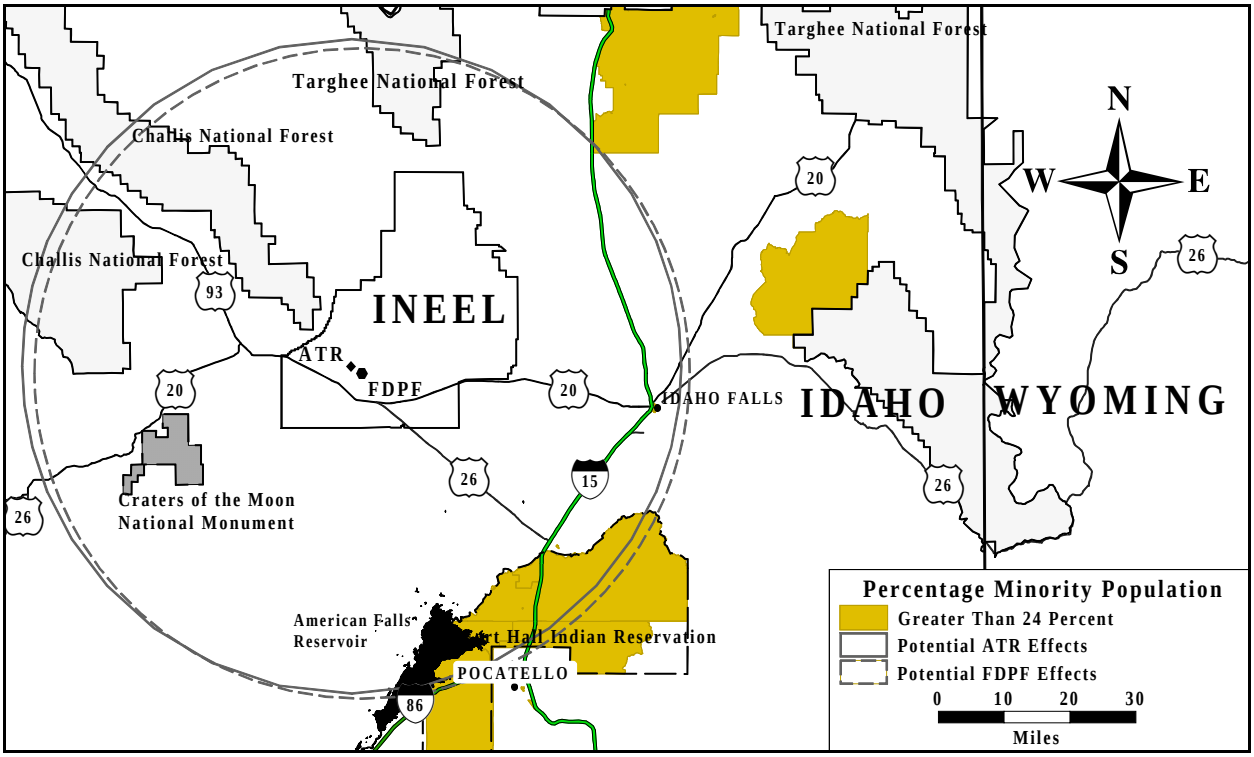

Figure K–2 Geographical Distribution of Minority Populations Residing Within 80 Kilometers (50 Miles) of ATR and FDPF at INEEL

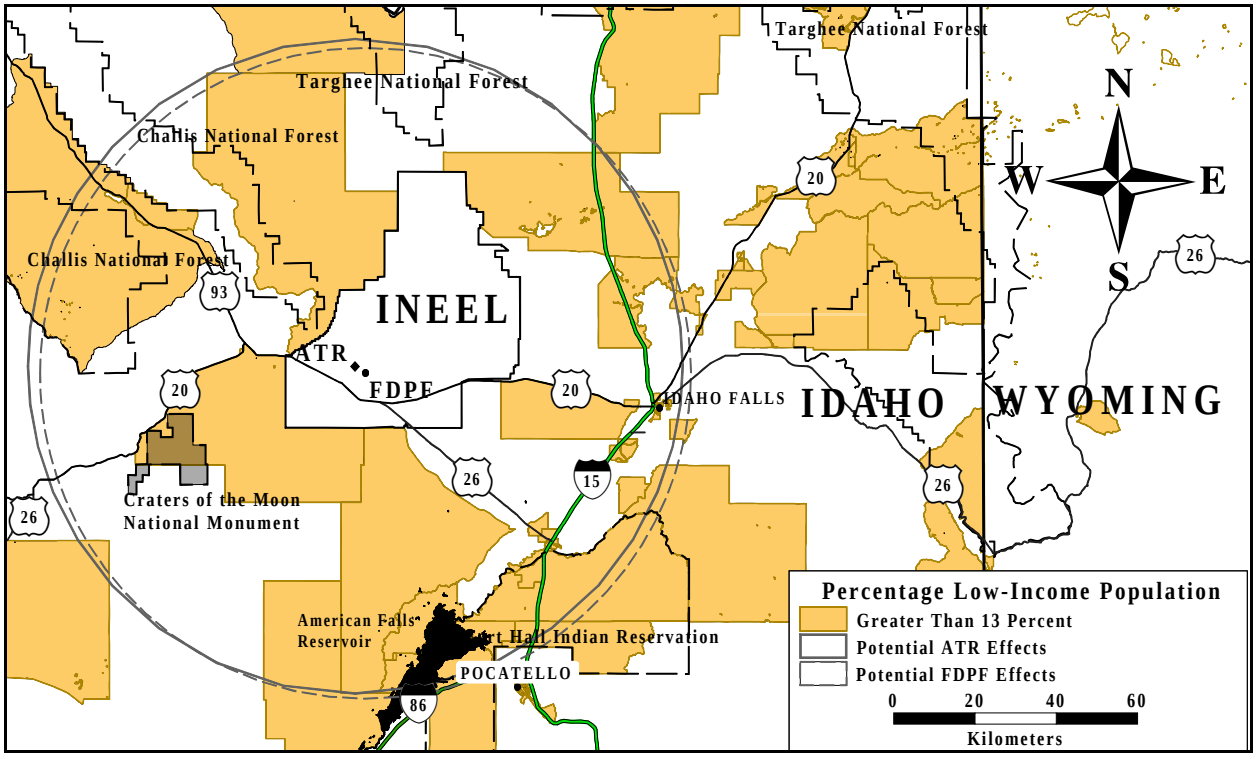

Figure K–3 Geographical Distribution of Low-Income Populations Residing Within 80 Kilometers (50 Miles) of ATR and FDPF at INEEL

of the Fort Hall Reservation. Block groups for which the percentage of the low-income population exceeds the national percentage (approximately 13 percent in 1990) are located throughout the potentially affected area. Thus, estimates of the effects of radiological or chemical releases to the atmosphere on minority populations could be noticeably influenced by assumptions concerning weather conditions existing during a given release. Low-income populations are located more uniformly throughout the potentially affected area, and estimates of the radiological effects on low-income populations would be less sensitive to assumptions concerning prevailing wind direction, wind speed, and atmospheric stability class. Appendix H discussed the radiological and nonradiological release models used in this NI PEIS.

As discussed in Chapter 4, normal operations that would result from implementation of the alternatives at INEEL would pose no significant incremental health or other risks to persons residing within the potentially affected area. Environmental justice concerns in the INEEL area include food consumption patterns of minority communities, such as the Fort Hall Indian Reservation, and low-income communities that are located throughout the potentially affected area. In order to assess potential health risks to minority and low-income populations, the health impacts due to ingestion of contaminated food in the potentially affected area were evaluated with the GENII computer model (see Appendix H). Health risks due to normal operations were evaluated under the assumption that all food consumed by residents in the potentially affected area during the 35-year operational period would be obtained locally and would be subject to radiological contamination that could result from normal operations. The maximum population dose to populations at risk near INEEL due to ingestion of radiologically contaminated food would be approximately 2.6×10^{-6} person-rem. The associated risk of a latent cancer fatality would be essentially zero. Thus, no credible pattern of food consumption by minority or low-income populations would result in a significant health risk attributable to radiological contamination of the food supply that could result from normal operations.

In the event of a radiological accident at one of the INEEL facilities, radiological contamination would be directed toward the Fort Hall Indian Reservation if the prevailing winds at the time of the accident were blowing from the northwest (see Figure K–2). However, accidents that could occur at INEEL under implementation of the alternatives would not be expected to result in a latent cancer fatality among the exposed population or the maximally exposed individual residing within the boundaries of the Fort Hall Reservation.

Implementation of the alternatives would thus pose no significant radiological risks to minority or low-income populations residing within the potentially affected area surrounding INEEL.

K.5.2 Results for ORR

As discussed in Chapter 2, HFIR is a candidate reactor for DOE's isotope supply program, and REDC is a candidate processing facility. These facilities are co-located at latitude 35° 55'7" north and longitude 84° 18'14" west. **Figure K–4** shows the racial and Hispanic composition of the minority population residing within 80 kilometers (50 miles) of HFIR and REDC at ORR in 1990 (DOC 1992) and those projected to reside in the potentially affected area in the year 2020. In the interval between 1990 and 2020, the percentage of the total population composed of minorities is projected to increase from approximately 6 to 8 percent. For comparison, during the 1990 census, minorities were found to comprise approximately one-quarter of the total national population. By the year 2020, minorities are projected to comprise approximately one-third of the total national population. The percentage of the minority population residing in the potentially affected area surrounding ORR was less than the corresponding national percentage in 1990, and is expected to remain so through the year 2020. Blacks are the largest minority group residing in the potentially affected area, while the Asian and Hispanic populations are projected to show the largest growth rates.

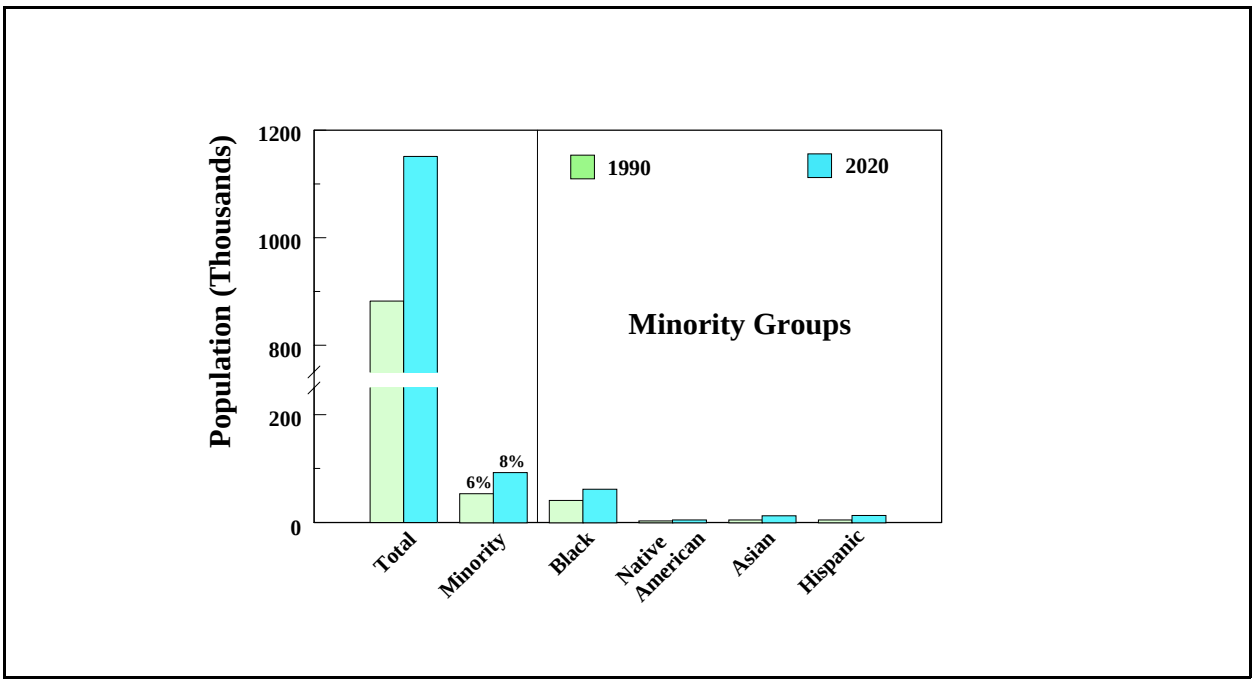

Figure K–4 Racial and Hispanic Composition of Populations Residing Within 80 Kilometers (50 Miles) of HFIR and REDC at ORR

Figure K–5 shows the geographical distribution of the minority population residing near ORR. Minority populations within the potentially affected area are concentrated largely in the area of Knoxville, Tennessee. **Figure K–6** provides an expanded view of the areas surrounding Knoxville and ORR. The minority community of Scarboro is adjacent to the northern boundary of ORR. Approximately 60 percent of the minority populations residing in the potentially affected area surrounding ORR also reside within 16 kilometers (10 miles) of Knoxville. Thus, estimates of the effects of radiological or chemical releases could be noticeably influenced by assumptions concerning weather conditions existing during a given release.

During the 1990 census, 16 percent of the residents within the potentially affected area surrounding ORR reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and nearly 16 percent of the residents of Tennessee reported incomes below the poverty threshold during the same year. Thus, the percentage of low-income population residing within the potentially affected area exceeded that for the Nation, but is equal to the corresponding percentage for Tennessee. **Figures K–7** and **K–8** show the geographical distribution of low-income residents surrounding ORR and the Knoxville-ORR area, respectively. Block groups for which the percentage of low-income residents exceeds the corresponding national percentage are located throughout the potentially affected area. Thus, estimates of the effects of radiological or chemical releases on low-income populations are less sensitive to assumptions concerning existing weather conditions.

As discussed in Chapter 4, normal operations that would result from implementation of the alternatives at ORR would pose no significant incremental health or other risks to persons residing within the potentially affected area. Environmental justice concerns in the ORR area include food consumption patterns of minority communities, such as those in Scarboro and Knoxville, and low-income communities that are located throughout the potentially affected area. In order to assess potential health risks to minority and low-income populations, the health impacts due to ingestion of contaminated food in the potentially affected area were evaluated with the GENII computer model (see Appendix H). Health risks due to normal operations were evaluated under the assumption that all food consumed by residents in the potentially affected area during the

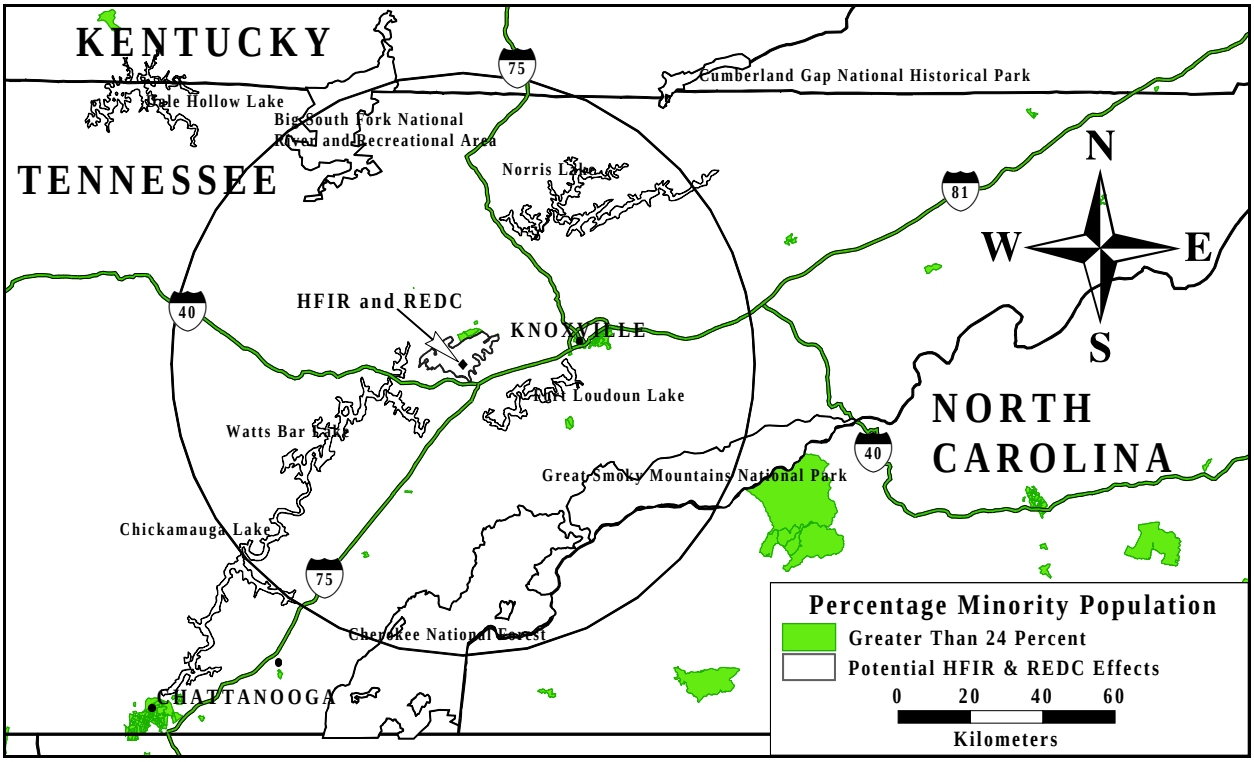

Figure K–5 Geographical Distribution of Minority Populations Residing Within 80 Kilometers (50 Miles) of HFIR and REDC at ORR

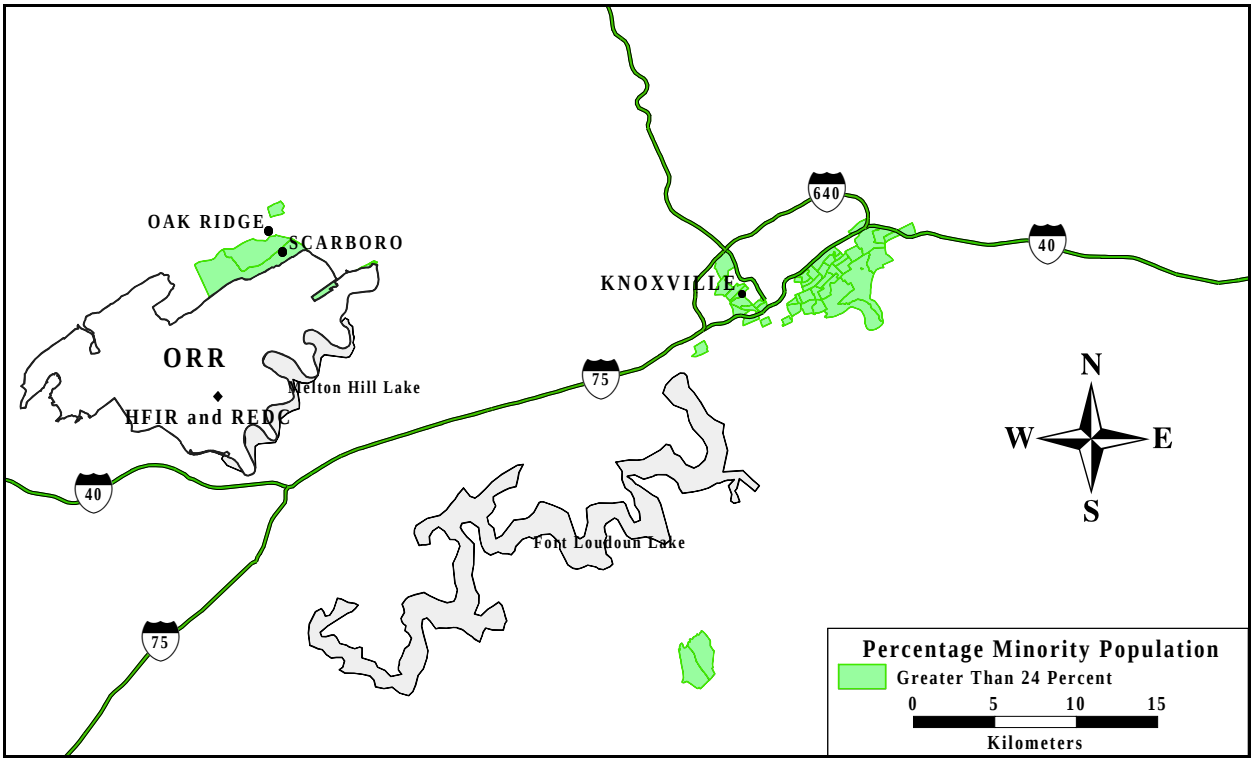

Figure K–6 Geographical Distribution of Minority Populations in the Knoxville, Tennessee Area

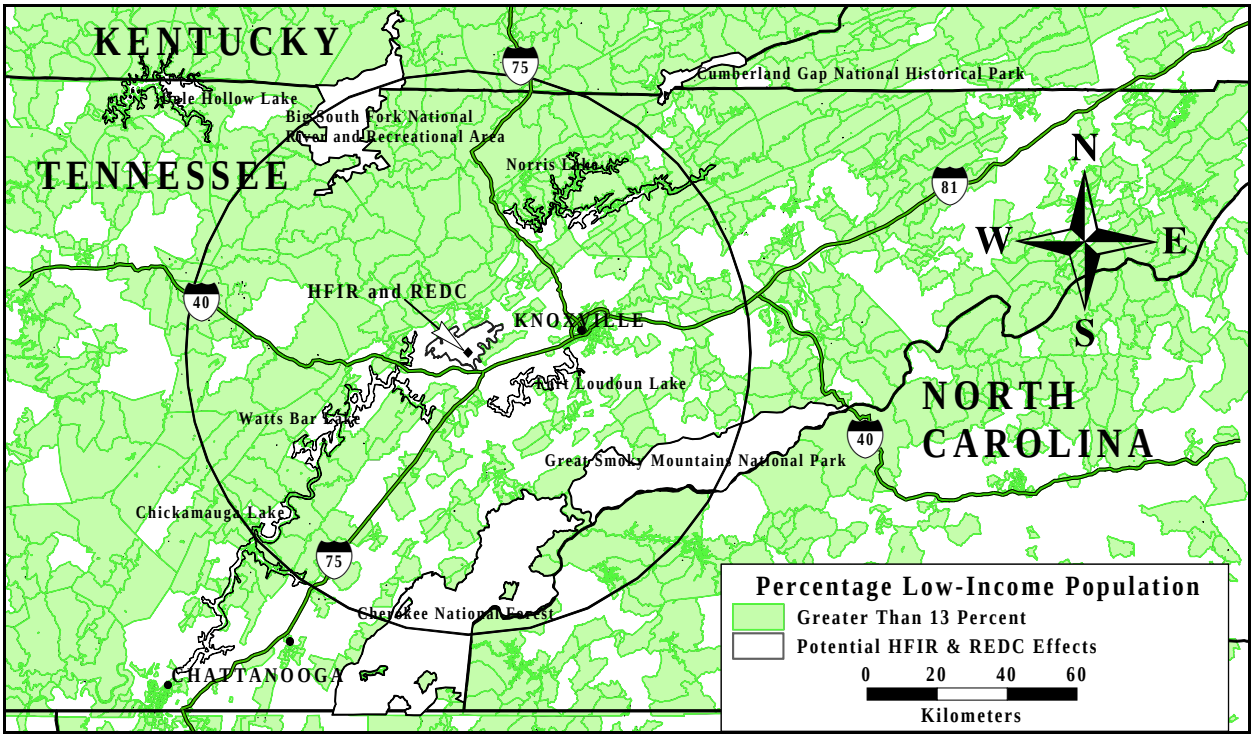

Figure K–7 Geographical Distribution of Low-Income Populations Residing Within 80 Kilometers (50 Miles) of HFIR and REDC at ORR

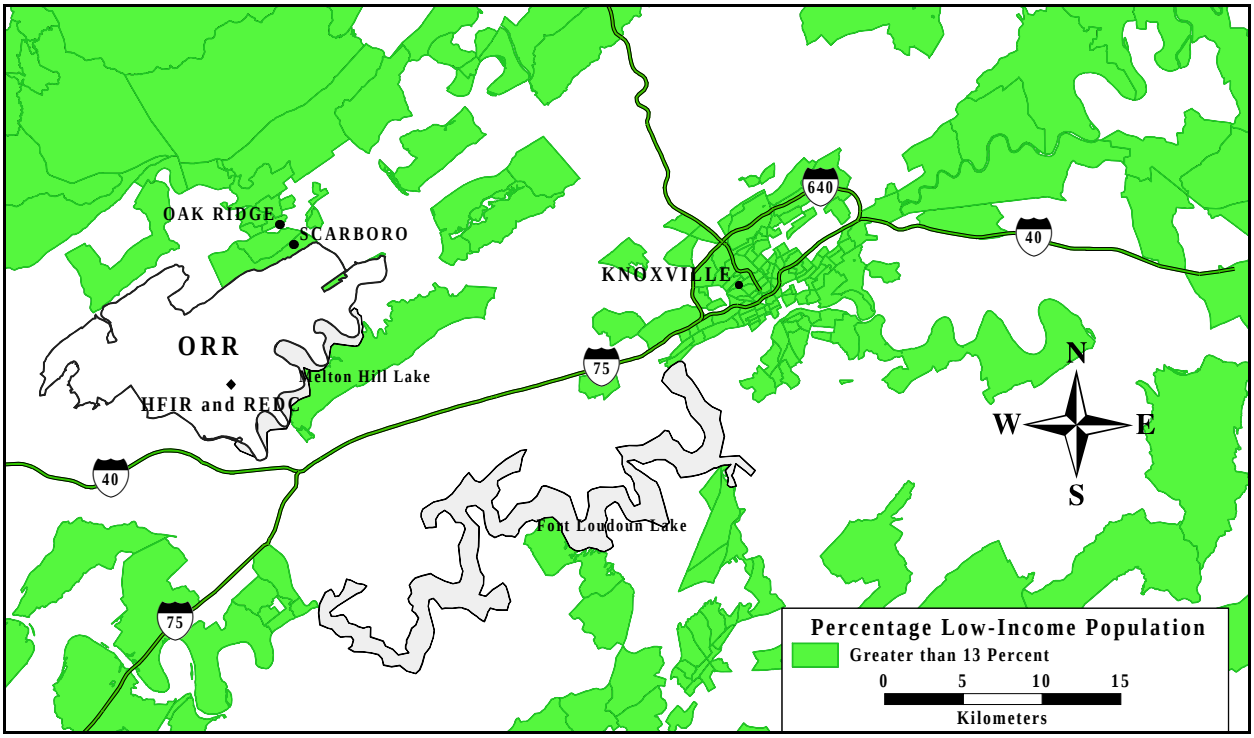

Figure K–8 Geographical Distribution of Low-Income Populations in the Knoxville, Tennessee Area

35-year operational period would be obtained locally and would be subject to radiological contamination that could result from normal operations. The maximum population dose to populations at risk near ORR due to ingestion of radiologically contaminated food would be approximately 2.6×10^{-7} person-rem. The associated risk of a latent cancer fatality would be essentially zero. Thus, no credible pattern of food consumption by minority or low-income populations would result in a significant health risk attributable to radiological contamination of the food supply that could result from normal operations.

As shown in Figure K–6, minorities at risk in the potentially affected area are concentrated in the community of Scarboro adjacent to the northern boundary of ORR and Knoxville, Tennessee. In the event of a radiological accident at one of the ORR facilities, radiological contamination would be directed toward Scarboro if the prevailing winds at the time of the accident were blowing from the south-southwest (see Figure K–6). As discussed in Chapter 4 (e.g., see Section 4.3.1.1.10), the largest radiological risks to the public that could result from implementation of the alternatives are those that could result from accidents at candidate fabrication and processing facilities.

Scarboro is the minority community closest to the boundary of ORR. The residents of Scarboro are among the populations at risk due to radiological accidents that could occur at REDC. Demographic data for Scarboro is contained in the block group designated by Federal Information Processing Standard (FIPS) code 470010201004. Projection of the population of Scarboro to the year 2020 shows that its total population is expected to be approximately 964 in 2020, and the minority population is expected to be approximately 877 in 2020. In order to conservatively estimate the risk of latent cancer fatalities resulting among minority residents of Scarboro from a beyond-design-basis earthquake at REDC, it was assumed that if such an earthquake were to occur, then all minority residents of Scarboro would receive a radiation dose of 26.3 rem. That is the dose that would be received by the maximally exposed individual residing at the ORR boundary in the direction north-northwest from REDC. Under this assumption, the resulting population dose would be approximately 23,065 person-rem. For an accident frequency of 10^{-5} (see Section 4.3.1.1.10), the annual risk to the projected minority population under these conservative assumptions would be 2.3×10^{-4} latent cancer fatality. Over the 35-year program duration, the expected latent cancer fatalities among the minority population would be approximately 0.008. No latent cancer fatalities among the minority population of Scarboro would be expected to result from a beyond-design-basis earthquake at REDC. In addition, if REDC were selected to process plutonium-238 under the nuclear infrastructure alternatives, specific safety analysis documentation would be developed to provide the authorization basis for REDC operations. If DOE site officials deem it necessary, facility and/or operational modification would be used to mitigate the impacts of severe accidents.

Approximately one-third of the residents of Scarboro reported incomes less than the poverty threshold in 1990. Under the assumption that one-third of the residents of Scarboro in 2020 would report incomes less than the poverty threshold, the dose received by low-income residents due to a beyond-design-basis earthquake at REDC would be approximately 8,450 person-rem. The annual risk to low-income residents of Scarboro would then be approximately 8.5×10^{-5} latent cancer fatality. Over the 35-year program duration, the expected latent cancer fatalities among the projected low-income population would be approximately 0.003. No latent cancer fatalities among the low-income population of Scarboro would be expected to result from a beyond-design-basis earthquake at REDC.

If the prevailing winds were blowing from the west-southwest during an accident at REDC, then radiological contamination resulting from the accident would be directed toward minority and low-income communities residing in the Knoxville area. The closest of these communities is approximately 30 kilometers (18.8 miles) from HFIR and REDC (see Figures K–6 and K–8), and therefore accidents at the reactor or fabrication and processing facilities would not be expected to pose a significant risk to the minority or low-income populations in the Knoxville area.

Implementation of the alternatives would thus pose no significant radiological risks to minority or low-income populations residing within the potentially affected area surrounding ORR.

K.5.3 Results for Hanford

As discussed in Chapter 2, candidate facilities at Hanford include the FFTF reactor, the FMEF processing and storage facility, and storage facilities in the Radiochemical Processing Laboratory (RPL) and Building 306–E. FFTF is located at latitude 46° 26′8″ north and longitude 119° 21′32″ west. FMEF is located at latitude 46º 26'7.0" north and longitude 119º 21'55.0" west. RPL is located at latitude 46° 22′7.7″ north and longitude 119° 16′43.7″ west. **Figure K–9** shows the racial and Hispanic composition of the minority population residing within 80 kilometers (50 miles) of FMEF in 1990 and those projected to reside in the potentially affected area in 2020. In the interval from 1990 to 2020, the percentage of the total population composed of minorities is projected to increase from approximately 25 percent to 44 percent. For comparison, during the 1990 census, minorities were found to comprise nearly one-quarter of the total national population. By the year 2020, minorities are projected to comprise approximately one-third of the total national population. The percentage of the minority population residing in the potentially affected area surrounding Hanford and FMEF was approximately equal to the corresponding national percentage in 1990, but is projected to exceed the corresponding national percentage by the year 2020. As indicated in Figure K–9, Hispanics are the largest minority group residing in the potentially affected area. The population at risk near FMEF is typical of populations at risk surrounding FFTF and RPL/306–E in the 300 Area. Total and minority populations residing in potentially affected areas surrounding FMEF, FFTF, and RPL differ by less than 1 percent.

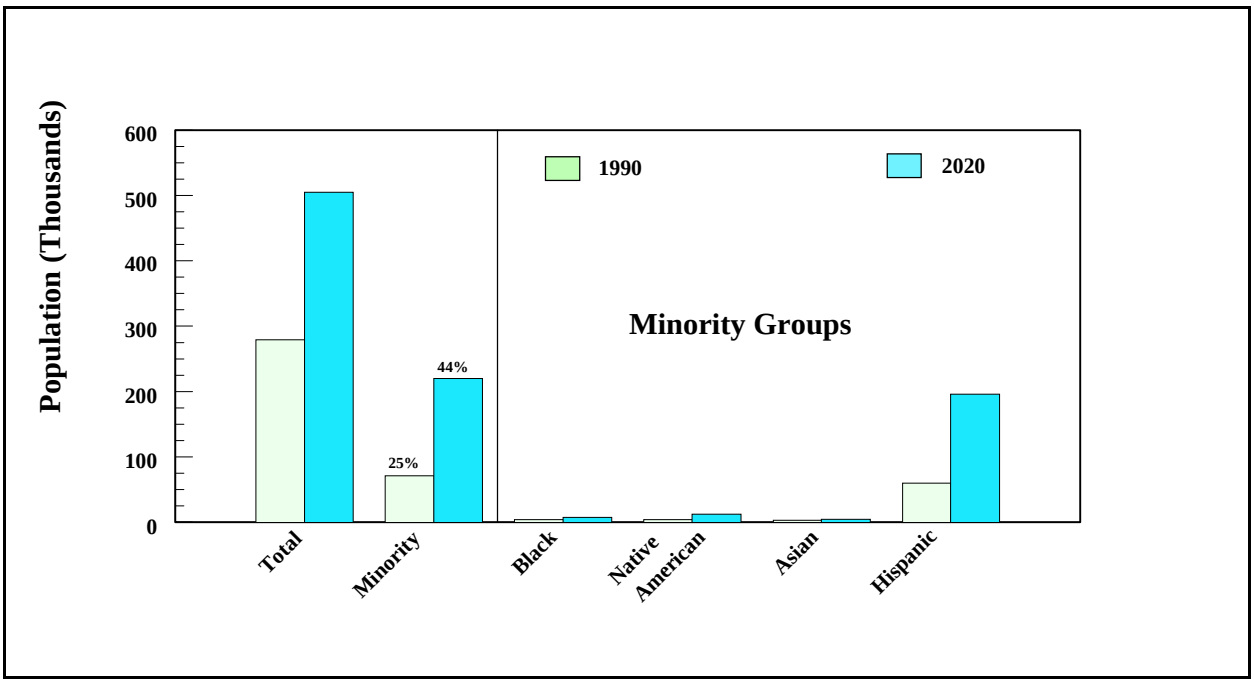

Figure K–9 Racial and Hispanic Composition of Populations Residing Within 80 Kilometers (50 Miles) of FFTF and RPL/306–E at Hanford

During the 1990 census, approximately 17 percent of the residents within the potentially affected area surrounding Hanford reported incomes below the poverty threshold. Slightly over 13 percent of the national population reported incomes below the poverty threshold, and approximately 11 percent of the residents of the State of Washington reported incomes below the poverty threshold during the same year. Thus, the percentage

of low-income population residing within the potentially affected area in 1990 exceeded both the national percentage and that for the State of Washington.

Figures K–10 and **K–11** show the geographical distribution of minority and low-income populations residing near the Hanford site in 1990. As indicated in the figures, block groups for which the percentage of minority residents exceeds the corresponding national percentage (approximately 24 percent in 1990) or for which the percentage low-income population exceeds the corresponding national percentage (approximately 13 percent in 1990) are located throughout the potentially affected area.

As discussed in Chapter 4, normal operations that would result from implementation of the alternatives at Hanford would pose no significant incremental health or other risks to persons residing within the potentially affected area. Environmental justice concerns in the Hanford area include food consumption patterns of minority communities, such the Yakama Indian Reservation, and low-income communities that are located throughout the potentially affected area. In order to assess potential health risks to minority and low-income populations, the health impacts due to ingestion of contaminated food in the potentially affected area were evaluated with the GENII computer model (see Appendix H). Health risks due to normal operations were evaluated under the assumption that all food consumed by residents in the potentially affected area during the 35-year operational period would be obtained locally and would be subject to radiological contamination that could result from normal operations. The maximum population dose to populations at risk near Hanford due to ingestion of radiologically contaminated food would be approximately 2.0 person-rem due to normal operations at FMEF and 4.6 person-rem due to normal operations at RPL. The associated risk would be approximately 0.001 latent cancer fatality due to normal FMEF operations and approximately 0.0023 latent cancer fatality due to normal operations at RPL. Thus, no credible pattern of food consumption by minority or low-income populations would result in a significant health risk attributable to radiological contamination of the food supply that could result from normal operations.

In the event of a radiological accident at one of the Hanford facilities, radiological contamination would be directed toward the Yakama Indian Reservation if the prevailing winds at the time of the accident were blowing from the northeast (see Figure K–11). However, accidents that could occur at Hanford under implementation of the alternatives would not be expected to result in a latent cancer fatality among the exposed population or the maximally exposed individual residing within the boundary of the Yakama Indian Reservation.

Implementation of the alternatives would thus pose no significant radiological risks to minority or low-income populations residing within the potentially affected area surrounding Hanford.

K.6 RESULTS FOR REPRESENTATIVE TRANSPORTATION ROUTES

As discussed in Chapter 4 and Appendix J, selection of the No Action Alternative or Alternative 5 would pose no significant radiological or nonradiological risks to the public. Hence, selection of the No Action Alternative would not be expected to result in disproportionately high and adverse risks for any group within the general population, including minority and low-income groups.

The highest transportation risks to the public would occur if Alternative 1, 3, or 4 were selected for implementation. Due largely to the radiological risks that could result from accidents during the air transportation of isotopes under these alternatives, the number of latent cancer fatalities over the 35-year program would be approximately 0.5. Because air transportation accidents could occur anywhere along the flight path, the associated radiological risks would not disproportionately fall on any particular population regardless of the racial, ethnic, and economic composition of that population.

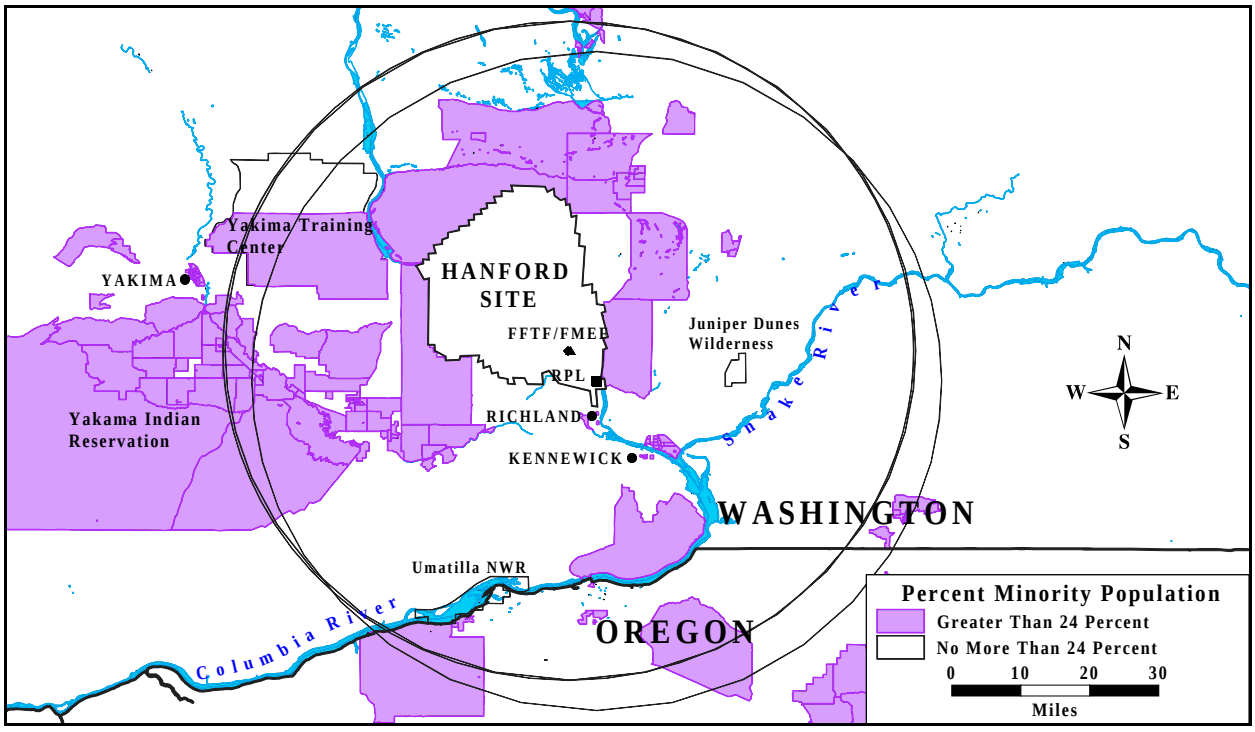

Figure K–10 Geographical Distribution of Minority Populations Residing Within 80 Kilometers (50 Miles) of FFTF and RPL/306–E at Hanford

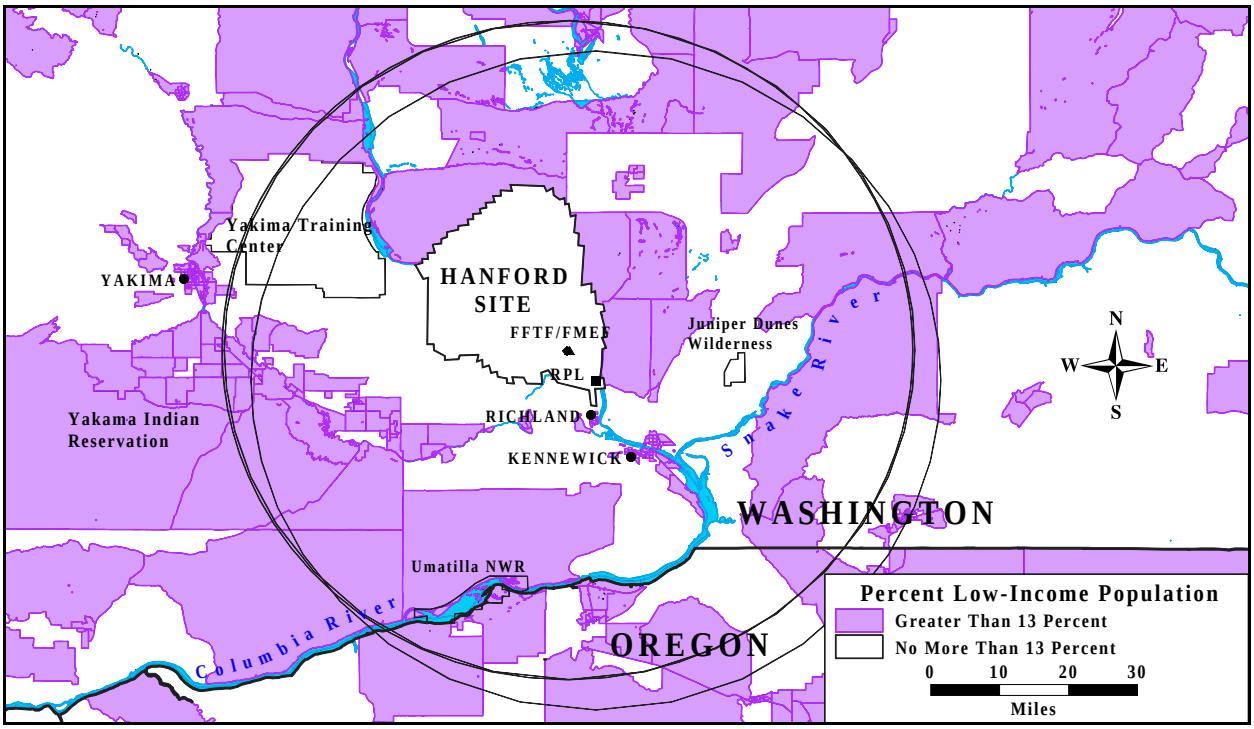

Figure K–11 Geographical Distribution of Low-Income Populations Residing Within 80 Kilometers (50 Miles) of FFTF, FMEF, and RPL/306–E at Hanford

Under Alternative 2, the highest transportation risks to the public result from vehicular collisions (without radiological consequences). No traffic fatalities would be expected. Traffic accidents could occur anywhere along the highway routes, and no identifiable group within the general population would be subject to disproportionate risks.

K.7 ENVIRONMENTAL JUSTICE FOR ALTERNATIVES 3 AND 4

As discussed in Chapter 2, under Alternatives 3 and 4, DOE would construct one or more new facilities for isotope production at sites yet to be selected. Environmental effects described in Chapter 4 for these alternatives are intended to be representative of the impacts on total populations that could occur for generic populations. However, the characterization of minority and low-income populations is site-specific. If Alternative 3 or Alternative 4 were selected for implementation at a specific site, an additional environmental justice analysis for that site would be conducted prior to implementation.

K.8 CUMULATIVE IMPACTS

As discussed in Section 4.8, implementation of the alternatives would be expected to have no significant impact on existing environments at INEEL, ORR, and Hanford. Cumulative radiological risks to the maximally exposed individual in the general population are essentially zero. Radiological impacts on the public that could result from implementation of the No Action Alternative, Alternative 1, Alternative 2, or Alternative 5 at these sites are small. No credible pattern in food consumption would result in significant risks to the public residing within potentially affected areas surrounding INEEL, ORR, and Hanford due to implementation of these alternatives.

Evaluations of environmental justice are site specific, and if Alternative 3 or Alternative 4 were selected for implementation, an analysis of the cumulative impacts on minority and low-income populations would be performed prior to implementation.

K.9 REFERENCES

Federal Register

59 FR 7629, Executive Office of the President, 1994, "Executive Order 12898 - Federal Actions To Address Environmental Justice in Minority Populations and Low-Income Populations," February 11.

Other References

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Appendix L
Socioeconomics Analysis

L.1 INTRODUCTION

This appendix includes the supporting data used for assessing the alternatives in the socioeconomics sections of this *Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility*. The socioeconomic analysis involved two major steps: (1) the characterization and projection of existing social, economic, and infrastructure conditions surrounding each of the candidate sites (i.e., the affected environment); and (2) the evaluation of potential changes in the socioeconomic conditions that could result from alternatives in the regions addressed (i.e., the environmental consequences). Data and analyses used to support the assessment made for the socioeconomic sections are presented in **Tables L–1** through **L–27**.

The socioeconomic environment is defined for two geographic regions: the regional economic area and the region of influence. Regional economic areas are used to assess potential effects on the regional economy, and regions of influence are used to assess effects that are more localized in political jurisdictions surrounding the sites.

The regional economic areas for each site encompasses a broad market that involves trade among and between regional industrial and service sectors and is characterized by strong economic linkages between the communities in the region. These linkages determine the nature and magnitude of multiplier effects of economic activity (purchases, earnings, and employment) at each site. Regional economic areas are defined by the U.S. Bureau of Economic Analysis and consist of an economic node that serves as the center of economic activity and the surrounding counties that are economically related and include the places of work and residences of the labor force.

Potential demographic impacts were assessed for each region of influence, a smaller geographic area where the housing market and local community services would be most affected. Site-specific regions of influence were identified as those counties where approximately 90 percent of the current U.S. Department of Energy and contractor employees reside. This residential distribution reflects existing commuting patterns and attractiveness of area communities for people employed at each site and is used to estimate the future distribution of in-migrating workers.

Table L–1 Oak Ridge Reservation Regional Economic Area Employment and Economy, 2000–2040

Regional Economic Area	2000	2005	2010	2020	2030	2040
Civilian labor force	493,669	526,450	549,317	593,551	642,649	697,307
Total employment	473,203	504,572	526,445	568,727	615,630	667,807
Unemployment rate (percent)	4.1	4.2	4.2	4.2	4.2	4.2

Source: DOL 2000; State of Tennessee 2000.

Table L–2 Oak Ridge Reservation Region of Influence Population, 2000–2040

County	2000	2005	2010	2020	2030	2040
Anderson	72,327	74,115	75,836	79,103	82,512	86,068
Knox	373,576	388,868	403,631	431,759	461,847	494,032
Loudon	39,526	42,126	44,718	49,988	55,880	62,467
Roane	50,897	52,596	54,427	58,107	62,035	66,229
Total	**536,326**	**557,705**	**578,612**	**618,957**	**662,274**	**708,796**

Source: Forstall 1995; State of Tennessee 2000.

Table L–3 Oak Ridge Reservation Region of Influence Total Number of Owner and Renter Housing Units, 2000–2040

County	2000	2005	2010	2020	2030	2040
Anderson	31,075	31,843	32,582	33,986	35,451	36,978
Knox	159,759	166,298	172,611	184,640	197,507	211,271
Loudon	16,434	17,515	18,593	20,784	23,234	25,972
Roane	21,914	22,646	23,434	25,018	26,710	28,516
Total	**229,182**	**238,302**	**247,220**	**264,429**	**282,902**	**302,737**

Source: DOC 1992; State of Tennessee 2000.

Table L–4 Oak Ridge Reservation Region of Influence Total Student Enrollment, 2000–2040

County	2000	2005	2010	2020	2030	2040
Anderson	6,849	7,018	7,181	7,491	7,814	8,150
Knox	52,266	54,405	56,471	60,406	64,616	69,119
Loudon	4,700	5,009	5,317	5,944	6,645	7,428
Roane	6,678	6,901	7,141	7,624	8,139	8,690
Total	**70,493**	**73,333**	**76,110**	**81,465**	**87,214**	**93,387**

Source: Davis 2000; Garza 2000; Groover 2000; McKinney 2000; Pierce 2000; State of Tennessee 2000.

Table L–5 Oak Ridge Reservation Region of Influence Total Number of Teachers, 2000–2040

County	Ratio of Students to Teachers	2000	2005	2010	2020	2030	2040
Anderson	13.2	518	531	543	567	591	616
Knox	17.3	3,014	3,137	3,256	3,483	3,726	3,986
Loudon	18.8	250	266	283	316	353	395
Roane	14.8	450	465	481	514	548	586
Total	**16.7**	**4,232**	**4,399**	**4,563**	**4,880**	**5,218**	**5,583**

Source: Davis 2000; Garza 2000; Groover 2000; McKinney 2000; Pierce 2000; State of Tennessee 2000.

Table L–6 Oak Ridge Reservation Region of Influence Total Number of Sworn Police Officers, 2000–2040

County	2000	2005	2010	2020	2030	2040
Anderson	158	162	165	173	180	188
Knox	1,159	1,207	1,253	1,340	1,433	1,533
Loudon	84	90	95	106	119	133
Roane	112	115	119	128	136	145
Total	**1,513**	**1,574**	**1,632**	**1,747**	**1,868**	**1,999**

Source: HPI 1999; State of Tennessee 2000.

Table L–7 Oak Ridge Reservation Region of Influence Total Number of Firefighters, 2000–2040

County	2000	2005	2010	2020	2030	2040
Anderson	285	292	299	311	325	339
Knox	625	651	675	722	773	826
Loudon	162	173	183	205	229	256
Roane	241	249	258	275	294	314
Total	**1,313**	**1,365**	**1,415**	**1,513**	**1,621**	**1,735**

Source: State of Tennessee 1998, 2000.

Table L–8 Oak Ridge Reservation Region of Influence Number of Hospital Beds, 2000–2040

County	2000	2005	2010	2020	2030	2040
Anderson	313	321	329	343	357	373
Knox	2,442	2,542	2,638	2,822	3,019	3,229
Loudon	55	59	62	69	78	87
Roane	155	161	166	177	189	202
Total	**2,965**	**3,083**	**3,195**	**3,411**	**3,643**	**3,891**

Source: AHA 1995; State of Tennessee 2000.

Table L–9 Oak Ridge Reservation Region of Influence Total Number of Doctors, 2000–2040

County	2000	2005	2010	2020	2030	2040
Anderson	185	190	194	202	211	220
Knox	1,341	1,396	1,449	1,550	1,658	1,774
Loudon	34	36	39	43	48	54
Roane	36	37	39	41	44	47
Total	**1,596**	**1,659**	**1,721**	**1,836**	**1,961**	**2,095**

Source: Randolph, Seidman, and Pasko 1995; State of Tennessee 2000.

Table L–10 INEEL Regional Economic Area Projected Employment and Economy, 2000–2040

Regional Economic Area	2000	2005	2010	2020	2030	2040
Civilian labor force	161,149	168,979	177,199	194,848	214,255	235,595
Total employment	153,440	160,884	168,784	185,662	204,229	224,652
Unemployment rate (percent)	4.8	4.8	4.7	4.7	4.7	4.6

Source: DOL 1999; Idaho Power 1996; State of Wyoming 1996.

Table L–11 INEEL Region of Influence Projected Population, 2000–2040

County	2000	2005	2010	2020	2030	2040
Bannock	78,600	81,808	85,147	92,240	99,924	108,248
Bingham	44,426	46,236	48,120	52,119	56,450	61,141
Bonneville	85,650	89,154	92,802	100,551	108,947	118,044
Jefferson	20,609	21,646	22,736	25,082	27,670	30,526
Total	**229,285**	**238,844**	**248,805**	**269,992**	**292,991**	**317,959**

Source: DOE 1999.

Table L–12 INEEL Region of Influence Projected Number of Owner and Renter Housing Units, 2000–2040

County	2000	2005	2010	2020	2030	2040
Bannock	30,275	31,510	32,796	35,528	38,487	41,693
Bingham	15,138	15,754	16,396	17,759	19,235	20,833
Bonneville	31,215	32,493	33,822	36,646	39,706	43,021
Jefferson	6,643	6,978	7,329	8,086	8,921	9,843
Total	**83,271**	**86,735**	**90,343**	**98,019**	**106,349**	**115,390**

Source: DOE 1999.

Table L–13 INEEL Region of Influence Projected Student Enrollment, 2000–2040

County	Capacity (percent)	2000	2005	2010	2020	2030	2040
Bannock	86.5	15,413	16,042	16,697	18,088	19,595	21,227
Bingham	84.7	11,867	12,350	12,853	13,921	15,078	16,331
Bonneville	91.8	19,782	20,592	21,434	23,224	25,163	27,265
Jefferson	90.6	5,879	6,175	6,486	7,155	7,893	8,708
Total	**88.4**	**52,941**	**55,159**	**57,470**	**62,388**	**67,729**	**73,530**

Source: DOE 1999.

Table L–14 INEEL Region of Influence Projected Number of Teachers, 2000–2040

County	Ratio of Students to Teachers	2000	2005	2010	2020	2030	2040
Bannock	17.9	863	899	935	1,013	1,097	1,188
Bingham	18.2	653	680	707	765	828	896
Bonneville	20.1	982	1,022	1,064	1,153	1,249	1,353
Jefferson	18.4	319	335	352	388	428	472
Total	**18.8**	**2,817**	**2,936**	**3,058**	**3,319**	**3,602**	**3,909**

Source: DOE 1999.

Table L–15 INEEL Region of Influence Projected Number of Sworn Police Officers, 2000–2040

County	2000	2005	2010	2020	2030	2040
Bannock	225	234	244	265	287	311
Bingham	56	58	61	66	72	78
Bonneville	191	199	207	224	243	263
Jefferson	29	30	32	35	39	43
Total	**501**	**521**	**544**	**590**	**641**	**695**

Source: DOE 1999.

Table L–16 INEEL Region of Influence Projected Number of Firefighters, 2000–2040

County	2000	2005	2010	2020	2030	2040
Bannock	188	196	204	221	240	260
Bingham	152	158	165	179	194	211
Bonneville	157	164	170	184	199	215
Jefferson	94	99	104	115	127	141
Total	**591**	**617**	**643**	**699**	**760**	**827**

Source: DOE 1999.

Table L–17 INEEL Region of Influence Projected Number of Hospital Beds, 2000–2040

County	2000	2005	2010	2020	2030	2040
Bannock	434	451	470	509	551	597
Bingham	268	279	290	314	340	368
Bonneville	329	343	357	387	420	456
Jefferson	–	–	–	–	–	–
Total	**1,031**	**1,073**	**1,117**	**1,210**	**1,311**	**1,421**

Source: DOE 1999.

Table L–18 INEEL Region of Influence Projected Number of Doctors, 2000–2040

County	2000	2005	2010	2020	2030	2040
Bannock	146	152	158	171	185	200
Bingham	23	24	25	27	29	32
Bonneville	172	179	186	201	217	235
Jefferson	5	6	6	7	8	10
Total	**346**	**361**	**375**	**406**	**439**	**477**

Source: DOE 1999.

Table L–19 Hanford Regional Economic Area Projected Employment and Economy, 2000–2040

Regional Economic Area	2000	2005	2010	2020	2030	2040
Civilian labor force	369,570	393,230	418,465	473,828	536,515	607,496
Total employment	328,709	349,790	372,278	421,605	477,467	540,732
Unemployment rate (percent)	11.1	11.0	11.0	11.0	11.0	11.0

Source: DOE 1999.

Table L–20 Hanford Region of Influence Projected Population, 2000–2040

County	2000	2005	2010	2020	2030	2040
Benton	149,100	157,549	166,476	185,870	207,524	231,701
Franklin	50,683	54,562	58,738	68,071	78,887	91,422
Total	**199,783**	**212,111**	**225,214**	**253,941**	**286,411**	**323,123**

Source: DOE 1999.

Table L–21 Hanford Region of Influence Projected Number of Owner and Renter Housing Units, 2000–2040

County	2000	2005	2010	2020	2030	2040
Benton	58,217	61,516	65,002	72,575	81,030	90,470
Franklin	17,806	19,168	20,635	23,914	27,714	32,118
Total	**76,023**	**80,684**	**85,637**	**96,489**	**108,744**	**122,588**

Source: DOE 1999.

Table L–22 Hanford Region of Influence Projected Student Enrollment, 2000–2040

County	Capacity (percent)	2000	2005	2010	2020	2030	2040
Benton	90.7	30,427	32,151	33,973	37,931	42,350	47,284
Franklin	97.7	10,896	11,730	12,628	14,636	16,963	19,660
Total	**92.5**	**41,323**	**43,881**	**46,601**	**52,567**	**59,313**	**66,944**

Source: DOE 1999.

Table L–23 Hanford Region of Influence Projected Number of Teachers, 2000–2040

County	Ratio of Students to Teachers	2000	2005	2010	2020	2030	2040
Benton	15.8	1,930	2,039	2,154	2,404	2,683	2,994
Franklin	16.8	647	697	750	869	1,007	1,167
Total	**16.0**	**2,577**	**2,736**	**2,904**	**3,273**	**3,690**	**4,161**

Source: DOE 1999.

Table L–24 Hanford Region of Influence Projected Number of Sworn Police Officers, 2000–2040

County	2000	2005	2010	2020	2030	2040
Benton	225	238	251	280	312	348
Franklin	79	85	92	107	125	146
Total	**304**	**323**	**343**	**387**	**437**	**494**

Source: DOE 1999.

Table L–25 Hanford Region of Influence Projected Number of Firefighters, 2000–2040

County	2000	2005	2010	2020	2030	2040
Benton	399	422	445	496	553	617
Franklin	267	288	310	360	418	485
Total	**666**	**710**	**755**	**856**	**971**	**1,102**

Source: DOE 1999.

Table L–26 Hanford Region of Influence Projected Number of Hospital Beds, 2000–2040

County	2000	2005	2010	2020	2030	2040
Benton	271	287	303	339	379	424
Franklin	143	154	166	193	224	260
Total	**414**	**441**	**469**	**532**	**603**	**684**

Source: DOE 1999.

Table L–27 Hanford Region of Influence Projected Number of Doctors, 2000–2040

County	2000	2005	2010	2020	2030	2040
Benton	225	238	251	280	312	348
Franklin	53	57	61	70	81	93
Total	**278**	**295**	**312**	**350**	**393**	**441**

Source: DOE 1999

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Appendix M
Ecological Resources

The scientific names of animal and plant species that exist in the areas of or on the candidate sites are listed in **Table M–1**. Species are grouped by common name and listed in alphabetical order.

Table M–1 Scientific Names of Animal and Plant Species

Common Name	Scientific Name
Mammals	
Black-tailed jackrabbit	*Lepus californicus*
Bobcat	*Lynx rufus*
Coyote	*Canis latrans*
Elk	*Cervus elaphus*
Gray bat	*Myotis grisescens*
Gray fox	*Urocyon cinereoargenteus*
Gray wolf	*Canis lupus*
Great Basin pocket mouse	*Perognathus parvus*
Indiana bat	*Myotis sodalis*
Mink	*Mustela vison*
Mountain lion	*Felis concolor*
Mule deer	*Odocoileus hemionus*
Pronghorn	*Antilocapra americana*
Pygmy rabbit	*Brachylagus idahoensis*
Raccoon	*Procyon lotor*
Townsend's ground squirrel	*Spermophilus townsendii*
Townsend’s western big-eared bat	*Corynorhinus townsendii*
White-footed mouse	*Peromyscus leucopus*
Whitetail deer	*Odocoileus virginianus*
Birds	
Aleutian Canada goose	*Branta canadensis leucopareia*
American kestrel	*Falco sparverius*
Bald eagle	*Haliaeetus leucocephalus*
Black tern	*Chlidonias niger*
Burrowing owl	*Athene cunicularia*
California gull	*Larus californicus*
Canada goose	*Branta canadensis*
Carolina chickadee	*Parus carolinensis*
Common loon	*Gavia immer*
Ferruginous hawk	*Buteo regalis*
Forster's tern	*Sterna forsteri*
Golden eagle	*Aquila chrysaetos*
Great blue heron	*Ardea herodias*
Great horned owl	*Bubo virginianus*
Horned lark	*Eremophila alpestris*
Loggerhead shrike	*Lanius ludovicianus*
Magpie	*Pica spp.*
Northern cardinal	*Cardinalis cardinalis*
Northern goshawk	*Accipiter gentilis*

Common Name	Scientific Name
Birds (Continued)	
Northern harrier	*Circus cyaneus*
Osprey	*Pandion haliaetus*
Peregrine falcon	*Falco peregrinus*
Prairie falcon	*Falco mexicanus*
Raven	*Corvus spp.*
Red-tailed hawk	*Buteo jamaicensis*
Ring-billed gull	*Larus delawarensis*
Sage grouse	*Centrocercus urophasianus*
Sage sparrow	*Amphispiza belli*
Swainson's hawk	*Buteo swainsoni*
Trumpeter swan	*Cygnus buccinator*
Western meadowlark	*Sturnella neglecta*
Wild turkey	*Meleagris gallopavo*
Reptiles	
Eastern garter snake	*Thamnophis sirtalis*
Gopher snake	*Pituophis melanoleucus*
Northern Pacific rattlesnake	*Crotalus viridis oreganus*
Short-horned lizard	*Phrynosoma douglassi*
Side-blotched lizard	*Uta stansburiana*
Amphibians	
American toad	*Bufo americanus*
Fish	
Blacknose dace	*Rhinichthys atratulus*
Bluegill	*Lepomis macrochirus*
Brook trout	*Salvelinus fontinalis*
Channel catfish	*Ictalurus punctatus*
Chinook salmon	*Oncorhynchus tshawytscha*
Coho salmon	*Oncorhynchus kisutch*
Common carp	*Cyprinus carpio*
Crappie	*Pomoxis spp.*
Freshwater drum	*Aplodinotus grunniens*
Kokanee salmon	*Oncorhynchus nerka*
Largemouth bass	*Micropterus salmoides*
Mountain whitefish	*Prosopium williamsoni*
Rainbow trout	*Salmo gaidneri*
Sauger	*Stizostedion canadense*
Shorthead sculpin	*Cottus confusus*
Smallmouth bass	*Micropterus dolomieui*
Sockeye salmon	*Oncorhynchus nerka*
Speckled dace	*Rhinichthys osculus*
Steelhead trout	*Oncorhynchus mykiss*
Stoneroller	*Campostoma spp.*
Sunfish	*Lepomis spp.*
Walleye	*Stizostedion vitreum*
White sturgeon	*Acipenser transmontanus*

Common Name	Scientific Name
Fish (Continued)	
Yellow perch	*Perca flavescens*
Plants	
Big sagebrush	*Artemisia tridentata*
Bitterbrush	*Purshia tridentata*
Bluebunch wheatgrass	*Agropyron spicatum*
Bluebrush	*Ceanothus thyrsiflorus*
Bottlebrush squirreltail	*Sitanion hystrix*
Bulrush	*Scirpus spp.*
Cattail	*Typha spp.*
Cedar	*Juniperus spp.*
Cheatgrass	*Bromus tectorum*
Cottonwood	*Populus spp.*
Crested wheatgrass	*Agropyron desertorum*
Dune scurfpea	*Psoralea lanceolata*
Giant wildrye	*Elymus condensatus*
Gray horsebrush	*Tetradymia canescens*
Gray rabbitbrush	*Chrysothamnus nauseous*
Greasewood	*Sarcobatus vermiculatus*
Green rabbitbrush	*Chrysothamnus greenei*
Hemlock	*Tsuga canadensis*
Hickory	*Carya spp.*
Indian ricegrass	*Oryzopsis hymenoides*
Juniper	*Juniperus spp.*
Low sagebrush	*Artemisia arbuscula*
Lupine	*Lupinus spp.*
Needle-and-thread grass	*Stipa comata*
Nuttall waterweed	*Elodea nuttallii*
Oak	*Quercus spp.*
Peachleaf willow	*Salix amygdaloides*
Pine	*Pinus spp.*
Piper’s daisy	*Erigeron piperianus*
Poverty-weed	*Monolepis mittaliana*
Prickly pear cactus	*Opuntia spp.*
Rabbitbrush	*Chrysothamnus spp.*
Rush	*Juncus spp.*
Russian thistle	*Salsola kali*
Sagebrush	*Artemisia spp.*
Saltbush	*Atriplex spp.*
Saltgrass	*Distichlis spicata*
Sandberg’s bluegrass	*Poa sandbergii*
Snow buckwheat	*Eriogonum niveum*
Spike rush	*Eleocharis spp.*
Spiny hopsaga	*Grayia spinosa*
Thickspike wheatgrass	*Agropyron dasytachyum*
Threetip sagebrush	*Artemisia tripartita*
Thyme buckwheat	*Eriogonum thymoides*

Common Name	Scientific Name
Plants (Continued)	
Tumble mustard	*Sisymbrium altissimum*
Utah juniper	*Juniperus osteosperma*
Ute ladies'-tresses	*Spiranthes diluvialis*
Watercress	*Rorippa nasturtium-aquaticum*
Western wheatgrass	*Agropyron smithii*
White mulberry	*Morus alba*
White pine	*Pinus strobus*
Willow	*Salix spp.*
Winterfat	*Eurotia lanata*
Yarrow	*Achillea millefolium*

Appendix N
The Public Scoping Process

N.1 Scoping Process Description

An early step in the development of a programmatic environmental impact statement (PEIS) following the guidance and requirements contained in the Council on Environmental Quality and the Department of Energy (DOE) regulations is to initiate "an early and open process for determining the scope of the issues to be addressed and for identifying the significant issues related to the proposed action." Major purposes of this scoping process include (1) informing the public about the proposed action; (2) obtaining input from the public and other concerned entities on significant issues that should be evaluated in this *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (*Nuclear Infrastructure Programmatic Environmental Impact Statement* [NI PEIS]); (3) identifying and eliminating from detailed study issues that are not significant or have been addressed by other environmental reviews; (4) determining appropriate roles for lead and cooperating agencies, as needed; (5) identifying other environmental review and consultation requirements; and (6) indicating how the preparation of this NI PEIS relates to the agency's planning and decision-making processes (40 CFR Section 1501.7). As shown in **Figure N–1**, the scoping process is one of the opportunities for public involvement required as part of the National Environmental Policy Act (NEPA) process.

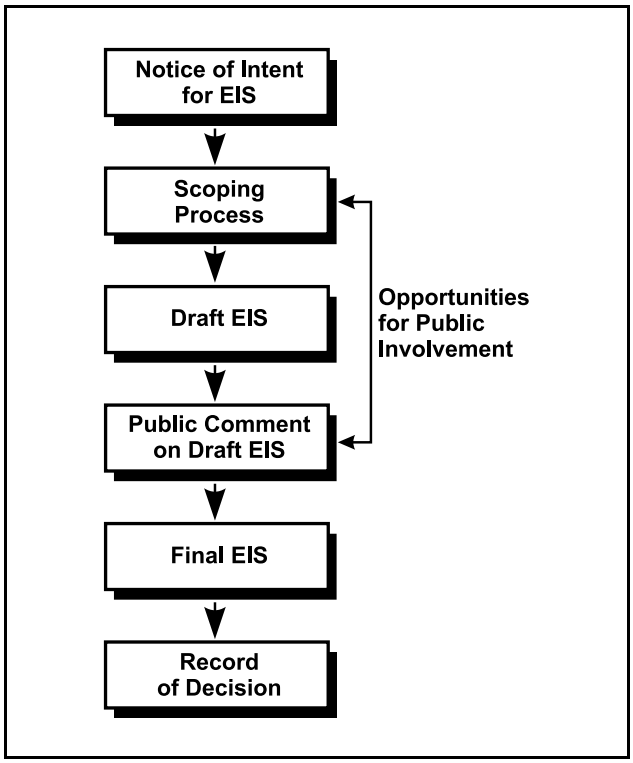


Figure N–1 NEPA Process

On October 5, 1998, DOE published a Notice of Intent in the Federal Register (63 FR 53398) to prepare an environmental impact statement (EIS) on the proposed production of plutonium-238 for use in advanced radioisotope power systems for future space missions. With that announcement, DOE began preparing the *Environmental Impact Statement for the Proposed Production of Plutonium-238 for Use in Advanced Radioisotope Power Systems for Future Space Missions* (*Plutonium-238 Production EIS*). The scope of the *Plutonium-238 Production EIS* was established through a public scoping process conducted from November 4, 1998, through January 4, 1999, as shown in **Table N–1**. As part of the scoping process for that EIS, DOE announced that the Fast Flux Test Facility (FFTF) would not be considered a reasonable alternative for the plutonium-238 production mission unless restart of the facility was proposed for other reasons.

The Secretary of Energy subsequently announced on August 18, 1999, that DOE would prepare this NI PEIS. Because plutonium-238 production would be among the missions considered in this NI PEIS, the scope of the *Plutonium-238 Production EIS* in its entirety was incorporated within the scope of this NI PEIS, and preparation of the *Plutonium-238 Production EIS* as a separate NEPA review was terminated. As such, DOE reviewed and considered all comments originally received from the public during the *Plutonium-238 Production EIS* scoping period in the preparation of this NI PEIS.

Table N–1 Schedule of Public Scoping Meetings

Location	Date	Approximate Number of Attendees
Plutonium-238 Production EIS Scoping Meetings—1998		
Idaho Falls, Idaho	November 12	20
Oak Ridge, Tennessee	November 17	30
Richland, Washington	November 19	200
	Total	250
NI PEIS Scoping Meetings—1999		
Oak Ridge, Tennessee	October 13	30
Idaho Falls, Idaho	October 15	20
Seattle, Washington	October 18	200
Portland, Oregon	October 19	300
Hood River, Oregon	October 20	300
Richland, Washington	October 21	200
Washington, D.C.	October 27	60
	Total	1,110

On September 15, 1999, DOE published a Notice of Intent in the Federal Register (64 FR 50064) to prepare this NI PEIS. DOE held scoping meetings in October 1999, as shown in Table N–1. This Notice of Intent initiated the 45-day scoping period. Based on comments from participants who attended the *Plutonium-238 Production EIS* scoping meetings, additional scoping meetings were held in the Pacific Northwest related to this NI PEIS. The Pacific Northwest meeting locations included Seattle and Richland, Washington, and Portland and Hood River, Oregon. Meetings were held in Oak Ridge, Tennessee, and Idaho Falls, Idaho, and a meeting was also added in Washington, D.C., to accommodate national interest groups and others.

As a result of previous experience and positive responses from attendees of other DOE/NEPA public meetings and hearings, DOE chose an interactive format for the scoping meetings. Each meeting began with a presentation by the DOE EIS Document Manager, who explained the proposed action and planned scope of the subject PEIS, and answered questions on the presentation. Afterwards, an impartial facilitator opened the floor for comments and concerns from the audience. DOE, national laboratory staff, and contractor personnel were available to respond to the questions and comments, as needed. A verbatim transcript of public comments and DOE responses was prepared by court reporters who attended each of the meetings. The transcripts were made available to the public at DOE reading rooms and on the Internet at www.nuclear.gov. Reading room locations are shown in **Table N–2**. Written statements or comments from the public were collected at the meetings. In addition, the public was encouraged to submit comments to DOE by U.S. mail, electronic mail, a toll-free fax line, and a toll-free telephone line until the end of the scoping comment period. Comments received after that date were considered to the extent practicable.

The comments received during the *Plutonium-238 Production EIS* and NI PEIS scoping periods are described in the following sections.

Table N–2 Public Information Centers

Location	Address
Idaho	U.S. Department of Energy Public Reading Room 1776 Science Center Drive Idaho Falls, ID 83415 (208) 526-1144
Oregon	Portland State University Bradford Price Miller Library Government Documents Section 951 Southwest Hall Portland, OR 97207 (502) 725-3690
Tennessee	U.S. Department of Energy Public Reading Room Building 1916-T-2, Suite 300 230 Warehouse Road Oak Ridge, TN 37830 (865) 241-4780
Washington	U.S. Department of Energy Public Reading Room Consolidated Information Center, Room 101L 2770 University Drive Richland, WA 99352 (509) 372-7443 Gonzaga University Foley Center Library East 502 Boone Spokane, WA 99258 (509) 323-6532 Richland Public Library 955 Northgate Drive Richland, WA 99352 (509) 942-7457 University of Washington Suzzallo Library Government Publications Room Box 352900 Seattle, WA 98195 (206) 543-1937
Washington, D.C.	U.S. Department of Energy Freedom of Information Public Reading Room Forrestal Building, Room IE-190 1000 Independence Ave, SW Washington, DC 20585 (202) 586-3142

N.1.1 Plutonium-238 Production EIS Scoping Comments

DOE received approximately 750 comments from about 245 citizens, interested groups, and Federal, state, and local officials during the public scoping period for the *Plutonium-238 Production EIS*. About 220 of these comments were presented orally during the public meetings. Approximately half of all the commentors (117) were supportive of the proposed plutonium-238 production mission, 105 were opposed to plutonium-238 production, and 23 commentors were neither for nor against the proposal.

In general, the people who attended the meetings in Idaho and Tennessee were supportive of DOE's proposed plans to produce plutonium-238 domestically for future space missions. In Richland, Washington, the meeting was attended by representatives of several stakeholder and environmental groups who voiced opposition to DOE's consideration of using FFTF for plutonium-238 production.

Other main concerns of the Richland, Washington, meeting participants were that DOE should not consider restarting FFTF, that DOE has worked hard over the years to change Hanford's mission from "production" to "cleanup," and that DOE should continue to honor its commitment to cleanup. There were concerns about the generation of additional waste streams at the site and the operational safety of FFTF. There was strong opposition to the restart of FFTF for any mission.

Of all the comments submitted during the scoping period, approximately 320 concerned the Hanford Site. Of these Hanford-specific comments, there were roughly equal numbers of comments supporting and opposing the use of FFTF and the Fuels and Materials Examination Facility for plutonium-238 production. The major concern of those in opposition was that DOE should not consider restarting FFTF and should continue to honor DOE's commitment to the cleanup at Hanford, rather than introducing a new production mission. The safety of FFTF, potential for further contamination, and generation of additional waste streams were concerns expressed by commentors.

About 65 of the comments concerned the use of plutonium-238 in space, or the National Aeronautics and Space Administration (NASA) Space Program in general. The majority of the space-related comments were in opposition to the use of plutonium-238 in space because of concerns about the safety of radioisotope power systems.

Approximately 45 of the comments addressed issues related to waste management, including concerns about the amount of waste generated by the program, the processing and final disposition of all generated wastes, and the impact of additional wastes on the ongoing environmental cleanup programs at the sites.

The cost of the plutonium-238 production program was another concern expressed by commentors. About 45 comments included requests for information related to the costs of each alternative. Several commentors also proposed additional alternatives for consideration in the EIS.

The following general issues and concerns are highlights of the comments made.

- It was suggested that additional irradiation service alternatives such as commercial light water reactors and accelerators should be considered.

- It was suggested that additional storage, target fabrication, and target processing alternatives should be considered, such as the Hot Fuel Examination Facility at Argonne National Laboratory–West and the H-Canyon and HB-Line at the Savannah River Site.

- Concerns were voiced about additional waste streams being generated, including what the waste streams would be and how they would be disposed. Concerns were also related to the generation and disposal of any transuranic waste since non-defense-related wastes are not eligible for disposal at the Waste Isolation Pilot Plant.

- It was suggested that the cost of implementing the various alternatives should be analyzed, and a cost breakdown for each alternative should be provided. Costs of concern that were mentioned included equipment and facility modifications, annual operating expenses, transportation, appropriate waste

storage and final disposal. Commentors were concerned about who would bear the costs of plutonium-238 production.

The scope of the proposed *Plutonium-238 Production EIS* was incorporated within the scope of this NI PEIS. DOE has considered all comments originally received from the public during the *Plutonium-238 Production EIS* scoping period in preparing this NI PEIS.

N.1.2 NI PEIS Scoping Comments

The written and oral comments received at the scoping meetings and the additional comments received via U.S. mail, electronic mail, and toll-free faxes and telephone calls during the public comment period were reviewed and considered by DOE in preparing this NI PEIS, along with all comments and input originally received from the public during the *Plutonium-238 Production EIS* scoping period.

In addition to the comments received on the *Plutonium-238 Production EIS*, DOE received approximately 7,000 comments from citizens, interest groups, and other stakeholders during the scoping comment period for this NI PEIS. Campaigns including the same or similar submittals made by multiple people accounted for more than 4,300 comments. Other submittals accounted for more than 2,600 comments. Of the comments received, there were more than 700 unique comments. In some cases, a commentor provided similar or identical comments both orally at the scoping meetings and in writing, so their comments were duplicated.

The comments obtained by DOE throughout the scoping process addressed several key issues. At the scoping meetings on this NI PEIS, the most prevalent concerns are shown below.

- The status of, and commitment to, cleanup at Hanford and the impact of FFTF restart on the existing waste problem.
- There was concern about the lack of justification for the identified missions.
- Commentors requested information about the cost of implementing the various alternatives.
- Commentors repeatedly emphasized the need for an additional alternative calling for the permanent deactivation of FFTF coupled with the No Action Alternative elements, that is, no plutonium-238 production and no additional research and development or medical isotope production beyond existing operating facility capabilities.

As shown in Table N–1, the number of people who commented at the scoping meetings conducted in Oak Ridge, Tennessee; Idaho Falls, Idaho; and Washington, D.C., was smaller in comparison to the meetings held in the Pacific Northwest. At the scoping meeting in Oak Ridge, Tennessee, a commentor was concerned with the relationship of this NI PEIS to other DOE programs and the relative merits of accelerator and reactor performance. The commentor stated the PEIS should include an explanation of mixed oxide fuel disposition. In addition, the commentor supported medical isotope production in Oak Ridge because it is near a transportation hub and some medical isotopes are short-lived; therefore, transportation is key.

At the scoping meetings in Idaho Falls, Idaho, most commentors supported the new missions at the Idaho National Engineering and Environmental Laboratory. The commentors also stated that the socioeconomic impacts of the alternatives need to be considered in the NI PEIS. A commentor stated that decisions in regard to medical isotope production should be based on the needs of the Nation as a whole and not on perceived commercial needs. The commentor also stated that incremental DOE and commercial investments in the

Advanced Test Facility would be sufficient to enhance reactor radioisotope production needs and meet the requirements of the nuclear medicine industry.

At the meetings held in the States of Washington and Oregon, the comments frequently were about FFTF, either supporting or opposing the use of FFTF to accomplish the proposed missions. Many of the commentors who attended the meetings in Seattle, Washington; Portland, Oregon; and Hood River, Oregon, were strongly opposed to the restart of FFTF. As a result of these public comments, DOE added a fifth alternative: the shutdown of FFTF without new production missions. Many commentors stated that the Hanford cleanup mission would be jeopardized, especially when DOE has not met Hanford cleanup milestones.

Most of the comments received at the Richland, Washington, meeting were in support of restarting FFTF. Supporters said restart would not hamper Hanford's cleanup mission, and stated that operation of FFTF could help save the lives of many people by producing isotopes to be used in new ways to treat cancer, heart disease, and other illnesses. Commentors were also concerned about the potential generation of radioactive and hazardous waste as a result of the proposed missions. Commentors stated that DOE should analyze projected waste streams, including health and safety risks and the cumulative impacts to the environment, as well as analyzing impacts from additional spent nuclear fuel storage.

In general, many commentors stated that the PEIS should include a detailed cost analysis of each alternative, including total life-cycle costs. Some commentors stated that DOE has a substantial investment of public resources and money and every effort needs to be made to use FFTF. Others opposed the use of FFTF as a waste of money and stated that keeping FFTF on standby is expensive.

At the scoping meeting in Washington, D.C., the commentors supported the need for medical isotope production. Several commentors were against the restart of FFTF and others stated that DOE needs to consider partnerships with private industry to generate necessary funds for restart. Some commentors thought that a cost study should be prepared and should include avoided future health care costs and cost savings to the national Medicare and Medicaid programs that could be realized by using nuclear isotopes in medical applications. Proliferation concerns were also raised as some commentors stated that: (1) the United States would be sending the wrong message by restarting FFTF; (2) a change in the U.S. nonproliferation policy would be required to import German mixed oxide fuel; and (3) the use of highly enriched uranium would be contrary to existing U.S. nonproliferation policy. Other concerns included waste generation, Hanford cleanup, and safety at FFTF.

N.2 HOW COMMENTS WERE HANDLED

As comments were received during both the *Plutonium-238 Production EIS* and NI PEIS scoping periods, they were logged in, assigned a unique document identification number and placed into the Administrative Record. Comments were identified and assigned to the general categories that are shown in **Table N–3**. By grouping the comments in these categories, DOE representatives were aided in deciding which comments were within or outside the scope of this NI PEIS and which comments needed to be added to the scope of this NI PEIS.

Once the comments were grouped into the categories of comments, they were all reviewed and taken into consideration in the preparation of this Draft NI PEIS. DOE received many comments that were found to be within the scope of this NI PEIS and these comments are addressed in the document. Comments that were added to the scope of this NI PEIS and those that were determined to be outside of the scope of this NI PEIS are summarized in the following sections. All comments were placed in this NI PEIS project Administrative Record.

Table N–3 Categories of Comments

Purpose and need, including medical isotopes, plutonium-238 production, research and development, general irradiation needs, and commercial isotopes
Site-specific recommendations and alternative selection, which included the No Action Alternative, Alternatives 1 through 4 that were included in the Notice of Intent, Alternative 5 that was added as a result of the scoping process, and other general alternative issues
EIS process, including the adequacy of the process, public involvement, the availability of information, the need for additional scoping meetings, and the extension of the comment period
Impacts to the environment in the areas of geology and soils, socioeconomics, environmental justice, cumulative impacts, and general environmental contamination from the proposed action
Public and occupational safety and health
Waste and spent nuclear fuel generation, storage, and disposition
Transportation
Decontamination and decommissioning of facilities
Applicable laws, regulations, and other requirements, including general issues, licensing of facilities, and external regulations of facilities
Costs, including FFTF investment and the cost of each alternative
Other miscellaneous issues, including the cleanup of Hanford, the suggestion that no new missions would be appropriate for Hanford, and antinuclear and non-proliferation-related comments

N.3 COMMENTS THAT WERE ADDED TO THE SCOPE OF THIS NI PEIS

Some comments were received during both the *Plutonium-238 Production EIS* and NI PEIS scoping periods that DOE representatives added to the scope of this NI PEIS. These comments are highlighted in the following sections.

N.3.1 Plutonium-238 Comments Added

- Commentors were opposed to any new missions or operations at Hanford that would generate additional nuclear waste. Information has been added to this NI PEIS addressing potential waste generation and other environmental impacts associated with the proposed missions.

- Commentors proposed the use of non-DOE facilities for irradiation services. The addition of non-DOE facilities, such as a commercial light water reactor, is an existing radiation source that DOE will consider to meet the requirements of the plutonium-238 production mission. The ability of the commercial light water reactor to meet the requirements of the other proposed isotope production missions is addressed in this NI PEIS.

- Commentors requested additional meetings in Washington and Oregon at the *Plutonium-238 Production EIS* scoping meetings in November 1998. The public meetings for this NI PEIS were expanded to include Seattle, Washington; Portland, Oregon; and Hood River Oregon, in addition to the Richland, Washington, meeting.

N.3.2 NI PEIS Comments Added

- Commentors strongly suggested that DOE shut down FFTF. In response to public comment, DOE has added to the NI PEIS a new alternative (Alternative 5) to permanently deactivate FFTF, with no new missions at Hanford.

- Commentors were concerned about the restart of FFTF and budget constraints. DOE has made a commitment that implementation of the Record of Decision will not divert or reprogram budgeted funds designated for Hanford cleanup, regardless of the alternative(s) selected.

- Commentors stated that DOE should adhere to the Tri-Party Agreement. DOE has completed more than 900 Tri-Party Agreement milestones and targets with more than a 98 percent on-time completion rate. The Tri-Party Agreement is a living document and is routinely modified in accordance with procedures defined by the agreement. DOE alone cannot modify the agreement. Implementation of any of the proposed missions at Hanford would not be in conflict with the land use plan or the Tri-Party Agreement.

- Commentors suggested that this NI PEIS should include information on the stewardship of FFTF. There would be no transfer of FFTF stewardship when it is in the standby, startup, or operating mode. This NI PEIS addresses the transition of FFTF stewardship after it is deactivated.

- Commentors suggested that European regulatory and government issues associated with the export of SNR–300 fuel be addressed in this NI PEIS. It is anticipated that a "Nuclear Infrastructure Nonproliferation Impacts Assessment" report would address the export issue.

N.4 OUT OF SCOPE COMMENTS

Some comments were received by DOE that were determined to be outside the scope of the environmental impact analyses of this NI PEIS. The major comments that fall into this group are summarized below for comments received related to both the *Plutonium-238 Production EIS* and this NI PEIS.

N.4.1 Plutonium-238 Out of Scope

- Commentors voiced both support and opposition to the NASA space program and the use of radioisotopes for space exploration in particular. This NI PEIS does not address the merits of the space program or the desirability of radioisotope use in space missions. Those questions fall under the purview of NASA. This NI PEIS is concerned only with the question of how to provide for the identified plutonium-238 requirements for space missions.

- Commentors requested additional information and analysis of the costs associated with the different alternatives. This NI PEIS is required by regulation to examine the environmental impacts of the identified alternatives. Cost information to support the Record of Decision is developed separate from this NI PEIS.

- Commentors voiced concern over the continued production of plutonium by the United States. This NI PEIS is limited to the examination of environmental impacts resulting from the production of plutonium-238 only. Plutonium-238 is not weapons-grade material. In fact, plutonium-238 is a contaminant in weapons-grade plutonium (plutonium-239) that degrades weapon performance.

N.4.2 NI PEIS Out of Scope

- Commentors requested additional analysis on medical isotope demand and the benefits associated with medical isotope use. This NI PEIS evaluates the environmental impacts of medical isotopes production given a projected production level. However, the demand and associated benefits are described in the report, *Expert Panel: Forecast Future Demand for Medical Isotopes* (Wagner et al. 1998).

- Commentors requested additional analysis of alternative power sources for space vehicles and the production requirements for plutonium-238. This NI PEIS evaluates the environmental impacts of producing plutonium-238 in the quantity required by the space program. NASA is responsible for evaluating alternative energy sources for space vehicles and determining the desirability of plutonium-238 usage for power.

- Commentors expressed both support and opposition to restarting FFTF and requested additional risk/benefit analysis and safeguards associated with the FFTF restart alternative. This NI PEIS evaluates the risks and environmental impacts associated with FFTF restart (and all other alternatives). The evaluation of benefits occurring from the implementation of any alternative is covered in separate documents and will be taken into consideration in developing the Record of Decision. The subject of additional safeguards for FFTF restart, that is, commitment to Nuclear Regulatory Commission oversight, is an operational issue to be considered only if FFTF restart is selected in the Record of Decision.

- Commentors requested additional analysis on the environmental effects of combining radioactive waste from alternative implementation and existing hazardous wastes, such as, pesticides. Some commentors voiced opposition to any additional radioactive waste production. This NI PEIS evaluates the environmental and health impacts of the wastes produced by the alternatives. Wastes produced from other activities, for example, pesticide application, are not required to be evaluated in this document. Additionally, this NI PEIS evaluates the impacts of wastes generated by the alternatives and identifies those alternatives that result in no additional waste produced.

- Commentors requested additional cost analysis for alternatives. This NI PEIS evaluates the environmental impacts associated with the implementation of the alternatives. Cost analysis is not required to be included in an EIS, although it would contribute to a decision made in the Record of Decision.

N.5 REFERENCES

Code of Federal Regulations

40 CFR Section 1501.7, "Scoping," Council on Environmental Quality.

Federal Register

63 FR 53398, U.S. Department of Energy, 1998, "Notice of Intent to Prepare an Environmental Impact Statement for the Proposed Production of Plutonium-238 for Use in Advanced Radioisotope Power Systems for Future Space Missions," October 5.

64 FR 50064, U.S. Department of Energy, 1999, "Notice of Intent to Prepare a Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (DOE/EIS-0310)," September 15.

Other

Wagner et al., 1998, *Expert Panel: Forecast Future Demand for Medical Isotopes*, presented in Arlington, VA, Medical University of South Carolina, Charleston, SC, September 25–26.

Appendix O Contractor Disclosure Statement
NEPA Disclosure Statement for the Preparation of a Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility

CEQ Regulations at 40 CFR 1506.5(c), which have been adopted by the DOE (10 CFR 1021), require contractors who will prepare an EIS to execute a disclosure specifying that they have no financial or other interest in the outcome of the project. The term "financial interest or other interest in the outcome of the project" for the purposes of this disclosure is defined in the March 23, 1981 guidance, "Forty Most Asked Questions Concerning CEQ's National Environmental Policy Act Regulations," 46 FR 18026–18038 at Question 17a and b.

"Financial or other interest in the outcome of the project 'includes' any financial benefit such as a promise of future construction or design work in the project, as well as indirect benefits the contractor is aware of (e.g., if the project would aid proposals sponsored by the firm's other clients)." 46 FR 18026–18038 at 18031.

In accordance with these requirements, the offeror and any proposed subcontractors hereby certify as follows: (check either (a) or (b) to assure consideration of your proposal).

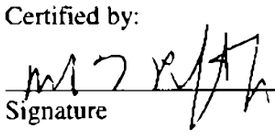
(a) ✔ Offeror/subcontractor have no financial interest in the outcome of the project.

(b) ___ Offeror and any proposed subcontractor have the following financial or other interest in the outcome of the project and hereby agree to divest themselves of such interest prior to award of this contract.

Financial or Other Interests:

1.
2.
3.

Certified by:



Signature

Richard T. Profant
Name

Corporate Vice President

7/5/00
Date

Science Applications International Corporation

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Financial or Other Interests:

1.
2.
3.

Certified by:

Signature

Dick Trevillian
Name

Vice President

7/3/00
Date

Parallax, Inc.

CEQ Regulations at 40 CFR 1506.5(c), which have been adopted by the DOE (10 CFR 1021), require contractors who will prepare an EIS to execute a disclosure specifying that they have no financial or other interest in the outcome of the project. The term "financial interest or other interest in the outcome of the project" for the purposes of this disclosure is defined in the March 23, 1981 guidance, "Forty Most Asked Questions Concerning CEQ's National Environmental Policy Act Regulations," 46 FR 18026–18038 at Question 17a and b.

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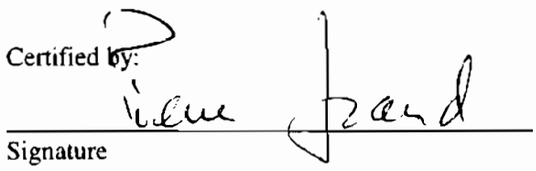
Financial or Other Interests:

1.
2.
3.

Certified by:



Signature

Pierre Grand
Name

President

July 5th 2000
Date

TECHSOURCE, Incorporated

CEQ Regulations at 40 CFR 1506.5(c), which have been adopted by the DOE (10 CFR 1021), require contractors who will prepare an EIS to execute a disclosure specifying that they have no financial or other interest in the outcome of the project. The term "financial interest or other interest in the outcome of the project" for the purposes of this disclosure is defined in the March 23, 1981 guidance, "Forty Most Asked Questions Concerning CEQ's National Environmental Policy Act Regulations," 46 FR 18026–18038 at Question 17a and b.

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(b) ___ Offeror and any proposed subcontractor have the following financial or other interest in the outcome of the project and hereby agree to divest themselves of such interest prior to award of this contract.

Financial or Other Interests:

1.
2.
3.

Certified by:

Paul Hogroian

Signature

Paul Hogroian, P.E.
Name

President

July 3, 2000
Date

Project Engineering and Management, Inc.

Appendix P
Nuclear Infrastructure Cost Report Summary

Cost Report for Alternatives
Presented in the

Draft Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility

August 2000

SUMMARY

S.1 INTRODUCTION AND BACKGROUND

The following is a summary of a report evaluating the costs associated with the U.S. Department of Energy (DOE) proposal to enhance its existing nuclear facility infrastructure to accommodate new and expanding missions in the areas of nuclear research and development and isotope production. DOE currently does not have sufficient steady-state irradiation sources to meet the Nation's projected needs for: (1) isotopes for medical and industrial uses, (2) fuel to power future U.S. National Aeronautics and Space Administration (NASA) spacecraft, and (3) nuclear research and development.

The alternatives for the proposed expanded isotope production missions that were evaluated in this Cost Report are presented in the *Draft Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (Nuclear Infrastructure Programmatic Environmental Impact Statement [NI PEIS]) (DOE 2000).

Costs of potential decisions are not typically evaluated in an environmental impact statement (EIS), but DOE recognizes that the financial implications of its future programs are important considerations for decision making and has resolved to inform the public about those costs. The findings of this Cost Report and public input received on the NI PEIS are among the factors that DOE will consider when preparing the Record of Decision.

The programmatic alternatives considered in this Cost Report focus on the use of irradiation facilities that are currently operating, could be brought online, or could be constructed and operated to meet DOE's irradiation needs. Thus, the report considers the following alternatives (presented in more detail in Chapter 2 of the NI PEIS):

- **No Action Alternative**, maintaining the status quo; that is, DOE's existing facilities would continue to meet their current mission requirements within their operating levels, and DOE would not enhance existing U.S. nuclear facility infrastructure or expand its current missions to accommodate new missions.

- **Alternative 1**, which includes resuming operation of the Fast Flux Test Facility (FFTF) at the Hanford Site (Hanford) in Richland, Washington

- **Alternative 2**, using only existing operational facilities (the Advanced Test Reactor [ATR] at Idaho National Engineering and Environmental Laboratory [INEEL], the High Flux Isotope Reactor [HFIR] at Oak Ridge National Laboratory [ORNL], or a generic commercial light water reactor [CLWR]) to accommodate the plutonium-238 production mission

- **Alternative 3**, constructing and operating one or two new accelerator(s) at an existing DOE site

- **Alternative 4**, constructing and operating a new research reactor at an existing DOE site

- **Alternative 5**, permanently deactivate Hanford's FFTF without enhancing U.S. nuclear facility infrastructure to accommodate new or expanded missions. Although Alternatives 2, 3, and 4 include the deactivation of FFTF, Alternative 5 is included as a stand-alone alternative in response to numerous public comments received during the scoping period for the NI PEIS.

The No Action Alternative and Alternatives 1 through 4 each have several options, evaluated in this Cost Report. These options involve primarily DOE facilities that could be used for fabrication, storage, and postirradiation processing of the targets necessary for the program missions. Among the facilities proposed are: (1) the Radiochemical Engineering Development Center (REDC) at ORNL, (2) the Fluorinel Dissolution Process Facility (FDPF) and/or the Chemical Processing Plant (CPP) Building 651 (CPP–651) (storage only) at INEEL, (3) the Fuels and Materials Examination Facility (FMEF) at Hanford, (4) Building 325, the Radiochemical Processing Laboratory (RPL), and Building 306–E at Hanford, and (5) a new facility to be constructed and operated at an existing DOE site to support the one or two new accelerator or new research reactor alternatives. **Table S-1** presents an overview of the alternatives and options evaluated in the NI PEIS.

S.2 DECISIONS TO BE MADE

In reaching programmatic decisions regarding potential expansion of its existing nuclear facility infrastructure, DOE will factor the analytical environmental results of the NI PEIS together with the findings presented in this Cost Report and the NI Nonproliferation Impacts Assessment[1], the *Nuclear Science and Technology Infrastructure Roadmap*, recommendations of the Nuclear Energy Research Advisory Committee (NERAC) and its various subcommittees, public input, and other DOE policy and programmatic considerations.

With the benefit of this broad base of information, DOE intends to make the following decisions:

- Whether to expand its current nuclear facility infrastructure to meet projected requirements for future medical and industrial isotope production, plutonium-238 production, and nuclear research and development.

- If a decision is made to expand DOE's existing nuclear facility infrastructure, whether to (1) construct new facilities (one or two accelerators or a research reactor), or (2) restart FFTF at Hanford as part of a nuclear infrastructure expansion program and, if not, whether to remove FFTF from standby mode and permanently deactivate it in preparation for its eventual decontamination and decommissioning.

- If a decision is made not to expand DOE's existing nuclear facility infrastructure, decide whether to (1) select from existing operating facilities those needed to support the proposed plutonium-238 mission, or (2) continue purchasing plutonium-238 from Russia to support future NASA space missions, and (3) whether DOE inventories of neptunium-237 should be relocated and stored for future plutonium-238 production needs. Existing operating facilities performing medical, research, and/or industrial isotope production and/or nuclear research and development missions would continue to support existing missions at current levels.

The programmatic decisions to be made in association with the NI PEIS are the responsibility of the DOE Office of Nuclear Energy, Science and Technology. In addition to the range of reasonable programmatic alternatives evaluated in the NI PEIS, DOE could choose to combine components of several alternatives in selecting the most appropriate strategy. For example, DOE could select a low-energy accelerator to produce medical, research, and industrial isotopes, and an existing operating reactor to produce plutonium-238 and conduct nuclear research and development. If alternatives were selected involving the siting, construction, and operation of one or two new accelerators or a new research reactor, appropriate site- and project-specific National Environmental Policy Act (NEPA) documentation, tiered from the NI PEIS, would be prepared.

[1]The DOE Office of Arms Control and Nonproliferation is analyzing the nonproliferation policy impacts of FFTF's restart, and of the other alternatives and their various options, and will be reporting its findings in the *Nonproliferation Impacts Assessment for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (Nuclear Infrastructure Nonproliferation Impacts Assessment).

Table S–1 Alternatives and Options Evaluated in the NI PEIS

			Plutonium-238 Production Mission		*Medical and Industrial Isotopes Production and Nuclear Research and Development Mission*	
	Option Number	*Irradiation Facility*	*Storage Facility*	*Target Fabrication and Processing Facility*	*Storage Facility*	*Target Fabrication and Processing Facility*
No Action Alternative	1	–	–	–	–	–
	2	–	REDC	–	–	–
	3	–	CPP–651	–	–	–
	4	–	FMEF	–	–	–
Alternative 1: Restart FFTF	1	FFTF [a]	REDC	REDC	RPL/306–E	RPL/306–E
	2	FFTF [a]	FDPF/CPP–651	FDPF	RPL/306–E	RPL/306–E
	3	FFTF [a]	FMEF	FMEF	FMEF	FMEF
	4	FFTF [b]	REDC	REDC	RPL/306–E	RPL/306–E
	5	FFTF [b]	FDPF/CPP–651	FDPF	RPL/306–E	RPL/306–E
	6	FFTF [b]	FMEF	FMEF	FMEF	FMEF
Alternative 2: Use Only Existing Operational Facilities	1	ATR	REDC	REDC	–	–
	2	ATR	FDPF/CPP–651	FDPF	–	–
	3	ATR	FMEF	FMEF	–	–
	4	CLWR	REDC	REDC	–	–
	5	CLWR	FDPF/CPP–651	FDPF	–	–
	6	CLWR	FMEF	FMEF	–	–
	7	HFIR and ATR	REDC	REDC	–	–
	8	HFIR and ATR	FDPF/CPP–651	FDPF	–	–
	9	HFIR and ATR	FMEF	FMEF	–	–
Alternative 3: Construct New Accelerator(s)	1	New	REDC	REDC	New [c]	New [c]
	2	New	FDPF/CPP–651	FDPF	New [c]	New [c]
	3	New	FMEF	FMEF	New [c]	New [c]
Alternative 4: Construct New Research Reactor	1	New	REDC	REDC	New [c]	New [c]
	2	New	FDPF/CPP–651	FDPF	New [c]	New [c]
	3	New	FMEF	FMEF	New [c]	New [c]
Alternative 5: Permanently Deactivate FFTF (with no new missions)	–	–	–	–	–	–

Key: RPL/306-E = Radiochemical processing Laboratory and Hanford 300 Area Building 306-E.

a. Hanford FFTF would start up and operate with onsite and German mixed oxide (MOX) fuel and then highly enriched uranium (HEU) fuel.

b. Hanford FFTF would start up and operate with only the onsite MOX fuel and then HEU fuel.

c. The new facility would not be required if a DOE site with available support capability and infrastructure is selected.

The programmatic decisions to be reached in association with the NI PEIS are schematically presented in **Figure S–1.** In accordance with the first-tier "yes or no" decision to be made (as seen in Figure S–1), alternatives analyzed in the NI PEIS were arranged into two groups—nonexpanded infrastructure alternatives, including the No Action Alternative and Alternatives 2 and 5; and expanded infrastructure alternatives, including Alternatives 1, 3, and 4. Cost estimates for the nonexpanded and expanded infrastructure alternatives were also arranged into these groups and are presented in Section S.3, Results and Conclusions.

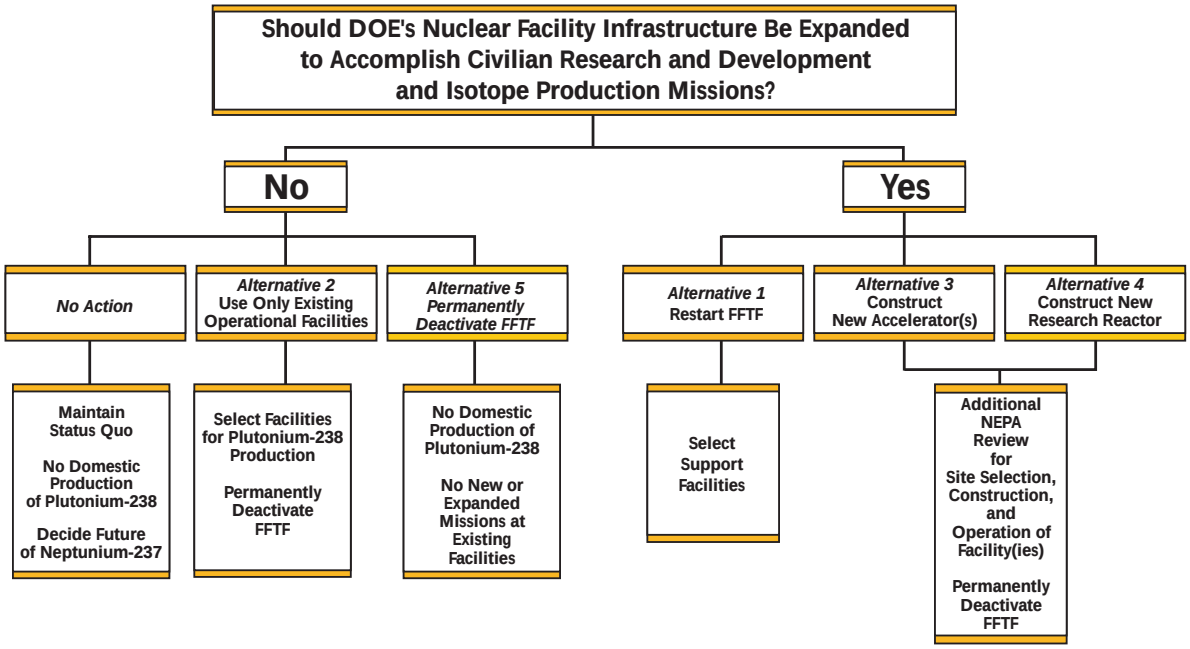

Figure S–1 Pending Decisions

S.3 RESULTS AND CONCLUSIONS

Summaries of cost estimates for the nonexpanded and expanded infrastructure alternatives identified in Figure S–1 are presented in **Tables S–2** and **S–3**. All figures shown represent millions of FY 2000 dollars. No credit was taken for projected revenues from medical and industrial isotope sales, or from fees paid by domestic or international users of facilities.

Nonexpanded Infrastructure Alternatives

A summary of the estimated costs of the nonexpanded infrastructure alternatives (the No Action Alternative and Alternatives 2 and 5 of the NI PEIS) is presented in Table S–2. Capital costs (costs of modifying existing facilities), costs for permanently deactivating FFTF (where appropriate), annual operating costs, and transportation costs are presented for irradiation facilities and neptunium-237 storage and plutonium-238 processing facilities. In addition, costs for the purchase and transport of Russian plutonium-238 are presented. DOE would continue its medical and industrial isotope production and nuclear research and development activities of the current operating levels of existing facilities.

Table S–2 Summary of Estimated Costs of Nonexpanded Infrastructure Alternatives (Millions of FY 2000 Dollars)

Cost Elements	*No Action*	*Alternatives*												
		Alternative 2: Use Only Existing Operational Facilities												*Alternative 5: Deactivate FFTF*
		ATR			*CLWR*			*ATR and HFIR*						
Irradiation Facilities														
FFTF in standby mode (annual) (A)	40.8													
FFTF deactivation (B)		281.2			281.2			281.2						281.2
Startup; target development, testing, and evaluation (C)		2			20			3.5						
Operations (annual) (D)		8.1			5.1			8.1						
Russian Plutonium-238														
Purchase 5 kilograms (11 pounds) of Russian Plutonium-238 (annual)	8.7 [a]													
Transport Russian Plutonium-238 to LANL (annual) (E)	0.14													
Total Annual Costs	**8.84**													
Processing Facility Alternative Options	**1**	**2**	**3**	**4**	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	
Neptunium-237 Storage and Plutonium-238 Processing Facilities		**REDC**	**CPP-651**	**FMEF**	**REDC**	**FDPF**	**FMEF**	**REDC**	**FDPF**	**FMEF**	**REDC**	**FDPF**	**FMEF**	
Modification and startup costs (F)		16.9	2.12	19.3	51.2	37.2	72.8	55.1	41.2	72.8	51.2	37.2	72.8	
Operations (annual) (G)		1.5	1.5	2.6	7.8	6.7	15.3	10.8	9.7	18.3	7.8	6.7	15.3	
Medical and Industrial Isotope/Nuclear Research and Development Processing Facilities [b]														
Modification or construction and startup costs														
Operations (annual)														
Combined Estimated Costs														
Total Costs (B+C+F)	**0**	16.9	2.12	19.3	334.4	320.4	356	356.3	342.4	374	335.9	321.9	357.5	281.2
Annual Costs (A+D+E+G)	49.6	51.1	51.1	52.2	15.9	14.8	23.4	15.9	14.8	23.4	15.9	14.8	23.4	0
Plutonium-238 Production Transportation														
Neptunium-237 from SRS (total)		1.4	7.1	8.5	1.4	7.1	8.5	1.4	7.1	8.5	1.4	7.1	8.5	
Total annual plutonium-238 production shipping and handling costs					0.39	0.24	0.32	0.41	0.40	0.46	0.34	0.29	0.35	
Medical and Industrial Isotope Transportation (annual) [b]														

Key: LANL = Los Alamos National Laboratory; SRS = Savannah River Site.

a. Based on FY 2000 contract year eight, $1.74 million per kilogram × 5 kilograms. Succeeding year purchase price escalated at a contractual 3.5 percent per year for the remaining two years of the contract.

b. DOE would continue its medical and industrial isotope production and nuclear research and development activities at the current operating levels of existing facilities.

Note: Shaded areas indicate that no costs would be incurred under that alternative and/or option.

Table S–3 Summary of Estimated Costs of Expanded Infrastructure Alternatives (Millions of FY 2000 Dollars)

Cost Elements	*Alternatives*								
	Alternative 1: Restart FFTF			*Alternative 3: Construct New Accelerator(s)*			*Alternative 4: Construct New Research Reactor*		
Irradiation Facilities									
Modification or construction and startup, including target development, testing, and evaluation	314			1,096.0			312		
FFTF deactivation				281.2			281.2		
Total costs (A)	**314**			**1,377.2**			**593.2**		
Operations (annual) [a] (B)	58.9			45.1			25		
Processing Facility Alternative Options	**1 and 4 [b]**	**2 and 5 [b]**	**3 and 6 [b]**	**1**	**2**	**3**	**1**	**2**	**3**
Plutonium-238 Production Facilities	**REDC**	**FDPF**	**FMEF**	**REDC**	**FDPF**	**FMEF**	**REDC**	**FDPF**	**FMEF**
Modification and startup costs (C)	55.1	41.2	72.8	51.2	37.2	72.8	51.2	37.2	72.8
Operations (annual) (D)	10.8	9.7	18.3	7.8	6.7	15.3	7.8	6.7	15.3
Medical and Industrial Isotope/Nuclear Research and Development Processing Facilities	**RPL/306-E**		**FMEF**	**New Processing Support Facility**			**New Processing Support Facility**		
Modification or construction and startup costs (E)	29.4		36.8	71.1			71.1		
Operations (annual) (F)	12.1		12.9	23.3			23.3		
Combined Estimated Costs									
Total Costs (A+C+E)	398.5	384.6	423.6	1,499.5	1,485.5	1,521.1	715.5	701.5	737.1
Annual Operating Costs [c] (B+D+F)	81.8	80.7	90.1	76.2	75.1	83.7	56.1	55	63.6
Plutonium-238 Production Transportation									
Neptunium-237 from SRS (total)	1.4	7.1	8.5	1.4	7.1	8.5	1.4	7.1	8.5
Total annual plutonium-238 production shipping and handling costs	0.41	0.28	0.28	1.54	1.50	1.54	2.39	2.37	2.42
Medical and Industrial Isotope Transportation (annual)	0.73	0.73	0.73	0.73	0.73	0.73	0.73	0.73	0.73

Key: SRS = Savannah River Site; RPL/306-E = Radiochemical Processing Laboratory and Hanford 300 Area Building 306-E.

a. Annual operating costs are an average of FFTF operating costs using onsite mixed oxide fuel (MOX) = $56.2 million, German MOX fuel = $56.7, highly enriched uranium (HEU) fuel = $63.9 million.

b. Options 1, 2, and 3 assume FFTF would use onsite MOX, German MOX, and then HEU fuel during operations. Options 4, 5, and 6 assume FFTF would use onsite MOX and then HEU fuel during operations.

c. Alternative 1 annual operating costs include an average of the FFTF operating costs.

Note: Shaded area indicates that no costs would be incurred under that alternative cost element.

- Under the No Action Alternative, FFTF would be maintained in its current standby mode at a cost of $40.8 million per year. The No Action Alternative would also include the annual purchase of 5 kilograms (11 pounds) of Russian plutonium-238 at an assumed annual cost of $8.84 million per year. Additional costs would depend on which option is chosen under the No Action Alternative. Option 1 would only incur the cost of maintaining FFTF in standby and the purchase of plutonium-238 from Russia. Options 2, 3, or 4 would involve the transport of neptunium-237 from SRS to REDC, CPP-651, or FMEF for long-term storage (costing $17 to 19 million for storage modifications and startup at REDC and FMEF and $2 million at CPP-651, which has existing storage capacity). Annual operating costs at all three storage sites would be approximately $1.5 to 2.6 million per year. The total costs of transporting neptunium-237 from SRS to storage facilities is a function of distance and would vary from $1.4 million for transport to REDC to $7.1 to 8.5 million to CPP-651 or FMEF, respectively.

- Alternative 2 would combine the use of existing irradiation facilities (ATR, ATR in combination with HFIR, or a CLWR) with the choice of three processing facilities (REDC, FDPF, or FMEF) to provide nine different options for producing plutonium-238. FFTF would be deactivated at a cost of $281 million constituting the major cost element of all options under Alternative 2. In addition, the following costs would be incurred:

  – Processing facility modification costs would be about $37 million for FDPF; $51 million for REDC; and $73 million for FMEF (for the addition of most process flowsheet items of equipment, within existing plant and services) for Options 1, 2, 3, 7, 8, and 9. An additional cost of $4 million for additional facility modifications was estimated for REDC and FDPF to fabricate stainless steel targets for the CLWR under Options 4, and 5.

  – Processing facility operating costs would be about $7 to 8 million per year for REDC and FDPF and $15 million per year for FMEF for Options 1, 2, 3, 7, 8, and 9. An additional cost of $3 million was estimated for REDC, FDPF, and FMEF for the fabrication of stainless steel targets for the CLWR under Options 4, 5, and 6.

  – Irradiation charges would be $8 million per year for ATR and ATR in combination with HFIR, and $5 million per year for the CLWR.

  – Total transportation costs for the shipment of neptunium-237 from SRS to processing facilities would be the same as previously described for the No Action Alternative. Differences in annual plutonium-238 production shipping and handling costs between the options are due to distance, the location of the irradiation facility, and the number of shipments. All shipments to and from irradiation facilities under this alternative would be by commercial truck.

- Alternative 5 would involve the deactivation of FFTF, at a cost of $281 million.

The sum of all facility modification costs for the nonexpanded infrastructure alternatives would be $0 to 19 million for the No Action Alternative; $320 to 374 million for Alternative 2; and $281 million for Alternative 5. The sum of all annual facility operating costs (less transportation) for this program would be $50 to 52 million for the No Action Alternative; $15 to 23 million for Alternative 2; and $0 for Alternative 5.

Expanded Infrastructure Alternatives

A summary of the estimated costs of the expanded infrastructure alternatives (Alternatives 1, 3, and 4 of the NI PEIS) is presented in Table S–3. Capital costs (costs of either modifying existing facilities or constructing

new facilities), costs for permanently deactivating FFTF (where appropriate), annual operating costs, and transportation costs are presented for irradiation and processing facilities.

With respect to irradiation facilities, which constitute the major cost element of these alternatives, it can be seen that:

- Capital costs would be in the order of $300 million for Alternative 1 (FFTF restart) and Alternative 4 (construction of a new research reactor), and more than $1 billion for Alternative 3 (construction of new accelerators). An additional burden of $281 million would be placed on Alternatives 3 and 4 for FFTF deactivation costs because these alternatives involve the construction of new facilities. Alternative 1, FFTF restart, would not incur this cost.

- The estimated annual costs of operating the irradiation facilities would be: $25 million per year for the new research reactor in Alternative 4; $45 million per year for the accelerators in Alternative 3; and $59 to 64 million per year for FFTF in Alternative 1.

It can also be seen that the other types of facilities used in the expanded infrastructure alternatives (isotope processing facilities and support facilities that fabricate targets for irradiation and chemically process irradiated targets to recover, package, and ship isotopes) are specific to the production of either (1) plutonium-238, or (2) medical and industrial isotopes.

- Costs of modifying REDC, FDPF, or FMEF to support plutonium-238 production, together with startup costs, would range from $37 to 73 million. The lower end of this range of front-end costs represents investments in REDC and FDPF, which have been built. FMEF has not been fully equipped nor operated, and would therefore require the higher modification costs to bring this facility online. Similarly, the annual operating costs for these facilities, would range from about $7 to 18 million per year, due to the availability of shared resources that can reduce operating costs, compared to a nonoperating facility like FMEF. An additional cost of $4 million for additional facility modifications at REDC and FDPF and $3 million operating costs at REDC, FDPF, and FMEF was estimated for the fabrication of stainless steel targets for the FFTF under Alternative 1.

- The mission to produce medical and industrial isotopes and expand nuclear research and development capabilities would be supported by either the modification of existing operational facilities at Hanford under Alternative 1 (RPL/Building 306–E or FMEF) or the construction of a new facility supporting either new accelerators (Alternative 3) or a new research reactor (Alternative 4). The investment for modifications or construction and startup would amount to about $29 to 37 million for the Hanford facilities and $71 million for a newly constructed processing support facility. Annual operating costs would be lower for the two existing facilities compared to a new processing support facility ($12 to 13 million per year for RPL/Building 306–E or FMEF and $23 million per year for a new processing support facility).

Transportation costs for the expanded infrastructure alternatives would be higher for the plutonium-238 production mission than the medical and industrial isotope mission, due to distances traveled, (e.g., REDC at ORNL to FFTF at Hanford versus shipping to the nearest air freight terminal) the number of shipments, and the cost of secure shipments. Differences in annual plutonium-238 production shipping and handling costs between the three alternatives are due to the cost of secure transport versus commercial truck and the number of shipments. Under Alternative 1, commercial trucks would be used to transport neptunium targets between processing facilities and FFTF. Alternative 3 would have the fewest number of shipments but requires the use of secure transport. Alternative 4 would have the same number of shipments and nearly the same shipping and handling costs as Alternative 1, but would require the use of secure transport to ship fabricated neptunium-237 targets from processing facilities to the new research reactor. The difference in the total costs of shipping

neptunium-237 from the Savannah River Site (SRS) to plutonium-238 processing facilities is a function of distance from SRS. These costs would range from a low of $1.4 million per year for REDC to about $7 to 8 million per year for FDPF and FMEF. By comparison, transportation costs in medical and industrial isotope production (involving intrasite transfers of relatively small targets and offsite transfers to the nearest air freight terminal) would amount to $0.73 million per year for each alternative.

The sum of all facility modification costs in the expanded infrastructure alternatives would be $385 to 424 million for Alternative 1; $1,485 to 1,521 million for Alternative 3; and $702 to 737 million for Alternative 4. The sum of all annual facility operating costs (less transportation) would be $82 to 90 million per year for Alternative 1; $75 to 84 million per year for Alternative 3; and $55 to 64 million per year for Alternative 4.

S.4 RISK ANALYSIS OF COST ESTIMATES

Although several types of contingencies can be defined, in general, a contingency refers to the cost that must be added to a base estimate to account for "unknown" costs. Two broad types of contingencies have been identified by Los Alamos National Laboratory (LANL) in the conceptual design report for a high-energy tritium production linear accelerator (LANL 1997). The most common type of contingency is an allowance for indeterminates, such as uncertainties in time, materials, or equipment items which may have inadvertently been omitted from the estimate. It should also be noted that the quality of the design basis for the development of the cost estimate is often a determinant of the magnitude of this type of contingency (Peters and Timmerhaus 1991). The Contingencies and Uncertainties columns in **Table S–4** reflect these types of uncertainties. A second type of contingency, often termed "risk contingency," is particularly applicable to projects involving new technologies (e.g., projects which require the preparation of cost estimates while nuclear research and development is still in progress). This contingency covers the cost effects of unforeseen design changes, altered performance requirements, or major schedule delays due to developmental problems. The Technical Risk and Schedule Risk columns in Table S–4 are indicative of risk contingency considerations.

The contingencies listed in Table S-4 that apply to the costs of the alternatives can be considered under these definitions:

No Action Alternative—Alternative cost involves little or no contingencies, technical or schedule risk, as no action is being taken other than the purchase and transport of Russian plutonium-238 to LANL and transport of neptunium-237 from SRS to long-term storage facilities at either REDC, CPP-651, or FMEF. There is a high uncertainty regarding the future purchase price for Russian plutonium-238 that could significantly affect the current estimated cost of this alternative. The current estimate for the cost for purchasing Russian plutonium-238 assumed that the contract price would be extended using the negotiated annual escalation rate of 3.5 percent for the duration of the project planning period described in the NI PEIS. The contract for the purchase of Russian plutonium-238 is in year eight, with two years remaining (DOE 1997). Beyond the last two years of the contract, the future price of Russian plutonium-238 is unknown.

Alternative 1: Restart FFTF—This alternative uses existing facilities and proven technologies, which implies relatively low contingencies (in the order of 10 to 20 percent), which is customary for this type of operation. The potential exists for schedule delays in the neptunium-237 and medical and industrial isotope stainless steel target development for FFTF. The schedule risk is considered low, because it was assumed that neptunium-237 and medical and industrial isotope target development and testing would be accomplished during FFTF startup. However, some schedule risk would remain if stainless steel targets should fail during testing or not meet performance requirements during target evaluation prior to isotope production.

Table S–4 Risk Analysis of Cost Estimates

Alternatives	*Contingencies*	*Uncertainties*	*Technical Risk*	*Schedule Risk*	*Discussion*
No Action	Low range	High	None	Low	Uncertainty: cost of Russian plutonium-238
Alternative 1: Restart FFTF	Low range	Low	None	Low	Schedule risk: neptunium-237 and medical and industrial isotope target development
Alternative 2: Use Only Existing Operational Facilities					
ATR and HFIR	Low range	Low	None	Low	Existing technology
CLWR	Moderate range	Moderate	Low	High	Schedule risk: neptunium-237 target development. Uncertainties: proprietary irradiation services costs and unknown target development cost
Alternative 3: Construct New Accelerator(s)					
High-energy linear accelerator	High range	High	High	Very high	Contingency: factor associated with preconceptual design and target/blanket development. Uncertainty: technology in development for this application. Schedule risks: target/blanket shipping cask development and certification
Low-energy cyclotron accelerator	Low range	Low	None	Low	Proven technology
Alternative 4: Construct New Research Reactor	High range	Moderate	Low	Moderate	Contingency: factor associated with preconceptual design, capability risk. Schedule risk: neptunium-237 target development
Alternative 5: Deactivate FFTF	Low range	None	None	Low	None

Alternative 2: Use Only Existing Operational Facilities—This alternative should have a low contingency of 20 percent or less because of existing technology. This alternative presents no technological requirements for modifications to existing operational facilities for the production of isotopes or the use of new technologies.

CLWR use is considered a low technological risk because it is a proven technology and an ongoing operation. However, the schedule risk is considered high because of uncertainties associated with the development of neptunium-237 targets for a CLWR (i.e., neptunium-237 target development, testing, and evaluation would have to fit in with the CLWR refueling cycle). If the neptunium-237 target fails during testing or does not meet performance requirements during target evaluation, additional target testing could not occur until the next refueling cycle (generally, another 18 months). CLWR irradiation services costs are also uncertain due to the proprietary nature of the industry.

Alternative 3: Construct New Accelerator(s)—This alternative involves the use of high-energy linear accelerator technology for the production of neutrons via spallation for isotope production. This technology places Alternative 3 in an area of high technological and schedule risks, and of high contingency factors in several areas of component development for the application of high-energy linear acceleration for plutonium-238 production.

Conversely, low-energy cyclotron accelerator use for the production of medical and industrial isotopes is a low-cost, proven technology, is currently used commercially, and has little or no schedule risk.

Alternative 4: Construct New Research Reactor—This alternative involves the use of proven research reactor technology, which implies low risk; however, the very nature of the preconceptual design requires that a high level of contingency be applied to the construction cost estimate and operating costs. The schedule risk for neptunium-237 target development is considered moderate, because even though the new research reactor design is based on proven research reactor and fuel technologies, it is preconceptual. Like FFTF, it was assumed that neptunium-237 and medical and industrial isotope target development, testing, and evaluation would be accomplished during construction and startup of the new research reactor. Unlike the CLWR, targets can be pulled from the new research reactor core at any time during testing for evaluation.

Alternative 5: Deactivate FFTF—This alternative involves only the deactivation of the FFTF reactor, which is currently in standby mode; except for uncertainties associated with the disposal of the sodium coolant, the deactivation of FFTF poses little or no technological risk and has a low-cost contingency.

Appendix Q
Nuclear Infrastructure Nonproliferation Impact Assessment
Executive Summary

NUCLEAR INFRASTRUCTURE NONPROLIFERATION IMPACT ASSESSMENT

For Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility

September 2000

United States Department of Energy
Office of Arms Control and Nonproliferation

EXECUTIVE SUMMARY

ES-1 PURPOSE, SCOPE, AND OBJECTIVE

This document assesses the potential nonproliferation impacts that might result from U.S. Department of Energy (hereafter referred to as the Department or DOE) nuclear infrastructure improvements as proposed and described in the *Draft Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (DOE/EIS 0310D), July, 2000 (hereafter referred to as the Draft NI PEIS). The DOE Office of Arms Control and Nonproliferation has prepared this *Nuclear Infrastructure Nonproliferation Impact Assessment for Accomplishing Expanded Civilian Nuclear Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (hereafter referred to as the NI NIA). Together with the Draft NI PEIS and an associated cost report, both being prepared by the DOE Office of Nuclear Energy, this assessment is being made available to the public as part of the Department's decision-making process to evaluate nuclear infrastructure improvement alternatives.

The United States has an annual requirement for the production of radioisotopes needed for medical, industrial, and scientific applications. The Department has an obligation to supply Pu-238 thermal/power supplies to support currently scheduled and future NASA missions. Civil nuclear energy research and development (R&D) is also required to support future U.S. nuclear energy production, civil nuclear waste disposal, and possible nuclear science applications (*e.g.*, space reactors for future NASA missions). These programmatic needs, particularly those emanating from the projected growth rate in the use of medical isotopes and the continued requirement to produce isotopes for other applications (*e.g.*, Pu-238 for NASA missions), have led the Department to consider various infrastructure improvement alternatives, including the utilization of existing and new facilities.

The Department issued a Notice of Intent on September 15, 1999 to prepare a PEIS for specified alternatives to accomplish these nuclear infrastructure missions.[1] The Notice of Intent identified alternatives as follows: 1) resume Fast Flux Test Facility (FFTF) operation; 2) construct and operate a new research reactor at a generic DOE site; 3) construct and operate one or more new neutron-producing accelerators at a generic DOE site; or 4) meet these projected mission needs utilizing existing reactor and accelerator facilities (other than FFTF). The Draft NI PEIS assesses the environmental impact of all these alternatives, though not in precisely this order. Furthermore, the Draft NI PEIS also evaluates a No Action Alternative and a fifth alternative: permanently deactivate FFTF with no new missions at any U.S. facilities. This NI NIA will follow the same delineation of alternatives as the Draft NI PEIS to assess the nonproliferation impact of actions that are proposed in the Draft NI PEIS.

The objective of the NI NIA is to evaluate the relationship between the missions, facilities, alternatives and options as described in the Draft NI PEIS, and the body of U.S. Government nonproliferation policy, U.S. laws and regulations, and international agreements. Based on that evaluation, the NI NIA presents conclusions and recommendations regarding the nonproliferation merits and drawbacks of the various activities proposed in the Draft NI PEIS to assist the Secretary of Energy to render a Record of Decision following publication of the Final NI PEIS.

[1] "Notice of Intent To Prepare a Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility," 64 *Fed. Reg.* 50064, 1999.

This assessment is limited to an evaluation of the direct and reasonably implied nonproliferation impact of the activities proposed in the Draft NI PEIS. Mission necessity, safety, environmental impact, effectiveness, costs, and life-cycle economics of activities described in the Draft NI PEIS are not considered to be central to the nonproliferation analysis reported in the NI NIA.

ES-2 FACILITIES, ALTERNATIVES, AND OPTIONS

The facilities identified by the Department in the Draft NI PEIS are presented in Table ES-1. The irradiation facilities are described and evaluated in Sections 4 and 5 and target fabrication and processing facilities are described and evaluated in Section 6. The facility type, name, location, acronym assigned in the NI NIA, and the operational status of each facility is shown in the table.

Table ES-1. Facilities Identified in the Draft NI PEIS

Type	Name	Acronym	Location	Status
Irradiation	Fast Flux Test Facility	FFTF	Hanford, WA	Standby
	Advanced Test Reactor	ATR	Idaho National Energy and Environmental Laboratory (INEEL), ID	Operational
	High Flux Isotope Reactor	HFIR	Oak Ridge National Laboratory (ORNL), TN	Operational
	Commercial Light Water Reactor	CLWR	Existing CLWR site to be determined	Operational
	New High-Energy Accelerator New Low-Energy Accelerator	- -	Existing DOE site to be determined	- -
	New Research Reactor	-	Existing DOE site to be determined	-
Target Fabrication and Processing	Radiochemical Engineering Development Center	REDC	Oak Ridge National Laboratory (ORNL), TN	Operational
	Fluorinel Dissolution Process Facility CPP-651	FDPF CPP-651	Idaho National Energy and Environmental Laboratory (INEEL), ID	FDPF: Non-operational Available CPP-651: Operational
	Fuels and Materials Examination Facility	FMEF	Hanford, WA	Non-operational Available
	Radiochemical Processing Laboratory Building 306-E	RPL 306-E	Hanford, WA	Operational
	New Support Facility	-	Existing DOE site to be determined	-

Using the facilities identified above, the Department has defined five potential alternatives and a No-Action Alternative to accomplish the missions described above. Table ES-2 defines the five alternatives and enumerates the options under each alternative (*i.e.*, the facility variations within each alternative). Each alternative and option is evaluated in Section 8. Under the No Action Alternative (all options) and Alternative 5, Pu-238 is purchased from Russia to meet NASA program requirements. Furthermore, under all options in Alternatives 2, 3, 4 and 5, FFTF is permanently deactivated. FFTF standby/deactivation is covered by a previous NEPA action that is not evaluated in this assessment, but the standby/deactivation activity is covered as a special case under the comprehensive FFTF nonproliferation assessment given in Section 4.[2]

[2] *Environmental Assessment – Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington*, DOE/EIA-0993, May, 1995.

Table ES-2. Alternatives and Options Defined in the Draft NI PEIS

Alternatives	Options	Irradiation Facility	Pu-238 Production Mission		Medical and Industrial Isotope Production and Nuclear Energy Research and Development Mission	
			Storage Facility	*Processing Facility*	*Storage Facility*	*Processing Facility*
No Action Alternative[d, e]	1	-	-	-	-	-
	2	-	REDC	-	-	-
	3	-	CPP-651	-	-	-
	4	-	FMEF	-	-	-
Alternative 1: *Restart FFTF*[g]	1	FFTF[a]	REDC	REDC	RPL/306-E	RPL/306-E
	2	FFTF[a]	FDPF/CPP-651	FDPF	RPL/306-E	RPL/306-E
	3	FFTF[a]	FMEF	FMEF	FMEF	FMEF
	4	FFTF[b]	REDC	REDC	RPL/306-E	RPL/306-E
	5	FFTF[b]	FDPF/CPP-651	FDPF	RPL/306-E	RPL/306-E
	6	FFTF[b]	FMEF	FMEF	FMEF	FMEF
Alternative 2: *Use Only Existing Operational Facilities*[f]	1	ATR	REDC	REDC	-	-
	2	ATR	FDPF/CPP-651	FDPF	-	-
	3	ATR	FMEF	FMEF	-	-
	4	CLWR	REDC	REDC	-	-
	5	CLWR	FDPF/CPP-651	FDPF	-	-
	6	CLWR	FMEF	FMEF	-	-
	7	HFIR/ATR	REDC	REDC	-	-
	8	HFIR/ATR	FDPF/CPP-651	FDPF	-	-
	9	HFIR/ATR	FMEF	FMEF	-	-
Alternative 3: *Construct New Accelerators*[f, g, h]	1	New	REDC	REDC	New[c]	New[c]
	2	New	FDPF/CPP-651	FDPF	New[c]	New[c]
	3	New	FMEF	FMEF	New[c]	New[c]
Alternative 4: *Construct New Research Reactor*[f]	1	New	REDC	REDC	New[c]	New[c]
	2	New	FDPF/CPP-651	FDPF	New[c]	New[c]
	3	New	FMEF	FMEF	New[c]	New[c]
Alternative 5: *Permanently Deactivate FFTF (with no new missions)*[d]	-	-	-	-	-	-

a) FFTF operates with MOX fuel for 21 years and uranium fuel for 14 years.
b) FFTF operates with MOX fuel for 6 years and uranium fuel for 29 years.
c) The New Support Facility would not be required if a DOE site with available support capability and infrastructure is selected.
d) Under the No Action Alternative (all options) and Alternative 5, Pu-238 is purchased from Russia to supply NASA programs.
e) Under the No Action Alternative, FFTF is maintained in standby mode indefinitely.
f) Under Alternatives 2, 3, and 4, the FFTF is permanently deactivated.
g) The ATW placeholder is not evaluated in this NI NIA. The ATW program will be the topic of a future ATW NIA.
h) A new low-energy accelerator might also be combined with reactor options under Alternative 2 to fulfill all proposed missions.

ES-3 NUCLEAR MATERIALS RELEVANT TO THIS ASSESSMENT

Mixed Oxide Reactor Fuel. Fresh and spent mixed oxide (MOX) fuel contains plutonium isotopes that are immediately useful as a fissile material in nuclear weapons following chemical separation from the uranium contained in the fuel matrix and metallurgical processing. MOX fuel (PuO_2 mixed with UO_2 in sintered pellet form) is intended as the initial fuel supply for the FFTF in the event that a Record of

Decision directs FFTF to restart. Two sources of fresh MOX fuel for FFTF have been identified in the Draft NI PEIS:

- FFTF MOX fuel currently stored at the Hanford site that was remaining when FFTF went into standby mode. There is enough Hanford MOX fuel to operate the reactor at 100 megawatts thermal (MWt) for about 6 years. This fuel is hereafter referred to as Hanford MOX fuel.
- Partially remanufactured German SNR-300 MOX fuel currently stored at Hanau, Germany and Dounreay, Scotland. This fuel would require some remanufacturing and would be imported to the United States for use in the FFTF. There is enough German SNR-300 MOX fuel to operate the FFTF at 100 MWt for about 15 years following the consumption of the Hanford MOX fuel. This fuel is hereafter referred to as German MOX fuel.

Highly Enriched Uranium Reactor Fuel. All uranium enriched in U-235 to or above 20% is called highly enriched uranium (HEU). HEU is special nuclear material (SNM). HEU fuel is required to operate two of the irradiation facilities proposed in the Draft NI PEIS: the High-Flux Isotope Reactor (HFIR) and the Advanced Test Reactor (ATR). Both research reactors use aluminum clad HEU plate fuel. The HEU contained in the HFIR and ATR plate fuel is 93% enriched such that it is immediately useful as a fissile material in nuclear weapons following chemical separation from the fuel matrix and metallurgical processing.

HEU fuel may be required to operate FFTF following the consumption of available MOX fuel supplies. FFTF can use HEU oxide fuel in the form of sintered pellets. The HEU contained in the FFTF oxide fuel is enriched to between 30 and 37%. International and domestic safeguards regulations treat uranium, that is enriched above 20%, as material that is usable as fissile material for nuclear weapons. However, higher assays are more readily usable than lower assays.

Low Enriched Uranium Reactor Fuel. Any uranium enriched in U-235 to less than 20% is called low enriched uranium (LEU). LEU is SNM. LEU fuel is required to operate two of the irradiation facilities proposed in the Draft NI PEIS: commercial light water reactor (CLWR) and new research reactor. A CLWR uses sintered LEU oxide fuel pellets enriched to between 3 and 4%. A new research reactor would use aluminum clad LEU oxide plate fuel enriched to slightly below 20%. In both cases, conversion to uranium hexafluoride, further enrichment and metallurgical processing would be required to obtain material that is readily usable for nuclear weapons.

In fiscal year 2001, the Department's RERTR program plans to study conversion of ATR to use LEU fuel. If a Record of Decision directs a restart of FFTF, the RERTR program will study the conversion of FFTF to LEU fuel. In both cases, If LEU fuel is found to be technically feasible, it would probably be enriched to slightly less than 20%. LEU fuel would require conversion to uranium hexafluoride, further enrichment and metallurgical processing to obtain material that is readily usable for nuclear weapons.

Neptunium. Neptunium is an alternate nuclear material (ANM). The utility of ANM in nuclear weapons is recognized by the U.S. Government and the international community. The Pu-238 production mission described in the Draft NI PEIS requires the production and irradiation of neptunium targets. Neptunium targets are typically made of purified, concentrated neptunium dioxide with an aluminum binder, canned or clad in aluminum. The production of Pu-238 requires the production of purified neptunium dioxide from neptunium solution followed by target fabrication, irradiation to build in Pu-238, chemical processing to separate and purify neptunium and Pu-238 from fission products and other waste products, and a repeat of the cycle to produce further Pu-238. Each cycle destroys neptunium since neptunium is converted to Pu-238 in the process.

Plutonium-238. Pu-238 is special nuclear material (SNM). However, isotopically concentrated Pu-238 (above 80%) is generally recognized to not constitute a nuclear proliferation threat. The IAEA exempts plutonium that contains more than 80% Pu-238 from international safeguards and DOE assigns this material to the lowest DOE safeguards grade. However, this material is rigorously protected against loss, theft and sabotage (through physical protection and accounting) and is strictly contained (to prevent accidental release) as a result of the health and safety risks presented by the material.

Target and Product Materials Associated with Isotope Production Missions. A wide variety of materials (radioactive and nonradioactive) are described in the Draft NI PEIS to produce targets for the production of medical and industrial isotopes. None of the materials listed as targets or products are materials of nuclear nonproliferation concern. As such, these materials are not relevant to this NI NIA.

Civil Nuclear Energy Research and Development Materials. The nuclear materials that might be involved in civil nuclear energy R&D are not described, or listed in detail in the Draft NI PEIS. However, example missions are described. This NI NIA focuses on the use of materials of nonproliferation concern (nuclear weapons-usable fissile materials: plutonium,[3] HEU and ANM) in facilities, alternatives, and options described in the Draft NI PEIS. Civil nuclear energy R&D studies on materials other than the materials of concern are not germane to this NI NIA.

ES-4 NONPROLIFERATION POLICY CONTEXT

In broad terms, the analysis performed in this assessment focuses on four major proliferation concerns that may be raised by the nuclear facilities and operations reviewed in the Draft NI PEIS:

- The concern that, pursuant to the Draft NI PEIS, the construction or operation of a facility in the United States that uses weapons-usable nuclear materials might encourage the development of similar facilities abroad, to the detriment of U.S. non-proliferation efforts aimed at discouraging the development of such facilities;
- The risk that weapons-usable nuclear material might be stolen from a U.S. nuclear facility constructed or operated pursuant to the Draft NI PEIS by agents of a country of proliferation concern or by a subnational organization or terrorist group;
- The risk that restrictions on voluntary or legally mandated international monitoring of certain U.S. facilities operated pursuant to the Draft NI PEIS might reduce confidence in U.S. pledges that it will never use for nuclear weapons certain weapons-usable nuclear materials that it has declared to be excess to defense needs; and
- The risk that activities proposed under the Draft NI PEIS might interfere with the implementation of anticipated future treaties, such as the Fissile Material Cutoff Treaty (FMCT).

The three weapons-usable nuclear materials whose use and processing are analyzed in this assessment, and which are discussed below, are HEU, plutonium,[4] and neptunium. Although HEU and plutonium have long been the subject of U.S. and international nonproliferation controls, neptunium, which to date has been separated in significant quantities only in nuclear-weapon states, became the subject of international regulation only in 1999.

The United States has long led global efforts to prevent the proliferation of nuclear weapons and to safeguard weapons-usable fissile materials against the risk of theft or diversion. Because the knowledge needed to make at least a crude nuclear weapon is now widespread, limited access to these essential

[3] The term "plutonium" is understood in this context to mean isotopic mixtures of plutonium other than isotopically concentrated Pu-238.
[4] Ibid.

ingredients of nuclear weapons is the principal technical barrier to nuclear proliferation in the world today. Hence, the United States has placed heavy emphasis on efforts to help monitor, protect, control, account for, and, ultimately, dispose of weapons-usable fissile materials worldwide.

Because of its pivotal role in preventing the proliferation of nuclear weapons and its own extensive nuclear programs and activities, the manner in which the United States manages its nuclear activities has a significant influence on other states. U.S. technical and policy choices frequently affect similar choices in other countries both by example and in the way these choices support U.S. diplomatic efforts. Thus, decisions of the type analyzed in the Draft NI PEIS that are taken in the United States can positively or negatively affect efforts to enhance the global nonproliferation regime and bolster the international norm against the acquisition of nuclear weapons. In recent years, the United States has sought to make its nuclear activities increasingly transparent in order to increase international confidence in the global arms control and nonproliferation regime and to encourage similar actions by other countries.

In order to practically evaluate the nonproliferation merits and drawbacks of the activities proposed in the Draft NI PEIS, this NI NIA analyzes the proposed missions, facilities, alternatives, and options within the context of U.S. nonproliferation policy. This body of policy is articulated in President Clinton's 1993 Nonproliferation and Export Control Policy Statement (see Appendix 10.2), other relevant U.S. laws and regulations, and international agreements. Most central to this assessment are policies concerning:

- Plutonium reprocessing;
- Civil use of HEU;
- Monitoring of ANM; and
- Support of anticipated FMCT negotiations.

ES-5 NONPROLIFERATION ASSESSMENT METHODS

Technical and Policy Factors. This NI NIA evaluates the nonproliferation impact of the activities proposed in the Draft NI PEIS by analyzing the various missions, facilities, alternatives, and options against three technical factors and four policy factors. The technical evaluation factors focus on assuring that weapons-usable fissile materials are physically difficult to either steal or divert, and that this material and associated processes are appropriately safeguarded. The three technical factors assess the degree to which an activity would be:

- Assuring against theft or diversion;
- Facilitating cost-effective international monitoring; and
- Resulting in final material forms from which retrieval is more difficult than from original material forms.

The four policy factors used in this assessment focus on the ability of the United States to maintain and strengthen international efforts to stem the spread of nuclear weapons, including the overall approach to limit, restrict, and minimize the use of weapons-usable fissile material in civilian applications. Furthermore, the policy factors also address the continued transparency of the U.S. domestic moratorium on fissile material production for nuclear weapons. The four policy factors include the degree to which an activity would be:

- Maintaining consistency with U.S. nonproliferation policy;
- Avoiding encouragement of plutonium reprocessing;
- Building confidence that the United States is not producing material for nuclear weapons; and

- Supporting negotiation of a verifiable Fissile Material Cutoff Treaty.

Evaluation Grading Scale. A qualitative grading scale on three levels is defined to indicate the degree to which particular missions, facilities, alternatives, or options meet U.S. nonproliferation objectives. The three levels in the grading scale are:

● *Fully Meets Nonproliferation Objectives.* A mission, facility, alternative, or option under a factor assessment *fully meets nonproliferation objectives* if: there *are no significant identified concerns* that can be raised demonstrating how the use of the facility or implementation of the alternative is contrary to U.S. nonproliferation objectives as defined by the assessment factor.

◑ *Might Raise Nonproliferation Concerns.* A mission, facility, alternative, or option under a factor assessment *might raise nonproliferation concerns* if: there *is significant uncertainty* as to whether the use of the facility or implementation of the alternative *might have an adverse effect* on U.S. nonproliferation objectives as defined by the assessment factor.

○ *Raises Nonproliferation Concerns.* A mission, facility, alternative, or option under a factor assessment *raises nonproliferation concerns* if: there *are significant identified concerns* that can be raised demonstrating how the use of the facility or implementation of the alternative is contrary to U.S. nonproliferation objectives as defined by the assessment factor.

ES-6 SUMMARY OF NONPROLIFERATION ASSESSMENTS

Table ES-3 shows the summary of the detailed facility assessment scores. Facilities and mission cases (*e.g.*, FFTF standby/deactivation, neptunium storage) are shown across rows and nonproliferation assessment technical and policy factors are shown down columns. *There are currently no U.S. nonproliferation policies, laws, regulations or international agreements that preclude the use of any of the facilities in the manner described in the Draft NI PEIS.* However, there are a few instances of nonproliferation concerns and uncertainties.

These concerns and uncertainties are associated with the use of processing facilities to recover Pu-238 and neptunium from irradiated neptunium targets as part of the Pu-238 production mission. In all facility cases (REDC, FDPF, and FMEF), the repeated separation and purification of neptunium (which is an unavoidable part of the process) raises *significant uncertainty* under the third technical factor associated with reduction in material attractiveness. This is always the case and is technically unavoidable (even if Pu-238 is purchased from Russia, this process is required in a Russian nuclear facility).

Other concerns and uncertainties surrounding the use of FDPF stem from concerns about transparency measures that could be required as part of an FMCT verification regime. The extent to which FDPF, as a former defense nuclear material production facility, would be available for international monitoring under an FMCT is currently unknown.

Irradiation facilities and missions, as described in the Draft NI PEIS, do not have any identified nonproliferation concerns or uncertainties. Although the intended fuel supply for FFTF includes two different sources for existing MOX fuel, an analysis of these MOX supply options identified significant mitigating factors that indicated substantial nonproliferation benefits to disposing of that attractive material as highly radioactive spent fuel (see Section 4). If HEU fuel is required for either FFTF (30 to 37% enriched) or ATR (93% enriched) it will be procured in strict accordance with U.S. nonproliferation policy following the principles outlined in the Schumer Amendment (see Appendix 10.3). The Schumer

amendment places restrictions on the export of HEU, requiring that facilities pursue conversion to LEU fuels and targets.

It should be added that operation of the FFTF does not set a precedent that may encourage other states to build new high-flux test reactors using MOX or HEU fuels. The FFTF case is unique: it involves an existing, previously operated facility and the irradiation of previously fabricated MOX fuel now in storage, conditions that are highly unlikely to arise elsewhere. Possible future use of HEU at the facility will be subject to the same strict scrutiny that the United States would wish to have applied by other states considering the use of such fuel.

Table ES-3. Assessments of Facilities as Described in the Draft NI PEIS

		Irradiation Facilities								***Target Fabrication and Processing Facilities***						***Np-237 Storage***		
		FFTF Restarted	FFTF Standby/Deactivated	ATR	HFIR	CLWR	New Low-Energy Accelerator	New High-Energy Accelerator	New Research Reactor	REDC	FDPF	FMEF	RPL	306-E	New Support Facility	REDC	CPP-651	FMEF
Technical Factors	Assuring Against Theft or Diversion	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●
	Facilitating Cost-Effective International Monitoring	●	●	●	●	●	●	●	●	●	○	●	●	●	●	●	●	●
	Resulting in Final Material Forms from which Retrieval is More Difficult than from Original Material Forms	●	●	●	●	●	●	●	●	◑	◑	◑	●	●	●	●	●	●
Policy Factors	Maintaining Consistency with U.S. Nonproliferation Policy	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●
	Avoiding Encouragement of Plutonium Reprocessing	●	●	●	●	●	●	●	●	●	◑	●	●	●	●	●	●	●
	Building Confidence that the U.S. is not Producing Material for Nuclear Weapons	●	●	●	●	●	●	●	●	●	◑	●	●	●	●	●	●	●
	Supporting Negotiation of a Verifiable FMCT	●	●	●	●	●	●	●	●	●	○	●	●	●	●	●	●	●

● *Fully meets nonproliferation objectives*
◑ *Might raise nonproliferation concerns*
○ *Raises nonproliferation concerns*

It should also be noted that although the ATR's defense program mission precludes it from international monitoring, there are no U.S. nonproliferation policy directives, international agreements or regulations that generically prevent civil programs from being conducted in current or former defense facilities –

ATR is currently hosting civil radioisotope production programs. However, when comparable alternatives exist that allow civil programs to be hosted in facilities that are eligible for international monitoring, it is preferable to maintain a separation between defense and civil programs.

Table ES-4 shows the detailed assessment grades for each alternative and option described in the Draft NI PEIS (the alternatives and options are shown in Table ES-2). The alternative and option assessments are performed using the methods described in Section 3 (incorporating each of the facility assessments with a generic transportation assessment in a "weak link" analysis). The generic transportation assessment (see Section 8.3) found no significant nonproliferation impact associated with nuclear material transportation.

In Alternatives 1 through 4 (U.S. Pu-238 production alternatives) the assessments are fully determined by the Pu-238 processing facility assessment (REDC, FDPF, and FMEF). Furthermore, under the No Action Alternative and Alternative 5, the alternative assessments are determined by a generic assessment of the Russian Pu-238 purchase option (presented in Section 8.2). The Russian Pu-238 purchase option suffers from similar nonproliferation uncertainties and concerns as FDPF. In addition, the status of Russian domestic safeguards of ANM is largely unknown. Moreover, since there is currently no Russian moratorium on spent fuel reprocessing, and neptunium recovery is part of the Russian reprocessing flowsheet, the Russian inventory of separated weapons-usable neptunium could continue to increase, even if smaller quantities of neptunium were destroyed in the production of Pu-238.

ES-7 CONCLUSIONS AND RECOMMENDATIONS

ES-7.1 OVERALL ASSESSMENT OF MISSIONS PROPOSED IN THE DRAFT NI PEIS

There are currently no U.S. nonproliferation policies, laws, regulations or international agreements that preclude the use of any of the facilities in the manner described in the Draft NI PEIS. The overall missions (independent of selected facilities) proposed in the Draft NI PEIS are evaluated by using the methods presented in Section 3.

Medical, Industrial, and Research Isotope Production. There *are no significant identified concerns* demonstrating how, within the bounds of the description given in the Draft NI PEIS, the pursuit of the medical, industrial, and research isotope production mission is contrary to U.S. nonproliferation objectives as defined by any assessment factor. Therefore, this mission is graded as ● *fully meets nonproliferation objectives.*

Plutonium-238 Production. With the exception of the third technical assessment factor, *reduction in attractiveness of material forms* (see Section 3), there *are no significant identified concerns* demonstrating how, within the bounds of the description given in the Draft NI PEIS, the pursuit of the Pu-238 production mission is contrary to U.S. nonproliferation objectives as defined by the remaining technical and policy assessment factors. Therefore, these remaining factors are graded as ● *fully meets nonproliferation objectives.* In the case of the third technical assessment factor, the process of producing, recovering, and purifying Pu-238 requires that neptunium also be recovered, purified, and recycled. However, in the event that Pu-238 production is resumed in the United States, the total separated stocks of neptunium will be reduced over time in an irreversible manner since there is a moratorium on U.S. spent fuel reprocessing – the activity that could lead to the production of additional stocks of separated neptunium. This overall reduction in a weapons-usable material stock is a partial mitigation of the identified concern. Even so, *there is significant uncertainty* raised with respect to the third technical assessment factor, and that single factor is graded as ◑ *might raise nonproliferation concerns.* However, it should be pointed out that this issue is unavoidable (unless the United States elects to neither produce

Table ES-4. Assessments of Alternatives and Options as Defined in the Draft NI PEIS

		Technical Factors			***Policy Factors***			
Alternatives	***Options***	Assuring Against Theft or Diversion	Facilitating Cost-Effective International Monitoring	Resulting in Final Material Forms from which Retrieval is More Difficult than from Original Material Forms	Maintaining Consistency with U.S. Nonproliferation Policy	Avoiding Encouragement of Plutonium Reprocessing	Building Confidence that the U.S. (Russia)* is not Producing Material for Nuclear Weapons	Supporting Negotiation of a Verifiable FMCT
No Action Alternative*	1	◑	◑	○	●	●	●	◑
	2	◑	◑	○	●	●	●	◑
	3	◑	◑	○	●	●	●	◑
	4	◑	◑	○	●	●	●	◑
Alternative 1: *Restart FFTF*	1	●	●	◑	●	●	●	●
	2	●	○	◑	●	◑	◑	○
	3	●	●	◑	●	●	●	●
	4	●	●	◑	●	●	●	●
	5	●	○	◑	●	◑	◑	○
	6	●	●	◑	●	●	●	●
Alternative 2: *Use Only Existing Operational Facilities*	1	●	●	◑	●	●	●	●
	2	●	○	◑	●	◑	◑	○
	3	●	●	◑	●	●	●	●
	4	●	●	◑	●	●	●	●
	5	●	○	◑	●	◑	◑	○
	6	●	●	◑	●	●	●	●
	7	●	●	◑	●	●	●	●
	8	●	○	◑	●	◑	◑	○
	9	●	●	◑	●	●	●	●
Alternative 3: *Construct New Accelerator(s)*	1	●	●	◑	●	●	●	●
	2	●	○	◑	●	◑	◑	○
	3	●	●	◑	●	●	●	●
Alternative 4: *Construct New Research Reactor*	1	●	●	◑	●	●	●	●
	2	●	○	◑	●	◑	◑	○
	3	●	●	◑	●	●	●	●
Alternative 5: *Permanently Deactivate FFTF (with no new missions)**	-	◑	◑	○	●	●	●	◑

* Under the No Action Alternative (Options 1-4) and Alternative 5, the Russian Pu-238 purchase option is considered.

● *Fully meets nonproliferation objectives*

◑ *Might raise nonproliferation concerns*

○ *Raises nonproliferation concerns*

nor purchase Pu-238) and impacts all alternatives and options, including the No Action Alternative and Alternative 5: permanently deactivate FFTF with no new missions at U.S. facilities.

Civil Nuclear Energy Research and Development. The DOE Office of Nuclear Energy has included Accelerator Transmutation of Waste (ATW) as one of many possible future civil nuclear energy R&D missions as a placeholder in the event that the U.S. Government decides to pursue this technology. Currently, the Department is performing technical paper studies and planning studies (*e.g.*, the "ATW Road Map") to assist Congress with fiscal and program planning. These efforts are also being reviewed by the independent Nuclear Energy Advisory Committee (NERAC) Subcommittee on the Accelerator Transmutation of Waste, which, in its report of May 23, 2000, recommended that, a study should be launched to identify potential proliferation concerns associated with ATW and possible approaches to mitigate identified concerns. A comprehensive nonproliferation impact assessment of the ATW program plan will be performed by the Office of Arms Control and Nonproliferation prior to proceeding beyond paper studies with actual fuels materials testing in support of ATW (or other technologies that include or imply closed fuel cycle technologies). As such, the nonproliferation impact of a possible future ATW program is not considered in this NI NIA since it is not a well-defined, principal identified mission at this time. It will, however, be considered in a future nonproliferation impact assessment if the ATW Program moves forward. With respect to other identified civil nuclear energy R&D missions, there *are no significant identified concerns* demonstrating how, within the bounds of the description given in the Draft NI PEIS, the pursuit of these missions is contrary to U.S. nonproliferation objectives as defined by any assessment factor. In fact, the development of proliferation resistant nuclear fuels and technologies are a significant feature of the intended R&D program. Therefore, this mission is graded as ● *fully meets nonproliferation objectives.*

ES-7.2 NONPROLIFERATION MOST AND LEAST FAVORABLE ALTERNATIVES AND OPTIONS

Since the assessments of alternatives and options are largely determined by the Pu-238 processing facility assessments, the options that use the REDC and FMEF have the most favorable assessments, and the options that use the FDPF have the least favorable assessments. The No Action Alternative and Alternative 5 use the Russian Pu-238 purchase option. These alternatives score between most and least favorable. As a result, the *most favorable nonproliferation alternatives and options* are:

- Alternative 1: Restart FFTF, Options 1, 3, 4, and 6
- Alternative 2: Use Only Existing Facilities, Options 1, 3, 4, 6, 7, and 9
- Alternative 3: Construct New Accelerator(s), Options 1 and 3
- Alternative 4: Construct New Research Reactor, Options 1 and 3

The *least favorable nonproliferation alternatives and options* are:

- Alternative 1: Restart FFTF, Options 2 and 5
- Alternative 2: Use Only Existing Facilities, Options 2, 5, and 8
- Alternative 3: Construct New Accelerator(s), Option 2
- Alternative 4: Construct New Research Reactor, Option 2

ES-7.3 SPECIAL CONSIDERATIONS FOR ALTERNATIVE 1: RESTART FFTF

If the Nuclear Infrastructure Record of Decision elects to restart FFTF (under any option), there are some special considerations. To codify the assumptions underlying the conclusion that restart of the FFTF fully

meets U.S. nonproliferation policy objectives, the Nuclear Infrastructure Record of Decision should include the following commitments:

- The FFTF will not be configured to operate as a breeder reactor (breeding ratio equal to or greater than one) or to optimize the production of plutonium.
- Spent MOX fuel irradiated in the FFTF will not be reprocessed.
- During the period that the FFTF is fueled with Hanford MOX fuel, an analysis will be undertaken by the RERTR program to determine whether the reactor can be fueled with LEU fuel, and if this is shown to be technically feasible, the reactor will be fueled with LEU fuel following the consumption of existing MOX fuel (Hanford and, possibly, German MOX fuel).
- A nonproliferation impact assessment will be prepared on the ATW program prior to the test irradiation of ATW fuels materials in the FFTF.
- The FFTF will remain available for international monitoring.

ES-7.4 NONPROLIFERATION UNCERTAINTIES, CONCERNS, AND MITIGATION APPROACH

There are a limited number of nonproliferation concerns and uncertainties that might be mitigated to increase the number of alternatives and options that have optimum nonproliferation qualities for the missions described in the Draft NI PEIS. These concerns are associated with the U.S. and Russian facilities used to process and recover Pu-238.

- If managed access can be granted to the FDPF, sufficient for verification of an FMCT, the uncertainties and concerns associated with the use of FDPF for the Pu-238 processing mission would be effectively mitigated (with the exception of the material forms technical factor).
- If the United States had sufficient confidence concerning the rigor of Russian controls on ANM, this uncertainty would be effectively mitigated.
- If Russia were to implement a moratorium on spent nuclear fuel reprocessing, the material forms technical factor would be partially mitigated to "◑ might raise nonproliferation concerns" – similar to the U.S. Pu-238 program assessments, since Russia would no longer be able to add to its stocks of separated neptunium.
- If managed access can be granted to the Russian facility responsible for Pu-238 and neptunium recovery, sufficient for verification of an FMCT, the uncertainties associated with the use of a Russian facility for the Pu-238 processing mission would be effectively mitigated (with the exception of the material forms technical factor).

Appendix R
NASA Mission Planning Correspondence

National Aeronautics and
Space Administration
Headquarters
Washington, DC 20546-0001

Reply to Attn of

S

MAY 2 2 2000

Mr. Earl Wahlquist
Associate Director
Office of Space and Defense
Power Systems
U.S. Department of Energy
1000 Independence Avenue, S.W.
Washington, DC 20585

Dear Mr. Wahlquist:

Recent changes in the NASA Deep Space Systems (DSS) program have led to modifications in our planned launch dates and spacecraft configurations for the Europa Orbiter, Pluto-Kuiper Express, and Solar Probe missions. These modifications have been driven by changes in the launch vehicle and technology maturity status, and are based upon the best information and the limited funds available.

Under a proposed plan, the Pluto-Kuiper Express mission would be launched in 2004. Due to cost profile constraints, the baseline mission plan is to utilize the F5 generator "as is" as the primary power source. It is expected that DOE and the Jet Propulsion Laboratory (JPL) in Pasadena, California, will work together to develop a test and inspection program to verify the flight worthiness of the F5 generator.

A proposed plan for the Europa Orbiter mission is to launch in 2006 (with a fallback option for a 2007 launch). A proposed plan for the Solar Probe mission is to launch in 2007 (with a fallback option for a 2008 launch). Both of these missions could possibly utilize the option of a proposed new power system based on a Stirling technology generator. NASA Headquarters and JPL will work closely with DOE to establish requirements that will ensure that this new power system will meet the flight needs of these missions.

As a result of the proposed DSS program changes, NASA Headquarters no longer has an identifiable planned requirement for Small Radioisotope Thermoelectric Generator (SRTG) power systems. Therefore, NASA Headquarters requests that all SRTG development efforts for DSS spacecraft missions be halted. In addition, investigations into the utilization of the E8 and Multi-Hundred Watt systems for DSS applications should be stopped. The closure of these activities will permit reprogramming the funds to support the start-up of the Stirling system development.

As always, your support for the development of spacecraft power systems is greatly appreciated.

Sincerely,

Earle K. Huckins III
Deputy Associate Administrator
for Space Science

cc:
S/E. Weiler
S/C. Pilcher
SD/D.Lavery
SD/M. Dahl
SD/J. Parrish
DOE/W. Magwood
DOE/L. Herrera

National Aeronautics and
Space Administration
Headquarters
Washington, DC 20546-0001

SEP 12 2000

Reply to Attn of S

TO: Jet Propulsion Laboratory
Attn: 180/904/Director

FROM: S/Associate Administrator for Space Science

SUBJECT: Pluto-Kuiper Express Mission and Reformulation of the Deep Space Systems Program

I have concluded, for programmatic and technical reasons, that it is not feasible to implement the Pluto-Kuiper Express (PKE) mission as currently conceived. I, therefore, direct you to immediately stop all work in support of PKE development as currently conceived.

In anticipation of a new science roadmap for Outer Planet Exploration, you are further directed to begin developing a technology plan that would support a mission to Pluto and the Kuiper Belt, to arrive at Pluto no later than 2020. Clearly, a central objective of this technology plan should be to reduce flight time to Pluto substantially from what is currently achievable.

You are also directed to continue all activities necessary to support a launch of the Europa Orbiter mission in January 2006.

The Acting Science Director for Solar System Exploration, Dr. Jay T. Bergstralh, will be in contact with your staff regarding other aspects of the new Outer Planets Science Roadmap. If you or your staff have any questions, please feel free to contact Dr. Bergstralh at (202) 358-0313.

Edward J. Weiler

cc:
B/Mr. Mal Peterson
S/Dr. E. Huckins
JPL/NMO/180-801/Dr. B. Parker

National Aeronautics and
Space Administration
Headquarters
Washington, DC 20546-0001

SEP 22 2000

Reply to Attn of SD

Mr. Earl Wahlquist
Associate Director
Office of Space and Defense
Power Systems
U.S. Department of Energy (DOE)
1000 Independence Avenue, SW
Washington, DC 20585

Dear Mr. Wahlquist:

The Office of Space Science recently concluded, for programmatic and technical reasons, that it is not feasible to implement the Pluto-Kuiper Express (PKE) mission as currently conceived. We have directed the Jet Propulsion Laboratory to stop all work in support of the current PKE development and to continue all activities necessary to support a launch of the Europa Orbiter (EO) mission in January 2006. Therefore, the mission planning guidance provided to you in our May 22, 2000, letter needs to be updated.

While the design baseline for the EO mission includes the use of Stirling technology radioisotope generators, a viable backup power system is needed in order to support the 2006 launch. Current power estimates indicate that both the F-5 and E-8 generators would be required to support the mission. Your mission planning should consider maintaining the viability of both F-5 and E-8 generators as EO backups.

NASA relies on DOE's capability to produce and provide Radioisotope Power Systems (such as Stirling) that are essential for planning and implementing deep space and long lived space exploration missions. Your continued support is appreciated.

Sincerely,

Earle K. Huckins III
Deputy Associate Administrator
for Space Science

DOE/EIS-0310
December 2000

FINAL Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility

Volume 3 Book 1–Comment Response Document

United States Department of Energy
Office of Nuclear Energy, Science and Technology
Washington, DC

Cover photograph and illustration identification, beginning at top center and going clockwise:

- Radioisotope tagged monoclonal antibodies, “smart bullets,” target malignant cells for diagnosis and treatment of diseases
- The Fast Flux Test Facility at the Hanford Site near Richland, Washington
- Illustration of a satellite that could use radioisotope power systems
- The High Flux Isotope Reactor at the Oak Ridge National Laboratory near Oak Ridge, Tennessee
- The Advanced Test Reactor at the Idaho National Engineering and Environmental Laboratory near Idaho Falls, Idaho
- Tip of a remote-handling arm, used for work in developing industrial and medical isotopes

AVAILABILITY OF THE FINAL NI PEIS

General questions regarding this PEIS or for a copy of this PEIS, please contact:

Colette E. Brown, Document Manager
Office of Space and Defense Power Systems (NE-50)
Office of Nuclear Energy, Science and Technology
U.S. Department of Energy
19901 Germantown Road
Germantown, MD 20874
Attention: NI PEIS
Telephone: (877) 562-4593
E-mail: Nuclear.Infrastructure-PEIS@hq.doe.gov

This PEIS is accessible on the Office of Nuclear Energy, Science and Technology web site at www.nuclear.gov.

Printed with soy ink on recycled paper

Cover Sheet

Responsible Agency: United States Department of Energy (DOE)

Title: *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility* (NI PEIS)

Locations: Idaho, Tennessee, Washington

Contacts: For copies of this programmatic environmental impact statement (PEIS), call toll-free (877) 562-4593

For additional information on this Final PEIS, contact:

Colette E. Brown, Document Manager
Office of Space and Defense Power Systems (NE-50)
Office of Nuclear Energy, Science and Technology
U.S. Department of Energy
19901 Germantown Road
Germantown, MD 20874
Attention: NI PEIS
Telephone: (877) 562-4593

For general information on the DOE National Environmental Policy Act (NEPA) process, contact:

Carol M. Borgstrom, Director
Office of NEPA Policy and Compliance (EH-42)
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585
Telephone: (202) 586-4600, or leave a message at (800) 472-2756

Abstract: Under the authority of the Atomic Energy Act of 1954, as amended, the DOE is responsible for ensuring the availability of isotopes for medical, industrial and research applications, meeting the nuclear material needs of other Federal agencies, and undertaking research and development activities related to development of nuclear power for civilian use. To meet these responsibilities, DOE maintains nuclear infrastructure capabilities that support various missions. Current estimates for the future needs of medical and industrial isotopes, plutonium-238, and research requirements indicate that the current infrastructure may soon be insufficient to meet the projected demands. DOE proposes to enhance these capabilities to provide for: (1) production of isotopes for medical and industrial uses, (2) production of plutonium-238 for use in advanced radioisotope power systems for future National Aeronautics and Space Administration (NASA) space exploration missions, and (3) the Nation's nuclear research and development needs for civilian application.

This NI PEIS evaluates the environmental impacts of a No Action Alternative (maintaining status quo), four alternative strategies to accomplish this mission, and an alternative to permanently deactivate the Fast Flux Test Facility (FFTF), with no new missions. Alternatives 2, 3, and 4 also include permanent deactivation of FFTF. The alternatives are:

No Action
1. Restart FFTF at Hanford, Washington
2. Use only existing operational facilities
3. Construct one or two new accelerators
4. Construct a new research reactor
5. Permanently deactivate FFTF (with no new missions)

The Preferred Alternative is Alternative 2, Option 7, Use Only Existing Operational Facilities. DOE would reestablish domestic production of plutonium-238, as needed, using the Advanced Test Reactor in Idaho and the High Flux Isotope Reactor in Tennessee, and would process irradiated plutonium-238 targets at the Radiochemical Engineering Development Center in Tennessee. DOE would permanently deactivate FFTF under the Preferred Alternative.

Public Comments: The Draft NI PEIS was issued for public review and comment on July 21, 2000. The comment period ended on September 18, 2000, although late comments were considered to the extent practicable. Public hearings were held to obtain comments on the Draft NI PEIS in Oak Ridge, Tennessee; Idaho Falls, Idaho; Hood River and Portland, Oregon; Seattle and Richland, Washington; and Arlington, Virginia. All comments were considered by DOE in preparing the Final NI PEIS, which also incorporates any new information received since issuance of the Draft NI PEIS. In response to comments on the Draft NI PEIS and as a result of information that was unavailable at the time of the issuance of the Draft PEIS, the Final PEIS contains revisions and new information, indicated by a sidebar in the margin. Volume 3 contains the comments received during the public review period for the Draft NI PEIS and DOE's responses to these comments. DOE will use the analyses presented in the Final NI PEIS as well as other information, including public input, costs, nonproliferation impacts, schedules, technical assurance, and other policy and programmatic objectives, in preparing the Record of Decision for accomplishing expanded civilian nuclear energy research and development and isotope production missions in the United States, including the role of FFTF. DOE will issue the Record of Decision no sooner than 30 days after the U.S. Environmental Protection Agency publishes a notice of availability of the Final NI PEIS in the Federal Register.

Reader's Guide

Volume 3, the *Comment Response Document*, is organized into three chapters:

- Chapter 1 - Overview of the Public Comment Process and the Comment Response Document
- Chapter 2 - Written Comments and DOE Responses
- Chapter 3 - Oral Comments Presented at the Public Hearings and DOE Responses

These chapters are divided among the three books of Volume 3 as follows:

- Book 1 - Chapter 1 and Chapter 2 (pages 2-1 through 2-931)
- Book 2 - Chapter 2 (pages 2-932 through 2-1914)
- Book 3 - Chapter 2 (pages 2-1915 through 2-2344) and Chapter 3

Chapter 1, "Overview of the Public Comment Process and the Comment Response Document," summarizes key issues raised during the comment period on the Draft NI PEIS. It also identifies major changes made to this NI PEIS after publication of the Draft in response to these comments and incorporates new information that was unavailable at the time of the issuance of the Draft NI PEIS.

Chapter 2, "Written Comments and DOE Responses," provides a side-by-side display of the written comments received (full-text reproductions) and DOE's responses. Individual comments are numbered in the margins of the comment document, and DOE responses to each numbered comment are provided on the right side of each page.

The comment document numbers in Chapter 2 are in ascending order but are not sequential. Each comment document was assigned a sequential log number as it was received. When the same comment document was submitted by many individuals, it was designated as a campaign. The campaigns were grouped together for the purpose of responding to comments, and do not appear in numerical order.

Chapter 3, "Oral Comments Presented at the Public Hearings and DOE Responses," provides a side-by-side display of the oral comments presented at the public hearings and DOE's responses. The speakers' names appear alphabetically by hearing location. Commentors who submitted their oral presentations in writing will find their submittals and DOE's responses in Chapter 2.

To Find a Specific Comment Document and DOE Response

Refer to the "List of Commentors" immediately following the Volume 3 Table of Contents. This list is organized alphabetically and contains the corresponding page number(s) to find the comment document. The public officials, organizations, and interest groups appear first, then individuals are listed. City and state government bodies are listed under "City of" or "State of." Members of Congress are listed alphabetically under "Members of Congress."

DOE has made a good faith effort to interpret the spelling of names that were either written on comments or were recorded on the telephone comment line.

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List of Tables

List of Commentors

Public Officials, Organizations, and Interest Groups

Individuals

Chapter 1

Overview of the Public Comment Process and the Comment Response Document

Chapter 1
Overview of the Public Comment Process and the Comment Response Document

In July 2000, the U.S. Department of Energy (DOE) published the *Draft Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (Nuclear Infrastructure Programmatic Environmental Impact Statement [NI PEIS]).* In accordance with the Council on Environmental Quality (CEQ) and DOE National Environmental Policy Act (NEPA) regulations, a Federal Register notice (65 FR 46443) announced the availability of the Draft NI PEIS and invited interested parties to provide comments. The Draft NI PEIS or Summary was distributed to approximately 6,000 individuals.

1.1 The Public Comment Process

NEPA regulations mandate a minimum 45-day comment period after the U.S. Environmental Protection Agency's (EPA) Notice of Availability of a draft EIS to provide an opportunity for the public to comment on the EIS analysis and results. The 45-day comment period on the Draft NI PEIS began on July 28, 2000, and to provide interested parties with additional time to comment, the deadline for transmittal of comments was changed from September 11, 2000 (as stated in the transmittal letter of the Draft NI PEIS and the Summary), to September 18, 2000. While the official comment period ended on September 18, 2000, DOE addressed late comments to the extent practicable and considered all comments received through October 31, 2000, in preparing the final NI PEIS. Comments that were received through September 25, 2000, along with corresponding responses, have been included in Chapter 2 of this volume. Direct responses are not included to comments that were received after September 25, 2000. However, all of these comments were considered and are characterized by other comments received during the comment period (for which a response has been provided).

1.2 Public Hearing Format

During the 52-day comment period, DOE held seven hearings to discuss the proposed action and to receive oral and written comments on the Draft NI PEIS. The hearings were held near the locations that would be affected by the proposed alternatives, as well as some additional locations in Oregon and Washington in response to stakeholder requests. In addition, a hearing was held in the Washington, D.C. area. The hearing schedule and estimated attendance at each hearing are presented in **Table 1–1**. These attendance estimates are based on the number of registration forms completed and returned to DOE at each hearing, as well as a rough "head count" of the audience, and may not include all those present.

Table 1–1 Hearing Schedule and Attendance

Hearing Location	Date	Estimated Attendance
Oak Ridge, Tennessee	August 22, 2000	15
Idaho Falls, Idaho	August 25, 2000	20
Hood River, Oregon	August 28, 2000	265
Portland, Oregon	August 29, 2000	320
Seattle, Washington	August 30, 2000	250
Richland, Washington	August 31, 2000	330
Arlington, Virginia	September 6, 2000	15
Total		1,215

An independent facilitator was present at each hearing to direct and clarify discussions and comments. A comment recorder also was present at each hearing to record the proceedings. At the hearings in Oregon and Washington, a second comment recorder was available in a separate room to receive comments from attendees who were not able to attend the entire session, or who wanted to give their comments and leave. Transcripts of the hearings are available in DOE public reading rooms and libraries listed in Chapter 7 of Volume 1.

DOE representatives were available to meet with the public for informal discussions prior to and after the hearings. In an effort to encourage interaction between members of the public and DOE representatives, DOE used an interactive format for the hearings. The format included a presentation, question and answer session, and a comment session. Each hearing opened with a welcome from the facilitator, followed by a presentation on the proposed action by a DOE representative. Next, the facilitator opened the question and answer session to give the audience a chance to ask questions about the material presented. This was followed by the comment session, during which attendees were randomly selected to provide their comments. Attendees received a numbered ticket from the staff at the registration table and the facilitator picked the tickets from a container to determine the order of speakers. To ensure that all attendees were given an opportunity to provide comments, each speaker was limited to 5 minutes. Those commentors who needed additional time were invited to speak again after everyone had an initial opportunity to provide their comments. Modifications to the format were made at each of the public hearings to fulfill any special requests of attendees.

1.3 COMMENTS ON THE DRAFT NI PEIS

The public was encouraged to submit comments on the Draft NI PEIS to DOE via U.S. mail, e-mail, telephone, fax, and at the public hearings. DOE received approximately 3,400 submittals containing over 6,200 comments addressing a wide range of issues. A number of written comments submitted during the hearings were also presented orally; those comments were counted once. All comments submitted to DOE during the comment period were given equal consideration in preparation of the Final NI PEIS. Comments determined to be beyond the scope of the NI PEIS were forwarded to the appropriate DOE office for consideration. **Table 1–2** lists the number of comments received by method of submission.

Table 1–2 Comment Submission Method

Method	Number of Submittals
U.S. mail	2,493
E-mail	332
Telephone	107
Fax	92
Comments submitted at hearings	439
Total	3,463

Upon receipt, all written submittals were date-stamped and assigned a sequential log number used in tracking during the comment response process. Oral comments presented at the hearings were similarly identified and assigned a sequential log number. All comments were then processed through the comment analysis and response system for inclusion in this document. Each comment was assigned to a specific category to facilitate response and provide an overview of the type of comments that DOE received. Documents identical in content are presented only once (e.g., a written comment that was presented orally at a hearing). Campaigns (e.g., identical comments submitted by numerous individuals) likewise are presented and responded to only once. However, campaign documents with additional comments are responded to separately. The comment categories are shown in **Table 1–3**.

Table 1–3 Comment Categories

Accelerator Design	Miscellaneous Cost Issues
Air Quality	NEPA Process (extension of comment period, public participation, availability of information, completeness of overall analysis, additional hearings, etc.)
Alternative 1 - Restart FFTF	No Action Alternative
Alternative 2 - Use Only Existing Operational Facilities	Noise
Alternative 3 - Construct New Accelerator(s)	Nonproliferation
Alternative 4 - Construct New Research Reactor	Nuclear Energy Research and Development
Alternative 5 - Permanently Deactivate FFTF (with no new missions)	Oak Ridge Reservation Site Issues
Applicable Laws, Regulations, and Other Requirements	Policy
Cost of Alternatives	Preferred Alternative
Cultural and Paleontological Resources	Processing Facilities
Cumulative Impacts and General Environmental Impacts	Production of Medical and Industrial Isotopes
Ecological Resources	Production of Plutonium-238
Environmental Justice	Public and Occupational Health and Safety - Facility Accidents
Existing Human Health Risks	Public and Occupational Health and Safety - Normal Operations
FFTF Investment	Purpose, Need, and Timing of Missions
General Alternative Issues (alternatives considered but dismissed, new alternatives, etc.)	Reactor Design
General Irradiation Needs	Relationship to Other DOE Programs
General Antinuclear	Scoping
Generic Support Facility Design	Socioeconomics
Geology and Soils	Transportation (incident-free and accidents)
Hanford Site Issues	Visual Resources
Idaho National Engineering and Environmental Laboratory Site Issues	Waste Management (includes spent fuel issues)
Irradiation Facilities	Water Resources
Land Resources	

Chapter 2 contains the comments (submitted in writing and by telephone) and the DOE responses presented in a side-by-side format, with each delineated comment receiving a separate response. Not all responses appear directly next to their corresponding comment due to the varying lengths of each response. However, all comments and responses are numbered with a comment identification number to facilitate matching a comment with its response. Where commentors presented support for, or opposition to, a specific alternative, this was noted. Where commentors provided additional statements supporting their positions, DOE responded in detail to those that needed clarification or were in error.

Chapter 3 contains the comments that were submitted during oral presentations at the public hearings held in August and September 2000. The chapter is organized alphabetically by speaker's name according to the hearing location. The format and response procedures used in Chapter 2 were followed in Chapter 3.

Commentors who submitted their oral presentations in writing will find their submittals and responses in Chapter 2. The full transcripts from each hearing are available at DOE reading rooms and libraries listed in Chapter 7 of Volume 1.

An alphabetical List of Commentors with corresponding page numbers has been provided immediately following the Volume 3 Table of Contents to assist the reader in finding specific comment documents and

DOE responses. Public officials, organizations, and interest groups appear first, then individuals are listed. City and state government bodies are listed under "City of" or "State of." Members of Congress are listed alphabetically under "Members of Congress."

1.4 ENVIRONMENTAL PROTECTION AGENCY RATING OF THE NI PEIS

EPA reviewed and rated the Draft NI PEIS as Environmental Concerns - Insufficient Information (EC-2). To a large extent, a lack of information in the Draft NI PEIS was the basis for their environmental concerns. EPA was also concerned that the cost and nonproliferation reports were not made available to the public until well into the comment period on the Draft NI PEIS. A copy of the EPA rating is included among the written comments in Chapter 2 of this volume.

1.5 ISSUES RAISED DURING THE PUBLIC COMMENT PERIOD ON THE DRAFT NI PEIS

During the public comment period on the Draft NI PEIS, DOE received approximately 3,400 submittals containing over 6,200 comments addressing a wide range of issues. DOE considered comments received after the close of the public comment period to the extent practicable (see Section 1.5.6).

The following discusses the major issues raised, and DOE's responses to these issues. Changes made in response to comments received on the Draft NI PEIS are described in Section 1.6.

Major issues raised addressed purpose and need for the proposed action; impact of FFTF on Hanford cleanup; waste management and spent nuclear fuel; cost of the various alternatives; nuclear nonproliferation policy; public involvement; and environmental impacts. Aside from comments on the proposed action and its environmental impacts, many commentors expressed support for or opposition to FFTF restart, the major point of public controversy associated with the NI PEIS.

1.5.1 Purpose and Need for the Proposed Action

Many commentors expressed the opinion that DOE failed to demonstrate a compelling argument for the projected need for medical isotopes, and that such medical isotopes could be produced or purchased elsewhere, particularly in Canada. In contrast, a large number of commentors expressed support for expanded isotope production by sharing personal stories of how medical isotopes had either saved a relative or friend, or could have saved them had isotopes been available. As presented in Section 1.2.1 of Volume 1, DOE sought independent analysis of trends in the use of medical isotopes, and established two advisory bodies, the Expert Panel and the Nuclear Energy Research Advisory Committee (NERAC). DOE has adopted these growth projections as a planning tool for evaluating the potential capability of the existing nuclear facility infrastructure to meet programmatic requirements. In the period since the initial estimates were made, the actual growth of medical isotope use has tracked at levels consistent with the Expert Panel findings. While Canada currently provides a large amount of the medical radioisotopes used in the United States, it only supplies a limited number of economically attractive commercial isotopes (primarily molybdenum-99), and it does not supply research isotopes or the diverse array of medical and industrial isotopes considered in the NI PEIS.

A number of commentors also questioned the suitability of using FFTF for producing research isotopes in light of findings presented in the NERAC Subcommittee for Isotope Research and Production Planning Report. While it would not be cost effective to restart FFTF for the singular purpose of producing small quantities of various research isotopes, sustained operation of FFTF for the production of larger quantities of both research and commercial isotopes would be viable if FFTF were operated in concert with producing plutonium-238 and conducting nuclear energy research and development for civilian applications. In recognition of these

constraints on its operational feasibility, the NI PEIS only evaluates the use of FFTF for isotope production when coupled with these other missions.

Commentors also questioned the need for the United States to reestablish domestic production of plutonium-238. In particular, commentors pointed to the availability of plutonium-238 that could be purchased from Russia, and recent guidance from NASA stating that DOE no longer needed to support certain radioisotope power systems. As discussed in Section 1.2.2 of Volume 1, DOE could purchase plutonium-238 from Russia. However, for supply reliability reasons and concern of nuclear nonproliferation, DOE's preference is to establish a domestic plutonium-238 production capability. Current NASA guidance to DOE is also discussed in Section 1.2.2. The May 22, 2000, correspondence from NASA identifies that it no longer has a planned requirement for Small Radioisotope Thermoelectric Generator (SRTG) power systems. This does not mean that NASA no longer requires DOE to provide the necessary plutonium-238 to support deep space missions. Rather, SRTG development efforts were stopped in order to permit reprogramming of funds to support development of a new radioisotope power system based on a Stirling radioisotope power systems technology generator. This new radioisotope power system, referred to in the subject correspondence, requires one-third less plutonium as its fuel source. Because the Stirling radioisotope power systems technology is developmental, NASA has requested in a September 22, 2000, letter to DOE that the plutonium-238 needed for a large radioisotope thermoelectric generator be maintained as a backup.

1.5.2 Impact of FFTF Restart on Hanford Cleanup

A number of commentors expressed concern that DOE's primary mission at Hanford needs to be cleanup, including compliance with the Tri-Party Agreement. Although beyond the scope of this NI PEIS, ongoing Hanford cleanup activities are high priority to DOE. Hanford environmental restoration activities are conducted in accordance with the Tri-Party (i.e., DOE's Richland Operations Office, EPA, and the State of Washington Department of Ecology) Agreement. This agreement specifies milestones and schedules for restoration of all parts of Hanford. FFTF milestones in the Tri-Party Agreement were placed in abeyance (suspension) by agreement of the three parties until a decision is made on the future of FFTF. Public meetings were held on this formal milestone change. DOE is fully committed to honoring this agreement.

A number of commentors also expressed concern that funding for Hanford cleanup would be diverted for FFTF restart and hamper the progress of cleanup activities. The U.S. Congress funds Hanford cleanup through the Office of the Assistant Secretary for Environmental Management (EM). Congress also funds FFTF through the Office of Nuclear Energy, Science and Technology (NE). The nuclear infrastructure missions described in Section 1.2 of Volume 1 would also be funded through NE, which has no funding connection to Hanford cleanup activities. As stated in Section N.3.2 of Volume 2, implementation of the nuclear infrastructure alternatives would not divert or reprogram budgeted funds designated for Hanford cleanup, regardless of the alternative(s) selected.

1.5.3 Waste Management and Spent Nuclear Fuel

A number of commentors expressed concern over the generation and disposition of waste resulting from the proposed action. In particular, commentors pointed to past DOE waste management practices and questioned whether wastes resulting from proposed NI PEIS activities would be properly managed. The NI PEIS addresses wastes produced for each alternative, as well as cumulative impacts related to waste production. Waste minimization programs at each of the alternative sites are also addressed. These programs would be implemented for the alternative selected in the Record of Decision. The waste generated from any of the alternatives considered in the NI PEIS would be managed (i.e., treated, stored, and disposed of) in a safe and environmentally protective manner and in compliance with all applicable Federal and state laws and regulations and applicable DOE orders.

A number of commentors expressed specific concern over the generation and disposition of waste resulting from FFTF restart and operation, and how this would impact Hanford's existing waste management infrastructure. Management of wastes that would be generated under implementation of Alternative 1 (Restart FFTF) is discussed in Section 4.3 of Volume 1 (e.g., Section 4.3.1.1.13). Section 4.3.1.1.13 was revised to clarify that the Hanford waste management infrastructure is analyzed in this NI PEIS for the management of waste resulting from FFTF restart and operation. This analysis is consistent with policy and DOE Order 435.1, *Radioactive Waste Management*, that DOE radioactive waste shall be treated, stored, and in the case of low-level waste, disposed of at the site where the waste is generated, if practical, or at another DOE facility. However, if DOE determines that use of the Hanford waste management infrastructure or other DOE sites is not practical or cost effective, DOE may issue an exemption under DOE Order 435.1 for the use of non-DOE facilities (i.e., commercial facilities) to store, treat, and dispose of such waste generated from the restart and operation of FFTF. In addition, Sections 4.3.3.1.13 and 4.4.3.1.13 also address the potential impacts associated with the waste generated from the target fabrication and processing in the Fuels and Materials Examination Facility (FMEF) and how this waste would be managed at the site.

A number of commentors also raised concern that processing of irradiated targets for production of plutonium-238 would generate high-level radioactive waste. DOE Manual 435.1, *Radioactive Waste Management*, defines high-level radioactive waste as "the highly radioactive waste material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that is determined, consistent with existing law, to require permanent isolation." DOE has prepared an implementation guide to M 435.1 to assist in implementing the requirements contained in that manual. For this particular "requirement," the definition of high-level radioactive waste, the guide is intended to facilitate the classification of indefinite waste as to whether or not they are high-level radioactive waste. It is recognized that the definition of high-level radioactive waste is not precise and is essentially a source-based definition that also alludes to concentrations of a given waste stream. Page II-8 of the guide notes that "For the purpose of managing high-level waste under DOE M 435.1-1 [sic], spent nuclear fuel includes spent driver elements and/or irradiated target elements that contain transuranium elements." This statement was included in the guide because the concentrations of long-lived isotopes are likely to be somewhat high during reprocessing and it also meets the source-based definition. As a result of reviewing this guide and to address the comments raised, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. As a result, the Waste Management sections (i.e., Sections 4.3.1.1.13, 4.3.2.1.13, 4.3.3.1.13, and 4.4.3.1.13) of this NI PEIS have been revised to reflect this different classification from what was assumed in the Draft NI PEIS. As discussed in these revised sections, irrespective of how the waste is classified (i.e., transuranic or high-level radioactive waste), the composition and characteristics are the same, and the waste management (i.e., treatment and onsite storage) for this NI PEIS would be the same. In addition, even if the waste were managed as high-level radioactive waste, it would have no impact on the existing high-level radioactive waste management infrastructure (e.g., high-level waste storage tanks) because the high-activity waste from processing the targets would be initially stored and vitrified within the processing facility (i.e., FMEF, the Radiochemical Engineering Development Center [REDC], or the Fluorinel Dissolution Process Facility [FDPF]).

Commentors also expressed concern over the potential impacts of spent nuclear fuel generation from FFTF restart and operation, particularly regarding human health risk. This NI PEIS estimates that about 16 metric tons of heavy metal spent nuclear fuel would be generated over 35 years of operation of FFTF. Hanford is currently managing about 2,000 metric tons of heavy metal spent nuclear fuel. As indicated in Table 4–173, the radiation risk to a maximally exposed individual from normal operational activities during management of the current stored nuclear fuel over 35 years is 1.4×10^{-8} latent cancer fatality. The risk to the maximally exposed individual that would be associated with the new nuclear infrastructure operations to restart FFTF and operate FMEF or the Radiochemical Processing Laboratory is 9.5×10^{-8} latent cancer fatality. Furthermore,

only a small fraction of this risk would be attributable to management of the additional spent nuclear fuel at FFTF. The annual dose to the maximally exposed individual from all current and reasonably foreseeable activities is less than 0.2 millirem. This dose is well within the DOE dose limits given in DOE Order 5400.5, *Radiation Protection of the Public and the Environment*. As discussed in that order, the dose limit from airborne emissions is 10 millirem per year, as required by EPA regulations under the Clean Air Act; the dose limit from drinking water is 4 millirem per year, consistent with the EPA drinking water criteria under the Safe Drinking Water Act; and the dose limit from all pathways combined is 100 millirem per year. The risk to the population from all activities at Hanford would be 0.21 latent cancer fatality over 35 years. DOE has committed to remove the spent nuclear fuel at Hanford for ultimate disposition in a geologic repository.

1.5.4 Cost of the Various Alternatives

Commentors expressed opinions about the costs related to the stated missions. Commentors stated that a cost-benefit analysis was necessary to show the value of production of medical isotopes balanced against facility costs, in particular, the restart of FFTF, and noted that perhaps facilities would be able to pay for themselves. There were concerns that FFTF restart would take funds away from the cleanup of Hanford. Commentors noted that the decommissioning costs were not included for the restart FFTF option in the *NI Cost Report*. Several commentors remarked that the expense of plutonium-238 production cannot be justified when DOE needs to clean up existing problems at its sites.

Although the costs of proposed actions are not required by NEPA and CEQ regulations to be included in a PEIS, DOE prepared a separate *NI Cost Report*. This report would provide additional pertinent information to the Secretary of Energy so that he may make an informed decision with respect to the alternatives presented in this Final NI PEIS. Pursuant to CEQ regulations (40 CFR Section 1505.1(e)), such a document comparing alternatives should be made available to the public prior to any decision being made. DOE mailed this document to more than 730 interested parties on August 24, 2000. This report was made available immediately upon release on the NE web site (http://www.nuclear.gov) and in the public reading rooms. DOE has also provided the summary of the *NI Cost Report* in Volume 2, Appendix P, in this Final NI PEIS.

1.5.5 Nuclear Nonproliferation Policy

Commentors expressed opinions about the nuclear nonproliferation implications of the proposed action. Commentors were concerned about keeping plutonium-238 out of the hands of third parties, and it was suggested that the purchase of plutonium-238 from Russia would stop proliferation of the material and the United States would know the disposition of the quantity purchased. Several commentors raised concerns about specific facilities described in the NI PEIS, including FDPF and FFTF. The use of highly enriched uranium fuel in FFTF was questioned related to a possible violation of U.S. nuclear nonproliferation policy. Conversely, the shutdown of FFTF that occurred previously was characterized as being done to discourage proliferation of nuclear weapons worldwide, but had instead weakened the U.S. position as a world leader in nuclear technology. There were comments about the timeliness of release of the *NI Nonproliferation Impact Assessment*, that no nonproliferation information was included in the Draft NI PEIS, and that nuclear nonproliferation policy should be considered by DOE in selection of its preferred alternative.

The plutonium being considered for production in this NI PEIS is plutonium-238, which is not the same isotope of plutonium that is used in nuclear weapons. The production of plutonium-238 does not present a nonproliferation concern. DOE developed the separate *NI Nonproliferation Impact Assessment*, published in September 2000, that analyzed the nonproliferation impacts of the actions considered in this PEIS and found that there are no U.S. nonproliferation policies, laws, regulations, or international agreements that preclude the use of any of the facilities in the manner described in the Draft NI PEIS. Although this policy analysis is not required under NEPA, it is an essential element in the decision-making process for the DOE nuclear

infrastructure. A summary of the *NI Nonproliferation Impact Assessment* is included in Volume 2, Appendix Q, of this Final NI PEIS. It is also available on the DOE NE web site (http://www.nuclear.gov).

1.5.6 Public Involvement

Commentors expressed opinions about the length of the comment period on the Draft NI PEIS, and said they wanted additional time to obtain and review relevant documents, including the *NI Cost Report* and *NI Nonproliferation Impact Assessment*. As identified in Section 1.1, the deadline for transmittal of comments was changed from September 11, 2000, to September 18, 2000 (as stated in the transmittal letters of the Draft PEIS and the Summary). While the official comment period ended on September 18, 2000, DOE addressed late comments to the extent practicable and considered all comments received through October 31, 2000, in preparing this Final NI PEIS. Comments that were received through September 30, 2000, along with corresponding responses, have been included in Chapter 2 of this volume. Direct responses are not included to comments that were received after September 30, 2000. However, all these comments were considered and are characterized by other comments received during the comment period (for which a response has been provided).

Many commentors expressed the opinion that public input is intended for "show only," and that DOE has already made its decisions. Commentors also stated that they had given the same comments over and over again and that DOE representatives were not listening. DOE policy encourages effective public participation in its decision-making process. In compliance with NEPA and CEQ regulations, DOE provided opportunity to the public to comment on the scope of the NI PEIS and the environmental impact analysis of DOE's proposed alternatives. DOE gave equal consideration to all comments. In preparing the Final NI PEIS, DOE carefully considered all comments received from the public.

Some commentors expressed opinions about the conduct of the hearings, both positive and negative. The public hearing format was designed to be fair. The public hearing format used was based on stakeholder input and was presented in the Notice of Availability (65 FR 46443 et seq.) for the Draft NI PEIS. This format was intended to encourage public participation, regardless of the motivation for attending the hearing. It provided an opportunity for the participants to meet one another, exchange information, and share concerns, with DOE personnel available throughout the course of each hearing to answer questions. The meetings were facilitated by an independent moderator to ensure that all persons wishing to speak had an opportunity to do so. Persons wishing to comment were selected at random from the audience rather than according to the order in which they registered. This was accomplished by a random number drawing. In addition to the comment recorder stationed at the main hearing, a second recorder was available in an adjacent room to receive comments without the need to await selection at the main proceeding. The hearing format promoted open and equal representation by all individuals and groups.

1.5.7 Environmental Impacts

A number of commentors questioned the results of the environmental impact analysis and cumulative impacts, specifically at Hanford. Many of these comments focused on concerns that the proposed action would result in negative impacts to the health of individuals residing in the Hanford region. The NI PEIS analyzes the impacts of the various alternatives, and the environmental impacts associated with all proposed nuclear infrastructure activities are addressed in detail in Chapter 4 of Volume 1. Specifically, the environmental impacts associated with operation of the Hanford facilities during normal operations and from postulated accidents are presented in Section 4.3. These assessments were made using well-established and accepted analytical methods, as described in Appendixes G through L in Volume 2. The analytical methodology is conservative by nature; the actual impacts to the environment would be expected to be less than calculated. All impacts have been shown to be small. No fatalities among workers or the general public would be

expected over the 35-year operational period. The impacts to the biosphere (air, water, and land) were also evaluated and determined to be small.

Some commentors raised specific concern over potential contamination of the Columbia River resulting from the restart of FFTF. However, FFTF is approximately 4.5 miles from the Columbia River. There are no discharges to the river from FFTF and no radioactive or hazardous discharges to groundwater. As indicated in analyses presented in Chapter 4 of Volume 1 (e.g., Sections 4.3.1.1.4, 4.3.3.1.4, 4.4.3.1.4, 4.5.3.2.4, and 4.6.3.2.4), there would be no discernible impacts to groundwater or surface water quality at Hanford from operation of Hanford facilities that would support the nuclear infrastructure missions described in Section 1.2 of Volume 1.

A number of commentors also expressed concern that DOE would expose individuals in the Pacific Northwest to risks associated with importing of weapons-grade plutonium. None of the proposed alternatives involve the shipment of any weapons-grade plutonium to any port in the United States. Alternative 1 does postulate that DOE might decide at some point to import mixed oxide fuel from Europe to fuel FFTF. At this time, however, DOE has not proposed to import this fuel through any specific port. If DOE ultimately decides to import fuel from Europe, it would perform a separate NEPA analysis to select a port. This review would address all relevant potential impacts of overseas and inland water transportation, shipboard fires, package handling, land transportation, as well as safeguards and security associated with the import of SNR–300 mixed oxide fuel through a variety of specific candidate ports on the west and east coasts. It would take into account all public comments, including local resolutions, concerning the desirability of bringing mixed oxide fuel into the proposed alternative ports.

In the event that DOE decides to enhance its nuclear infrastructure, it would not expose any population to high, unacceptable risks under any alternative. Any transportation activities that would be conducted by DOE would comply with U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Transportation regulations. Associated transatlantic shipments would comply with International Atomic Energy Agency requirements. In Section J.6.2 of Volume 2, DOE reviewed the potential maximum impacts from the marine transportation of mixed oxide fuel from Europe to a representative military port (Charleston, South Carolina), and overland transportation to Hanford. Also in that section, the results of a bounding analysis show that the maximum potential radiological risks to the surrounding public from mixed oxide fuel shipments would be extremely small (e.g., less than 1 chance in a trillion for a latent cancer fatality per shipment from severe accidents at docks and in channels and less than 1 chance in 50 billion for a latent cancer fatality per shipment from overland highway accidents).

1.6 CHANGES FROM THE DRAFT NI PEIS

In response to comments on the Draft NI PEIS and as a result of information that was unavailable at the time of its issuance, this Final NI PEIS contains revisions and new information. These revisions and new information are indicated by sidebars. A brief discussion of the most important changes included in this Final NI PEIS is provided in the following paragraphs.

Chapter 1

Purpose and Need for Agency Action

As a result of public comments, additional discussion was incorporated to address DOE's production of medical, research, and industrial isotopes relative to global isotope production and availability. In addition, the discussion of the need for plutonium-238 production for space missions was expanded and updated to reflect the most recent planning guidance provided by NASA to DOE.

Issues Raised During the Public Comment Period on the Draft NI PEIS

Section 1.5, Issues Raised During the Public Comment Period on the Draft NI PEIS, was added to this Final NI PEIS.

Related NEPA Reviews

The Final NI PEIS was revised to add descriptions of the *Final Environmental Impact Statement, Management of Spent Nuclear Fuel from the K Basins at the Hanford Site, Richland, Washington* (DOE/EIS-0245F), and the *Environmental Assessment, Management of Hanford Site Non-Defense Production Reactor Spent Nuclear Fuel (*DOE/EA-1185). The impacts of these NEPA actions were factored into the assessment of potential cumulative impacts resulting from the NI PEIS proposed action.

This Final NI PEIS was also revised to reflect recent Records of Decision that have been issued for the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (DOE/EIS-0218F), the *Final Environmental Impact Statement for Treating Transuranic (TRU)/Alpha Low-Level Waste at the Oak Ridge National Laboratory, Oak Ridge, Tennessee* (DOE/EIS-0305), and the *Final Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (DOE/EIS-0306).

Changes from the Draft NI PEIS

Section 1.8, Changes from the Draft NI PEIS, was added to this Final NI PEIS.

Chapter 2

Transportation Requirements

Additional U.S. ports were named as candidates for receiving mixed oxide fuel from Europe.

Alternatives Considered and Dismissed

Information was provided to explain why the Isotope Production Facility at LANL, the Brookhaven LINAC (Linear Accelerator) Isotope Producer and the Alternating Gradient Synchrotron accelerator complex at Brookhaven National Laboratory, and CLWRs were not considered reasonable alternatives for the production of medical isotopes.

Information was also provided to explain why increasing the power levels at ATR and/or HFIR or installing rapid radioisotope retrieval systems would be insufficient to meet the long-term growth projection needs and therefore were dismissed as reasonable alternatives.

Preferred Alternative

The discussion of DOE's preferred alternative for accomplishing the proposed action, that is, Alternative 2, Use Only Existing Operational Facilities, Option 7, is included in this Final NI PEIS.

Summary of Environmental Impacts

Section 2.7 was revised in response to comments that it was difficult to compare environmental impacts among alternatives. Although estimates of the environmental impacts that would result from implementation of the

alternatives are the same as those in the Draft NI PEIS, the tables and accompanying text were reformatted for ease in comparing environmental impacts among alternatives and among options within alternatives. Section 2.7 was also revised to focus on incremental impacts that would result from implementation of the alternatives. Baseline environmental impacts were removed from the comparisons among alternatives and options. This information is now presented in Chapter 3.

Chapter 3

Affected Environment

Additional information was provided on the environmental baseline at each site, including graphics to more clearly illustrate existing surface water and groundwater conditions. Estimates of existing impacts for current HFIR/REDC operations were added to Sections 3.2.3.2 (Air Quality), 3.2.9.1.2 (Radiation Exposure and Risk), and 3.2.11.1 (Waste Inventories and Activities). Similarly, estimates for current ATR operations were added to Sections 3.3.3.2 (Air Quality), 3.3.9.1.2 (Radiation Exposure and Risk), and 3.3.11.1 (Waste Inventories and Activities). Estimates of existing impacts of maintaining FFTF in standby were added to Section 3.4.3.1 (Air Quality). Information was also provided on the impacts of the range fires affecting Hanford and INEEL during the summer of 2000. In addition, site data were updated to reflect recent measurements and analyses.

In response to public comments on the Draft NI PEIS, additional information on health studies conducted in the Hanford area was also incorporated.

Chapter 4

Air Quality

Stack parameters used for the air quality modeling were added. In response to public comment, estimates of the ambient air quality concentrations from FFTF sources were added to the deactivation section.

Water Resources

New water use and sanitary wastewater generation increments for REDC and FDPF were added to reflect the revised additional workforce required at these facilities and to be consistent with FMEF. Water use and wastewater generation rates for the new accelerator(s) and new research reactor alternatives were also revised. These changes were also incorporated into the waste management analyses.

Ecological and Cultural and Palentological Resources

These sections were updated to reflect that consultations concerning threatened and endangered species and cultural resources were conducted with appropriate Federal and state agencies. Consultations were also conducted with interested Native American tribes. No major issues were raised as a result of these consultations.

Socioeconomics

Section 4.3.1.1.8 was revised to reflect changes in the number of workers associated with FFTF operations and deactivation. The associated impacts on community services were also incorporated. In addition, the number of workers at the Oak Ridge Reservation was revised to reflect the entire site workforce rather than just the number of workers at the Oak Ridge National Laboratory.

Normal Operations

Based on more recent site data on occupational radiation exposure for workers at REDC, all worker health impacts for target processing at REDC, FMEF, and FDPF and for neptunium target storage at REDC, Chemical Processing Plant–651, and FMEF were updated. Also, low-energy accelerator source terms were modified to properly reflect normal operational emissions resulting in modifications to the population health impacts for all options of Alternative 3.

Facility Accidents

The high-energy accelerator analysis was redone to incorporate a more accurate revised source term, and the incremental risks for currently operating reactors were added to the tables. An additional analysis addressing industrial accidents was also performed and incorporated into Chapter 4.

Transportation

The neptunium inventory was revised to use the recently declassified actual inventory. The number of actual shipments from SRS to the processing facilities and the transportation risk estimates were modified accordingly.

Waste Management

The analysis for the Draft NI PEIS assumed that the waste generated from the processing of irradiated neptunium-237 targets is transuranic waste. However, as a result of comments received during the public comment period, DOE is considering whether the waste from processing of irradiated neptunium-237 targets should be classified as high-level radioactive waste and not transuranic waste. The Waste Management sections (i.e., Sections 4.3.1.1.13, 4.3.2.1.13, 4.3.3.1.13, and 4.4.3.1.13) were revised to reflect this different classification from what was assumed in the Draft NI PEIS.

Spent Nuclear Fuel Management

These sections were revised to quantify the generation of spent fuel from 35 years of operation and to state that dry spent nuclear fuel storage at the FFTF site is similar to NRC-approved methods currently being used for interim storage of commercial spent nuclear fuel. In addition, based on public comments, a reference was added about the K Basins spent fuel storage.

Cumulative Impacts

Cumulative impact tables in Section 4.8 were revised to present the contributions from each of the various site actions anticipated during the course of the operational period evaluated in this NI PEIS.

The air quality tables were also revised to incorporate the revised baseline from Chapter 3. In addition, waste management tables were revised to include the sites' treatment, storage, and disposal capacities for easier comparison of the waste generations by waste type to the waste management capacities at the sites.

Chapter 5

In response to public comments, a list of organizations that DOE contacted during the consultation process was added.

Volume 2

Summaries of the *NI Cost Report* and *NI Nonproliferation Impact Assessment* were added as Appendixes P and Q, respectively. NASA mission guidance correspondence was added as Appendix R.

Volume 3

Volume 3 of the NI PEIS was added to present the comments received during the public review period for the Draft NI PEIS and DOE's responses to these comments.