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Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement

Volume II
Appendices

September 1997

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Abstract:

The purpose of *the Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement* (SEIS-II) is to provide information on environmental impacts regarding the Department of Energy's (DOE) proposed disposal operations at WIPP. To that end, SEIS-II has been prepared to assess the potential impacts of continuing the phased development of WIPP as a geologic repository for the safe disposal of transuranic (TRU) waste. SEIS-II evaluates a Proposed Action, three Action Alternatives based on the waste management options presented in the *Final Waste Management Programmatic Environmental Impact Statement*, and two No Action Alternatives. The Proposed Action describes the treatment and disposal of the Basic Inventory of TRU waste over a 35-year period. The Basic Inventory is that waste currently permitted in WIPP based on current laws and agreements. The Action Alternatives propose the treatment of the Basic Inventory and an Additional Inventory as well as the transportation of the treated waste to WIPP for disposal over a 150- to 190-year period. The three Action Alternatives include the treatment of TRU waste at consolidation sites to meet WIPP planning-basis Waste Acceptance Criteria, the thermal treatment of TRU waste to meet Land Disposal Restrictions, and the treatment of TRU waste by a shred and grout process. The No Action Alternatives propose the dismantling and closure of WIPP and storage of the waste. One No Action Alternative proposes treating the waste thermally before placing it in retrievable storage.

SEIS-II evaluates environmental impacts resulting from the various treatment options; the transportation of TRU waste to WIPP using truck, a combination of truck and regular rail service, and a combination of truck and dedicated rail service; and the disposal of this waste in the repository. Evaluated impacts include those to the general environment and to human health. Additional issues associated with the implementation of the alternatives are discussed to provide further understanding of the decisions to be reached and to provide the opportunity for public input on improving DOE's Environmental Management Program.

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APPENDIX A
WASTE INVENTORY

This appendix provides information on the characteristics and quantities of transuranic (TRU) waste that may be disposed of at the Waste Isolation Pilot Plant (WIPP). This information is necessary for assessing the potential impacts from the transportation of TRU waste, from WIPP operations, and from the long-term performance of WIPP.

A.1 INTRODUCTION

TRU waste has been generated since the 1940s as part of the nuclear defense research and production activities of the Federal government. Several types of operations generate TRU waste: (1) nuclear weapons development and manufacturing, (2) prior plutonium recovery, (3) research and development, (4) environmental restoration, and decontamination and decommissioning activities, (5) waste management programs, and (6) testing and research at facilities under U.S. Department of Energy (DOE or the Department) contract.

Until about 1970, TRU waste, along with low-level waste, was disposed of in shallow trenches without an intent to retrieve it. In 1970, it was determined that TRU waste should be isolated and disposed of in a different manner than low-level waste. Thus, the Atomic Energy Commission, a DOE predecessor agency, adopted a policy requiring that waste containing TRU elements be placed in containers that could be retrieved from storage within 20 years.

The WIPP Land Withdrawal Act (LWA) (Public Law 102-579) limits (1) the volume of TRU waste that can be disposed of at WIPP to 175,600 cubic meters (6.2 million cubic feet); (2) the total activity of remote-handled (RH) TRU waste to 5.1 million curies (Ci); (3) the activity of RH-TRU waste averaged over the volume of a disposal container to 23 Ci per liter; and (4) the RH-TRU waste volume having a surface dose rate that exceeds 100 rem per hour to 5 percent. DOE and the State of New Mexico agreed to limit the volume of RH-TRU waste to no more than 7,080 cubic meters (250,000 cubic feet) (DOE 1981). This limit results in disposal of RH-TRU waste with total curies below the LWA limit. WIPP capacities for contact-handled (CH) TRU waste and RH-TRU waste are 168,500 cubic meters (5,950,000 cubic feet) and 7,080 cubic meters (250,000 cubic feet), respectively.

As explained in Chapter 2, TRU waste is broadly categorized to include (1) defense wastes of the type that were subject to previous WIPP-related National Environmental Policy Act (NEPA) reviews and (2) other defense and nondefense TRU wastes for which DOE retains management responsibility. TRU wastes analyzed in the *Final Environmental Impact Statement for the Waste Isolation Pilot Plant* (FEIS) (DOE 1980) and the *Final Supplement Environmental Impact Statement for the Waste Isolation Pilot Plant* (SEIS-I) (DOE 1990) included wastes resulting from defense activities and programs that were placed in retrievable storage pursuant to the 1970 Atomic Energy Commission policy. TRU wastes that were reasonably expected to be generated by ongoing defense activities and programs were also analyzed. For the purpose of this *Waste Isolation Pilot Plant Disposal Phase Draft Supplemental Environmental Impact Statement* (SEIS-II), this TRU waste inventory is referred to as the "Basic Inventory."

Defense and nondefense TRU wastes that had not previously been analyzed by FEIS and SEIS-I include (1) nondefense and commercial TRU waste, (2) defense TRU waste commingled with polychlorinated biphenyls (PCB), and (3) defense (and perhaps some nondefense) TRU waste disposed of prior to the Atomic Energy Commission policy of 1970. This TRU waste inventory is referred to as the "Additional Inventory."

A.1.1 Changes and New Information Since SEIS-I

The following major changes in information and assumptions regarding TRU waste differ from those used in SEIS-I (DOE 1990).

- Additional sites that have, or expect to generate, TRU waste have been identified. TRU waste volumes at these sites, however, account for only a small percent of the total TRU waste volume.

- Estimates of the future generation of TRU waste differ for some sites due to changes in those sites' missions.

- More detailed descriptions of the volume and physical characteristics of waste streams have been developed at many sites.

- More detailed descriptions of the radionuclide content of waste streams have been developed at many sites.

In SEIS-I, the volume of TRU waste was estimated at 159,000 cubic meters (5.6 million cubic feet) for CH-TRU waste and 2,690 cubic meters (95,000 cubic feet) for RH-TRU waste. These estimates were based on current volumes of stored waste and waste expected to be generated through the year 2013 (DOE 1990). The volume of TRU waste for the SEIS-II Basic Inventory is estimated at 135,000 cubic meters (4.7 million cubic feet) for CH-TRU waste and 35,000 cubic meters (1.2 million cubic feet) for RH-TRU waste. These estimates are based on current volumes of stored waste and waste expected to be generated through the year 2033. (More recent estimates of these volumes are presented in Appendix J.)

There is a high level of uncertainty and a current lack of consistent data regarding waste to be produced by future potential decontamination and decommissioning (D&D) and environmental restoration (ER) activities. DOE developed ER and D&D inventories through the 1996 *Baseline Environmental Management Report* (BEMR) (DOE 1996d). However, the BEMR data were not available in time to be included in the *WIPP Transuranic Waste Baseline Inventory Report, Revision 3* (BIR-3) (DOE 1996c) or SEIS-II. Battelle Columbus Laboratories, Bettis Atomic Power Laboratories (Bettis), and the Hanford Site (Hanford) reported D&D and ER projections that were included in BIR-3. Consistent with the *Compliance Certification Application for the Waste Isolation Pilot Plant* (DOE 1996f) (CCA), SEIS-II analyses used the waste volumes reported in BIR-3 and the *Integrated Data Base Report* (DOE 1994a) (IDB).

Revision 1 of BIR reported Hanford's submittal as approximately 46,000 cubic meters (1.6 million cubic feet) of projected RH-TRU waste, of which 43,000 cubic meters (1.5 million cubic feet) were called "suspect" RH-TRU waste due to insufficient information. Reevaluation of the 46,000 cubic meters (1.6 million cubic feet) of projected RH-TRU waste by Hanford personnel has resulted in a decrease of the reported projected RH-TRU waste to approximately 21,500 cubic meters (760,000 cubic feet) (through the year 2022) for BIR-2 and BIR-3. Additional evaluations of the reported Hanford RH-TRU waste volumes are ongoing, and the results will be reported in future revisions of BIR. Hanford now reports approximately 200 cubic meters (7,000 cubic feet) of stored RH-TRU wastes, with the rest of the waste still to be generated.

To conservatively estimate potential impacts, analyses in SEIS-II are based on the larger total volumes in the earlier BIR-3 (DOE 1996c), which incorporates the waste volumes of its second revision (BIR-2) (DOE 1995d), although analyses based on *The National Transuranic Waste Management Plan* (DOE 1996e) are also included (see Appendix J). The SEIS-II estimates, based on BIR-3, include 62,000 cubic meters (2.2 million cubic feet) of stored defense CH-TRU waste and 3,600 cubic meters (127,000 cubic feet) of stored defense RH-TRU waste. SEIS-II estimates also include 73,000 cubic meters (2.6 million cubic feet) of newly generated defense CH-TRU waste and 32,000 cubic meters (1.1 million cubic feet) of newly generated defense RH-TRU waste that will be generated through the year 2033.

A.1.2 Data Sources

Six main data sources were used to develop the TRU waste volume estimates, radionuclide inventories, and hazardous constituent inventories in SEIS-II. These six data sources are the following:

- *WIPP Transuranic Waste Baseline Inventory Report*, Revision 3 (BIR-3) (DOE 1996c), which incorporates by reference the waste volume estimates in the *WIPP Transuranic Waste Baseline Inventory Report*, Revision 2 (BIR-2) (DOE 1995d)
- *Integrated Data Base Report-1994: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections, and Characteristics* (IDB) (DOE 1994a)
- *Waste Isolation Pilot Plant Safety Analysis Report* (SAR) (DOE 1995c)
- *Waste Isolation Pilot Plant Safety Analysis Report* (SAR) (DOE 1997a)
- *Comment Responses and Revisions to the Resource Conservation and Recovery Act* (RCRA) *Part B Permit Application* (DOE 1996a)
- *Waste Acceptance Criteria for the Waste Isolation Pilot Plant* (WAC), Revision 5 (DOE 1996b)

The BIR-3 database contains detailed physical descriptions of TRU waste as well as TRU waste volumes and radionuclide inventories associated with individual waste streams at generator-storage sites. The database contains both currently stored waste volumes and waste volumes expected to be generated through the year 2022. BIR-3, both WIPP SARs, and the WIPP RCRA Part B Permit Application were used to estimate the inventory of hazardous constituents. Limits for several waste form characteristics, such as thermal power limits, were obtained from the planning-basis WAC.

Not all of the BIR-3 waste streams have associated radionuclide inventories. For example, only about 80 percent of the CH-TRU waste stream volumes and about 15 percent of the RH-TRU waste stream volumes have reported radionuclide inventories. Where possible, BIR-3 radionuclide data were used to analyze the impacts due to TRU waste handling, shipping, and accidents. To supplement missing radionuclide site information, however, an estimate of the total radionuclide inventory for a particular site as reported in the IDB was used. Because data presented in the IDB are site-wide, not waste-stream based, the IDB data were used to estimate the total radionuclide inventory of WIPP in the performance assessment calculations of long-term performance.

A.2 WASTE CHARACTERIZATION

DOE has developed waste matrix codes which organize waste streams by their physical and chemical properties. Over 900 waste streams listed in BIR-3 have been grouped into the following 11 waste matrix code groups or final TRU waste forms: combustible, filter, graphite, heterogeneous, inorganic nonmetal, lead/cadmium metal, uncategorized metal, salt, soil, solidified inorganic, and solidified organic (DOE 1996c). A brief description of each final TRU waste form is given in Table A-1.

Table A-1
Final TRU Waste Form Code Group Definitions

Final TRU Waste Form Code Group	Definition
Combustible	Debris that is approximately 95 percent or more, by volume, combustible materials. Examples of combustible debris are materials constructed of plastic, rubber, wood, paper, and cloth.
Filter	Debris that is approximately 50 percent or more, by volume, High Efficiency Particulate Air (HEPA) filters or additional filters constructed of more than one material type (e.g., metal, inorganic nonmetal, and combustibles).
Graphite	Debris that is approximately 95 percent or more, by volume, graphite-based solid materials. Graphite debris includes crucibles, graphite components, and pure graphite.
Heterogeneous	Debris that is at least 50 percent by volume materials that do not meet criteria for assignment into other categories. For example, waste that is a mixture of metal and combustible debris, neither of which comprises 95 percent or more of the waste by volume.
Inorganic nonmetal	Debris that is approximately 95 percent or more, by volume, inorganic nonmetal material. Examples of waste in this group include glass and ceramics.
Lead/cadmium metal	Debris that is approximately 95 percent or more, by volume, metal that contains bulk lead or cadmium as part of the matrix. Examples of this waste include glovebox parts with lead clad in stainless-steel or cadmium sheets.
Uncategorized metal	Debris that is approximately 95 percent or more, by volume, metal but either lacks sufficient information to enable characterization into one of the other categories or contains both lead and cadmium as part of the bulk matrix.
Salt	Debris that is at least 50 percent by volume salts. Stable pyrochemical salt is an example of this group.
Soil	Debris that is approximately 95 percent or more, by volume, soil. This includes sand, silt, and rock/gravel where rock/gravel volumes total less than 50 percent of the matrix.
Solidified inorganic	Debris that is at least 50 percent by volume inorganic process residues. This group includes solidified sludges and small particles.
Solidified organic	Debris that is at least 50 percent by volume organic process residues. These are defined as process residues with a base structure that is primarily organic. The matrix may contain some inorganic solids content such that approximately 20 percent by weight of the waste would remain as residue ash/solids following incineration. Examples include organic resins, organic sludges and solidified organic liquids.

A.2.1 Planning-Basis WAC

TRU waste must be certified to meet planning-basis WAC before it is transported to WIPP (DOE 1989). WAC established conditions that govern the physical, radiological, chemical composition, and packaging requirements of TRU waste. With a broader scope, WAC Revision 4 (DOE 1991) consolidated all of the requirements for TRU waste storage at WIPP into one document. WAC Revision 5 (DOE 1996b) was published in April 1996, providing an update to the requirements.

In developing TRU waste transportation and disposal volumes, the following criteria consistent with planning-basis WAC were incorporated:

- A TRUPACT-II's maximum gross weight must not exceed 8,730 kilograms (19,250 pounds) and the maximum gross weight of the canister of the RH-72B cask is 3,630 kilograms (8,000 pounds).
- The total gross weight for a truck shipment is 36,300 kilograms (80,000 pounds).
- The maximum thermal power (heat-generating capacity) is 40 watts (W) for a TRUPACT-II container and 300 W for an RH-TRU waste canister.
- The maximum plutonium-239 (Pu-239) equivalent activity (PE-Ci) for untreated CH-TRU waste is 80 PE-Ci for a drum, and 130 PE-Ci for a standard waste box. Untreated CH-TRU waste in 55-gallon drums may contain up to 1,800 PE-Ci of activity if overpacked in standard waste boxes or 10-drum overpacks. Drums containing solidified or vitrified CH-TRU waste may contain up to 1,800 PE-Ci of activity per drum. RH-TRU waste canisters may not exceed 1,000 PE-Ci.

Planning-basis WAC for the TRU waste weight, thermal power, and PE-Ci are discussed in the following sections. These factors are used to determine the number of shipments required for waste streams of varying densities and thermal properties.

A.2.1.1 Weight Limits for Packaging TRU Waste

Weight limits apply to the packaging of CH-TRU waste in 55-gallon drums, TRUPACT-IIs, and RH-72B casks. A CH-TRU waste drum shall not exceed 454 kilograms (1,000 pounds). This limit includes the weight of the drum itself, approximately 27 kilograms (60 pounds). Once the CH-TRU waste drum is loaded onto the TRUPACT-II, additional weight limits apply. In addition, the RH-72B cask is limited to a total waste canister payload of 3,630 kilograms (8,000 pounds).

Below are the weight values associated with TRUPACT-IIs and the tractor and trailer, as taken from the *CH-TRU Waste Packaging Optimization Report* (DOE 1995a):

- Maximum gross vehicle weight 36,300 kilograms (80,000 pounds)
- Maximum tractor and trailer weight 12,700 kilograms (28,000 pounds)
- Maximum loaded individual TRUPACT-II weight 8,730 kilograms (19,250 pounds)
- Average empty individual TRUPACT-II weight 5,760 kilograms (12,705 pounds)

- Weight of pallet, slip sheets, and guide tubes — 120 kilograms (265 pounds) per TRUPACT-II

- Seven-pack of empty dunnage drums weight — 190 kilograms (420 pounds)

Due to the maximum gross vehicle weight restriction, a shipment consisting of three TRUPACT-IIs per trailer, the maximum number of TRUPACT-IIs for a truck shipment, would not exceed the maximum TRUPACT-II weight limit. The average weight of the contents of a drum in this case, as indicated in Table A-2, is up to 142 kilograms (312 pounds).

Table A-2
CH-TRU Waste Shipping Weights [a]

Number of TRUPACT-IIs per Trailer	Number of Drums per Shipment	Payload per Shipment (kilograms)	Average Weight of Drum Contents (kilograms)	Dunnage Weight per TRUPACT-II (kilograms)
3	42	5,950	Up to 142	0
2	28	5,710	142 to 204	0
2	14	5,330	204 to 381	190
2	12	5,272	381 to 427 [b]	218

[a] Adapted from Table A-1 of the *CH-TRU Waste Packaging Optimization Report* (DOE 1995a).
[b] The maximum drum weight is 454 kilograms.

When two TRUPACT-IIs are shipped with no dunnage, the maximum TRUPACT-II payload for the shipment is equal to two fully loaded TRUPACT-IIs, 5,710 kilograms (12,590 pounds). For shipments of higher density waste, it may be necessary to use dunnage to meet the TRUPACT-II maximum weight restriction while allowing the drums to approach their maximum weight. Table A-2 shows two cases that evaluate the average drum weight when using dunnage. The first case involves a seven-pack of empty drums, to demonstrate the average case. The second case involves eight empty drums per TRUPACT-II, which nearly maximizes the allowable weight of the drum.

RH-TRU waste truck shipments are limited to one RH-72B cask by the maximum gross vehicle weight. Assuming the RH-72B cask is loaded with three drums and spacers, the maximum payload in an RH-72B cask is 3,629 kilograms (8,000 pounds). Because only one cask can be shipped by truck per shipment, the cask can always be maximally loaded. Below is a listing of other weights associated with the RH-72B cask, as taken from the *Safety Analysis Report for the RH-72B Waste Shipping Package* (DOE 1994c):

- Gross RH-72B cask weight — 20,412 kilograms (45,000 pounds)

- RH-72B outer cask weight — 12,647 kilograms (27,883 pounds)

- Inner vessel weight — 1,825 kilograms (4,023 pounds)

- Weight of impact limiters 2,311 kilograms (5,094 pounds)
- Loaded canister weight 3,629 kilograms (8,000 pounds)

Both the TRUPACT-II and the RH-72B cask can be maximally loaded when shipped by rail. Three TRUPACT-IIs or two RH-72B casks can be loaded onto a single, standard railcar.

A.2.1.2 Planning-Basis WAC Thermal Power Limits

Some heat is generated by TRU waste due to the decay energy of the different radioactive isotopes. The amount of heat generated in a given volume depends on the radionuclide activity and the average energy of the alpha particles as they are released during decay.

In addition to heat, hydrogen gas is generated when high-energy alpha particles strike polymers such as plastic. The amount of hydrogen gas generated is a function of the amount of energy deposited by ionizing radiation in the hydrogenous material present in the TRU waste. Thermal power limits have been established by the *TRUPACT Content Codes* (TRUCON) to ensure that the concentration of flammable gas within the innermost plastic bag of the waste configuration is less than 5 percent after a 60-day period (DOE 1994b). These thermal power levels, expressed in terms of watts per waste drum, have been developed as a surrogate for direct calculation of the ionizing energy deposition. The 60-day period is assumed to be the maximum time a container would remain in an unvented, loaded TRUPACT-II. As might be expected, the more layers of plastic used in packaging materials, the more restrictive the thermal power limit. Thermal power limits were obtained from Table 6-1 of TRUCON and are included later in this appendix (see Tables A-16 through A-18 and A-20 through A-23).

A.2.1.3 PE-Ci

The PE-Ci concept was developed to eliminate the dependency of radiological analyses on the specific radionuclide composition of TRU waste streams. The inhalation hazard of radionuclides was normalized to the hazard associated with Pu-239. Because SEIS-II evaluates the radiation hazard associated with individual radionuclides, PE-Ci values were used to evaluate whether the radionuclide inventory of specific waste matrix code groups meet the PE-Ci limits specified in planning-basis WAC (DOE 1996b). Additional detail on the PE-Ci concept can be obtained from Appendix B of the WIPP SAR (DOE 1995c and 1997a). Specific calculations for a mix of radionuclides are provided in Table A-2 of the WIPP SAR (DOE 1995c and 1997a).

PE-Ci values were calculated for all final TRU waste forms. Radionuclide weighting factors, derived from the normalized inhalation hazards, were taken from the planning-basis WAC (DOE 1996b). One final waste form, CH-TRU salt waste at Idaho National Engineering and Environmental Laboratory (INEEL), slightly exceeds the PE-Ci limits under Action Alternative 2. A value of 142.6 PE-Ci per cubic meter for 8.85 cubic meters (313 cubic feet) of stored waste was calculated. Under the volume reduction assumptions for thermal processing, this yields an equivalent value of 407.5 PE-Ci per cubic meter for 3.1 cubic meters (109 cubic feet) of processed waste. This waste stream would require a volume dilution to 3.3 cubic meters (117 cubic feet) to meet the PE-Ci limit.

All TRU waste volume calculations, including the PE-Ci calculations described above, were performed on integrated data for final TRU waste forms. Each final TRU waste form is comprised of several (8 to 10 on average) individual waste streams. In addition, residue data from Rocky Flats Environmental Technology Site (RFETS) included TRU waste volumes for nine final TRU waste forms. The radionuclide inventory, however, was only given as the radionuclide activity for the total volume. The treatment of the waste streams into final TRU waste forms provides the integrated values for the PE-Ci calculations. Because the radionuclide activity of the waste streams is averaged, however, situations may exist where the activity of a single waste stream, or the combination of particular waste streams, would be high enough to exceed the PE-Ci limits.

A.2.1.4 Fissile-Gram Equivalents

The WAC has addressed criticality concerns with the imposition of a limit on the amount of fissile-gram equivalents (FGE) of plutonium that can be loaded in an individual drum or shipped in a TRUPACT-II. Currently, the WAC plutonium limits are 200 FGE per drum and 325 FGE per TRUPACT-II. Fissile-gram equivalents of plutonium are calculated by multiplying weighting factors by the mass of plutonium isotopes and adding these products. The computational method and FGE factors are described in Table A-2 of the WIPP SAR (DOE 1995c and 1997a).

As noted in Section A.2.1.3, all TRU waste volume calculations were performed on integrated data to obtain final TRU waste forms. Each final TRU waste form is comprised of 8 to 10 individual waste streams. With the exception of the RFETS residue wastes, all integrated waste forms meet the FGE requirements of the WAC. Based on the waste stream characteristics of the RFETS data and the gas generation limits given in Table 6.1 of the TRUCON (1994b), the RFETS residue wastes would require 1,205 shipments to WIPP.

The U.S. Nuclear Regulatory Commission (NRC) has recently approved the use of pipe overpacks in the TRUPACT-II shipping container (NRC 1997). This change will allow a pipe overpack in a drum to contain 200 FGE; therefore, if the TRUPACT-II is allowed to carry 14 such drums, the TRUPACT-II limit would be 2,800 FGE. DOE now estimates (Rivera 1997) that residue cleanup activities at RFETS will generate 45,800 drums of waste, of which approximately 22,000 drums will contain pipe overpacks.

RFETS PLUTONIUM RESIDUES

DOE is currently preparing an environmental impact statement on the management of certain plutonium residues and scrub alloy stored at RFETS, as announced in a Notice of Intent (NOI) on November 19, 1996 (61 Federal Register [FR] 58866). The SEIS-II Basic Inventory includes approximately 4,200 cubic meters (148,322 cubic feet) of plutonium residues at RFETS (see Table A-17). This residual volume includes approximately 3,000 kilograms (6,614 pounds) of plutonium, which is consistent with the estimated plutonium activity at RFETS in the NOI. The NOI also states the RFETS residues have an estimated mass of 106,600 kilograms (235,013 pounds), but about 42,300 kilograms (93,255 pounds) would not meet the required safeguards for disposal. The NOI further notes the presence of plutonium scrub alloy at RFETS (approximately 700 kilograms [1,543 pounds] with 200 curies of plutonium) and an additional 18,400 kilograms (40,565 pounds) of plutonium residues stored among SRS, Hanford, LLNL, and LANL. The BIR-3, which were used as the basis for the SEIS-II inventory estimates, contain no specific information on scrub alloys or on plutonium residues at other sites. Because the information on these materials is considered uncertain, neither has been included for consideration in SEIS-II nor has their final disposition been evaluated or determined. SEIS-II assumes that plutonium residues at RFETS can be transported to WIPP using pipe overpacks. Further discussion has been provided in Section A.2.1.4.

Although the use of pipe overpacks has been approved by the NRC, it has not yet been integrated into the WAC. For the purpose of analyses in SEIS-II, it has been assumed that the pipe overpack will be incorporated into the WAC. Therefore, approximately 1,205 shipments will be required to transport the RFETS residue waste pipe overpacks to the WIPP site.

A.3 WASTE VOLUMES

BIR-3 contains detailed information on waste streams and volumes according to generator-storage site for the following: (1) TRU waste currently in retrievable storage (the stored volume) and (2) TRU waste expected to be generated in 28 years (the projected volume). In the context of SEIS-II, "TRU waste volumes" refers to the total TRU waste volume disposed of over the lifetime of WIPP. To be conservative, SEIS-II analyses were based upon an upper TRU waste generation limit taken from BIR-3.

A.3.1 Basic Inventory Volumes

Under the Proposed Action, it is assumed that WIPP operations would begin in 1998 and continue for 35 years, ending TRU waste receipt operations in 2033. Projected TRU waste volumes ($V_{projected}$) are given in BIR-3; however, these values reflect volumes estimated in the year 2022. To estimate the total TRU waste inventory volume at a generator-storage site in 2033, the following calculation was used:

$$V_{site} = V_{stored} + [38\,(R_{generate})] \qquad \text{(Equation A-1)}$$

where

V_{site} = estimated TRU waste volume through the year 2033

V_{stored} = TRU waste volume stored at the generator-storage site through 1995

38 = the number of years of waste generation (35 plus 3 years until 1998)

$R_{generate}$ = ($V_{projected}$) / 28 years; this is the TRU waste volume generation rate

The Basic Inventory TRU waste volume for each generator-storage site is given in Table 2-2, presented in this appendix as Table A-3. These are the final TRU waste form volumes based on the minimum level of treatment necessary to meet planning-basis WAC. Generator-storage site volumes for the year 2022 are provided solely for comparison with the volumes in BIR-3; they are not used in further calculations.

A.3.2 Additional Inventory Volumes

TRU waste in the Additional Inventory includes the following: (1) TRU waste commingled with PCBs at Hanford, INEEL, and Mound Plant (Mound); (2) commercial/nondefense waste at ARCO Medical Products Company (ARCO), West Valley Demonstration Project (WVDP), Oak Ridge National Laboratory (ORNL), Lawrence Berkeley Laboratory (LBL), and Knolls Atomic Power Laboratory (Knolls); and (3) previously disposed of TRU waste at a number of sites. These volumes are shown in Table 2-3, presented here as Table A-4.

Table A-3
Basic Inventory TRU Waste Volumes [a]

Site [d]	Stored (1995) (cubic meters) CH-TRU	Stored (1995) (cubic meters) RH-TRU	Estimated Total through 2022 [b] (cubic meters) CH-TRU	Estimated Total through 2022 [b] (cubic meters) RH-TRU	Estimated Total through 2033 [c] (cubic meters) CH-TRU	Estimated Total through 2033 [c] (cubic meters) RH-TRU
Hanford Site (Hanford)	12,000	200	46,000	22,000	57,000	29,000
Los Alamos National Laboratory (LANL)	11,000	94	18,000	190	21,000	230
Idaho National Engineering and Environmental Laboratory (INEEL)	28,000	220	28,000	220	28,000	220
Argonne National Laboratory - West (ANL-W)	7	19	750	1,300	1,000	1,700
Argonne National Laboratory - East (ANL-E)	25	--- [e]	150	---	200	---
Savannah River Site (SRS)	2,900	---	9,600	---	12,000	---
Rocky Flats Environmental Technology Site (RFETS)	4,900	---	9,300	---	11,000	---
Oak Ridge National Laboratory (ORNL)	1,300	2,500	1,600	2,900	1,700	3,100
Lawrence Livermore National Laboratory (LLNL)	230	---	940	---	1,200	---
Nevada Test Site (NTS)	620	---	630	---	630	---
Mound Plant (Mound)	300	---	300	---	300	---
Bettis Atomic Power Laboratory (Bettis)	---	---	120	7	170	9
Sandia National Laboratories - Albuquerque (SNL)	7	---	14	---	17	---
Paducah Gaseous Diffusion Plant (PGDP)	---	---	6	---	8	---
U.S. Army Materiel Command (USAMC)	3	---	3	---	3	---
Energy Technology Engineering Center (ETEC)	2	6	2	7	2	7
University of Missouri Research Reactor (U of Mo)	1	---	1	---	1	---
Ames Laboratory - Iowa State University (Ames)	---	---	1	---	1	---
Battelle Columbus Laboratories (BCL)	---	580	---	580	---	580
Totals	**62,000**	**3,600**	**116,000**	**27,000**	**135,000**	**35,000**

[a] The inventory for SEIS-II is based on BIR-3 (DOE 1996c), which takes into account potential thermal treatment at some sites. The thermal treatment is not necessarily for PCB-commingled waste. Volumes have been rounded. Actual totals may differ due to rounding.
[b] Post-1970 defense TRU waste volumes through 2022 are estimated in BIR-2 (1995d).
[c] The Proposed Action, described in Chapter 3, is based on operation of WIPP for 35 years through 2033.
[d] Sites in boldface were included in SEIS-I. INEEL and ANL-W are located near each other and are counted as a single site in SEIS-II; however, ANL-W is listed separately to indicate its contribution to the inventory.
[e] Dashes indicate no waste.

BIR-3 contains estimates of the total volume of previously disposed of TRU waste by site; however, it does not divide these wastes into CH-TRU and RH-TRU waste categories. The ratio of CH-TRU to RH-TRU waste volumes from the Basic Inventory, therefore, was used to divide the previously disposed of TRU waste into CH-TRU and RH-TRU waste categories on a site-by-site basis. In addition, some site information exists for TRU waste commingled with PCBs, commercial, and nondefense TRU waste in BIR-3. The volumes of these types of waste, expected to be generated by 2033, are also given in Table A-4.

A.3.3 Waste Volumes for the Proposed Action

Under the Proposed Action, CH-TRU waste would be consolidated at 10 sites and RH-TRU waste consolidated at four sites before shipment to WIPP. Figure 3-1 identifies the treatment sites for the Proposed Action. Table 3-1, presented here as Table A-5, identifies the treatment sites and

Table A-4
Additional Inventory TRU Waste Volumes [a, b]

Site [c]	PCB (cubic meters)		Commercial/Nondefense (cubic meters)		Previously Disposed of (cubic meters)		Total (cubic meters)	
	CH-TRU	RH-TRU	CH-TRU	RH-TRU	CH-TRU	RH-TRU	CH-TRU	RH-TRU
Hanford Site (Hanford)	240	---	---	---	63,000	1,000	63,000	1,000
Los Alamos National Laboratory (LANL)	---	---	---	---	14,000	120	14,000	120
Idaho National Engineering and Environmental Laboratory (INEEL)	460	---	---	---	57,000	440	57,000	440
Savannah River Site (SRS)	---	---	---	---	4,900	---	4,900	---
Oak Ridge National Laboratory (ORNL)	---	---	5	---	61	120	66	120
Mound Plant (Mound)	19	---	---	---	---	---	19	---
Sandia National Laboratories - Albuquerque (SNL)	---	---	---	---	1	---	1	---
ARCO Medical Products Company (ARCO)	---	---	1	---	---	---	1	---
Knolls Atomic Power Laboratory (Knolls)	---	---	---	81	---	---	---	81
Lawrence Berkeley Laboratory (LBL)	---	---	2	---	---	---	2	---
West Valley Demonstration Project (WVDP)	---	---	190	370	---	1,400	190	1,700
Totals	**720**	**---**	**200**	**450**	**138,000**	**3,100**	**139,000**	**3,500**

[a] The volume of TRU waste represents the 1995 existing and projected waste (DOE 1995d). The thermal treatment, though, is not necessarily for PCB-commingled waste.
[b] Actual totals may differ due to rounding.
[c] Sites in boldface also store post-1970 defense TRU waste, see Table 2-2. The remaining four sites currently have no post-1970 defense TRU waste.

Table A-5
TRU Waste Volumes (Basic Inventory) for the Proposed Action [a]

Site [d]	Site Volume Through 2033 (cubic meters)		Pretreatment Consolidated Volume [b] (cubic meters)		Post-Treatment Disposal Volume (cubic meters)		Disposal Volume (cubic meters) [c]
	CH-TRU	RH-TRU	CH-TRU	RH-TRU	CH-TRU [e]	RH-TRU [f]	RH-TRU
Hanford Site (Hanford)	57,000	29,000	57,000	29,000	57,000	42,000	2,800
Los Alamos National Laboratory (LANL)	21,000	230	21,000	230	21,000	330	330
Idaho National Engineering and Environmental Laboratory (INEEL) [g]	28,000	220	29,000	2,000	30,000	2,800	2,800
Argonne National Laboratory - West (ANL-W)	1,000	1,700	--- [h]	---	---	---	---
Argonne National Laboratory - East (ANL-E)	200	---	200	---	200	---	---
Savannah River Site (SRS)	12,000	---	12,000	---	12,000	---	---
Rocky Flats Environmental Technology Site (RFETS)	11,000	---	11,000	---	17,000	---	---
Oak Ridge National Laboratory (ORNL)	1,700	3,100	1,800	3,700	1,900	5,300	1,100
Lawrence Livermore National Laboratory (LLNL)	1,200	---	1,200	---	1,200	---	---
Nevada Test Site (NTS)	630	---	630	---	630	---	---
Mound Plant (Mound)	300	---	300	---	340	---	---
Bettis Atomic Power Laboratory (Bettis)	170	9	---	---	---	---	---
Sandia National Laboratories - Albuquerque (SNL)	17	---	---	---	---	---	---
Paducah Gaseous Diffusion Plant (PGDP)	8	---	---	---	---	---	---
U.S. Army Materiel Command (USAMC)	3	---	---	---	---	---	---
Energy Technology Engineering Center (ETEC)	2	7	---	---	---	---	---
University of Missouri Research Reactor (U of Mo)	1	---	---	---	---	---	---
Ames Laboratory - Iowa State University (Ames)	1	---	---	---	---	---	---
Battelle Columbus Laboratories (BCL)	---	580	---	---	---	---	---
Total	**135,000**	**35,000**	**135,000**	**35,000**	**143,000 [i]**	**50,000**	**7,080 [j]**
Disposal Volume Allowed by WIPP LWA and Agreement for Consultation and Cooperation (C & C)	**---**	**---**	**---**	**---**	**168,500 [i]**	**7,080**	**---**

[a] The inventory for SEIS-II is based on BIR-3, which took into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment does not necessarily include PCB-commingled waste. Volumes have been rounded. Actual totals may differ due to rounding. The site volumes through 2033 match the final columns on Table 2-2.

[b] Volumes include consolidation of waste as indicated on Figure 3-1.

[c] All LANL and INEEL RH-TRU waste is assumed to be disposed of; RH-TRU waste disposed of for Hanford would be approximately 2,800 cubic meters; that for ORNL would be 1,100 cubic meters (after consolidation).

[d] Sites in boldface were included in SEIS-I.

[e] Post-treatment volumes have been adjusted to meet packaging and transportation requirements to meet the planning-basis WAC.

[f] Values represent WIPP emplacement volumes, except for Hanford and ORNL.

[g] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, are counted as one site in SEIS-II.

[h] Dashes indicate no waste.

[i] Though 143,000 cubic meters of CH-TRU waste are part of the Basic Inventory, additional CH-TRU waste may become a part of that inventory should RCRA or Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) action lead to retrieval of previously disposed of waste. Therefore, SEIS-II assesses the impact of the entire disposal volume allowed, 168,500 cubic meters.

[j] The Waste Isolation Pilot Plant Land Withdrawal Act (LWA) limits the total RH-TRU waste curie content of WIPP to 5.1 million. Under the Proposed Action, the total curie content associated with the 7,080 cubic meters is less than 1 million curies.

waste volumes for the Proposed Action. These volumes are projected for the year 2033. Table A-5 differs from Table 3-1 in that an additional column entitled "Disposal Volume" for RH-TRU waste has been added.

The volumes in BIR-3 represent WIPP disposal volumes. They do not, however, account for volume dilution required for some CH-TRU waste streams in order to meet thermal power (hydrogen gas generation) limits for transportation in the TRUPACT-II containers. The CH-TRU waste volumes in the "Post-Treatment Disposal Volumes" column in Table A-5 include volume expansion due to packaging to meet thermal power limits. These are the anticipated TRU waste volumes that would be disposed of at WIPP.

The volume expansion for CH-TRU waste is based on the thermal loading (W/cubic meter) of the final waste form and on limits to the thermal loading per container as set by TRUCON. The TRU waste radioisotopic inventory from BIR-3 was used to determine the thermal loading for each site on a final waste form basis. If the calculated thermal power exceeds the thermal power limit, then volume expansion is required. Less than 4 percent of the stored CH-TRU waste (other than the RFETS residues) requires volume expansion to meet thermal power limits, provided that plastic wrap is not used when the waste drums are filled (bagless posting). When the RFETS residue waste is considered, this total climbs to nearly 9 percent of the waste. If bagless posting is not used, disposal volumes for CH-TRU waste nearly double.

To determine the volume due to expansion ($V_{Expansion}$), the thermal output for the final TRU waste form ($W_{Calculated}$) in watts per cubic meter is divided by the watt limit per cubic meter for the final TRU waste form (W_{Limit}). $W_{Calculated}$ is calculated from the radionuclide inventory waste stream. W_{Limit} values were obtained from Table 6-1 in TRUCON and converted from watts per drum to watts per cubic meter to facilitate calculations and comparisons (DOE 1994b). If the fraction ($W_{Calculated}/W_{limit}$) is greater than 1, it is multiplied by the initial TRU waste volume ($V_{Initial}$) before packaging. Otherwise, the volume due to expansion would remain the same as the volume before packaging. The equation is as follows:

$$\text{For } \left(\frac{W_{Calculated}}{W_{Limit}}\right) > 1;\ V_{Expansion} = \left(\frac{W_{Calculated}}{W_{Limit}}\right) \times V_{Initial} \qquad \text{(Equation A-2a)}$$

$$\text{For } \left(\frac{W_{Calculated}}{W_{Limit}}\right) < 1;\ V_{Expansion} = V_{Initial} \qquad \text{(Equation A-2b)}$$

Thermal processing is assumed to destroy the materials that generate hydrogen gas, so volume expansion due to thermal power is not applied to thermally processed waste. The Savannah River Site (SRS) has indicated that it would package or treat waste as required to meet planning-basis WAC, including gas generation limitations (Williams 1996). The SRS operations are expected to include both thermal processing and transferal of some high-activity waste from the CH-TRU waste to the RH-TRU waste category. It is assumed that there is no net increase or decrease in the volume of CH-TRU waste and RH-TRU waste at the SRS as a result of this processing.

With the RH-TRU waste volume limit at WIPP of 7,080 cubic meters (250,000 cubic feet), the volume disposed of was calculated using the capacity of the waste containers rather than the volume of the waste within the containers. An RH-TRU waste canister has a volume capacity of 0.89 cubic meter (32 cubic feet); therefore, the volume disposed of for one canister is 0.89 cubic meter (32 cubic feet). Three drums would be placed in an RH-TRU waste canister.

It was assumed that all RH-TRU waste would be contained in 55-gallon drums prior to being inserted into RH-TRU waste canisters, and would remain in place by spacers within the canister. The RH-TRU waste canister would, therefore, account for only 0.624 cubic meter (22 cubic feet) of RH-TRU waste at a site. This type of packaging would require approximately 1.43 times as much volume in WIPP as the volumes of waste stored in drums at the sites. This assumption results in an overestimated total volume disposed of relative to current storage practices. Also, for the purpose of the Proposed Action analysis, it was assumed that all sites except Hanford and ORNL are able to send all RH-TRU waste to WIPP. The waste would go to a treatment site and then to WIPP. Hanford and ORNL would only send about 6.7 percent of their RH-TRU waste to WIPP under the Proposed Action. As with CH-TRU waste, the RH-TRU waste disposal volumes are given in the "Post-Treatment Disposal Volumes" column in Table A-5.

A.3.4 Waste Volumes for Action Alternative 1

Under Action Alternative 1, CH-TRU waste would be consolidated at 10 sites and RH-TRU waste consolidated at four sites before shipment to WIPP based on the *Final Waste Management Programmatic Environmental Impact Statement* (WM PEIS) (DOE 1997b) Decentralized option. The treatment sites under Action Alternative 1 are shown in Figure 3-2. The CH-TRU waste volumes for Action Alternative 1 are given in Table 3-2, presented here as Table A-6. Similarly, the RH-TRU waste volumes under Action Alternative 1 are given in Table 3-3, presented here as Table A-7. The most notable difference under this alternative, as compared to the Proposed Action, is that all of the TRU waste identified in Table A-4 would be sent to WIPP, except for the TRU waste commingled with PCBs.

Under Action Alternative 1, it is assumed that all waste is packaged to meet planning-basis WAC requirements. Table A-6 columns containing CH-TRU waste post-treatment disposal volumes account for the volume expansion that is assumed necessary to meet thermal power limits; therefore, these volumes would be disposed of at WIPP. Tables A-6 and A-7 are divided into columns which identify the amount of waste under the Basic Inventory and Additional Inventory assumed under this alternative.

A.3.5 Waste Volumes for Action Alternative 2

Three important differences impact waste volumes under Action Alternative 2 as compared to the Proposed Action. These differences are: (1) all RH-TRU waste would be disposed of at WIPP rather than being limited to 7,080 cubic meters (250,000 cubic feet), (2) all of the waste identified in Tables A-3 and A-4 would be sent to WIPP, and (3) all waste would be subjected to thermal treatment to meet RCRA Land Disposal Restrictions (LDR) for hazardous constituents. A 65-percent reduction in the TRU waste volume to be disposed of was assumed due to LDR thermal treatment of both CH-TRU and RH-TRU waste (DOE 1995b). In an operational sense, the volume change for thermal processing would depend heavily on the physical characteristics of the individual waste stream.

Table A-6
CH-TRU Waste Volumes for Action Alternative 1 [a]

Site [d]	Site Volume Through 2033 (cubic meters) Basic Inventory	Site Volume Through 2033 Additional Inventory [e]	Site Volume Through 2033 Total Inventory	Pretreatment Consolidated Volume [b] (cubic meters) Basic Inventory	Pretreatment Consolidated Volume Additional Inventory [e]	Pretreatment Consolidated Volume Total Inventory	Post-Treatment Disposal Volume [c] (cubic meters) Basic Inventory	Post-Treatment Disposal Volume Additional Inventory [e]	Post-Treatment Disposal Volume Total Inventory
Hanford Site (Hanford)	57,000	63,000	120,000	57,000	63,000	120,000	57,000	63,000	120,000
Los Alamos National Laboratory (LANL)	21,000	14,000	35,000	21,000	14,000	35,000	21,000	14,000	35,000
Idaho National Engineering and Environmental Laboratory (INEEL) [f]	28,000	57,000	85,000	30,000	57,000	86,000	30,000	57,000	87,000
Argonne National Laboratory - West (ANL-W)	1,000	--- [g]	1,000	---	---	---	---	---	---
Argonne National Laboratory - East (ANL-E)	200	---	200	200	---	200	200	---	200
Savannah River Site (SRS)	12,000	4,900	17,000	12,000	4,900	17,000	12,000	4,900	17,000
Rocky Flats Environmental Technology Site (RFETS)	11,000	---	11,000	11,000	---	11,000	17,000	---	17,000
Oak Ridge National Laboratory (ORNL)	1,700	56	1,700	1,800	260	2,100	1,800	260	2,100
Lawrence Livermore National Laboratory (LLNL)	1,200	---	1,200	1,200	---	1,200	1,200	---	1,200
Nevada Test Site (NTS)	630	---	630	630	---	630	630	---	630
Mound Plant (Mound)	300	---	300	300	---	300	340	---	340
Bettis Atomic Power Laboratory (Bettis)	170	---	170	---	---	---	---	---	---
Sandia National Laboratories - Albuquerque (SNL)	17	1	18	---	---	---	---	---	---
Paducah Gaseous Diffusion Plant (PGDP)	8	---	8	---	---	---	---	---	---
U.S. Army Materiel Command (USAMC)	3	---	3	---	---	---	---	---	---
Energy Technical Engineering Center (ETEC)	2	---	2	---	---	---	---	---	---
University of Missouri Research Reactor (U of Mo)	1	---	1	---	---	---	---	---	---
Ames Laboratory - Iowa State University (Ames)	1	---	1	---	---	---	---	---	---
ARCO Medical Products Company (ARCO)	---	1	1	---	---	---	---	---	---
Lawrence Berkeley Laboratory (LBL)	---	2	2	---	---	---	---	---	---
West Valley Demonstration Project (WVDP)	---	190	190	---	---	---	---	---	---
Total	**135,000**	**139,000**	**273,000**	**135,000**	**138,000**	**273,000**	**143,000**	**138,000**	**281,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily for PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.

[b] Volumes include consolidation of waste as indicated on Figure 3-2.

[c] Post-treatment volumes have been adjusted to meet packaging and transportation criteria.

[d] Sites in boldface were included in SEIS-I.

[e] TRU waste commingled with PCBs is not included because there would be no thermal treatment under this alternative.

[f] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.

[g] Dashes indicate no waste.

Table A-7
RH-TRU Waste Volumes for Action Alternative 1 [a]

Site [d]	Site Volume Through 2033 (cubic meters)			Pretreatment Consolidated Volume [b] (cubic meters)			Post-Treatment Disposal Volume [c] (cubic meters)		
	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory
Hanford Site (Hanford)	29,000	1,000	30,000	29,000	1,000	30,000	42,000	1,500	43,000
Los Alamos National Laboratory (LANL)	230	120	350	230	120	350	330	170	490
Idaho National Engineering Laboratory (INEEL) [f]	220	440	660	2,000	440	2,400	2,800	630	3,400
Argonne National Laboratory - West (ANL-W)	1,700	--- [g]	1,700	---	---	---	---	---	---
Oak Ridge National Laboratory (ORNL)	3,100	120	3,200	3,700	2,000	5,600	5,200	2,700	8,000
Bettis Atomic Power Laboratory (Bettis)	9	---	9	---	---	---	---	---	---
Energy Technical Engineering Center (ETEC)	7	---	7	---	---	---	---	---	---
Battelle Columbus Laboratories (BCL)	580	---	580	---	---	---	---	---	---
Knolls Atomic Power Laboratory (Knolls)	---	80	80	---	---	---	---	---	---
West Valley Demonstration Project (WVDP)	---	1,700	1,700	---	---	---	---	---	---
Total	**35,000**	**3,500**	**39,000**	**35,000**	**3,500**	**39,000**	**50,000**	**5,000**	**55,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily of PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.
[b] Volumes include consolidation of waste as indicated on Figure 3-2.
[c] Post-treatment volumes have been adjusted to meet packaging and transportation criteria.
[d] Sites in boldface were included in SEIS-I.
[e] Additional Inventory (no TRU waste commingled with PCBs is included).
[f] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.
[g] Dashes indicate no waste.

Using an aggregate volume reduction factor of 0.35 for all waste treated by thermal processing, the resulting disposal volume for CH-TRU waste was calculated as:

$$V_{Disposal} = (0.35)\, V_{Consolidated} \qquad \text{(Equation A-3)}$$

For RH-TRU waste, the disposal volume resulting from thermal treatment was calculated in a similar manner. A volume expansion factor of 1.43 was applied to account for the placement of three waste drums in an RH-TRU waste canister (Equation A-4).

$$V_{Disposal} = (0.35)\, V_{Consolidated} \times 1.43 \qquad \text{(Equation A-4)}$$

$V_{Consolidated}$ refers to the TRU waste volume prior to thermal treatment. $V_{Disposal}$ refers to the TRU waste volume resulting from thermal treatment that would be disposed of at WIPP.

Thermal processing produces waste in the form of a slag. A density change assumption, therefore, is made such that a 55-gallon drum containing the slag would weigh 454 kilograms (1,000 pounds). Waste density values are used in the determination of the number of shipments (Section A.3.9).

See Table A-2 for the CH-TRU waste average drum weights used to determine the number of shipments.

Three separate waste treatment site options, based on the WM PEIS Regionalized 2, Regionalized 3, and Centralized Alternatives exist under Action Alternative 2. These subalternatives are Action Alternative 2A, 2B, and 2C, respectively. Both the type of waste treatment and the waste volume remain the same for all the Action Alternative 2 subalternatives; only the locations of the waste treatment sites are different. The treatment sites for Action Alternative 2A, 2B, and 2C are indicated in Figures 3-3, 3-4, and 3-5, respectively.

CH-TRU waste volumes for treatment sites under Action Alternative 2A, based on the WM PEIS Regionalized 2 Alternative, are shown in Table 3-4 and presented here as Table A-8. CH-TRU waste volumes for treatment sites under Action Alternative 2B, based on the WM PEIS Regionalized 3 Alternative, are shown in Table 3-6 and presented here as Table A-9. CH-TRU waste volumes for treatment sites under Action Alternative 2C, based on the WM PEIS Centralized Alternative, are shown in Table 3-8 presented here as Table A-10.

RH-TRU waste treatment sites are the same for each of the three WM PEIS waste treatment scenarios. The corresponding RH-TRU waste volumes under Action Alternative 2 are given in Tables 3-5, 3-7, and 3-9. These tables are consolidated here in Table A-11. The total volume of RH-TRU waste to be disposed of at WIPP would be 19,000 cubic meters (670,000 cubic feet). Waste volumes for Action Alternative 2 are all projected to the year 2033.

Accounting both for the large volume reduction factor after waste treatment and for placing three 55-gallon drums into each RH-TRU waste canister, there would be 12,400 cubic meters (438,000 cubic feet) more RH-TRU waste disposed of under Action Alternative 2 than under the Proposed Action.

A.3.6 Waste Volumes for Action Alternative 3

Under Action Alternative 3, CH-TRU waste would be consolidated at five sites and RH-TRU waste would be consolidated at two sites before shipment to WIPP. Figure 3-6 shows the treatment sites under this alternative. CH-TRU waste volumes under Action Alternative 3 are given in Table 3-10 and presented here as Table A-12. The RH-TRU waste volumes under Action Alternative 3 are given in Table 3-11 and presented here as Table A-13. Both the Basic Inventory (Table A-3) and the Additional Inventory, with the exception of TRU waste commingled with PCBs (Table A-4), are sent to WIPP. Shred and grout would be the dominant waste treatment process.

The post-treatment disposal volumes of CH-TRU waste (Table A-12) indicate the volume expansions assumed necessary to meet thermal power limits and due to the shred and grout process. These volumes would be disposed of at WIPP. Post-treatment disposal volumes of RH-TRU waste (Table A-13) account for volume adjustments pertaining to the placement of three 55-gallon drums into the RH-TRU waste canisters and volume expansion from the shred and grout process.

Table A-8
CH-TRU Waste Volumes for Action Alternative 2A [a]

Site [d]	Site Volume Through 2033 (cubic meters) Basic Inventory	Site Volume Through 2033 (cubic meters) Additional Inventory [e]	Site Volume Through 2033 (cubic meters) Total Inventory	Pretreatment Consolidated Volume [b] (cubic meters) Basic Inventory	Pretreatment Consolidated Volume [b] (cubic meters) Additional Inventory [e]	Pretreatment Consolidated Volume [b] (cubic meters) Total Inventory	Post-Treatment Disposal Volume [c] (cubic meters) Basic Inventory	Post-Treatment Disposal Volume [c] (cubic meters) Additional Inventory [e]	Post-Treatment Disposal Volume [c] (cubic meters) Total Inventory
Hanford Site (Hanford)	57,000	63,000	120,000	59,000	63,000	122,000	21,000	22,000	43,000
Los Alamos National Laboratory (LANL)	21,000	14,000	35,000	21,000	14,000	35,000	7,400	4,900	12,000
Idaho National Engineering and Environmental Laboratory (INEEL) [f]	28,000	57,000	85,000	30,000	57,000	87,000	10,000	31,000	41,000
Argonne National Laboratory - West (ANL-W)	1,000	--- [g]	1,000	---	---	---	---	---	---
Argonne National Laboratory-East (ANL-E)	200	---	200	---	---	---	---	---	---
Savannah River Site (SRS)	12,000	4,900	17,000	14,000	5,000	20,000	5,000	1,800	6,800
Rocky Flats Environmental Technology Site (RFETS)	11,000	---	11,000	11,000	---	11,000	3,800	---	3,800
Oak Ridge National Laboratory (ORNL)	1,700	66	1,700	---	---	---	---	---	---
Lawrence Livermore National Laboratory (LLNL)	1,200	---	1,200	---	---	---	---	---	---
Nevada Test Site (NTS)	630	---	630	---	---	---	---	---	---
Mound Plant (Mound)	300	20	320	---	---	---	---	---	---
Bettis Atomic Power Laboratory (Bettis)	170	---	170	---	---	---	---	---	---
Sandia National Laboratories - Albuquerque (SNL)	17	1	18	---	---	---	---	---	---
Paducah Gaseous Diffusion Plant (PGDP)	8	---	8	---	---	---	---	---	---
U.S. Army Materiel Command (USAMC)	3	---	3	---	---	---	---	---	---
Energy Technical Engineering Center (ETEC)	2	---	2	---	---	---	---	---	---
University of Missouri Research Reactor (U of Mo)	1	---	1	---	---	---	---	---	---
Ames Laboratory - Iowa State University (Ames)	1	---	1	---	---	---	---	---	---
ARCO Medical Products Company (ARCO)	---	1	1	---	---	---	---	---	---
Lawrence Berkeley Laboratory (LBL)	---	2	2	---	---	---	---	---	---
West Valley Demonstration Project (WVDP)	---	190	190	---	---	---	---	---	---
Total	**135,000**	**139,000**	**274,000**	**135,000**	**139,000**	**274,000**	**47,000**	**60,000**	**107,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily for PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.

[b] Volumes include consolidation of waste as indicated on Figure 3-3.

[c] Post-treatment volumes have been adjusted to meet packaging and transportation criteria.

[d] Sites in boldface were included in SEIS-I.

[e] TRU waste commingled with PCBs is included.

[f] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.

[g] Dashes indicate no waste.

Table A-9
CH-TRU Waste Volumes for Action Alternative 2B [a]

Site [d]	Site Volume Through 2033 (cubic meters)			Pretreatment Consolidated Volume [b] (cubic meters)			Post-Treatment Disposal Volume [c] (cubic meters)		
	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory
Hanford Site (Hanford)	57,000	63,000	120,000	59,000	63,000	122,000	21,000	22,000	43,000
Los Alamos National Laboratory (LANL)	21,000	14,000	35,000	--- [f]	---	---	---	---	---
Idaho National Engineering and Environmental Laboratory (INEEL) [g]	28,000	57,000	85,000	62,000	71,000	133,000	22,000	36,000	57,000
Argonne National Laboratory-West (ANL-W)	1,000	---	1,000	---	---	---	---	---	---
Argonne National Laboratory - East (ANL-E)	200	---	200	---	---	---	---	---	---
Savannah River Site (SRS)	12,000	4,900	17,000	14,000	5,200	20,000	5,000	1,800	6,800
Rocky Flats Environmental Technology Site (RFETS)	11,000	---	11,000	---	---	---	---	---	---
Oak Ridge National Laboratory (ORNL)	1,700	66	1,700	---	---	---	---	---	---
Lawrence Livermore National Laboratory (LLNL)	1,200	---	1,200	---	---	---	---	---	---
Nevada Test Site (NTS)	630	---	630	---	---	---	---	---	---
Mound Plant (Mound)	300	19	320	---	---	---	---	---	---
Bettis Atomic Power Laboratory (Bettis)	170	---	170	---	---	---	---	---	---
Sandia National Laboratories - Albuquerque (SNL)	17	1	18	---	---	---	---	---	---
Paducah Gaseous Diffusion Plant (PGDP)	8	---	8	---	---	---	---	---	---
U.S. Army Materiel Command (USAMC)	3	---	3	---	---	---	---	---	---
Energy Technical Engineering Center (ETEC)	2	---	2	---	---	---	---	---	---
University of Missouri Research Reactor (U of Mo)	1	---	1	---	---	---	---	---	---
Ames Laboratory - Iowa State University (Ames)	1	---	1	---	---	---	---	---	---
ARCO Medical Products Company (ARCO)	---	1	1	---	---	---	---	---	---
Lawrence Berkeley Laboratory (LBL)	---	2	2	---	---	---	---	---	---
West Valley Demonstration Project (WVDP)	---	190	190	---	---	---	---	---	---
Total	**135,000**	**139,000**	**274,000**	**135,000**	**139,000**	**274,000**	**47,000**	**60,000**	**107,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily for PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.

[b] Volumes include consolidation of waste as indicated on Figure 3-4.

[c] Post-treatment volumes have been adjusted to meet packaging and transportation criteria.

[d] Sites in boldface were included in SEIS-I.

[e] TRU waste commingled with PCBs is included.

[f] Dashes indicate no waste.

[g] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.

Table A-10
CH-TRU Waste Volumes for Action Alternative 2C [a]

Site [b]	Site Volume Through 2033 (cubic meters)			Pretreatment Consolidated Volume (cubic meters)			Post-Treatment Disposal Volume (cubic meters)		
	Basic Inventory	Additional Inventory [c]	Total Inventory	Basic Inventory	Additional Inventory [c]	Total Inventory	Basic Inventory	Additional Inventory [c]	Total Inventory
Hanford Site (Hanford)	57,000	63,000	120,000	--- [d]	---	---	---	---	---
Los Alamos National Laboratory (LANL)	21,000	14,000	35,000	---	---	---	---	---	---
Idaho National Engineering and Environmental Laboratory (INEEL) [e]	28,000	57,000	85,000	---	---	---	---	---	---
Argonne National Laboratory-West (ANL-W)	1,000	---	1,000	---	---	---	---	---	---
Argonne National Laboratory-East (ANL-E)	200	---	200	---	---	---	---	---	---
Savannah River Site (SRS)	12,000	4,900	17,000	---	---	---	---	---	---
Rocky Flats Environmental Technology Site (RFETS)	11,000	---	11,000	---	---	---	---	---	---
Oak Ridge National Laboratory (ORNL)	1,700	66	1,700	---	---	---	---	---	---
Lawrence Livermore National Laboratory (LLNL)	1,200	---	1,200	---	---	---	---	---	---
Nevada Test Site (NTS)	630	---	630	---	---	---	---	---	---
Mound Plant (Mound)	300	19	320	---	---	---	---	---	---
Bettis Atomic Power Laboratory (Bettis)	170	---	170	---	---	---	---	---	---
Sandia National Laboratories - Albuquerque (SNL)	17	1	18	---	---	---	---	---	---
Paducah Gaseous Diffusion Plant (PGDP)	8	---	8	---	---	---	---	---	---
U.S. Army Materiel Command (USAMC)	3	---	3	---	---	---	---	---	---
Energy Technical Engineering Center (ETEC)	2	---	2	---	---	---	---	---	---
University of Missouri Research Reactor (U of Mo)	1	---	1	---	---	---	---	---	---
Ames Laboratory - Iowa State University (Ames)	1	---	1	---	---	---	---	---	---
ARCO Medical Products Company (ARCO)	---	1	1	---	---	---	---	---	---
Lawrence Berkeley Laboratory (LBL)	---	2	2	---	---	---	---	---	---
West Valley Demonstration Project (WVDP)	---	190	190	---	---	---	---	---	---
WIPP [f]	---	---	---	135,000	139,000	274,000	47,000	60,000	107,000
Total	**135,000**	**139,000**	**274,000**	**135,000**	**139,000**	**274,000**	**47,000**	**60,000**	**107,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily for PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.

[b] Sites in boldface were included in SEIS-I.

[c] TRU waste commingled with PCBs is included.

[d] Dashes indicate no waste.

[e] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.

[f] TRU waste is consolidated and treated at WIPP.

Table A-11
RH-TRU Waste Volumes for Action Alternatives 2A, 2B, and 2C [a]

Site [d]	Site Volume Through 2033 (cubic meters)			Pretreatment Consolidated Volume [b] (cubic meters)			Post-Treatment Disposal Volume [c] (cubic meters)		
	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory
Hanford Site (Hanford)	29,000	1,000	30,000	32,000	1,600	33,000	16,000	920	17,000
Los Alamos National Laboratory (LANL)	230	120	350	--- [f]	---	---	---	---	---
Idaho National Engineering and Environmental Laboratory (INEEL) [g]	220	440	660	---	---	---	---	---	---
Argonne National Laboratory-West (ANL-W)	1,700	---	1,700	---	---	---	---	---	---
Oak Ridge National Laboratory (ORNL)	3,100	120	3,200	3,700	1,900	5,600	1,800	960	2,800
Bettis Atomic Power Laboratory (Bettis)	9	---	9	---	---	---	---	---	---
Energy Technical Engineering Center (ETEC)	7	---	7	---	---	---	---	---	---
Battelle Columbus Laboratories (BCL)	580	---	580	---	---	---	---	---	---
Knolls Atomic Power Laboratory (Knolls)	---	81	81	---	---	---	---	---	---
West Valley Demonstration Project (WVDP)	---	1,700	1,700	---	---	---	---	---	---
Total	**35,000**	**3,500**	**39,000**	**35,000**	**3,500**	**39,000**	**18,000**	**1,900**	**19,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily for PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.
[b] Volumes include consolidation of waste as indicated on Figure 3-5.
[c] Post-treatment volumes have been adjusted to meet packaging and transportation criteria.
[d] Sites in boldface were included in SEIS-I.
[e] TRU waste commingled with PCBs is included.
[f] Dashes indicate no waste.
[g] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.

Using an aggregate volume expansion factor of 1.2 of all waste treated by the shred and grout process, the resulting disposal volume for CH-TRU waste is calculated as:

$$V_{Disposal} = (1.2)\, V_{Consolidated} \qquad \text{(Equation A-5)}$$

As in Equation A-4, an additional volume expansion factor of 1.43 is applied to RH-TRU waste calculations to account for the placement of three waste drums in an RH-TRU waste canister (Equation A-6).

$$V_{Disposal} = (1.2)\, V_{Consolidated} \times 1.43 \qquad \text{(Equation A-6)}$$

The waste density would increase for the shred and grout process because much of the waste drum's void space would be filled. A density increase factor of 1.263 is used for waste streams that are already solidified, and a density increase factor of 2.357 is used for all other waste streams (DOE 1995b).

Table A-12
CH-TRU Waste Volumes for Action Alternative 3 [a]

Site [d]	Site Volume Through 2033 (cubic meters)			Pretreatment Consolidated Volume [b] (cubic meters)			Post-Treatment Disposal Volume [c] (cubic meters)		
	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory
Hanford Site (Hanford)	57,000	63,000	120,000	59,000	63,000	121,000	70,000	75,000	146,000
Los Alamos National Laboratory (LANL)	21,000	14,000	35,000	21,000	14,000	35,000	25,000	17,000	42,000
Idaho National Engineering and Environmental Laboratory (INEEL) [f]	28,000	57,000	85,000	30,000	57,000	86,000	37,000	68,000	105,000
Argonne National Laboratory - West (ANL-W)	1,000	--- [g]	1,000	---	---	---	---	---	---
Argonne National Laboratory - East (ANL-E)	200	---	200	---	---	---	---	---	---
Savannah River Site (SRS)	12,000	4,900	17,000	14,000	5,000	20,000	17,000	6,200	23,000
Rocky Flats Environmental Technology Site (RFETS)	11,000	---	11,000	11,000	---	11,000	19,000	---	19,000
Oak Ridge National Laboratory (ORNL)	1,700	66	1,700	---	---	---	---	---	---
Lawrence Livermore National Laboratory (LLNL)	1,200	---	1,200	---	---	---	---	---	---
Nevada Test Site (NTS)	630	---	630	---	---	---	---	---	---
Mound Plant (Mound)	300	---	300	---	---	---	---	---	---
Bettis Atomic Power Laboratory (Bettis)	170	---	170	---	---	---	---	---	---
Sandia National Laboratories - Albuquerque (SNL)	17	1	18	---	---	---	---	---	---
Paducah Gaseous Diffusion Plant (PGDP)	8	---	8	---	---	---	---	---	---
U.S. Army Materiel Command (USAMC)	3	---	3	---	---	---	---	---	---
Energy Technical Engineering Center (ETEC)	2	---	2	---	---	---	---	---	---
University of Missouri Research Reactor (U of Mo)	1	---	1	---	---	---	---	---	---
Ames Laboratory - Iowa State University (Ames)	1	---	1	---	---	---	---	---	---
ARCO Medical Products Company (ARCO)	---	1	1	---	---	---	---	---	---
Lawrence Berkeley Laboratory (LBL)	---	2	2	---	---	---	---	---	---
West Valley Demonstration Project (WVDP)	---	190	190	---	---	---	---	---	---
Total	**135,000**	**138,000**	**273,000**	**135,000**	**138,000**	**273,000**	**168,000**	**166,000**	**334,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily for PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.
[b] Volumes include consolidation of waste as indicated on Figure 3-6.
[c] Post-treatment volumes have been adjusted to meet packaging and transportation criteria.
[d] Sites in boldface were included in SEIS-I.
[e] TRU waste commingled with PCBs is not included because there would be no thermal treatment under this alternative.
[f] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.
[g] Dashes indicate no waste.

Table A-13
RH-TRU Waste Volumes for Action Alternative 3 [a]

Site [d]	Site Volume Through 2033 (cubic meters)			Pretreatment Consolidated Volume [b] (cubic meters)			Post-Treatment Disposal Volume [c] (cubic meters)		
	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory	Basic Inventory	Additional Inventory [e]	Total Inventory
Hanford Site (Hanford)	29,000	1,000	30,000	32,000	1,600	33,000	54,000	2,700	57,000
Los Alamos National Laboratory (LANL)	230	120	350	--- [f]	---	---	---	---	---
Idaho National Engineering and Environmental Laboratory (INEEL) [g]	220	440	660	---	---	---	---	---	---
Argonne National Laboratory - West (ANL-W)	1,700	---	1,700	---	---	---	---	---	---
Oak Ridge National Laboratory (ORNL)	3,100	120	3,200	3,700	1,900	5,600	6,300	3,300	10,000
Bettis Atomic Power Laboratory (Bettis)	9	---	9	---	---	---	---	---	---
Energy Technical Engineering Center (ETEC)	7	---	7	---	---	---	---	---	---
Battelle Columbus Laboratories (BCL)	580	---	580	---	---	---	---	---	---
Knolls Atomic Power Laboratory (Knolls)	---	81	81	---	---	---	---	---	---
West Valley Demonstration Project (WVDP)	---	1,700	1,700	---	---	---	---	---	---
Total	**35,000**	**3,500**	**39,000**	**35,000**	**3,500**	**39,000**	**60,000**	**6,000**	**66,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily for PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.
[b] Volumes include consolidation of waste as indicated on Figure 3-6.
[c] Post-treatment volumes have been adjusted to meet packaging and transportation criteria.
[d] Sites in boldface were included in SEIS-I.
[e] TRU waste commingled with PCBs is not included because there would be no thermal treatment under this alternative.
[f] Dashes indicate no waste.
[g] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.

A.3.7 Waste Volumes for No Action Alternative 1

The TRU waste volumes under No Action Alternative 1A are identical to those under Action Alternative 2A (see Tables A-8 and A-11). Likewise, TRU waste volumes under No Action Alternative 1B are identical to those under Action Alternative 2B (see Tables A-9 and A-11).

Two TRU waste treatment scenarios exist under No Action Alternative 1. No Action Alternative 1A assumes the same waste treatment sites as under the WM PEIS Regionalized 2 Alternative. No Action Alternative 1B assumes the same waste treatment sites as the WM PEIS (DOE 1997b) Regionalized 3 Alternative. Waste treatment would be the same as that in Action Alternative 2; however, TRU waste would be placed in retrievable storage at the treatment sites rather than being sent to WIPP.

A.3.8 Waste Volumes for No Action Alternative 2

TRU waste volumes for No Action Alternative 2, projected to the year 2033, are given in Table 3-16 and presented here as Table A-14. As shown in this table, all RH-TRU waste in the Basic Inventory was included in the analysis. For the purpose of analysis, no consolidation of waste was assumed.

A.3.9 Number of Waste Shipments

The number of shipments required to transport waste from the treatment sites to WIPP primarily depends on the type of waste (CH-TRU or RH-TRU), the waste volume, and the waste density. Some final CH-TRU waste forms are dense enough that TRUPACT-II weight limits impact the number of waste drums that can be carried in one trip (Section A.2.1.1). Material parameter data are found in BIR-3 for most of the 900 waste streams. From the material parameter data, average waste densities were derived for the final waste form at each waste generator-storage site. The

Table A-14
TRU Waste Volumes for No Action Alternative 2 [a]

Site [b]	Site Volume Through 2033 (cubic meters)		Stored Volume (1995) (cubic meters)		Newly Generated Post-Treatment Volume (cubic meters)	
	CH-TRU	RH-TRU	CH-TRU	RH-TRU	CH-TRU	RH-TRU
Hanford Site (Hanford)	57,000	29,000	12,000	200	45,000	29,000
Los Alamos National Laboratory (LANL)	21,000	230	11,000	90	10,000	130
Idaho National Engineering and Environmental Laboratory (INEEL) [c]	28,000	220	28,000	220	---	---
Argonne National Laboratory - West (ANL-W)	1,000	1,700	7	20	1,000	1,700
Argonne National Laboratory - East (ANL-E)	200	--- [d]	25	---	180	---
Savannah River Site (SRS)	12,000	---	2,900	---	9,200	---
Rocky Flats Environmental Technology Site (RFETS)	11,000	---	4,900	---	6,000	---
Oak Ridge National Laboratory (ORNL)	1,700	3,100	1,300	2,500	350	600
Lawrence Livermore National Laboratory (LLNL)	1,200	---	230	---	960	---
Nevada Test Site (NTS)	630	---	620	---	10	---
Mound Plant (Mound)	300	---	300	---	---	---
Bettis Atomic Power Laboratory (Bettis)	170	9	---	---	170	9
Sandia National Laboratories - Albuquerque (SNL)	17	---	7	---	10	---
Paducah Gaseous Diffusion Plant (PGDP)	8	---	---	---	8	---
U.S. Army Materiel Command (USAMC)	3	---	3	---	---	---
Energy Technology Engineering Center (ETEC)	2	7	2	6	---	1
University of Missouri Research Reactor (U of Mo)	1	---	1	---	1	---
Ames Laboratory - Iowa State University (Ames)	1	---	1	---	1	---
Battelle Columbus Laboratories (BCL)	---	580	---	5,801	---	---
Total	**135,000**	**35,000**	**62,000**	**3,600**	**73,000**	**32,000**

[a] The inventory for SEIS-II is based on BIR-3, which takes into account potential thermal treatment at some sites (DOE 1996c). The thermal treatment is not necessarily for PCB-commingled waste. Site volumes through 2033 are the sum of similar columns on Tables 2-2 and 2-3. Volumes have been rounded. Actual totals may differ due to rounding.
[b] Sites in boldface were included in SEIS-I.
[c] INEEL and ANL-W are considered as one site in the WM PEIS and, therefore, counted as one site in SEIS-II.
[d] Dashes indicate no waste.

average density data were used to estimate the number of truck shipments required to move waste from each site. Table A-15 provides the number of truck shipments from waste treatment sites to WIPP according to waste inventory (Basic or Additional) and alternative, where applicable. It was assumed that thermal processing would result in a dense waste stream such that a standard waste drum would weigh 1,000 pounds (454 kilograms).

The number of RH-TRU waste shipments assumes the use of one RH-TRU waste canister per shipment. For the Proposed Action, the number of CH-TRU waste shipments has been adjusted to the WIPP allowable volume of approximately 168,500 cubic meters (5.95 million cubic feet) rather than the 143,000 cubic meters (5 million cubic feet) shown in Table A-5.

When considering rail analysis, it was assumed that one railcar would carry six TRUPACT-IIs or two RH-72B casks, twice as much as a truck. A maximum of three railcars would be used per shipment; thus, 18 TRUPACT-IIs or six RH-72B casks could be transported per shipment. As identified in Table A-2, certain CH-TRU waste density configurations require that waste shipments be made using only two TRUPACT-IIs rather than three. Under the Proposed Action and Action Alternative 1, about 38 percent of the TRU waste shipments would fall into this category. All of the waste shipments for Action Alternative 2 reflect two TRUPACT-IIs per shipment. Under Action Alternative 3, about 86 percent of the waste shipments would consist of two TRUPACTS-IIs rather than three.

The number of RH-TRU waste shipments from a treatment site can be calculated from the corresponding post-treatment disposal volume. RH-TRU waste post-treatment disposal volumes have been calculated for the Proposed Action and each alternative, and are presented in a series of tables in this appendix. Assuming that an RH-TRU waste shipment consists of a single waste canister, the number of shipments is determined by dividing the RH-TRU waste post-treatment volume by 0.89 cubic meter (32 cubic feet), the volume of a single RH-TRU waste canister.

In addition to the post-treatment disposal volume, calculations for the number of CH-TRU waste shipments require the waste density of each final waste form at each treatment site. The waste density, calculated from material parameters in BIR-3, is adjusted for volume expansion to meet thermal power or PE-Ci limits, or density changes due to thermal or shred and grout treatment. The CH-TRU waste post-treatment disposal volumes presented in the tables within this appendix reflect site totals; they do not contain details on the final waste form by site. Average densities for each final waste form, before adjustments, are given in Tables A-16 through A-18 and A-20 through A-23. Waste defined as “Possible Future Wastes” in Table A-23 represents waste that would most likely be sent to WIPP. This waste was included as part of the Basic Inventory for the purpose of SEIS-II analysis.

Calculating the number of CH-TRU waste shipments from a site begins with the post-treatment disposal volume of a final waste form. From this, the number of drums is determined using the assumption that the volume of each drum is 0.208 cubic meter (7.3 cubic feet). Next, the adjusted waste density is used to determine the number of drums that can be moved in one shipment (Table A-2). The process is repeated for each final waste form at the treatment site. The result is the total number of waste shipments from the treatment site.

Example calculations for the number of shipments from a site are presented for CH-TRU waste at Los Alamos National Laboratory (LANL) under the Proposed Action in Table A-19. First, a volume multiplier is calculated using Equation A-2 to determine if the waste form must be

Table A-15
Number of Truck Shipments to WIPP by Alternative [a]

Sites	Proposed Action	Action Alternative 1			Action Alternative 2A			Action Alternative 2B		
	Basic Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste										
Hanford	13,666	11,562	7,167	18,729	8,230	8,813	17,043	8,230	8,813	17,043
LANL	5,009	4,238	1,590	5,828	2,952	1,947	4,899	---	---	---
INEEL	5,782	4,892	6,474	11,366	4,178	12,388	16,566	8,653	14,335	22,988
ANL-W	---	---	---	---	---	---	---	---	---	---
SRS	2,238	1,893	558	2,451	2,020	723	2,743	2,020	723	2,743
RFETS	2,485	2,102	0	2,102	1,524	0	1,524	---	---	---
ORNL	251	212	8	220	---	---	---	---	---	---
LLNL	162	137	0	137	---	---	---	---	---	---
NTS	86	73	0	73	---	---	---	---	---	---
Mound	59	50	23	73	---	---	---	---	---	---
ANL-E	28	24	0	24	---	---	---	---	---	---
Bettis	---	---	---	---	---	---	---	---	---	---
SNL	---	---	---	---	---	---	---	---	---	---
PGDP	---	---	---	---	---	---	---	---	---	---
USAMC	---	---	---	---	---	---	---	---	---	---
ETEC	---	---	---	---	---	---	---	---	---	---
U of Mo	---	---	---	---	---	---	---	---	---	---
Ames	---	---	---	---	---	---	---	---	---	---
ARCO	---	---	---	---	---	---	---	---	---	---
LBL	---	---	---	---	---	---	---	---	---	---
WVDP	---	---	---	---	---	---	---	---	---	---
Totals	29,766	25,183	15,820	41,003	18,904	23,871	42,775	18,903	23,871	42,774
RH-TRU Waste										
Hanford	3,178	47,156	1,651	48,807	17,730	1,031	18,761	17,730	1,031	18,761
LANL	367	367	190	557	---	---	---	---	---	---
INEEL	3,136	3,136	711	3,847	---	---	---	---	---	---
ORNL	1,276	5,875	3,076	8,951	2,057	1,077	3,134	2,057	1,077	3,134
Totals	7,957	56,534	5,628	62,162	19,787	2,108	21,895	19,787	2,108	21,895

[a] The transportation analysis conservatively assumes the number of rail shipments is one-half of the truck shipments.

Table A-15
Number of Truck Shipments to WIPP By Alternative — Continued [a]

	Action Alternative 2C			Action Alternative 3		
Sites	**Basic Inventory**	**Additional Inventory**	**Total Inventory**	**Basic Inventory**	**Additional Inventory**	**Total Inventory**
CH-TRU Waste						
Hanford	11,562	7,194	18,756	24,531	8,600	33,131
LANL	4,236	1,590	5,826	7,628	1,907	9,535
INEEL	4,776	6,639	11,415	10,386	7,769	18,155
ANL-W	116	0	116	---	---	---
SRS	1,893	558	2,451	2,885	706	3,591
RFETS	2,102	0	2,102	2,897	0	2,897
ORNL	192	8	200	---	---	---
LLNL	137	0	137	---	---	---
NTS	73	0	73	---	---	---
Mound	50	3	53	---	---	---
ANL-E	22	0	22	---	---	---
Bettis	20	0	20	---	---	---
SNL-AL	2	1	3	---	---	---
PGDP	1	0	1	---	---	---
USAMC	1	0	1	---	---	---
ETEC	1	0	1	---	---	---
U of Mo	1	0	1	---	---	---
Ames	1	0	1	---	---	---
ARCO	0	1	1	---	---	---
LBL	0	1	1	---	---	---
WVDP	0	23	23	---	---	---
Totals	25,188	16,018	41,206	48,327	18,982	67,309
RH-TRU Waste						
Hanford	17,730	1,031	18,761	60,789	3,076	63,865
LANL	---	---	---	---	---	---
INEEL	---	---	---	---	---	---
ORNL	2,057	1,077	3,134	7,050	3,691	10,741
Totals	19,787	2,108	21,895	67,839	6,767	74,606

[a] The transportation analysis conservatively assumes the number of rail shipments is one-half of the truck shipments.

Table A-16
CH-TRU Final Waste Form Data [a]

Site	Final Waste Form	Type [b]	Stored Volume (cubic meters)	Projected Volume (cubic meters)	Thermal Power Loading (W/cubic meter)	Thermal Power Limit [c] Bagless (W/cubic meter)	PE Ci/cubic meter	Average Density (kg/cubic meter)
Ames	Solidified Inorganics	CH-MTRU	0.0	0.4	0.3668	13.7400	---	793.2
ANL-E	Lead/Cadmium Metal	CH-MTRU	1.1	1.3	---	1.0820	---	330.0
	Solidified Inorganics	CH-MTRU	5.2	0.0	0.0001	13.7400	---	355.4
	Solidified Organics	CH-MTRU	0.2	0.0	---	13.7400	---	697.2
	Uncategorized Metal	CH-TRU	5.0	128.5	0.1090	1.0820	---	302.9
ANL-W	Combustible	CH-MTRU	0.0	2.0	0.2609	0.5413	7.85	401.2
	Heterogeneous	CH-MTRU	1.7	0.0	0.0648	0.5413	1.77	512.3
	Combustible	CH-TRU	0.0	99.6	---	0.5413	---	401.2
	Heterogeneous	CH-TRU	4.8	345.4	---	0.5413	---	512.3
	Uncategorized Metal	CH-TRU	0.0	293.8	---	1.0820	---	423.7
Bettis	Heterogeneous	CH-TRU	0.0	123.3	0.3782	0.5413	0.04	332.7
ETEC	Heterogeneous	CH-TRU	1.7	0.0	---	0.5413	0.14	512.3
Hanford	Combustible	CH-MTRU	455.7	1,247.3	0.2288	0.5413	8.06	133.9
	Lead/Cadmium Metal	CH-MTRU	14.2	34.5	0.1391	1.0820	4.90	328.2
	Solidified Organics	CH-MTRU	7.4	9.4	0.1315	13.7400	4.63	160.7
	Uncategorized Metal	CH-MTRU	444.9	19,635.2	0.1971	1.0820	0.14	419.4
	Heterogeneous,	CH-TRU	11,190.8	6,271.3	0.3931	0.5413	8.91	865.5
	Inorganic Non-Metal	CH-TRU	34.7	69.1	0.1466	1.0820	5.28	204.7
	Soils	CH-TRU	119.5	5,961.7	1.4339	13.7400	1.08	635.7
	Solidified Inorganics	CH-TRU	12.9	7.1	0.1423	13.7400	4.90	432.5
INEEL	Filter	CH-MTRU	131.0	0.0	0.9120	0.5413	27.72	96.6
	Inorganic Non-Metal	CH-MTRU	2,066.1	0.0	0.5747	1.0820	18.05	1,827.6
	Solidified Organics	CH-MTRU	789.7	0.0	0.0620	13.7400	2.12	894.9
	Inorganic Non-Metal	CH-TRU	853.6	0.0	0.4966	1.0820	---	2,463.3
	Combustible	CH-MTRU	3,239.2	0.0	0.0936	0.5413	2.96	725.8
	Graphite	CH-MTRU	410.6	0.0	0.1647	1.0820	5.71	310.9
	Heterogeneous	CH-MTRU	6,334.2	0.0	0.0755	0.5413	1.97	415.5
	Lead/Cadmium Metal	CH-MTRU	14.4	0.0	---	1.0820	---	313.5
	Salt	CH-MTRU	8.8	0.0	4.5238	13.7400	142.63	203.6
	Solidified Inorganics	CH-MTRU	3,991.7	0.0	0.5209	13.7400	15.80	674.6
	Uncategorized Metal	CH-MTRU	5,789.8	0.0	0.2055	1.0820	5.88	306.3
	Combustible	CH-TRU	65.8	0.0	3.0165	0.5413	76.67	280.3
	Graphite	CH-TRU	87.6	0.0	0.3372	1.0820	11.61	338.8
	Heterogeneous	CH-TRU	4,274.3	0.0	0.0633	0.5413	0.40	318.1
	Salt	CH-TRU	11.7	0.0	1.3889	13.7400	37.33	318.3
	Uncategorized Metal	CH-TRU	77.1	0.0	10.7713	1.0820	88.22	403.6
LANL	Combustible	CH-MTRU	266.3	698.9	0.7201	0.5413	20.90	354.1
	Solidified Inorganics	CH-MTRU	4,883.2	1,952.3	1.2818	13.7400	17.02	1,248.3
	Uncategorized Metal	CH-MTRU	2,561.9	1,118.2	0.3720	1.0820	12.14	417.7
	Combustible	CH-TRU	1,555.2	1,677.3	1.1597	0.5413	35.73	354.1
	Heterogeneous	CH-TRU	16.0	29.1	0.0529	0.5413	7.99	346.3
	Soils	CH-TRU	110.6	29.1	0.0706	13.7400	1.98	1,200.0
	Solidified Inorganics	CH-TRU	5.0	81.5	4.0527	13.7400	111.66	1,004.8
	Solidified Organics	CH-TRU	1.5	29.1	0.1459	13.7400	4.49	1,296.0
	Uncategorized Metal	CH-TRU	1,652.5	1,735.6	1.1597	1.0820	32.38	525.0

[a] Dashed line indicates that information was not available.
[b] MTRU denotes TRU mixed waste.
[c] Values were obtained from Table 6-1 of TRUCON (DOE 1994b) and converted from watts per drum to watts per cubic meter to facilitate calculations.

Table A-16
CH-TRU Final Waste Form Data — Continued [a]

Site	Final Waste Form	Type [b]	Stored Volume (cubic meter)	Projected Volume (cubic meter)	Thermal Power Loading (W/cubic meter)	Thermal Power Limit [c] Bagless (W/cubic meter)	PE Ci/cubic meter	Average Density (kg/cubic meter)
LLNL	Heterogeneous	CH-MTRU	8.6	0.0	0.2682	0.5413	4.45	221.3
	Solidified Organics	CH-MTRU	1.0	5.8	0.0872	13.7400	3.08	268.0
	Filter	CH-TRU	15.5	32.3	0.1801	0.5413	3.36	213.9
	Heterogeneous	CH-TRU	190.2	663.8	0.1269	0.5413	4.36	165.3
	Salt	CH-TRU	0.6	3.0	0.2268	13.7400	7.94	342.0
	Solidified Inorganics	CH-TRU	14.4	5.8	0.0939	13.7400	3.31	268.0
Mound	Combustible	CH-MTRU	1.7	0.0	3.1350	0.5413	86.97	401.2
	Solidified Inorganics	CH-MTRU	1.9	0.0	0.2405	13.7400	6.60	1,074.6
	Combustible	CH-TRU	5.4	0.0	3.0542	0.5413	83.90	401.2
	Filter	CH-TRU	0.8	0.0	0.0653	0.5413	1.55	96.0
	Heterogeneous	CH-TRU	0.6	0.0	0.9581	0.5413	27.80	512.3
	Soils	CH-TRU	177.2	0.0	0.0076	13.7400	0.22	846.8
	Solidified Inorganics	CH-TRU	4.2	0.0	0.0040	13.7400	0.11	1,143.0
	Uncategorized Metal	CH-TRU	82.5	0.0	0.1613	1.0820	3.86	423.7
NTS	Heterogeneous	CH-MTRU	613.3	9.0	0.1667	0.5413	5.29	225.2
	Solidified Inorganics	CH-TRU	5.7	0.0	0.2243	13.7400	7.78	272.0
ORNL	Heterogeneous	CH-MTRU	697.6	256.3	0.0051	0.5413	0.15	251.8
	Heterogeneous	CH-TRU	606.5	0.0	0.0633	0.5413	1.89	251.8
PGDP	Inorganic Non-Metal	CH-MTRU	0.0	1.9	0.0031	1.0820	0.08	63.6
RFETS	Combustible	CH-MTRU	151.4	736.5	0.0908	0.5413	3.00	131.2
	Filter	CH-MTRU	2.1	325.4	0.1273	0.5413	4.42	84.9
	Heterogeneous	CH-MTRU	1.2	0.0	0.0232	0.5413	0.81	362.0
	Inorganic Non-Metal	CH-MTRU	16.5	291.7	0.0499	1.0820	1.73	249.8
	Lead/Cadmium Metal	CH-MTRU	4.0	298.3	0.1349	1.0820	4.41	245.2
	Solidified Inorganics	CH-MTRU	150.2	1,193.3	0.0894	13.7400	0.56	714.9
	Solidified Organics	CH-MTRU	109.8	0.0	0.0276	13.7400	1.33	983.5
	Uncategorized Metal	CH-MTRU	1.5	0.0	0.8873	1.0820	30.47	215.5
	Combustible	CH-TRU	34.1	124.6	0.0668	0.5413	2.16	79.5
	Filter	CH-TRU	70.0	152.2	0.2361	0.5413	8.15	90.5
	Graphite	CH-TRU	13.7	47.6	0.2060	1.0820	7.17	279.1
	Heterogeneous	CH-TRU	2.6	0.0	0.0232	0.5413	0.19	370.8
	Inorganic Non-metal	CH-TRU	41.8	575.0	0.0836	1.0820	2.90	219.0
	Salt	CH-TRU	0.0	325.9	---	13.7400	---	536.4
	Solidified Inorganics	CH-TRU	15.2	64.3	1.1542	13.7400	39.53	685.4
	Solidified Organics	CH-TRU	0.0	31.1	0.0701	13.7400	2.44	349.1
	Uncategorized Metal	CH-TRU	91.9	236.1	0.0836	1.0820	2.89	321.3
SNL	Heterogeneous	CH-TRU	6.7	7.5	0.0522	0.5413	0.94	166.1
SRS	Heterogeneous	CH-MTRU	2,611.6	5,465.2	2.9701	0.5413	83.78	173.2
	Solidified Inorganics	CH-MTRU	200.2	1,169.6	1.7429	13.7400	48.26	2,185.5
	Uncategorized Metal	CH-MTRU	70.4	120.5	2.9701	1.0820	83.78	387.1
	Heterogeneous	CH-TRU	0.0	10.7	3.0504	0.5413	83.64	161.5
U of Mo	Heterogeneous	CH-MTRU	0.2	0.8	0.0513	0.5413	1.54	101.3
USAMC	Heterogeneous	CH-TRU	2.5	0.0	0.0028	0.5413	0.09	512.3

[a] Dashed line indicates that information was not available.
[b] MTRU denotes TRU mixed waste.
[c] Values were obtained from Table 6-1 of TRUCON (DOE 1994b) and converted from watts per drum to watts per cubic meter to facilitate calculations.

Table A-17
Final Waste Form Data for RFETS CH-TRU Waste Residues [a, b]

Final Waste Form	Type	Stored Volume (cubic meters)	Projected Volume (cubic meters)	Thermal Power Loading (W/cubic meter)	Thermal Power Limit Bagless (W/cubic meter)	PE-Ci/cubic meter	Average Density (kg/cubic meter)
Combustible	CH-TRU	184.4	0.0	2.5205	0.5413	83.4	---
Filter	CH-TRU	592.8	0.0	2.5205	0.5413	83.4	---
Graphite	CH-TRU	47.1	0.0	2.5205	1.0820	83.4	---
Heterogeneous	CH-TRU	3.8	0.0	2.5205	0.5413	83.4	---
Inorganic Non-Metal	CH-TRU	2,509.6	0.0	2.5205	1.0820	83.4	---
Lead/Cadmium Metal	CH-TRU	7.6	0.0	2.5205	1.0820	83.4	---
Salt	CH-TRU	376.5	0.0	2.5205	13.7400	83.4	---
Solidified Inorganics	CH-TRU	404.6	0.0	2.5205	13.7400	83.4	---
Uncategorized Metal	CH-TRU	55.5	0.0	2.5205	1.0820	83.4	---

[a] RFETS residue data is part of the Basic Inventory.
[b] Dashed line indicates that information was not available.

Table A-18
Final Waste Form Data for PCB-Commingled Waste [a]

Site	Final Waste Form	Type [b]	Stored Volume (cubic meters)	Projected Volume (cubic meters)	Thermal Power Loading (W/cubic meter)	Thermal Power Limit Bagless (W/cubic meter)	PE-Ci/cubic meter	Average Density (kg/cubic meter)
Hanford	Uncategorized Metal	CH-MTRU	66.1	113.4	---	---	---	255.7
	Lead/Cadmium Metal	CH-MTRU	3.8	0.0	---	---	---	327.9
	Solidified Organics	CH-MTRU	2.1	8.3	---	---	---	160.4
	Solidified Organics	CH-MTRU	1.2	2.5	---	---	---	153.2
	Uncategorized Metal	CH-MTRU	0.2	0.0	---	---	---	338.1
INEEL	Inorganic Non-Metal	CH-MTRU	108.6	0.0	---	---	---	2,500.0
	Solidified Organics	CH-MTRU	352.8	0.0	---	---	---	903.9
Mound	Unknown	CH-TRU	19.0	0.0	---	---	---	---
	Unknown	CH-TRU	0.2	0.0	---	---	---	---

[a] Dashed line indicates that information was not available.
[b] MTRU denotes TRU mixed waste.

Table A-19
Example LANL Data for Shipment Calculations

Final Waste Form	Watts per Cubic Meter	Watt Limit with Bagless Posting	Volume Multiplier for Watts	Average Density (kg/cubic meter)	Adjusted Average Density (kg/cubic meter)	Shipment Volume Multiplier for Density	Site Volume	Shipment Volume
Combustible	0.7201	0.5413	1	354.1	354.1	1	1,214.8	1,214.8
Lead/Cadmium Metal Waste	Missing	1.0820	1	313.5	313.5	1	1.9	1.9
Solidified Inorganics	1.2818	13.7400	1	1,248.3	1,248.3	3	7,532.7	22,598.2
Uncategorized Metal	0.3720	1.0820	1	417.7	417.7	1	4,079.5	4,079.5
Combustible	1.1597	0.5413	1	354.1	354.1	1	3,831.5	3,831.5
Heterogeneous	0.0529	0.5413	1	346.3	346.3	1	55.5	55.5
Soils	0.0706	13.7400	1	1,200.0	1,200.0	3	150.1	450.3
Solidified Inorganics	4.0527	13.7400	1	1,004.8	1,004.8	3	115.6	346.9
Solidified Organics	0.1459	13.7400	1	1,296.0	1,296.0	3	41.0	122.9
Uncategorized Metal	1.1597	1.0820	1.0718	525.0	489.8	1	4,007.9	4,295.7

Table A-20
RH-TRU Final Waste Form Data [a]

Site	Final Waste Form	Type [b]	Stored Volume (cubic meters	Projected Volume (cubic meters)	Thermal Power Loading (W/cubic meter)	Thermal Power Limit Bagless (W/cubic meter)	PE-Ci/cubic meter	Average Density (kg/cubic meter)
ANL-W	Inorganic Nonmetal	RH-MTRU	0.0	21.4	---	337.1	---	132.5
	Solidified Inorganics	RH-MTRU	1.8	28.5	---	337.1	---	1,074.6
	Heterogeneous	RH-TRU	0.0	1,208.6	---	337.1	---	512.3
	Lead/Cadmium Metal	RH-TRU	0.0	6.2	---	337.1	---	423.7
BCL	Heterogeneous	RH-TRU	580.5	0.0	---	337.1	0.01	2,000.0
Bettis	Heterogeneous	RH-TRU	0.0	6.7	6.3620	337.1	45.45	885.0
ETEC	Lead/Cadmium Metal	RH-MTRU	0.9	0.0	---	337.1	13.19	313.5
Hanford	Heterogeneous	RH-MTRU	0.0	2,617.5	---	337.1	---	432.4
	Lead/Cadmium Metal	RH-MTRU	2.7	60.5	0.0012	337.1	0.06	592.0
	Uncategorized Metal	RH-MTRU	0.0	0.0	0.0003	337.1	0.00	519.1
	Heterogeneous	RH-TRU	199.4	1,448.9	1.9994	337.1	0.61	460.2
	Uncategorized Metal	RH-TRU	0.0	17,400.4	0.0003	337.1	0.00	283.5
INEEL	Combustible	RH-MTRU	21.4	0.0	0.2655	337.1	9.17	294.0
	Heterogeneous	RH-MTRU	43.9	0.0	0.1421	337.1	4.82	210.3
	Inorganic Non-Metal	RH-MTRU	33.2	0.0	1.2867	337.1	39.34	1,930.6
	Lead/Cadmium Metal	RH-MTRU	3.6	0.0	---	337.1	---	313.5
	Solidified Inorganics	RH-MTRU	65.3	0.0	0.6192	337.1	18.68	466.1
	Solidified Organics	RH-MTRU	3.6	0.0	0.5015	337.1	17.04	842.1
	Uncategorized Metal	RH-MTRU	22.5	0.0	0.0672	337.1	2.34	221.4
	Heterogeneous	RH-TRU	5.9	0.0	0.0048	337.1	0.09	378.2
	Inorganic Non-Metal	RH-TRU	13.1	0.0	4.4674	337.1	39.85	2,500.0
	Uncategorized Metal	RH-TRU	8.3	0.0	22.2868	337.1	204.34	341.5
LANL	Uncategorized Metal	RH-MTRU	16.9	33.8	0.9587	337.1	9.57	557.5
	Combustible	RH-TRU	15.1	49.0	0.1466	337.1	2.92	354.1
	Heterogeneous	RH-TRU	11.6	0.0	0.1990	337.1	3.17	302.9
	Uncategorized Metal	RH-TRU	50.7	16.0	0.0002	337.1	0.00	567.4
ORNL	Heterogeneous	RH-MTRU	1,347.5	240.3	0.0005	337.1	0.01	251.8
	Solidified Inorganics	RH-MTRU	1,036.9	206.5	---	337.1	0.01	793.3
	Heterogeneous	RH-TRU	84.5	0.0	0.0107	337.1	0.09	251.8

[a] Dashed line indicates that information was not available.
[b] MTRU denotes TRU mixed waste.

Table A-21
Previously Disposed TRU Final Waste Form Data [a]

Site	Final Waste Form	Type	Stored Volume (cubic meters)	Projected Volume (cubic meters)	Thermal Power Loading (W/cubic meter)	Thermal Power Limit Bagless (W/cubic meter)	PE-Ci/cubic meter	Average Density (kg/cubic meter)
Hanford	Buried	CH-TRU	62,599.1	0.0	---	---	---	---
		RH-TRU	1,029.9	0.0	---	---	---	---
INEEL	Buried	CH-TRU	56,556.5	0.0	---	---	---	---
		RH-TRU	443.5	0.0	---	---	---	---
LANL	Buried	CH-TRU	13,881.5	0.0	---	---	---	---
		RH-TRU	118.5	0.0	---	---	---	---
ORNL	Buried	CH-TRU	60.8	0.0	---	---	---	---
		RH-TRU	115.2	0.0	---	---	---	---
SNL	Buried	CH-TRU	1.3	0.0	---	---	---	---
SRS	Buried	CH-TRU	4,874.0	0.0	---	---	---	---
WVDP	Unknown, Buried Assume RH-TRU	RH-TRU	1,353.2	0.0	---	---	---	---

[a] Dashed line indicates that information was not available.

Table A-22
Commercial/Nondefense TRU Final Waste Form Data [a]

Site	Final Waste Form	Type [b]	Stored Volume (cubic meters)	Projected Volume (cubic meters)	Thermal Power Loading (W/cubic meter)	Thermal Power Limit Bagless (W/cubic meter)	PE-Ci/cubic meter	Average Density (kg/cubic meter)
LBL	Heterogeneous	CH-TRU	0.6	1.0	---	---	---	1,565.0
ORNL	Heterogeneous	CH-MTRU [b]	1.0	0.0	---	---	---	251.8
	Unknown	CH-TRU	4.4	0.0	---	---	---	251.8
Knolls	Heterogeneous	RH-TRU	2.5	51.0	---	---	---	252.9
	Heterogeneous	RH-MTRU	0.0	6.9	---	---	---	253.8
ARCO	Heterogeneous	CH-TRU	0.2	0.2	---	---	---	---
WVDP	Unknown	CH-MTRU	10.0	0.0	---	---	---	---
	Solidified Inorganics	CH-TRU	0.2	0.0	---	---	---	---
	Solidified Inorganics	CH-TRU	0.4	0.0	---	---	---	---
	Unknown	CH-TRU	0.2	0.0	---	---	---	---
	Solidified Inorganics	CH-MTRU	1.5	0.0	---	---	---	---
	Heterogeneous	CH-TRU	3.7	0.0	---	---	---	---
	Unknown	CH-TRU	0.6	0.0	---	---	---	---
	Inorganic Non-Metal	CH-TRU	0.2	0.0	---	---	---	---
	Unknown	CH-TRU	0.4	0.0	---	---	---	---
	Heterogeneous	CH-TRU	13.0	88.0	---	---	---	---
	Heterogeneous	CH-TRU	2.1	29.1	---	---	---	---
	Uncategorized Metal	CH-TRU	0.2	0.0	---	---	---	---
	Lead/Cadmium Metal	CH-MTRU	2.1	0.0	---	---	---	---
	Solidified Organics	CH-MTRU	0.4	0.0	---	---	---	---
	Filter	RH-TRU	3.8	0.0	---	---	---	---
	Filter	RH-TRU	49.0	46.0	---	---	---	---
	Unknown	RH-TRU	17.8	0.0	---	---	---	---
	Uncategorized Metal	RH-TRU	89.9	0.0	---	---	---	---

[a] Dashed line indicates that information was not available.
[b] MTRU denotes TRU mixed waste.

Table A-23
Possible Future TRU Final Waste Form Data [a, b]

Site	Final Waste Form	Type [c]	Stored Volume (cubic meters)	Projected Volume (cubic meters)	Thermal Power Loading (W/cubic meter)	Thermal Power Limit Bagless (W/cubic meter)	PE-Ci/cubic meter	Average Density (kg/cubic meter)
ANL-E	Uncategorized - Unknown	CH-TRU	13.3	0.0	---	---	---	---
ETEC	Uncategorized - Unknown	RH-MTRU	5.4	0.8	---	---	---	---
Mound	Uncategorized - Unknown	CH-TRU	23.0	0.0	---	---	---	---
	Uncategorized - Unknown	CH-TRU	4.2	0.0	---	---	---	---
ORNL	Uncategorized - Unknown	CH-TRU	17.7	0.0	---	---	---	---
	Uncategorized - Unknown	RH-TRU	0.9	0.0	---	---	---	---
PGDP	Uncategorized - Unknown	CH-MTRU	0.0	1.9	---	---	---	---
	Uncategorized - Unknown	CH-MTRU	0.0	1.9	---	---	---	---
Hanford	Uncategorized - Unknown	CH-MTRU	0.4	0.0	---	---	---	---
	Uncategorized - Unknown	CH-MTRU	0.2	0.0	---	---	---	---
	Uncategorized - Unknown	CH-TRU	0.2	0.0	---	---	---	---
	Uncategorized - Unknown	CH-TRU	1.5	0.8	---	---	---	---
	Uncategorized - Unknown	CH-TRU	19.0	61.0	---	---	---	---
	Uncategorized - Unknown	CH-TRU	0.4	0.0	---	---	---	---

[a] Dashed line indicates that information was not available.
[b] This waste volume has been included as part of the Basic Inventory.
[c] MTRU denotes TRU mixed waste.

expanded (void space) to reduce the gas generation to acceptable limits for shipment in the TRUPACT-II. This multiplier is calculated for every final waste form. Next, the average density for the final waste form is divided by the volume multiplier from the watt calculations to obtain the average density of the expanded waste form. In this example, the uncategorized metal is the only waste form that requires volume expansion to meet the gas generation limits. Next, a shipment volume multiplier is determined for every final waste form by computing the weight of a drum as 0.208 x (Adjusted Average Density) and looking up the shipment volume multiplier in Table A-2. Finally, the shipment volume is determined as the product of the site volume, the density volume multiplier, and the watt multiplier. The shipment volume is simply a mathematical construct that incorporates the amount of void space needed to completely fill three TRUPACT-IIs on a truck.

The total shipment volume is the sum of the shipment volume over all waste forms. For LANL, this number is 36,997.2 cubic meters (1,306,543.8 cubic feet). Each truck can carry three TRUPACT-IIs for a total of 42 drums, with a total waste volume of 8.736 cubic meters (308.5 cubic feet). Thus, the shipment volume of 36,997.2 cubic meters (1,306,543.8 cubic feet) requires 4,235 shipments. LANL also serves as a consolidation site and receives enough additional waste for two more shipments, for a total of 4,237 shipments to WIPP. Finally, the analysis for the Proposed Action assumes that the repository takes as much CH-TRU waste as is allowed under the LWA. This assumption requires the volume of waste to be increased by 18.2 percent. Thus, the total number of shipments from LANL to WIPP under the Proposed Action is 5,009 ($= 4{,}237 \times 1.182$).

A.3.10 Final Waste Form Data

The data from which the volume, density, and PE-Ci results were derived are presented in Table A-16. Cells with dashed lines indicate that BIR-3 data were not available to calculate the values. Where thermal power or PE-Ci data were missing, it was assumed that the waste form met planning-basis WAC requirements without volume expansion. When density data were missing, it was assumed that a CH-TRU waste shipment can handle three fully loaded TRUPACT-II's, except under Action Alternative 2 where a shipment consists of 12 drums and an RH-TRU waste shipment can handle three drums.

A.4 RADIONUCLIDE INVENTORY

The WIPP facility has been designed with the intent to use it for disposal of TRU waste. TRU wastes are defined to be wastes containing more than 100 nanocuries of alpha-emitting TRU radionuclides per gram of waste, with half-lives greater than 20 years, subject to a number of conditions (see Chapter 1). TRU waste at the generator-storage sites typically include non-TRU radionuclides commingled with the TRU waste. Data on all of the radionuclides are presented in this appendix. The four non-TRU radionuclides with the greatest activity in the waste are strontium-90 (Sr-90), yttrium-90 (Y-90), cesium-137 (Cs-137), and barium-137m (Ba-137m).

A.4.1 Inventory Information in 1995

The radionuclide activity by site was obtained from IDB (DOE 1994a), and the activity was adjusted to account for decay to December 31, 1995. Radionuclide information for CH-TRU waste was available for the following sites: Argonne National Laboratory-East (ANL-E), ARCO, U.S. Army Materiel Command (USAMC), Energy Technical Engineering Center (ETEC), Hanford, INEEL, LBL, LANL, Lawrence Livermore National Laboratory (LLNL), Mound, University of Missouri (U of Mo), Nevada Test Site (NTS), ORNL, Paducah Gaseous Diffusion Plant (PGDP), and RFETS. More than 99 percent of the CH-TRU waste volume in the Basic Inventory is located at these sites.

RH-TRU waste radionuclide information was only available from the following sites: ETEC, Hanford, INEEL, LANL, NTS, ORNL. In the Basic Inventory, approximately 98 percent of the RH-TRU waste volume is located at these sites.

The percentages of RH-TRU waste volumes at these sites in the Total Inventory for Action Alternatives 1, 2, 3, No Action Alternative 1, and No Action Alternative 2 are 93, 84, 93, 81, and 94 percent, respectively. Most of the remaining RH-TRU waste volume is accounted for at Battelle-Columbus; however, no information was available for this site.

Four sites (NTS, ANL-E, SRS, and Sandia National Laboratories (SNL)) provided RH-TRU waste radionuclide inventories in the IDB and BIR-3; however, they did not identify any waste volumes associated with RH-TRU waste. For the purpose of analyses, the assumption was made that these sites do not have any RH-TRU waste.

Due to a lack of radionuclide information for some waste streams in BIR-3, site radionuclide information for all but RFETS and ORNL were obtained from IDB. RFETS residue data were only found in BIR-3 and, therefore, were added to RFETS IDB data. ORNL's CH-TRU waste data for Cs-137 and Ba-137m radionuclide concentrations in the IDB were found to exceed the 200 millirem per hour limit to be designated as CH-TRU waste. Likewise, the data in BIR-3 for ORNL exceeded this limit. As an example, waste stream OR-W044 reported 520 cubic meters (18,360 cubic feet) of waste with a concentration of 0.489 Ci per cubic meter. ORNL staff confirmed that only 13 drums (2.704 cubic meters [95 cubic feet]) within this waste stream have the 0.489 Ci per cubic meter concentration, and yet it was applied to the entire waste stream. SEIS-II uses the Cs-137 data from BIR-3 rather than from IDB. OR-W044 waste stream data were adjusted to show Cs-137 in only 13 drums (2.704 cubic meters [95 cubic feet]). Under these assumptions, a total stored inventory of 1.328 Ci for Cs-137 and about 1.256 Ci for Ba-137m were determined. The radionuclide inventories (expressed as both activity in curies and mass in grams) for stored CH-TRU waste in 1995 according to generator-storage site are given in Tables A-24 and A-25, respectively. The radionuclide inventories (expressed as both activity in curies and mass in grams) for stored RH-TRU waste in 1995 according to generator-storage site are given in Tables A-26 and A-27, respectively.

The site radionuclide information represents the radionuclide inventory in a given volume as determined from IDB or BIR-3. The radionuclide inventory by site for an alternative was calculated using the following equation:

$$I_{Alternative} = I_{1995} + \left(\frac{I_{1995}}{V_{IDB}}\right)(V_{Estimated} - V_{Stored}) \qquad \text{(Equation A-7)}$$

where

$I_{Alternative}$ = total site radionuclide inventory (in Ci) used in the analysis for an alternative

I_{1995} = inventory in 1995 from IDB

V_{IDB} = volume associated with I_{1995}

$V_{Estimated}$ = estimated volume from Tables 2-2 and 2-3 to be generated through 2033 for an alternative

V_{Stored} = stored, post-treatment 1995 volume based on Tables 2-2 and 2-3.

Table A-24
Radionuclide Inventories (curies) for Stored CH-TRU Waste in 1995 [a]

Isotope	ANL-E	ARCO	USAMC	ETEC	Hanford	INEEL	LBL	LANL	LLNL	MOUND
Pu-238	2.11E+00	3.70E+02	---	1.11E-01	8.05E+04	5.98E+04	2.32E-04	1.15E+05	7.65E+01	4.97E+02
Pu-241	5.43E+01	---	---	6.22E+00	3.78E+04	1.50E+05	4.48E-07	1.62E+03	1.63E+03	---
Pu-239	3.28E+01	---	1.80E+01	1.79E+00	2.63E+04	4.01E+04	8.45E-06	7.91E+04	1.64E+02	6.28E+00
Am-241	5.89E+00	---	---	5.19E-01	4.73E+03	9.01E+04	9.17E-02	1.17E+04	1.44E+02	---
Pu-240	9.42E+00	---	---	6.12E-01	6.14E+03	9.84E+03	5.14E-03	1.01E+02	6.44E+01	---
Cs-137	---	---	---	---	6.83E+02	6.04E+01	---	4.81E+01	1.66E-06	---
Ba-137m	---	---	---	---	6.46E+02	5.71E+01	---	4.55E+01	1.57E-06	---
Cm-244	---	---	---	---	6.83E+01	4.93E+02	8.70E-02	1.56E+02	6.54E+01	---
Y-90	---	---	---	2.00E-01	6.92E+02	1.96E+00	---	4.44E+01	---	---
Sr-90	---	---	---	2.00E-01	6.92E+02	1.96E+00	---	4.44E+01	---	---
U-233	3.00E-02	---	---	1.20E-11	8.00E+01	8.99E+02	4.81E-03	4.46E+01	5.95E-09	---
Pu-242	1.00E-02	---	---	5.00E-05	3.80E-01	9.45E-01	1.01E-02	4.85E+02	2.02E-02	---
U-234	---	1.05E-03	---	1.93E-06	5.37E+01	6.18E+00	4.73E-09	6.06E+00	3.29E-03	2.47E-02
Pa-233	---	---	---	9.49E-07	2.72E-01	8.53E-01	6.32E-06	3.22E-02	4.71E-04	---
Np-237	---	---	---	9.49E-07	2.72E-01	8.53E-01	6.32E-06	3.22E-02	4.71E-04	---
Co-60	---	---	---	---	---	6.24E+01	---	7.91E-06	---	---
Eu-155	---	---	---	---	1.06E-03	3.83E-01	---	2.41E-01	---	---
Cf-252	---	---	---	---	3.52E+01	2.19E-03	---	---	---	---
Pb-212	---	---	---	---	5.18E-02	2.62E+01	---	6.16E-03	---	---
Ra-224	---	---	---	---	5.18E-02	2.62E+01	---	1.32E-03	---	---
Bi-212	---	---	---	---	5.18E-02	2.62E+01	---	1.32E-03	---	---
Po-216	---	---	---	---	5.18E-02	2.62E+01	---	1.32E-03	---	---
Rn-220	---	---	---	---	5.18E-02	2.62E+01	---	1.32E-03	---	---
Th-228	---	---	---	---	5.18E-02	2.62E+01	---	1.32E-03	---	---
U-232	---	---	---	---	---	2.53E+01	---	1.67E-03	---	---
Np-239	9.52E-02	---	---	---	9.01E-02	3.79E-01	3.85E-02	3.83E+00	2.45E-02	---
Am-243	9.52E-02	---	---	---	9.01E-02	3.79E-01	3.85E-02	3.83E+00	2.45E-02	---
Tc-99	---	---	---	---	9.51E-06	2.16E-03	---	1.02E-02	---	---
Po-212	---	---	---	---	3.32E-02	1.68E+01	---	8.48E-04	---	---
Cm-245	---	---	---	---	1.68E+01	9.09E-06	2.27E-06	1.60E-06	---	---
Tl-208	---	---	---	---	1.86E-02	9.42E+00	---	4.76E-04	---	---
U-237	---	---	---	1.53E-04	9.27E-01	3.67E+00	---	3.98E-02	4.00E-02	---
Ra-226	---	---	---	---	3.15E-05	4.80E-02	3.37E-02	9.05E-01	3.99E-08	4.47E-09
Po-218	---	---	1.40E-11	---	3.15E-05	4.80E-02	3.37E-02	9.05E-01	---	4.47E-09
Rn-222	---	---	---	---	3.15E-05	4.80E-02	3.37E-02	9.05E-01	---	4.47E-09
Bi-214	---	---	---	---	3.15E-05	4.80E-02	3.37E-02	9.04E-01	---	4.46E-09
Pb-214	---	---	---	---	3.15E-05	4.80E-02	3.37E-02	9.04E-01	---	4.46E-09
Po-214	---	---	---	---	3.15E-05	4.80E-02	3.37E-02	9.04E-01	---	4.46E-09
Ag-109m	---	---	---	---	---	---	---	6.56E+00	---	---
Cd-109	---	---	---	---	---	---	---	6.55E+00	---	---
Pa-234m	---	---	---	---	5.86E+00	1.16E-01	---	2.38E-02	3.03E-02	---
Th-234	---	---	---	---	5.86E+00	1.16E-01	---	2.36E-02	3.03E-02	---
U-238	---	---	---	---	5.86E+00	1.16E-01	---	2.36E-02	3.03E-02	---
Pm-147	---	---	---	---	4.78E-02	2.62E+00	---	2.00E+00	---	---
U-235	---	---	1.77E-08	1.06E-08	1.71E+00	6.18E-02	---	5.27E-01	5.93E-04	1.05E-07
Th-231	---	---	1.77E-08	1.06E-08	1.71E+00	6.18E-02	---	5.27E-01	1.76E-03	1.05E-07
Ac-225	---	---	---	---	1.31E-01	1.52E+00	5.45E-06	8.06E-02	---	---
Th-229	---	---	---	---	1.31E-01	1.52E+00	5.75E-06	8.06E-02	---	---
Ra-225	---	---	---	---	1.31E-01	1.52E+00	5.45E-06	8.06E-02	---	---
At-217	---	---	---	---	1.31E-01	1.52E+00	5.45E-06	8.06E-02	---	---
Bi-213	---	---	---	---	1.31E-01	1.52E+00	5.45E-06	8.06E-02	---	---
Fr-221	---	---	---	---	1.31E-01	1.52E+00	5.45E-06	8.06E-02	---	---
Pb-209	---	---	---	---	1.31E-01	1.52E+00	5.45E-06	8.06E-02	---	---
Po-213	---	---	---	---	1.28E-01	1.49E+00	5.33E-06	7.89E-02	---	---
C-14	---	---	---	---	1.60E+00	1.66E-01	---	2.00E-07	---	---
Bi-210	---	---	---	---	5.30E-06	2.70E-02	8.96E-03	2.80E-01	---	5.20E-10
Po-210	---	---	---	---	5.30E-06	2.70E-02	8.96E-03	2.80E-01	---	5.20E-10
Pb-210	---	---	---	---	5.30E-06	2.70E-02	8.96E-03	2.80E-01	---	5.20E-10
Eu-152	---	---	---	---	7.34E-07	1.62E-01	---	4.18E-04	1.33E-06	---
Cm-243	---	---	---	---	1.52E-02	---	---	1.11E+00	---	---
Eu-154	---	---	---	---	6.22E-05	6.43E-01	---	2.45E-02	5.25E-07	---

[a] Dashed line indicates that the radionuclide was not present or that information was not reported.

Table A-24
Radionuclide Inventories (curies) for Stored CH-TRU Waste in 1995 — Continued [a]

Isotope	U of MO	NTS	ORNL	PGDP	Pantex	RFETS	RFETS Residues	SRS	Total
Pu-238	---	3.15E+04	3.50E+03	---	---	3.43E+02	8.14E+03	5.53E+05	8.52E+05
Pu-241	6.32E-03	2.40E+02	4.79E+04	---	---	5.23E+04	1.02E+06	1.12E+05	1.42E+06
Pu-239	2.46E-02	2.76E+03	2.72E+03	5.57E+01	5.55E-02	9.98E+03	1.74E+05	9.35E+03	3.44E+05
Am-241	3.24E-01	2.84E+02	1.61E+03	---	---	1.10E+04	1.09E+05	2.01E+03	2.30E+05
Pu-240	---	2.66E+01	9.48E+02	---	---	7.22E+03	3.98E+04	2.31E+03	6.64E+04
Cs-137	---	3.60E-01	1.33E+00	---	---	---	---	7.51E+00	8.01E+02
Ba-137m	---	3.41E-01	1.26E+00	---	---	---	---	7.11E+00	7.57E+02
Cm-244	---	2.28E+02	1.06E+03	---	---	---	---	1.17E+03	3.24E+03
Y-90	---	3.10E-01	1.48E+03	---	---	---	---	6.98E+00	2.22E+03
Sr-90	---	3.10E-01	1.48E+03	---	---	---	---	6.98E+00	2.22E+03
U-233	1.78E-09	1.81E+00	1.77E+02	1.42E-03	---	1.29E+01	---	3.75E+00	1.22E+03
Pu-242	---	8.70E-02	2.37E-01	---	---	9.63E-05	---	3.75E-01	4.87E+02
U-234	2.98E-13	1.26E-02	1.57E+01	---	---	4.81E-03	---	2.56E+01	1.07E+02
Pa-233	2.28E-04	5.78E-03	7.32E-01	5.50E+01	---	1.70E-02	---	8.59E+00	6.55E+01
Np-237	2.28E-04	5.78E-03	7.27E-01	5.50E+01	---	1.70E-02	---	8.59E+00	6.55E+01
Co-60	---	---	1.84E-06	---	---	---	---	3.56E-01	6.28E+01
Eu-155	---	3.80E-03	---	---	---	---	---	5.28E+01	5.34E+01
Cf-252	---	1.70E-02	1.60E-01	---	---	---	---	3.62E-01	3.58E+01
Pb-212	---	1.64E-02	2.83E-01	---	---	---	---	9.20E-03	2.66E+01
Ra-224	---	1.71E-02	2.83E-01	---	---	---	---	9.20E-03	2.66E+01
Bi-212	---	1.64E-02	2.83E-01	---	---	---	---	9.20E-03	2.66E+01
Po-216	---	1.64E-02	2.83E-01	---	---	---	---	9.20E-03	2.66E+01
Rn-220	---	1.64E-02	2.83E-01	---	---	---	---	9.20E-03	2.66E+01
Th-228	---	1.64E-02	2.83E-01	---	---	---	---	9.20E-03	2.66E+01
U-232	---	1.65E-02	2.90E-01	---	---	---	---	8.94E-02	2.57E+01
Np-239	---	1.22E+00	1.49E+01	---	---	---	---	7.55E-01	2.13E+01
Am-243	---	1.22E+00	1.16E+01	---	---	---	---	7.55E-01	1.81E+01
Tc-99	---	5.99E-05	1.78E+01	---	---	---	---	4.50E-06	1.78E+01
Po-212	---	1.05E-02	1.82E-01	---	---	---	---	5.89E-03	1.70E+01
Cm-245	---	9.44E-06	3.35E-05	---	---	---	---	---	1.68E+01
Tl-208	---	5.89E-03	1.02E-01	---	---	---	---	3.31E-03	9.55E+00
U-237	1.55E-07	5.88E-03	1.18E+00	---	---	1.28E+00	---	1.52E+00	8.66E+00
Ra-226	---	2.50E-01	6.54E+00	---	---	---	---	7.30E-06	7.77E+00
Po-218	---	2.50E-01	6.49E+00	---	---	---	---	7.30E-06	7.73E+00
Rn-222	---	2.50E-01	6.49E+00	---	---	---	---	7.30E-06	7.73E+00
Bi-214	---	2.49E-01	6.49E+00	---	---	---	---	7.30E-06	7.72E+00
Pb-214	---	2.49E-01	6.49E+00	---	---	---	---	7.30E-06	7.72E+00
Po-214	---	2.49E-01	6.49E+00	---	---	---	---	7.30E-06	7.72E+00
Ag-109m	---	---	---	---	---	---	---	---	6.56E+00
Cd-109	---	---	---	---	---	---	---	---	6.55E+00
Pa-234m	1.16E-07	3.46E-04	4.26E-02	---	---	---	---	5.71E-03	6.08E+00
Th-234	1.16E-07	3.18E-04	4.26E-02	---	---	---	---	5.71E-03	6.08E+00
U-238	1.16E-07	1.64E-04	4.26E-02	---	---	---	---	5.71E-03	6.08E+00
Pm-147	---	1.05E-01	1.94E-02	---	---	---	---	1.24E-05	4.80E+00
U-235	4.44E-11	1.17E-02	1.33E-02	3.29E-07	1.09E-10	4.78E-05	---	5.84E-03	2.33E+00
Th-231	3.75E-08	6.15E-05	1.45E-02	3.29E-07	1.09E-10	4.78E-05	---	5.84E-03	2.32E+00
Ac-225	---	2.41E-03	2.07E-01	4.02E-07	---	---	---	1.31E-05	1.94E+00
Th-229	---	2.41E-03	2.07E-01	4.02E-07	---	---	---	1.31E-05	1.94E+00
Ra-225	---	2.41E-03	2.07E-01	4.02E-07	4.02E-07	---	---	1.31E-05	1.94E+00
At-217	---	2.41E-03	2.07E-01	4.02E-07	---	---	---	1.31E-05	1.94E+00
Bi-213	---	2.41E-03	2.07E-01	4.02E-07	---	---	---	1.31E-05	1.94E+00
Fr-221	---	2.41E-03	2.07E-01	4.02E-07	---	---	---	1.31E-05	1.94E+00
Pb-209	---	2.41E-03	2.07E-01	4.02E-07	---	---	---	1.31E-05	1.94E+00
Po-213	---	2.36E-03	2.02E-01	3.93E-07	---	---	---	1.28E-05	1.90E+00
C-14	---	2.50E-04	---	---	---	---	---	---	1.77E+00
Bi-210	---	6.69E-02	1.26E+00	---	---	---	---	1.11E-06	1.65E+00
Po-210	---	6.69E-02	1.26E+00	---	---	---	---	1.11E-06	1.65E+00
Pb-210	---	6.69E-02	1.26E+00	---	---	---	---	1.11E-06	1.65E+00
Eu-152	---	1.06E+00	6.18E-04	---	---	---	---	---	1.22E+00
Cm-243	---	---	---	---	---	---	---	---	1.12E+00
Eu-154	---	4.28E-01	---	---	---	---	---	2.84E-04	1.10E+00

[a] Dashed line indicates that the radionuclide was not present or that information was not reported.

Table A-25
Radionuclide Inventories (grams) for Stored CH-TRU Waste in 1995 [a]

Isotope	ANL-E	ARCO	USAMC	ETEC	Hanford	INEEL	LBL	LANL	LLNL	Mound
Pu-238	1.23E-01	2.16E+01	---	6.46E-03	4.70E+03	3.49E+03	1.35E-05	6.69E+03	4.46E+00	2.90E+01
Pu-241	5.27E-01	---	---	6.03E-02	3.66E+02	1.45E+03	4.34E-09	1.57E+01	1.58E+01	---
Pu-239	5.27E+02	---	2.90E+02	2.88E+01	4.22E+05	6.45E+05	1.36E-04	1.27E+06	2.64E+03	1.01E+02
Am-241	1.71E+00	---	---	1.51E-01	1.38E+03	2.62E+04	2.67E-02	3.40E+03	4.18E+01	---
Pu-240	4.13E+01	---	---	2.68E+00	2.69E+04	4.32E+04	2.25E-02	4.44E+02	2.82E+02	---
Cs-137	---	---	---	---	7.85E+00	6.93E-01	---	5.52E-01	1.91E-08	---
Ba-137m	---	---	---	---	1.20E-06	1.06E-07	---	8.45E-08	2.92E-15	---
Cm-244	---	---	---	---	8.43E-01	6.08E+00	1.07E-03	1.93E+00	8.07E-01	---
Y-90	---	---	---	3.68E-07	1.27E-03	3.60E-06	---	8.16E-05	---	---
Sr-90	---	---	---	1.47E-03	5.07E+00	1.44E-02	---	3.25E-01	---	---
U-233	3.11E+00	---	---	1.24E-09	8.28E+03	9.31E+04	4.98E-01	4.62E+03	6.16E-07	---
Pu-242	2.54E+00	---	---	1.27E-02	9.67E+01	2.40E+02	2.56E+00	1.23E+05	5.13E+00	---
U-234	---	1.69E-01	---	3.09E-04	8.61E+03	9.89E+02	7.58E-07	9.72E+02	5.27E-01	3.96E+00
Pa-233	---	---	---	4.57E-11	1.31E-05	4.11E-05	3.04E-10	1.55E-06	2.27E-08	---
Np-237	---	---	---	1.35E-03	3.85E+02	1.21E+03	8.95E-03	4.57E+01	6.67E-01	---
Co-60	---	---	---	---	---	6.34E-02	---	8.04E-09	---	---
Eu-155	---	---	---	---	2.27E-06	8.23E-04	---	5.18E-04	---	---
Cf-252	---	---	---	---	6.55E-02	4.06E-06	---	---	---	---
Pb-212	---	---	---	---	3.71E-08	1.88E-05	---	4.41E-09	---	---
Ra-224	---	---	---	---	3.23E-07	1.64E-04	---	8.26E-09	---	---
Bi-212	---	---	---	---	3.54E-09	1.79E-06	---	9.04E-11	---	---
Po-216	---	---	---	---	8.57E-17	4.33E-14	---	2.19E-18	---	---
Rn-220	---	---	---	---	5.62E-11	2.84E-08	---	1.43E-12	---	---
Th-228	---	---	---	---	6.31E-05	3.19E-02	---	1.61E-06	---	---
U-232	---	---	---	---	---	1.18E+00	---	7.82E-05	---	---
Np-239	4.09E-07	---	---	---	3.87E-07	1.63E-06	1.65E-07	1.64E-05	1.05E-07	---
Am-243	4.77E-01	---	---	---	4.52E-01	1.90E+00	1.93E-01	1.92E+01	1.23E-01	---
Tc-99	---	---	---	---	5.61E-04	1.27E-01	---	6.02E-01	---	---
Po-212	---	---	---	---	1.10E-16	5.54E-14	---	2.80E-18	---	---
Cm-245	---	---	---	---	9.77E+01	5.29E-05	1.32E-05	9.33E-06	---	---
Tl-208	---	---	---	---	6.38E-11	3.23E-08	---	1.63E-12	---	---
U-237	---	---	---	1.87E-09	1.13E-05	4.49E-05	---	4.87E-07	4.90E-07	---
Ra-226	---	---	---	---	3.18E-05	4.85E-02	3.40E-02	9.14E-01	4.03E-08	4.51E-09
Po-218	---	---	4.94E-20	---	1.11E-13	1.70E-10	1.19E-10	3.20E-09	---	1.58E-17
Rn-222	---	---	---	---	2.05E-10	3.11E-07	2.18E-07	5.87E-06	---	2.90E-14
Bi-214	---	---	---	---	7.10E-13	1.08E-09	7.58E-10	2.04E-08	---	1.01E-16
Pb-214	---	---	---	---	9.60E-13	1.46E-09	1.03E-09	2.76E-08	---	1.36E-16
Po-214	---	---	---	---	9.79E-20	1.49E-16	1.05E-16	2.81E-15	---	1.39E-23
Ag-109m	---	---	---	---	---	---	---	2.52E-09	---	---
Cd-109	---	---	---	---	---	---	---	2.48E-03	---	---
Pa-234m	---	---	---	---	8.52E-09	1.68E-10	---	3.46E-11	4.40E-11	---
Th-234	---	---	---	---	2.53E-04	5.00E-06	---	1.02E-06	1.31E-06	---
U-238	---	---	---	---	1.74E+07	3.44E+05	---	7.02E+04	9.00E+04	---
Pm-147	---	---	---	---	5.14E-05	2.82E-03	---	2.16E-03	---	---
U-235	---	---	8.20E-03	4.90E-03	7.90E+05	2.86E+04	---	2.44E+05	2.74E+02	4.86E-02
Th-231	---	---	3.33E-14	1.99E-14	3.21E-06	1.16E-07	---	9.89E-07	3.31E-09	1.97E-13
Ac-225	---	---	---	---	2.25E-06	2.62E-05	9.39E-11	1.39E-06	---	---
Th-229	---	---	---	---	6.14E-01	7.13E+00	2.70E-05	3.79E-01	---	---
Ra-225	---	---	---	---	3.33E-06	3.87E-05	1.39E-10	2.05E-06	---	---
At-217	---	---	---	---	8.03E-17	9.34E-16	3.35E-21	4.96E-17	---	---
Bi-213	---	---	---	---	6.80E-09	7.91E-08	2.84E-13	4.20E-09	---	---
Fr-221	---	---	---	---	7.36E-10	8.56E-09	3.07E-14	4.54E-10	---	---
Pb-209	---	---	---	---	2.83E-08	3.29E-07	1.18E-12	1.75E-08	---	---
Po-213	---	---	---	---	1.01E-17	1.18E-16	4.23E-22	6.25E-18	---	---
C-14	---	---	---	---	3.59E-01	3.73E-02	---	4.48E-08	---	---
Po-210	---	---	---	---	1.17E-09	5.98E-06	1.99E-06	6.22E-05	---	1.15E-13
Pb-210	---	---	---	---	6.93E-08	3.53E-04	1.17E-04	3.67E-03	---	6.80E-12
Eu-152	---	---	---	---	4.15E-09	9.20E-04	---	2.37E-06	7.53E-09	---
Cm-243	---	---	---	---	2.94E-04	---	---	2.14E-02	---	---
Eu-154	---	---	---	---	2.36E-07	2.44E-03	---	9.28E-05	1.99E-09	---

[a] Dashed line indicates that the radionuclide was not present or that information was not reported.

Table A-25
Radionuclide Inventories (grams) for Stored CH-TRU Waste in 1995 — Continued [a]

Isotope	U of MO	NTS	ORNL	PGDP	RFETS	RFETS Residues	SRS	Grand Total
Pu-238	---	1.84E+03	2.04E+02	---	2.00E+01	4.75E+02	3.23E+04	4.97E+04
Pu-241	6.13E-05	2.32E+00	4.65E+02	---	5.07E+02	9.85E+03	1.09E+03	1.38E+04
Pu-239	3.96E-01	4.44E+04	4.38E+04	8.95E+02	1.60E+05	2.79E+06	1.50E+05	5.53E+06
Am-241	9.42E-02	8.27E+01	4.70E+02	---	3.19E+03	3.16E+04	5.84E+02	6.69E+04
Pu-240	---	1.17E+02	4.15E+03	---	3.17E+04	1.74E+05	1.01E+04	2.91E+05
Cs-137	---	4.14E-03	1.53E-02	---	---	---	8.63E-02	9.20E+00
Ba-137m	---	6.33E-10	2.33E-09	---	---	---	1.32E-08	1.41E-06
Cm-244	---	2.82E+00	1.31E+01	---	---	---	1.44E+01	4.00E+01
Y-90	---	5.69E-07	2.71E-03	---	---	---	1.28E-05	4.08E-03
Sr-90	---	2.27E-03	1.08E+01	---	---	---	5.11E-02	1.63E+01
U-233	1.85E-07	1.88E+02	1.83E+04	1.47E-01	1.34E+03	---	3.89E+02	1.26E+05
Pu-242	---	2.21E+01	6.03E+01	---	2.45E-02	---	9.55E+01	1.24E+05
U-234	4.78E-11	2.01E+00	2.51E+03	---	7.71E-01	---	4.10E+03	1.72E+04
Pa-233	1.10E-08	2.78E-07	3.52E-05	2.65E-03	8.21E-07	---	4.14E-04	3.15E-03
Np-237	3.23E-01	8.19E+00	1.03E+03	7.79E+04	2.42E+01	---	1.22E+04	9.28E+04
Co-60	---	---	1.87E-09	---	---	---	3.61E-04	6.38E-02
Eu-155	---	8.16E-06	---	---	---	---	1.13E-01	1.15E-01
Cf-252	---	3.17E-05	2.98E-04	---	---	---	6.73E-04	6.65E-02
Pb-212	---	1.18E-08	2.03E-07	---	---	---	6.59E-09	1.90E-05
Ra-224	---	1.07E-07	1.77E-06	---	---	---	5.74E-08	1.66E-04
Bi-212	---	1.12E-09	1.93E-08	---	---	---	6.28E-10	1.81E-06
Po-216	---	2.71E-17	4.69E-16	---	---	---	1.52E-17	4.39E-14
Rn-220	---	1.78E-11	3.07E-10	---	---	---	9.97E-12	2.88E-08
Th-228	---	2.00E-05	3.45E-04	---	---	---	1.12E-05	3.24E-02
U-232	---	7.68E-04	1.35E-02	---	---	---	4.17E-03	1.20E+00
Np-239	---	5.26E-06	6.39E-05	---	---	---	3.25E-06	9.16E-05
Am-243	---	6.13E+00	5.83E+01	---	---	---	3.79E+00	9.06E+01
Tc-99	---	3.53E-03	1.05E+03	---	---	---	2.65E-04	1.05E+03
Po-212	---	3.47E-17	5.99E-16	---	---	---	1.94E-17	5.62E-14
Cm-245	---	5.49E-05	1.95E-04	---	---	---	---	9.77E+01
Tl-208	---	2.02E-11	3.49E-10	---	---	---	1.13E-11	3.27E-08
U-237	1.90E-12	7.19E-08	1.44E-05	---	1.57E-05	---	1.86E-05	1.06E-04
Ra-226	---	2.52E-01	6.61E+00	---	---	---	7.38E-06	7.86E+00
Po-218	---	8.82E-10	2.29E-08	---	---	---	2.58E-14	2.73E-08
Rn-222	---	1.62E-06	4.21E-05	---	---	---	4.74E-11	5.01E-05
Bi-214	---	5.62E-09	1.46E-07	---	---	---	1.64E-13	1.74E-07
Pb-214	---	7.60E-09	1.98E-07	---	---	---	2.22E-13	2.35E-07
Po-214	---	7.75E-16	2.02E-14	---	---	---	2.27E-20	2.40E-14
Ag-109m	---	---	---	---	---	---	---	2.52E-09
Cd-109	---	---	---	---	---	---	---	2.48E-03
Pa-234m	1.69E-16	5.04E-13	6.19E-11	---	---	---	8.30E-12	8.84E-09
Th-234	5.02E-12	1.37E-08	1.84E-06	---	---	---	2.46E-07	2.62E-04
U-238	3.46E-01	4.88E+02	1.27E+05	---	---	---	1.70E+04	1.81E+07
Pm-147	---	1.12E-04	2.08E-05	---	---	---	1.33E-08	5.17E-03
U-235	2.05E-05	5.41E+03	6.16E+03	1.52E-01	2.21E+01	---	2.70E+03	1.08E+06
Th-231	7.04E-14	1.15E-10	2.73E-08	6.18E-13	8.97E-11	---	1.10E-08	4.36E-06
Ac-225	---	4.15E-08	3.57E-06	6.91E-12	---	---	2.25E-10	3.34E-05
Th-229	---	1.13E-02	9.71E-01	1.89E-06	---	---	6.15E-05	9.11E+00
Ra-225	---	6.13E-08	5.27E-06	1.02E-11	---	---	3.33E-10	4.94E-05
At-217	---	1.48E-18	1.27E-16	2.47E-22	---	---	8.05E-21	1.19E-15
Bi-213	---	1.25E-10	1.08E-08	2.09E-14	---	---	6.81E-13	1.01E-07
Fr-221	---	1.36E-11	1.17E-09	2.26E-15	---	---	7.38E-14	1.09E-08
Pb-209	---	5.21E-10	4.48E-08	8.69E-14	---	---	2.83E-12	4.20E-07
Po-213	---	1.87E-19	1.60E-17	3.11E-23	---	---	1.01E-21	1.50E-16
C-14	---	5.60E-05	---	---	---	---	---	3.96E-01
Bi-210	---	5.39E-07	1.02E-05	---	---	---	8.93E-12	1.33E-05
Po-210	---	1.48E-05	2.80E-04	---	---	---	2.46E-10	3.65E-04
Pb-210	---	8.76E-04	1.66E-02	---	---	---	1.45E-08	2.16E-02
Eu-152	---	6.00E-03	3.50E-06	---	---	---	---	6.92E-03
Cm-243	---	---	---	---	---	---	---	2.17E-02
Eu-154	---	1.62E-03	---	---	---	---	1.08E-06	4.15E-03

[a] Dashed line indicates that the radionuclide was not present or that information was not reported

Table A-26
Radionuclide Inventories (curies) for Stored RH-TRU Waste in 1995 [a]

Isotope	ETEC	Hanford	INEEL	Knolls	LANL	ORNL	WVDP	Total
Y-90	2.62E+00	6.46E+03	1.70E+03	5.70E+01	1.24E+02	3.52E+04	1.96E+01	4.36E+04
Sr-90	2.62E+00	6.46E+03	1.70E+03	5.70E+01	1.24E+02	3.52E+04	1.96E+01	4.36E+04
Cs-137	2.62E+00	6.98E+03	1.90E+03	5.71E+01	1.35E+02	9.78E+03	5.35E+01	1.89E+04
Ba-137m	2.48E+00	6.61E+03	1.80E+03	5.40E+01	1.28E+02	9.25E+03	5.06E+01	1.79E+04
Pu-241	---	4.67E+03	4.81E+01	7.77E-01	---	3.97E-07	---	4.72E+03
Eu-152	---	---	1.14E-01	---	5.09E-04	3.66E+03	---	3.66E+03
Eu-154	---	---	7.90E-01	1.40E+00	3.50E-02	1.77E+03	---	1.78E+03
Cm-244	---	---	9.63E-02	---	---	9.44E+02	---	1.10E+03
Co-60	2.30E+00	3.36E+02	1.30E+01	2.75E-01	4.17E+00	6.14E+02	---	9.70E+02
Pu-239	4.00E-01	3.35E+02	2.98E+01	3.30E-03	9.28E+01	9.85E+01	---	5.59E+02
Am-241	5.85E-02	1.93E+02	4.68E+01	5.07E-02	---	2.42E+02	5.39E-01	4.83E+02
Eu-155	---	---	3.35E-01	1.81E-01	1.77E+00	3.51E+02	---	3.53E+02
Pu-240	---	1.67E+02	2.48E+01	3.10E-03	---	1.07E+00	---	1.93E+02
Th-231	4.73E-10	1.46E-01	6.42E-03	---	8.78E-03	1.86E+02	---	1.86E+02
U-235	4.73E-10	1.46E-01	5.38E-03	---	8.78E-03	1.86E+02	---	1.86E+02
Pu-238	---	4.67E+01	6.09E+01	9.27E-01	3.90E+00	2.81E+01	1.98E+01	1.69E+02
Cm-243	---	---	1.45E-02	---	---	1.48E+02	---	1.48E+02
Cs-134	---	---	5.38E+01	4.73E+00	2.42E-02	9.57E+00	---	6.81E+01
U-233	---	4.15E-01	3.91E-01	---	---	5.73E+01	---	5.81E+01
Pm-147	---	---	1.49E+01	4.34E+00	1.13E+01	---	---	3.34E+01
Rh-106	---	---	6.65E-02	4.98E-01	3.38E-01	3.21E+01	---	3.30E+01
Ru-106	---	---	6.65E-02	4.98E-01	3.38E-01	3.21E+01	---	3.30E+01
Pr-144	---	---	3.93E+00	1.54E+00	1.58E-02	1.51E+01	---	2.05E+01
Ce-144	---	---	3.98E+00	1.56E+00	1.60E-02	1.20E+01	---	1.75E+01
C-14	---	---	4.00E-02	---	---	6.11E+00	---	6.15E+00
Kr-85	---	---	5.95E+00	---	---	---	---	5.95E+00
Sb-125	---	---	9.81E-01	5.33E-01	2.79E+00	---	---	4.30E+00
Cf-252	---	---	---	---	---	3.86E+00	---	3.86E+00
Ni-63	---	---	3.50E+00	---	---	---	---	3.50E+00
U-238	---	1.03E-02	3.57E-03	---	2.00E-05	3.37E+00	---	3.38E+00
Pa-234m	---	1.03E-02	1.38E-03	---	2.00E-05	3.37E+00	---	3.38E+00
Th-234	---	1.03E-02	1.38E-03	---	2.00E-05	3.37E+00	---	3.38E+00
U-232	---	---	---	---	---	1.76E+00	---	1.76E+00
Po-216	---	1.49E-03	2.65E-05	---	---	1.68E+00	---	1.69E+00
Bi-212	---	1.49E-03	2.65E-05	---	---	1.68E+00	---	1.68E+00
Pb-212	---	1.49E-03	2.65E-05	---	---	1.68E+00	---	1.68E+00
Ra-224	---	1.49E-03	2.65E-05	---	---	1.68E+00	---	1.68E+00
Rn-220	---	1.49E-03	2.65E-05	---	---	1.68E+00	---	1.68E+00
Th-228	---	1.49E-03	2.65E-05	---	---	1.68E+00	---	1.68E+00
U-234	---	1.29E+00	1.51E-01	4.98E-06	1.11E-05	2.02E-03	4.94E-04	1.45E+00
Po-212	---	9.54E-04	1.70E-05	---	---	1.07E+00	---	1.07E+00
Te-125m	---	---	2.39E-01	1.30E-01	6.88E-01	---	---	1.06E+00

[a] Dashed line indicates that the radionuclide was not present or that information was not reported.

Rearranging terms, this equation can be written as:

$$I_{Alternative} = I_{1995}\left[1 + \left(\frac{V_{Estimated} - V_{Stored}}{V_{IDB}}\right)\right] \text{ or}$$

$$I_{Alternative} = I_{1995}\, V_{factor} \qquad \text{(Equation A-8)}$$

where

V_{factor} = complex volume variable used to estimate the total site radionuclide inventory for each alternative

Table A-27
Radionuclide Inventories (grams) for Stored RH-TRU Waste in 1995 [a]

Isotope	ETEC	Hanford	INEEL	Knolls	LANL	ORNL	WVDP	Total
Y-90	4.81E-06	1.19E-02	3.11E-03	1.05E-04	2.28E-04	6.47E-02	3.60E-05	8.01E-02
Sr-90	1.92E-02	4.73E+01	1.24E+01	4.17E-01	9.08E-01	2.58E+02	1.43E-01	3.19E+02
Cs-137	3.01E-02	8.02E+01	2.18E+01	6.56E-01	1.55E+00	1.12E+02	6.15E-01	2.17E+02
Ba-137m	4.61E-09	1.23E-05	3.33E-06	1.00E-07	2.38E-07	1.72E-05	9.39E-08	3.32E-05
Pu-241	---	4.53E+01	4.67E-01	7.53E-03	---	3.86E-09	---	4.57E+01
Eu-152	---	---	6.44E-04	---	2.88E-06	2.07E+01	---	2.07E+01
Eu-154	---	---	2.99E-03	5.30E-03	1.32E-04	6.72E+00	---	6.72E+00
Cm-244	---	---	1.19E-03	---	---	1.17E+01	---	1.17E+01
Co-60	2.34E-03	3.42E-01	1.32E-02	2.80E-04	4.24E-03	6.24E-01	---	9.86E-01
Pu-239	6.44E+00	5.39E+03	4.79E+02	5.31E-02	1.49E+03	1.58E+03	---	8.95E+03
Am-241	1.70E-02	5.62E+01	1.36E+01	1.48E-02	---	7.04E+01	1.57E-01	1.40E+02
Eu-155	---	---	7.21E-04	3.89E-04	3.81E-03	7.54E-01	---	7.59E-01
Pu-240	---	7.30E+02	1.09E+02	1.36E-02	---	4.69E+00	---	8.44E+02
Th-231	8.89E-16	2.74E-07	1.21E-08	---	1.65E-08	3.49E-04	---	3.49E-04
U-235	2.19E-04	6.75E+04	2.49E+03	---	4.06E+03	8.59E+07	---	8.60E+07
Pu-238	---	2.73E+00	3.55E+00	5.41E-02	2.28E-01	1.64E+00	1.16E+00	9.36E+00
Cm-243	---	---	2.81E-04	---	---	2.87E+00	---	2.87E+00
Cs-134	---	---	1.68E-02	1.48E-03	7.56E-06	3.00E-03	---	2.13E-02
U-233	---	4.29E+01	4.05E+01	---	---	5.93E+03	---	6.02E+03
Pm-147	---	---	1.60E-02	4.68E-03	1.22E-02	---	---	3.29E-02
Rh-106	---	---	4.87E-09	3.64E-08	2.47E-08	2.35E-06	---	2.42E-06
Ru-106	---	---	1.98E-05	1.49E-04	1.01E-04	9.58E-03	---	9.85E-03
Pr-144	---	---	5.21E-08	2.04E-08	2.10E-10	1.99E-07	---	2.72E-07
Ce-144	---	---	1.25E-03	4.87E-04	5.02E-06	3.75E-03	---	5.49E-03
C-14	---	---	8.96E-03	---	---	1.37E+00	---	1.38E+00
Kr-85	---	---	1.51E-02	---	---	---	---	1.51E-02
Sb-125	---	---	9.50E-04	5.16E-04	2.70E-03	---	---	4.16E-03
Cf-252	---	---	---	---	---	7.17E-03	---	7.17E-03
Ni-63	---	---	5.92E-02	---	---	---	---	5.92E-02
U-238	---	3.05E+04	1.06E+04	---	5.95E+01	1.00E+07	---	1.01E+07
Pa-234m	---	1.49E-11	2.01E-12	---	2.91E-14	4.90E-09	---	4.92E-09
Th-234	---	4.43E-07	5.97E-08	---	8.64E-10	1.45E-04	---	1.46E-04
U-232	---	---	---	---	---	8.24E-02	---	8.24E-02
Po-216	---	2.46E-18	4.38E-20	---	---	2.77E-15	---	2.77E-15
Bi-212	---	1.02E-10	1.81E-12	---	---	1.14E-07	---	1.14E-07
Pb-212	---	1.07E-09	1.90E-11	---	---	1.20E-06	---	1.20E-06
Ra-224	---	9.29E-09	1.65E-10	---	---	1.04E-05	---	1.05E-05
Rn-220	---	1.61E-12	2.87E-14	---	---	1.81E-09	---	1.82E-09
Th-228	---	1.81E-06	3.23E-08	---	---	2.04E-03	---	2.04E-03
U-234	---	2.07E+02	2.42E+01	7.98E-04	1.78E-03	3.23E-01	7.92E-02	2.32E+02
Po-212	---	3.15E-18	5.60E-20	---	---	3.54E-15	---	3.54E-15
Te-125m	---	---	1.33E-05	7.22E-06	3.82E-05	---	---	5.86E-05

[a] Dashed line indicates that the radionuclide was not present or that information was not reported.

CH-TRU waste volume factors (V_{Factor}) are presented in Table A-28. To meet the LWA limit of 168,500 cubic meters (5.95 million cubic feet), radionuclide activity was adjusted by a factor of approximately 1.182 under the Proposed Action. Likewise, CH-TRU waste volume factors for the Proposed Action were adjusted by this factor. Because volume factors are determined according to waste treatment site, and do not depend upon which wastes are included at WIPP, or how wastes are treated prior to disposal, the volume factors among the WM PEIS (DOE 1997b) regionalized options within an alternative are the same. Those sites without radionuclide inventory estimates in IDB were not included in the table.

Table A-29 gives RH-TRU waste volume factors according to site. Here, the Proposed Action radionuclide loading was adjusted down to meet the WIPP regulatory limit of 7,080 cubic meters

Table A-28
Volume Factors Used to Estimate Total CH-TRU Waste Radionuclide Inventories at Each Site [a]

Site	Proposed Action	Action Alternatives 1 and 3	Action Alternative 2 and No Action Alternative 1	No Action Alternative 2
Ames Laboratory - Iowa State University (Ames)	N/A	N/A	N/A	N/A
ARCO Medical Products Company (ARCO)	0.000	2.500	2.500	0.000
Argonne National Laboratory- East (ANL-E)	9.587	8.111	8.111	8.111
Argonne National Laboratory- West (ANL-W)	N/A	N/A	N/A	N/A
Bettis Atomic Power Laboratory (Bettis)	N/A	N/A	N/A	N/A
Energy Technology Engineering Center (ETEC)	1.182	1.000	1.000	1.000
Hanford Site (Hanford)	5.524	9.762	9.782	4.673
Idaho National Engineering and Environmental Laboratory (INEEL)	1.182	3.009	3.026	1.000
Lawrence Berkeley Laboratory (LBL)	0.000	3.262	3.262	0.000
Lawrence Livermore National Laboratory (LLNL)	6.130	5.186	5.186	5.186
Los Alamos National Laboratory (LANL)	2.249	3.158	3.158	1.903
Mound Plant (Mound)	1.182	1.000	1.064	1.000
Nevada Test Site (NTS)	1.205	1.020	1.020	1.020
Oak Ridge National Laboratory (ORNL)	1.493	1.313	1.313	1.263
Paducah Gaseous Diffusion Plant (PGDP) [b]	1.604	1.357	1.357	1.357
Rocky Flats Environmental Technology Site (RFETS) [c]	11.185	9.462	9.462	9.462
Sandia National Laboratories (SNL)	N/A	N/A	N/A	N/A
Savannah River Site (SRS) [d]	2.362	2.528	2.528	1.998
U.S. Army Materiel Command (USAMC)	1.182	1.000	1.000	1.000
University of Missouri Research Reactor (U of MO)	7.599	6.429	6.429	6.429
West Valley Demonstration Project (WVDP)	N/A	N/A	N/A	N/A

[a] Volume factors are used in Equation A-8
[b] PDGP is assumed to be the projected BIR-3 volume
[c] RFETS residue data are not included in the volume factors
[d] IDB volume of 9,200 cubic meters used for SRS

N/A = Not Applicable

Table A-29
Volume Factors Used to Estimate Total RH-TRU Waste Radionuclide Inventories at Each Site [a]

Site	Proposed Action	Action Alternative 1	Action Alternative 2	Action Alternative 3	No Action Alternative 1	No Action Alternative 2
ETEC	1.179	1.179	1.179	1.179	1.179	1.179
Hanford	9.779	150.708	150.708	150.708	145.610	150.708
INEEL	1.000	3.009	3.009	3.009	1.000	3.009
Knolls [b]	0.000	6.712	6.712	6.712	0.000	6.712
LANL	2.421	3.677	3.677	3.677	2.421	3.677
ORNL	0.084	1.292	1.292	1.292	1.246	1.292
WVDP	0.000	5.606	5.606	5.606	0.000	5.606

[a] Volume factors are used in Equation A-8
[b] IDB volume for Knolls assumed to be 20 percent of the BIR-3 projected volume

(250,000 cubic feet). As stated previously, volume factors among the WM PEIS regionalized options within an alternative are the same.

A.4.2 Radionuclide Estimates for the Proposed Action

The following sections discuss the radionuclide estimates used for risk analyses of the Proposed Action.

A.4.2.1 CH-TRU Waste

Assuming that the volume of CH-TRU waste disposed of at WIPP would be the maximum allowed under the LWA, the Proposed Action CH-TRU waste radionuclide inventory, summed over all radionuclides, would be approximately 5.8×10^6 Ci. The Proposed Action CH-TRU waste radionuclide inventory for each treatment site is given in Table A-30. Only radionuclides with total inventories of 1 Ci or more were included in the analyses.

When radionuclide inventory data were not available for a site, the average radionuclide concentration taken from the other sites were assigned. RFETS residue data were not included in the determination of the average radionuclide concentrations due to the unique characteristics of the residues.

Plutonium-238 (Pu-238) is a major contributor to the total radionuclide activity, accounting for 32 percent of the total activity. Approximately half of the Pu-238 is generated at SRS. These SEIS-II data are higher than that reported in SEIS-I, where Pu-238 represented only about 1 percent of the total CH-TRU waste activity.

A.4.2.2 RH-TRU Waste

The RH-TRU waste radionuclide inventory according to treatment site under the Proposed Action is given in Table A-31. Radionuclides with total inventories of less than 1 Ci were not included in the analysis. The total radionuclide loading for the Proposed Action is 4.3×10^5 Ci, which is less than the LWA imposed limit of 5.1×10^6 Ci.

A.4.3 Radionuclide Estimates for Action Alternative 1

The following sections describe the radionuclide estimates for the risk analyses conducted for Action Alternative 1.

A.4.3.1 CH-TRU Waste

CH-TRU waste radionuclide inventory according to treatment site for Action Alternative 1 is given in Table A-32. Radionuclides with total inventories of less than 1 Ci were not included in the analysis.

When radionuclide inventory data were not available for a site, the average radionuclide concentration taken from the other sites were assigned. As previously mentioned, RFETS residue data were excluded when computing the average concentration of CH-TRU waste. The total radionuclide loading for CH-TRU waste under Action Alternative 1 is 7.3×10^6 Ci.

Table A-30
Radionuclide Inventories (curies) for CH-TRU Waste by Treatment Site for the Proposed Action [a]

Isotope	Hanford	INEEL	LANL	LLNL	MOUND	NTS	ORNL	RFETS	SRS	ANL-E	Total
Pu-238	4.45E+05	8.40E+04	2.58E+05	4.69E+02	5.88E+02	3.79E+04	7.44E+03	1.20E+04	1.31E+06	2.02E+01	2.151E+06
Pu-241	2.09E+05	1.84E+05	3.76E+03	1.00E+04	0.00E+00	2.96E+02	7.26E+04	1.60E+06	2.64E+05	5.21E+02	2.344E+06
Pu-239	1.45E+05	5.01E+04	1.78E+05	1.01E+03	7.42E+00	3.33E+03	4.63E+03	2.85E+05	2.21E+04	3.14E+02	6.897E+05
Am-241	2.61E+04	1.08E+05	2.63E+04	8.81E+02	0.00E+00	3.43E+02	2.73E+03	2.31E+05	4.74E+03	5.65E+01	4.008E+05
Pu-240	3.39E+04	1.21E+04	2.35E+02	3.95E+02	0.00E+00	3.28E+01	1.48E+03	1.21E+05	5.45E+03	9.03E+01	1.742E+05
Cs-137	3.77E+03	8.40E+01	1.08E+02	1.02E-05	0.00E+00	4.34E-01	4.09E+00	0.00E+00	1.77E+01	0.00E+00	3.988E+03
Ba-137m	3.57E+03	7.95E+01	1.03E+02	9.64E-06	0.00E+00	4.11E-01	3.87E+00	0.00E+00	1.68E+01	0.00E+00	3.772E+03
Cm-244	3.77E+02	6.33E+02	3.53E+02	4.01E+02	0.00E+00	2.75E+02	1.59E+03	0.00E+00	2.76E+03	0.00E+00	6.390E+03
Y-90	3.82E+03	3.75E+01	1.01E+02	0.00E+00	0.00E+00	6.10E-01	2.21E+03	0.00E+00	1.65E+01	0.00E+00	6.189E+03
Sr-90	3.82E+03	3.75E+01	1.01E+02	0.00E+00	0.00E+00	6.10E-01	2.21E+03	0.00E+00	1.65E+01	0.00E+00	6.187E+03
U-233	4.42E+02	1.08E+03	1.01E+02	3.65E-08	0.00E+00	2.18E+00	2.67E+02	1.45E+02	8.86E+00	2.88E-01	2.048E+03
Pu-242	2.10E+00	8.83E+00	1.09E+03	1.24E-01	0.00E+00	1.05E-01	1.64E+00	1.08E-03	8.86E-01	9.59E-02	1.105E+03
U-234	2.97E+02	9.00E+00	1.37E+01	2.02E-02	2.92E-02	1.51E-02	2.37E+01	5.38E-02	6.04E+01	0.00E+00	4.037E+02
Pa-233	1.50E+00	2.04E+00	8.97E-02	2.89E-03	0.00E+00	6.97E-03	8.95E+01	1.91E-01	2.03E+01	0.00E+00	1.136E+02
Np-237	1.50E+00	2.04E+00	8.97E-02	2.89E-03	0.00E+00	6.97E-03	8.95E+01	1.91E-01	2.03E+01	0.00E+00	1.136E+02
Co-60	0.00E+00	7.48E+01	1.65E-02	0.00E+00	0.00E+00	0.00E+00	1.65E-01	0.00E+00	8.40E-01	0.00E+00	7.580E+01
Eu-155	5.83E-03	1.30E+00	5.56E-01	0.00E+00	0.00E+00	4.58E-03	1.41E-01	0.00E+00	1.25E+02	0.00E+00	1.267E+02
Cf-252	1.95E+02	5.69E-01	9.42E-03	0.00E+00	0.00E+00	2.05E-02	3.33E-01	0.00E+00	8.55E-01	0.00E+00	1.965E+02
Pb-212	2.86E-01	3.14E+01	2.08E-02	0.00E+00	0.00E+00	1.98E-02	4.93E-01	0.00E+00	2.17E-02	0.00E+00	3.225E+01
Ra-224	2.86E-01	3.14E+01	9.97E-03	0.00E+00	0.00E+00	2.07E-02	4.93E-01	0.00E+00	2.17E-02	0.00E+00	3.224E+01
Bi-212	2.86E-01	3.14E+01	9.97E-03	0.00E+00	0.00E+00	1.98E-02	4.93E-01	0.00E+00	2.17E-02	0.00E+00	3.224E+01
Po-216	2.86E-01	3.14E+01	9.97E-03	0.00E+00	0.00E+00	1.98E-02	4.93E-01	0.00E+00	2.17E-02	0.00E+00	3.224E+01
Rn-220	2.86E-01	3.14E+01	9.97E-03	0.00E+00	0.00E+00	1.98E-02	4.93E-01	0.00E+00	2.17E-02	0.00E+00	3.224E+01
Th-228	2.86E-01	3.14E+01	9.97E-03	0.00E+00	0.00E+00	1.98E-02	4.93E-01	0.00E+00	2.17E-02	0.00E+00	3.224E+01
U-232	0.00E+00	3.03E+01	1.05E-02	0.00E+00	0.00E+00	1.98E-02	5.01E-01	0.00E+00	2.11E-01	0.00E+00	3.108E+01
Np-239	4.98E-01	7.86E-01	8.61E+00	1.50E-01	0.00E+00	1.47E+00	2.23E+01	0.00E+00	1.78E+00	9.12E-01	3.648E+01
Am-243	4.98E-01	7.35E-01	8.61E+00	1.50E-01	0.00E+00	1.47E+00	1.74E+01	0.00E+00	1.78E+00	9.12E-01	3.158E+01
Tc-99	5.26E-05	2.85E-01	2.77E-02	0.00E+00	0.00E+00	7.22E-05	2.66E+01	0.00E+00	1.06E-05	0.00E+00	2.695E+01
Po-212	1.84E-01	2.01E+01	6.39E-03	0.00E+00	0.00E+00	1.27E-02	3.16E-01	0.00E+00	1.39E-02	0.00E+00	2.066E+01
Cm-245	9.27E+01	2.66E-01	4.42E-03	0.00E+00	0.00E+00	1.14E-05	4.42E-02	0.00E+00	0.00E+00	0.00E+00	9.301E+01
Tl-208	1.03E-01	1.13E+01	3.58E-03	0.00E+00	0.00E+00	7.10E-03	1.77E-01	0.00E+00	7.81E-03	0.00E+00	1.158E+01
U-237	5.12E+00	4.48E+00	9.18E-02	2.45E-01	0.00E+00	7.26E-03	1.78E+00	1.43E+01	3.59E+00	0.00E+00	2.965E+01
Ra-226	1.74E-04	1.80E-01	2.04E+00	2.45E-07	5.28E-09	3.01E-01	9.78E+00	0.00E+00	1.72E-05	0.00E+00	1.230E+01
Po-218	1.74E-04	1.79E-01	2.04E+00	0.00E+00	5.28E-09	3.01E-01	9.71E+00	0.00E+00	1.72E-05	0.00E+00	1.223E+01
Rn-222	1.74E-04	1.79E-01	2.04E+00	0.00E+00	5.28E-09	3.01E-01	9.71E+00	0.00E+00	1.72E-05	0.00E+00	1.223E+01
Bi-214	1.74E-04	1.79E-01	2.04E+00	0.00E+00	5.28E-09	3.01E-01	9.71E+00	0.00E+00	1.72E-05	0.00E+00	1.222E+01
Pb-214	1.74E-04	1.79E-01	2.04E+00	0.00E+00	5.28E-09	3.01E-01	9.71E+00	0.00E+00	1.72E-05	0.00E+00	1.222E+01
Po-214	1.74E-04	1.79E-01	2.04E+00	0.00E+00	5.28E-09	3.01E-01	9.71E+00	0.00E+00	1.72E-05	0.00E+00	1.222E+01
Ag-109m	0.00E+00	1.04E-01	1.47E+01	0.00E+00	0.00E+00	0.00E+00	1.72E-02	0.00E+00	0.00E+00	0.00E+00	1.487E+01
Cd-109	0.00E+00	1.04E-01	1.47E+01	0.00E+00	0.00E+00	0.00E+00	1.72E-02	0.00E+00	0.00E+00	0.00E+00	1.486E+01
Pa-234m	3.24E+01	2.33E-01	5.51E-02	1.86E-01	0.00E+00	4.17E-04	7.95E-02	0.00E+00	1.35E-02	0.00E+00	3.293E+01
Th-234	3.24E+01	2.33E-01	5.47E-02	1.86E-01	0.00E+00	3.84E-04	7.95E-02	0.00E+00	1.35E-02	0.00E+00	3.293E+01
U-238	3.24E+01	2.33E-01	5.47E-02	1.86E-01	0.00E+00	1.98E-04	7.95E-02	0.00E+00	1.35E-02	0.00E+00	3.293E+01
Pm-147	2.64E-01	3.18E+00	4.51E+00	0.00E+00	0.00E+00	1.26E-01	4.15E-02	0.00E+00	2.92E-05	0.00E+00	8.118E+00
U-235	9.44E+00	1.10E-01	1.19E+00	3.63E-03	1.24E-07	1.41E-02	2.60E-02	5.35E-04	1.38E-02	0.00E+00	1.080E+01
Th-231	9.44E+00	1.10E-01	1.19E+00	1.08E-02	1.24E-07	7.41E-05	2.78E-02	5.35E-04	1.38E-02	0.00E+00	1.079E+01
Ac-225	7.22E-01	1.83E+00	1.82E-01	0.00E+00	0.00E+00	2.90E-03	3.14E-01	0.00E+00	3.09E-05	0.00E+00	3.047E+00
Th-229	7.22E-01	1.83E+00	1.82E-01	0.00E+00	0.00E+00	2.90E-03	3.14E-01	0.00E+00	3.09E-05	0.00E+00	3.046E+00
Ra-225	7.22E-01	1.83E+00	1.82E-01	0.00E+00	0.00E+00	2.90E-03	3.14E-01	0.00E+00	3.09E-05	0.00E+00	3.046E+00
At-217	7.22E-01	1.83E+00	1.82E-01	0.00E+00	0.00E+00	2.90E-03	3.14E-01	0.00E+00	3.09E-05	0.00E+00	3.046E+00
Bi-213	7.22E-01	1.83E+00	1.82E-01	0.00E+00	0.00E+00	2.90E-03	3.14E-01	0.00E+00	3.09E-05	0.00E+00	3.046E+00
Fr-221	7.22E-01	1.83E+00	1.82E-01	0.00E+00	0.00E+00	2.90E-03	3.14E-01	0.00E+00	3.09E-05	0.00E+00	3.046E+00
Pb-209	7.22E-01	1.83E+00	1.82E-01	0.00E+00	0.00E+00	2.90E-03	3.14E-01	0.00E+00	3.09E-05	0.00E+00	3.046E+00
Po-213	7.06E-01	1.79E+00	1.78E-01	0.00E+00	0.00E+00	2.84E-03	3.07E-01	0.00E+00	3.03E-05	0.00E+00	2.980E+00
C-14	8.83E+00	2.25E-01	4.65E-04	0.00E+00	0.00E+00	3.01E-04	4.64E-03	0.00E+00	0.00E+00	0.00E+00	9.063E+00
Bi-210	2.93E-05	5.80E-02	6.31E-01	0.00E+00	6.14E-10	8.06E-02	1.89E+00	0.00E+00	2.62E-06	0.00E+00	2.662E+00
Po-210	2.93E-05	5.80E-02	6.31E-01	0.00E+00	6.14E-10	8.06E-02	1.89E+00	0.00E+00	2.62E-06	0.00E+00	2.662E+00
Pb-210	2.93E-05	5.80E-02	6.31E-01	0.00E+00	6.14E-10	8.06E-02	1.89E+00	0.00E+00	2.62E-06	0.00E+00	2.662E+00
Eu-152	4.05E-06	2.11E-01	1.26E-03	8.15E-06	0.00E+00	1.28E+00	4.14E-03	0.00E+00	0.00E+00	0.00E+00	1.493E+00
Cm-243	8.40E-02	1.78E-02	2.49E+00	0.00E+00	0.00E+00	0.00E+00	2.95E-03	0.00E+00	0.00E+00	0.00E+00	2.594E+00
Eu-154	3.44E-04	7.78E-01	5.54E-02	3.22E-06	0.00E+00	5.16E-01	2.88E-03	0.00E+00	6.71E-04	0.00E+00	1.353E+00

[a] This table lists only radionuclides with activities of 1 curie or more. Information about radionuclides with lower activities is contained in Table 4-6 of the Compliance Certification Application (DOE 1996f).

Table A-31
Radionuclide Inventories (curies) for RH-TRU Waste by Treatment Site for the Proposed Action [a]

Isotope	Hanford	INEEL	LANL	ORNL	Total
Y-90	6.32E+04	2.45E+04	3.01E+02	1.07E+04	9.88E+04
Sr-90	6.32E+04	2.45E+04	3.00E+02	1.07E+04	9.87E+04
Cs-137	6.83E+04	1.18E+04	3.27E+02	4.19E+03	8.46E+04
Ba-137m	6.46E+04	1.12E+04	3.10E+02	3.96E+03	8.00E+04
Pu-241	4.56E+04	2.52E+03	0.00E+00	8.40E+02	4.90E+04
Eu-152	0.00E+00	1.92E+03	1.23E-03	9.59E+02	2.88E+03
Eu-154	0.00E+00	9.31E+02	8.47E-02	4.64E+02	1.40E+03
Cm-244	0.00E+00	5.79E+02	0.00E+00	2.76E+02	8.55E+02
Co-60	3.29E+03	5.22E+02	1.01E+01	2.24E+02	4.05E+03
Pu-239	3.28E+03	3.23E+02	2.25E+02	1.08E+02	3.93E+03
Am-241	1.89E+03	3.00E+02	0.00E+00	1.06E+02	2.30E+03
Eu-155	0.00E+00	1.85E+02	4.29E+00	9.22E+01	2.82E+02
Pu-240	1.63E+03	1.26E+02	0.00E+00	3.44E+01	1.79E+03
Th-231	1.43E+00	9.74E+01	2.13E-02	4.86E+01	1.48E+02
U-235	1.43E+00	9.74E+01	2.13E-02	4.86E+01	1.48E+02
Pu-238	4.57E+02	1.50E+02	9.44E+00	3.25E+01	6.48E+02
Cm-243	0.00E+00	7.77E+01	0.00E+00	3.88E+01	1.17E+02
Cs-134	0.00E+00	8.95E+01	5.85E-02	1.29E+01	1.02E+02
U-233	4.06E+00	3.08E+01	0.00E+00	1.51E+01	5.00E+01
Pm-147	0.00E+00	3.24E+01	2.73E+01	5.95E+00	6.57E+01
Rh-106	0.00E+00	1.74E+01	8.18E-01	8.56E+00	2.67E+01
Ru-106	0.00E+00	1.74E+01	8.18E-01	8.56E+00	2.67E+01
Pr-144	0.00E+00	1.47E+01	3.84E-02	4.92E+00	1.97E+01
Ce-144	0.00E+00	1.32E+01	3.88E-02	4.13E+00	1.73E+01
C-14	0.00E+00	3.26E+00	0.00E+00	1.61E+00	4.87E+00
Kr-85	0.00E+00	9.07E+00	0.00E+00	1.06E+00	1.01E+01
Sb-125	0.00E+00	3.24E+00	6.75E+00	7.66E-01	1.08E+01
Cf-252	0.00E+00	2.02E+00	0.00E+00	1.01E+00	3.03E+00
Ni-63	0.00E+00	5.34E+00	0.00E+00	6.23E-01	5.96E+00
U-238	1.00E-01	1.78E+00	4.84E-05	8.84E-01	2.76E+00
Pa-234m	1.00E-01	1.77E+00	4.84E-05	8.84E-01	2.76E+00
Th-234	1.00E-01	1.77E+00	4.84E-05	8.84E-01	2.76E+00
U-232	0.00E+00	9.25E-01	0.00E+00	4.62E-01	1.39E+00
Po-216	1.46E-02	8.84E-01	0.00E+00	4.40E-01	1.34E+00
Bi-212	1.46E-02	8.79E-01	0.00E+00	4.39E-01	1.33E+00
Pb-212	1.46E-02	8.79E-01	0.00E+00	4.39E-01	1.33E+00
Ra-224	1.46E-02	8.79E-01	0.00E+00	4.39E-01	1.33E+00
Rn-220	1.46E-02	8.79E-01	0.00E+00	4.39E-01	1.33E+00
Th-228	1.46E-02	8.79E-01	0.00E+00	4.39E-01	1.33E+00
U-234	1.26E+01	9.08E-01	2.69E-05	2.57E-01	1.38E+01
Po-212	9.33E-03	5.63E-01	0.00E+00	2.81E-01	8.53E-01
Te-125m	0.00E+00	7.94E-01	1.67E+00	1.88E-01	2.65E+00

[a] This table lists only radionuclides with activities of 1 curie or more. Information about radionuclides with lower activities is contained in Table 4-6 of the Compliance Certification Application (DOE 1996f).

Table A-32
Radionuclide Inventories (curies) for CH-TRU Waste by Treatment Site for Action Alternative 1

Isotope	Hanford	INEEL	LANL	LLNL	MOUND	NTS	ORNL	RFETS	SRS	ANL-E	Total
Pu-238	7.86E+05	1.93E+05	3.62E+05	3.97E+02	3.06E+03	3.21E+04	7.74E+03	1.14E+04	1.40E+06	1.71E+01	2.79E+06
Pu-241	3.69E+05	4.58E+05	5.24E+03	8.46E+03	1.22E+03	2.50E+02	6.40E+04	1.51E+06	2.83E+05	4.40E+02	2.70E+06
Pu-239	2.57E+05	1.23E+05	2.50E+05	8.52E+02	5.24E+02	2.82E+03	4.12E+03	2.68E+05	2.36E+04	2.66E+02	9.30E+05
Am-241	4.62E+04	2.73E+05	3.69E+04	7.46E+02	3.69E+02	2.90E+02	2.44E+03	2.12E+05	5.07E+03	4.78E+01	5.77E+05
Pu-240	6.00E+04	3.00E+04	3.28E+02	3.34E+02	8.09E+01	2.78E+01	1.31E+03	1.08E+05	5.84E+03	7.64E+01	2.06E+05
Cs-137	6.67E+03	1.94E+02	1.52E+02	8.62E-06	2.43E+00	3.67E-01	3.85E+00	0.00E+00	1.90E+01	0.00E+00	7.04E+03
Ba-137m	6.31E+03	1.84E+02	1.44E+02	8.16E-06	2.30E+00	3.47E-01	3.64E+00	0.00E+00	1.80E+01	0.00E+00	6.66E+03
Cm-244	6.67E+02	1.53E+03	4.95E+02	3.39E+02	9.83E+00	2.33E+02	1.40E+03	0.00E+00	2.95E+03	0.00E+00	7.63E+03
Y-90	6.76E+03	4.11E+01	1.41E+02	0.00E+00	6.74E+00	5.16E-01	1.94E+03	0.00E+00	1.77E+01	0.00E+00	8.91E+03
Sr-90	6.76E+03	4.11E+01	1.41E+02	0.00E+00	6.74E+00	5.16E-01	1.94E+03	0.00E+00	1.76E+01	0.00E+00	8.91E+03
U-233	7.81E+02	2.72E+03	1.41E+02	3.08E-08	3.70E+00	1.85E+00	2.36E+02	1.23E+02	9.49E+00	2.43E-01	4.02E+03
Pu-242	3.74E+00	1.06E+01	1.53E+03	1.05E-01	1.48E+00	8.88E-02	1.59E+00	9.11E-04	9.49E-01	8.11E-02	1.55E+03
U-234	5.25E+02	2.03E+01	1.92E+01	1.70E-02	3.50E-01	1.28E-02	2.08E+01	4.55E-02	6.47E+01	0.00E+00	6.50E+02
Pa-233	2.65E+00	3.60E+00	1.20E-01	2.44E-03	1.99E-01	5.89E-03	7.58E+01	1.61E-01	2.17E+01	0.00E+00	1.04E+02
Np-237	2.65E+00	3.60E+00	1.20E-01	2.44E-03	1.99E-01	5.89E-03	7.57E+01	1.61E-01	2.17E+01	0.00E+00	1.04E+02
Co-60	0.00E+00	1.89E+02	1.79E-02	0.00E+00	1.91E-01	0.00E+00	1.65E-01	0.00E+00	8.99E-01	0.00E+00	1.90E+02
Eu-155	1.03E-02	2.00E+00	7.76E-01	0.00E+00	1.62E-01	3.87E-03	1.41E-01	0.00E+00	1.33E+02	0.00E+00	1.37E+02
Cf-252	3.44E+02	5.73E-01	1.02E-02	0.00E+00	1.09E-01	1.74E-02	3.04E-01	0.00E+00	9.15E-01	0.00E+00	3.46E+02
Pb-212	5.06E-01	7.93E+01	2.70E-02	0.00E+00	8.07E-02	1.67E-02	4.42E-01	0.00E+00	2.33E-02	0.00E+00	8.04E+01
Ra-224	5.06E-01	7.93E+01	1.17E-02	0.00E+00	8.07E-02	1.75E-02	4.42E-01	0.00E+00	2.33E-02	0.00E+00	8.04E+01
Bi-212	5.06E-01	7.93E+01	1.17E-02	0.00E+00	8.07E-02	1.67E-02	4.42E-01	0.00E+00	2.33E-02	0.00E+00	8.04E+01
Po-216	5.06E-01	7.93E+01	1.17E-02	0.00E+00	8.07E-02	1.67E-02	4.42E-01	0.00E+00	2.33E-02	0.00E+00	8.04E+01
Rn-220	5.06E-01	7.93E+01	1.17E-02	0.00E+00	8.07E-02	1.67E-02	4.42E-01	0.00E+00	2.33E-02	0.00E+00	8.04E+01
Th-228	5.06E-01	7.93E+01	1.17E-02	0.00E+00	8.07E-02	1.67E-02	4.42E-01	0.00E+00	2.33E-02	0.00E+00	8.04E+01
U-232	0.00E+00	7.66E+01	1.26E-02	0.00E+00	7.81E-02	1.68E-02	4.49E-01	0.00E+00	2.26E-01	0.00E+00	7.74E+01
Np-239	1.01E+00	1.48E+00	1.21E+01	1.27E-01	6.47E-02	1.25E+00	1.96E+01	0.00E+00	1.91E+00	7.72E-01	3.83E+01
Am-243	1.01E+00	1.43E+00	1.21E+01	1.27E-01	5.48E-02	1.25E+00	1.53E+01	0.00E+00	1.91E+00	7.72E-01	3.40E+01
Tc-99	9.29E-05	2.89E-01	3.74E-02	0.00E+00	5.41E-02	6.11E-05	2.34E+01	0.00E+00	1.14E-05	0.00E+00	2.38E+01
Po-212	3.24E-01	5.08E+01	7.52E-03	0.00E+00	5.17E-02	1.07E-02	2.83E-01	0.00E+00	1.49E-02	0.00E+00	5.15E+01
Cm-245	1.64E+02	2.66E-01	4.77E-03	0.00E+00	5.09E-02	9.62E-06	4.42E-02	0.00E+00	0.00E+00	0.00E+00	1.64E+02
Tl-208	1.82E-01	2.85E+01	4.22E-03	0.00E+00	2.90E-02	6.01E-03	1.59E-01	0.00E+00	8.36E-03	0.00E+00	2.89E+01
U-237	9.04E+00	1.12E+01	1.28E-01	2.07E-01	2.63E-02	6.14E-03	1.57E+00	1.21E+01	3.84E+00	0.00E+00	3.81E+01
Ra-226	1.10E-01	2.68E-01	2.86E+00	2.07E-07	2.36E-02	2.54E-01	8.61E+00	0.00E+00	1.84E-05	0.00E+00	1.21E+01
Po-218	1.10E-01	2.67E-01	2.86E+00	0.00E+00	2.35E-02	2.54E-01	8.54E+00	0.00E+00	1.84E-05	0.00E+00	1.21E+01
Rn-222	1.10E-01	2.67E-01	2.86E+00	0.00E+00	2.35E-02	2.54E-01	8.54E+00	0.00E+00	1.84E-05	0.00E+00	1.21E+01
Bi-214	1.10E-01	2.67E-01	2.86E+00	0.00E+00	2.34E-02	2.54E-01	8.54E+00	0.00E+00	1.84E-05	0.00E+00	1.21E+01
Pb-214	1.10E-01	2.67E-01	2.86E+00	0.00E+00	2.34E-02	2.54E-01	8.54E+00	0.00E+00	1.84E-05	0.00E+00	1.21E+01
Po-214	1.10E-01	2.67E-01	2.86E+00	0.00E+00	2.34E-02	2.54E-01	8.54E+00	0.00E+00	1.84E-05	0.00E+00	1.21E+01
Ag-109m	0.00E+00	1.04E-01	2.07E+01	0.00E+00	1.99E-02	0.00E+00	1.72E-02	0.00E+00	0.00E+00	0.00E+00	2.08E+01
Cd-109	0.00E+00	1.04E-01	2.07E+01	0.00E+00	1.99E-02	0.00E+00	1.72E-02	0.00E+00	0.00E+00	0.00E+00	2.08E+01
Pa-234m	5.72E+01	4.45E-01	7.69E-02	1.57E-01	1.84E-02	3.53E-04	7.19E-02	0.00E+00	1.44E-02	0.00E+00	5.80E+01
Th-234	5.72E+01	4.45E-01	7.63E-02	1.57E-01	1.84E-02	3.24E-04	7.19E-02	0.00E+00	1.44E-02	0.00E+00	5.80E+01
U-238	5.72E+01	4.45E-01	7.63E-02	1.57E-01	1.84E-02	1.67E-04	7.19E-02	0.00E+00	1.44E-02	0.00E+00	5.80E+01
Pm-147	4.66E-01	7.97E+00	6.33E+00	0.00E+00	1.46E-02	1.07E-01	3.80E-02	0.00E+00	3.12E-05	0.00E+00	1.49E+01
U-235	1.67E+01	2.23E-01	1.66E+00	3.08E-03	7.07E-03	1.19E-02	2.36E-02	4.52E-04	1.48E-02	0.00E+00	1.86E+01
Th-231	1.67E+01	2.23E-01	1.66E+00	9.15E-03	7.04E-03	6.27E-05	2.52E-02	4.52E-04	1.48E-02	0.00E+00	1.86E+01
Ac-225	1.28E+00	4.60E+00	2.55E-01	0.00E+00	5.89E-03	2.46E-03	2.77E-01	0.00E+00	3.31E-05	0.00E+00	6.42E+00
Th-229	1.28E+00	4.60E+00	2.55E-01	0.00E+00	5.89E-03	2.46E-03	2.77E-01	0.00E+00	3.31E-05	0.00E+00	6.42E+00
Ra-225	1.28E+00	4.60E+00	2.55E-01	0.00E+00	5.89E-03	2.46E-03	2.77E-01	0.00E+00	3.31E-05	0.00E+00	6.42E+00
At-217	1.28E+00	4.60E+00	2.55E-01	0.00E+00	5.89E-03	2.46E-03	2.77E-01	0.00E+00	3.31E-05	0.00E+00	6.42E+00
Bi-213	1.28E+00	4.60E+00	2.55E-01	0.00E+00	5.89E-03	2.46E-03	2.77E-01	0.00E+00	3.31E-05	0.00E+00	6.42E+00
Fr-221	1.28E+00	4.60E+00	2.55E-01	0.00E+00	5.89E-03	2.46E-03	2.77E-01	0.00E+00	3.31E-05	0.00E+00	6.42E+00
Pb-209	1.28E+00	4.60E+00	2.55E-01	0.00E+00	5.89E-03	2.46E-03	2.77E-01	0.00E+00	3.31E-05	0.00E+00	6.42E+00
Po-213	1.25E+00	4.50E+00	2.50E-01	0.00E+00	5.76E-03	2.40E-03	2.71E-01	0.00E+00	3.24E-05	0.00E+00	6.28E+00
C-14	1.56E+01	5.29E-01	5.02E-04	0.00E+00	5.36E-03	2.55E-04	4.64E-03	0.00E+00	0.00E+00	0.00E+00	1.61E+01
Bi-210	2.93E-02	1.07E-01	8.86E-01	0.00E+00	5.00E-03	6.82E-02	1.66E+00	0.00E+00	2.80E-06	0.00E+00	2.76E+00
Po-210	2.93E-02	1.07E-01	8.86E-01	0.00E+00	5.00E-03	6.82E-02	1.66E+00	0.00E+00	2.80E-06	0.00E+00	2.76E+00
Pb-210	2.93E-02	1.07E-01	8.86E-01	0.00E+00	5.00E-03	6.82E-02	1.66E+00	0.00E+00	2.80E-06	0.00E+00	2.76E+00
Eu-152	7.16E-06	5.08E-01	1.67E-03	6.90E-06	3.71E-03	1.08E+00	4.03E-03	0.00E+00	0.00E+00	0.00E+00	1.60E+00
Cm-243	1.48E-01	1.78E-02	3.50E+00	0.00E+00	3.41E-03	0.00E+00	2.95E-03	0.00E+00	0.00E+00	0.00E+00	3.67E+00
Eu-154	6.07E-04	1.95E+00	7.77E-02	2.72E-06	3.33E-03	4.37E-01	2.88E-03	0.00E+00	7.19E-04	0.00E+00	2.48E+00

A.4.3.2 RH-TRU Waste

The RH-TRU waste radionuclide inventory according to treatment site under Action Alternative 1 is given in Table A-33. Radionuclides with total inventories of less than 1 Ci were not included in the analysis. The total radionuclide loading for RH-TRU waste under Action Alternative 1 is 5.1×10^6 Ci.

A.4.4 Radionuclide Estimates for Action Alternative 2

The following sections present the radionuclide estimates for the risk analyses of Action Alternative 2.

A.4.4.1 CH-TRU Waste

The CH-TRU waste radionuclide inventory according to treatment site under Action Alternative 2A is given in Table A-34. Treatment based on the Action Alternative 2B is given in Table A-35. Under Action Alternative 2C, all waste is treated at WIPP. The radionuclide inventory values for Action Alternative 2C, therefore, may be obtained from the "Total" column of either Table A-34 or Table A-35. The total radionuclide loading for CH-TRU waste under both Action Alternatives 2A and 2B is 7.3×10^6 Ci.

A.4.4.2 RH-TRU Waste

RH-TRU waste handling would be the same under Action Alternatives 2A, 2B, and 2C. The radionuclide inventories for all are represented in Table A-36. The total radionuclide loading for RH-TRU waste under Action Alternative 2 would be 5.1×10^6 Ci.

A.4.5 Radionuclide Estimates for Action Alternative 3

The following sections discuss radionuclide estimates for risk analyses for Action Alternative 3.

A.4.5.1 CH-TRU Waste

The CH-TRU waste radionuclide inventory according to treatment site under Action Alternative 3 is given in Table A-37. The total radionuclide loading for CH-TRU waste under this alternative would be 7.3×10^6 Ci.

A.4.5.2 RH-TRU Waste

The RH-TRU waste radionuclide inventory according to treatment site under Action Alternative 3 is given in Table A-38. The total radionuclide loading for RH-TRU waste under this alternative would be 5.1×10^6 Ci.

A.4.6 Radionuclide Estimates for No Action Alternative 1

The CH-TRU waste radionuclide inventory for the treatment sites for No Action Alternative 1 would be the same as those under Action Alternatives 2A and 2B. The radionuclide information is given in Tables A-34 and A-35, respectively.

Table A-33
Radionuclide Inventories (curies) for RH-TRU Waste by Treatment Site for Action Alternative 1

Isotope	Hanford	INEEL	LANL	ORNL	Total
Y-90	9.74E+05	2.80E+04	4.56E+02	5.38E+04	1.06E+06
Sr-90	9.74E+05	2.80E+04	4.56E+02	5.38E+04	1.06E+06
Cs-137	1.05E+06	1.56E+04	4.97E+02	1.67E+04	1.09E+06
Ba-137m	9.96E+05	1.48E+04	4.70E+02	1.58E+04	1.03E+06
Pu-241	7.03E+05	2.62E+03	0.00E+00	8.45E+02	7.07E+05
Eu-152	0.00E+00	1.92E+03	1.87E-03	5.39E+03	7.31E+03
Eu-154	0.00E+00	9.33E+02	1.29E-01	2.62E+03	3.55E+03
Cm-244	0.00E+00	5.79E+02	0.00E+00	1.42E+03	2.00E+03
Co-60	5.07E+04	5.48E+02	1.53E+01	9.69E+02	5.22E+04
Pu-239	5.05E+04	3.83E+02	3.41E+02	2.27E+02	5.15E+04
Am-241	2.91E+04	3.94E+02	0.00E+00	4.02E+02	2.99E+04
Eu-155	0.00E+00	1.86E+02	6.52E+00	5.17E+02	7.10E+02
Pu-240	2.51E+04	1.76E+02	0.00E+00	3.57E+01	2.53E+04
Th-231	2.20E+01	9.75E+01	3.23E-02	2.73E+02	3.93E+02
U-235	2.20E+01	9.75E+01	3.23E-02	2.73E+02	3.93E+02
Pu-238	7.04E+03	2.72E+02	1.43E+01	1.84E+02	7.51E+03
Cm-243	0.00E+00	7.77E+01	0.00E+00	2.18E+02	2.96E+02
Cs-134	0.00E+00	1.98E+02	8.89E-02	5.62E+01	2.54E+02
U-233	6.25E+01	3.16E+01	0.00E+00	8.44E+01	1.79E+02
Pm-147	0.00E+00	6.24E+01	4.15E+01	3.51E+01	1.39E+02
Rh-106	0.00E+00	1.75E+01	1.24E+00	5.07E+01	6.94E+01
Ru-106	0.00E+00	1.75E+01	1.24E+00	5.07E+01	6.94E+01
Pr-144	0.00E+00	2.26E+01	5.83E-02	3.34E+01	5.61E+01
Ce-144	0.00E+00	2.12E+01	5.90E-02	2.91E+01	5.03E+01
C-14	0.00E+00	3.34E+00	0.00E+00	8.99E+00	1.23E+01
Kr-85	0.00E+00	2.10E+01	0.00E+00	1.06E+00	2.21E+01
Sb-125	0.00E+00	5.21E+00	1.03E+01	4.34E+00	1.98E+01
Cf-252	0.00E+00	2.02E+00	0.00E+00	5.67E+00	7.69E+00
Ni-63	0.00E+00	1.24E+01	0.00E+00	6.23E-01	1.30E+01
U-238	1.55E+00	1.78E+00	7.36E-05	4.96E+00	8.29E+00
Pa-234m	1.55E+00	1.78E+00	7.36E-05	4.96E+00	8.28E+00
Th-234	1.55E+00	1.78E+00	7.36E-05	4.96E+00	8.28E+00
U-232	0.00E+00	9.25E-01	0.00E+00	2.59E+00	3.52E+00
Po-216	2.24E-01	8.84E-01	0.00E+00	2.47E+00	3.57E+00
Bi-212	2.24E-01	8.79E-01	0.00E+00	2.46E+00	3.57E+00
Pb-212	2.24E-01	8.79E-01	0.00E+00	2.46E+00	3.57E+00
Ra-224	2.24E-01	8.79E-01	0.00E+00	2.46E+00	3.57E+00
Rn-220	2.24E-01	8.79E-01	0.00E+00	2.46E+00	3.57E+00
Th-228	2.24E-01	8.79E-01	0.00E+00	2.46E+00	3.57E+00
U-234	1.95E+02	1.21E+00	4.08E-05	2.63E-01	1.96E+02
Po-212	1.44E-01	5.63E-01	0.00E+00	1.58E+00	2.29E+00
Te-125m	0.00E+00	1.27E+00	2.53E+00	1.06E+00	4.87E+00

Table A-34
Radionuclide Inventories (curies) for CH-TRU Waste by Treatment Site for Action Alternative 2A

Isotope	Hanford	INEEL	LANL	RFETS	SRS	Total
Pu-238	7.88E+05	2.26E+05	3.62E+05	1.14E+04	1.41E+06	2.80E+06
Pu-241	3.78E+05	4.60E+05	5.24E+03	1.51E+06	3.49E+05	2.70E+06
Pu-239	2.58E+05	1.27E+05	2.50E+05	2.68E+05	2.85E+04	9.31E+05
Am-241	4.70E+04	2.75E+05	3.69E+04	2.12E+05	7.93E+03	5.79E+05
Pu-240	6.04E+04	3.02E+04	3.28E+02	1.08E+05	7.31E+03	2.06E+05
Cs-137	6.68E+03	1.96E+02	1.52E+02	0.00E+00	2.53E+01	7.05E+03
Ba-137m	6.32E+03	1.85E+02	1.44E+02	0.00E+00	2.39E+01	6.67E+03
Cm-244	1.01E+03	1.77E+03	4.95E+02	0.00E+00	4.36E+03	7.64E+03
Y-90	6.77E+03	4.17E+01	1.41E+02	0.00E+00	1.97E+03	8.92E+03
Sr-90	6.77E+03	4.16E+01	1.41E+02	0.00E+00	1.97E+03	8.92E+03
U-233	7.82E+02	2.74E+03	1.41E+02	1.23E+02	2.49E+02	4.04E+03
Pu-242	3.86E+00	1.07E+01	1.53E+03	9.11E-04	4.10E+00	1.55E+03
U-234	5.26E+02	2.04E+01	1.92E+01	4.55E-02	8.59E+01	6.51E+02
Pa-233	2.66E+00	3.62E+00	1.20E-01	1.61E-01	9.77E+01	1.04E+02
Np-237	2.66E+00	3.62E+00	1.20E-01	1.61E-01	9.77E+01	1.04E+02
Co-60	0.00E+00	1.90E+02	1.79E-02	0.00E+00	1.25E+00	1.91E+02
Eu-155	1.03E-02	2.01E+00	7.76E-01	0.00E+00	1.34E+02	1.37E+02
Cf-252	3.45E+02	5.91E-01	1.02E-02	0.00E+00	1.33E+00	3.47E+02
Pb-212	5.07E-01	7.98E+01	2.70E-02	0.00E+00	5.46E-01	8.09E+01
Ra-224	5.07E-01	7.98E+01	1.17E-02	0.00E+00	5.46E-01	8.08E+01
Bi-212	5.07E-01	7.98E+01	1.17E-02	0.00E+00	5.46E-01	8.08E+01
Po-216	5.07E-01	7.98E+01	1.17E-02	0.00E+00	5.46E-01	8.08E+01
Rn-220	5.07E-01	7.98E+01	1.17E-02	0.00E+00	5.46E-01	8.08E+01
Th-228	5.07E-01	7.98E+01	1.17E-02	0.00E+00	5.46E-01	8.08E+01
U-232	0.00E+00	7.70E+01	1.26E-02	0.00E+00	7.52E-01	7.78E+01
Np-239	1.13E+00	2.73E+00	1.21E+01	0.00E+00	2.23E+01	3.83E+01
Am-243	1.13E+00	2.68E+00	1.21E+01	0.00E+00	1.81E+01	3.40E+01
Tc-99	9.31E-05	2.89E-01	3.74E-02	0.00E+00	2.35E+01	2.38E+01
Po-212	3.25E-01	5.11E+01	7.52E-03	0.00E+00	3.50E-01	5.18E+01
Cm-245	1.64E+02	2.66E-01	4.77E-03	0.00E+00	9.50E-02	1.65E+02
Tl-208	1.82E-01	2.87E+01	4.22E-03	0.00E+00	1.96E-01	2.90E+01
U-237	9.27E+00	1.13E+01	1.28E-01	1.21E+01	5.44E+00	3.82E+01
Ra-226	1.10E-01	5.23E-01	2.86E+00	0.00E+00	8.63E+00	1.21E+01
Po-218	1.10E-01	5.22E-01	2.86E+00	0.00E+00	8.57E+00	1.21E+01
Rn-222	1.10E-01	5.22E-01	2.86E+00	0.00E+00	8.57E+00	1.21E+01
Bi-214	1.10E-01	5.22E-01	2.86E+00	0.00E+00	8.56E+00	1.21E+01
Pb-214	1.10E-01	5.22E-01	2.86E+00	0.00E+00	8.56E+00	1.21E+01
Po-214	1.10E-01	5.22E-01	2.86E+00	0.00E+00	8.56E+00	1.21E+01
Ag-109m	0.00E+00	1.04E-01	2.07E+01	0.00E+00	3.71E-02	2.08E+01
Cd-109	0.00E+00	1.04E-01	2.07E+01	0.00E+00	3.71E-02	2.08E+01
Pa-234m	5.75E+01	4.47E-01	7.69E-02	0.00E+00	1.05E-01	5.81E+01
Th-234	5.75E+01	4.47E-01	7.63E-02	0.00E+00	1.05E-01	5.81E+01
U-238	5.75E+01	4.47E-01	7.63E-02	0.00E+00	1.05E-01	5.81E+01
Pm-147	4.67E-01	8.12E+00	6.33E+00	0.00E+00	5.26E-02	1.50E+01
U-235	1.67E+01	2.36E-01	1.66E+00	4.52E-04	4.54E-02	1.87E+01
Th-231	1.67E+01	2.24E-01	1.66E+00	4.52E-04	4.70E-02	1.87E+01
Ac-225	1.28E+00	4.63E+00	2.55E-01	0.00E+00	2.83E-01	6.44E+00
Th-229	1.28E+00	4.63E+00	2.55E-01	0.00E+00	2.83E-01	6.44E+00
Ra-225	1.28E+00	4.63E+00	2.55E-01	0.00E+00	2.83E-01	6.44E+00
At-217	1.28E+00	4.63E+00	2.55E-01	0.00E+00	2.83E-01	6.44E+00
Bi-213	1.28E+00	4.63E+00	2.55E-01	0.00E+00	2.83E-01	6.44E+00
Fr-221	1.28E+00	4.63E+00	2.55E-01	0.00E+00	2.83E-01	6.44E+00
Pb-209	1.28E+00	4.63E+00	2.55E-01	0.00E+00	2.83E-01	6.44E+00
Po-213	1.25E+00	4.53E+00	2.50E-01	0.00E+00	2.76E-01	6.30E+00
C-14	1.56E+01	5.32E-01	5.02E-04	0.00E+00	9.99E-03	1.62E+01
Bi-210	2.93E-02	1.76E-01	8.86E-01	0.00E+00	1.67E+00	2.76E+00
Po-210	2.93E-02	1.76E-01	8.86E-01	0.00E+00	1.67E+00	2.76E+00
Pb-210	2.93E-02	1.76E-01	8.86E-01	0.00E+00	1.67E+00	2.76E+00
Eu-152	1.41E-05	1.59E+00	1.67E-03	0.00E+00	7.73E-03	1.60E+00
Cm-243	1.49E-01	1.78E-02	3.50E+00	0.00E+00	6.35E-03	3.67E+00
Eu-154	6.11E-04	2.40E+00	7.77E-02	0.00E+00	6.92E-03	2.49E+00

Table A-35
Radionuclide Inventories (curies) for CH-TRU Waste by Treatment Site for Action Alternative 2B

Isotope	Hanford	INEEL	SRS	Total
Pu-238	7.88E+05	6.00E+05	1.41E+06	2.80E+06
Pu-241	3.78E+05	1.98E+06	3.49E+05	2.70E+06
Pu-239	2.58E+05	6.45E+05	2.85E+04	9.31E+05
Am-241	4.70E+04	5.24E+05	7.93E+03	5.79E+05
Pu-240	6.04E+04	1.39E+05	7.31E+03	2.06E+05
Cs-137	6.68E+03	3.48E+02	2.53E+01	7.05E+03
Ba-137m	6.32E+03	3.29E+02	2.39E+01	6.67E+03
Cm-244	1.01E+03	2.27E+03	4.36E+03	7.64E+03
Y-90	6.77E+03	1.83E+02	1.97E+03	8.92E+03
Sr-90	6.77E+03	1.83E+02	1.97E+03	8.92E+03
U-233	7.82E+02	3.01E+03	2.49E+02	4.04E+03
Pu-242	3.86E+00	1.54E+03	4.10E+00	1.55E+03
U-234	5.26E+02	3.96E+01	8.59E+01	6.51E+02
Pa-233	2.66E+00	3.91E+00	9.77E+01	1.04E+02
Np-237	2.66E+00	3.91E+00	9.77E+01	1.04E+02
Co-60	0.00E+00	1.90E+02	1.25E+00	1.91E+02
Eu-155	1.03E-02	2.79E+00	1.34E+02	1.37E+02
Cf-252	3.45E+02	6.01E-01	1.33E+00	3.47E+02
Pb-212	5.07E-01	7.98E+01	5.46E-01	8.09E+01
Ra-224	5.07E-01	7.98E+01	5.46E-01	8.08E+01
Bi-212	5.07E-01	7.98E+01	5.46E-01	8.08E+01
Po-216	5.07E-01	7.98E+01	5.46E-01	8.08E+01
Rn-220	5.07E-01	7.98E+01	5.46E-01	8.08E+01
Th-228	5.07E-01	7.98E+01	5.46E-01	8.08E+01
U-232	0.00E+00	7.71E+01	7.52E-01	7.78E+01
Np-239	1.13E+00	1.48E+01	2.23E+01	3.83E+01
Am-243	1.13E+00	1.48E+01	1.81E+01	3.40E+01
Tc-99	9.31E-05	3.26E-01	2.35E+01	2.38E+01
Po-212	3.25E-01	5.11E+01	3.50E-01	5.18E+01
Cm-245	1.64E+02	2.71E-01	9.50E-02	1.65E+02
Tl-208	1.82E-01	2.87E+01	1.96E-01	2.90E+01
U-237	9.27E+00	2.35E+01	5.44E+00	3.82E+01
Ra-226	1.10E-01	3.38E+00	8.63E+00	1.21E+01
Po-218	1.10E-01	3.38E+00	8.57E+00	1.21E+01
Rn-222	1.10E-01	3.38E+00	8.57E+00	1.21E+01
Bi-214	1.10E-01	3.38E+00	8.56E+00	1.21E+01
Pb-214	1.10E-01	3.38E+00	8.56E+00	1.21E+01
Po-214	1.10E-01	3.38E+00	8.56E+00	1.21E+01
Ag-109m	0.00E+00	2.08E+01	3.71E-02	2.08E+01
Cd-109	0.00E+00	2.08E+01	3.71E-02	2.08E+01
Pa-234m	5.75E+01	5.24E-01	1.05E-01	5.81E+01
Th-234	5.75E+01	5.23E-01	1.05E-01	5.81E+01
U-238	5.75E+01	5.23E-01	1.05E-01	5.81E+01
Pm-147	4.67E-01	1.45E+01	5.26E-02	1.50E+01
U-235	1.67E+01	1.90E+00	4.54E-02	1.87E+01
Th-231	1.67E+01	1.89E+00	4.70E-02	1.87E+01
Ac-225	1.28E+00	4.88E+00	2.83E-01	6.44E+00
Th-229	1.28E+00	4.88E+00	2.83E-01	6.44E+00
Ra-225	1.28E+00	4.88E+00	2.83E-01	6.44E+00
At-217	1.28E+00	4.88E+00	2.83E-01	6.44E+00
Bi-213	1.28E+00	4.88E+00	2.83E-01	6.44E+00
Fr-221	1.28E+00	4.88E+00	2.83E-01	6.44E+00
Pb-209	1.28E+00	4.88E+00	2.83E-01	6.44E+00
Po-213	1.25E+00	4.78E+00	2.76E-01	6.30E+00
C-14	1.56E+01	5.32E-01	9.99E-03	1.62E+01
Bi-210	2.93E-02	1.06E+00	1.67E+00	2.76E+00
Po-210	2.93E-02	1.06E+00	1.67E+00	2.76E+00
Pb-210	2.93E-02	1.06E+00	1.67E+00	2.76E+00
Eu-152	1.41E-05	1.59E+00	7.73E-03	1.60E+00
Cm-243	1.49E-01	3.51E+00	6.35E-03	3.67E+00
Eu-154	6.11E-04	2.48E+00	6.92E-03	2.49E+00

Table A-36
Radionuclide Inventories (curies) for RH-TRU Waste by Treatment Site for Action Alternatives 2A, 2B, and 2C

Isotope	Hanford	ORNL	Total
Y-90	1.00E+06	5.38E+04	1.06E+06
Sr-90	1.00E+06	5.38E+04	1.06E+06
Cs-137	1.07E+06	1.67E+04	1.09E+06
Ba-137m	1.01E+06	1.58E+04	1.03E+06
Pu-241	7.06E+05	8.45E+02	7.07E+05
Eu-152	1.92E+03	5.39E+03	7.31E+03
Eu-154	9.33E+02	2.62E+03	3.55E+03
Cm-244	5.79E+02	1.42E+03	2.00E+03
Co-60	5.13E+04	9.69E+02	5.22E+04
Pu-239	5.12E+04	2.27E+02	5.15E+04
Am-241	2.95E+04	4.02E+02	2.99E+04
Eu-155	1.93E+02	5.17E+02	7.10E+02
Pu-240	2.53E+04	3.57E+01	2.53E+04
Th-231	1.19E+02	2.73E+02	3.93E+02
U-235	1.19E+02	2.73E+02	3.93E+02
Pu-238	7.33E+03	1.84E+02	7.51E+03
Cm-243	7.77E+01	2.18E+02	2.96E+02
Cs-134	1.98E+02	5.62E+01	2.54E+02
U-233	9.41E+01	8.44E+01	1.79E+02
Pm-147	1.04E+02	3.51E+01	1.39E+02
Rh-106	1.87E+01	5.07E+01	6.94E+01
Ru-106	1.87E+01	5.07E+01	6.94E+01
Pr-144	2.27E+01	3.34E+01	5.61E+01
Ce-144	2.12E+01	2.91E+01	5.03E+01
C-14	3.34E+00	8.99E+00	1.23E+01
Kr-85	2.10E+01	1.06E+00	2.21E+01
Sb-125	1.55E+01	4.34E+00	1.98E+01
Cf-252	2.02E+00	5.67E+00	7.69E+00
Ni-63	1.24E+01	6.23E-01	1.30E+01
U-238	3.33E+00	4.96E+00	8.29E+00
Pa-234m	3.32E+00	4.96E+00	8.28E+00
Th-234	3.32E+00	4.96E+00	8.28E+00
U-232	9.25E-01	2.59E+00	3.52E+00
Po-216	1.11E+00	2.47E+00	3.57E+00
Bi-212	1.10E+00	2.46E+00	3.57E+00
Pb-212	1.10E+00	2.46E+00	3.57E+00
Ra-224	1.10E+00	2.46E+00	3.57E+00
Rn-220	1.10E+00	2.46E+00	3.57E+00
Th-228	1.10E+00	2.46E+00	3.57E+00
U-234	1.96E+02	2.63E-01	1.96E+02
Po-212	7.07E-01	1.58E+00	2.29E+00
Te-125m	3.80E+00	1.06E+00	4.87E+00

Table A-37
Radionuclide Inventories (curies) for CH-TRU Waste by Treatment Site for Action Alternative 3

Isotope	Hanford	INEEL	LANL	RFETS	SRS	Total
Pu-238	7.86E+05	2.25E+05	3.62E+05	1.14E+04	1.41E+06	2.79E+06
Pu-241	3.77E+05	4.58E+05	5.24E+03	1.51E+06	3.49E+05	2.70E+06
Pu-239	2.57E+05	1.26E+05	2.50E+05	2.68E+05	2.85E+04	9.30E+05
Am-241	4.69E+04	2.73E+05	3.69E+04	2.12E+05	7.93E+03	5.77E+05
Pu-240	6.03E+04	3.01E+04	3.28E+02	1.08E+05	7.31E+03	2.06E+05
Cs-137	6.67E+03	1.95E+02	1.52E+02	0.00E+00	2.53E+01	7.04E+03
Ba-137m	6.31E+03	1.84E+02	1.44E+02	0.00E+00	2.39E+01	6.66E+03
Cm-244	1.01E+03	1.77E+03	4.95E+02	0.00E+00	4.36E+03	7.63E+03
Y-90	6.76E+03	4.16E+01	1.41E+02	0.00E+00	1.97E+03	8.91E+03
Sr-90	6.76E+03	4.16E+01	1.41E+02	0.00E+00	1.97E+03	8.91E+03
U-233	7.81E+02	2.73E+03	1.41E+02	1.23E+02	2.49E+02	4.02E+03
Pu-242	3.85E+00	1.07E+01	1.53E+03	9.11E-04	4.10E+00	1.55E+03
U-234	5.25E+02	2.03E+01	1.92E+01	4.55E-02	8.59E+01	6.50E+02
Pa-233	2.66E+00	3.61E+00	1.20E-01	1.61E-01	9.77E+01	1.04E+02
Np-237	2.66E+00	3.61E+00	1.20E-01	1.61E-01	9.77E+01	1.04E+02
Co-60	0.00E+00	1.89E+02	1.79E-02	0.00E+00	1.25E+00	1.90E+02
Eu-155	1.03E-02	2.00E+00	7.76E-01	0.00E+00	1.34E+02	1.37E+02
Cf-252	3.44E+02	5.91E-01	1.02E-02	0.00E+00	1.33E+00	3.46E+02
Pb-212	5.06E-01	7.93E+01	2.70E-02	0.00E+00	5.46E-01	8.04E+01
Ra-224	5.06E-01	7.93E+01	1.17E-02	0.00E+00	5.46E-01	8.04E+01
Bi-212	5.06E-01	7.93E+01	1.17E-02	0.00E+00	5.46E-01	8.04E+01
Po-216	5.06E-01	7.93E+01	1.17E-02	0.00E+00	5.46E-01	8.04E+01
Rn-220	5.06E-01	7.93E+01	1.17E-02	0.00E+00	5.46E-01	8.04E+01
Th-228	5.06E-01	7.93E+01	1.17E-02	0.00E+00	5.46E-01	8.04E+01
U-232	0.00E+00	7.66E+01	1.26E-02	0.00E+00	7.52E-01	7.74E+01
Np-239	1.13E+00	2.73E+00	1.21E+01	0.00E+00	2.23E+01	3.83E+01
Am-243	1.13E+00	2.68E+00	1.21E+01	0.00E+00	1.81E+01	3.40E+01
Tc-99	9.29E-05	2.89E-01	3.74E-02	0.00E+00	2.35E+01	2.38E+01
Po-212	3.24E-01	5.08E+01	7.52E-03	0.00E+00	3.50E-01	5.15E+01
Cm-245	1.64E+02	2.66E-01	4.77E-03	0.00E+00	9.50E-02	1.64E+02
Tl-208	1.82E-01	2.85E+01	4.22E-03	0.00E+00	1.96E-01	2.89E+01
U-237	9.25E+00	1.12E+01	1.28E-01	1.21E+01	5.44E+00	3.81E+01
Ra-226	1.10E-01	5.22E-01	2.86E+00	0.00E+00	8.63E+00	1.21E+01
Po-218	1.10E-01	5.21E-01	2.86E+00	0.00E+00	8.57E+00	1.21E+01
Rn-222	1.10E-01	5.21E-01	2.86E+00	0.00E+00	8.57E+00	1.21E+01
Bi-214	1.10E-01	5.21E-01	2.86E+00	0.00E+00	8.56E+00	1.21E+01
TPb-214	1.10E-01	5.21E-01	2.86E+00	0.00E+00	8.56E+00	1.21E+01
Po-214	1.10E-01	5.21E-01	2.86E+00	0.00E+00	8.56E+00	1.21E+01
Ag-109m	0.00E+00	1.04E-01	2.07E+01	0.00E+00	3.71E-02	2.08E+01
Cd-109	0.00E+00	1.04E-01	2.07E+01	0.00E+00	3.71E-02	2.08E+01
Pa-234m	5.73E+01	4.45E-01	7.69E-02	0.00E+00	1.05E-01	5.80E+01
Th-234	5.73E+01	4.45E-01	7.63E-02	0.00E+00	1.05E-01	5.80E+01
U-238	5.73E+01	4.45E-01	7.63E-02	0.00E+00	1.05E-01	5.80E+01
Pm-147	4.66E-01	8.08E+00	6.33E+00	0.00E+00	5.26E-02	1.49E+01
U-235	1.67E+01	2.35E-01	1.66E+00	4.52E-04	4.54E-02	1.86E+01
Th-231	1.67E+01	2.23E-01	1.66E+00	4.52E-04	4.70E-02	1.86E+01
Ac-225	1.28E+00	4.60E+00	2.55E-01	0.00E+00	2.83E-01	6.42E+00
Th-229	1.28E+00	4.60E+00	2.55E-01	0.00E+00	2.83E-01	6.42E+00
Ra-225	1.28E+00	4.60E+00	2.55E-01	0.00E+00	2.83E-01	6.42E+00
At-217	1.28E+00	4.60E+00	2.55E-01	0.00E+00	2.83E-01	6.42E+00
Bi-213	1.28E+00	4.60E+00	2.55E-01	0.00E+00	2.83E-01	6.42E+00
Fr-221	1.28E+00	4.60E+00	2.55E-01	0.00E+00	2.83E-01	6.42E+00
Pb-209	1.28E+00	4.60E+00	2.55E-01	0.00E+00	2.83E-01	6.42E+00
Po-213	1.25E+00	4.50E+00	2.50E-01	0.00E+00	2.76E-01	6.28E+00
C-14	1.56E+01	5.29E-01	5.02E-04	0.00E+00	9.99E-03	1.61E+01
Bi-210	2.93E-02	1.75E-01	8.86E-01	0.00E+00	1.67E+00	2.76E+00
Po-210	2.93E-02	1.75E-01	8.86E-01	0.00E+00	1.67E+00	2.76E+00
Pb-210	2.93E-02	1.75E-01	8.86E-01	0.00E+00	1.67E+00	2.76E+00
Eu-152	1.41E-05	1.59E+00	1.67E-03	0.00E+00	7.73E-03	1.60E+00
Cm-243	1.48E-01	1.78E-02	3.50E+00	0.00E+00	6.35E-03	3.67E+00
Eu-154	6.10E-04	2.39E+00	7.77E-02	0.00E+00	6.92E-03	2.48E+00

Table A-38
Radionuclide Inventories (curies) for RH-TRU Waste by Treatment Site for Action Alternative 3

Isotope	Hanford	ORNL	Total
Y-90	1.00E+06	5.38E+04	1.06E+06
Sr-90	1.00E+06	5.38E+04	1.06E+06
Cs-137	1.07E+06	1.67E+04	1.09E+06
Ba-137m	1.01E+06	1.58E+04	1.03E+06
Pu-241	7.06E+05	8.45E+02	7.07E+05
Eu-152	1.92E+03	5.39E+03	7.31E+03
Eu-154	9.33E+02	2.62E+03	3.55E+03
Cm-244	5.79E+02	1.42E+03	2.00E+03
Co-60	5.13E+04	9.69E+02	5.22E+04
Pu-239	5.12E+04	2.27E+02	5.15E+04
Am-241	2.95E+04	4.02E+02	2.99E+04
Eu-155	1.93E+02	5.17E+02	7.10E+02
Pu-240	2.53E+04	3.57E+01	2.53E+04
Th-231	1.19E+02	2.73E+02	3.93E+02
U-235	1.19E+02	2.73E+02	3.93E+02
Pu-238	7.33E+03	1.84E+02	7.51E+03
Cm-243	7.77E+01	2.18E+02	2.96E+02
Cs-134	1.98E+02	5.62E+01	2.54E+02
U-233	9.41E+01	8.44E+01	1.79E+02
Pm-147	1.04E+02	3.51E+01	1.39E+02
Rh-106	1.87E+01	5.07E+01	6.94E+01
Ru-106	1.87E+01	5.07E+01	6.94E+01
Pr-144	2.27E+01	3.34E+01	5.61E+01
Ce-144	2.12E+01	2.91E+01	5.03E+01
C-14	3.34E+00	8.99E+00	1.23E+01
Kr-85	2.10E+01	1.06E+00	2.21E+01
Sb-125	1.55E+01	4.34E+00	1.98E+01
Cf-252	2.02E+00	5.67E+00	7.69E+00
Ni-63	1.24E+01	6.23E-01	1.30E+01
U-238	3.33E+00	4.96E+00	8.29E+00
Pa-234m	3.32E+00	4.96E+00	8.28E+00
Th-234	3.32E+00	4.96E+00	8.28E+00
U-232	9.25E-01	2.59E+00	3.52E+00
Po-216	1.11E+00	2.47E+00	3.57E+00
Bi-212	1.10E+00	2.46E+00	3.57E+00
Pb-212	1.10E+00	2.46E+00	3.57E+00
Ra-224	1.10E+00	2.46E+00	3.57E+00
Rn-220	1.10E+00	2.46E+00	3.57E+00
Th-228	1.10E+00	2.46E+00	3.57E+00
U-234	1.96E+02	2.63E-01	1.96E+02
Po-212	7.07E-01	1.58E+00	2.29E+00
Te-125m	3.80E+00	1.06E+00	4.87E+00

The RH-TRU waste radionuclide inventory for the treatment sites under this alternative would be the same as those under Action Alternatives 2A and 2B. This radionuclide information is given in Table A-36.

A.4.7 Radionuclide Estimates for No Action Alternative 2

The following sections discuss radionuclide estimates used for the No Action Alternative 2 analyses.

A.4.7.1 CH-TRU Waste

The CH-TRU waste radionuclide inventory for the seven largest generator-storage sites under No Action Alternative 2 is given in Table A-39.

No Action Alternative 2 long-term performance calculations are initiated at the assumed time of loss of institutional control at the generator-storage sites. A decayed radionuclide inventory was required for long-term performance assessment calculations. The CH-TRU waste radionuclide inventory at treatment sites decayed to the year 2133 is given in Table A-40. Only those radionuclides with activities greater than 1 Ci and half-lives greater than approximately 10 minutes have been retained.

A.4.7.2 RH-TRU Waste

The RH-TRU waste radionuclide inventory at each treatment site for No Action Alternative 2 is given in Table A-41. Again, No Action Alternative 2 long-term performance calculations require the use of a decayed radionuclide inventory to account for the 100 years after WIPP would have ceased operation. These values are given in Table A-42. Only those radionuclides with activities greater than 1 Ci and half-lives greater than approximately 10 minutes have been retained.

A.4.8 Radionuclide Concentrations for Drilling Intrusions

The long-term performance assessment analyses presented in Chapter 5 include examination of the effects of drilling into waste disposal rooms. Estimates of the median and 75th percentile concentrations of individual nuclides for drilling intrusions into CH-TRU waste are provided in Table A-43. These values are representative of uncompacted waste, are decayed through the end of 1995, and are based on 457 waste streams. The concentration estimates were based on 1,000 replicates of random sampling of waste streams where the probability of selection of a particular waste stream was proportional to the volume of the waste stream. Each replicate was formed by integrating data from three separate waste streams because the CH-TRU waste drums would be stacked three high in the disposal rooms. The samples were then ordered on the basis of the concentration of Am-241 because more than 85 percent of the dose to the drilling crew came from Am-241.

Only a limited amount of information was available for the RH-TRU waste streams, and the random sampling scheme did not yield representative results. Therefore, the RH-TRU waste concentrations used for all drilling analyses were average results, with no differentiation between median and 75th percentile values.

Table A-39
Radionuclide Inventories (curies) for CH-TRU Waste by Generator-Storage Site for No Action Alternative 2

Isotope	Hanford	INEEL	LANL	LLNL	ORNL	RFETS	SRS
Pu-238	3.76E+05	7.31E+04	2.18E+05	3.97E+02	6.64E+03	1.14E+04	1.10E+06
Pu-241	1.76E+05	1.56E+05	3.20E+03	8.46E+03	6.16E+04	1.51E+06	2.24E+05
Pu-239	1.23E+05	4.28E+04	1.51E+05	8.52E+02	3.98E+03	2.68E+05	1.87E+04
Am-241	2.21E+04	9.20E+04	2.22E+04	7.46E+02	2.36E+03	2.12E+05	4.01E+03
Pu-240	2.87E+04	1.03E+04	2.00E+02	3.34E+02	1.27E+03	1.08E+05	4.61E+03
Cs-137	3.19E+03	7.30E+01	9.17E+01	8.62E-06	3.78E+00	0.00E+00	1.50E+01
Ba-137m	3.02E+03	6.91E+01	8.68E+01	8.16E-06	3.58E+00	0.00E+00	1.42E+01
Cm-244	3.19E+02	5.44E+02	2.98E+02	3.39E+02	1.35E+03	0.00E+00	2.33E+03
Y-90	3.24E+03	3.71E+01	8.51E+01	0.00E+00	1.87E+03	0.00E+00	1.40E+01
Sr-90	3.23E+03	3.71E+01	8.51E+01	0.00E+00	1.87E+03	0.00E+00	1.39E+01
U-233	3.74E+02	9.18E+02	8.52E+01	3.08E-08	2.27E+02	1.23E+02	7.50E+00
Pu-242	1.78E+00	8.66E+00	9.23E+02	1.05E-01	1.58E+00	9.11E-04	7.50E-01
U-234	2.51E+02	7.87E+00	1.16E+01	1.70E-02	2.01E+01	4.55E-02	5.11E+01
Pa-233	1.27E+00	1.89E+00	7.85E-02	2.44E-03	7.57E+01	1.61E-01	1.72E+01
Np-237	1.27E+00	1.89E+00	7.85E-02	2.44E-03	7.57E+01	1.61E-01	1.72E+01
Co-60	0.00E+00	6.34E+01	1.65E-02	0.00E+00	1.65E-01	0.00E+00	7.11E-01
Eu-155	4.93E-03	1.23E+00	4.73E-01	0.00E+00	1.41E-01	0.00E+00	1.06E+02
Cf-252	1.65E+02	5.69E-01	9.42E-03	0.00E+00	2.96E-01	0.00E+00	7.23E-01
Pb-212	2.42E-01	2.66E+01	1.87E-02	0.00E+00	4.28E-01	0.00E+00	1.84E-02
Ra-224	2.42E-01	2.66E+01	9.52E-03	0.00E+00	4.28E-01	0.00E+00	1.84E-02
Bi-212	2.42E-01	2.66E+01	9.52E-03	0.00E+00	4.28E-01	0.00E+00	1.84E-02
Po-216	2.42E-01	2.66E+01	9.52E-03	0.00E+00	4.28E-01	0.00E+00	1.84E-02
Rn-220	2.42E-01	2.66E+01	9.52E-03	0.00E+00	4.28E-01	0.00E+00	1.84E-02
Th-228	2.42E-01	2.66E+01	9.52E-03	0.00E+00	4.28E-01	0.00E+00	1.84E-02
U-232	0.00E+00	2.57E+01	9.96E-03	0.00E+00	4.34E-01	0.00E+00	1.79E-01
Np-239	4.21E-01	7.17E-01	7.29E+00	1.27E-01	1.88E+01	0.00E+00	1.51E+00
Am-243	4.21E-01	6.65E-01	7.29E+00	1.27E-01	1.47E+01	0.00E+00	1.51E+00
Tc-99	4.45E-05	2.84E-01	2.41E-02	0.00E+00	2.25E+01	0.00E+00	8.99E-06
Po-212	1.55E-01	1.71E+01	6.10E-03	0.00E+00	2.74E-01	0.00E+00	1.18E-02
Cm-245	7.84E+01	2.66E-01	4.42E-03	0.00E+00	4.42E-02	0.00E+00	0.00E+00
Tl-208	8.71E-02	9.57E+00	3.42E-03	0.00E+00	1.54E-01	0.00E+00	6.60E-03
U-237	4.33E+00	3.81E+00	7.80E-02	2.07E-01	1.51E+00	1.21E+01	3.04E+00
Ra-226	1.47E-04	1.71E-01	1.72E+00	2.07E-07	8.28E+00	0.00E+00	1.46E-05
Po-218	1.47E-04	1.70E-01	1.72E+00	0.00E+00	8.22E+00	0.00E+00	1.46E-05
Rn-222	1.47E-04	1.70E-01	1.72E+00	0.00E+00	8.22E+00	0.00E+00	1.46E-05
Bi-214	1.47E-04	1.70E-01	1.72E+00	0.00E+00	8.22E+00	0.00E+00	1.46E-05
Pb-214	1.47E-04	1.70E-01	1.72E+00	0.00E+00	8.22E+00	0.00E+00	1.46E-05
Po-214	1.47E-04	1.70E-01	1.72E+00	0.00E+00	8.21E+00	0.00E+00	1.46E-05
Ag-109m	0.00E+00	1.04E-01	1.25E+01	0.00E+00	1.72E-02	0.00E+00	0.00E+00
Cd-109	0.00E+00	1.04E-01	1.25E+01	0.00E+00	1.72E-02	0.00E+00	0.00E+00
Pa-234m	2.74E+01	2.12E-01	4.69E-02	1.57E-01	6.97E-02	0.00E+00	1.14E-02
Th-234	2.74E+01	2.12E-01	4.65E-02	1.57E-01	6.97E-02	0.00E+00	1.14E-02
U-238	2.74E+01	2.12E-01	4.65E-02	1.57E-01	6.97E-02	0.00E+00	1.14E-02
Pm-147	2.23E-01	2.70E+00	3.82E+00	0.00E+00	3.71E-02	0.00E+00	2.47E-05
U-235	7.99E+00	9.87E-02	1.00E+00	3.08E-03	2.30E-02	4.52E-04	1.17E-02
Th-231	7.99E+00	9.85E-02	1.00E+00	9.15E-03	2.45E-02	4.52E-04	1.17E-02
Ac-225	6.10E-01	1.55E+00	1.54E-01	0.00E+00	2.67E-01	0.00E+00	2.61E-05
Th-229	6.10E-01	1.55E+00	1.54E-01	0.00E+00	2.66E-01	0.00E+00	2.61E-05
Ra-225	6.10E-01	1.55E+00	1.54E-01	0.00E+00	2.66E-01	0.00E+00	2.61E-05
At-217	6.10E-01	1.55E+00	1.54E-01	0.00E+00	2.66E-01	0.00E+00	2.61E-05
Bi-213	6.10E-01	1.55E+00	1.54E-01	0.00E+00	2.66E-01	0.00E+00	2.61E-05
Fr-221	6.10E-01	1.55E+00	1.54E-01	0.00E+00	2.66E-01	0.00E+00	2.61E-05
Pb-209	6.10E-01	1.55E+00	1.54E-01	0.00E+00	2.66E-01	0.00E+00	2.61E-05
Po-213	5.97E-01	1.52E+00	1.51E-01	0.00E+00	2.61E-01	0.00E+00	2.56E-05
C-14	7.47E+00	1.94E-01	4.65E-04	0.00E+00	4.64E-03	0.00E+00	0.00E+00
Bi-210	2.48E-05	5.31E-02	5.34E-01	0.00E+00	1.60E+00	0.00E+00	2.22E-06
Po-210	2.48E-05	5.31E-02	5.34E-01	0.00E+00	1.60E+00	0.00E+00	2.22E-06
Pb-210	2.48E-05	5.30E-02	5.34E-01	0.00E+00	1.60E+00	0.00E+00	2.22E-06
Eu-152	3.43E-06	1.82E-01	1.12E-03	6.90E-06	4.00E-03	0.00E+00	0.00E+00
Cm-243	7.11E-02	1.78E-02	2.11E+00	0.00E+00	2.95E-03	0.00E+00	0.00E+00
Eu-154	2.91E-04	6.61E-01	4.69E-02	2.72E-06	2.88E-03	0.00E+00	5.68E-04

Table A-40
Decayed Radionuclide Inventories (curies) Adjusted for Decay Through 2133 for CH-TRU Waste by Generator-Storage Site for No Action Alternative 2

Isotopes	Hanford	INEEL	LANL	LLNL	ORNL	RFETS	SRS
Pu-238	1.71E+05	3.26E+04	9.88E+04	1.80E+02	2.94E+03	5.17E+03	4.61E+05
Pu-239	1.23E+05	4.25E+04	1.51E+05	8.50E+02	3.90E+03	2.67E+05	1.86E+04
Am-241	2.41E+04	8.27E+04	1.90E+04	8.81E+02	3.82E+03	2.25E+05	9.79E+03
Pu-240	2.85E+04	1.01E+04	1.98E+02	3.31E+02	1.27E+03	1.07E+05	4.56E+03
Pu-241	1.52E+03	1.27E+03	2.58E+01	6.87E+01	5.06E+02	1.23E+04	1.12E+03
Pu-242	3.74E+02	9.16E+02	8.52E+01	1.10E-06	2.29E+02	1.23E+02	7.51E+00
U-233	1.78E+00	7.91E+00	9.23E+02	1.05E-01	1.46E+00	9.11E-04	7.50E-01
U-234	3.26E+02	2.18E+01	5.43E+01	9.48E-02	2.16E+01	2.28E+00	2.80E+02
Cs-137	3.20E+02	1.05E+01	9.23E+00	8.63E-07	2.98E+02	0.00E+00	1.19E+00
Y-90	2.97E+02	3.09E+00	7.81E+00	0.00E+00	1.74E+02	0.00E+00	1.01E+00
Sr-90	2.97E+02	3.09E+00	7.81E+00	0.00E+00	1.74E+02	0.00E+00	1.01E+00
Cm-245	6.95E+00	1.17E+01	6.47E+00	7.36E+00	2.97E+01	0.00E+00	3.45E+01
Cm-244	7.80E+01	2.38E-01	3.96E-03	0.00E+00	3.96E-02	0.00E+00	0.00E+00
Th-234	2.74E+01	2.03E-01	4.64E-02	1.57E-01	6.89E-02	0.00E+00	1.14E-02
U-238	2.74E+01	2.03E-01	4.64E-02	1.57E-01	6.89E-02	0.00E+00	1.14E-02
Tc-99	4.18E-01	6.31E-01	7.21E+00	1.26E-01	1.48E+01	0.00E+00	1.50E+00
Pa-233	4.18E-01	6.31E-01	7.21E+00	1.26E-01	1.48E+01	0.00E+00	1.50E+00
Np-237	4.45E-05	2.57E-01	2.37E-02	0.00E+00	2.28E+01	0.00E+00	8.99E-06
Np-239	1.27E+00	1.79E+00	7.69E-02	2.44E-03	1.15E+00	1.61E-01	1.72E+01
Am-243	1.27E+00	1.79E+00	7.69E-02	2.44E-03	1.15E+00	1.61E-01	1.72E+01
Th-229	4.12E+00	1.02E+01	9.53E-01	5.31E-09	2.42E+00	1.16E+00	7.76E-02
Ra-225	4.12E+00	1.01E+01	9.53E-01	5.31E-09	2.42E+00	1.16E+00	7.75E-02
Ac-225	4.12E+00	1.01E+01	9.53E-01	5.30E-09	2.42E+00	1.16E+00	7.75E-02
Pb-212	4.55E-17	1.01E+01	3.65E-03	0.00E+00	1.70E-01	0.00E+00	6.38E-02
Ra-224	4.55E-17	1.01E+01	3.65E-03	0.00E+00	1.70E-01	0.00E+00	6.38E-02
Th-228	4.53E-17	1.01E+01	3.65E-03	0.00E+00	1.70E-01	0.00E+00	6.38E-02
U-232	5.56E-03	1.53E-01	1.65E+00	1.13E-06	8.03E+00	1.81E-05	3.38E-03
Ra-226	1.42E-04	1.52E-01	1.65E+00	1.99E-07	8.03E+00	0.00E+00	1.39E-05
Rn-222	0.00E+00	9.81E+00	3.55E-03	0.00E+00	1.65E-01	0.00E+00	6.21E-02
U-235	1.38E-04	1.50E-01	1.62E+00	1.92E-07	7.84E+00	0.00E+00	1.37E-05
Th-231	1.38E-04	1.50E-01	1.62E+00	1.92E-07	7.84E+00	0.00E+00	1.37E-05
Pb-210	1.38E-04	1.49E-01	1.62E+00	1.92E-07	7.84E+00	0.00E+00	1.37E-05
Bi-210	8.00E+00	9.51E-02	1.00E+00	3.08E-03	2.26E-02	4.52E-04	1.17E-02
Po-210	8.00E+00	9.51E-02	1.00E+00	3.08E-03	2.26E-02	4.52E-04	1.17E-02
C-14	7.40E+00	1.90E-01	4.15E-04	0.00E+00	4.14E-03	0.00E+00	0.00E+00

Table A-41
Radionuclide Inventories (curies) for RH-TRU Waste by Generator-Storage Site for No Action Alternative 2

Isotope	Hanford	INEEL	LANL	ORNL	Total
Y-90	9.41E+05	2.45E+04	3.01E+02	5.16E+04	1.02E+06
Sr-90	9.41E+05	2.45E+04	3.00E+02	5.16E+04	1.02E+06
Cs-137	1.02E+06	1.18E+04	3.27E+02	1.55E+04	1.04E+06
Ba-137M	9.62E+05	1.12E+04	3.10E+02	1.47E+04	9.88E+05
Pu-241	6.79E+05	2.52E+03	0.00E+00	8.39E+02	6.83E+05
Eu-152	0.00E+00	1.92E+03	1.23E-03	5.22E+03	7.13E+03
Eu-154	0.00E+00	9.31E+02	8.47E-02	2.52E+03	3.46E+03
Cm-244	0.00E+00	5.79E+02	0.00E+00	1.37E+03	1.95E+03
Co-60	4.90E+04	5.21E+02	1.01E+01	9.38E+02	5.04E+04
Pu-239	4.88E+04	3.23E+02	2.25E+02	2.22E+02	4.96E+04
Am-241	2.81E+04	3.00E+02	0.00E+00	3.87E+02	2.88E+04
Eu-155	0.00E+00	1.85E+02	4.29E+00	5.00E+02	6.89E+02
Pu-240	2.43E+04	1.26E+02	0.00E+00	3.56E+01	2.44E+04
Th-231	2.13E+01	9.74E+01	2.13E-02	2.64E+02	3.83E+02
U-235	2.13E+01	9.74E+01	2.13E-02	2.64E+02	3.83E+02
Pu-238	6.80E+03	1.50E+02	9.44E+00	6.51E+01	7.03E+03
Cm-243	0.00E+00	7.77E+01	0.00E+00	2.11E+02	2.89E+02
Cs-134	0.00E+00	8.95E+01	5.85E-02	2.40E+01	1.14E+02
U-233	6.04E+01	3.08E+01	0.00E+00	8.17E+01	1.73E+02
Pm-147	0.00E+00	3.24E+01	2.73E+01	5.95E+00	6.57E+01
Rh-106	0.00E+00	1.74E+01	8.18E-01	4.59E+01	6.40E+01
Ru-106	0.00E+00	1.74E+01	8.18E-01	4.59E+01	6.40E+01
Pr-144	0.00E+00	1.47E+01	3.84E-02	2.24E+01	3.71E+01
Ce-144	0.00E+00	1.32E+01	3.88E-02	1.81E+01	3.13E+01
C-14	0.00E+00	3.26E+00	0.00E+00	8.70E+00	1.20E+01
Kr-85	0.00E+00	9.07E+00	0.00E+00	1.06E+00	1.01E+01
Sb-125	0.00E+00	3.24E+00	6.75E+00	7.66E-01	1.08E+01
Cf-252	0.00E+00	2.02E+00	0.00E+00	5.49E+00	7.51E+00
Ni-63	0.00E+00	5.34E+00	0.00E+00	6.23E-01	5.96E+00
U-238	1.49E+00	1.78E+00	4.84E-05	4.80E+00	8.07E+00
Pa-234m	1.49E+00	1.77E+00	4.84E-05	4.80E+00	8.06E+00
Th-234	1.49E+00	1.77E+00	4.84E-05	4.80E+00	8.06E+00
U-232	0.00E+00	9.24E-01	0.00E+00	2.51E+00	3.44E+00
Po-216	2.17E-01	8.84E-01	0.00E+00	2.39E+00	3.49E+00
Bi-212	2.17E-01	8.79E-01	0.00E+00	2.38E+00	3.48E+00
Pb-212	2.17E-01	8.79E-01	0.00E+00	2.38E+00	3.48E+00
Ra-224	2.17E-01	8.79E-01	0.00E+00	2.38E+00	3.48E+00
Rn-220	2.17E-01	8.79E-01	0.00E+00	2.38E+00	3.48E+00
Th-228	2.17E-01	8.79E-01	0.00E+00	2.38E+00	3.48E+00
U-234	1.88E+02	9.08E-01	2.69E-05	2.60E-01	1.89E+02
Po-212	1.39E-01	5.63E-01	0.00E+00	1.53E+00	2.23E+00
Te-125m	0.00E+00	7.93E-01	1.67E+00	1.88E-01	2.65E+00

Table A-42
Decayed Radionuclide Inventories (curies) Adjusted for Decay Through 2133 for RH-TRU Waste by Generator-Storage Site for No Action Alternative 2

Isotope	Hanford	INEEL	LANL	ORNL
Cs-137	1.02E+05	1.18E+03	3.27E+01	1.55E+03
Y-90	8.64E+04	2.25E+03	2.75E+01	4.75E+03
Sr-90	8.64E+04	2.25E+03	2.75E+01	4.75E+03
Pu-239	4.87E+04	3.22E+02	2.24E+02	2.22E+02
Am-241	4.37E+04	3.29E+02	0.00E+00	3.54E+02
Pu-240	2.40E+04	1.26E+02	0.00E+00	3.89E+01
Pu-241	5.52E+03	2.05E+01	0.00E+00	6.82E+00
Pu-238	3.08E+03	6.80E+01	4.28E+00	2.96E+01
Th-231	2.13E+01	9.74E+01	2.13E-02	2.65E+02
U-235	2.13E+01	9.74E+01	2.13E-02	2.65E+02
U-234	1.89E+02	9.38E-01	1.88E-03	2.74E-01
U-233	6.04E+01	3.08E+01	0.00E+00	8.17E+01
Cm-244	0.00E+00	1.26E+01	0.00E+00	2.97E+01
Eu-152	0.00E+00	1.06E+01	6.80E-06	2.88E+01
Cm-243	0.00E+00	6.81E+00	0.00E+00	1.85E+01
C-14	0.00E+00	3.22E+00	0.00E+00	8.61E+00
U-238	1.49E+00	1.78E+00	4.84E-05	4.80E+00
Th-234	1.49E+00	1.78E+00	4.84E-05	4.80E+00
Ni-63	0.00E+00	2.60E+00	0.00E+00	3.03E-01

Table A-43
Radionuclide Concentrations (curies/cubic meter)
for Drilling Intrusions into CH-TRU Waste

CH-TRU Radionuclide	Proposed Action		Action Alternative 1		Action Alternative 2		Action Alternative 3	
	50% Level	75% Level	50% Level	75% Level	50% Level	75% Level	50% Level	75% Level
Am-241	1.89E+00	2.83E+00	1.58E+00	2.37E+00	4.16E+00	6.24E+00	1.33E+00	1.99E+00
Am-243	8.79E-06	1.32E-05	7.35E-06	1.10E-05	1.94E-05	2.91E-05	6.17E-06	9.26E-06
Ba-137m	4.74E-03	3.43E-03	3.96E-03	2.87E-03	1.05E-02	7.57E-03	3.33E-03	2.41E-03
Bi-214	1.13E-06	1.69E-06	9.41E-07	1.41E-06	2.48E-06	3.72E-06	7.91E-07	1.19E-06
C-14	9.79E-10	1.47E-09	8.19E-10	1.23E-09	2.16E-09	3.24E-09	6.88E-10	1.03E-09
Cm-243	4.05E-07	6.07E-07	3.39E-07	5.08E-07	8.93E-07	1.34E-06	2.84E-07	4.26E-07
Cm-244	2.66E-02	3.99E-02	2.23E-02	3.34E-02	5.87E-02	8.80E-02	1.87E-02	2.80E-02
Co-60	5.53E-03	8.30E-03	4.63E-03	6.94E-03	1.22E-02	1.83E-02	3.88E-03	5.83E-03
Cs-137	5.76E-02	8.24E-02	4.82E-02	6.89E-02	1.27E-01	1.82E-01	4.05E-02	5.79E-02
Eu-152	4.11E-06	6.17E-06	3.44E-06	5.16E-06	9.06E-06	1.36E-05	2.89E-06	4.33E-06
Eu-154	4.04E-05	6.06E-05	3.38E-05	5.07E-05	8.91E-05	1.34E-04	2.84E-05	4.26E-05
Eu-155	2.71E-05	4.06E-05	2.26E-05	3.40E-05	5.97E-05	8.95E-05	1.90E-05	2.85E-05
Np-237	8.25E-06	1.24E-05	6.90E-06	1.04E-05	1.82E-05	2.73E-05	5.79E-06	8.69E-06
Np-239	1.10E-13	1.65E-13	9.22E-14	1.38E-13	2.43E-13	3.65E-13	7.74E-14	1.16E-13
Pa-233	5.72E-08	8.58E-08	4.78E-08	7.18E-08	1.26E-07	1.89E-07	4.02E-08	6.03E-08
Pb-210	4.94E-07	7.40E-07	4.13E-07	6.19E-07	1.09E-06	1.63E-06	3.47E-07	5.20E-07
Pb-212	8.33E-12	1.25E-11	6.96E-12	1.05E-11	1.84E-11	2.75E-11	5.85E-12	8.77E-12
Pb-214	1.13E-06	1.69E-06	9.41E-07	1.41E-06	2.48E-06	3.72E-06	7.91E-07	1.19E-06
Pm-147	1.36E-02	2.04E-02	1.14E-02	1.70E-02	2.99E-02	4.49E-02	9.54E-03	1.43E-02
Po-214	1.13E-06	1.69E-06	9.41E-07	1.41E-06	2.48E-06	3.72E-06	7.91E-07	1.19E-06
Po-218	1.13E-06	1.69E-06	9.41E-07	1.41E-06	2.48E-06	3.72E-06	7.91E-07	1.19E-06
Pu-238	7.98E+00	2.37E+00	6.67E+00	1.98E+00	1.76E+01	5.23E+00	5.60E+00	1.66E+00
Pu-239	1.91E+00	1.34E+00	1.60E+00	1.12E+00	4.22E+00	2.94E+00	1.34E+00	9.38E-01
Pu-240	4.47E-01	3.29E-01	3.74E-01	2.75E-01	9.86E-01	7.25E-01	3.14E-01	2.31E-01
Pu-241	1.24E+01	9.44E+00	1.04E+01	7.89E+00	2.74E+01	2.08E+01	8.72E+00	6.63E+00
Pu-242	3.51E-05	3.20E-05	2.94E-05	2.68E-05	7.74E-05	7.06E-05	2.47E-05	2.25E-05
Ra-225	1.28E-17	1.92E-17	1.07E-17	1.61E-17	2.82E-17	4.23E-17	8.99E-18	1.35E-17
Ra-226	1.13E-06	1.69E-06	9.41E-07	1.41E-06	2.48E-06	3.72E-06	7.91E-07	1.19E-06
Rn-222	1.13E-06	1.69E-06	9.41E-07	1.41E-06	2.48E-06	3.72E-06	7.91E-07	1.19E-06
Sr-90	5.29E-02	7.56E-02	4.42E-02	6.32E-02	1.17E-01	1.67E-01	3.72E-02	5.31E-02
Tc-99	1.74E-07	2.60E-07	1.45E-07	2.18E-07	3.83E-07	5.74E-07	1.22E-07	1.83E-07
Th-228	8.33E-12	1.25E-11	6.96E-12	1.05E-11	1.84E-11	2.75E-11	5.85E-12	8.77E-12
Th-234	2.18E-09	3.27E-09	1.83E-09	2.74E-09	4.81E-09	7.22E-09	1.53E-09	2.30E-09
U-232	7.06E-04	1.06E-03	5.90E-04	8.85E-04	1.56E-03	2.33E-03	4.96E-04	7.43E-04
U-233	2.11E-02	3.17E-02	1.77E-02	2.65E-02	4.66E-02	6.98E-02	1.48E-02	2.22E-02
U-234	1.32E-04	1.19E-04	1.10E-04	9.91E-05	2.91E-04	2.61E-04	9.26E-05	8.32E-05
U-235	2.60E-06	2.12E-06	2.17E-06	1.77E-06	5.72E-06	4.67E-06	1.82E-06	1.49E-06
U-238	2.47E-05	4.11E-06	2.07E-05	3.43E-06	5.45E-05	9.05E-06	1.74E-05	2.88E-06
Y-90	4.89E-03	3.59E-03	4.09E-03	3.00E-03	1.08E-02	7.92E-03	3.44E-03	2.52E-03

A.5 HAZARDOUS CONSTITUENTS INVENTORY

TRU mixed waste is defined as any TRU waste that is commingled with a hazardous waste regulated by RCRA (40 Code of Federal Regulations 261, Subparts C and D). There are two classes of RCRA-regulated constituents of concern relative to WIPP, metals and volatile organic compounds (VOC). For the purposes of conducting analyses that bound adverse impacts, all TRU wastes to be emplaced in WIPP were assumed to be TRU mixed waste. As compared with the approximate 60 percent of the volume of stored waste classified as TRU mixed wastes (see Table A-16). No attempt was made to forecast future operational procedures at generator-storage sites that would treat TRU and TRU mixed waste streams differently.

A.5.1 Metals Inventory

BIR-3 does not contain detailed information on hazardous constituents. An inventory of hazardous metals, however, was developed for fire and explosion accident scenarios in the SAR (DOE 1995c, Table 5.1-2). These inventory values, which correspond to a 110-kilogram (243-pound) drum, are: 0.023 kilograms (0.051 pounds) for beryllium; 3.3×10^{-4} kilograms (7.3×10^{-4} pounds) for cadmium; 0.91 kilograms (2 pounds) for lead; and 0.39 kilograms (0.86 pounds) for mercury.

Using the average material parameter data in BIR-3, a drum of CH-TRU waste is expected to contain approximately 120 kilograms (265 pounds) of waste. Similarly, a drum of RH-TRU waste is expected to contain 105 kilograms (230 pounds) of waste. CH-TRU waste metal concentrations were adjusted from the 110-kilogram drum in SAR to the 120-kilogram drum for SEIS-II. Similarly, the RH-TRU waste metal concentrations were adjusted down to the 105-kilogram drum for SEIS-II. The BIR-3 lead inventory value of 464 kilograms per cubic meter (29 pounds per cubic feet), which includes lead for shielding, was used for the RH-TRU waste concentration. Concentrations of metals used in SEIS-II analyses for both CH-TRU and RH-TRU waste are listed in Table A-44.

Total inventory values for hazardous metals analyzed in SEIS-II are given in Table A-45. These values are based on the metal concentration in Table A-44 as well as the Basic Inventory and Additional Inventory.

Table A-44
Concentration of Hazardous Metals

Metal	CH-TRU Waste Inventory (kg/cubic meter)	RH-TRU Waste Inventory (kg/cubic meter)
Beryllium	1.21E-01	1.21E-01
Cadmium	1.73E-03	1.74E-03
Lead	4.79E+00	4.64E+02
Mercury	2.05E+00	2.05E+00

Table A-45
Inventory (kilograms) of Hazardous Metals by Alternative

Alternatives	CH-TRU Waste	RH-TRU Waste	Total
Lead Inventory			
Proposed Action	8.08E+05	3.29E+06	4.09E+06
Action Alternative 1	1.31E+06	1.80E+07	1.93E+07
Action Alternative 2	1.31E+06	1.80E+07	1.93E+07
Action Alternative 3	1.31E+06	1.80E+07	1.93E+07
No Action Alternative 1	1.31E+06	1.80E+07	1.93E+07
No Action Alternative 2	6.46E+05	1.64E+07	1.70E+07
Beryllium Inventory			
Proposed Action	2.04E+04	8.58E+02	2.13E+04
Action Alternative 1	3.31E+04	4.70E+03	3.78E+04
Action Alternative 2	3.32E+04	4.70E+03	3.79E+04
Action Alternative 3	3.31E+04	4.70E+03	3.78E+04
No Action Alternative 1	3.32E+04	4.70E+03	3.79E+04
No Action Alternative 2	1.63E+04	4.27E+03	2.06E+04
Cadmium Inventory			
Proposed Action	2.93E+02	1.23E+01	3.05E+02
Action Alternative 1	4.75E+02	6.74E+01	5.42E+02
Action Alternative 2	4.76E+02	6.74E+01	5.43E+02
Action Alternative 3	4.75E+02	6.74E+01	5.42E+02
No Action Alternative 1	4.76E+02	6.74E+01	5.43E+02
No Action Alternative 2	2.34E+02	6.13E+01	2.96E+02
Mercury Inventory			
Proposed Action	3.46E+05	1.45E+04	3.61E+05
Action Alternative 1	5.61E+05	7.97E+04	6.40E+05
Action Alternative 2	5.62E+05	7.97E+04	6.42E+05
Action Alternative 3	5.61E+05	7.97E+04	6.40E+05
No Action Alternative 1	5.62E+05	7.97E+04	6.42E+05
No Action Alternative 2	2.77E+05	7.25E+04	3.49E+05

A.5.2 Inventory of Volatile Organic Compounds

Other than to indicate the presence of a relatively small volume of PCB-commingled waste, the BIR-3 database does not contain information on organic compounds. PCB-commingled waste is only considered under Action Alternative 2 and No Action Alternative 1. Further, PCB waste would be thermally treated, which would completely destroy the PCBs. Risk analyses, therefore, were not performed for PCB-commingled TRU waste.

The *Comment Responses and Revisions to the Resource Conservation and Recovery Act Part B Permit Application*, published after SAR, provided more recent headspace sampling data and was used to estimate concentrations of VOCs (DOE 1996a, Table C2-4). The permit application summarizes the results of a headspace sampling and analysis study conducted on RFETS CH-TRU waste. Approximately 930 drums of varying waste types were sampled. Average concentrations of the VOCs that present the greatest potential risk to human health are expressed as parts per million per volume in Table A-46. Where compounds were not detected in the sampling process,

Table A-46
Average Concentrations (ppmv) of Volatile Organic Compounds as Reported from the RCRA Part B Permit Application

Final TRU Waste Form	Carbon Tetrachloride	Chlorobenzene	Chloroform	Methyl Ethyl Ketone	Methylene Chloride
Combustible	566.52	1.54	41.09	7.60	12.29
Filter	1.44	0.18	0.19	5.11	0.48
Graphite	0.10	0.03	0.06	8.09	0.91
Heterogeneous	91.07	9.46	18.99	62.05	143.13
Inorganic Non-Metal	3.27	0.16	1.03	7.29	2.56
Lead/Cadmium Metal	255.28	4.94	6.86	42.56	8.61
Salt	4.32	0.18	0.23	5.50	0.56
Soils	0	0	0	0	0
Solidified Inorganics	316.51	1.29	1.15	6.83	8.05
Solidified Organics	8,319.32	94.30	135.98	717.96	214.47
Uncategorized Metal	9.58	13.40	7.94	39.34	1,941.71
Final TRU Waste Form	**1,1,2,2-Tetrachloroethane**	**Toluene**	**1,1-Dichloroethene**	**1,2-Dichloroethane**	**1,1,1-Trichloroethane**
Combustible	96.25	1.75	1.98	1.57	7.38
Filter	16.08	0.14	0.32	0.26	11.63
Graphite	8.06	0.03	0.04	0.03	0.63
Heterogeneous	711.98	7.64	14.42	7.62	24.11
Inorganic Non-Metal	29.33	0.17	1.01	0.16	4.31
Lead/Cadmium Metal	510.47	4.89	11.62	5.23	8.97
Salt	4.86	0.04	0.17	0.04	5.04
Soils	0	0	0	0	0
Solidified Inorganics	125.09	1.26	2.45	1.06	6.26
Solidified Organics	4,543.96	81.87	88.25	81.40	204.59
Uncategorized Metal	126.88	7.06	7.52	7.02	29.03

one-half of the detection limit was used for calculating average concentrations. The best available data was used to calculate concentrations of VOCs, but DOE recognizes that the data may not reflect actual usage of hazardous chemicals at all sites. Conservative assumptions, such as the assumption that all waste is mixed waste, were used to ensure the impacts stated in this document reflect a reasonable upper limit of likely impacts.

The average concentrations for the VOCs were computed for the various final TRU waste forms. SEIS-II volume data was divided into final TRU waste forms in order to compute a volume-weighted average concentration for the VOCs, according to site. The weighted average concentrations for CH-TRU waste under No Action Alternative 2 are given in Table A-47.

Concentrations of VOCs were directly used in the analyses and are given in Tables A-47 and A-48 for No Action Alternative 2. Both No Action Alternative 2 and the Proposed Action address disposal of the same waste; therefore, the same concentrations were appropriate for both. The same concentration was assumed to apply to all of the waste under Action Alternatives 1 and 3. Finally, the thermal processing assumed under Action Alternative 2 and No Action Alternative 1, removes any VOCs.

Table A-47
Volume-Weighted Average Concentrations (ppmv)
of Volatile Organic Compounds in CH-TRU Waste by Treatment Site
for Proposed Action, Action Alternative 1, Action Alternative 3, and No Action Alternative 2 [a]

Volatile Organic Compound	Hanford	LANL	INEEL	SRS	RFETS	ORNL	LLNL	NTS	Mound	ANL-E
Carbon Tetrachloride	60.20	271.20	372.68	122.89	274.93	90.19	152.06	93.09	22.44	28.93
Chlorobenzene	9.65	6.20	9.41	8.33	2.67	9.37	9.45	9.39	3.75	12.16
Chloroform	11.84	13.65	17.73	16.13	8.98	18.81	18.53	18.83	3.20	7.35
Methyl Ethyl Ketone	40.40	21.00	54.23	53.43	19.57	61.45	62.88	61.55	11.22	36.64
Methylene Chloride	964.63	753.36	480.29	157.99	91.73	141.75	133.50	141.92	532.07	1,734.17
1,1,2,2-Tetrachloroethane	309.63	128.16	452.87	613.68	141.52	705.13	692.38	706.71	41.00	128.39
Toluene	6.05	3.78	7.04	6.69	2.23	7.57	7.68	7.59	2.02	6.49
1,1-Dichloroethene	8.60	4.47	10.16	12.51	3.21	14.28	13.99	14.31	2.18	7.03
1,2-Dichloroethane	6.01	3.64	6.96	6.64	2.18	7.55	7.66	7.56	2.00	6.45
1,1,1-Trichloroethane	22.31	15.67	23.18	21.56	10.12	23.88	24.44	23.95	8.32	26.42

[a] Volatiles are assumed destroyed during thermal treatment in Action Alternative 2 and No Action Alternative 1.

Table A-48
Volume-Weighted Average Concentrations (ppmv)
of Volatile Organic Compounds in RH-TRU Waste by Treatment Site
for Proposed Action, Action Alternative 1, Action Alternative 3, and No Action Alternative 2 [a]

Volatile Organic Compound	Hanford	LANL	INEEL	ORNL
Carbon Tetrachloride	26.13	212.58	119.03	172.05
Chlorobenzene	12.61	8.97	8.80	6.53
Chloroform	10.08	20.34	17.44	12.58
Methyl Ethyl Ketone	43.76	29.16	56.88	42.19
Methylene Chloride	1,586.14	1,161.64	172.75	94.56
1,1,2,2-Tetrachloroethane	241.67	145.58	638.56	500.94
Toluene	7.17	5.20	7.05	5.35
1,1-Dichloroethene	8.87	5.89	13.07	10.12
1,2-Dichloroethane	7.13	5.11	7.02	5.26
1,1,1-Trichloroethane	28.01	21.05	22.56	17.69

[a] Volatiles are assumed destroyed during thermal treatment in Action Alternative 2 and No Action Alternative 1.

The RCRA Part B Permit Application sampling data were taken from drums containing CH-TRU waste. In the absence of further information, the same concentrations were assumed to apply to RH-TRU waste. The weighted average concentrations by site for RH-TRU waste for No Action Alternative 2 are given in Table A-48.

For analyses that require a total inventory, such as the groundwater analysis of No Action Alternative 2, headspace data were used to calculate a total inventory of organic contaminants. Rault's Law and the assumption that the maximum average quantity of organic liquid in TRU waste is 1-weight percent were used. The concentrations of volatile organic material in thermally treated waste forms were assumed to be zero, because these contaminants are removed during high-temperature processing. The calculated VOC inventories for both CH-TRU and RH-TRU waste according to site are shown in Tables A-49 and A-50.

Table A-49
Inventory (grams) of Volatile Organic Compounds in CH-TRU Waste for the Proposed Action, Action Alternative 1, Action Alternative 3, and No Action Alternative 2 [a]

Compound	Hanford	INEEL	LANL	ORNL	RFETS	SRS	LLNL
Carbon Tetrachloride	139,660	438,499	230,351	6,730	120,509	59,830	7,333
Chloroform	20,560	15,608	8,674	1,050	2,946	5,878	669
1,1-Dichloroethylene	3,759	2,253	715	201	265	1,147	127
Methylene Chloride	698,580	176,405	199,743	3,302	12,551	24,010	2,010
1,1,2,2-Tetrachloroethane	16,234,213	12,042,561	2,460,196	1,189,188	1,401,930	6,752,290	754,557
Chlorobenzene	216,336	106,937	50,878	6,756	11,324	39,169	4,401
Methyl Ethyl Ketone	136,670	93,043	26,013	6,686	12,509	37,925	4,421
Toluene	56,419	33,311	12,894	2,272	3,939	13,090	1,489
1,2-Dichloroethane	24,633	14,463	5,461	994	1,691	5,705	652
1,1,1-Trichloroethane	47,554	25,053	12,230	1,637	4,074	9,645	1,083

[a] Volatiles are assumed destroyed during thermal treatment in Action Alternative 2 and No Action Alternative 1.

Table A-50
Inventory (grams) of Volatile Organic Compounds in RH-TRU Waste for the Proposed Action, Action Alternative 1, Action Alternative 3, and No Action Alternative 2 [a]

Compound	Hanford	INEEL	LANL	ORNL
Carbon Tetrachloride	31,024	9,397	1,959	25,450
Chloroform	8,958	1,031	140	1,392
1,1-Dichloroethylene	1,983	194	10	282
Methylene Chloride	587,915	4,257	3,342	4,366
1,1,2,2-Tetrachloroethane	6,485,477	1,139,327	30,327	1,674,680
Chlorobenzene	144,586	6,712	798	9,323
Methyl Ethyl Ketone	75,757	6,547	392	9,100
Toluene	34,223	2,238	193	3,181
1,2-Dichloroethane	14,949	978	83	1,374
1,1,1-Trichloroethane	30,552	1,636	178	2,405

[a] Volatiles are assumed destroyed during thermal treatment in Action Alternative 2 and No Action Alternative 1.

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APPENDIX B
SUMMARY OF THE WASTE MANAGEMENT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT AND ITS USE IN DETERMINING HUMAN HEALTH IMPACTS AT TREATMENT SITES

The *Final Waste Management Programmatic Environmental Impact Statement* (WM PEIS) (DOE 1997) is a nationwide study examining the environmental impacts of managing five types of radioactive and hazardous wastes that result primarily from nuclear defense activities – the development, production, and testing of nuclear weapons at a variety of sites located around the United States. The five waste types are the following: low-level mixed waste, low-level waste, transuranic (TRU) waste, high-level waste, and hazardous waste.

For each waste-type system, facilities are needed to treat, store, and/or dispose of the waste. In the WM PEIS, the Department of Energy (DOE or the Department) has not only examined, in an integrated fashion, the impacts of complex-wide waste management for each waste type but also the specific cumulative impacts for all the waste facilities at a given site. The WM PEIS provides information on the impacts of various siting alternatives, which DOE will use in deciding where to locate additional treatment, storage, and/or disposal capacity for each waste type. However, the location of a facility at a selected site will not be decided until completion of a subsequent sitewide or project-specific National Environmental Policy Act (NEPA) review.

B.1 RELATIONSHIP OF DIFFERENT LEVELS OF NEPA DOCUMENTS

In accordance with DOE NEPA regulations, three types of NEPA documentation may be prepared: programmatic, sitewide, and project-level. Programmatic documents, such as the WM PEIS, provide environmental input into decisions on broad agency actions, such as the adoption of new plans, programs, and policies to guide future actions. Sitewide NEPA documents, such as this document (SEIS-II), provide the opportunity for considering changes in the overall operating mode of a DOE site, including mission change, and provide a current environmental baseline at the site. Project-level NEPA documents evaluate the impacts of a specific project at a specific location on a site and are intended to provide environmental input into the manner in which the facility should be constructed and operated. Sitewide NEPA documents, which evaluate projects that could be implemented in the near-term at a site, may also serve as project-level NEPA documents for specified projects.

B.2 WASTE MANAGEMENT DECISIONS TO BE MADE BY DOE

The WM PEIS is intended to provide environmental information to assist DOE in determining where to consolidate waste and where it should modify existing waste management facilities or construct new facilities. The TRU waste management facilities proposed in the WM PEIS are treatment and storage facilities. DOE needs to identify sites for waste management facilities in order to protect public health and safety, comply with federal law, and minimize adverse effects to the environment. If sites are selected for TRU waste treatment and storage facilities, DOE intends to select the sites using the WM PEIS analysis but will not select the level of treatment needed. Treatment level decisions will be made using SEIS-II analyses. Specific locations for the waste management facilities within a site will be selected on the basis of subsequent sitewide or project-level NEPA documents.

B.3 OVERVIEW OF THE WM PEIS TRU WASTE ANALYSES

SEIS-II refers to relevant information, primarily concerning treatment sites, from several documents, including the *Title 40 CFR Part 191 Compliance Certification Application for the Waste Isolation Pilot Plant* (DOE 1996b), the *Resource Conservation and Recovery Act (RCRA) Part B Application* (DOE 1996a) and the WM PEIS (DOE 1997); SEIS-II updates and scales information with more recent information from these other documents. The following sections present an overview of information in the WM PEIS that is relevant to SEIS-II.

B.3.1 TRU Waste

TRU waste analyzed in the WM PEIS considers both contact-handled (CH) TRU and remote-handled (RH) TRU waste placed in retrievable storage across the DOE complex since 1970 and projected to be generated for 20 years. For the purposes of WM PEIS analyses, DOE included the small amount of nondefense TRU waste.

In addition, approximately 60 percent of the TRU waste also contains hazardous constituents as defined by the Resource Conservation and Recovery Act (RCRA); this waste is called TRU mixed waste. For purposes of the WM PEIS analyses, DOE assumed that the entire inventory of TRU waste was TRU mixed waste.

Management activities associated with TRU waste that are discussed in the WM PEIS include (1) retrieving TRU waste from storage and transporting it to a treatment facility; (2) sorting and treating the TRU waste as appropriate, packaging the waste, and certifying the waste for shipment to the Waste Isolation Pilot Plant (WIPP) for disposal; (3) storing certified waste; and (4) transporting the TRU waste to WIPP for disposal. For all of its alternatives except its no action alternative, the WM PEIS assumed that TRU waste would be disposed of at WIPP.

B.3.2 TRU Waste Generator Sites and Inventories

Sixteen sites are identified in the WM PEIS that have or are expected to generate or manage TRU waste.

Major sites identified in the WM PEIS include the following:

- Argonne National Laboratory-East (ANL-E) near Chicago, Illinois
- Hanford Site (Hanford) at Richland, Washington
- Idaho National Engineering and Environmental Laboratory (INEEL) near Idaho Falls, Idaho
- Lawrence Livermore National Laboratory (LLNL) near San Francisco, California
- Los Alamos National Laboratory (LANL) at Los Alamos, New Mexico

- Mound Plant (Mound) at Miamisburg, Ohio
- Nevada Test Site (NTS) near Las Vegas, Nevada
- Oak Ridge Reservation (ORR) at Oak Ridge, Tennessee (identified as Oak Ridge National Laboratory [ORNL] in SEIS-II)
- Rocky Flats Environmental Technology Site (RFETS) near Golden, Colorado
- Savannah River Site (SRS) at Aiken, South Carolina.

Identified as smaller generators are the following:

- Energy Technology Engineering Center (ETEC) at Canoga Park, California
- Lawrence Berkeley Laboratory (LBL) at Berkeley, California
- Paducah Gaseous Diffusion Plant (PGDP) at Paducah, Kentucky
- Sandia National Laboratories (SNL) at Albuquerque, New Mexico
- University of Missouri at Columbia (U of Mo), Missouri
- West Valley Demonstration Project (WVDP) at West Valley, New York. The small amount of waste from this site originated from commercial reprocessing of spent nuclear fuel and so is not defense related.

The WM PEIS analyzes the potential environmental impacts for managing approximately 67,000 cubic meters (2.4 million cubic feet) of retrievably stored CH-TRU waste and about 1,700 cubic meters (60,000 cubic feet) of retrievably stored RH-TRU waste. Approximately 95 percent of the existing CH-TRU waste and RH-TRU waste is stored at Hanford, INEEL, LANL, ORR, RFETS, and SRS.

An additional 47,000 cubic meters (1.7 million cubic feet) of CH-TRU waste and 17,000 cubic meters (600,000 cubic feet) of RH-TRU waste was assumed to be generated over the next 20 years (excluding TRU waste that would result from environmental restoration activities), for a total of about 132,000 cubic meters (4.7 million cubic feet) of retrievably stored TRU waste. The inventory and annual generator rates for the WM PEIS were obtained from the *Interim Mixed Waste Inventory Report* (DOE 1993) and the *Integrated Data Base for 1992* (DOE 1992). Updated information on waste volumes was used for Hanford and SRS. Updated data for TRU waste were taken from two sources: the *Mixed Waste Inventory Summary Report* (MWIR 95) (DOE 1995a) and the *Transuranic Waste Baseline Inventory Report, Revision 2* (BIR-2) (DOE 1995b), with most of the new information taken from MWIR 95. Table B-1 presents the waste volumes as used in WM PEIS risk calculations; this table is the same as WM PEIS Table 8.1-1. SEIS-II analyses differ slightly from the WM PEIS regarding TRU waste volumes, years of generation, and the number of sites producing waste as discussed in Appendix A. These changes presented in SEIS-II were necessary to evaluate the most recent information and Departmental planning assumptions that were available for analysis. For example, SEIS-II waste volumes include environmental restoration wastes in the Additional Inventory; SEIS-II also assumes a 35-year operations period; and the number of TRU waste sites was expanded to include smaller sites captured in the updated inventory.

Table B-1
Transuranic Waste Volumes of the WM PEIS (cubic meters)

	CH-TRU Waste			RH-TRU Waste			
Site [a]	**Inventory**	**20-Year Projected Generation**	**Estimated Inventory + 20 Year Generation**	**Inventory**	**20-Year Projected Generation**	**Estimated Inventory + 20 Year Generation**	**Total**
ANL-E	15	940	960	---	340	340	1,300
ETEC	0.02	---	0.02	---	---	---	0.02
Hanford	12,000	24,000	36,000	200	15,400	16,000	52,000
INEEL	38,000	280	38,000	110	500	610	39,000
LANL	8,200	2,500	11,000	79	10	89	11,000
LBL	0.8	0.2	1	---	---	---	1
LLNL	200	1,500	1,700	---	---	---	1,700
Mound	274	1,200	1,500	---	---	---	1,500
NTS	610	---	610	---	---	---	610
ORR	670	360	1,000	1,300	360	1,700	2,700
PGDP	14	---	14	---	---	---	14
RFETS	1,500	4,800	6,200	---	---	---	6,200
SNL	1	---	1	---	---	---	1
SRS	5,100	11,500	16,600	---	---	---	16,600
U of Mo	---	2	2	---	---	---	2
WVDP	0.5	---	0.5	---	---	---	0.5
Total	**67,000**	**47,000**	**114,000**	**1,700**	**17,000**	**18,000**	**132,000**

[a] WIPP, the seventeenth site, does not currently have any TRU waste.

Note: Volume data are rounded from field estimates and columns and rows do not add. Waste volume projections contained in this and other WM PEIS tables were based on 1993 or earlier data and may vary from the latest site estimates at the time of publication.

Source: WM PEIS, Table 8.1-1

B.3.3 Waste Treatment

There are three alternative waste treatments considered in the WM PEIS: treatment to the Waste Acceptance Criteria (WAC); shredding and using grout; and treatment to the RCRA Land Disposal Restrictions (LDR). Compliance with WAC is the minimum level of treatment required. The shred and grout treatment would be used to further stabilize the waste and reduce the rate of potential gas generation. Treatment to meet LDRs would further stabilize and consolidate waste and destroy volatile organic compounds (VOCs) in the waste. For more information on these treatment technologies, see Chapter 2.

B.3.4 Alternatives

As stated above, the WM PEIS was prepared to support decisions on where to treat and store TRU waste. To assist DOE in making decisions regarding the sites at which it should locate waste management facilities, the WM PEIS considers four categories of alternatives for each waste type: the no action alternative, decentralized alternatives that would minimize the transportation of waste between sites, regionalized alternatives that would locate waste management facilities at several sites throughout the nation, and a centralized alternative that would locate large waste management facilities at only one site for CH-TRU waste and two sites for RH-TRU waste. For TRU waste, DOE considers more than one regionalized alternative in order to vary the number of sites having waste management facilities and the sites at which the facilities could be located. This variation among alternatives allows flexibility when considering the future configuration of waste management facilities. These TRU waste alternatives are summarized in the following subsections. All WM PEIS action alternatives discussed below assume that the waste would be shipped to WIPP for disposal.

B.3.4.1 Decentralized Alternative

Under the WM PEIS Decentralized Alternative, DOE would, as needed, treat and package TRU waste to meet WAC. The treatment and packaging would occur at all sites. After treatment, CH-TRU waste would be shipped to the nearest one of the 10 sites with the larger amount of TRU waste for storage prior to disposal in WIPP.

B.3.4.2 Regionalized Alternatives

The WM PEIS regionalized alternatives would consolidate TRU waste for treatment and storage prior to disposal. Three TRU waste regionalized alternatives are analyzed, with varying degrees of treatment at six and four sites, and storage at those sites prior to disposal in WIPP.

Regionalized 1

Under the WM PEIS Regionalized 1 Alternative, CH-TRU waste would be shipped from the 10 smallest generators to the four sites with the largest volumes of TRU waste (Hanford, INEEL, LANL, and SRS). In addition, RFETS would continue to treat its own waste, but would not receive waste from off site. RH-TRU waste would be shipped from ANL-E, INEEL, and LANL to Hanford or ORR for treatment. At all six treatment sites, TRU waste would be treated using a shred and grout process (referred to in the WM PEIS as the “reduce gas generation potential”). The six treatment sites proposed under this alternative have 95 percent of current and anticipated TRU waste inventories.

Regionalized 2

Under the WM PEIS Regionalized 2 Alternative, DOE would use the same waste consolidation configuration as in Regionalized 1, except that the TRU waste would be treated to meet the LDRs.

Regionalized 3

Under the WM PEIS Regionalized 3 Alternative, the consolidation of waste for treatment at four sites (Hanford, INEEL, ORR, and SRS) where approximately 80 percent of TRU waste is already located or is expected to be generated is considered. CH-TRU waste would be treated at

Hanford, INEEL, and SRS; RH-TRU waste would be treated at Hanford and ORR. Under this alternative, TRU waste would be treated to meet the LDRs.

B.3.4.3 Centralized Alternative

Under the WM PEIS Centralized Alternative, DOE would ship all CH-TRU waste to WIPP for treatment to meet the LDRs and for disposal. RH-TRU waste would be shipped to Hanford and ORR for treatment to meet the LDRs and eventually disposed of in WIPP.

B.3.4.4 No Action Alternative

Under the WM PEIS No Action Alternative, DOE would continue to characterize, process, and package newly generated TRU waste based on the current WAC for storage at sites where existing or planned facilities are available. DOE would continue to store TRU waste in existing storage facilities for the duration of this analysis (20 years) and would not ship TRU waste for off-site storage; there would be no disposal. All sites are assumed to have adequate capabilities to package and store TRU waste generated in the future. Eleven sites have projected future TRU waste generation, including five sites generating both CH-TRU and RH-TRU waste. The WM PEIS No Action Alternative does not assess the health risks, environmental impacts, or costs of removing TRU waste from retrievable storage and packaging it.

B.4 INCORPORATION OF WM PEIS INTO SEIS-II ANALYSES

WM PEIS analyses form the basis of the SEIS-II analysis of generator site impacts. These impacts, adjusted for different inventories and other analysis assumptions and combined with the SEIS-II analyses of impacts from waste disposal at WIPP and lag storage at the generator sites, present a comprehensive picture of the potential human health impacts complex-wide from management, treatment, and disposal of TRU waste.

The WM PEIS examines potential impacts of management and treatment of the various waste types. Impact areas evaluated in the WM PEIS for all of the waste types include human health risks, air quality, water resources, ecological resources, socioeconomics, land use, environmental justice, infrastructure, cultural resources, and cost.

The relevant portions of the WM PEIS have been summarized and incorporated in SEIS-II. Where appropriate, the WM PEIS impacts have been adjusted to reflect recent information such as revised estimates of future waste generation, cumulative impacts, and potential future activities at the sites. Life-cycle costs and transportation analyses have been reexamined and revised with the results presented in Chapter 5 and methods presented in Appendices D and E, respectively. Human health impacts from the WM PEIS have also been adjusted to reflect waste inventory differences and other factors considered under the SEIS-II alternatives.

For routine operations involving treatment, health impacts in the WM PEIS are evaluated for the off-site population, the on-site worker population not involved in treatment, and waste management workers directly involved in treatment activities. Impacts are quantified using two approaches: analysis of population health risk impacts and analysis of individual health risk impacts. Population impacts focus on the total number of people in each population who may experience adverse health impacts if a particular alternative were implemented.

B.5 USING HUMAN HEALTH IMPACTS AT TREATMENT SITES FROM THE WM PEIS

SEIS-II focuses on impacts from disposal of TRU waste. However, human health impacts from management and treatment of TRU waste at the generator sites, addressed in the WM PEIS, may be a major contributor to the overall risk of disposing of TRU waste and preparing it for disposal.

Overall, in the WM PEIS the numerically largest health risks result from alternatives where TRU waste is treated to meet the LDRs (the WM PEIS Regionalized 2, Regionalized 3, and Centralized alternatives). These alternatives assume the use of thermal destruction of organic waste to meet the LDRs. This treatment method results in emissions of radionuclides that result in additional off-site cancer risks; the maximally exposed individuals (MEI) are at LANL, INEEL, and WIPP. Although postulated waste management worker fatalities primarily result from physical hazards, fatalities are lower when TRU waste is treated to planning-basis WAC or by a shred and grout process than when TRU waste is treated to meet the LDRs.

Because of differences between the WM PEIS and SEIS-II, it was necessary to adjust the impacts from the WM PEIS before they could be used in SEIS-II. SEIS-II analyses use different TRU waste volumes and radionuclide inventories than those in the WM PEIS, and include environmental restoration wastes in the Additional Inventory. SEIS-II alternatives also differ from the WM PEIS by having more years of waste generation and site operation, more sites producing waste, and, in some cases, the waste inventory and the manner of waste consolidation. Human health impacts adjusted from the WM PEIS are those occurring as a result of routine waste treatment and management operations and do not include accidents involving workers or members of the public. Therefore, only those impacts resulting from routine releases and exposure to radioactive material and hazardous chemicals, resulting in potential latent fatal cancers (LCFs) or cancer incidence, respectively, were adjusted.

Radiation-related human health impacts for members of the public and noninvolved workers at the treatment/generator sites were adjusted based on differences in (1) waste volumes treated at the major treatment sites and (2) site-specific concentrations of key radionuclides. These adjustments resulted in a volume ratio (VR) and a concentration ratio (CR), respectively. Radiation-related human health impacts for involved workers and all estimates of cancer incidence from exposure to hazardous chemicals were adjusted only on differences in the waste volumes handled and treated. Key radionuclide and exposure pathway information for involved workers and quantitative hazardous chemical information were not available in the WM PEIS.

Equation B-1 was used to calculate the adjusted radiation dose and LCFs from waste treatment to the off-site population, the MEI, the noninvolved worker population, and the maximally exposed noninvolved worker for the SEIS-II.

SEIS-II treatment impact = VR x CR x (WM PEIS treatment impact) (Equation B-1)

Equation B-2 was used to calculate the adjusted radiation dose and LCFs for the involved worker population and the hazardous chemical cancer incidence for all populations and individuals.

SEIS-II treatment impact = VR x (WM PEIS treatment impact) (Equation B-2) where

VR = SEIS-II/WM PEIS waste volume ratio

and CR = SEIS-II/WM PEIS key radionuclide concentration ratio

with both VR and CR varied for each treatment site, alternative, inventory type (Total, Additional, or Basic) and waste type (CH-TRU or RH-TRU). Table B-2 presents the VR and key radionuclide CR for each treatment site, alternative, inventory type (Total, Basic, and Additional) and waste type (CH-TRU and RH-TRU).

Waste volume information for SEIS-II was taken from the "Pre-Treatment Consolidated Volume" columns of Tables A-5 through A-14 of Appendix A. The WM PEIS TRU waste volumes used were those presented in Table B-1 and in Table 8.1-1 of the WM PEIS. The volume ratio calculations were done for each of three SEIS-II inventories: the Basic Inventory, the Additional Inventory, and the Total Inventory except for the Proposed Action and No Action Alternative 2 for which the Basic and Total Inventory are the same (for more information on these inventories, see Chapters 2 and 3 and Appendix A).

Key radionuclides are those defined in Appendix D of the WM PEIS as the single radionuclide contributing the highest risk of latent cancer fatality at each site under each alternative. Key radionuclides contributing the highest risk to off-site populations are listed in WM PEIS Table D.3.4-18 for CH-TRU waste and in WM PEIS Table D.3.4-34 for RH-TRU waste. These key radionuclide concentrations were also used to adjust impacts to the MEI, noninvolved worker population and the maximally exposed noninvolved worker.

Once these key radionuclides were identified, radionuclide concentrations for the SEIS-II alternatives were calculated using the radionuclide inventory and volume data shown in Appendix A. Concentrations were determined by dividing the total activity per year of a particular radionuclide by the total annual volume in cubic meters per year.

The concentrations of the WM PEIS key radionuclides at the various sites were taken from the tables in Appendix B of *Transuranic Waste Inventory, Characteristics, Generation, and Facility Assessment for Treatment, Storage, and Disposal Alternatives Considered in the U.S. Department of Energy Waste Management Programmatic Environmental Impact Statement* (ANL 1995). The tables used for WM PEIS CH-TRU waste radionuclide concentrations were as follows (for an explanation of the SEIS-II alternatives, see Chapter 3 of this document):

Table B-2 was used for the Proposed Action, Action Alternative 1, and No Action Alternative 2

- Table B-3 was used for Action Alternative 3
- Table B-4 was used for Action Alternative 2A and No Action Alternative 1A
- Table B-5 was used for Action Alternative 2B and No Action Alternative 1B
- Table B-6 was used for Action Alternative 2C

Table B-2
Key Radionuclide Concentration and Volume Adjustment Factors for CH-TRU and RH-TRU Waste

Site	Key Radionuclide [a]	SEIS/WM PEIS Concentration Ratio	Volume Ratio Total	Volume Ratio Basic	Volume Ratio Additional
Proposed Action (Decentralized) [b]					
CH-TRU Waste					
ANL-E	Plutonium-239	0.47	0.21	Same as Total	N/A
Hanford	Plutonium-238	0.26	1.60		N/A
INEEL	Americium-241	1.86	0.77		N/A
LANL	Americium-241	0.07	1.91		N/A
LLNL	Plutonium-239	0.49	0.70		N/A
NTS	Plutonium-239	40.67	1.04		N/A
RFETS	Americium-241	6.72	1.75		N/A
SRS	Plutonium-238	0.25	0.73		N/A
RH-TRU Waste					
Hanford	Plutonium-239	0.14	1.89	Same as Total	N/A
INEEL	Plutonium-241	0.04	3.21		N/A
LANL	Plutonium-239	0.56	2.57		N/A
ORNL	Curium-244	0.42	2.16		N/A
Action Alternative 1 (Decentralized) [b]					
CH-TRU Waste					
ANL-E	Plutonium-239	0.40	0.21	0.21	-- [c]
Hanford	Plutonium-238	0.22	3.34	1.60	1.74
INEEL	Americium-241	1.59	2.26	0.77	1.49
LANL	Americium-241	0.06	3.18	1.91	1.26
LLNL	Plutonium-239	0.42	0.70	0.70	-- [c]
NTS	Plutonium-239	34.41	1.04	1.04	-- [c]
RFETS	Americium-241	6.17	1.75	1.75	-- [c]
SRS	Plutonium-238	0.19	1.02	0.73	0.29
RH-TRU Waste					
Hanford	Plutonium-239	0.22	1.95	1.89	0.07
INEEL	Plutonium-241	0.04	3.93	3.21	0.73
LANL	Plutonium-239	0.56	3.90	2.57	1.33
ORNL	Curium-244	0.42	3.29	2.16	1.13
Action Alternative 2A and No Action Alternative 1A (Regionalized 2) [b]					
CH-TRU Waste					
Hanford	Plutonium-238	0.24	3.22	1.56	1.67
INEEL	Americium-241	1.62	2.25	0.77	1.48
LANL	Americium-241	0.06	3.18	1.91	1.26
RFETS	Americium-241	10.67	1.75	1.75	-- [c]
SRS	Plutonium-238	3.85	0.98	0.72	0.26
RH-TRU Waste					
Hanford	Plutonium-239	1.54	2.04	1.94	0.10
ORNL	Curium-244	0.16	2.71	1.78	0.93

[a] WM PEIS key radionuclides are found in Table D.3.4-18 for CH-TRU waste and in Table D.3.4-34 for RH-TRU waste.
[b] The WM PEIS alternative is shown in parenthesis.
[c] No waste in this inventory.

N/A = Not Applicable

Table B-2
Key Radionuclide Concentration and Volume
Adjustment Factors for CH-TRU and RH-TRU Waste — Continued

Site	Key Radionuclide [a]	SEIS/WM PEIS Concentration Ratio	Volume Ratio		
Action Alternative 2B and No Action Alternative 1B (Regionalized 3) [b]					
CH-TRU Waste					
Hanford	Plutonium-238	4.14	**Total**	**Basic**	**Additional**
INEEL	Americium-241	1.29	2.38	1.11	1.27
SRS	Plutonium-238	3.85	0.98	0.72	0.26
RH-TRU Waste					
Hanford	Plutonium-239	1.54	2.04	1.94	0.10
ORNL	Curium-244	0.16	2.71	1.78	0.93
Action Alternative 2C (Centralized) [b]					
CH-TRU Waste					
WIPP	Plutonium-238	1.43	2.41	1.19	1.22
RH-TRU Waste					
Hanford	Plutonium-239	1.54	2.04	1.94	0.10
ORNL	Curium-244	0.16	2.71	1.78	0.93
Action Alternative 3 (Regionalized 1) [b]					
CH-TRU Waste					
Hanford	Plutonium-238	4.14	3.22	1.56	1.66
INEEL	Americium-241	6.16	2.24	0.77	1.46
LANL	Americium-241	16.35	3.18	1.91	1.26
RFETS	Americium-241	0.09	1.75	1.75	
SRS	Plutonium-238	3.85	0.97	0.72	0.26
RH-TRU Waste					
Hanford	Plutonium-239	0.22	2.04	1.94	0.10
ORNL	Curium-244	0.16	2.71	1.78	0.93
No Action Alternative 2 (Decentralized) [b]					
CH-TRU Waste					
ANL-E	Plutonium-239	2.50	0.19	Same as Total	N/A
Hanford	Plutonium-238	4.47	1.26		N/A
INEEL	Americium-241	0.63	0.03		N/A
LANL	Americium-241	16.35	0.91		N/A
LLNL	Plutonium-239	2.39	0.56		N/A
NTS	Plutonium-239	0.03	0.02		N/A
RFETS	Americium-241	0.16	0.96		N/A
SRS	Plutonium-238	4.66	0.55		N/A
RH-TRU Waste					
Hanford	Plutonium-239	0.22	1.89	Same as Total	N/A
INEEL	Plutonium-241	0.01	2.57		N/A
LANL	Plutonium-239	0.89	3.21		N/A
ORNL	Curium-244	0.19	2.16		N/A

[a] WM PEIS key radionuclides are found in Table D.3.4-18 for CH-TRU waste and in Table D.3.4-34 for RH-TRU waste.
[b] The WM PEIS alternative is shown in parenthesis.
[c] No waste in this inventory.

N/A = Not Applicable

The tables used for WM PEIS RH-TRU waste radionuclide concentrations were as follows:

- Table B-8 was used for the Proposed Action, Action Alternative 1, and No Action Alternative 2
- Table B-9 was used for Action Alternative 3
- Table B-10 was used for Action Alternatives 2A, 2B, and 2C and No Action Alternatives 1A and 1B.

The WM PEIS presents only the total site-specific impacts (Volume II; impacts not broken out by CH-TRU or RH-TRU waste) and the total programmatic impact from CH-TRU and RH-TRU waste (Appendix D of the WM PEIS). For most sites, this does not present a problem because most are principally either a CH-TRU or a RH-TRU waste site. For these sites, the impact from CH-TRU and RH-TRU waste was apportioned by the relative volumes of CH-TRU and RH-TRU waste treated at the site. At Hanford, impacts to the offsite population, the MEI, the noninvolved worker population, and the maximally exposed noninvolved worker were apportioned by the differences between the Regionalized 2 and Regionalized 3 alternatives (which have identical human health impacts at Hanford, where both CH-TRU and RH-TRU waste are treated) and the Centralized alternative, where only RH-TRU waste is treated at Hanford. However, under the SEIS-II alternatives, the CH-TRU and RH-TRU waste volumes at Hanford are very similar, and impacts to the involved worker population would be expected to be significantly higher from handling CH-TRU waste than from handling the same volume of RH-TRU waste. Therefore, impacts to the Hanford involved worker population from CH-TRU and RH-TRU waste were apportioned using the ratio of CH-TRU and RH-TRU waste programmatic impacts for each WM PEIS alternative, shown in Appendix D of the WM PEIS (Tables D.3.4-3 and D.3.4-23). Calculated impacts from hazardous chemicals are generally higher from RH-TRU waste than from CH-TRU waste, while radiological impacts are higher from CH-TRU waste than from RH-TRU waste.

No impacts are expected to any of the analyzed groups from exposure to hazardous chemicals; and there is no expectation of LCFs in the MEI, noninvolved worker population, or the noninvolved worker MEI. For RH-TRU waste treatment, there is no expectation of cancer incidence or LCF from exposure to hazardous chemicals or radionuclides. SEIS-II estimates of waste treatment impacts adjusted from WM PEIS human health impacts are principally noted for the off-site populations and for waste treatment worker populations. The adjusted human health impacts from DOE site treatment of CH-TRU and RH-TRU waste are presented by site for each of the SEIS-II alternatives, Total, Basic, and Additional inventories, in Tables B-3 through B-19. Radiation-related LCFs may be expected in the population under Action Alternatives 2A, 2B, and 2C and No Action Alternatives 1A and 1B. There is a calculated expectation of up to 2.4 LCFs for the Total Inventory under Action Alternative 2A and No Action Alternative 1A. Up to 2.3 LCFs may be expected under Action Alternative 2B and No Action Alternative 1B, and about 1 LCF (0.9) may be expected under Action Alternative 2C.

Table B-3
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals for the Proposed Action Total (Basic) Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker				Maximally Exposed	Noninvolved Worker				Maximally Exposed	Noninvolved Worker		
	Population	Individual	Population	Individual	Worker Population	Population	Individual	Population	Individual	Worker Population	Population	Individual	Population	Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	1.9E-02	4.1E-07	9.7E-04	1.6E-06	4.9E+02	4.7E-05	9.9E-10	2.4E-06	4.0E-09	9.0E-01	1.9E-02	4.1E-07	9.7E-04	1.6E-06	5.0E+02
LANL	1.5E-02	1.5E-06	1.3E-03	1.0E-06	6.8E+02	1.3E-03	1.3E-07	1.1E-04	8.5E-08	7.4E+00	1.6E-02	1.6E-06	1.5E-03	1.1E-06	6.9E+02
INEEL	3.2E-03	3.9E-07	9.5E-04	8.1E-07	4.8E+02	4.3E-06	5.2E-10	1.3E-06	1.1E-09	3.2E+01	3.2E-03	3.9E-07	9.6E-04	8.2E-07	5.1E+02
SRS	2.8E-02	2.6E-07	2.9E-03	2.6E-06	1.2E+02	0	0	0	0	0	2.8E-02	2.6E-07	2.9E-03	2.6E-06	1.2E+02
RFETS	2.2E-01	2.9E-06	1.1E-02	6.7E-06	3.3E+01	0	0	0	0	0	2.2E-01	2.9E-06	1.1E-02	6.7E-06	3.3E+01
ORNL	0	0	0	0	0	1.4E-03	4.4E-08	4.8E-05	4.4E-08	1.5E+01	1.4E-03	4.4E-08	4.8E-05	4.4E-08	1.5E+01
LLNL	2.4E-03	3.8E-08	1.2E-04	4.2E-08	9.8E-01	0	0	0	0	0	2.4E-03	3.8E-08	1.2E-04	4.2E-08	9.8E-01
NTS	9.7E-06	2.5E-09	2.7E-05	1.2E-07	5.4E-01	0	0	0	0	0	9.7E-06	2.5E-09	2.7E-05	1.2E-07	5.4E-01
ANL-E	4.0E-04	2.2E-09	2.1E-06	2.1E-09	4.6E+00	0	0	0	0	0	4.0E-04	2.2E-09	2.1E-06	2.1E-09	4.6E+00
Total	2.9E-01		1.7E-02		1.8E+03	2.7E-03		1.6E-04		5.5E+01	2.9E-01		1.8E-02		1.9E+03
Radiation-Related LCFs															
Hanford	9.7E-06	2.0E-10	3.9E-07	6.6E-10	2.0E-01	2.4E-08	5.0E-13	9.4E-10	1.6E-12	3.6E-04	9.7E-06	2.0E-10	3.9E-07	6.6E-10	2.0E-01
LANL	7.5E-06	7.5E-10	5.4E-07	4.1E-10	2.7E-01	6.3E-07	6.3E-11	4.5E-08	3.4E-11	3.0E-03	8.2E-06	8.2E-10	5.8E-07	4.4E-10	2.8E-01
INEEL	1.6E-06	2.0E-10	3.8E-07	3.3E-10	1.9E-01	2.1E-09	2.6E-13	5.1E-10	4.3E-13	1.3E-02	1.6E-06	2.0E-10	3.8E-07	3.3E-10	2.0E-01
SRS	1.4E-05	1.3E-10	1.2E-06	1.0E-09	4.9E-02	0	0	0	0	0	1.4E-05	1.3E-10	1.2E-06	1.0E-09	4.9E-02
RFETS	1.1E-04	1.5E-09	4.4E-06	2.7E-09	1.3E-02	0	0	0	0	0	1.1E-04	1.5E-09	4.4E-06	2.7E-09	1.3E-02
ORNL	0	0	0	0	0	7.2E-07	2.2E-11	1.9E-08	1.8E-11	5.9E-03	7.2E-07	2.2E-11	1.9E-08	1.8E-11	5.9E-03
LLNL	1.2E-06	1.9E-11	4.7E-08	1.7E-11	3.9E-04	0	0	0	0	0	1.2E-06	1.9E-11	4.7E-08	1.7E-11	3.9E-04
NTS	4.9E-09	1.2E-12	1.1E-08	4.9E-11	2.2E-04	0	0	0	0	0	4.9E-09	1.2E-12	1.1E-08	4.9E-11	2.2E-04
ANL-E	2.0E-07	1.1E-12	8.3E-10	8.3E-13	1.8E-03	0	0	0	0	0	2.0E-07	1.1E-12	8.3E-10	8.3E-13	1.8E-03
Total	1.5E-04		7.0E-06		7.3E-01	1.4E-06		6.6E-08		2.2E-02	1.5E-04		7.0E-06		7.5E-01
Chemicals - Cancer Incidence															
Hanford	1.3E-10	0	8.0E-11	1.6E-13	3.5E-08	6.3E-13	0	3.8E-13	7.4E-16	6.4E-11	1.3E-10	0	8.0E-11	1.6E-13	3.5E-08
LANL	1.3E-09	1.6E-13	6.5E-10	4.0E-13	6.6E-08	1.4E-11	1.7E-15	7.0E-12	4.3E-15	7.2E-10	1.3E-09	1.6E-13	6.5E-10	4.0E-13	6.7E-08
INEEL	1.7E-09	2.4E-13	2.3E-09	1.9E-12	6.9E-06	1.2E-10	1.6E-14	1.5E-10	1.3E-13	4.7E-07	1.9E-09	2.6E-13	2.4E-09	2.0E-12	7.4E-06
SRS	1.4E-11	0	6.9E-12	8.0E-15	3.1E-09	0	0	0	0	0	1.4E-11	0	6.9E-12	8.0E-15	3.1E-09
RFETS	8.4E-10	0	1.7E-10	8.6E-14	1.2E-08	0	0	0	0	0	8.4E-10	0	1.7E-10	8.6E-14	1.2E-08
ORNL	0	0	0	0	0	3.5E-07	1.7E-11	1.1E-07	1.1E-10	6.5E-06	3.5E-07	1.7E-11	1.1E-07	1.1E-10	6.5E-06
LLNL	9.1E-08	2.0E-12	2.1E-08	6.3E-12	6.2E-07	0	0	0	0	0	9.1E-08	2.0E-12	2.1E-08	6.3E-12	6.2E-07
NTS	3.9E-13	0	5.3E-12	2.2E-14	9.1E-10	0	0	0	0	0	3.9E-13	0	5.3E-12	2.2E-14	9.1E-10
ANL-E	3.1E-10	0	6.5E-12	5.9E-15	4.0E-10	0	0	0	0	0	3.1E-10	0	6.5E-12	5.9E-15	4.0E-10
Total	9.6E-08		2.4E-08		7.7E-06	3.5E-07		1.1E-07		6.9E-06	4.4E-07		1.4E-07		1.5E-05

Table B-4
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 1 Total Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker
	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population
Radiation Dose (in rem or person-rem)															
Hanford	3.4E-02	7.2E-07	1.7E-03	2.9E-06	1.0E+03	7.8E-05	1.6E-09	3.9E-06	6.6E-09	9.4E-01	3.4E-02	7.2E-07	1.7E-03	2.9E-06	1.0E+03
LANL	2.1E-02	2.1E-06	1.9E-03	1.4E-06	1.1E+03	1.9E-03	1.9E-07	1.7E-04	1.3E-07	1.1E+01	2.3E-02	2.3E-06	2.1E-03	1.6E-06	1.1E+03
INEEL	8.1E-03	9.9E-07	2.4E-03	2.1E-06	1.4E+03	5.3E-06	6.4E-10	1.6E-06	1.3E-09	3.9E+01	8.1E-03	9.9E-07	2.4E-03	2.1E-06	1.4E+03
SRS	3.0E-02	2.8E-07	3.2E-03	2.8E-06	1.7E+02	0	0	0	0	0	3.0E-02	2.8E-07	3.2E-03	2.8E-06	1.7E+02
RFETS	2.1E-01	2.7E-06	1.0E-02	6.2E-06	3.3E+01	0	0	0	0	0	2.1E-01	2.7E-06	1.0E-02	6.2E-06	3.3E+01
ORNL	0	0	0	0	0	2.2E-03	6.7E-08	7.4E-05	6.7E-08	2.2E+01	2.2E-03	6.7E-08	7.4E-05	6.7E-08	2.2E+01
LLNL	2.0E-03	3.2E-08	1.0E-04	3.5E-08	9.8E-01	0	0	0	0	0	2.0E-03	3.2E-08	1.0E-04	3.5E-08	9.8E-01
NTS	8.2E-06	2.1E-09	2.3E-05	1.0E-07	5.4E-01	0	0	0	0	0	8.2E-06	2.1E-09	2.3E-05	1.0E-07	5.4E-01
ANL-E	3.3E-04	1.8E-09	1.8E-06	1.8E-09	4.6E+00	0	0	0	0	0	3.3E-04	1.8E-09	1.8E-06	1.8E-09	4.6E+00
Total	3.0E-01		1.9E-02		3.8E+03	4.2E-03		2.5E-04		7.4E+01	3.1E-01		2.0E-02		3.9E+03
Radiation-Related LCFs															
Hanford	1.7E-05	3.6E-10	6.8E-07	1.2E-09	4.1E-01	3.9E-08	8.2E-13	1.6E-09	2.7E-12	3.7E-04	1.7E-05	3.6E-10	6.9E-07	1.2E-09	4.1E-01
LANL	1.1E-05	1.1E-09	7.6E-07	5.7E-10	4.5E-01	9.6E-07	9.6E-11	6.8E-08	5.1E-11	4.5E-03	1.2E-05	1.2E-09	8.2E-07	6.2E-10	4.6E-01
INEEL	4.1E-06	4.9E-10	9.6E-07	8.2E-10	5.6E-01	2.6E-09	3.2E-13	6.2E-10	5.3E-13	1.6E-02	4.1E-06	5.0E-10	9.6E-07	8.2E-10	5.8E-01
SRS	1.5E-05	1.4E-10	1.3E-06	1.1E-09	6.9E-02	0	0	0	0	0	1.5E-05	1.4E-10	1.3E-06	1.1E-09	6.9E-02
RFETS	1.0E-04	1.4E-09	4.1E-06	2.5E-09	1.3E-02	0	0	0	0	0	1.0E-04	1.4E-09	4.1E-06	2.5E-09	1.3E-02
ORNL	0	0	0	0	0	1.1E-06	3.3E-11	2.9E-08	2.7E-11	8.9E-03	1.1E-06	3.3E-11	2.9E-08	2.7E-11	8.9E-03
LLNL	1.0E-06	1.6E-11	4.0E-08	1.4E-11	3.9E-04	0	0	0	0	0	1.0E-06	1.6E-11	4.0E-08	1.4E-11	3.9E-04
NTS	4.1E-09	1.1E-12	9.1E-09	4.1E-11	2.2E-04	0	0	0	0	0	4.1E-09	1.1E-12	9.1E-09	4.1E-11	2.2E-04
ANL-E	1.7E-07	9.2E-13	7.0E-10	7.0E-13	1.8E-03	0	0	0	0	0	1.7E-07	9.2E-13	7.0E-10	7.0E-13	1.8E-03
Total	1.5E-04		7.8E-06		1.5E+00	2.1E-06		1.0E-07		2.9E-02	1.5E-04		7.9E-06		1.5E+00
Chemicals - Cancer Incidence															
Hanford	2.8E-10	0	1.7E-10	3.3E-13	7.3E-08	6.6E-13	0	3.9E-13	7.7E-16	6.6E-11	2.8E-10	0	1.7E-10	3.3E-13	7.3E-08
LANL	2.1E-09	2.6E-13	1.1E-09	6.6E-13	1.1E-07	2.1E-11	2.6E-15	1.1E-11	6.6E-15	1.1E-09	2.2E-09	2.6E-13	1.1E-09	6.7E-13	1.1E-07
INEEL	5.1E-09	7.1E-13	6.7E-09	5.5E-12	2.0E-05	1.4E-10	2.0E-14	1.9E-10	1.6E-13	5.7E-07	5.2E-09	7.3E-13	6.8E-09	5.7E-12	2.1E-05
SRS	1.9E-11	0	9.7E-12	1.1E-14	4.3E-09	0	0	0	0	0	1.9E-11	0	9.7E-12	1.1E-14	4.3E-09
RFETS	8.4E-10	0	1.7E-10	8.6E-14	1.2E-08	0	0	0	0	0	8.4E-10	0	1.7E-10	8.6E-14	1.2E-08
ORNL	0	0	0	0	0	5.3E-07	2.7E-11	1.7E-07	1.6E-10	9.9E-06	5.3E-07	2.7E-11	1.7E-07	1.6E-10	9.9E-06
LLNL	9.1E-08	2.0E-12	2.1E-08	6.3E-12	6.2E-07	0	0	0	0	0	9.1E-08	2.0E-12	2.1E-08	6.3E-12	6.2E-07
NTS	3.9E-13	0	5.3E-12	2.2E-14	9.1E-10	0	0	0	0	0	3.9E-13	0	5.3E-12	2.2E-14	9.1E-10
ANL-E	3.1E-10	0	6.5E-12	5.9E-15	4.0E-10	0	0	0	0	0	3.1E-10	0	6.5E-12	5.9E-15	4.0E-10
Total	1.0E-07		2.9E-08		2.1E-05	5.3E-07		1.7E-07		1.0E-05	6.3E-07		2.0E-07		3.2E-05

Table B-5
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2A and No Action Alternative 1A Total Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	2.6E+02	5.3E-03	1.2E+01	2.2E-02	1.0E+03	4.1E+00	8.5E-05	2.0E-01	3.5E-04	2.0E+01	2.6E+02	5.4E-03	1.3E+01	2.2E-02	1.0E+03
LANL	2.5E+02	2.5E-02	2.3E+01	1.7E-02	1.1E+03	0	0	0	0	0	2.5E+02	2.5E-02	2.3E+01	1.7E-02	1.1E+03
INEEL	5.4E+01	6.5E-03	1.6E+01	1.3E-02	1.3E+03	0	0	0	0	0	5.4E+01	6.5E-03	1.6E+01	1.3E-02	1.3E+03
SRS	1.7E+01	1.6E-04	1.8E+00	1.6E-03	1.9E+02	0	0	0	0	0	1.7E+01	1.6E-04	1.8E+00	1.6E-03	1.9E+02
RFETS	4.1E+03	5.6E-02	2.1E+02	1.3E-01	3.2E+01	0	0	0	0	0	4.1E+03	5.6E-02	2.1E+02	1.3E-01	3.2E+01
ORNL	0	0	0	0	0	4.0E+01	1.2E-03	1.4E+00	1.2E-03	6.2E+02	4.0E+01	1.2E-03	1.4E+00	1.2E-03	6.2E+02
Total	4.7E+03		2.6E+02		3.6E+03	4.4E+01		1.6E+00		6.4E+02	4.7E+03		2.6E+02		4.2E+03
Radiation-Related LCFs															
Hanford	1.3E-01	2.6E-06	5.0E-03	8.7E-06	4.0E-01	2.1E-03	4.3E-08	8.0E-05	1.4E-07	8.1E-03	1.3E-01	2.7E-06	5.0E-03	8.8E-06	4.1E-01
LANL	1.3E-01	1.3E-05	9.2E-03	6.7E-06	4.3E-01	0	0	0	0	0	1.3E-01	1.3E-05	9.2E-03	6.7E-06	4.3E-01
INEEL	2.7E-02	3.2E-06	6.3E-03	5.3E-06	5.2E-01	0	0	0	0	0	2.7E-02	3.2E-06	6.3E-03	5.3E-06	5.2E-01
SRS	8.5E-03	7.9E-08	7.2E-04	6.3E-07	7.4E-02	0	0	0	0	0	8.5E-03	7.9E-08	7.2E-04	6.3E-07	7.4E-02
RFETS	2.1E+00	2.8E-05	8.2E-02	5.0E-05	1.3E-02	0	0	0	0	0	2.1E+00	2.8E-05	8.2E-02	5.0E-05	1.3E-02
ORNL	0	0	0	0	0	2.0E-02	6.1E-07	5.4E-04	4.9E-07	2.5E-01	2.0E-02	6.1E-07	5.4E-04	4.9E-07	2.5E-01
Total	2.3E+00		1.0E-01		1.4E+00	2.2E-02		6.2E-04		2.6E-01	2.4E+00		1.0E-01		1.7E+00
Chemicals - Cancer Incidence															
Hanford	4.2E-10	0	2.4E-10	4.8E-13	5.9E-07	1.1E-12	0	6.1E-13	1.2E-15	1.2E-08	4.2E-10	0	2.4E-10	4.8E-13	6.1E-07
LANL	3.5E-09	4.1E-13	1.7E-09	1.0E-12	1.2E-06	0	0	0	0	0	3.5E-09	4.1E-13	1.7E-09	1.0E-12	1.2E-06
INEEL	3.8E-09	5.1E-13	4.9E-09	4.2E-12	4.9E-05	0	0	0	0	0	3.8E-09	5.1E-13	4.9E-09	4.2E-12	4.9E-05
SRS	2.8E-11	0	1.5E-11	1.7E-14	3.6E-08	0	0	0	0	0	2.8E-11	0	1.5E-11	1.7E-14	3.6E-08
RFETS	1.5E-09	0	3.0E-10	1.5E-13	1.1E-07	0	0	0	0	0	1.5E-09	0	3.0E-10	1.5E-13	1.1E-07
ORNL	0	0	0	0	0	3.0E-07	1.5E-11	1.0E-07	9.5E-11	1.4E-05	3.0E-07	1.5E-11	1.0E-07	9.5E-11	1.4E-05
Total	9.1E-09		7.1E-09		5.1E-05	3.0E-07		1.0E-07		1.4E-05	3.1E-07		1.1E-07		6.4E-05

Table B-6
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2B and No Action Alternative 1B Total Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker
	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population
							Radiation Dose (in rem or person-rem)								
Hanford	4.4E+03	9.0E-02	2.1E+02	3.7E-01	1.0E+03	4.1E+00	8.5E-05	2.0E-01	3.5E-04	2.0E+01	4.4E+03	9.1E-02	2.1E+02	3.7E-01	1.0E+03
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	2.5E+02	3.0E-02	7.5E+01	6.3E-02	1.4E+03	0	0	0	0	0	2.5E+02	3.0E-02	7.5E+01	6.3E-02	1.4E+03
SRS	1.7E+01	1.6E-04	1.8E+00	1.6E-03	1.9E+02	0	0	0	0	0	1.7E+01	1.6E-04	1.8E+00	1.6E-03	1.9E+02
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	4.0E+01	1.2E-03	1.4E+00	1.2E-03	6.2E+02	4.0E+01	1.2E-03	1.4E+00	1.2E-03	6.2E+02
Total	4.7E+03		2.9E+02		2.6E+03	4.4E+01		1.6E+00		6.4E+02	4.7E+03		2.9E+02		3.3E+03
							Radiation-Related LCFs								
Hanford	2.2E+00	4.5E-05	8.5E-02	1.5E-04	4.0E-01	2.1E-03	4.3E-08	8.0E-05	1.4E-07	8.1E-03	2.2E+00	4.5E-05	8.5E-02	1.5E-04	4.1E-01
INEEL	1.2E-01	1.5E-05	3.0E-02	2.5E-05	5.8E-01	0	0	0	0	0	1.2E-01	1.5E-05	3.0E-02	2.5E-05	5.8E-01
SRS	8.5E-03	7.9E-08	7.2E-04	6.3E-07	7.4E-02	0	0	0	0	0	8.5E-03	7.9E-08	7.2E-04	6.3E-07	7.4E-02
ORNL	0	0	0	0	0	2.0E-02	6.1E-07	5.4E-04	4.9E-07	2.5E-01	2.0E-02	6.1E-07	5.4E-04	4.9E-07	2.5E-01
Total	2.3E+00		1.2E-01		1.1E+00	2.2E-02		6.2E-04		2.6E-01	2.3E+00		1.2E-01		1.3E+00
							Chemicals - Cancer Incidence								
Hanford	4.2E-10	0	2.4E-10	4.8E-13	5.9E-07	1.1E-12	0	6.1E-13	1.2E-15	1.2E-08	4.2E-10	0	2.4E-10	4.8E-13	6.1E-07
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	4.0E-09	5.6E-13	5.4E-09	4.4E-12	7.7E-05	0	0	0	0	0	4.0E-09	5.6E-13	5.4E-09	4.4E-12	7.7E-05
SRS	2.8E-11	0	1.5E-11	1.7E-14	3.6E-08	0	0	0	0	0	2.8E-11	0	1.5E-11	1.7E-14	3.6E-08
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	3.0E-07	1.5E-11	1.0E-07	9.5E-11	1.4E-05	3.0E-07	1.5E-11	1.0E-07	9.5E-11	1.4E-05
Total	4.4E-09		5.6E-09		7.8E-05	3.0E-07		1.0E-07		1.4E-05	3.0E-07		1.1E-07		9.2E-05

Table B-7
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2C Total Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	0	0	0	0	0	4.1E+00	8.4E-05	1.9E-01	3.4E-04	2.0E+01	4.1E+00	8.4E-05	1.9E-01	3.4E-04	2.0E+01
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	4.0E+01	1.2E-03	1.4E+00	1.2E-03	6.2E+02	4.0E+01	1.2E-03	1.4E+00	1.2E-03	6.2E+02
WIPP	1.8E+03	4.8E-01	1.4E+02	5.5E-01	9.9E+01	0	0	0	0	0	1.8E+03	4.8E-01	1.4E+02	5.5E-01	7.3E+02
Total	1.8E+03		1.4E+02		9.9E+01	4.4E+01		1.5E+00		6.4E+02	1.8E+03		1.5E+02		1.4E+03
Radiation-Related LCFs															
Hanford	0	0	0	0	0	2.0E-03	4.2E-08	7.8E-05	1.4E-07	8.1E-03	2.0E-03	4.2E-08	7.8E-05	1.4E-07	8.1E-03
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	2.0E-02	6.1E-07	5.4E-04	4.9E-07	2.5E-01	2.0E-02	6.1E-07	5.4E-04	4.9E-07	2.5E-01
WIPP	9.0E-01	2.4E-04	5.8E-02	2.2E-04	4.0E-02	0	0	0	0	0	1	0	0	0	2.9E-01
Total	9.0E-01		5.8E-02		4.0E-02	2.2E-02		6.2E-04		2.6E-01	9.2E-01		5.9E-02		5.5E-01
Chemicals - Cancer Incidence															
Hanford	0	0	0	0	0	1.6E-10	0	9.5E-11	1.9E-13	3.5E-09	1.6E-10	0	9.5E-11	1.9E-13	3.5E-09
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	3.0E-07	1.5E-11	1.0E-07	9.5E-11	1.4E-05	3.0E-07	1.5E-11	1.0E-07	9.5E-11	1.4E-05
WIPP	1.3E-09	5.1E-13	6.7E-10	3.1E-12	6.4E-05	0	0	0	0	0	1.3E-09	5.1E-13	6.7E-10	3.1E-12	6.4E-05
Total	1.3E-09		6.7E-10		6.4E-05	3.0E-07		1.0E-07		1.4E-05	3.0E-07		1.0E-07		7.8E-05

Table B-8
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 3 Total Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	1.1E+00	2.3E-05	5.3E-02	9.2E-05	1.0E+03	1.5E-04	3.0E-09	7.2E-06	1.2E-08	1.1E+01	1.1E+00	2.3E-05	5.3E-02	9.2E-05	1.1E+03
LANL	6.7E+00	7.2E-04	6.2E-01	4.6E-04	1.1E+03	0	0	0	0	0	6.7E+00	7.2E-04	6.2E-01	4.6E-04	1.1E+03
INEEL	3.9E-02	4.9E-06	1.2E-02	1.0E-05	1.4E+03	0	0	0	0	0	3.9E-02	4.9E-06	1.2E-02	1.0E-05	1.4E+03
SRS	1.0E+00	9.4E-06	1.1E+00	9.4E-05	1.9E+02	0	0	0	0	0	1.0E+00	9.4E-06	1.1E+00	9.4E-05	1.9E+02
RFETS	4.9E-03	6.6E-08	2.5E-04	1.5E-07	3.3E+01	0	0	0	0	0	4.9E-03	6.6E-08	2.5E-04	1.5E-07	3.3E+01
ORNL	0	0	0	0	0	7.4E-04	2.2E-08	2.5E-05	2.3E-08	1.9E+01	7.4E-04	2.2E-08	2.5E-05	2.3E-08	1.9E+01
Total	8.8E+00		1.8E+00		3.8E+03	8.9E-04		3.2E-05		3.0E+01	8.8E+00		1.8E+00		3.8E+03
Radiation-Related LCFs															
Hanford	5.4E-04	1.1E-08	2.1E-05	3.7E-08	4.2E-01	7.3E-08	1.5E-12	2.9E-09	4.9E-12	4.5E-03	5.4E-04	1.1E-08	2.1E-05	3.7E-08	4.2E-01
LANL	3.3E-03	3.6E-07	2.5E-04	1.9E-07	4.5E-01	0	0	0	0	0	0	0	0	0	4.5E-01
INEEL	2.0E-05	2.4E-09	4.7E-06	4.0E-09	5.5E-01	0	0	0	0	0	0	0	0	0	5.5E-01
SRS	5.1E-04	4.7E-09	4.3E-04	3.7E-08	7.4E-02	0	0	0	0	0	0	0	0	0	7.4E-02
RFETS	2.5E-06	3.3E-11	9.9E-08	6.0E-11	1.3E-02	0	0	0	0	0	0	0	0	0	1.3E-02
ORNL	0	0	0	0	0	3.7E-07	1.1E-11	9.9E-09	9.1E-12	7.5E-03	3.7E-07	1.1E-11	9.9E-09	9.1E-12	7.5E-03
Total	4.4E-03		7.1E-04		1.5E+00	4.4E-07		1.3E-08		1.2E-02	4.4E-03		7.1E-04		1.5E+01
Chemicals - Cancer Incidence															
Hanford	4.2E-10	0	2.5E-10	5.1E-13	3.5E-07	1.1E-12	0	6.4E-13	1.3E-15	3.7E-09	4.2E-10	0	2.5E-10	5.1E-13	3.5E-07
LANL	3.0E-09	6.6E-13	1.5E-09	9.5E-13	4.7E-07	0	0	0	0	0	3.0E-09	6.6E-13	1.5E-09	9.5E-13	4.7E-07
INEEL	5.1E-09	7.0E-13	6.8E-09	5.7E-12	3.3E-05	0	0	0	0	0	5.1E-09	7.0E-13	6.8E-09	5.7E-12	3.3E-05
SRS	3.0E-11	0	1.5E-11	1.8E-14	2.1E-08	0	0	0	0	0	3.0E-11	0	1.5E-11	1.8E-14	2.1E-08
RFETS	1.1E-09	0	2.3E-10	1.1E-13	3.7E-08	0	0	0	0	0	1.1E-09	0	2.3E-10	1.1E-13	3.7E-08
ORNL	0	0	0	0	0	4.3E-07	2.2E-11	1.4E-07	1.4E-10	9.2E-06	4.3E-07	2.2E-11	1.4E-07	1.4E-10	9.2E-06
Total	9.6E-09		8.8E-09		3.4E-05	4.3E-07		1.4E-07		9.2E-06	4.4E-07		1.5E-07		4.3E-05

Table B-9
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2 Total (Basic/Newly Generated) Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	2.6E-01	5.4E-06	1.3E-02	2.2E-05	3.9E+02	7.6E-05	1.6E-09	3.8E-06	6.4E-09	9.0E-01	2.6E-01	5.4E-06	1.3E-02	2.2E-05	3.9E+02
LANL	1.6E+00	1.6E-04	1.4E-01	1.1E-04	3.2E+02	2.0E-03	2.0E-07	1.8E-04	1.4E-07	7.4E+00	1.6E+00	1.6E-04	1.4E-01	1.1E-04	3.3E+02
INEEL	4.3E-05	5.2E-09	1.3E-05	1.1E-08	1.9E+01	7.8E-07	9.5E-11	2.3E-07	2.0E-10	3.2E+01	4.4E-05	5.3E-09	1.3E-05	1.1E-08	5.1E+01
SRS	3.8E-01	3.6E-06	4.1E-02	3.6E-05	9.4E+01	0	0	0	0	0	3.8E-01	3.6E-06	4.1E-02	3.6E-05	9.4E+01
RFETS	3.0E-03	3.9E-08	1.5E-04	8.9E-08	1.8E+01	0	0	0	0	0	3.0E-03	3.9E-08	1.5E-04	8.9E-08	1.8E+01
ORNL	0	0	0	0	0	6.5E-04	2.0E-08	2.2E-05	2.0E-08	1.5E+01	6.5E-04	2.0E-08	2.2E-05	2.0E-08	1.5E+01
LLNL	9.2E-03	1.5E-07	4.6E-04	1.6E-07	7.8E-01	0	0	0	0	0	9.2E-03	1.5E-07	4.6E-04	1.6E-07	7.8E-01
NTS	1.3E-10	3.4E-14	3.7E-10	1.7E-12	1.0E-02	0	0	0	0	0	1.3E-10	3.4E-14	3.7E-10	1.7E-12	1.0E-02
ANL-E	1.9E-03	1.0E-08	1.0E-05	1.0E-08	4.2E+00	0	0	0	0	0	1.9E-03	1.0E-08	1.0E-05	1.0E-08	4.2E+00
Total	2.3E+00		2.0E-01		8.5E+02	2.7E-03		2.1E-04		5.5E+01	2.3E+00		2.0E-01		9.1E+02
Radiation-Related LCFs															
Hanford	1.3E-04	2.7E-09	5.2E-06	8.7E-09	0.2	3.8E-08	8.0E-13	1.5E-09	2.6E-12	3.6E-04	1.3E-04	2.7E-09	5.2E-06	8.8E-09	0.2
LANL	8.1E-04	8.1E-08	5.8E-05	4.4E-08	0.1	1.0E-06	1.0E-10	7.2E-08	5.4E-11	3.0E-03	8.1E-04	8.1E-08	5.8E-05	4.4E-08	0.1
INEEL	2.2E-08	2.6E-12	5.1E-09	4.3E-12	7.4E-03	3.9E-10	4.7E-14	9.2E-11	7.8E-14	1.3E-02	2.2E-08	2.7E-12	5.2E-09	4.4E-12	2.0E-02
SRS	1.9E-04	1.8E-09	1.6E-05	1.4E-08	3.7E-02	0	0	0	0	0	1.9E-04	1.8E-09	1.6E-05	1.4E-08	3.7E-02
RFETS	1.5E-06	1.9E-11	5.8E-08	3.5E-11	7.3E-03	0	0	0	0	0	1.5E-06	1.9E-11	5.8E-08	3.5E-11	7.3E-03
ORNL	0	0	0	0	0	3.2E-07	9.9E-12	8.7E-09	7.9E-12	5.9E-03	3.2E-07	9.9E-12	8.7E-09	7.9E-12	5.9E-03
LLNL	4.6E-06	7.4E-11	1.8E-07	6.4E-11	3.1E-04	0	0	0	0	0	4.6E-06	7.4E-11	1.8E-07	6.4E-11	3.1E-04
NTS	6.7E-14	1.7E-17	1.5E-13	6.7E-16	4.2E-06	0	0	0	0	0	6.7E-14	1.7E-17	1.5E-13	6.7E-16	4.2E-06
ANL-E	9.5E-07	5.2E-12	4.0E-09	4.0E-12	1.7E-03	0	0	0	0	0	9.5E-07	5.2E-12	4.0E-09	4.0E-12	1.7E-03
Total	1.1E-03		8.0E-05		0.3	1.4E-06		8.2E-08		2.2E-02	1.1E-03		8.0E-05		0.4
Chemicals - Cancer Incidence															
Hanford	1.1E-10	0	6.3E-11	1.2E-13	2.8E-08	6.3E-13	0	3.8E-13	7.4E-16	6.4E-11	1.1E-10	0	6.3E-11	1.2E-13	2.8E-08
LANL	6.1E-10	7.5E-14	3.1E-10	1.9E-13	3.2E-08	1.4E-11	1.7E-15	7.0E-12	4.3E-15	7.2E-10	6.3E-10	7.7E-14	3.1E-10	1.9E-13	3.2E-08
INEEL	6.8E-11	9.4E-15	8.9E-11	7.4E-14	2.7E-07	1.2E-10	1.6E-14	1.5E-10	1.3E-13	4.7E-07	1.8E-10	2.6E-14	2.4E-10	2.0E-13	7.4E-07
SRS	1.0E-11	0	5.2E-12	6.1E-15	2.3E-09	0	0	0	0	0	1.0E-11	0	5.2E-12	6.1E-15	2.3E-09
RFETS	4.6E-10	0	9.2E-11	4.7E-14	6.5E-09	0	0	0	0	0	4.6E-10	0	9.2E-11	4.7E-14	6.5E-09
ORNL	0	0	0	0	0	3.5E-07	1.7E-11	1.1E-07	1.1E-10	6.5E-06	3.5E-07	1.7E-11	1.1E-07	1.1E-10	6.5E-06
LLNL	7.3E-08	1.6E-12	1.7E-08	5.0E-12	4.9E-07	0	0	0	0	0	7.3E-08	1.6E-12	1.7E-08	5.0E-12	4.9E-07
NTS	7.6E-15	0	1.0E-13	4.2E-16	1.8E-11	0	0	0	0	0	7.6E-15	0	1.0E-13	4.2E-16	1.8E-11
ANL-E	2.9E-10	0	5.9E-12	5.3E-15	3.6E-10	0	0	0	0	0	2.9E-10	0	5.9E-12	5.3E-15	3.6E-10
Total	7.4E-08		1.7E-08		8.3E-07	3.5E-07		1.1E-07		6.9E-06	4.2E-07		1.3E-07		7.8E-06

Table B-10
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 1 Basic Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker				Maximally Exposed	Noninvolved Worker				Maximally Exposed	Noninvolved Worker		
	Population	Individual	Population	Individual	Worker Population	Population	Individual	Population	Individual	Worker Population	Population	Individual	Population	Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	1.6E-02	3.5E-07	8.2E-04	1.4E-06	4.9E+02	7.6E-05	1.6E-09	3.8E-06	6.4E-09	9.0E-01	1.6E-02	3.5E-07	8.2E-04	1.4E-06	5.0E+02
LANL	1.3E-02	1.3E-06	1.1E-03	8.6E-07	6.8E+02	1.3E-03	1.3E-07	1.1E-04	8.5E-08	7.4E+00	1.4E-02	1.4E-06	1.2E-03	9.4E-07	6.9E+02
INEEL	2.8E-03	3.4E-07	8.2E-04	7.0E-07	4.8E+02	4.3E-06	5.2E-10	1.3E-06	1.1E-09	3.2E+01	2.8E-03	3.4E-07	8.2E-04	7.0E-07	5.1E+02
SRS	2.1E-02	2.0E-07	2.2E-03	2.0E-06	1.2E+02	0	0	0	0	0	2.1E-02	2.0E-07	2.2E-03	2.0E-06	1.2E+02
RFETS	2.1E-01	2.7E-06	1.0E-02	6.2E-06	3.3E+01	0	0	0	0	0	2.1E-01	2.7E-06	1.0E-02	6.2E-06	3.3E+01
ORNL	0	0	0	0	0	1.4E-03	4.4E-08	4.8E-05	4.4E-08	1.5E+01	1.4E-03	4.4E-08	4.8E-05	4.4E-08	1.5E+01
LLNL	2.0E-03	3.2E-08	1.0E-04	3.5E-08	9.8E-01	0	0	0	0	0	2.0E-03	3.2E-08	1.0E-04	3.5E-08	9.8E-01
NTS	8.2E-06	2.1E-09	2.3E-05	1.0E-07	5.4E-01	0	0	0	0	0	8.2E-06	2.1E-09	2.3E-05	1.0E-07	5.4E-01
ANL-E	3.3E-04	1.8E-09	1.8E-06	1.8E-09	4.6E+00	0	0	0	0	0	3.3E-04	1.8E-09	1.8E-06	1.8E-09	4.6E+00
Total	2.6E-01		1.5E-02		1.8E+03	2.8E-03		1.7E-04		5.5E+01	2.6E-01		1.5E-02		1.9E+03
Radiation-Related LCFs															
Hanford	8.2E-06	1.7E-10	3.3E-07	5.6E-10	2.0E-01	3.8E-08	8.0E-13	1.5E-09	2.6E-12	3.6E-04	8.2E-06	1.7E-10	3.3E-07	5.6E-10	2.0E-01
LANL	6.4E-06	6.4E-10	4.5E-07	3.4E-10	2.7E-01	6.3E-07	6.3E-11	4.5E-08	3.4E-11	3.0E-03	7.0E-06	7.0E-10	5.0E-07	3.8E-10	2.8E-01
INEEL	1.4E-06	1.7E-10	3.3E-07	2.8E-10	1.9E-01	2.1E-09	2.6E-13	5.1E-10	4.3E-13	1.3E-02	1.4E-06	1.7E-10	3.3E-07	2.8E-10	2.0E-01
SRS	1.1E-05	9.8E-11	9.0E-07	7.9E-10	4.9E-02	0	0	0	0	0	1.1E-05	9.8E-11	9.0E-07	7.9E-10	4.9E-02
RFETS	1.0E-04	1.4E-09	4.1E-06	2.5E-09	1.3E-02	0	0	0	0	0	1.0E-04	1.4E-09	4.1E-06	2.5E-09	1.3E-02
ORNL	0	0	0	0	0	7.2E-07	2.2E-11	1.9E-08	1.8E-11	5.9E-03	7.2E-07	2.2E-11	1.9E-08	1.8E-11	5.9E-03
LLNL	1.0E-06	1.6E-11	4.0E-08	1.4E-11	3.9E-04	0	0	0	0	0	1.0E-06	1.6E-11	4.0E-08	1.4E-11	3.9E-04
NTS	4.1E-09	1.1E-12	9.1E-09	4.1E-11	2.2E-04	0	0	0	0	0	4.1E-09	1.1E-12	9.1E-09	4.1E-11	2.2E-04
ANL-E	1.7E-07	9.2E-13	7.0E-10	7.0E-13	1.8E-03	0	0	0	0	0	1.7E-07	9.2E-13	7.0E-10	7.0E-13	1.8E-03
Total	1.3E-04		6.1E-06		7.3E-01	1.4E-06		6.6E-08		2.2E-02	1.3E-04		6.2E-06		7.5E-01
Chemicals - Cancer Incidence															
Hanford	1.3E-10	0	8.0E-11	1.6E-13	3.5E-08	6.3E-13	0	3.8E-13	7.4E-16	6.4E-11	1.3E-10	0	8.0E-11	1.6E-13	3.5E-08
LANL	1.3E-09	1.6E-13	6.5E-10	4.0E-13	6.6E-08	1.4E-11	1.7E-15	7.0E-12	4.3E-15	7.2E-10	1.3E-09	1.6E-13	6.5E-10	4.0E-13	6.7E-08
INEEL	1.7E-09	2.4E-13	2.3E-09	1.9E-12	6.9E-06	1.2E-10	1.6E-14	1.5E-10	1.3E-13	4.7E-07	1.9E-09	2.6E-13	2.4E-09	2.0E-12	7.4E-06
SRS	1.4E-11	0	6.9E-12	8.0E-15	3.1E-09	0	0	0	0	0	1.4E-11	0	6.9E-12	8.0E-15	3.1E-09
RFETS	8.4E-10	0	1.7E-10	8.6E-14	1.2E-08	0	0	0	0	0	8.4E-10	0	1.7E-10	8.6E-14	1.2E-08
ORNL	0	0	0	0	0	3.5E-07	1.7E-11	1.1E-07	1.1E-10	6.5E-06	3.5E-07	1.7E-11	1.1E-07	1.1E-10	6.5E-06
LLNL	9.1E-08	2.0E-12	2.1E-08	6.3E-12	6.2E-07	0	0	0	0	0	9.1E-08	2.0E-12	2.1E-08	6.3E-12	6.2E-07
NTS	3.9E-13	0	5.3E-12	2.2E-14	9.1E-10	0	0	0	0	0	3.9E-13	0	5.3E-12	2.2E-14	9.1E-10
ANL-E	3.1E-10	0	6.5E-12	5.9E-15	4.0E-10	0	0	0	0	0	3.1E-10	0	6.5E-12	5.9E-15	4.0E-10
Total	9.6E-08		2.4E-08		7.7E-06	3.5E-07		1.1E-07		6.9E-06	4.4E-07		1.4E-07		1.5E-05

Table B-11
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2A and No Action Alternative 1A Basic Inventory

	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker
Site	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population
Radiation Dose (in rem or person-rem)															
Hanford	1.2E+02	2.5E-03	6.0E+00	1.0E-02	4.8E+02	3.9E+00	8.1E-05	1.9E-01	3.4E-04	1.9E+01	1.3E+02	2.6E-03	6.2E+00	1.1E-02	5.0E+02
LANL	1.5E+02	1.5E-02	1.4E+01	1.0E-02	6.5E+02	0	0	0	0	0	1.5E+02	1.5E-02	1.4E+01	1.0E-02	6.5E+02
INEEL	1.9E+01	2.2E-03	5.4E+00	4.6E-03	4.5E+02	0	0	0	0	0	1.9E+01	2.2E-03	5.4E+00	4.6E-03	4.5E+02
SRS	1.2E+01	1.2E-04	1.3E+00	1.2E-03	1.4E+02	0	0	0	0	0	1.2E+01	1.2E-04	1.3E+00	1.2E-03	1.4E+02
RFETS	4.1E+03	5.6E-02	2.1E+02	1.3E-01	3.2E+01	0	0	0	0	0	4.1E+03	5.6E-02	2.1E+02	1.3E-01	3.2E+01
ORNL	0	0	0	0	0	2.6E+01	8.0E-04	8.9E-01	8.0E-04	4.1E+02	2.6E+01	8.0E-04	8.9E-01	8.0E-04	4.1E+02
Total	4.4E+03		2.3E+02		1.7E+03	3.0E+01		1.1E+00		4.3E+02	4.4E+03		2.3E+02		2.2E+03
Radiation-Related LCFs															
Hanford	6.2E-02	1.3E-06	2.4E-03	4.2E-06	1.9E-01	2.0E-03	4.1E-08	7.7E-05	1.3E-07	7.7E-03	6.4E-02	1.3E-06	2.5E-03	4.3E-06	2.0E-01
LANL	7.5E-02	7.5E-06	5.6E-03	4.0E-06	2.6E-01	0	0	0	0	0	7.5E-02	7.5E-06	5.6E-03	4.0E-06	2.6E-01
INEEL	9.3E-03	1.1E-06	2.2E-03	1.8E-06	1.8E-01	0	0	0	0	0	9.3E-03	1.1E-06	2.2E-03	1.8E-06	1.8E-01
SRS	6.2E-03	5.8E-08	5.3E-04	4.6E-07	5.5E-02	0	0	0	0	0	6.2E-03	5.8E-08	5.3E-04	4.6E-07	5.5E-02
RFETS	2.1E+00	2.8E-05	8.2E-02	5.0E-05	1.3E-02	0	0	0	0	0	2.1E+00	2.8E-05	8.2E-02	5.0E-05	1.3E-02
ORNL	0	0	0	0	0	1.3E-02	4.0E-07	3.5E-04	3.2E-07	1.6E-01	1.3E-02	4.0E-07	3.5E-04	3.2E-07	1.6E-01
Total	2.2E+00		9.3E-02		7.0E-01	1.5E-02		4.3E-04		1.7E-01	2.2E+00		9.3E-02		8.7E-01
Chemicals - Cancer Incidence															
Hanford	2.0E-10	0	1.2E-10	2.3E-13	2.9E-07	1.0E-12	0	5.8E-13	1.2E-15	1.1E-08	2.0E-10	0	1.2E-10	2.3E-13	3.0E-07
LANL	2.1E-09	2.5E-13	1.0E-09	6.3E-13	7.0E-07	0	0	0	0	0	2.1E-09	2.5E-13	1.0E-09	6.3E-13	7.0E-07
INEEL	1.3E-09	1.7E-13	1.7E-09	1.4E-12	1.7E-05	0	0	0	0	0	1.3E-09	1.7E-13	1.7E-09	1.4E-12	1.7E-05
SRS	2.1E-11	0	1.1E-11	1.2E-14	2.7E-08	0	0	0	0	0	2.1E-11	0	1.1E-11	1.2E-14	2.7E-08
RFETS	1.5E-09	0	3.0E-10	1.5E-13	1.1E-07	0	0	0	0	0	1.5E-09	0	3.0E-10	1.5E-13	1.1E-07
ORNL	0	0	0	0	0	2.0E-07	1.0E-11	6.6E-08	6.2E-11	9.1E-06	2.0E-07	1.0E-11	6.6E-08	6.2E-11	9.1E-06
Total	5.1E-09		3.1E-09		1.8E-05	2.0E-07		6.6E-08		9.1E-06	2.0E-07		6.9E-08		2.7E-05

Table B-12
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2B and No Action Alternative 1B Basic Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	2.1E+03	4.4E-02	1.0E+02	1.8E-01	4.8E+02	3.9E+00	8.1E-05	1.9E-01	3.4E-04	1.9E+01	2.1E+03	4.4E-02	1.0E+02	1.8E-01	5.0E+02
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	1.1E+02	1.4E-02	3.5E+01	2.9E-02	6.7E+02	0	0	0	0	0	1.1E+02	1.4E-02	3.5E+01	2.9E-02	6.7E+02
SRS	1.2E+01	1.2E-04	1.3E+00	1.2E-03	1.4E+02	0	0	0	0	0	1.2E+01	1.2E-04	1.3E+00	1.2E-03	1.4E+02
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	2.6E+01	8.0E-04	8.9E-01	8.0E-04	4.1E+02	2.6E+01	8.0E-04	8.9E-01	8.0E-04	4.1E+02
Total	2.2E+03		1.4E+02		1.3E+03	3.0E+01		1.1E+00		4.3E+02	2.3E+03		1.4E+02		1.7E+03
Radiation-Related LCFs															
Hanford	1.1E+00	2.2E-05	4.1E-02	7.2E-05	1.9E-01	2.0E-03	4.1E-08	7.7E-05	1.3E-07	7.7E-03	1.1E+00	2.2E-05	4.1E-02	7.2E-05	2.0E-01
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	5.7E-02	7.0E-06	1.4E-02	1.2E-05	2.7E-01	0	0	0	0	0	5.7E-02	7.0E-06	1.4E-02	1.2E-05	2.7E-01
SRS	6.2E-03	5.8E-08	5.3E-04	4.6E-07	5.5E-02	0	0	0	0	0	6.2E-03	5.8E-08	5.3E-04	4.6E-07	5.5E-02
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.3E-02	4.0E-07	3.5E-04	3.2E-07	1.6E-01	1.3E-02	4.0E-07	3.5E-04	3.2E-07	1.6E-01
Total	1.1E+00		5.6E-02		5.2E-01	1.5E-02		4.3E-04		1.7E-01	1.1E+00		5.6E-02		6.9E-01
Chemicals - Cancer Incidence															
Hanford	2.0E-10	0	1.2E-10	2.3E-13	2.9E-07	1.0E-12	0	5.8E-13	1.2E-15	1.1E-08	2.0E-10	0	1.2E-10	2.3E-13	3.0E-07
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	1.8E-09	2.6E-13	2.5E-09	2.1E-12	3.6E-05	0	0	0	0	0	1.8E-09	2.6E-13	2.5E-09	2.1E-12	3.6E-05
SRS	2.1E-11	0	1.1E-11	1.2E-14	2.7E-08	0	0	0	0	0	2.1E-11	0	1.1E-11	1.2E-14	2.7E-08
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	2.0E-07	1.0E-11	6.6E-08	6.2E-11	9.1E-06	2.0E-07	1.0E-11	6.6E-08	6.2E-11	9.1E-06
Total	2.1E-09		2.6E-09		3.6E-05	2.0E-07		6.6E-08		9.1E-06	2.0E-07		6.8E-08		4.5E-05

Table B-13
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2C Basic Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker				Maximally Exposed	Noninvolved Worker				Maximally Exposed	Noninvolved Worker		
	Population	Individual	Population	Individual	Worker Population	Population	Individual	Population	Individual	Worker Population	Population	Individual	Population	Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	0	0	0	0	0	3.9E+00	8.0E-05	1.8E-01	3.3E-04	1.9E+01	3.9E+00	8.0E-05	1.8E-01	3.3E-04	1.9E+01
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	2.6E+01	8.0E-04	8.9E-01	8.0E-04	4.1E+02	2.6E+01	8.0E-04	8.9E-01	8.0E-04	4.1E+02
WIPP	8.8E+02	2.4E-01	7.1E+01	2.7E-01	3.6E+02	0	0	0	0	0	8.8E+02	2.4E-01	7.1E+01	2.7E-01	3.6E+02
Total	8.8E+02		7.1E+01		3.6E+02	3.0E+01		1.1E+00		4.3E+02	9.1E+02		7.2E+01		7.9E+02
Radiation-Related LCFs															
Hanford	0	0	0	0	0	1.9E-03	4.0E-08	7.4E-05	1.3E-07	7.7E-03	1.9E-03	4.0E-08	7.4E-05	1.3E-07	7.7E-03
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.3E-02	4.0E-07	3.5E-04	3.2E-07	1.6E-01	1.3E-02	4.0E-07	3.5E-04	3.2E-07	1.6E-01
WIPP	4.4E-01	1.2E-04	2.9E-02	1.1E-04	1.4E-01	0	0	0	0	0	0	0	0	0	1.4E-01
Total	4.4E-01		2.9E-02		1.4E-01	1.5E-02		4.3E-04		1.7E-01	4.6E-01		2.9E-02		3.2E-01
Chemicals - Cancer Incidence															
Hanford	0	0	0	0	0	1.5E-10	0	9.1E-11	1.8E-13	3.3E-09	1.5E-10	0	9.1E-11	1.8E-13	3.3E-09
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	2.0E-07	1.0E-11	6.6E-08	6.2E-11	9.1E-06	2.0E-07	1.0E-11	6.6E-08	6.2E-11	9.1E-06
WIPP	6.5E-10	2.5E-13	3.3E-10	1.5E-12	3.2E-05	0	0	0	0	0	6.5E-10	2.5E-13	3.3E-10	1.5E-12	3.2E-05
Total	6.5E-10		3.3E-10		3.2E-05	2.0E-07		6.6E-08		9.1E-06	2.0E-07		6.6E-08		4.1E-05

Table B-14
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 3 Basic Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	5.3E-01	1.1E-05	2.6E-02	4.4E-05	5.1E+02	1.4E-04	2.9E-09	6.8E-06	1.2E-08	1.1E+01	5.3E-01	1.1E-05	2.6E-02	4.4E-05	5.2E+02
LANL	4.0E+00	4.3E-04	3.7E-01	2.8E-04	6.8E+02	0	0	0	0	0	4.0E+00	4.3E-04	3.7E-01	2.8E-04	6.8E+02
INEEL	1.4E-02	1.7E-06	4.0E-03	3.5E-06	4.8E+02	0	0	0	0	0	1.4E-02	1.7E-06	4.0E-03	3.5E-06	4.8E+02
SRS	7.5E-01	6.9E-06	8.0E-01	6.9E-05	1.4E+02	0	0	0	0	0	7.5E-01	6.9E-06	8.0E-01	6.9E-05	1.4E+02
RFETS	4.9E-03	6.6E-08	2.5E-04	1.5E-07	3.3E+01	0	0	0	0	0	4.9E-03	6.6E-08	2.5E-04	1.5E-07	3.3E+01
ORNL	0	0	0	0	0	4.9E-04	1.5E-08	1.6E-05	1.5E-08	1.2E+01	4.9E-04	1.5E-08	1.6E-05	1.5E-08	1.2E+01
Total	5.3E+00		1.2E+00		1.8E+03	6.3E-04		2.3E-05		2.3E+01	5.3E+00		1.2E+00		1.9E+03
Radiation-Related LCFs															
Hanford	2.6E-04	5.5E-09	1.0E-05	1.8E-08	2.0E-01	7.0E-08	1.5E-12	2.7E-09	4.7E-12	4.3E-03	2.6E-04	5.5E-09	1.0E-05	1.8E-08	0.2
LANL	2.0E-03	2.2E-07	1.5E-04	1.1E-07	2.7E-01	0	0	0	0	0	2.0E-03	2.2E-07	1.5E-04	1.1E-07	0.3
INEEL	6.8E-06	8.4E-10	1.6E-06	1.4E-09	1.9E-01	0	0	0	0	0	6.8E-06	8.4E-10	1.6E-06	1.4E-09	0.2
SRS	3.7E-04	3.5E-09	3.2E-04	2.8E-08	5.5E-02	0	0	0	0	0	3.7E-04	3.5E-09	3.2E-04	2.8E-08	5.5E-02
RFETS	2.5E-06	3.3E-11	9.9E-08	6.0E-11	1.3E-02	0	0	0	0	0	2.5E-06	3.3E-11	9.9E-08	6.0E-11	1.3E-02
ORNL	0	0	0	0	0	2.4E-07	7.3E-12	6.5E-09	5.9E-12	4.9E-03	2.4E-07	7.3E-12	6.5E-09	5.9E-12	4.9E-03
Total	2.7E-03		4.8E-04		7.3E-01	3.1E-07		9.2E-09		9.2E-03	2.7E-03		4.8E-04		7.4E-01
Chemicals - Cancer Incidence															
Hanford	2.0E-10	0	1.2E-10	2.5E-13	1.7E-07	1.0E-12	0	6.1E-13	1.2E-15	3.6E-09	2.0E-10	0	1.2E-10	2.5E-13	1.7E-07
LANL	1.8E-09	4.0E-13	8.9E-10	5.7E-13	2.8E-07	0	0	0	0	0	1.8E-09	4.0E-13	8.9E-10	5.7E-13	2.8E-07
INEEL	1.7E-09	2.4E-13	2.4E-09	2.0E-12	1.1E-05	0	0	0	0	0	1.7E-09	2.4E-13	2.4E-09	2.0E-12	1.1E-05
SRS	2.2E-11	0	1.1E-11	1.3E-14	1.6E-08	0	0	0	0	0	2.2E-11	0	1.1E-11	1.3E-14	1.6E-08
RFETS	0	0	0	0	0	0	0	0	0	0	1.1E-09	0	2.3E-10	1.1E-13	3.7E-08
ORNL	0	0	0	0	0	2.8E-07	1.4E-11	9.2E-08	8.9E-11	6.0E-06	2.8E-07	1.4E-11	9.2E-08	8.9E-11	6.0E-06
Total	4.9E-09		3.6E-09		1.2E-05	2.8E-07		9.2E-08		6.1E-06	2.9E-07		9.6E-08		1.8E-05

Table B-15
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 1 Additional Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	1.8E-02	3.8E-07	8.9E-04	1.5E-06	5.4E+02	2.6E-06	5.6E-11	1.3E-07	2.2E-10	3.2E-02	1.8E-02	3.8E-07	8.9E-04	1.5E-06	5.4E+02
LANL	8.4E-03	8.4E-07	7.5E-04	5.7E-07	4.5E+02	6.5E-04	6.5E-08	5.8E-05	4.4E-08	3.8E+00	9.1E-03	9.1E-07	8.1E-04	6.1E-07	4.5E+02
INEEL	5.4E-03	6.5E-07	1.6E-03	1.4E-06	9.2E+02	9.7E-07	1.2E-10	2.9E-07	2.5E-10	7.2E+00	5.4E-03	6.5E-07	1.6E-03	1.4E-06	9.3E+02
SRS	8.5E-03	7.9E-08	9.1E-04	7.9E-07	5.0E+01	0	0	0	0	0	8.5E-03	7.9E-08	9.1E-04	7.9E-07	5.0E+01
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	7.5E-04	2.3E-08	2.5E-05	2.3E-08	7.7E+00	7.5E-04	2.3E-08	2.5E-05	2.3E-08	7.7E+00
Total	4.0E-02		4.1E-03		2.0E+03	1.4E-03		8.4E-05		1.9E+01	4.2E-02		4.2E-03		2.0E+03
Radiation-Related LCFs															
Hanford	8.9E-06	1.9E-10	3.6E-07	6.0E-10	2.2E-01	1.3E-09	2.8E-14	5.3E-11	9.0E-14	1.3E-05	8.9E-06	1.9E-10	3.6E-07	6.0E-10	2.2E-01
LANL	4.2E-06	4.2E-10	3.0E-07	2.3E-10	1.8E-01	3.3E-07	3.3E-11	2.3E-08	1.8E-11	1.5E-03	4.5E-06	4.5E-10	3.2E-07	2.4E-10	1.8E-01
INEEL	2.7E-06	3.3E-10	6.3E-07	5.4E-10	3.7E-01	4.9E-10	5.9E-14	1.2E-10	9.8E-14	2.9E-03	2.7E-06	3.3E-10	6.3E-07	5.4E-10	3.7E-01
SRS	4.3E-06	4.0E-11	3.6E-07	3.2E-10	2.0E-02	0	0	0	0	0	4.3E-06	4.0E-11	3.6E-07	3.2E-10	2.0E-02
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	3.7E-07	1.1E-11	1.0E-08	9.2E-12	3.1E-03	3.7E-07	1.1E-11	1.0E-08	9.2E-12	3.1E-03
Total	2.0E-05		1.7E-06		7.8E-01	7.0E-07		3.4E-08		7.5E-03	2.1E-05		1.7E-06		7.9E-01
Chemicals - Cancer Incidence															
Hanford	1.5E-10	0	8.7E-11	1.7E-13	3.8E-08	2.2E-14	0	1.3E-14	2.6E-17	2.2E-12	1.5E-10	0	8.7E-11	1.7E-13	3.8E-08
LANL	8.5E-10	1.0E-13	4.3E-10	2.6E-13	4.4E-08	7.3E-12	8.9E-16	3.6E-12	2.2E-15	3.7E-10	8.6E-10	1.0E-13	4.3E-10	2.7E-13	4.4E-08
INEEL	3.4E-09	4.7E-13	4.4E-09	3.7E-12	1.3E-05	2.6E-11	3.7E-15	3.4E-11	2.9E-14	1.1E-07	3.4E-09	4.7E-13	4.4E-09	3.7E-12	1.4E-05
SRS	5.6E-12	0	2.8E-12	3.2E-15	1.2E-09	0	0	0	0	0	5.6E-12	0	2.8E-12	3.2E-15	1.2E-09
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.8E-07	0	5.9E-08	5.6E-11	3.4E-06	1.8E-07	9.1E-12	5.9E-08	5.6E-11	3.4E-06
Total	4.4E-09		4.9E-09		1.4E-05	1.8E-07		5.9E-08		3.5E-06	1.9E-07		6.4E-08		1.7E-05

Table B-16
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2A and No Action Alternative 1A Additional Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker				Maximally Exposed	Noninvolved Worker				Maximally Exposed	Noninvolved Worker		
	Population	Individual	Population	Individual	Worker Population	Population	Individual	Population	Individual	Worker Population	Population	Individual	Population	Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	1.3E+02	2.7E-03	6.4E+00	1.1E-02	5.2E+02	2.0E-01	4.1E-06	9.6E-03	1.7E-05	9.7E-01	1.3E+02	2.7E-03	6.4E+00	1.1E-02	5.2E+02
LANL	1.0E+02	1.0E-02	9.2E+00	6.7E-03	4.3E+02	0	0	0	0	0	1.0E+02	1.0E-02	9.2E+00	6.7E-03	4.3E+02
INEEL	3.5E+01	4.2E-03	1.0E+01	8.7E-03	8.6E+02	0	0	0	0	0	3.5E+01	4.2E-03	1.0E+01	8.7E-03	8.6E+02
SRS	4.5E+00	4.2E-05	4.7E-01	4.2E-04	4.9E+01	0	0	0	0	0	4.5E+00	4.2E-05	4.7E-01	4.2E-04	4.9E+01
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.4E+01	4.2E-04	4.6E-01	4.2E-04	2.1E+02	1.4E+01	4.2E-04	4.6E-01	4.2E-04	2.1E+02
Total	2.7E+02		2.6E+01		1.8E+03	1.4E+01		4.7E-01		2.2E+02	2.9E+02		2.7E+01		2.1E+03
Radiation-Related LCFs															
Hanford	6.6E-02	1.4E-06	2.6E-03	4.5E-06	2.1E-01	9.9E-05	2.0E-09	3.9E-06	6.7E-09	3.9E-04	6.6E-02	1.4E-06	2.6E-03	4.5E-06	2.1E-01
LANL	5.0E-02	5.0E-06	3.7E-03	2.7E-06	1.7E-01	0	0	0	0	0	5.0E-02	5.0E-06	3.7E-03	2.7E-06	1.7E-01
INEEL	1.8E-02	2.1E-06	4.2E-03	3.5E-06	3.4E-01	0	0	0	0	0	1.8E-02	2.1E-06	4.2E-03	3.5E-06	3.4E-01
SRS	2.2E-03	2.1E-08	1.9E-04	1.7E-07	2.0E-02	0	0	0	0	0	2.2E-03	2.1E-08	1.9E-04	1.7E-07	2.0E-02
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	6.9E-03	2.1E-07	1.9E-04	1.7E-07	8.6E-02	6.9E-03	2.1E-07	1.9E-04	1.7E-07	8.6E-02
Total	1.4E-01		1.1E-02		7.4E-01	7.0E-03		1.9E-04		8.6E-02	1.4E-01		1.1E-02		8.3E-01
Chemicals - Cancer Incidence															
Hanford	2.2E-10	0	1.2E-10	2.5E-13	3.1E-07	5.1E-14	0	2.9E-14	5.9E-17	5.8E-10	2.2E-10	0	1.2E-10	2.5E-13	3.1E-07
LANL	1.4E-09	1.6E-13	6.8E-10	4.1E-13	4.6E-07	0	0	0	0	0	1.4E-09	1.6E-13	6.8E-10	4.1E-13	4.6E-07
INEEL	2.5E-09	3.3E-13	3.2E-09	2.8E-12	3.2E-05	0	0	0	0	0	2.5E-09	3.3E-13	3.2E-09	2.8E-12	3.2E-05
SRS	7.5E-12	0	3.9E-12	4.4E-15	9.5E-09	0	0	0	0	0	7.5E-12	0	3.9E-12	4.4E-15	9.5E-09
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.0E-07	5.3E-12	3.4E-08	3.3E-11	4.7E-06	1.0E-07	5.3E-12	3.4E-08	3.3E-11	4.7E-06
Total	4.1E-09		4.0E-09		3.3E-05	1.0E-07		3.4E-08		4.7E-06	2.4E-07		6.8E-08		3.8E-05

Table B-17
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2B and No Action Alternative 1A Additional Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker
	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population
Radiation Dose (in rem or person-rem)															
Hanford	2.3E+03	4.7E-02	1.1E+02	1.9E-01	5.2E+02	2.0E-01	4.1E-06	9.6E-03	1.7E-05	9.7E-01	2.3E+03	4.7E-02	1.1E+02	1.9E-01	5.2E+02
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	1.3E+02	1.6E-02	4.0E+01	3.4E-02	7.8E+02	0	0	0	0	0	1.3E+02	1.6E-02	4.0E+01	3.4E-02	7.8E+02
SRS	4.5E+00	4.2E-05	4.7E-01	4.2E-04	4.9E+01	0	0	0	0	0	4.5E+00	4.2E-05	4.7E-01	4.2E-04	4.9E+01
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.4E+01	4.2E-04	4.6E-01	4.2E-04	2.1E+02	1.4E+01	4.2E-04	4.6E-01	4.2E-04	2.1E+02
Total	2.4E+03		1.5E+02		1.3E+03	1.4E+01		4.7E-01		2.2E+02	2.4E+03		1.5E+02		1.6E+03
Radiation-Related LCFs															
Hanford	1.1E+00	2.3E-05	4.4E-02	7.7E-05	2.1E-01	9.9E-05	2.0E-09	3.9E-06	6.7E-09	3.9E-04	1.1E+00	2.3E-05	4.4E-02	7.7E-05	2.1E-01
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	6.6E-02	8.1E-06	1.6E-02	1.4E-05	3.1E-01	0	0	0	0	0	6.6E-02	8.1E-06	1.6E-02	1.4E-05	3.1E-01
SRS	2.2E-03	2.1E-08	1.9E-04	1.7E-07	2.0E-02	0	0	0	0	0	2.2E-03	2.1E-08	1.9E-04	1.7E-07	2.0E-02
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	6.9E-03	2.1E-07	1.9E-04	1.7E-07	8.6E-02	6.9E-03	2.1E-07	1.9E-04	1.7E-07	8.6E-02
Total	1.2E+00		6.0E-02		5.4E-01	7.0E-03		1.9E-04		8.6E-02	1.2E+00		6.1E-02		6.2E-01
Chemicals - Cancer Incidence															
Hanford	2.2E-10	0	1.2E-10	2.5E-13	3.1E-07	5.1E-14	0	2.9E-14	5.9E-17	5.8E-10	2.2E-10	0	1.2E-10	2.5E-13	3.1E-07
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	2.1E-09	3.0E-13	2.9E-09	2.4E-12	4.1E-05	0	0	0	0	0	2.1E-09	3.0E-13	2.9E-09	2.4E-12	4.1E-05
SRS	7.5E-12	0	3.9E-12	4.4E-15	9.5E-09	0	0	0	0	0	7.5E-12	0	3.9E-12	4.4E-15	9.5E-09
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.0E-07	5.3E-12	3.4E-08	3.3E-11	4.7E-06	1.0E-07	5.3E-12	3.4E-08	3.3E-11	4.7E-06
Total	2.3E-09		3.0E-09		4.2E-05	1.0E-07		3.4E-08		4.7E-06	1.0E-07		3.7E-08		4.6E-05

Table B-18
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 2C Additional Inventory

	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker		Maximally Exposed	Noninvolved Worker		Worker
Site	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population	Population	Individual	Population	Individual	Population
Radiation Dose (in rem or person-rem)															
Hanford	0	0	0	0	0	2.0E-01	4.1E-06	9.3E-03	1.7E-05	9.7E-01	2.0E-01	4.1E-06	9.3E-03	1.7E-05	9.7E-01
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.4E+01	4.2E-04	4.6E-01	4.2E-04	2.1E+02	1.4E+01	4.2E-04	4.6E-01	4.2E-04	2.1E+02
WIPP	9.1E+02	2.5E-01	7.4E+01	2.8E-01	3.7E+02	0	0	0	0	0	9.1E+02	2.5E-01	7.4E+01	2.8E-01	3.7E+02
Total	9.1E+02		7.4E+01		3.7E+02	1.4E+01		4.7E-01		2.2E+02	9.2E+02		7.4E+01		5.9E+02
Radiation-Related LCFs															
Hanford	0	0	0	0	0	9.8E-05	2.0E-09	3.7E-06	6.6E-09	3.9E-04	9.8E-05	2.0E-09	3.7E-06	6.6E-09	3.9E-04
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	6.9E-03	2.1E-07	1.9E-04	1.7E-07	8.6E-02	6.9E-03	2.1E-07	1.9E-04	1,7E-07	8.6E-02
WIPP	4.6E-01	1.2E-04	2.9E-02	1.1E-04	1.5E-01	0	0	0	0	0	0	0	0	0	1.5E-01
Total	4.6E-01		2.9E-02		1.5E-01	7.0E-03		1.9E-04		8.6E-02	4.6E-01		3.0E-02		2.3E-01
Chemicals - Cancer Incidence															
Hanford	0	0	0	0	0	7.7E-12	0	4.6E-12	8.9E-15	1.7E-10	7.7E-12	0	4.6E-12	8.9E-15	1.7E-10
LANL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INEEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
SRS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.0E-07	5.3E-12	3.4E-08	3.3E-11	4.7E-06	1.0E-07	5.3E-12	3.4E-08	3.3E-11	4.7E-06
WIPP	6.7E-10	2.6E-13	3.4E-10	1.6E-12	3.3E-05	0	0	0	0	0	6.7E-10	2.6E-13	3.4E-10	1.6E-12	3.3E-05
Total	6.7E-10		3.4E-10		3.3E-05	1.0E-07		3.4E-08		4.7E-06	1.0E-07		3.5E-08		3.7E-05

Table B-19
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals of Action Alternative 3 Additional Inventory

Site	Contact-Handled (CH-TRU) Waste					Remote-Handled (RH-TRU) Waste					CH-TRU Waste + RH-TRU Waste				
	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population	Population	Maximally Exposed Individual	Noninvolved Worker Population	Noninvolved Worker Individual	Worker Population
Radiation Dose (in rem or person-rem)															
Hanford	5.6E-01	1.2E-05	2.7E-02	4.7E-05	5.4E+02	7.0E-06	1.5E-10	3.4E-07	5.9E-10	5.4E-01	5.6E-01	1.2E-05	2.7E-02	4.7E-05	5.4E+02
LANL	2.7E+00	2.9E-04	2.5E-01	1.8E-04	4.5E+02	0	0	0	0	0	2.7E+00	2.9E-04	2.5E-01	1.8E-04	4.5E+02
INEEL	2.6E-02	3.2E-06	7.6E-03	6.6E-06	9.1E+02	0	0	0	0	0	2.6E-02	3.2E-06	7.6E-03	6.6E-06	9.1E+02
SRS	2.7E-01	2.5E-06	2.9E-01	2.5E-05	4.9E+01	0	0	0	0	0	2.7E-01	2.5E-06	2.9E-01	2.5E-05	4.9E+01
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	2.5E-04	7.6E-09	8.5E-06	7.8E-09	6.4E+00	2.5E-04	7.6E-09	8.5E-06	7.8E-09	6.4E+00
Total	3.5E+00		5.7E-01		1.9E+03	2.6E-04		8.9E-06		7.0E+00	3.5E+00		5.7E-01		2.0E+03
Radiation-Related LCFs															
Hanford	2.8E-04	5.8E-09	1.1E-05	1.9E-08	2.2E-01	3.5E-09	7.3E-14	1.4E-10	2.4E-13	2.1E-04	2.8E-04	5.8E-09	1.1E-05	1.9E-08	0.2
LANL	1.3E-03	1.4E-07	9.8E-05	7.4E-08	1.8E-01	0	0	0	0	0	0	0	0	0	0.2
INEEL	1.3E-05	1.6E-09	3.1E-06	2.6E-09	3.6E-01	0	0	0	0	0	0	0	0	0	0.4
SRS	1.3E-04	1.2E-09	1.1E-04	9.8E-09	1.9E-02	0	0	0	0	0	0	0	0	0	1.9E-02
RFETS	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
ORNL	0	0	0	0	0	1.3E-07	3.8E-12	3.4E-09	3.1E-12	2.6E-03	1.3E-07	3.8E-12	3.4E-09	3.1E-12	2.6E-03
Total	1.8E-03		2.3E-04		7.8E-01	1.3E-07		3.5E-09		2.8E-03	1.8E-03		2.3E-04		7.8E-01
Chemicals - Cancer Incidence															
Hanford	2.2E-10	0	1.3E-10	2.6E-13	1.8E-07	5.1E-14	0	3.1E-14	6.3E-17	1.8E-10	2.2E-10	0	1.3E-10	2.6E-13	1.8E-07
LANL	1.2E-09	2.6E-13	5.9E-10	3.8E-13	1.9E-07	0	0	0	0	0	1.2E-09	2.6E-13	5.9E-10	3.8E-13	1.9E-07
INEEL	3.3E-09	4.6E-13	4.5E-09	3.7E-12	2.2E-05	0	0	0	0	0	3.3E-09	4.6E-13	4.5E-09	3.7E-12	2.2E-05
SRS	7.9E-12	0	3.8E-12	4.6E-15	5.6E-09	0	0	0	0	0	7.9E-12	0	3.8E-12	4.6E-15	5.6E-09
RFETS	0	0	0	0	0	0	0	0	0	0	0.0E+00	0	0.0E+00	0.0E+00	0.0E+00
ORNL	0	0	0	0	0	1.5E-07	0	4.8E-08	4.7E-11	3.2E-06	1.5E-07	7.5E-12	4.8E-08	4.7E-11	3.2E-06
Total	4.7E-09		5.2E-09		2.2E-05	1.5E-07		4.8E-08		3.2E-06	1.5E-07		5.4E-08		2.5E-05

LCFs may also be expected in waste treatment workers under all alternatives and options except No Action Alternative 2. There is a calculated expectation of 1.7 radiation-related LCFs under Action Alternative 2A and No Action Alternative 1A, 1.5 LCFs under Action Alternatives 1 and 3, and 1.3 LCFs under Action Alternative 2B and No Action Alternative 1B. About 1 LCF (0.8 and 0.6, respectively) may be expected under the Proposed Action and Action Alternative 2C, while only 0.4 LCF would be expected under No Action Alternative 2.

Human health impacts taken from the WM PEIS, on which the SEIS-II estimates are based, are presented in Tables B-20 through B-24.

Table B-20
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals for WM PEIS Decentralized Alternative

Site	Population	MEI	Non-Involved Worker Population	Non-Involved Worker Individual	Worker Population
Radiation Dose (rem or person-rem) [a]					
Hanford	4.6E-02	9.7E-07	2.30E-03	3.9E-06	3.10E+02
LANL	1.1E-01	1.1E-05	9.80E-03	7.4E-06	3.60E+02
INEEL	2.3E-03	2.8E-07	6.80E-04	5.8E-07	6.30E+02
SRS	1.5E-01	1.4E-06	1.60E-02	1.4E-05	1.70E+02
RFETS	1.9E-02	2.5E-07	9.40E-04	5.7E-07	1.90E+01
ORNL	1.6E-03	4.9E-08	5.40E-05	4.9E-08	6.80E+00
LLNL	6.9E-03	1.1E-07	3.40E-04	1.2E-07	1.40E+00
NTS	2.3E-07	5.9E-11	6.40E-07	2.9E-09	5.20E-01
ANL-E	4.0E-03	2.2E-08	2.10E-05	2.1E-08	2.20E+01
Totals	3.4E-01		3.0E-02		1.5E+03
Radiation-Related LCFs [a, b]					
Hanford	2.3E-05	4.9E-10	9.2E-07	1.6E-09	1.2E-01
LANL	5.5E-05	5.5E-09	3.9E-06	3.0E-09	1.4E-01
INEEL	1.2E-06	1.4E-10	2.7E-07	2.3E-10	2.5E-01
SRS	7.5E-05	7.0E-10	6.4E-06	5.6E-09	6.8E-02
RFETS	9.5E-06	1.3E-10	3.8E-07	2.3E-10	7.6E-03
ORNL	8.0E-07	2.5E-11	2.2E-08	2.0E-11	2.7E-03
LLNL	3.5E-06	5.5E-11	1.4E-07	4.8E-11	5.6E-04
NTS	1.2E-10	3.0E-14	2.6E-10	1.2E-12	2.1E-04
ANL-E	2.0E-06	1.1E-11	8.4E-09	8.4E-12	8.8E-03
Totals	1.7E-04		1.2E-05		6.1E-01
Hazardous Chemical Cancer Incidence [a]					
Hanford	8.4E-11	0	5.0E-11	9.8E-14	2.2E-08
LANL	6.8E-10	8.3E-14	3.4E-10	2.1E-13	3.5E-08
INEEL	2.3E-09	3.2E-13	3.0E-09	2.5E-12	9.2E-06
SRS	1.9E-11	0	9.5E-12	1.1E-14	4.2E-09
RFETS	4.8E-10	0	9.6E-11	4.9E-14	6.8E-09
ORNL	1.6E-07	8.1E-12	5.2E-08	5.0E-11	3.0E-06
LLNL	1.3E-07	2.8E-12	3.0E-08	8.9E-12	8.8E-07
NTS	3.8E-13	0	5.1E-12	2.1E-14	8.8E-10
ANL-E	1.5E-09	0	3.1E-11	2.8E-14	1.9E-09
Totals	3.0E-07		8.6E-08		1.3E-05

[a] Radiation doses and hazardous chemical cancer incidences are taken from the following tables in Volume II of the WM PEIS: Hanford II-5.3-2, II-5.3-4; LANL II-7.3-2, II-7.3-4; INEEL II-6.3-2, II-6.3-4; SRS II-16.3-2, II-16.3-4; RFETS II-14.3-2, II-14.3-4; ORNL II-10.3-2, II-10.3-4; LLNL II-8.3-2, II-8.3-4; NTS II-9.3-2, II-9.3-4; ANL-E II-2.3-2, II-2.3-4.
[b] Calculated from the radiation dose, using 5 E-4 per rem for population and MEI, and 4 E-4 per rem for NIW population, NIW individual, and worker population.

Table B-21
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals for WM PEIS Regionalized 1 Alternative

Site	Population	MEI	Non-Involved Worker Population	Non-Involved Worker Individual	Worker Population
Radiation Dose (rem or person-rem) [a]					
Hanford	8.2E-02	1.7E-06	4.0E-03	6.9E-06	3.3E+02
LANL	1.3E-01	1.4E-05	1.2E-02	9.0E-06	3.6E+02
INEEL	2.9E-03	3.6E-07	8.6E-04	7.4E-07	6.3E+02
SRS	2.7E-01	2.5E-06	2.9E-01	2.5E-05	1.9E+02
RFETS	3.0E-02	4.0E-07	1.5E-03	9.1E-07	1.9E+01
ORNL	1.7E-03	5.1E-08	5.7E-05	5.2E-08	6.9E+00
LLNL	7.3E-03	1.2E-07	3.6E-04	1.2E-07	1.4E+00
NTS	3.0E-07	7.8E-11	8.4E-07	3.9E-09	5.0E-01
ANL-E	3.5E-03	1.9E-08	1.8E-05	1.8E-08	2.2E+01
Totals	5.3E-01		3.1E-01		1.6E+03
Radiation-Related LCFs [b]					
Hanford	4.1E-05	8.5E-10	1.6E-06	2.8E-09	1.3E-01
LANL	6.5E-05	7.0E-09	4.8E-06	3.6E-09	1.4E-01
INEEL	1.5E-06	1.8E-10	3.4E-07	3.0E-10	2.5E-01
SRS	1.4E-04	1.3E-09	1.2E-04	1.0E-08	7.6E-02
RFETS	1.5E-05	2.0E-10	6.0E-07	3.6E-10	7.6E-03
ORNL	8.5E-07	2.6E-11	2.3E-08	2.1E-11	2.8E-03
LLNL	3.7E-06	6.0E-11	1.4E-07	4.8E-11	5.6E-04
NTS	1.5E-10	3.9E-14	3.4E-10	1.6E-12	2.0E-04
ANL-E	1.8E-06	9.5E-12	7.2E-09	7.2E-12	8.8E-03
Totals	2.6E-04		1.2E-04		6.2E-01
Hazardous Chemical Cancer Incidence [a]					
Hanford	1.3E-10	0	7.9E-11	1.6E-13	1.1E-07
LANL	9.5E-10	2.1E-13	4.7E-10	3.0E-13	1.5E-07
INEEL	2.3E-09	3.2E-13	3.1E-09	2.6E-12	1.5E-05
SRS	3.1E-11	0	1.5E-11	1.8E-14	2.2E-08
RFETS	6.3E-10	0	1.3E-10	6.5E-14	2.1E-08
ORNL	1.6E-07	8.1E-12	5.2E-08	5.0E-11	3.4E-06
LLNL	2.2E-07	4.8E-12	5.2E-08	1.5E-11	1.9E-06
NTS	1.1E-12	0	1.5E-11	6.2E-14	4.4E-09
ANL-E	2.9E-09	1.3E-14	6.0E-11	5.4E-14	2.8E-09
Totals	3.9E-07		1.1E-07		2.1E-05

[a] Radiation doses and hazardous chemical cancer incidences are taken from the following tables in Volume II of the WM PEIS: Hanford II-5.3-2, II-5.3-4; LANL II-7.3-2, II-7.3-4; INEEL II-6.3-2, II-6.3-4; SRS II-16.3-2, II-16.3-4; RFETS II-14.3-2, II-14.3-4; ORNL II-10.3-2, II-10.3-4; LLNL II-8.3-2, II-8.3-4; NTS II-9.3-2, II-9.3-4; ANL-E II-2.3-2, II-2.3-4.

[b] Calculated from the radiation dose, using 5 E-4 per rem for population and MEI, and 4 E-4 per rem for NIW population, NIW individual, and worker population.

Table B-22
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals for WM PEIS Regionalized 2 Alternative

Site	Population	MEI	Non-Involved Worker Population	Non-Involved Worker Individual	Worker Population
Radiation Dose (rem or person-rem) [a]					
Hanford	3.3E+02	6.8E-03	1.6E+01	2.8E-02	3.2E+02
LANL	1.3E+03	1.3E-01	1.2E+02	8.7E-02	3.4E+02
INEEL	1.5E+01	1.8E-03	4.4E+00	3.7E-03	5.9E+02
SRS	4.5E+00	4.2E-05	4.8E-01	4.2E-04	1.9E+02
RFETS	2.2E+02	3.0E-03	1.1E+01	6.7E-03	1.8E+01
ORNL	9.2E+01	2.8E-03	3.1E+00	2.8E-03	2.3E+02
LLNL	7.3E-03	1.2E-07	3.6E-04	1.2E-07	1.4E+00
NTS	3.0E-07	7.8E-11	8.4E-07	3.9E-09	5.0E-01
ANL-E	3.5E-03	1.9E-08	1.8E-05	1.8E-08	2.2E+01
Totals	2.0E+03		1.5E+02		1.7E+03
Radiation-Related LCFs [b]					
Hanford	1.7E-01	3.4E-06	6.4E-03	1.1E-05	1.3E-01
LANL	6.5E-01	6.5E-05	4.8E-02	3.5E-05	1.4E-01
INEEL	7.5E-03	9.0E-07	1.8E-03	1.5E-06	2.4E-01
SRS	2.3E-03	2.1E-08	1.9E-04	1.7E-07	7.6E-02
RFETS	1.1E-01	1.5E-06	4.4E-03	2.7E-06	7.2E-03
ORNL	4.6E-02	1.4E-06	1.2E-03	1.1E-06	9.2E-02
LLNL	3.7E-06	6.0E-11	1.4E-07	4.8E-11	5.6E-04
NTS	1.5E-10	3.9E-14	3.4E-10	1.6E-12	2.0E-04
ANL-E	1.8E-06	9.5E-12	7.2E-09	7.2E-12	8.8E-03
Totals	9.8E-01		6.2E-02		6.8E-01
Hazardous Chemical Cancer Incidence [a]					
Hanford	1.3E-10	0	7.5E-11	1.5E-13	1.9E-07
LANL	1.1E-09	1.3E-13	5.4E-10	3.3E-13	3.7E-07
INEEL	1.7E-09	2.3E-13	2.2E-09	1.9E-12	2.2E-05
SRS	2.9E-11	0	1.5E-11	1.7E-14	3.7E-08
RFETS	8.4E-10	0	1.7E-10	8.5E-14	6.2E-08
ORNL	1.1E-07	5.7E-12	3.7E-08	3.5E-11	5.1E-06
LLNL	2.2E-07	4.8E-12	5.2E-08	1.5E-11	1.9E-06
NTS	1.1E-12	0	1.5E-11	6.2E-14	4.4E-09
ANL-E	2.9E-09	1.3E-14	6.0E-11	5.4E-14	2.8E-09
Totals	3.4E-07		9.2E-08		3.0E-05

[a] Radiation doses and hazardous chemical cancer incidences are taken from the following tables in Volume II of the WM PEIS: Hanford II-5.3-2, II-5.3-4; LANL II-7.3-2, II-7.3-4; INEEL II-6.3-2, II-6.3-4; SRS II-16.3-2, II-16.3-4; RFETS II-14.3-2, II-14.3-4; ORNL II-10.3-2, II-10.3-4; LLNL II-8.3-2, II-8.3-4; NTS II-9.3-2, II-9.3-4; ANL-E II-2.3-2, II-2.3-4.

[b] Calculated from the radiation dose, using 5 E-4 per rem for population and MEI, and 4 E-4 per rem for NIW population, NIW individual, and worker population.

Table B-23
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals for WM PEIS Regionalized 3 Alternative

Site	Population	MEI	Non-Involved Worker Population	Non-Involved Worker Individual	Worker Population
Radiation Dose (rem or person-rem) [a]					
Hanford	3.3E+02	6.8E-03	1.6E+01	2.8E-02	3.2E+02
LANL	1.4E-01	1.5E-05	1.3E-02	9.7E-06	3.6E+02
INEEL	8.2E+01	1.0E-02	2.5E+01	2.1E-02	6.2E+02
SRS	4.5E+00	4.2E-05	4.8E-01	4.2E-04	1.9E+02
RFETS	2.4E-02	3.3E-07	2.1E-03	7.4E-07	7.4E+01
ORNL	9.2E+01	2.8E-03	3.1E+00	2.8E-03	2.3E+02
LLNL	7.3E-03	1.2E-07	3.6E-04	1.2E-07	1.4E+00
NTS	3.0E-07	7.8E-11	8.4E-07	3.9E-09	5.0E-01
ANL-E	3.5E-03	1.9E-08	1.8E-05	1.8E-08	2.2E+01
Totals	5.1E+02		4.5E+01		1.8E+03
Radiation-Related LCFs [b]					
Hanford	1.7E-01	3.4E-06	6.4E-03	1.1E-05	1.3E-01
LANL	7.0E-05	7.5E-09	5.2E-06	3.9E-09	1.4E-01
INEEL	4.1E-02	5.0E-06	1.0E-02	8.4E-06	2.5E-01
SRS	2.3E-03	2.1E-08	1.9E-04	1.7E-07	7.6E-02
RFETS	1.2E-05	1.7E-10	8.4E-07	3.0E-10	3.0E-02
ORNL	4.6E-02	1.4E-06	1.2E-03	1.1E-06	9.2E-02
LLNL	3.7E-06	6.0E-11	1.4E-07	4.8E-11	5.6E-04
NTS	1.5E-10	3.9E-14	3.4E-10	1.6E-12	2.0E-04
ANL-E	1.8E-06	9.5E-12	7.2E-09	7.2E-12	8.8E-03
Totals	2.5E-01		1.8E-02		7.3E-01
Hazardous Chemical Cancer Incidence [a]					
Hanford	1.3E-10	0	7.5E-11	1.5E-13	1.9E-07
LANL	1.9E-09	2.3E-13	9.4E-10	5.7E-13	1.7E-07
INEEL	1.7E-09	2.4E-13	2.3E-09	1.9E-12	3.3E-05
SRS	2.9E-11	0	1.5E-11	1.7E-14	3.7E-08
RFETS	1.9E-09	9.0E-15	2.3E-10	1.2E-13	2.6E-08
ORNL	1.1E-07	5.7E-12	3.7E-08	3.5E-11	5.1E-06
LLNL	2.2E-07	4.8E-12	5.2E-08	1.5E-11	1.9E-06
NTS	1.1E-12	0	1.5E-11	6.2E-14	4.4E-09
ANL-E	2.9E-09	1.3E-14	6.0E-11	5.4E-14	2.8E-09
Totals	3.4E-07		9.3E-08		4.0E-05

[a] Radiation doses and hazardous chemical cancer incidences are taken from the following tables in Volume II of the WM PEIS: Hanford II-5.3-2, II-5.3-4; LANL II-7.3-2, II-7.3-4; INEEL II-6.3-2, II-6.3-4; SRS II-16.3-2, II-16.3-4; RFETS II-14.3-2, II-14.3-4; ORNL II-10.3-2, II-10.3-4; LLNL II-8.3-2, II-8.3-4; NTS II-9.3-2, II-9.3-4; ANL-E II-2.3-2, II-2.3-4.

[b] Calculated from the radiation dose, using 5 E-4 per rem for population and MEI, and 4 E-4 per rem for NIW population, NIW individual, and worker population.

Table B-24
Human Health Impacts Associated with TRU Waste Treatment from Radionuclides and Chemicals for WM PEIS Centralized Alternative

Site	Population	MEI	Non-Involved Worker Population	Non-Involved Worker Individual	Worker Population
Radiation Dose (rem or person-rem) [a]					
Hanford	1.3E+00	2.7E-05	6.2E-02	1.1E-04	4.1E+02
LANL	1.4E-01	1.5E-05	1.3E-02	9.7E-06	3.6E+02
INEEL	3.2E-03	3.9E-07	9.4E-04	8.1E-07	6.1E+02
SRS	6.8E-02	6.4E-07	7.3E-03	6.4E-06	2.6E+02
RFETS	2.4E-02	3.3E-07	1.2E-03	7.4E-07	7.4E+01
ORNL	9.2E+01	2.8E-03	3.1E+00	2.8E-03	2.3E+02
LLNL	7.3E-03	1.2E-07	3.6E-04	1.2E-07	1.4E+00
NTS	3.0E-07	7.8E-11	8.4E-07	3.9E-09	5.0E-01
ANL-E	3.5E-03	1.9E-08	1.8E-05	1.8E-08	2.2E+01
WIPP	5.2E+02	1.4E-01	4.2E+01	1.6E-01	4.1E+01
Totals	6.1E+02		4.5E+01		2.0E+03
Radiation-Related LCFs [b]					
Hanford	6.5E-04	1.4E-08	2.5E-05	4.4E-08	1.6E-01
LANL	7.0E-05	7.5E-09	5.2E-06	3.9E-09	1.4E-01
INEEL	1.6E-06	2.0E-10	3.8E-07	3.2E-10	2.4E-01
SRS	3.4E-05	3.2E-10	2.9E-06	2.6E-09	1.0E-01
RFETS	1.2E-05	1.7E-10	4.8E-07	3.0E-10	3.0E-02
ORNL	4.6E-02	1.4E-06	1.2E-03	1.1E-06	9.2E-02
LLNL	3.7E-06	6.0E-11	1.4E-07	4.8E-11	5.6E-04
NTS	1.5E-10	3.9E-14	3.4E-10	1.6E-12	2.0E-04
ANL-E	1.8E-06	9.5E-12	7.2E-09	7.2E-12	8.8E-03
WIPP	2.6E-01	7.0E-05	1.7E-02	6.4E-05	1.6E-02
Totals	3.1E-01		1.8E-02		8.0E-01
Hazardous Chemical Cancer Incidence [a]					
Hanford	7.9E-11	0	4.7E-11	9.2E-14	7.1E-08
LANL	1.9E-09	2.3E-13	9.4E-10	5.7E-13	1.7E-07
INEEL	6.6E-09	9.1E-13	8.7E-09	7.3E-12	4.4E-05
SRS	1.4E-11	0	6.8E-12	8.0E-15	4.0E-09
RFETS	1.1E-09	9.0E-15	2.3E-10	1.2E-13	2.6E-08
ORNL	1.1E-07	5.7E-12	3.7E-08	3.5E-11	5.1E-06
LLNL	2.2E-07	4.8E-12	5.2E-08	1.5E-11	1.9E-06
NTS	1.1E-12	0	1.5E-11	6.2E-14	4.4E-09
ANL-E	2.9E-09	1.3E-14	6.0E-11	5.4E-14	2.8E-09
WIPP	5.5E-10	2.1E-13	2.8E-10	1.3E-12	4.6E-06
Totals	3.4E-07		9.9E-08		5.6E-05

[a] Radiation doses and hazardous chemical cancer incidences are taken from the following tables in Volume II of the WM PEIS: Hanford II-5.3-2, II-5.3-4; LANL II-7.3-2, II-7.3-4; INEEL II-6.3-2, II-6.3-4; SRS II-16.3-2, II-16.3-4; RFETS II-14.3-2, II-14.3-4; ORNL II-10.3-2, II-10.3-4; LLNL II-8.3-2, II-8.3-4; NTS II-9.3-2, II-9.3-4; ANL-E II-2.3-2, II-2.3-4; WIPP II-17.3-2, II-17.3-4.

[b] Calculated from the radiation dose, using 5 E-4 per rem for population and MEI, and 4 E-4 per rem for NIW population, NIW individual, and worker population.

B.6 REFERENCES CITED IN APPENDIX B

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APPENDIX C
AIR QUALITY

This appendix describes the methods used for analyzing potential impacts to air quality at the Waste Isolation Pilot Plant (WIPP) and the major treatment sites from routine emissions of nonradological air pollutants during normal operations of the facility. Pollutants addressed in this appendix include nitrogen dioxide (NO_2), sulfur dioxide (SO_2), particulate matter with an aerodynamic diameter of 10 microns or less (PM_{10}), carbon monoxide (CO), lead (Pb) and volatile organic compounds (VOC) as ozone precursors. Lead and ozone would not be expected to be released in amounts of concern at WIPP. In formulating inputs for air quality modeling, a series of simplifying conservative assumptions have been used and are identified.

C.1 MODELS

The Industrial Source Complex (ISC3) computer code is used to estimate the annual air quality impacts. The ISC3 code consists of a short-term model (ISCST3) and a long-term model (ISCLT3). The long-term model was used to estimate annual air quality impacts. The short-term model was not used because the hourly meteorological data required by the model were not available at the time of the analysis. The model uses steady-state Gaussian plume algorithms to estimate pollutant concentrations from a wide variety of sources associated with industrial complexes (EPA 1995c). The model is appropriate for either flat or rolling terrain, modeling domains with a radius of 50 kilometers (31 miles) or less from the point of release, and either urban or rural environments. The ISC3 code is approved by the U.S. Environmental Protection Agency (EPA) for specific regulatory applications and is designed for use on personal computers. Input requirements for the ISC3 code include source configuration and pollutant emission parameters. The user may define point, line, area, or volume sources. The ISCLT3 code uses a joint frequency distribution of wind direction, wind speed, and atmospheric stability to compute pollutant transport and dispersion. Plume rise, stack tip downwash, and building wake can be computed, and plume depletion by deposition taken into account.

To calculate the short-term (24 hours or less) criteria pollutant impacts, the SCREEN3 model is used. SCREEN3 is a screening model used to estimate short-term air pollutant concentrations, including estimates of maximum ground-level concentrations from a single source (EPA 1995b). The model uses a steady-state Gaussian plume algorithm to calculate the concentration from a single point, area, or volume source of simple geometry. The model can be applied to both simple and complex terrain for modeling domains out to 100 kilometers (62 miles). Input requirements for SCREEN3 include source configuration information and pollutant emission parameters. Plume rise, building wake downwash, and plume impaction on complex terrain can be computed. While specific meteorological values of wind speed and stability can be entered to calculate pollutant transport and diffusion, the model can also calculate worst-case maximum concentrations, examining a range of stability classes and wind speeds to identify the most conservative meteorological conditions. Output of the SCREEN3 model is 1-hour maximum concentration at specified distances. Adjustment factors can be applied to estimate concentrations for averaging periods up to 24 hours. Averaging times and their corresponding adjustment or multiplying factors are shown in Table C-1 (EPA 1992). The SCREEN3 model is approved by EPA for specific screening procedures and is designed to run on personal computers.

Table C-1
Multiplying Factors to Estimate Maximum Concentration at Various Averaging Times for a Given 1-hour Maximum Concentration

Averaging Time	Multiplying Factor
3 hours	0.9
8 hours	0.7
24 hours	0.4

C.2 RECEPTORS

Maximum ground-level pollutant concentrations for regulation-specific time periods are reported at the maximally impacted receptor location. To determine maximum short-term impacts for exposure periods from 1 to 24 hours, pollutant concentrations for receptors along the "Off Limits Area" boundary are reported. Receptors along the "Off Limits Area" boundary are used because points along this boundary are the closest unrestricted access points to a member of the public. For long-term impacts, pollutant concentrations for receptors along the Land Withdrawal Boundary are reported. Points within the Land Withdrawal Boundary were not considered because of the limited time any member of the public would spend at an on-site location over the course of a year.

ISCLT3 model runs were done using a Cartesian grid of receptors spaced at 300-meter (984-foot) intervals along the Land Withdrawal Boundary. The model runs indicated that the maximum long-term concentrations would occur at a point approximately 3,000 meters (9,840 feet) north of the source. The maximum short-term concentrations for a ground level source would be at the closest point on the "Off Limits Area" boundary to the source, approximately 1,200 meters (3,940 feet) for the excavation exhaust stack.

C.3 SOURCE TERMS AND IMPACTS

The increase in airborne concentration of criteria pollutants is assumed to result from the routine operation of WIPP. Principal emission sources of particulates would be from fugitive salt dust, the excavation and disposal of salt, and fuel combustion from backup diesel-powered electrical generators and excavation and support equipment. Emissions of particulates are conservatively assumed to be emitted entirely as PM_{10}. Principal sources of NO_2, SO_2, and CO are from fuel combustion.

In all but one case, pollutants are assumed to be released from a point source. The exception to this is that wind erosion of the salt pile is assumed to come from an area source with a center located 360 meters (1,180 feet) north of the exhaust stack (see Figure 2-3). Plume rise is calculated for emissions from two back-up diesel generators and emissions out of the exhaust stack using stack parameters. Stack parameters for the generator (WEC 1993) and the exhaust shaft (DOE 1996) are given in Table C-2. Building wake downwash is not used in the model runs; however, there should be little difference between the concentration with and without building wake effects due to the large distances to receptors.

Table C-2
Stack Parameters for the Back-up Generator and Exhaust Shaft

Parameter	Generator	Excavation Exhaust
Stack Height	4.57 meters	8.2 meters
Stack Diameter	0.305 meter	4.4 meters
Stack Temperature	619.3 Kelvin	288 Kelvin
Velocity of Gas	10.18 meters per second	4.66 meters per second

To calculate the annual and 24-hour impacts under the Proposed Action, the following operating schedule was used: (1) excavation operations performed in two 8-hour shifts per day, 5 days per week, 52 weeks per year; (2) contact-handled (CH) transuranic (TRU) waste disposal operations performed in two 8-hour shifts per day, 4 days per week, 52 weeks per year; (3) remote-handled (RH) TRU waste disposal operations performed in two 8-hour shifts per day, 4 days per week, 52 weeks per year; and (4) maintenance performed in one 8-hour shift per week, 52 weeks per year. This same schedule is assumed for Action Alternatives 1, 2, and 3. To calculate annual pollutant concentrations using the ISCLT3 model, a joint frequency distribution of wind speed, wind direction, and atmospheric stability data from Carlsbad, New Mexico, for the years 1990 through 1994 was used. A description of the meteorological data is found in Appendix F, Human Health.

No meteorological input is required for estimating the short-term averaged concentrations using the SCREEN3 model. SCREEN3 estimates the maximum concentration by examining a range of wind speed and stability classes.

C.3.1 Salt Dust Emissions

Salt dust emissions would result from wind erosion of the salt piles, emissions of salt through the ventilation system exhaust, and dust released from transferring the salt from the repository to the salt storage pile. Salt dust emissions were estimated using a ground-level release, a 1-meter-(3.3 feet)-per-second wind, and an atmospheric stability class of F (stable).

Fugitive dust emissions from the salt pile have been conservatively estimated to be 6.44 kilograms per hectare (5.75 pounds per acre) per day of particulates (Tillman 1988b). The total area of the active salt pile for all alternatives except No Action Alternatives 1 and 2, because the salt piles would be disposed of during decommissioning, is assumed to be 12 hectares (30 acres) (DOE 1980). To estimate the annual emission, it is assumed that daily emission of fugitive dust is the same for each day of the year. The estimated daily emission rate is conservative because it does not take into account crusting, which would greatly reduce salt emission, and it assumes that the entire pile contains loose salt available for resuspension, when actually only a small portion of the pile would contain loose salt. The annual emission rate is also conservative. It is assumed that fugitive dust emissions occur each day when, in reality, fugitive dust emissions would only occur during high wind events.

The emission of particulates through the ventilation system is assumed to be 1.45 kilograms (3.20 pounds) per hour (Tillman 1988a). Emission of salt through the ventilation system is assumed to be continuous during an entire 8-hour shift.

The dust released from transferring the salt from the repository to the storage pile is assumed to be 10 grams (0.36 ounces) per ton of salt moved to the pile (DOE 1980). On average, the amount of salt brought to the surface in all of the alternatives except No Action Alternatives 1 and 2 is assumed to be 5.4×10^5 kilograms (600 tons) per day and 1.4×10^8 kilograms (1.6×10^5 tons) per year (Ashford 1996).

Table C-3 summarizes the source term for PM_{10} emissions of salt dust for all alternatives except No Action Alternatives 1 and 2. Estimated impacts on air quality from salt dust emissions are shown in Table C-4.

Table C-3
Source Term for Calculating Salt Dust Emission Impacts for All Alternatives Except No Action Alternatives 1 and 2

Source	Averaging Time	Mass of Pollutant	Emission Rate
Wind Erosion	Annual	29,000 kilograms per year	7.5E-6 grams per square meter per second
	24 hour	78 kilograms per day	7.5E-6 grams per square meter per second
Emission from Ventilation	Annual	6,000 kilograms per year	0.19 grams per second
	24 hour	23 kilograms per day	0.27 grams per second
Emission during Transfer	Annual	1,600 kilograms per year	0.049 grams per second
	24 hour	6 kilograms per day	0.069 grams per second

Table C-4
Estimated Impacts on Air Quality from Salt Dust Emissions for All Alternatives Except No Action Alternatives 1 and 2

Pollutant	Averaging Time	Maximum Concentration (micrograms per cubic meter)	Regulatory Limit (micrograms per cubic meter)	Percent of Regulatory Limit
PM_{10}	Annual	0.65	50 [a]	1.3
	24 hour	78	150 [a]	52

[a] Primary Federal Ambient Air Quality Standard (40 CFR Part 50)

C.3.2 Backup Generators

Four ambient air pollutants, NO_2, SO_2, CO, and PM_{10}, were assumed to be emitted by two 1,500-horsepower back-up diesel generators. A permit has been obtained by WIPP for the operation of these two back-up diesel generators (NMED 1993). Permit conditions have remained unchanged since 1993, with limits on the emissions of NO_2, SO_2, CO, and particulates. The generators are allowed to run, at most, 480 hours per year. Table C-5 summarizes the annual and hourly emission rate limits for the back-up diesel generators.

Table C-5
Maximum Annual and Hourly Emission Rates from Two Back-up Generators

Pollutant	Averaging Time	Maximum Emission Rate
NO_2	Annual	10,000 kilograms per year
	Hourly	21 kilograms per hour
SO_2	Annual	730 kilograms per year
	Hourly	1.4 kilograms per hour
CO	Annual	2,220 kilograms per year
	Hourly	4.6 kilograms per hour
PM_{10}	Annual	730 kilograms per year
	Hourly	1.5 kilograms per hour

The source term for long-term impacts is assumed to be the maximum allowable annual emission rate of each pollutant as defined in the permit and the maximum allowable hourly emission rate as defined by the permit for short-term impacts. In addition, the backup generators were assumed to run for 6 hours per day for the short-term impacts. The maximum normal operating schedule is 6 hours per month, with 2 hours per month dedicated to ensure proper operation and the remaining 4 hours per month dedicated to periodic operational maintenance on the generators (WEC 1993).

Table C-6 summarizes the source terms for the emission of the four criteria pollutants for all alternatives except No Action Alternatives 1 and 2. The two no action alternatives are not included because they do not assume the use of the generators. Estimated potential air quality impacts are presented in Table C-7.

Table C-6
Source Term for Calculating Back-up Diesel Generator Emission Impacts

Pollutant	Averaging Time	Mass of Pollutant (kilograms) per Averaging Time	Emission Rate (grams per second)
NO_2	Annual	10,000	0.32
	24 hour	130	1.5
SO_2	Annual	730	0.023
	24 hour	8.4	0.098
	3 hour	1.4	0.39
CO	8 hour	27	0.95
	1 hour	4.6	1.3
PM_{10}	Annual	730	0.023
	24 hour	9.0	0.10

Table C-7
Criteria Pollutant Air Quality Impacts from the Emissions of Two Back-up Generators

Pollutant	Averaging Time	Maximum Concentration (micrograms per cubic meter)	Regulatory Limit (micrograms per cubic meter)	Percent of Regulatory Limit
NO_2	Annual	0.12	84 [a]	0.15
	24 hour	54	168 [a]	32
SO_2	Annual	0.0088	47 [a]	0.019
	24 hour	3.6	234 [a]	1.5
	3 hour	32	1,170 [b]	2.8
CO	8 hour	62	8,900 [a]	0.71
	1 hour	120	13,400 [a]	0.87
PM_{10}	Annual	0.0088	50 [c]	0.018
	24 hour	3.8	150 [c]	2.6

[a] New Mexico Ambient Air Quality Standard (AQCR 201) corrected for altitude
[b] Secondary Federal Ambient Air Quality Standard (40 CFR Part 50) corrected for altitude
[c] Primary Federal Ambient Air Quality Standard (40 CFR Part 50)

C.3.3 Above-Ground Diesel Equipment

In addition to the two back-up generators, the diesel equipment used on the surface during WIPP operations includes: one diesel dump truck used to haul the salt from the repository to the storage pile, a fire water pump, and an emergency hoist (Hollen 1996). All other equipment on the surface is electric.

To estimate the emissions from the diesel dump truck, emission rates for the criteria pollutants for heavy-duty construction equipment found in EPA's *Supplement A to Compilation of Air Pollution Emission Factors, Volume 2: Mobile Sources* (EPA 1991, Table 2-7.1) were used. These emission rates are summarized in Table C-8. Pollutant emissions are estimated using the emission rate for diesel industrial engines found in EPA's *Compilation of Air Pollution Emission Factors, Volume 1: Stationary Point and Area Sources, 5th Edition* (EPA 1995a, Table 3.3-2) for the fire water pump and emergency hoist. These emission rates are summarized in Table C-9.

Truck operation is assumed to be 40 percent of an 8-hour shift, to account for start up and shutdown of the excavation operation during a shift, operator break and lunch periods, and the time when salt is being loaded in the truck. For the other diesel equipment used during routine operations, it is assumed that the fire water pump (188 horsepower) is in operation for 30 minutes per week and the emergency hoist (115 horsepower) is in operation for 30 minutes per month (Hollen 1996).

Table C-10 summarizes the source term for emissions of the above-ground diesel equipment for all alternatives except No Action Alternatives 1 and 2, which do not assume operation of any of the above-ground diesel equipment. The potential air quality impacts are shown in Table C-11.

Table C-8
Estimated Emission Rates for Diesel Dump Truck

Pollutant	Estimated Emission Rate (grams per hour of operation)
NO_2	1,900
SO_2	210
CO	820
PM_{10}	120

Table C-9
Estimated Emission Rates for Industrial Diesel Engines

Pollutant	Estimated Emission Rate (pounds per horsepower-hour)
NO_2	0.031
SO_2	0.00205
CO	0.00668
PM_{10}	0.00220

Table C-10
Source Term for Calculating Impacts from Surface Diesel Equipment

Pollutant	Averaging Time	Mass of Pollutant (kilograms) per Averaging Time	Emission Rate (grams per second)
NO_2	Annual	3,200	0.10
	24 hour	14	0.16
SO_2	Annual	350	0.011
	24 hour	1.5	0.017
	3 hour	0.76	0.070
CO	8 hour	3.1	0.11
	1 hour	1.3	0.35
PM_{10}	Annual	200	0.0063
	24 hour	0.89	0.010

Table C-11
Criteria Pollutant Air Quality Impacts from Surface Diesel Equipment Emissions

Pollutant	Averaging Time	Maximum Concentration (micrograms per cubic meter)	Regulatory Limit (micrograms per cubic meter)	Percent of Regulatory Limit
NO_2	Annual	0.050	84 [a]	0.060
	24 hour	33	168 [a]	20
SO_2	Annual	0.0057	47 [a]	0.012
	24 hour	3.4	234 [a]	1.5
	3 hour	32	1,170 [b]	2.7
CO	8 hour	38	8,900 [a]	0.43
	1 hour	180	13,400 [a]	1.3
PM_{10}	Annual	0.0031	50 [c]	0.0062
	24 hour	2.1	150 [c]	1.4

[a] New Mexico Ambient Air Quality Standard (AQCR 201) corrected for altitude
[b] Secondary Federal Ambient Air Quality Standard (40 CFR Part 50) corrected for altitude
[c] Primary Federal Ambient Air Quality Standard (40 CFR Part 50)

C.3.4 Underground Diesel Equipment

A variety of diesel equipment is used in underground excavation, disposal, and maintenance operations, although only a few pieces of equipment are in operation at any one time (WEC 1995). To estimate the impacts from underground equipment, the pollutant emissions are estimated using the emission rates for diesel industrial engines found in EPA's *Compilation of Air Pollution Emission Factors, Volume 1: Stationary Point and Area Sources, 5th Edition* (EPA 1995a, Table 3.3-2). These emission rates are summarized in Table C-9 above.

The pollution emission for the underground diesel equipment depends upon the usage, the rated power available, and the load factor (the power actually used divided by the power available for each engine). Underground diesel equipment usage is assumed to be 40 percent of an 8-hour shift, to account for start up and shutdown of the excavation operation during a shift, operator break and lunch periods, and the time in which the equipment is not in use. To estimate the power available, a typical number and type of equipment are assumed for the excavation operation, CH-TRU waste handling, RH-TRU waste handling, and maintenance. Table C-12 shows the estimated equipment used in normal excavation, disposal, and maintenance operations (Roland 1996) along with the total available power for equipment used during a particular operation (WEC 1995). With the exception of impacts of less than three hours, a load factor of 40 percent is assumed because the equipment will normally not be running at full power. For impacts of less than three hours, the equipment is assumed to be running at full power and a load factor of 100 percent is used.

Table C-13 summarizes the source term for pollutant emissions from underground equipment. The potential air quality impacts resulting from underground equipment use are shown in Table C-14. The short-term impacts are highly conservative because they assume that the emissions are directly from the stack and do not enter into the repository where they would be diluted.

Table C-12
Diesel Equipment and Total Available Power for Underground Operations

Operation	Equipment Used	Total Available Power (horsepower)
Excavation	4 Trucks, 2 Load Haul Dumps	709
CH-TRU Waste Disposal	2 CH-TRU Waste Transports and 1 6-ton Forklift	298
RH-TRU Waste Disposal	1 40-ton Forklift and 1 20-ton Forklift	416
Maintenance	Scissor Lift, Arc Welder, 2 Tractors, Lube Truck	255

Table C-13
Source Term for Calculating Impacts from Underground Diesel Equipment Emissions

Pollutant	Averaging Time	Mass of Pollutant (kilograms) per Averaging Time	Emission Rate (grams per second)
NO_2	Annual	12,000	0.39
	24 hour	51	0.59
SO_2	Annual	810	0.026
	24 hour	3.4	0.039
	3 hour	1.59	0.15
CO	8 hour	5.5	0.19
	1 hour	4.3	1.2
PM_{10}	Annual	870	0.0275
	24 hour	3.6	0.042

Table C-14
Criteria Pollutant Air Quality Impacts from Underground Diesel Equipment Emissions

Pollutant	Averaging Time	Maximum Concentration (micrograms per cubic meter)	Regulatory Limit (micrograms per cubic meter)	Percent of Regulatory Limit
NO_2	Annual	0.11	84 [a]	0.13
	24 hour	23	168 [a]	14
SO_2	Annual	0.0073	47 [a]	0.015
	24 hour	1.5	234 [a]	0.63
	3 hour	13	1,170 [b]	1.1
CO	8 hour	13	8,900 [a]	0.14
	1 hour	110	13,400 [a]	0.85
PM_{10}	Annual	0.0078	50 [c]	0.016
	24 hour	1.6	150 [c]	1.1

[a] New Mexico Ambient Air Quality Standard (AQCR 201) corrected for altitude
[b] Secondary Federal Ambient Air Quality Standard (40 CFR Part 50) corrected for altitude
[c] Primary Federal Ambient Air Quality Standard (40 CFR Part 50)

C.3.5 Decommissioning of WIPP

The plans for decommissioning WIPP are described in Section 3.1.3.5. The potential air quality impacts would mainly come from construction of the berm and permanent markers, dismantling the above-ground structures (approximately 8 hectares [20 acres] of buildings outside the surface marker area), and reclamation of the salt stored on the surface. The impacts would be from fugitive dust due to construction operations and emissions from heavy-duty construction equipment.

Impacts from the dismantling of the above-ground building and construction of the berm and permanent markers would be similar to the impacts described in the *Final Environmental Impact Statement for the Waste Isolation Pilot Plant* (FEIS), which describes the construction impacts for building the WIPP site. The same type of construction equipment would be used. Although the area of the berm varies between the Proposed Action and each of the action alternatives, the yearly usage of equipment and the number of acres of land disturbed by construction in a year is assumed to be similar. No Action Alternatives 1 and 2 would not require the building of the berm or setting up the permanent markers and would have smaller air quality impacts than those for the other alternatives.

Air quality impacts would also be due to reclaiming the stored salt on the surface. Stored salt could be used to close the shafts of the repository or act as a base for the berm. Emissions of criteria pollutants and the resulting air quality impacts from the shaft closure operations would be similar to those described in the FEIS. These include emissions from a mined-salt drier, loading the salt into a crusher, transporting the salt into the repository, and salt emitted out of the ventilation system. Impacts for using the reclaimed salt as the base to the berm would be similar to those for using the reclaimed salt to close the shafts since drying, crushing, and transport of the salt would be required for both. As the existing salt pile is smaller than the 12-hectare (30-acre) storage pile assumed under the Proposed Action and the action alternatives, emission and impacts from reclaiming the salt pile in No Action Alternatives 1 and 2 would be less than for the other alternatives.

C.3.6 Volatile Organic Compounds

VOC emissions were estimated using drum headspace volatile emission data (detailed in Appendix F). Emission rates for all of the individual VOCs listed were summed, and the one panel-equivalent of CH-TRU waste and RH-TRU waste (about 85,000 drum-equivalents for all alternatives) was assumed to be continuously releasing volatiles. The estimated total VOC emissions were 540 kilograms (1,200 pounds) per year. Because VOCs are ozone precursors, they were compared to the ozone release limit of 40 tons (80,000 pounds) per year in the New Mexico Prevention of Significant Deterioration (PSD) regulations (Air Quality Control Regulation [AQCR]). These emissions did not vary greatly among the Proposed Action, Action Alternative 1, and Action Alternative 3. No VOCs would be present in the thermally treated waste under Action Alternative 2 and No Action Alternative 1.

C.3.7 Criteria Air Pollutants at Generator-Storage and Treatment Sites

U.S. Department of Energy (DOE or the Department) evaluated air quality impacts at each proposed waste management site based on estimated increases in emissions of the six criteria air pollutants. Pollutant emission estimates were made for the construction and operation and maintenance activities of the waste management facilities.

In those areas where air pollution standards are not met (nonattainment areas), activities that introduce new sources of emissions are regulated under the General Conformity Rule. In areas where air pollution standards are met (attainment areas), regulations for the PSD of ambient air quality apply. In both cases, a permit is required for sources which will result in emissions equal to or greater than the limits set by pertinent regulations.

Criteria air pollutants can be emitted from construction equipment and from vehicles used to drive to the construction site; both are considered to be mobile sources. Criteria air pollutants are also emitted during operation and maintenance of waste management facilities (stationary sources) and by vehicles that are driven to the facility or used to transport waste (mobile sources). DOE evaluated air quality impacts for these pollutants at each site by comparing the estimated increases in tons per year to the allowable emission limits under either the General Conformity Rules in nonattainment areas or the PSD regulations in attainment areas.

Table C-15 shows the percent of standard/guidelines for emissions of criteria air pollutants during operation and maintenance at nine of the ten major generator-storage sites. Data are taken from the *Final Waste Management Programmatic Environmental Impact Statement* (WM PEIS) (DOE 1997). Data for the Mound Plant were not included in the WM PEIS.

Table C-16 shows the percent of the General Conformity Rule for emissions of criteria air pollutants during construction at four of the ten major generator-storage sites, where data are available. Data are taken from the WM PEIS (DOE 1997).

Table C-15
Percent of Standard/Guidelines for Criteria Air Pollutants
During Operations and Maintenance

Pollutant	Proposed Action	Action Alternative 1	Action Alternative 2A	Action Alternative 2B	Action Alternative 2C	Action Alternative 3	No Action Alternative 1A	No Action Alternative 1B	No Action Alternative 2
Argonne National Laboratory-East									
CO	0	0	0	0	0	0	0	0	0
NO_2	9	9	8	8	8	5	8	8	9
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	0	0	0	0	0	0	0	0	0
SO_2	0	0	0	0	0	0	0	0	0
VOC	5	5	5	5	5	5	5	5	5
Hanford Site									
CO	0	0	0	0	0	0	0	0	0
NO_2	2	2	2	2	1	2	2	2	2
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	1	1	2	2	1	1	2	2	1
SO_2	0	0	0	0	0	0	0	0	0
VOC	0	0	0	0	0	0	0	0	0
Idaho National Engineering and Environmental Laboratory									
CO	0	0	0	1	0	0	0	1	0
NO_2	1	1	2	4	0	2	2	4	1
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	1	1	10	17	0	1	10	17	1
SO_2	0	0	4	8	0	0	4	8	0
VOC	0	0	0	0	0	0	0	0	0
Lawrence Livermore National Laboratory									
CO	8	8	9	9	9	6	9	9	8
NO_2	2	2	2	2	2	1	2	2	2
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	0	0	0	0	0	0	0	0	0
SO_2	0	0	0	0	0	0	0	0	0
VOC	2	2	2	2	2	1	2	2	2
Los Alamos National Laboratory									
CO	0	0	0	0	0	0	0	0	0
NO_2	1	1	2	0	0	1	2	0	1
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	0	0	5	0	0	0	5	0	0
SO_2	0	0	2	0	0	0	2	0	0
VOC	0	0	0	0	0	0	0	0	0
Nevada Test Site									
CO	3	3	0	0	0	0	0	0	3
NO_2	0	0	0	0	0	0	0	0	0
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	0	0	0	0	0	0	0	0	0
SO_2	0	0	0	0	0	0	0	0	0
VOC	0	0	0	0	0	0	0	0	0
Oak Ridge National Laboratory									
CO	0	0	0	0	0	0	0	0	0
NO_2	1	1	1	1	1	0	1	1	1
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	0	0	2	2	2	0	2	2	0
SO_2	0	0	1	1	1	0	1	1	0
VOC	0	0	0	0	0	0	0	0	0
Rocky Flats Environmental Technology Site									
CO	17	17	24	5	5	20	24	5	17
NO_2	3	3	5	1	1	4	5	1	3
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	0	0	0	0	0	0	0	0	0
SO_2	0	0	1	0	0	0	1	0	0
VOC	4	4	6	1	1	5	6	1	4
Savannah River Site									
CO	0	0	0	0	0	0	0	0	0
NO_2	1	1	2	2	0	1	2	2	1
Pb	0	0	0	0	0	0	0	0	0
PM_{10}	0	0	9	9	0	1	9	9	0
SO_2	0	0	4	4	0	0	4	4	0
VOC	0	0	0	0	0	0	0	0	0

Source: WM PEIS (DOE 1997)

Table C-16
Percent of General Conformity Rule for Criteria Air Pollutants During Construction [a]

Pollutant	Proposed Action	Action Alternative 1	Action Alternative 2A	Action Alternative 2B	Action Alternative 2C	Action Alternative 3	No Action Alternative 1A	No Action Alternative 1B	No Action Alternative 2
Argonne National Laboratory-East									
CO	-	-	-	-	-	-	-	-	-
NO_2	49	49	34	34	34	40	34	34	49
Pb	-	-	-	-	-	-	-	-	-
PM_{10}	1	1	1	1	1	0	1	1	1
SO_2	-	-	-	-	-	-	-	-	-
VOC	17	17	8	8	8	16	8	8	17
Lawrence Livermore National Laboratory									
CO	13	13	20	20	20	9	20	20	13
NO_2	7	7	10	10	10	3	10	10	7
Pb	0	0	0	0	0	0	0	0	-
PM_{10}	0	0	0	0	0	0	0	0	-
SO_2	0	0	0	0	0	0	0	0	-
VOC	4	4	5	5	5	2	5	5	4
Nevada Test Site									
CO	5	5	0	0	0	0	0	0	5
NO_2	-	-	-	-	-	-	-	-	-
Pb	-	-	-	-	-	-	-	-	-
PM_{10}	0	0	0	0	0	0	0	0	0
SO_2	-	-	-	-	-	-	-	-	-
VOC	-	-	-	-	-	-	-	-	-
Rocky Flats Environmental Technology Site									
CO	19	19	29	7	7	20	29	7	19
NO_2	11	11	15	4	4	11	15	4	11
Pb	-	-	-	-	-	-	-	-	-
PM_{10}	1	1	1	0	0	1	1	0	1
SO_2	-	-	-	-	-	-	-	-	-
VOC	6	6	8	2	2	6	8	2	6

[a] Dashed line indicates that emissions of this criteria pollutant are assumed to be negligible.

Source: WM PEIS (DOE 1997)

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APPENDIX D
LIFE-CYCLE COSTS AND ECONOMIC IMPACTS

This appendix discusses the technical approach and sources of information used in the estimation of life-cycle costs and economic impacts of *Waste Isolation Pilot Plant Disposal Phase Supplemental Environmental Impact Statement* (SEIS-II) alternatives.

D.1 LIFE-CYCLE COSTS

Life-cycle costs were determined in three areas: waste management facility costs at the large waste consolidation sites, waste transportation costs to the Waste Isolation Pilot Plant (WIPP) site, and WIPP operations costs.

Appendix J compares the estimates of the SEIS-II transuranic (TRU) waste volumes with those from the *The National Transuranic Waste Management Plan* (DOE 1996). These estimates are lower than those used in the SEIS-II analysis. Since costs depend in large part on waste volume, the costs could be lower.

D.1.1 Waste Treatment Facility Costs

The *Final Waste Management Programmatic Environmental Impact Statement* (WM PEIS) cost estimates for waste management facilities include the following major cost elements (DOE 1997):

- Preoperations costs - These include the costs of technology site adaptation, including bench tests and demonstrations; statutory and regulatory permitting; plant setup costs; and related generic design, project management, and contingencies.

- Facility construction costs - These include the costs of buildings, equipment, and related design; construction and project management; and contingencies.

- Operations and maintenance costs - These include the costs of annual operations, maintenance, utilities, contractor supervision and overhead, and related project management and contingencies.

- Decontamination and decommissioning costs - These include the costs of demolition of facilities, environmental closure, post closure and monitoring activities.

The WM PEIS also provided an equivalent accounting or breakdown of waste management facility costs on the basis of waste management functions performed at sites, including the following:

- Waste retrieval and characterization - These include the costs to retrieve and characterize the constituents prior to and following treatment.

- Waste treatment costs - These include the costs to build, operate and maintain waste treatment facilities.

- Waste storage costs - These include the costs to build, operate, and maintain storage capacity at the large waste consolidation sites.

The cost information provided in data tables in the WM PEIS provided a benchmark for estimating waste management facility costs for the SEIS-II alternatives. Tables D-1 through D-8 present these estimates. Costs were calculated by adjusting the site costs provided in the WM PEIS (Tables II-2.3-12 through II-17.3-12) by the ratio of SEIS-II waste volumes relative to the WM PEIS waste volumes.

As an example, consider the case of the Proposed Action reported in Table D-1. In the WM PEIS, the Hanford Site (Hanford) would process a total of 25,300 cubic meters (895,000 cubic feet) of waste, whereas in SEIS-II, it would process a total of 86,900 cubic meters (3,070,000 cubic feet), or 3.43 times the reported WM PEIS volume. Accordingly, the adjusted waste management facility costs at Hanford under the Proposed Action are approximately $6.2 billion -- the product of the original WM PEIS cost estimate of $1.81 billion and the volume adjustment factor -- that is, $1.81 x 3.43 = $6.2 billion (in 1994 dollars). The volume adjustment factors corresponding to the various SEIS-II Alternatives are shown in Table D-9.

This adjustment process was applied to each site for the various SEIS-II alternatives. Because WM PEIS does not contain cost information for the Mound Plant (Mound), its benchmark cost estimate reflects the average costs of other sites having relatively small waste volumes.

Table D-1
Waste Treatment Facility Costs for the Proposed Action in Millions of 1994 Dollars [a]

Cost Components	Life-Cycle Component				Functional Area			Costs [b]
	Preoperations	Construction	Operations and Maintenance	Decontamination and Decommissioning	Retrieval and Characterization	Treatment	Storage	
WM PEIS Decentralized Alternative Site Costs								
INEEL/ANL-W	83	419	725	461	930	705	54	1,688
Hanford	110	403	960	338	584	1,188	39	1,810
SRS	30	134	251	90	15	457	33	505
LANL	50	237	397	234	454	438	27	919
RFETS	21	95	227	34	46	311	20	377
ORNL	33	151	286	81	109	415	28	551
Mound	12	37	62	12	12	99	12	124
LLNL	17	63	132	21	0	221	14	233
NTS	6	27	46	17	0	84	12	96
ANL-E	25	89	183	33	0	308	22	330
Total	---	---	---	---	---	---	---	6,633
SEIS-II Proposed Action Alternative Site Costs								
INEEL/ANL-W	67	338	584	371	749	568	44	1,361
Hanford	378	1,384	3,298	1,161	2,006	4,081	134	6,221
SRS	21	95	178	64	11	324	23	358
LANL	96	455	762	449	871	840	52	1,763
RFETS	37	166	398	60	81	545	35	661
ORNL	67	308	584	165	223	848	57	1,128
Mound	1	7	13	4	5	19	1	25
LLNL	12	44	93	15	0	155	10	165
NTS	6	28	48	18	0	87	12	99
ANL-E	4	13	28	5	0	48	3	51
Total	689	2,838	5,986	2,312	3,946	7,515	371	11,832
Total Discounted at 4.1 Percent	N/A	N/A	N/A	N/A	N/A	N/A	N/A	6,560

[a] Volumes and dollars have been rounded. Actual totals may differ due to rounding.
[b] "Costs" equal the sum of the costs for Preoperations, Construction, Operations and Maintenance, and Decontamination and Decommissioning and are also equal to the sum of the costs for Retrieval and Characterization, Treatment, and Storage.

N/A = Not Applicable

Table D-2
Waste Treatment Facility Costs for Action Alternative 1 in Millions of 1994 Dollars [a]

Cost Components	Life-Cycle Component				Functional Area			Costs [c]
	Preoperations	Construction	Operations and Maintenance	Decontamination and Decommissioning	Retrieval and Characterization	Treatment	Storage	
WM PEIS Decentralized Alternative Site Costs								
INEEL/ANL-W	83	419	725	461	930	705	54	1,688
Hanford	110	403	960	338	584	1,188	39	1,811
SRS	30	134	251	90	15	457	33	505
LANL	50	237	397	234	454	438	27	918
RFETS	21	95	227	34	46	311	20	377
ORNL	33	151	286	81	109	415	28	551
Mound	12	37	62	12	12	99	12	124
LLNL	17	63	132	21	0	221	14	234
NTS	6	27	46	17	0	84	12	96
ANL-E	25	89	183	33	0	308	22	330
Total	---	---	---	---	---	---	---	6,634
SEIS-II Action Alternative 1 Site Costs								
INEEL/ANL-W [b]	189	956	1,876	1,052	2,122	1,609	347	4,078
Hanford [b]	655	2,398	6,132	2,011	3,475	7,069	652	11,196
SRS	30	134	250	90	15	455	33	503
LANL	159	754	1,263	744	1,444	1,393	86	2,923
RFETS	37	166	398	60	81	545	35	661
ORNL [b]	94	430	867	231	311	1,183	132	1,626
Mound	1	6	13	4	5	19	1	25
LLNL	2	7	16	2	0	26	2	28
NTS	12	53	90	33	0	165	24	189
ANL-E	12	43	89	16	0	150	11	161
Total	1,191	4,947	10,994	4,243	7,453	12,614	1,323	21,390
Total Discounted at 4.1 Percent	N/A	N/A	N/A	N/A	N/A	N/A	N/A	11,862

[a] Volumes and dollars have been rounded. Actual totals may differ due to rounding.
[b] Lag storage costs are included for Hanford ($420 million), INEEL ($223 million), and ORNL ($52 million).
[c] "Costs" equal the sum of the costs for Preoperations, Construction, Operations and Maintenance, and Decontamination and Decommissioning and are also equal to the sum of the costs for Retrieval and Characterization, Treatment, and Storage.

N/A = Not Applicable

The adjustment process described above is appropriate provided that linear relationships exist between costs and waste volumes and that treatment rates considered for the SEIS-II alternatives fall within the range of operations considered in the WM PEIS. To verify that these cost conditions are satisfied, guidance provided in Feizollahi and Shropshire (1994, Table 1-1) was considered. Specifically, cost relationships involving treatment processing rates (in kilograms per hour), storage input/throughput rates (in cubic meters per hour), and storage total volumetric requirements (in cubic meters) were considered.

The operation rates implied by a 35-year period do not exceed the maximum rate (or scale) of operations referenced in Feizollahi and Shropshire (1994) at any of the sites for any of the SEIS-II alternatives. However, the operations rates implied by a 35-year period did fall below the minimum boundary identified in Feizollahi and Shropshire (1994) at several sites having relatively small consolidated waste volumes. Consequently, the rate of operations at these sites was increased by reducing the period of operations to less than 35 years. As a result, operation rates at Mound, Lawrence Livermore National Laboratory (LLNL), Argonne National Laboratory - East (ANL-E) and the Nevada Test Site (NTS) were increased by shortening the period of operations to less than 35 years.

Table D-3
Waste Treatment Facility Costs for Action Alternative 2A in Millions of 1994 Dollars [a]

Cost Components	Life-Cycle Component: Preoperations	Construction	Operations and Maintenance	Decontamination and Decommissioning	Functional Area: Retrieval and Characterization	Treatment	Storage	Costs [c]
WM PEIS Regionalized 2 Alternative Site Costs								
INEEL/ANL-W	125	549	875	493	930	1,023	90	2,042
Hanford	187	679	1,241	384	584	1,813	95	2,491
SRS	87	327	509	100	15	982	28	1,024
LANL	56	250	480	245	454	539	37	1,031
RFETS	34	145	303	49	46	457	29	531
ORNL	64	207	335	72	109	554	14	678
Total	---	---	---	---	---	---	---	7,797
SEIS-II Action Alternative 2 A Site Costs								
INEEL/ANL-W	281	1,234	1,967	1,108	2,091	2,300	202	4,593
Hanford [b]	1,144	4,153	7,983	2,348	3,572	11,096	969	15,637
SRS	100	376	586	115	17	1,130	32	1,179
LANL	176	788	1,512	772	1,430	1,698	117	3,245
RFETS	60	254	531	86	81	801	51	933
ORNL	132	428	693	149	225	1,146	29	1,400
Other sites [d]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	703
Total	1,893	7,233	13,272	4,578	7,416	18,171	1,400	27,690
Total Discounted at 4.1 Percent	N/A	N/A	N/A	N/A	N/A	N/A	N/A	15,356

[a] Volumes and dollars have been rounded. Actual totals may differ due to rounding.
[b] Lag storage costs are included for Hanford ($388 million).
[c] "Costs" equal the sum of the costs for Preoperations, Construction, Operations and Maintenance, and Decontamination and Decommissioning and are also equal to the sum of the costs for Retrieval and Characterization, Treatment, and Storage.
[d] Total cost of preparing waste for shipment to consolidation and treatment sites.

N/A = Not Applicable

Table D-4
Waste Treatment Facility Costs for Action Alternative 2B in Millions of 1994 Dollars [a]

Cost Components	Life-Cycle Component: Preoperations	Construction	Operations and Maintenance	Decontamination and Decommissioning	Functional Area: Retrieval and Characterization	Treatment	Storage	Costs [c]
WM PEIS Regionalized 3 Alternative Site Costs								
INEEL/ANL-W	157	680	1,140	508	930	1,456	100	2,485
Hanford	187	679	1,241	384	584	1,813	95	2,491
SRS	87	327	509	100	15	982	28	1,023
ORNL	64	207	335	72	109	554	14	678
Total	---	---	---	---	---	---	---	6,677
SEIS-II Action Alternative 2B Site Costs								
INEEL/ANL-W	539	2,335	3,915	1,745	3,194	5,001	343	8,527
Hanford [b]	1,144	4,541	7,590	2,348	3,572	11,088	969	15,629
SRS	100	376	586	115	17	1,130	32	1,179
ORNL	132	428	693	149	225	1,146	29	1,400
Other sites [d]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	3,805
Total	1,915	7,680	12,784	4,357	7,008	18,365	1,373	30,551
Total Discounted at 4.1 Percent	N/A	N/A	N/A	N/A	N/A	N/A	N/A	16,942

[a] Volumes and dollars have been rounded. Actual totals may differ due to rounding.
[b] Lag storage costs are included for Hanford ($388 million).
[c] "Costs" equal the sum of the costs for Preoperations, Construction, Operations and Maintenance, and Decontamination and Decommissioning and are also equal to the sum of the costs for Retrieval and Characterization, Treatment, and Storage.
[d] Total cost of preparing waste for shipment to consolidation and treatment sites.

N/A = Not Applicable

Table D-5
Waste Treatment Facility Costs for Action Alternative 2C in Millions of 1994 Dollars [a]

Cost Components	Life-Cycle Component: Preoperations	Construction	Operations and Maintenance	Decontamination and Decommissioning	Functional Area: Retrieval and Characterization	Treatment	Storage	Costs [c]
WM PEIS Centralized Alternative Site Costs								
Hanford	128	445	917	323	584	1,180	49	1,813
ORNL	64	207	335	72	109	554	14	678
WIPP	185	832	1,243	86	0	2,346	0	2,346
Total								4,837
SEIS-II Action Alternative 2C Site Costs								
Hanford [b]	167	580	1,585	421	759	1,539	452	2,750
ORNL	133	429	695	149	226	1,149	29	1,404
WIPP	478	2,148	3,209	222	0	6,057	0	6,057
Other sites [d]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	18,490
Totals	778	3,157	5,489	792	985	8,745	481	28,701
Total Discounted at 4.1 percent	N/A	N/A	N/A	N/A	N/A	N/A	N/A	15,909

[a] Volumes and dollars have been rounded. Actual totals may differ due to rounding.
[b] Lag storage costs are included for Hanford ($388 million).
[c] "Costs" equal the sum of the costs for Preoperations, Construction, Operations and Maintenance, and Decontamination and Decommissioning and are also equal to the sum of the costs for Retrieval and Characterization, Treatment, and Storage.
[d] Total cost of preparing waste for shipment to consolidation and treatment sites.

N/A = Not Applicable

Table D-6
Waste Treatment Facility Costs for Action Alternative 3 in Millions of 1994 Dollars [a]

Cost Components	Life-Cycle Component: Preoperations	Construction	Operations and Maintenance	Decontamination and Decommissioning	Functional Area: Retrieval	Treatment	Storage	Costs [c]
WM PEIS Regionalized 1 Alternative Site Costs								
INEEL/ANL-W	105	450	777	489	930	801	91	1,821
Hanford	156	480	1,116	371	584	1,446	93	2,123
SRS	68	246	421	120	15	780	61	855
LANL	56	260	405	240	454	468	39	961
RFETS	22	98	257	45	46	346	30	422
ORNL	45	120	247	68	109	353	19	480
Total								6,662
SEIS-II Action Alternative 3 Site Costs								
INEEL/ANL-W [b]	235	1,006	2,004	1,094	2,080	1,791	469	4,340
Hanford [b]	953	2,931	7,315	2,265	3,566	8,830	1,068	13,464
SRS	78	283	484	138	17	897	70	984
LANL	176	819	1,276	756	1,430	1,474	123	3,027
RFETS	39	172	450	79	81	606	53	740
ORNL [b]	93	248	663	141	225	730	191	1,146
Other sites [d]	N/A	N/A	N/A	N/A	N/A	A	N/A	641
Total	1,574	5,459	12,192	4,473	7,399	14,328	1,974	24,342
Total Discounted at 4.1 Percent	N/A	N/A	N/A	N/A	N/A	N/A	N/A	13,498

[a] Volumes and dollars have been rounded. Actual totals may differ due to rounding.
[b] Lag storage costs are included for Hanford ($500 million), INEEL ($266 million), and ORNL ($152 million).
[c] "Costs" equal the sum of the costs for Preoperations, Construction, Operations and Maintenance, and Decontamination and Decommissioning and are also equal to the sum of the costs for Retrieval and Characterization, Treatment, and Storage.
[d] Total cost of preparing waste for shipment to consolidation and treatment sites.

N/A = Not Applicable

Table D-7
Waste Treatment Facility Costs for No Action Alternative 1A in Millions of 1994 Dollars [a]

Cost Components	Life-Cycle Component				Functional Area			Costs [b]
	Preoperations	Construction	Operations and Maintenance	Decommissioning	Retrieval and Characterization	Treatment	Storage	
WM PEIS Regionalized 2 Alternative Site Costs								
INEEL/ANL-W	125	549	875	493	930	1,023	90	2,042
Hanford	187	679	1,241	384	584	1,813	95	2,491
SRS	87	327	509	100	15	982	28	1,023
LANL	56	250	480	245	454	539	37	1,031
RFETS	34	145	303	49	46	457	29	531
ORNL	64	207	335	72	109	554	14	678
Total	---	---	---	---	---	---	---	7,796
SEIS-II No Action Alternative 1A Site Costs								
INEEL/ANL-W	281	1,321	2,219	1,108	2,091	2,300	541	4,932
Hanford	1,144	4,242	8,236	2,348	3,572	11,096	1,311	15,979
SRS	100	415	795	115	17	1,130	280	1,427
LANL	176	835	1,727	772	1,430	1,698	379	3,507
RFETS	60	289	736	86	81	801	291	1,173
ORNL	132	460	895	149	225	1,146	263	1,634
Other sites [c]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	703
Total	1,893	7,562	14,608	4,578	7,416	18,171	3,065	29,355
Total Discounted at 4.1 Percent	N/A	N/A	N/A	N/A	N/A	N/A	N/A	16,282

[a] Volumes and dollars have been rounded. Actual totals may differ due to rounding.
[b] "Costs" equal the sum of the costs for Preoperations, Construction, Operations and Maintenance, and Decommissioning and are also equal to the sum of the costs for Retrieval and Characterization, Treatment, and Storage.
[c] Total cost of preparing waste for shipment to consolidation and treatment sites.

N/A = Not Applicable

Table D-8
Waste Treatment Facility Costs for No Action Alternative 1B in Millions of 1994 Dollars [a]

Cost Components	Life-Cycle Component				Functional Area			Costs [b]
	Preoperations	Construction	Operations and Maintenance	Decommissioning	Retrieval and Characterization	Treatment	Storage	
WM PEIS Regionalized 3 Alternative Site Costs								
INEEL/ANL-W	157	680	1,140	508	930	1,456	100	2,485
Hanford	187	679	1,241	384	584	1,813	95	2,491
SRS	87	327	509	100	15	982	28	1,024
ORNL	64	207	335	72	109	554	14	678
Total	---	---	---	---	---	---	---	6,678
SEIS-II No Action Alternative 1B Site Costs								
INEEL/ANL-W	539	2,444	4,187	1,745	3,194	5,001	724	8,919
Hanford	1,144	4,242	8,231	2,348	3,572	11,088	1,311	15,971
SRS	100	415	795	115	17	1,130	280	1,427
ORNL	132	460	895	149	225	1,146	263	1,634
Other sites [c]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	3,805
Total	1,915	7,561	14,108	4,357	7,008	18,365	2,578	31,756
Total Discounted at 4.1 Percent	N/A	N/A	N/A	N/A	N/A	N/A	N/A	17,610

[a] Volumes and dollars have been rounded. Actual totals may differ due to rounding.
[b] "Costs" equal the sum of the costs for Preoperations, Construction, Operations and Maintenance, and Decommissioning and are also equal to the sum of the costs for Retrieval and Characterization, Treatment, and Storage.
[c] Total cost of preparing waste for shipment to consolidation and treatment sites.

N/A = Not Applicable

Table D-9
Volume Adjustment Factors by Alternative

Consolidated Site	Proposed Action	Action Alternative 1	Action Alternatives 2A, 3, and No Action 1A	Action Alternative 2B and No Action 1B	Action Alternative 2C
INEEL/ANL-W	0.81	2.28	2.25	3.43	N/A
Hanford	3.43	5.95	6.12	6.12	1.30
SRS	0.71	1.00	1.15	1.15	N/A
LANL	1.92	3.18	3.15	N/A	N/A
RFETS	1.75	1.75	1.75	N/A	N/A
ORNL	2.04	2.85	2.07	2.07	2.07
Mound	0.20	0.20	N/A	N/A	N/A
LLNL	0.70	0.12	N/A	N/A	N/A
NTS	1.04	1.96	N/A	N/A	N/A
ANL-E	0.15	0.49	N/A	N/A	N/A
WIPP	N/A	N/A	N/A	N/A	2.58

N/A = Not Applicable

In the cases of Action Alternatives 1, 2, and 3, the site storage costs at Hanford, Idaho National Engineering and Environmental Laboratory (INEEL), and Oak Ridge National Laboratory (ORNL) extend beyond the 35-year waste processing period. These storage-cost adjustments are made to account for extended storage periods ranging from 100 to 155 years due to WIPP emplacement limitations for remote-handled (RH) TRU waste. Thus, the storage costs at these sites reflect periodic expenditures made every 35 years over the waste work-off period and are calculated on the basis of the remaining site inventories at the end of each period. In this way, storage costs decrease on a periodic basis over the life of the alternative as waste inventories are shipped to WIPP for disposal. Care was taken to reduce the number of sites requiring extended storage by adjusting treatment-shipment rates among the various sites, provided it did not interfere with waste handling limitations or compromise WIPP emplacement capacity. This resulted in fewer lag storage sites than the number considered in Appendix F.

The waste management facility costs, waste transportation costs, and WIPP operations costs presented in this appendix and in Chapter 5 are presented in discounted present value form. The approach for discounting the value of these costs is straightforward. In present value terms, annual costs incurred t-years from the present are discounted by multiplying them by the discount factor $(1/1+r)^t$, where “r” reflects an inflation-adjusted discount rate. A discount rate of 4.1 percent was used in the analysis of life-cycle costs presented in Chapter 5. Over the project life-cycle ($T = 35$ years), the discounted annual costs are represented by $C\ (1 + x^1 + x^2 + x^3 + \ldots + x^{35})$, where $x = (1/1 + r)$ and $T = 35$ is the end year of the project.

Table D-10 provides a sample of present value calculations for a 35-year period using hypothetical annual costs of $50 million, $100 million, $500 million, and $1,000 million and inflation-adjusted discount rates of 0 percent, 3 percent, and 5 percent. The present value of the cost estimates reported in this appendix, Chapter 5 and in the summary reflect the case of a 4.1 percent discount rate; thus, they are relatively higher than if they were discounted at 5 percent and relatively lower than if they were discounted at 3 percent.

Table D-10
Present Values of Hypothetical Annual Costs Incurred Over 35 Years

Inflation-Adjusted Discount Rate	Present Value of Annual Expenditures: $50 million for 35 years	Present Value of Annual Expenditures: $100 million for 35 years	Present Value of Annual Expenditures: $500 million for 35 years	Present Value of Annual Expenditures: $1,000 million per year for 35 years
r = 0 percent	$1,750 million	$3,500 million	$17,500 million	$35,000 million
r = 3 percent	$1,074 million	$2,149 million	$10,744 million	$21,487 million
r = 5 percent	$819 million	$1,637 million	$8,187 million	$16,374 million

D.1.2 Waste Transportation Costs

The waste transportation costs for contact-handled (CH) TRU and RH-TRU waste vary by alternative, depending on the mode of transportation and the total mileage and number of shipments required for disposal of the waste volumes. Two sources of information for the estimation of waste transportation costs were used: (1) the *Engineered Alternatives Cost/Benefit Study Final Report* (DOE 1995) for information concerning mileage between sites and the fixed and variable costs per truck shipment between sites; and (2) the *Comparative Study of the Waste Isolation Pilot Plant (WIPP) Transportation Alternatives* (DOE 1994) for information concerning the shipment costs of CH-TRU waste and RH-TRU waste via regular-class rail service and dedicated-class rail service.

The number of shipments reflects potential weight, thermal power, or plutonium-239 equivalent curies (PE-Ci) limitations. These relationships are described in Appendix A of this document. Truck shipments are limited to a maximum of three TRUPACT-IIs or one RH-72B per shipment. Rail shipments (either regular-class or dedicated-class) are limited to a maximum of six TRUPACT-IIs or two RH-72Bs per rail car. CH-TRU waste shipments may include as many as 42 drums of low-density waste (three TRUPACT-IIs) or as few as 12 drums of high-density waste (two TRUPACT-IIs). For RH-TRU waste, no limitations were assumed in meeting container and shipping specifications.

Approximately 38 percent of the waste shipments for the Proposed Action and Action Alternative 1 reflect two TRUPACT-IIs per shipment (the remainder involves three TRUPACT-IIs per shipment). All of the waste shipments for Action Alternative 2 reflect two TRUPACT-IIs per shipment. Approximately 86 percent of the shipments for Action Alternative 3 involve two TRUPACT-IIs per shipment. These percentages are based on stored waste stream data and are extrapolated to the other wastes for Action Alternatives 1 and 3.

Total shipments cover the number of shipments made from the small sites to consolidation sites for treatment and storage, and the number of shipments of treated waste volumes to WIPP. Total mileage depends on the number of shipments required to transport the treated waste volumes from each site to WIPP and the round-trip mileage between each site and WIPP. Total cost covers fixed and variable costs of transporting waste volumes between sites under the various options. Various cost parameters were used in the estimation of waste transportation costs, including the following:

- Fixed costs by truck - the number of truck shipments required in the transportation of treated waste volumes multiplied by an average fixed cost per shipment of $9,260

- Variable costs by truck - the number of round trip miles between waste origin and destination multiplied by an average variable cost of $0.12 per mile

- Total cost by rail (regular or dedicated rail service) - the number of rail cars shipped multiplied by a round-trip carload charge that varies from site to site. The round-trip carload charge ranges from $13,880 for Rocky Flats Environmental Technology Site (RFETS) under regular rail service up to $247,314 for Hanford under dedicated rail service.

The total waste transportation costs for truck and rail transportation are presented in Table D-11. Note that the transportation cost impacts for combined CH-TRU waste and RH-TRU waste shipments range from a minimum of $33 million under No Action Alternative 1A using regular rail, to a maximum of $15.69 billion under Action Alternative 3, using dedicated rail.

Table D-11
Truck and Rail Transportation Costs
by Alternative in Millions of 1994 Dollars

Action Alternative	Total Shipments [a]	Total Mileage	Total Cost
Proposed Action	38,290	105,856,389	1,588
Action Alternative 1 (Truck)	107,179	334,124,688	4,909
Action Alternative 1 (Regular Rail)	58,827	--	1,642
Action Alternative 1 (Dedicated Rail)	58,827	--	11,315
Action Alternative 2A (Truck)	67,488	203,120,649	2,990
Action Alternative 2A (Regular Rail)	37,603	--	1,007
Action Alternative 2A (Dedicated Rail)	37,603	--	6,863
Action Alternative 2B (Truck)	72,371	225,452,149	3,301
Action Alternative 2B (Regular Rail)	40,037	--	1,144
Action Alternative 2B (Dedicated Rail)	40,037	--	7,484
Action Alternative 2C (Truck)	66,132	197,247,516	2,913
Action Alternative 2C (Regular Rail)	37,561	--	909
Action Alternative 2C (Dedicated Rail)	37,561	--	6,544
Action Alternative 3 (Truck)	149,671	467,115,976	6,837
Action Alternative 3 (Regular Rail)	83,483	--	2,344
Action Alternative 3 (Dedicated Rail)	83,483	--	15,693
No Action Alternative 1A (Truck)	2,818	335,592,652	74
No Action Alternative 1A (Regular Rail)	1,409	--	33
No Action Alternative 1A (Dedicated Rail)	1,409	--	251
No Action Alternative 1B (Truck)	7,702	14,031,655	235
No Action Alternative 1B (Regular Rail)	3,851	--	157
No Action Alternative 1B (Dedicated Rail)	3,851	--	1,184
No Action Alternative 2	0	0	0

[a] Shipment numbers include shipments to consolidate waste at treatment sites or sites consolidating waste for shipment to WIPP.

D.1.3 WIPP Operations Budget

The life-cycle cost of WIPP operations will depend on the period of emplacement operations, which varies across the different SEIS-II alternatives. The analysis relied on budgetary information provided by Krznarich (1996). This budgetary information projects an average annual WIPP operations budget of $150 million over the period of emplacement operations and an average annual WIPP project budget of $180 million. Life-cycle cost impacts were based on the WIPP operations budget, while economic impacts in the region of influence (ROI) were based on the larger WIPP project budget. Other budgetary items, such as federal transfer payments made to state and local government agencies involving WIPP operations, have not been considered in the analyses of Chapter 5.

D.2 ECONOMIC IMPACTS IN THE WIPP ROI

This section discusses the methods and assumptions used in the analysis of the economic impacts which the Proposed Action and other SEIS-II alternatives may have in the ROI. Lansford et al. (1996) provides the most recent account of WIPP's economic effects on the New Mexico economy.

This work provides information about the distribution of WIPP fiscal year (FY) 1995 expenditures across industrial sectors in the ROI. This information was combined with projected WIPP operating budgets to construct a regional input-output model of the ROI for the analysis of SEIS-II economic impacts. This technical approach identifies the direct and indirect economic effects that the Proposed Action and the SEIS-II alternatives would have on regional employment, income, and output of goods and services.

Readers should note that the economic effects reported in this document both for WIPP and the treatment sites do not necessarily reflect the "creation" of new impacts. The reported impacts are estimates of what effects would be "supported" by specific levels of ROI expenditures, which may include substantial portions of existing, recurring expenditures, in addition to some new expenditures for new program facets presented in the alternatives.

The regional input-output model was developed using the IMpact analysis for PLANning (IMPLAN) regional economic modeling framework. A detailed discussion of the IMPLAN regional modeling methods can be found in the *IMPLAN Professional: User's Guide, Analysis Guide, Data Guide* (MIG 1997). A summary of the assumptions and limitations of the IMPLAN model is provided below.

- IMPLAN is a non-survey, input-output model that uses an adaptation of the 538-sector national input-output transactions table, otherwise known as the "national table." The most recent national table was issued by the Bureau of Economic Analysis (BEA 1994) and represents the industrial technologies in-place in 1987.

- IMPLAN provides the flexibility to update the 1987-level technology of any industry, as represented in the national table, to an improved representation of the technology currently being employed.

- IMPLAN reflects technology relationships that tend to be relatively stable over 10- to 15-year spans for most industries. Rapidly changing industries, such as the computer industry, are exceptions and should be evaluated in cases where they are affected by the economic impact scenario.

- IMPLAN, like other static input-output models, assumes that all resources required to satisfy an economic expansion are either immediately available within the study region or can be immediately imported.

To analyze economic impacts of a particular policy option using IMPLAN, one must have knowledge of the *net* change in "final demand" purchases made within the regional economy. Lansford et al. (1994, 1995, and 1996) provided FY 1993 through FY 1995 WIPP expenditure data for the State of New Mexico. The three years vary greatly in the pattern of their expenditures across the industries of the New Mexico economy, indicating that several "one-time" or "periodic" purchases are made in a given year. The economic impact model used a composite expenditure pattern based on the average pattern across FY 1993 through FY 1995 to catch some of the year-to-year variation in expenditure patterns at WIPP. The projected WIPP budget estimates were provided by Krznarich (1996). Based on this regional information, the IMPLAN model was used to generate a set of economic multipliers for the Proposed Action which were applied to WIPP-related expenditures in the ROI to determine the economic impact for each alternative. The economic activity multiplier for the ROI is estimated to be 2.80, meaning that for each dollar of WIPP-related budget expenditure in the ROI, an additional $1.80 is generated in the ROI economy. The ROI employment multiplier for direct WIPP employment is estimated to be 3.23, meaning that for each direct WIPP job, an additional 2.23 ROI jobs are supported (these may be full- or part-time jobs). The ROI labor income (employment compensation) multiplier is estimated to be 2.45, meaning that for each dollar of WIPP-related income generated in the ROI, another $1.45 of additional income is generated in the ROI.

As frequently noted, the Carlsbad area of the ROI receives the majority of the WIPP-related economic impact. Most of the WIPP personnel live and shop in the Carlsbad area. Many support services are located in Carlsbad, and WIPP-related spin-off businesses have located in Carlsbad. Typically, public economic data are not resolved to any finer detail than the county level, which presents the problem of how to resolve economic impacts that are focussed on a locality within the ROI.

Adjustments to the county-level economic data can be employed to estimate the economic impact in a given locality within a county. Such an adjustment was made to the ROI economic impact estimates to separate the impact on the Carlsbad area from the total ROI economic impact. In lieu of developing a comprehensive survey of all businesses in the ROI, the approach involved using publicly available phone book data (PhoneDisc 1997) to generate establishment counts by Standard Industrial Classification (SIC) code. Business phone listings for all businesses in Eddy and Lea counties were extracted and compiled based on locality and SIC code. This effectively created a census of all businesses (that list a phone number) in the ROI from which a location quotient was developed. Use of location quotients attempts to estimate the amount of local production that gets consumed locally. Schaffer and Chu (1969) are typically credited with pioneering this approach to a difficult question.

The question at issue for this analysis is, given that economic impact models estimate only county- or multi-county-level economic impacts, what portion of the two-county ROI impact occurs in the Carlsbad area of the ROI? The establishment counts by industry sector were used to develop the following location quotient for a given industry, *i*:

$$LQ_{i,Carlsbad} = (E_{i,Carlsbad} / E_{Carlsbad}) / (E_{i,ROI} / E_{ROI}),$$

where *E* is the number of establishments – gleaned from the 1997 phone book data. Based on this formulation, ROI industry-level economic impact estimates for the Proposed Action were multiplied by the Carlsbad industry-specific location quotients to estimate the proportion of ROI impact that could be expected to occur in the Carlsbad area. Under this approach, 84.6 percent of the ROI output impacts, 89.7 percent of the ROI employment impacts, and 89.8 percent of the ROI labor income impacts are estimated to occur in the Carlsbad area of the ROI, with the balance being distributed throughout all other localities. Table D-12 provides estimates of these economic impacts based on the location quotient method.

Table D-12
Annual Split of Total ROI Economic Impacts of the Proposed Action

Industry	Carlsbad Output (millions of 1994 dollars)	Carlsbad Employment (jobs, full- and part-time)	Carlsbad Income (millions of 1994 dollars)	Other ROI Output (millions of 1994 dollars)	Other ROI Employment (jobs, full- and part-time)	Other ROI Income (millions of 1994 dollars)
Ag & Mining	2.063	21	0.445	2.169	11	0.291
Construction	6.824	109	2.998	0.363	6	0.159
Manufacturing	14.304	152	3.217	12.899	75	1.736
Trans. Comm. Utilities	16.950	168	3.741	6.350	43	1.494
Trade	28.352	493	11.471	4.355	58	1.566
FIRE	25.308	154	2.766	1.350	8	0.148
Services	63.313	874	30.428	3.288	45	1.572
Government	14.886	215	11.825	0.790	11	0.627
WIPP Direct Effect	**95.501**	**979**	**46.255**	**17.536**	**116**	**5.246**
WIPP Total Effect	**267.499**	**3,164**	**113.146**	**49.102**	**374**	**12.839**
Multipliers	**2.80**	**3.23**	**2.45**	**2.80**	**3.23**	**2.45**

The location quotient method is simplistic because these estimations are usually made with income and employment data, but such data were not available at the level of industry and geographic detail needed for a sub-county analysis. Caution should be used when applying these estimates because by ignoring some local importing and exporting behavior among localities and the rest of the country and the effect of establishment size, the estimates are conservative and would tend to overstate the Carlsbad impact, rather than understate it, forming the likely upper bound on potential impacts. The limits of this approach required that ROI economic multipliers be applied uniformly to Carlsbad and the entire ROI, but since Carlsbad is a subset of the ROI, its economic multipliers can be reasoned to be lower. However, these location quotient estimates were used to develop the information presented in Chapter 5 and the Summary.

The continued operation of WIPP at the anticipated funding levels would continue to have a stabilizing effect in the ROI economy, which has historically varied according to price trends in the oil and gas industry. Using these estimates, the Proposed Action would account for as much as 7.9 percent of all employment in the ROI. This proportion would be higher when just considering the local economy of the Carlsbad portion of the ROI. It is important to note, however, that some SEIS-II alternatives call for project life spans that reach as far as the year 2310, including the institutional control period. The models used to develop economic impact information are valid for application over perhaps a 10- to 15-year period. Long-term economic forecasting models may be applicable for a 20-50 year period. No economic impact model exists that can be reliably applied to extremely long-term planning horizons (50 to 300+ years). The only claim that can be made with these impact estimates is that they are estimated with documented methods that are consistent across the alternatives, thus allowing a relative comparison.

The potential closure of WIPP presents the most noticeable economic impact in the ROI. This event is reflected in No Action Alternatives 1 and 2. These alternatives specify that, beginning in 1998, the WIPP will begin a 10-year period of decommissioning. By the year 2008, the site will be fully dismantled and returned to near-original condition with no TRU waste disposal activities having taken place. This implies that federal budget support of WIPP will be ramping down from current levels to near zero by the year 2008. This budgetary impact on the ROI can be considered in one of two ways. Under one scenario, the WIPP site can be assumed to undergo a "straight-line" decommissioning period, meaning that budgets would be cut by equal increments over the period until the site is permanently closed in the year 2008. Alternatively, WIPP closure could involve a "front-loaded" decommissioning period where more than half of the WIPP work force would be released in the first three years and thereafter decline at a much slower rate over the remaining years. The workforce would be declining rapidly and at the same time convert from an operations workforce to a decommissioning workforce until permanent closure in the year 2008.

The economic impacts reported in Chapter 5 are based on the straight-line decommissioning assumption. Corresponding to this case, the flow of federal dollars for WIPP salaries and expenditures in the ROI would decline steadily for 10 years prior to permanent closure of the facility. The decline in direct WIPP-related business would translate into additional declines in employment and salaries in the ROI workforce and industries that supply WIPP-related goods and services. Less of the support services currently provided in the ROI would remain viable over the course of WIPP closure. Similarly, the ROI service sector would decline as individuals leave the region in search of employment opportunities. At the same time, local and state government agencies would be affected by reductions in sales tax revenue and the region's tax base. Finally, the reduction in federal payments to government agencies in the ROI for WIPP-related services would have an effect on road and highway improvements, emergency preparedness, and other public services currently provided in the ROI. These impacts are not explicitly included in the analyses of Chapter 5.

D.3 ECONOMIC IMPACTS IN THE TREATMENT SITE ECONOMIC REGIONS

This section discusses the methods and assumptions used in the analysis of the impacts which the Proposed Action and the SEIS-II alternatives would have on regional employment, income, and output of goods and services at ANL-E, Hanford, INEEL, LANL, LLNL, Mound, NTS, Oak Ridge Reservation (ORR), RFETS, and SRS. The WM PEIS (DOE 1997) provides the most recent economic assessment of waste management activities at these treatment sites. That assessment provides information about economic impacts of the TRU waste management program for the DOE complex. The intent here is to provide further documentation for estimates found in Chapter 5.

First, economic regions were developed for the treatment sites based on the county location of the site and the associated total labor market for that county. Journey-to-work data (BEA 1996) were used to identify counties contributing to the base county's labor market, and counties were included as part of the ROI to the point that at least 95 percent of the labor market had been identified (see Table D-13). Next, the IMPLAN modeling tool was used with county data already available from this and other studies to create models for three sites. Impact models for the Hanford, LANL, and WIPP sites had been developed for previous studies and were adapted for this analysis. Each model provides estimates of output per unit of cost, employment per unit of cost, and income per unit of cost for the three ROIs mentioned. Next, FY 1996 workforce estimates for each site were divided by the 1995 ROI population for each site to estimate a consistent proportion of site employment to total ROI population. This measure was used to group the treatment sites according to similar site employment-to-population proportions. This resulted in separate groupings that linked LANL and INEEL; Hanford, ORR, and SRS; and WIPP, ANL-E, LLNL, Mound, NTS and RFETS. Based on these groupings, the impact factors from the LANL model were applied to INEEL, the impact factors from the Hanford model were applied to ORR and SRS, and WIPP impact factors were applied to ANL-E, LLNL, Mound, NTS, and RFETS (see Table D-14).

Table D-13
Determination of Treatment Site Economic Region

Treatment Site	County Location of Site	Counties Included in Economic ROI	Cumulative Percentage of Labor Market
ANL-E	Cook, IL	Cook, IL	83.7
		Du Page, IL	89.8
		Lake, IL	93.0
		Will, IL	95.2
Hanford	Benton, WA	Benton, WA	84.0
		Franklin, WA	92.9
		Yakima, WA	97.2
INEEL	Bingham, ID	Bingham, ID	83.7
		Bonneville, ID	92.5
		Bannock, ID	97.5
LANL	Los Alamos, NM	Los Alamos, NM	58.9
		Santa Fe, NM	78.3
		Rio Arriba, NM	95.6
LLNL	Alameda, CA	Alameda, CA	70.9
		Contra Costa, CA	84.2
		Santa Clara, CA	88.1
		San Francisco, CA	91.1
		San Mateo, CA	93.2
		San Joaquin, CA	95.1
Mound	Montgomery, OH	Montgomery, OH	76.1
		Greene, OH	84.8
		Warren, OH	88.1
		Miami, OH	91.4
		Clark, OH	94.2
		Butler, OH	95.6
NTS	Nye, NV	Nye, NV	58.9
		Clark, NV	95.4
ORR	Anderson, TN	Anderson, TN	51.3
		Knox, TN	76.0
		Roane, TN	87.3
		Campbell, TN	90.7
		Morgan, TN	93.5
		Loudon, TN	95.9
RFETS	Adams, CO	Adams, CO	51.8
		Jefferson, CO	66.7
		Arapahoe, CO	78.4
		Denver, CO	89.2
		Boulder, CO	94.4
		Weld, CO	97.3
SRS	Barnwell, SC	Barnwell, SC	70.9
		Bamberg, SC	79.3
		Aiken, SC	86.2
		Allendale, SC	91.7
		Orangeburg, SC	94.4
		Richmond, GA	97.0

Table D-14
Development of Economic Impact Scaling Factors for Treatment Sites

Treatment Site	FY 1996 Site Workforce	1995 Economic ROI Population	Site Employment as a Percentage of ROI Population	Economic Impact Model Used for Scaling	ROI Industrial Output per Unit of Cost	ROI Employment per Unit of Cost	ROI Labor Income per Unit of Cost
ANL-E	4,500	6,763,664	0.07	WIPP	2.80	19.7	0.70
Hanford	10,587	389,564	2.72	Hanford	1.70	32.0	1.62
INEEL	9,000	146,383	6.15	LANL	1.81	48.5	1.20
LANL	7,000	171,977	4.07	LANL	1.81	48.5	1.20
LLNL	7,300	5,696,565	0.13	WIPP	2.80	19.7	0.70
Mound	1,180	1,403,308	0.08	WIPP	2.80	19.7	0.70
NTS	5,700	1,017,156	0.56	WIPP	2.80	19.7	0.70
ORR	17,114	572,917	2.99	Hanford	1.70	32.0	1.62
RFETS	3,418	2,133,595	0.16	WIPP	2.80	19.7	0.70
SRS	16,000	466,042	3.43	Hanford	1.70	32.0	1.62

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APPENDIX E
TRANSPORTATION

This appendix supports results of transportation analyses presented in Chapter 5 of this document. Plans for transporting transuranic (TRU) waste to the Waste Isolation Pilot Plant (WIPP) and the impacts associated with that transportation are discussed.

E.1 INTRODUCTION

Since the Department of Energy (DOE or the Department) prepared the *Final Supplement Environmental Impact Statement for the Waste Isolation Pilot Plant* (SEIS-I) in 1990, new information has become available concerning the Department's TRU waste and its management. This new information falls into several categories, namely:

- ***Waste characterization:*** Due to improved site waste characterization, the volume and characteristics of waste to be transported are better understood than was the case for SEIS-I. The improved data permits more thorough analyses of the impacts associated with transporting TRU waste. Additionally, site-specific waste characterization data have been examined thoroughly to ensure that each site would meet all of the required shipping limits; including weight, volume, curie, and thermal power limits.

- ***The addition of small quantity sites under the Proposed Action and the alternatives:*** Additional sites with small quantities of TRU waste and sites with additional sources of TRU waste (previously disposed of, polychlorinated biphenyl (PCB)-commingled, commercial, and nondefense) have been added to the analyses, and the impacts of transporting waste from these sites have now been considered. Currently, DOE intends for these smaller sites to ship their TRU waste to one of the larger quantity sites for processing. Decisions on the locations of consolidation/treatment sites for TRU waste, if any, will be based upon the analysis in the *Final Waste Management Programmatic Environmental Impact Statement* (WM PEIS) (DOE 1997).

- ***Waste treatment agreements:*** To date, DOE has submitted Federal Facility Compliance Act (FFCAct) site treatment plans for its treatment sites, and orders or settlements have been reached with affected states and the U.S. Environmental Protection Agency (EPA). In most cases, TRU waste would be retrieved, characterized, treated, certified to meet the WIPP Waste Acceptance Criteria (WAC), and then stored pending disposal. Some of the smaller sites would ship TRU waste to another DOE site for characterization, treatment, certification to meet WAC, and then storage. As a possible exception, DOE has reached a negotiated settlement with the State of Idaho that would allow DOE to procure, construct, and operate a mixed waste treatment facility. It is currently planned that this mixed waste treatment facility would use a thermal process that would result in a waste form that would meet Resource Conservation and Recovery Act (RCRA) Land Disposal Restrictions (LDR). Treated waste would also meet WAC. Additional information regarding the management of TRU waste in the DOE complex may be found in the WM PEIS.

- ***Route changes:*** There have been changes on the routes to WIPP coming from Argonne National Laboratory - East (ANL-E), the Mound Plant (Mound), Lawrence Livermore National Laboratory (LLNL), and Oak Ridge National Laboratory (ORNL). These routes, with the exception of the LLNL route, which was modified to bypass Los Angeles, California, now feed into Interstate-20 and enter New Mexico from the south. No waste, therefore, would travel on Interstate-40 from the east into New Mexico. Additionally, minor changes in some of the routes surrounding the major generator-storage sites have been decided upon and potential bypass routes around Santa Fe, New Mexico; Roswell, New Mexico; and Carlsbad, New Mexico are under consideration.

- ***HIGHWAY and INTERLINE codes updated with 1990 census data:*** The HIGHWAY (Johnson et al. 1993a) and INTERLINE (Johnson et al. 1993b) computer codes allow for flexibility in calculating highway/rail mileage and population statistics, respectively. Each code provides mileage and population densities for designated highway and rail routes and have been updated with the most current (1990) census data.

- ***RH-72B shipping cask:*** The RH-72B shipping cask was referred to as the NUPAC in SEIS-I. Testing on the NUPAC was done on a 5/8 scale model, identified as the NUPAC 125B. Shipping canisters would be placed within the cask for added shielding and stabilization. The Safety Analysis Report for Packaging application for the RH-72B shipping cask was submitted to the Nuclear Regulatory Commission (NRC) for evaluation in December of 1996.

Four types of impacts were assessed in this second supplemental environmental impact statement (SEIS-II) regarding the transportation of TRU waste to WIPP. The first was the number of traffic accidents, fatalities, and injuries likely to occur as a result of transporting TRUPACT-IIs and RH-72B casks round trip between WIPP and the generator-storage sites. The second type, accident-free radiological impacts, would be associated with the external radiation present around a TRU waste package as it is being shipped. This radiation exposes the general public and transportation workers to very low levels of radiation both during transportation and while a shipment is stopped. The third type of impact, pollution health effects, would be the result of vehicle emissions (diesel exhaust) while traveling through urban areas. The final impact would be associated with accidents that are severe enough to breach the TRU waste packages, releasing some of the radioactive and hazardous material being shipped. For these accidents, two sets of radionuclide inventories were developed for both contact-handled (CH) TRU and remote-handled (RH) TRU waste shipments. The first inventory was based upon limits presented in planning-basis WAC (DOE 1996a) and the second inventory was based upon site-average radionuclide inventories.

Each type of impact presented in this appendix depends upon the route characteristics, the number of shipments via each route; and, for accidents, the accident environment associated with a particular mode of transportation and the behavior of the waste form and packaging in that environment. Thus, this section will summarize important route characteristics, TRU waste shipments using each route, and the impacts of potential transportation accidents.

Table 3-17 of Chapter 3 presents the features of the Proposed Action and the alternatives that are critical to transportation impact analyses, including the consolidation and treatment of TRU waste and TRU waste emplacement time frames. For a comprehensive description of the TRU waste transportation system, the reader is referred to Appendix A of this document and Appendices C, L, and M of SEIS-I (DOE 1990). Appendix L describes the design, testing, and certification of the shipping containers and casks used for TRU waste; Appendix C describes emergency-response training and capabilities; and Appendix M describes the management plan of the carrier.

E.2 TRUCK TRANSPORTATION AND HIGHWAY ROUTES

Although WIPP has facilities designed to receive TRU waste shipments either by truck or by rail, the Proposed Action considers shipment only by truck. Under the action alternatives, TRU waste transportation by truck is analyzed, and those results are used to assess the transportation impacts associated with shipments by rail.

To ensure that transportation operations proceed safely and efficiently, DOE has developed operating plans and provided communications facilities, including a dual satellite-based vehicle tracking system. DOE has awarded a contract to Colorado Allstate Trucking (CAST) Transportation, herein referred to as the "carrier." This contract, which runs for one year with options for four 1-year extensions, contains provisions for the safe and efficient transport of TRU waste and for responses to transportation emergencies. One of the contract provisions requires that the carrier prepare a management plan. The carrier's plan has been prepared and is similar to the plan discussed in Appendix M of SEIS-I (DOE 1990). Key provisions of the contract include the following:

- The carrier will provide tractors and drivers which are dedicated to contract requirements. Drivers are to be technically qualified and experienced and must complete training in twenty-eight training categories, including hazardous and radioactive materials transportation. Tractors are to be domiciled and maintained within 80 kilometers (50 miles) of WIPP. Tractors will be dispatched with a DOE-owned trailer and empty shipping containers.

- DOE will operate a transportation operations control center, the Central Monitoring Room (CMR), 24 hours per day, 7 days per week. This center will maintain day-to-day contact with the carrier and drivers. As required by DOE Order 460.2, the Transportation Tracking and Communications System (TRANSCOM) Control Center located in Oak Ridge, Tennessee, has the ability to track and communicate with shipment vehicles using the DOE TRANSCOM system. Tracking information can be disseminated to DOE users and other stakeholders, as necessary.

- The carrier will be required to meet federal regulatory requirements for the transportation of radioactive and hazardous materials, including: driver training in accordance with Title 49 of the Code of Federal Regulations (CFR), Part 172 (49 CFR 172), Subpart G entitled "Emergency Response Information"; 49 CFR Part 177.825 entitled "Routing and Training Requirements for Class 7 (Radioactive) Materials; 49 CFR Part 391 entitled "Qualifications of Drivers"; 49 CFR 397, Subpart D entitled "Routing of Class 7 (Radioactive) Materials"; the Commercial Motor Vehicle Safety Act of 1986 and subsequent amendments; and manifesting requirements for mixed waste specified in 40 CFR.

- The carrier is required to comply with emergency response guidelines for hazardous material and hazardous waste transporters outlined in the *Emergency Planning, Response, and Recovery Roles and Responsibilities for TRU Waste Transportation Accidents* (DOE 1995a).

In the event of an accident, carrier drivers would notify emergency first-responders via cellular phone and the CMR at WIPP via TRANSCOM. A senior DOE official and/or the DOE Carlsbad Area Office (CAO) Incident/Accident Team Leader would assist the state-provided on-scene commander. DOE resources would be made available to local authorities, as appropriate, to support the mitigation of the accident; including, but not limited to, package recovery and site cleanup. In the event of an accident such as a fire, breach, release, or suspected radioactive contamination, the carrier would follow established procedures for obtaining any needed federal, state, or local assistance and technical advice.

Drivers would carry instructions regarding the appropriate actions to be taken in the event of an accident and would be trained in packaging recovery procedures. In addition, the TRANSCOM system provides an electronic version of the U.S. Department of Transportation (DOT) Emergency Response Guidebook information, which is specific to each shipment material, to all TRANSCOM users. According to 49 CFR Part 390.15(b), the carrier shall maintain a register containing information on the accident for a period of one year after the accident occurs. Any carrier accident, no matter how minor, would be reported to the CAO Transportation Manager, the WIPP Traffic Manager, the CMR operator, and DOE Headquarters (HQ) Emergency Operations Center (EOC) via the DOE Albuquerque Operations Office. If not already notified by the carrier, the CMR operator would notify the shipper.

The carrier's management plan provides procedures to be followed regarding adverse weather conditions, delays, and parking during TRU waste shipments. Weather conditions would be constantly monitored, and drivers would be alerted to possible severe weather conditions. Delays may occur as the result of problems at the sites, weather conditions, or maintenance checks. Schedule delays of two or more hours from the shipping, receiving, and transit time would be immediately reported to the CMR, which would then notify the

TRANSCOM SYSTEM

DOE has developed a transportation tracking and communications system that is used to track truck and rail shipments. This satellite-based system, the TRANSCOM System, has been in operation since 1989. Since its inception, the TRANSCOM System has tracked over 500 shipments for DOE. The use of TRANSCOM is mandated by DOE Order 460.2, *Departmental Materials Transportation and Packaging Management*.

The mission of the TRANSCOM System is to provide tracking and communications for shipments of radioactive materials, hazardous materials, and other high-visibility shipping campaigns, as specified by DOE. The TRANSCOM System is managed and operated at the TRANSCOM Control Center (TCC) in Oak Ridge, Tennessee, for DOE.

The TRANSCOM System provides the TCC staff, shippers, carriers, receivers, and state, Tribal, and federal users with the ability to view information about shipments and communicate with each other during shipment tracking. Information about shipment contents, points of contact, routes, status, locations, and emergency response information is available to local emergency response teams and each user. The information is displayed in tabular and graphical form using a series of national, state, and county maps. The vehicle location can be determined to within a few meters, with position updates as frequently as every 60 seconds. Drivers are alerted to adverse weather or road conditions.

shipper or receiver, as appropriate. There would be no “deadlines” for a shipment to be received at WIPP. If a shipment were delayed, a new scheduled time of arrival would be arranged.

The carrier’s management plan follows instructions provided by the Western Governor’s Association (WGA) and DOE regarding the selection of suitable site parking areas. In the event of a shipping layover, carriers would use designated DOE or Department of Defense parking sites or an area designated by the affected state as a safe parking area. If a designated site were not available, the driver would select an appropriate site based on criteria such as nearby population, access, and security. The driver would also notify the nearest state police district office to confirm the appropriateness of the location. Motor vehicles transporting hazardous waste material other than Class A or Class B explosives are required not to be parked on or within 1.5 meters (5 feet) of the traveled portion of a public street or road, except for brief periods when the necessities of operation make it impracticable to park in any other place.

E.2.1 HIGHWAY Code

The HIGHWAY computer code (Johnson et al. 1993a) was used to determine the various truck routes used for these analyses. HIGHWAY is a computerized road atlas that details more than 386,000 kilometers (240,000 miles) of interstate and other highways. The user can specify the routing criteria to constrain the route selection. HIGHWAY calculates the total route length and the distances traveled through rural, suburban, and urban population zones. The HIGHWAY model contains an HM-164 and a WIPP default routing option. The HM-164 option, when activated, specifies a route that would comply with DOT regulations for highway route-controlled quantities (HRCQ) of radioactive material. The WIPP default routing option provides the New Mexico-specified routes to WIPP and uses routes defined by the HM-164 option for routes outside of New Mexico. When determining the route selection for the transportation analyses, the default settings were used in most cases, and the WIPP routing code option was activated. In this way, the most direct route that complies with DOT regulations would be selected by HIGHWAY. Population densities along each route were derived from 1990 census data. Rural, suburban, and urban areas are characterized according to the following breakdown:

- Rural mean population density of 6 persons per square kilometer (16 per square mile)

- Suburban mean population density of 719 persons per square kilometer (1,863 per square mile)

- Urban mean population density of 3,861 persons per square kilometer (10,003 per square mile)

E.2.2 Regulations Applicable to Highway Route Selection

On behalf of the carrier, DOE has coordinated shipping routes with the affected states. As a matter of policy, DOE has determined that all shipments, whether or not the definition of HRCQ has been met, will use the preferred routes in 49 CFR Part 397 Subpart D. Preferred routes consist of interstate-system highways, interstate bypasses or beltways around cities, and state-designated routes. This routing rule permits states and Indian tribes to designate routes in accordance with DOT guidelines or equivalent routing analysis. Interstate highways must be used in the absence of routes designated by states or tribes, unless a deviation is necessary.

WESTERN GOVERNORS ASSOCIATION

The Western Governors Association (WGA) WIPP Transport Advisory Group has cooperated with DOE to develop a safe transport program for waste shipments to WIPP. A memorandum of agreement between the western states and DOE *(Regional Protocol for the Safe Transport of Transuranic Waste to the WIPP)* has been signed. Also, under a contract with the WGA, the Western Interstate Energy Board prepared, *Safe Parking Areas for WIPP Shipments* in 1990, which will be strictly adhered to by the drivers. DOE sites have been designated for use as safe parking areas, and DOE has reached an agreement with the Department of Defense for the use of its facilities along the WIPP route for emergency parking. If no DOE, Department of Defense, or state-designated parking areas can be reached safely, the driver will be directed to select a safe parking area while avoiding highly populated areas, areas with difficult access or poor lighting, and crowded parking areas. The driver will then notify the State Police and the central dispatcher of the truck's location.

State-designated routes are preferred routes, selected in accordance with DOT guidelines (49 CFR Part 397.101[b] and [c]) or an equivalent routing analysis which adequately considers the overall impact to the public. The designation of routes must be preceded by substantive consultation with affected local jurisdictions and with any other states along such routes to ensure the consideration of impacts and the continuity of designated routes.

A state routing agency is an entity that is authorized to use a state legal process pursuant to 49 CFR 397, Subpart D to impose routing requirements, enforceable by state agencies, on carriers of radioactive material without regard to intrastate jurisdictional boundaries. This would include a common agency of more than one state such as one established by interstate compact. This term also includes Indian tribal authorities with police power to regulate and enforce highway routing requirements within their lands.

DOT regulations in 49 CFR 397, Subpart D provide routing and training requirements for carriers of radioactive material to ensure that the vehicles used for such transportation operate on routes that would minimize potential radiological impacts. Deviations from the state-designated route are allowed for necessary rest, fuel, and vehicle repair stops; to pick up, deliver, or transfer radioactive materials; and for emergency conditions that would make continued use of the preferred route unsafe. As required by 49 CFR Part 397.101(g), the carrier must prepare a route plan and supply a copy of the plan to the Federal Highway Administration, Traffic Control Division, the driver, and the shipper. Any deviation from the preferred route, and the reason for it, must be reported in an amendment to the route plan within thirty days following the shipment.

E.2.3 Proposed Routes

The proposed routes for transporting TRU waste by truck, as determined using HIGHWAY, are shown in Figure E-1. Figure E-2 details the routes through New Mexico. For the purposes of SEIS-II analyses, routes were selected based on the preferred routes defined in 49 CFR Part 173.403(l) and to be consistent with existing routing practices and applicable routing regulations and guidelines. These routes may not be the actual routes that would be used to transport TRU waste in the future. Details on the routes through New Mexico and from the 10 sites with the greatest number of shipments of waste to WIPP are presented below.

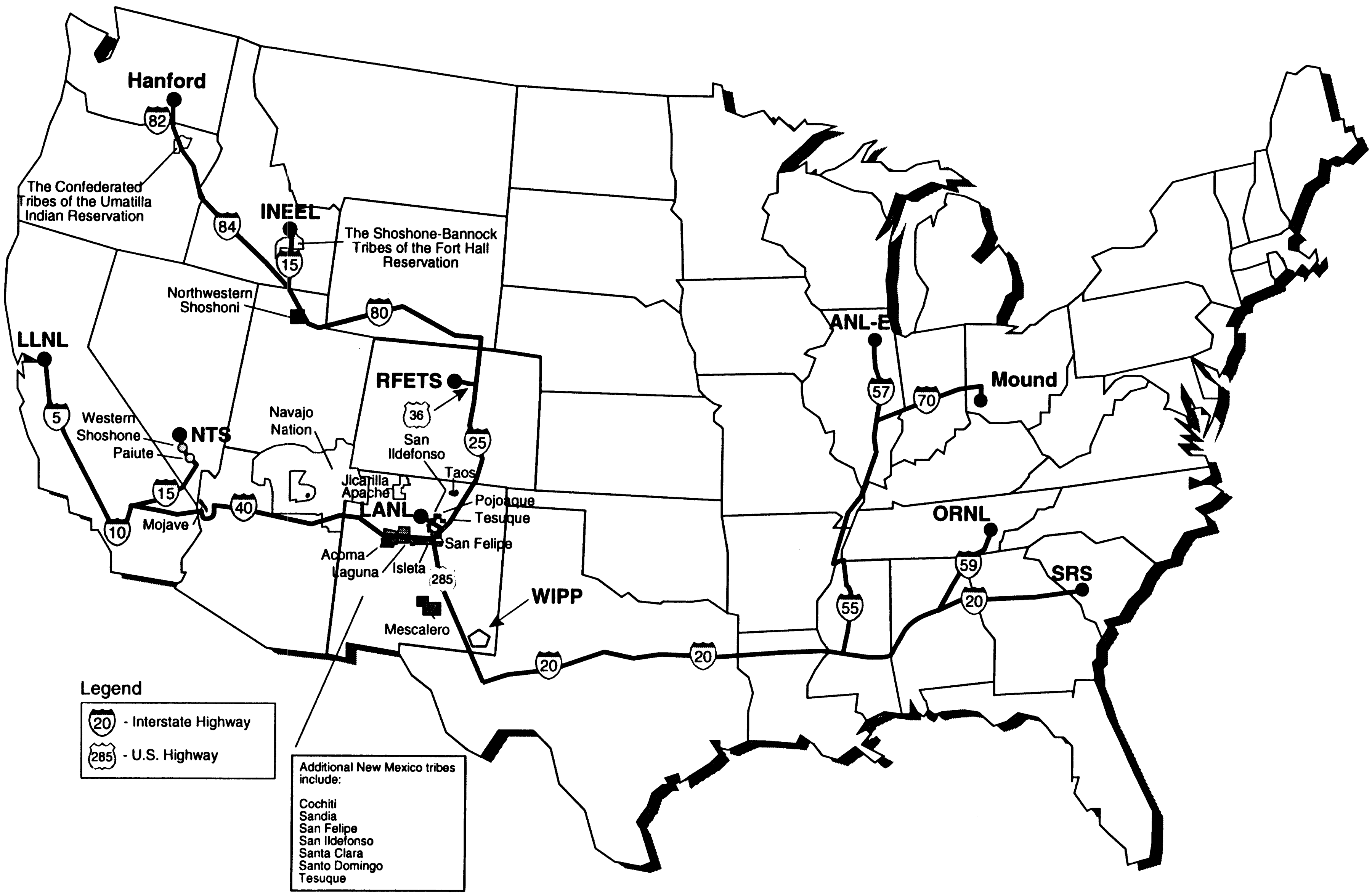

Figure E-1
Proposed TRU Waste Truck Transportation Routes on U.S. and Interstate Highways

Figure E-2
Proposed TRU Waste Truck Transportation Routes in New Mexico

Routes through the State of New Mexico. As shown in Figure E-1, all transportation routes converge in New Mexico. Transportation and routing have been identified in several agreements with the State of New Mexico. The agreements recognize that movements between interstate highways and WIPP would involve local highways, and because New Mexico is the host state, these highways would have a relatively concentrated service. Therefore, DOE agreed to support the state in efforts to obtain congressional funds necessary to repair and upgrade designated highway segments.

Route from Mound, Ohio. The proposed route is as follows:

Local Road to State 725, in Miamisburg, OH, 1.6 kilometers (1 mile)
State 725 to I-75, near Miamisburg, OH, 5 kilometers (3 miles)
I-75 to I-70, near Vandalia, OH, 29 kilometers (18 miles)
I-70 to I-465, southeast of Indianapolis, IN, 172 kilometers (107 miles)
I-465 to I-70, southwest of Indianapolis, IN, 32 kilometers (20 miles)
I-70 to I 57, near Teutopolis, IL, 211 kilometers (131 miles)
I-57 to I-55, near Sikeston, MO, 290 kilometers (180 miles)
I-55 to I-220, north of Jackson, MS, 541 kilometers (336 miles)
I-220 to I-20, west of Jackson, MS, 18 kilometers (11 miles)
I-20 to I-220, east of Shreveport, LA, 328 kilometers (204 miles)
I-220 to I-20, around the north side of Shreveport, LA, 29 kilometers (18 miles)
I-20 to US-285, at Pecos, TX, 978 kilometers (608 miles)
US-285 to US 180/62, at Carlsbad, NM, 137 kilometers (85 miles)
US 180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from ANL-E, Illinois. The proposed route is as follows:

Cass Avenue to I-55, north of ANL-E, 1.6 kilometers (1 mile)
I-55 to I-294, southwest of Chicago, IL, 8 kilometers (5 miles)
I-294 to I-80, south of Chicago, IL, 29 kilometers (18 miles)
I-80 to I-57, near Tinley Park, 8 kilometers (5 miles)
I-57 to I-55, near Sikeston, MO, 592 kilometers (368 miles)
I-55 to I-220, north of Jackson, MS, 541 kilometers (336 miles)
I-220 to I-20, west of Jackson, MS, 18 kilometers (11 miles)
I-20 to I-220, east of Shreveport, LA, 328 kilometers (204 miles)
I-220 to I-20, around the north side of Shreveport, LA, 29 kilometers (18 miles)
I-20 to US-285, at Pecos, TX, 978 kilometers (608 miles)
US-285 to US 180/62, at Carlsbad, NM, 137 kilometers (85 miles)
US 180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from ORNL, Tennessee. The proposed route is as follows:

Bethel Valley Road to State 95, west of ORNL, 3 kilometers (2 miles)
State 95 to I-40, south of Oak Ridge, TN, 5 kilometers (3 miles)
I-40 to I-75, southwest of Knoxville, TN, 6 kilometers (4 miles)
I-75 to I-24, east of Chattanooga, TN, 133 kilometers (83 miles)
I-24 to I-59, southwest of Chattanooga, TN, 31 kilometers (19 miles)
I-59 to I-459, northeast of Birmingham, AL, 195 kilometers (121 miles)

I-459 to I-20, southwest of Birmingham, AL, 53 kilometers (33 miles)
I-20 to I-220, east of Shreveport, LA, 684 kilometers (425 miles)
I-220 to I-20, around the north side of Shreveport, LA, 29 kilometers (18 miles)
I-20 to US-285, at Pecos, TX, 978 kilometers (608 miles)
US-285 to US-180/62, at Carlsbad, NM, 137 kilometers (85 miles)
US-180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from Savannah River Site (SRS), South Carolina. The proposed route is as follows:

SRS access road to State 19, north of SRS, 14 kilometers (9 miles)
State 19 to I-20, north of Aiken, SC, 29 kilometers (18 miles)
I-20 to I-285, east of Atlanta, GA, 249 kilometers (155 miles)
I-285 to I-20, around the south side of metro Atlanta, 45 kilometers (28 miles)
I-20 to I-459, northeast of Birmingham, AL, 204 kilometers (127 miles)
I-459 to I-20, southwest of Birmingham, AL, 47 kilometers (29 miles)
I-20 to I-220, east of Shreveport, LA, 684 kilometers (425 miles)
I-220 to I-20, around the north side of Shreveport, LA, 29 kilometers (18 miles)
I-20 to US-285, at Pecos, TX, 978 kilometers (608 miles)
US-285 to US-180/62, at Carlsbad, NM, 137 kilometers (85 miles)
US-180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from Hanford Site (Hanford), Washington. The proposed route is as follows:

Route 4S to State 240, in Hanford Reservation, 6 kilometers (4 miles)
State 240 to I-182, in Richland, WA, 11 kilometers (7 miles)
I-182 to I-82, southwest of Richland, WA, 8 kilometers (5 miles)
I-82 to I-84, near Hermiston, OR, 66 kilometers (41 miles)
I-84 to I-80, near Echo, UT, 949 kilometers (590 miles)
I-80 to I-25, at Cheyenne, WY, 626 kilometers (389 miles)
I-25 to US-285, southeast of Santa Fe (near Lamy), 767 kilometers (477 miles)
US-285 to US-180/62, at Carlsbad, NM, 414 kilometers (257 miles)
US-180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from Idaho National Engineering and Environmental Laboratory (INEEL), Idaho. The proposed route is as follows:

Plant Road to US-26, on INEEL Reservation, 1.6 kilometers (1 mile)
US-26 to I-15, near Blackfoot, ID, 64 kilometers (40 miles)
I-15 to I-84, southwest of Ogden, UT, 243 kilometers (151 miles)
I-84 to I-80, near Echo, UT, 63 kilometers (39 miles)
I-80 to I-25, at Cheyenne, WY, 626 kilometers (389 miles)
I-25 to US-285, southeast of Santa Fe (near Lamy), 767 kilometers (477 miles)
US-285 to US-180/62, at Carlsbad, NM, 414 kilometers (257 miles)
US-180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from Rocky Flats Environmental Technology Site (RFETS), Colorado. The proposed route is as follows:

Rocky Flats Access Road to State 93, west of RFETS, 3 kilometers (2 miles)
State 93 to State 128, south of Boulder, CO, 5 kilometers (3 miles)
State 128 to US-36, near Broomfield, CO, 13 kilometers (8 miles)
US-36 to I-25, north of Denver, CO, 14 kilometers (9 miles)
I-25 to US-285, southeast of Santa Fe (near Lamy), 767 kilometers (477 miles)
US-285 to US-180/62, at Carlsbad, NM, 414 kilometers (257 miles)
US-180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from Los Alamos National Laboratory (LANL), New Mexico. The proposed route is as follows:

Los Alamos Truck Route to State 4, east of Los Alamos, 10 kilometers (6 miles)
State 4 to State 502, east of Los Alamos, NM, 1.6 kilometers (1 mile)
State 502 to US-84, at Pojoaque, NM, 19 kilometers (12 miles)
US-84 to I-25, south of Santa Fe, NM, 32 kilometers (20 miles)
I-25 to US-285, southeast of Santa Fe (near Lamy), 13 kilometers (8 miles)
US-285 to US-180/62, at Carlsbad, NM, 414 kilometers (257 miles)
US-180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from LLNL, California. The proposed route is as follows:

Local Road to I-580, northeast of Livermore, CA, 5 kilometers (3 miles)
I-580 to I-5, near Vernalis, CA, 39 kilometers (24 miles)
I-5 to I-210, north of Los Angeles, CA, 468 kilometers (291 miles)
I-210 to I-10, near Pomona, CA, 77 kilometers (48 miles)
I-10 to I-15, in Ontario, CA, 27 kilometers (17 miles)
I-15 to I-40, in Barstow, CA, 119 kilometers (74 miles)
I-40 to US-285, near Clines Corner, NM, 1,191 kilometers (740 miles)
US-285 to US-180/62, at Carlsbad, NM, 349 kilometers (217 miles)
US-180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

Route from Nevada Test Site (NTS), Nevada. The proposed route, which is currently being renegotiated with the state, is as follows:

Local Road to US-95, south of Mercury, NV, 10 kilometers (6 miles)
US-95 to State Road 373, near Amargon Valley, NV, 39 kilometers (24 miles)
State Road 373 to State Road 127, near Shosone, CA, 81 kilometers (50 miles)
State Road 127 to I-15, near Baker, CA, 90 kilometers (56 miles)
I-15 to I-40, in Barstow, CA, 101 kilometers (63 miles)
I-40 to US-285, near Clines Corner, NM, 1,191 kilometers (740 miles)
US-285 to US-180/62, at Carlsbad, NM, 349 kilometers (217 miles)
US-180/62 to WIPP North Access Road, 43 kilometers (27 miles)
WIPP North Access Road to WIPP, 18 kilometers (11 miles)

E.3 NONRADIOLOGICAL IMPACTS FROM TRUCK TRANSPORTATION

This section addresses the impacts of traffic accidents and vehicle emissions associated with transporting TRU waste to WIPP. These impacts are not related to the radioactive material or hazardous chemicals being transported and would be the same as the impacts resulting from the transportation of nonhazardous material. Accident impacts were calculated as the number of injuries and fatalities that would be expected due to additional truck traffic along the proposed routes, and were calculated on a per-shipment basis, then totaled for all shipments over the transportation period. Calculations were based on data presented in *Longitudinal Review of State-Level Accident Statistics for Carriers of Interstate Freight* (Saricks and Kvitek 1994). These data, route mileages, and the number of shipments along each route were used to determine the aggregate impacts from transporting TRU waste. Impacts from vehicle emissions (pollution-related health effects) were calculated as the number of excess latent cancer fatalities (LCF) in the exposed population due to truck exhaust emissions.

All of the analyses were based on the impact-per-shipment from the major generator-storage sites to WIPP. Results were adjusted to account for small quantity site shipments to the major generator-storage sites (see Section E.6).

E.3.1 Methodology

Whenever material is shipped, the possibility of a traffic accident which could result in vehicular damage, injury, or death exists. Even when drivers are trained in defensive driving and take great care, there is a risk of a traffic accident. To date, shipments of empty TRUPACT-IIs have logged more than 1,529,000 kilometers (950,000 miles) without accident. However, for the SEIS-II analyses, truck accident statistics that were compiled for each state by highway type were used (Saricks and Kvitek 1994). No assumption was made that carrier drivers would be better trained or would be more careful than other truck drivers on the nation's roads.

It is important to note that the accident rates used in this assessment were computed using all interstate shipments, regardless of the cargo. Saricks and Kvitek (1994) point out that shippers and carriers of radioactive material generally have a higher-than-average awareness of transportation impacts and prepare for such shipments accordingly. Still, these effects were not considered. Separate accident rates for travel in rural, suburban, and urban population density zones in each state were used. The total accident impact for a case, therefore, depends on the total distance traveled in the various population zones within each state and does not rely on national-average accident statistics.

As discussed in Appendix A, the volumes of waste that are currently in storage and projected to be generated through the year 2033 were estimated from information provided in the *Transuranic Waste Baseline Inventory Report, Revision 3* (BIR-3) (DOE 1996b). Using the methods discussed in Appendix A, the volumes of CH-TRU and RH-TRU waste transported under the Proposed Action and each alternative were calculated. Also, the number of shipments needed to transport this waste were calculated on a site-by-site basis.

For truck shipments, it was assumed that CH-TRU waste would be packaged in Type A 55-gallon drums and transported in TRUPACT-IIs. Each TRUPACT-II would carry a maximum of two 7-packs of drums, and each shipment could carry two or three TRUPACT-II containers, a

PROPOSED USE OF HALFPACKS FOR TRANSPORTATION

A new CH-TRU waste packaging, the HALFPACK, is being developed by DOE to transport heavier-than-average drums of CH-TRU waste to WIPP. The role of the HALFPACK is to supplement the existing TRUPACT-II packaging by efficiently transporting heavy drums weighing up to 454 kilograms (1,000 pounds). The TRUPACT-II was specifically developed to transport 14 drums, each weighing an average of 227 kilograms (500 pounds).

The ability of the HALFPACK to efficiently transport heavy drums was achieved by reducing the height of the existing TRUPACT-II design by 70 centimeters (30 inches). This reduction in weight by approximately 771 kilograms (1,700 pounds) brings the empty weight of a HALFPACK to 4,581 kilograms (10,100 pounds). Although the capacity of the HALFPACK would be reduced to seven 55-gallon drums, the drums could weigh 454 kilograms (1,000 pounds) each, for a total payload capacity of 3,295 kilograms (7,265 pounds) including the pallet assembly. In accordance with the weight distribution restrictions placed on legal-weight transport truck and trailer vehicles by the U.S. Department of Transportation, the weight reduction will allow three HALFPACKs to carry 21 heavy drums per shipment instead of 14 heavy drums per shipment with TRUPACT-IIs, thereby reducing the total number of shipments overall. The table below shows the reduction in the total number of shipments under each alternative in SEIS-II if the HALFPACK were used from the onset of the transportation activities. The estimated reduction would range from more than 40 percent under Action Alternatives 2A and 2B to approximately 10 percent under Action Alternatives 1 and 2C.

Reduction in Shipments for Each Alternative by Using HALFPACK

	Number of Shipments to WIPP for the Total Inventory			
Alternatives	**With HALFPACK**	**TRUPACT-II Only**	**Difference (m3)**	**Percent Difference**
Proposed Action				
CH-TRU Waste	24,962	29,766	(4,804)	84%
Action Alternative 1				
CH-TRU Waste	36,938	41,003	(4,065)	90%
Action Alternative 2A				
CH-TRU Waste	24,444	42,775	(18,331)	57%
Action Alternative 2B				
CH-TRU Waste	24,444	42,774	(18,331)	57%
Action Alternative 2C				
CH-TRU Waste	37,082	41,206	(4,124)	90%
Action Alternative 3				
CH-TRU Waste	52,484	67,309	(14,825)	78%

Another benefit of the HALFPACK is its ability to transport overpack drums. An overpack drum consists of a 55-gallon drum of questionable integrity placed inside a new 85-gallon drum. The internal cavity of the HALFPACK can accommodate four of the taller overpack drums and therefore avoids the costs of repackaging and re-certification of the waste into new 55-gallon drums.

The HALFPACK will be developed in accordance with the criteria for Type B packaging by the Nuclear Regulatory Commission (NRC) as set forth in 10 CFR Part 71. Once the performance and certification criteria have been demonstrated, a Safety Analysis Report will be presented to the NRC for approval (currently scheduled for July 1998). Upon NRC approval of the HALFPACK, a Certificate of Compliance would be issued to DOE that would grant the authority to use and operate the HALFPACK.

maximum of 42 drums per shipment. The number of drums per TRUPACT-II would vary depending on waste restrictions. These restrictions include limits on weight, volume, thermal power, plutonium-239 equivalent curies (PE-Ci), and fissile-gram equivalents. It was assumed that RH-TRU waste would be transported in RH-72B casks, with only one cask per shipment. The number of CH-TRU and RH-TRU waste shipments to WIPP under the Proposed Action and the action alternatives are shown in Tables E-1 and E-2, respectively. Shipments under No Action Alternative 1 reflect the number of shipments needed to consolidate TRU waste at treatment sites. TRU waste is not shipped to WIPP under this alternative.

Under the Proposed Action and the action alternatives, there would be shipments from the small quantity sites to some of the 10 major generator-storage sites to consolidate waste prior to shipment to WIPP. These small quantity sites and the number of shipments are presented in Table E-3. Since none of the sites presented in Table E-3 would treat their waste before shipment, the number of shipments would be the same under the Proposed Action and the action alternatives. Under Action Alternatives 2 and 3, and No Action Alternative 1, some of the ten major generator-storage sites would ship their waste to one of the other major sites prior to shipment to WIPP for consolidation or treatment. These shipment numbers are presented in Table E-4.

The HIGHWAY code was used to estimate the mileages from the various sites to WIPP and to estimate the corresponding population density fractions. Additionally, mileages from the small quantity sites to the major generator sites were determined using HIGHWAY. The TRU waste origin and destination sites, total route miles, and the miles within each population zone for the 10 major generator-storage sites shipping CH-TRU waste are presented in Table E-5. This table also indicates the alternative under which the route would apply. Similar information is presented in Table E-6 for RH-TRU waste shipments. In Table E-7, the destination sites and mileages through each population zone for the small quantity sites are given.

The rural, suburban, and urban population route mileages along a given route were multiplied by state-specific rural, suburban, and urban accident, injury, or fatality rates to obtain route-specific impacts. The impacts were then summed over the route and divided by the total route mileage. For all but the State of New Mexico, the accident rate data for federally-aided interstate highways were used. For New Mexico, much of the waste travels on US-285; therefore, the rate data for federally-aided primary highways were used.

Route-specific per shipment accident, injury, and fatality rates (see Table E-8) were multiplied by the appropriate number of route shipments (Tables E-1 and E-2) to obtain the aggregate number of accidents, injuries, and fatalities. These impacts are shown in Table E-9. Because this analysis is not dependent on whether a truck is transporting full or empty TRU waste containers, twice the one-way mileage was used.

The distance traveled in an urban population zone and the impact factor for particulate and sulfur dioxide truck emissions (Rao et al. 1982) were used to estimate additional urban-area pollution health effects due to TRU waste shipments. The impact factor, 9.9×10^{-8} LCFs per kilometer (1.6×10^{-7} LCFs per mile), estimates the number of LCFs per urban mile traveled. The volume of particulates and sulfur dioxide emitted in an urban area by a single truck shipment would be quite small. A million or more simultaneous pollutant-generating shipments would be needed to achieve the minimum pollutant volume of particulates and sulfur dioxide required to cause one LCF. The LCFs attributed to diesel exhaust exposure in an urban area are very small relative to the impact of accidents, fatalities, or injuries.

Table E-1
Number of Shipments for CH-TRU Waste

Waste Origin Site	Proposed Action Scaled to WIPP Max.	Action Alternative 1			Action Alternative 2A			Action Alternative 2B			Action Alternative 2C		
	Basic Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
Argonne National Laboratory -East	28	24	---	24	---	---	---	---	---	---	22	---	22
Hanford Site	13,666	11,562	7,167	18,729	8,230	8,813	17,043	8,230	8,813	17,043	11,562	7,194	18,756
Idaho National Engineering and Environmental Laboratory/Argonne National Laboratory-West	5,782	4,892	6,474	11,366	4,178	12,388	16,566	8,653	14,335	22,988	4,892	6,639	11,531
Lawrence Livermore National Laboratory	162	137	---	137	---	---	---	---	---	---	137	---	137
Los Alamos National Laboratory	5,009	4,238	1,590	5,828	2,952	1,947	4,899	---	---	---	4,236	1,590	5,826
Mound Plant	59	50	23	73	---	---	---	---	---	---	50	3	53
Nevada Test Site	86	73	---	73	---	---	---	---	---	---	73	---	73
Oak Ridge National Laboratory	251	212	8	220	---	---	---	---	---	---	192	8	200
Rocky Flats Environmental Technology Site	2,485	2,102	---	2,102	1,524	---	1,524	---	---	---	2,102	---	2,102
Savannah River Site	2,238	1,893	558	2,451	2,020	723	2,743	2,020	723	2,743	1,893	558	2,451
Total	**29,766**	**25,183**	**15,820**	**41,003**	**18,904**	**23,871**	**42,775`**	**18,903**	**23,871**	**42,774**	**25,159**	**15,992**	**41,151**

Note: Shipments for No Action Alternatives 1A and 1B are the number of shipments to consolidate TRU waste from large generator sites to treatment sites. No TRU waste would be shipped to WIPP.

Table E-1
Number of Shipments for CH-TRU Waste — Continued

Waste Origin Site	Action Alternative 3			No Action Alternative 1A			No Action Alternative 1B		
	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
Argonne National Laboratory - East	---	---	---	22	---	22	22	---	22
Hanford Site	24,531	8,600	33,131	---	---	---	---	---	---
Idaho National Engineering and Environmental Laboratory / Argonne National Laboratory-West	10,386	7,769	18,155	---	---	---	---	---	---
Lawrence Livermore National Laboratory	---	---	---	137	---	137	137	---	137
Los Alamos National Laboratory	7,628	1,907	9,535	---	---	---	4,236	1,590	5,826
Mound Plant	---	---	---	50	3	53	50	3	53
Nevada Test Site	---	---	---	73	---	73	73	---	73
Oak Ridge National Laboratory	---	---	---	192	8	200	192	8	200
Rocky Flats Environmental Technology Site	2,897	---	2,897	---	---	---	2,102	---	2,102
Savannah River Site	2,885	706	3,591	---	---	---	---	---	---
Total	**48,327**	**18,982**	**67,309**	**474**	**11**	**485**	**6,812**	**1601**	**8,413**

Note: Shipments for No Action Alternatives 1A and 1B are the number of shipments to consolidate TRU waste from large generator sites to treatment sites. No TRU waste would be shipped to WIPP.

Table E-2
Number of Shipments for RH-TRU Waste

Waste Origin Site	Proposed Action	Alternative 1			Alternative 2A			Alternative 2B			Alternative 2C		
	Basic Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
Hanford Site	3,178	47,156	1,651	48,807	17,730	1,031	18,761	17,730	1,031	18,761	17,730	1,031	18,761
Idaho National Engineering and Environmental Laboratory / Argonne National Laboratory-West	3,136	3,136	711	3,847	---	---	---	---	---	---	---	---	---
Los Alamos National Laboratory	367	367	190	557	---	---	---	---	---	---	---	---	---
Oak Ridge National Laboratory	1,276	5,875	3,076	8,951	2,057	1,077	3,134	2,057	1,077	3,134	2,057	1,077	3,134
Total	**7,957**	**56,534**	**5,628**	**62,162**	**19,787**	**2,108**	**21,895**	**19,787**	**2,108**	**21,895**	**19,787**	**2,108**	**21,895**

Note: Shipments for No Action Alternatives 1A and 1B are the number of shipments to consolidate TRU waste from large generator sites to treatment sites. No TRU waste would be shipped to WIPP.

Table E-2
Number of Shipments for RH-TRU Waste — Continued

Waste Origin Site	Action Alternative 3			No Action Alternative 1A			No Action Alternative 1B		
	Basic Inventory	**Additional Inventory**	**Total Inventory**	**Basic Inventory**	**Additional Inventory**	**Total Inventory**	**Basic Inventory**	**Additional Inventory**	**Total Inventory**
Hanford Site	60,789	3,076	63,865	---	---	---	---	---	---
Idaho National Engineering and Environmental Laboratory / Argonne National Laboratory-West	---	---	---	3,136	711	3,847	3,136	711	3,847
Los Alamos National Laboratory	---	---	---	367	190	557	367	190	557
Oak Ridge National Laboratory	7,050	3,691	10,741	---	---	---	---	---	---
Total	**67,839**	**6,767**	**74,606**	**3,503**	**901**	**4,404**	**3,503**	**901**	**4,404**

Note: Shipments for No Action Alternatives 1A and 1B are the number of shipments to consolidate TRU waste from large generator sites to treatment sites. No TRU waste would be shipped to WIPP.

Table E-3
Number of CH-TRU and RH-TRU Waste Shipments to Consolidation/Treatment Sites from Sites with Smaller Quantities of TRU Waste

Origin Site with Smaller Quantities of Waste	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste			
Ames Laboratory	1	0	1
ARCO	0	1	1
Bettis Atomic Power Laboratory	20	0	20
Energy Technology Engineering Center	1	0	1
Lawrence Berkeley Laboratory	0	1	1
Paducah Gaseous Diffusion Plant	1	0	1
Sandia National Laboratories	2	1	3
University of Missouri	1	0	1
USAMC	1	0	1
West Valley Demonstration Project	0	23	23
Total	27	26	53
RH-TRU Waste			
Battelle Columbus Laboratory	931	0	931
Bettis Atomic Power Laboratory	15	0	15
Energy Technology Engineering Center	12	0	12
Knolls	0	130	130
West Valley Demonstration Project	0	2,762	2,762
Total	958	2,892	3,850

E.3.2 Results

Table E-9 summarizes the aggregate nonradiological impacts (accidents, injuries, fatalities, and pollution-related LCFs) associated with the transportation of TRU waste from the 10 major generator-storage sites by truck to WIPP. Impacts would be dependent on the number of shipments, which in turn, would be dependent on waste inventories and the differences in consolidation schemes and treatment options.

E.4 RADIOLOGICAL IMPACTS FROM TRUCK TRANSPORTATION

In this section, the impact analyses of the transportation of radioactive material are discussed. The impacts fall into two general categories, radiological impacts under normal transportation conditions and radiological impacts in the event of an accident.

These analyses were conducted in a manner consistent with WIPP-specific transportation analyses in SEIS-I (DOE 1990), the *Engineered Alternatives Cost/Benefit Study Final Report* (DOE 1995b), the WM PEIS (DOE 1997), and the *Comparative Study of Waste Isolation Pilot Plant (WIPP) Transportation Alternatives* (DOE 1994). The methods used were established by the NRC in the late 1970s.

Table E-4
CH-TRU and RH-TRU Waste Shipments from Major Generator-Storage Sites to Treatment Sites

Waste Origin Site	Action Alternative 2A			Action Alternative 2B			Action Alternative 2C		
	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste									
Argonne National Laboratory - East	22	---	22	22	---	22	---	---	---
Idaho National Engineering and Environmental Laboratory / Argonne National Laboratory-West	---	---	---	---	---	---	---	---	---
Lawrence Livermore National Laboratory	137	---	137	137	---	137	---	---	---
Los Alamos National Laboratory	---	---	---	4,236	1,590	5,826	---	---	---
Mound Plant	50	3	53	50	3	53	---	---	---
Nevada Test Site	73	---	73	73	---	73	---	---	---
Oak Ridge National Laboratory	192	8	200	192	8	200			
Rocky Flats Environmental Technology Site	---	---	---	2,102	---	2,102	---	---	---
Savannah River Site	---	---	---	---	---	---	---	---	---
Total	**474**	**11**	**485**	**6,812**	**1,601**	**8,413**	**---**	**---**	**---**
RH-TRU Waste									
Argonne National Laboratory - East	---	---	---	---	---	---	---	---	---
Idaho National Engineering and Environmental Laboratory / Argonne National Laboratory-West	3,136	711	3,847	3,136	711	3,847	3,136	711	3,847
Lawrence Livermore National Laboratory	---	---	---	---	---	---	---	---	---
Los Alamos National Laboratory	367	190	557	367	190	557	367	190	557
Mound Plant	---	---	---	---	---	---	---	---	---
Nevada Test Site	---	---	---	---	---	---	---	---	---
Rocky Flats Environmental Technology Site	---	---	---	---	---	---	---	---	---
Savannah River Site	---	---	---	---	---	---	---	---	---
Total	**3,503**	**901**	**4,404**	**3,503**	**901**	**4,404**	**3,503**	**901**	**4,404**

Table E-4
CH-TRU and RH-TRU Waste Shipments From Major Generator-Storage Sites to Treatment Sites — Continued

Waste Origin Site	Action Alternative 3			No Action Alternative 1A			No Action Alternative 1B		
	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste									
Argonne National Laboratory - East	22	---	22	22	---	22	22	---	22
Idaho National Engineering Laboratory Argonne National Laboratory-West	---	---	---	---	---	---	---	---	---
Lawrence Livermore National Laboratory	137	---	137	137	---	137	137	---	137
Los Alamos National Laboratory	---	---	---	---	---	---	4,236	1,590	5,826
Mound Plant	50	---	50	50	3	53	50	3	53
Nevada Test Site	73	---	73	73	---	73	73	---	73
Oak Ridge National Laboratory	192	8	200	192	8	200	192	8	200
Rocky Flats Environmental Technology Site	---	---	---	---	---	---	2,102	---	2,102
Savannah River Site	---	---	---	---	---	---	---	---	---
Total	**474**	**8**	**482**	**474**	**11**	**485**	**6,812**	**1,601**	**8,413**
RH-TRU Waste									
Argonne National Laboratory - East	---	---	---	---	---	---	---	---	---
Idaho National Engineering Laboratory Argonne National Laboratory-West	3,136	711	3,847	3,136	711	3,847	3,136	711	3,847
Lawrence Livermore National Laboratory	---	---	---	---	---	---	---	---	---
Los Alamos National Laboratory	367	190	557	367	190	557	367	190	557
Mound Plant	---	---	---	---	---	---	---	---	---
Nevada Test Site	---	---	---	---	---	---	---	---	---
Rocky Flats Environmental Technology Site	---	---	---	---	---	---	---	---	---
Savannah River Site	---	---	---	---	---	---	---	---	---
Total	**3,503**	**901**	**4,404**	**3,503**	**901**	**4,404**	**3,503**	**901**	**4,404**

Table E-5
CH-TRU Waste Transportation in Miles (kilometers)

Originating Site	Destination Site	Total One-way Truck Mileage [a]	Population Zone: Rural	Suburban	Urban	Applicable to the Following
Argonne National Laboratory-East	WIPP	1,696 (2,729)	1,412 (2,272)	259 (417)	25 (40)	Proposed Action, Action Alternative 1, and Action Alternative 2C
	SRS	877 (1,411)	587 (945)	266 (428)	24 (38)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
Hanford Site	WIPP	1,807 (2,908)	1,645 (2,647)	144 (232)	18 (29)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, and Action Alternative 3
Idaho National Engineering and Environmental Laboratory/ANL-W	WIPP	1,392 (2,241)	1,263 (2,033)	114 (184)	15 (24)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, and Action Alternative 3
Lawrence Livermore National Laboratory	WIPP	1,452 (2,337)	1,304 (2,099)	100 (161)	48 (77)	Proposed Action, Action Alternative 1, and Action Alternative 2C
	Hanford	890 (1,432)	675 (1,086)	184 (296)	31 (50)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
Los Alamos National Laboratory	WIPP	341 (549)	318 (512)	21 (34)	2 (3)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2C, and Action Alternative 3
	INEEL	1,145 (1,843)	1,025 (1,650)	104 (167)	16 (26)	Action Alternative 2B, and No Action Alternative 1B
Mound Plant	WIPP	1,764 (2,838)	1,359 (2,187)	382 (614)	23 (37)	Proposed Action, Action Alternative 1, and Action Alternative 2C
	SRS	639 (1,028)	424 (682)	205 (330)	10 (16)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
Nevada Test Site	WIPP	1,214 (1,954)	1,137 (1,830)	64 (103)	13 (21)	Proposed Action, Action Alternative 1, and Action Alternative 2C
	INEEL	712 (1,146)	600 (966)	92 (148)	20 (32)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
Oak Ridge National Laboratory	WIPP	1,439 (2,316)	1,160 (1,867)	265 (426)	14 (23)	Proposed Action, Action Alternative 1, and Action Alternative 2C
	SRS	357 (575)	245 (394)	109 (175)	3 (5)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
Rocky Flats Environmental Technology Site	WIPP	704 (1,133)	619 (996)	71 (114)	14 (23)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2C, and Action Alternative 3
	INEEL	732 (1,178)	669 (1,077)	54 (87)	9 (14)	Action Alternative 2B, and No Action Alternative 1B
Savannah River Site	WIPP	1,535 (2,470)	1,203 (1,936)	315 (507)	17 (27)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, and Action Alternative 3

[a] Total mileages shown may differ from the sum of the population mileages due to rounding.

Table E-6
RH-TRU Waste Transportation in Miles (kilometers)

Originating Site	Destination Site	Total One-way Truck Mileage [a]	Population Zone			Applicable to the Following
			Rural	Suburban	Urban	
Hanford (Richland) Site	WIPP	1,807 (2,908)	1,645 (2,647)	144 (232)	18 (29)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, and Action Alternative 3
Idaho National Engineering and Environmental Laboratory/ANL-W	WIPP	1,392 (2,240)	1,263 (2,033)	114 (183)	15 (24)	Proposed Action, and Action Alternative 1
	Hanford	600 (966)	550 (885)	47 (76)	3 (5)	Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
Los Alamos National Laboratory	WIPP	342 (550)	318 (512)	22 (35)	2 (3)	Proposed Action, and Action Alternative 1
	Hanford	1,560 (2,511)	1,407 (2,264)	135 (217)	18 (29)	Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
Oak Ridge National Laboratory	WIPP	1,438 (2,314)	1,160 (1,867)	265 (426)	13 (21)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, and Action Alternative 3
Savannah River Site	ORNL	357 (575)	245 (394)	109 (175)	3 (5)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B, No Action Alternative 2

[a] Total mileages shown may differ from the sum of the population mileages due to rounding.

The computer codes used for these analyses have been extensively documented elsewhere (Johnson et al. 1993a and 1993b, NRC 1977, and Neuhauser and Kanipe 1992). MICROSHIELD 4.21 was used to calculate the transportation index (TI) (radiation exposure rate at 1 meter [3.3 feet]) and RADTRAN was used to calculate radiological impacts (Neuhauser and Kanipe 1992). RADTRAN is the product of almost 15 years of development and is a flexible analytical tool for calculating the population impacts under both normal transportation and transportation accidents. The major RADTRAN input parameters for truck transportation are summarized in Table E-10. The RISKIND computer code (ANL 1995) was used to determine the maximally exposed individual (MEI) in an accident.

To evaluate the radiological impacts of accidents, it was necessary to consider the probability of an accident occurring and the potential consequences of that accident. This analysis included the following steps:

- Identifying the physical, chemical, and radiological characteristics of the waste
- Identifying the system that would be used for shipping (types of shipping containers, number of containers per shipment, etc.)
- Identifying potential accident scenarios in which radioactive material may be released

Table E-7
Mileages (kilometers) to Treatment Sites from Sites with Small Quantities of TRU Waste

Originating Site	Destination Site	Total One-way Truck Mileage [a]	Population Zone Rural	Suburban	Urban	Applicable to the Following
CH-TRU Waste						
Ames Laboratory	ORNL	893 (1,438)	669 (1,077)	209 (337)	15 (24)	Proposed Action, Action Alternative 1, and No Action Alternative 2
	INEEL	1,293 (2,081)	1,199 (1,930)	85 (137)	9 (15)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
	WIPP	1,255 (2,020)	1,121 (1,804)	118 (190)	16 (26)	Action Alternative 2C
Bettis Atomic Power Laboratory	ORNL	606 (975)	415 (668)	178 (286)	13 (21)	Proposed Action, Action Alternative 1, and No Action Alternative 2
	SRS	686 (1,104)	485 (781)	188 (303)	12 (19)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
	WIPP	1,803 (2,902)	1,453 (2,338)	318 (512)	31 (50)	Action Alternative 2C
U.S. Army Materiel Command	ORNL	701 (1,128)	497 (800)	191 (307)	13 (21)	Proposed Action, Action Alternative 1, and No Action Alternative 2
	INEEL	1,429 (2,300)	1,323 (2,129)	98 (158)	8 (13)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
	WIPP	1,391 (2,239)	1,245 (2,004)	131 (211)	14 (23)	Action Alternative 2C
Paducah Gaseous Diffusion Plant	ORNL	316 (509)	252 (406)	60 (97)	4 (6)	Proposed Action, Action Alternative 1, and No Action Alternative 2
	SRS	587 (945)	393 (632)	178 (286)	17 (27)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
	WIPP	1,360 (2,189)	1,174 (1,889)	171 (275)	14 (23)	Action Alternative 2C
Energy Technology Engineering Center	NTS	375 (603)	269 (433)	61 (98)	45 (72)	Proposed Action, Action Alternative 1, and No Action Alternative 2
	INEEL	958 (1,542)	755 (1,216)	142 (229)	62 (100)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
	WIPP	1,151 (1,853)	997 (1,605)	101 (163)	53 (85)	Action Alternative 2C
Sandia National Laboratories	LANL	104 (167)	82 (132)	17 (27)	5 (8)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 3, and No Action Alternative 2
	INEEL	1,170 (1,884)	1,041 (1,676)	110 (177)	18 (29)	Action Alternative 2B and No Action Alternative 1B
	WIPP	311 (501)	288 (464)	19 (31)	4 (6)	Action Alternative 2C

[a] Total mileages shown may differ from the sum of the population mileages due to rounding.

Table E-7
Mileages (kilometers) to Treatment Sites from Sites with Small Quantities of TRU Waste — Continued

Originating Site	Destination Site	Total One-way Truck Mileage [a]	Population Zone Rural	Suburban	Urban	Applicable to the Following
CH-TRU Waste						
University of Missouri	ORNL	610 (982)	476 (766)	120 (193)	14 (23)	Proposed Action, Action Alternative 1, and No Action Alternative 2
	SRS	881 (1,418)	617 (993)	238 (383)	26 (42)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
	WIPP	1,145 (1,843)	1,018 (1,638)	110 (177)	18 (29)	Action Alternative 2C
ARCO Medical Products Company	ORNL	658 (1,059)	411 (662)	241 (388)	6 (10)	Action Alternative 1
	SRS	724 (1,165)	463 (745)	253 (407)	8 (13)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
	WIPP	2,056 (3,310)	1,608 (2,589)	420 (676)	27 (43)	Action Alternative 2C
Lawrence Berkeley Laboratory	Hanford	870 (1,400)	668 (1,075)	167 (269)	35 (56)	Action Alternative 1, Action Alternative 2B, Action Alternative 3, and No Action Alternative 1B
	WIPP	1,522 (2,450)	1,320 (2,125)	131 (211)	71 (114)	Action Alternative 2C
West Valley Demonstration Project	ORNL	749 (1,206)	467 (752)	265 (427)	17 (27)	Action Alternative 1
	SRS	902 (1,452)	635 (1,022)	259 (417)	9 (15)	Action Alternative 2A, Action Alternative 2B, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
RH-TRU Waste						
Battelle Columbus Laboratory	ORNL	395 (636)	269 (433)	117 (188)	9 (15)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, No Action Alternative 1A, Action Alternative 3, No Action Alternative 1B, and No Action Alternative 2
Bettis Atomic Power Laboratory	ORNL	606 (976)	415 (668)	178 (287)	13 (21)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, No Action Alternative 1A, Action Alternative 3, No Action Alternative 1B, and No Action Alternative 2
Knolls	ORNL	883 (1,422)	589 (948)	284 (457)	10 (16)	Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, Action Alternative 3, No Action Alternative 1A, and No Action Alternative 1B
Energy Technology Engineering Center	Hanford	1,203 (1,937)	982 (1,581)	179 (288)	41 (66)	Proposed Action, Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, No Action Alternative 1A, Action Alternative 3, No Action Alternative 1B, and No Action Alternative 2
Sandia National Laboratories	LANL	104 (167)	82 (132)	17 (27)	5 (8)	Proposed Action, Action Alternative 1, No Action Alternative 1B, and No Action Alternative 2
	Hanford	1,586 (2,553)	1,424 (2,293)	141 (227)	21 (34)	Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, Action Alternative 3, and No Action Alternative 1B
West Valley Demonstration Project	ORNL	749 (1,206)	467 (752)	265 (427)	17 (27)	Action Alternative 1, Action Alternative 2A, Action Alternative 2B, Action Alternative 2C, No Action Alternative 1A, Action Alternative 3, and No Action Alternative 1B

[a] Total mileages shown may differ from the sum of the population mileages due to rounding.

Table E-8
Nonradiological Impacts per Shipment
from Accidents and Pollution Related Health Effects

Originating Site	Destination Site	Impact Category	Population Zone			
			Rural	Suburban	Urban	Total [a]
Argonne National Laboratory - East	WIPP	Accidents	7.7E-4	3.6E-4	5.0E-5	1.2E-3
		Injuries	7.1E-4	1.8E-4	2.1E-5	9.1E-4
		Fatalities	9.1E-5	3.0E-5	3.4E-6	1.2E-4
		Vehicle Pollution (LCFs) [b]	---	---	8.0E-6	8.0E-6
	SRS	Accidents	3.3E-4	4.5E-4	4.6E-5	8.3E-4
		Injuries	3.3E-4	4.4E-4	4.4E-5	8.2E-4
		Fatalities	3.4E-5	3.3E-5	3.1E-6	7.0E-5
		Vehicle Pollution (LCFs) [b]	---	---	7.6E-6	7.6E-6
Hanford Site	WIPP	Accidents	1.6E-3	2.3E-4	3.2E-5	1.9E-3
		Injuries	1.0E-3	1.8E-4	2.6E-5	1.3E-3
		Fatalities	1.5E-4	1.3E-5	1.7E-6	1.7E-4
		Vehicle Pollution (LCFs) [b]	---	---	5.8E-6	5.8E-6
Idaho National Engineering and Environmental Laboratory/ ANL-W	WIPP	Accidents	1.4E-3	2.1E-4	3.0E-5	1.6E-3
		Injuries	8.2E-4	1.6E-4	2.4E-5	1.0E-3
		Fatalities	1.3E-4	1.2E-5	1.7E-6	1.5E-4
		Vehicle Pollution (LCFs) [b]	---	---	4.9E-6	4.9E-6
	Hanford	Accidents	4.0E-4	3.3E-5	2.1E-6	4.4E-4
		Injuries	3.5E-4	2.7E-5	1.7E-6	3.8E-4
		Fatalities	3.0E-5	9.8E-7	6.0E-8	3.1E-5
		Vehicle Pollution (LCFs) [b]	---	---	1.1E-6	1.1E-6
Lawrence Livermore National Laboratory	WIPP	Accidents	9.6E-4	1.4E-4	5.2E-5	1.2E-3
		Injuries	9.5E-4	1.3E-4	4.9E-5	1.1E-3
		Fatalities	1.4E-4	1.3E-5	4.4E-6	1.5E-4
		Vehicle Pollution (LCFs) [b]	---	---	1.5E-5	1.5E-5
	Hanford	Accidents	4.3E-4	1.8E-4	2.9E-5	6.4E-4
		Injuries	3.7E-4	1.7E-4	2.8E-5	5.7E-4
		Fatalities	3.7E-5	1.2E-5	1.9E-6	5.0E-5
		Vehicle Pollution (LCFs) [b]	---	---	9.9E-6	9.9E-6
Los Alamos National Laboratory	WIPP	Accidents	4.9E-4	3.3E-5	3.2E-6	5.2E-4
		Injuries	4.7E-4	3.2E-5	3.1E-6	5.1E-4
		Fatalities	7.1E-5	4.8E-6	4.7E-7	7.6E-5
		Vehicle Pollution (LCFs) [b]	---	---	6.7E-7	6.7E-7
	Hanford	Accidents	1.2E-3	2.0E-4	3.2E-5	1.5E-3
		Injuries	1.1E-3	1.6E-4	2.7E-5	1.3E-3
		Fatalities	9.3E-5	1.1E-5	1.8E-6	1.1E-4
		Vehicle Pollution (LCFs) [b]	---	---	5.9E-6	5.9E-6
	INEEL	Accidents	9.4E-4	1.7E-4	3.0E-5	1.1E-3
		Injuries	8.5E-4	1.4E-4	2.6E-5	1.0E-3
		Fatalities	7.5E-5	9.6E-6	1.8E-6	8.6E-5
		Vehicle Pollution (LCFs) [b]	---	---	5.0E-6	5.0E-6

[a] Totals may differ from the sum due to rounding.
[b] Dashed lines indicate that vehicle pollution impacts apply only to urban population zones.

Table E-8
Nonradiological Impacts per Shipment
from Accidents and Pollution Related Health Effects — Continued

Originating Site	Destination Site	Impact Category	Population Zone: Rural	Suburban	Urban	Total [a]
Mound Plant	WIPP	Accidents	8.0E-4	3.4E-4	3.3E-5	1.2E-3
		Injuries	7.4E-4	3.3E-4	3.2E-5	1.1E-3
		Fatalities	9.2E-5	2.9E-5	2.3E-6	1.2E-4
		Vehicle Pollution (LCFs) [b]	---	---	7.5E-6	7.5E-6
	SRS	Accidents	2.3E-4	3.2E-4	1.5E-5	5.7E-4
		Injuries	2.4E-4	3.1E-4	1.6E-5	5.7E-4
		Fatalities	2.8E-5	2.3E-5	9.8E-7	5.1E-5
		Vehicle Pollution (LCFs) [b]	---	---	3.3E-6	3.3E-6
Nevada Test Site	WIPP	Accidents	8.7E-4	1.4E-4	3.2E-5	1.0E-3
		Injuries	8.6E-4	1.3E-4	2.9E-5	1.0E-3
		Fatalities	1.2E-4	1.4E-5	3.3E-6	1.4E-4
		Vehicle Pollution (LCFs) [b]	---	---	4.3E-6	4.3E-6
	INEEL	Accidents	4.3E-4	6.8E-5	1.6E-5	5.1E-4
		Injuries	3.9E-4	6.0E-5	1.4E-5	4.6E-4
		Fatalities	3.0E-5	5.4E-6	1.4E-6	3.7E-5
		Vehicle Pollution (LCFs) [b]	---	---	6.5E-6	6.5E-6
Oak Ridge National Laboratory	WIPP	Accidents	5.6E-4	2.3E-4	1.1E-5	8.1E-4
		Injuries	5.7E-4	3.4E-4	1.4E-5	9.2E-4
		Fatalities	8.1E-5	2.8E-5	1.2E-6	1.1E-4
		Vehicle Pollution (LCFs) [b]	---	---	4.4E-6	4.4E-6
	SRS	Accidents	1.5E-4	1.8E-4	5.3E-6	3.3E-4
		Injuries	1.6E-4	1.8E-4	4.9E-6	3.4E-4
		Fatalities	1.9E-5	1.5E-5	4.1E-7	3.4E-5
		Vehicle Pollution (LCFs) [b]	---	---	9.6E-7	9.6E-7
Rocky Flats Environmental Technology Site	WIPP	Accidents	7.0E-4	1.7E-4	3.05E-5	9.0E-4
		Injuries	6.9E-4	1.5E-4	2.7E-5	8.7E-4
		Fatalities	9.0E-5	1.2E-5	1.8E-6	1.0E-4
		Vehicle Pollution (LCFs) [b]	---	---	4.5E-6	4.5E-6
	INEEL	Accidents	6.4E-4	5.9E-5	1.4E-5	7.1E-4
		Injuries	5.6E-4	3.6E-5	1.2E-5	6.0E-4
		Fatalities	4.4E-5	1.9E-6	7.2E-7	4.6E-5
		Vehicle Pollution (LCFs) [b]	---	---	2.9E-6	2.9E-6
Savannah River Site	WIPP	Accidents	6.3E-4	3.7E-4	1.9E-5	1.0E-3
		Injuries	6.1E-4	3.6E-4	1.8E-5	9.9E-4
		Fatalities	8.5E-5	2.8E-5	1.3E-6	1.1E-4
		Vehicle Pollution (LCFs) [b]	---	---	5.5E-6	5.5E-6

[a] Totals may differ from the sum due to rounding.
[b] Dashed lines indicate that vehicle pollution impacts apply only to urban population zones.

Table E-9
Aggregate Nonradiological Impacts for CH-TRU and RH-TRU Waste Transport [a]

Nonradiological Aggregate Impacts	Proposed Action Scaled to WIPP Max	Action Alternative 1			Action Alternative 2A			Action Alternative 2B			Action Alternative 2C		
	Basic Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste													
Accidents	43	36	25	62	28	38	66	38	42	80	36	26	62
Injuries	30	26	17	43	20	25	45	27	28	55	26	17	43
Fatalities	4	4	2	6	3	3	6	3	4	7	4	3	7
Vehicle Pollution (LCF)	0.1	0.1	0.08	0.2	0.09	0.1	0.2	0.1	0.1	0.3	0.1	0.08	0.2
RH-TRU Waste													
Accidents	13	100	9	109	38	6	43	38	6	43	38	6	43
Injuries	9	68	8	76	26	5	31	26	5	31	25	5	31
Fatalities	1	9	1	10	3	1	4	3	1	4	3	1	4
Vehicle Pollution (LCF)	0.04	0.3	0.04	0.3	0.1	0.03	0.2	0.1	0.03	0.2	0.1	0.03	0.2
Total CH-TRU and RH-TRU Waste													
Accidents	56	136	34	171	66	44	109	76	48	123	74	32	105
Injuries	39	94	25	119	46	30	76	53	33	86	51	22	74
Fatalities	5	13	3	16	6	4	10	6	5	11	7	4	11
Vehicle Pollution (LCF)	0.1	0.4	0.1	0.5	0.2	0.1	0.4	0.2	0.1	0.5	0.2	0.1	0.4

[a] Total Inventory numbers may differ due to rounding.

Note: LCF = Latent Cancer Fatalities

Table E-9
Aggregate Nonradiological Impacts for CH-TRU and RH-TRU Waste Transport — Continued [a]

Nonradiological Aggregate Impacts	Action Alternative 3			No Action Alternative 1A			No Action Alternative 1B		
	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste									
Accidents	73	31	104	0.3	0.03	0.3	7	2	8
Injuries	51	20	71	0.2	0.03	0.3	6	2	8
Fatalities	7	3	10	0.02	2.1E-3	0.02	0.5	0.1	0.6
Vehicle Pollution (LCF)	0.2	0.09	0.3	2.5E-3	9.6E-5	2.6E-3	0.03	8.0E-3	0.04
RH-TRU Waste									
Accidents	124	12	135	2	3	5	2	3	5
Injuries	85	10	94	2	2	4	2	2	4
Fatalities	11	1	12	0.2	0.2	0.3	0.2	0.2	0.3
Vehicle Pollution (LCF)	0.4	0.05	0.4	6.5E-3	0.02	0.03	6.5E-03	0.02	0.03
Total CH-TRU and RH-TRU Waste									
Accidents	197	43	239	2	3	5	9	5	13
Injuries	136	30	165	2	2	4	8	4	12
Fatalities	18	4	22	0.2	0.2	0.3	0.7	0.3	1
Vehicle Pollution (LCF)	0.6	0.1	0.7	9.0E-03	0.02	0.03	0.04	0.03	0.07

[a] Total Inventory numbers may differ due to rounding.

Note 1: LCF = Latent Cancer Fatalities
Note 2: Impacts for No Action Alternatives 1A and 1B reflect the number of shipments to consolidate TRU waste from generator-storage sites to treatment sites. No TRU waste would be shipped to WIPP.

Table E-10
RADTRAN Input for Transportation Analyses

Parameter	CH-TRU Waste	RH-TRU Waste
Configuration Data		
Mode of Transportation	Truck	Truck
Package type	TRUPACT-II	72B cask
Packages/Shipment	3	1
Package characteristic dimension, meter	7.4	3.6
Movement Data		
Shipment distance	Site/alternative specific (See Table E-5 and E-6)	
Population density	Route/alternative specific per Highway or Interline Routing Models	
Shipment Speed (kilometer/hour)		
Urban	24.2	24.2
Suburban	40.3	40.3
Rural	88.6	88.6
Stop time (hour) per kilometer	0.01	0.01
Other Normal Input	RADTRAN 4 default values	RADTRAN 4 default values
Normal Exposure Data		
Transportation Index, TI (mrem/hour)	Site/alternative specific	Site/alternative specific
Number of crew members	2	2
Effective distance from source to crew	10	10
Number of people per public vehicle	2	2
Number of people exposed per stop	50	50
Exposure distance while stopped, meters	20	20
Accident Exposure Data		
Number of accident severity categories	8	8
Accident severity category frequency	(NUREG-0170 values)	(NUREG-0170 values)
Radioactive contents/parameters	See Table E-17	See Table E-17
Release fractions	See Table E-20	See Table E-21
Other accident inputs	RADTRAN 4 default values	RADTRAN 4 default values

Note: Accident, injury, and fatality rates were determined for each route using state-specific data (Saricks and Kvitek 1994)

- Assigning a probability to each release scenario
- Estimating the amount and type of material likely to be released in each scenario (the release fraction)
- Evaluating the consequences of such a release, most often in terms of radiation dose to workers and the public

These analyses were performed to estimate the radiological impacts due to the shipment of TRU waste from the major generator-storage sites to WIPP.

E.4.1 Radiological Impacts Due to Accident-Free Transportation

Accident-free radiological impacts would occur during the routine transportation of radioactive material and are the result of public and worker exposures to external radiation (at levels allowed by transportation regulations). The dose rates would be low and would typically be less than that of natural background radiation.

E.4.1.1 Methodology

To determine the radiological impacts of accident-free transportation, the TRU waste volumes, the number of shipments that would be required to transport the waste, the number of miles traveled, and a breakdown of the miles traveled according to urban, suburban, and rural population zones were used. To estimate the accident-free exposures, the speeds of the vehicle were assumed to be 15, 25, and 55 miles per hour in an urban, suburban, and rural population zone, respectively. The use of these speeds resulted in an overestimation of the accident-free exposures. They were not representative of the actual speeds the vehicles would travel in each population zone. The actual speeds are dependent on the posted speed limit or the maximum speed of the vehicle (65 miles per hour). However, the speeds used to estimate accident-free exposures would account for the potential for increased traffic in an urban population zone where an average speed could be as low as 15 miles per hour due to rush-hour traffic.The accident-free radiological impacts, expressed in person-rem, were converted to LCFs using a conversion factor of 1 person-rem = 5 x 10^{-4} LCFs for nonoccupational doses and 1 person-rem = 4 x 10^{-4} LCFs for occupational doses (ICRP 1991).

Among the more important RADTRAN input parameters are: the TI, the frequency of stops, the number of people exposed and their distances from the package surface, and the speed of the vehicle used for transportation. The following categories of accident-free occupational and nonoccupational exposures were assessed using RADTRAN (the nomenclature provided in the output are identified in parentheses):

- ***Along Route*** *(Off Link Exposure):* Exposure to individuals adjacent to routes of travel
- ***Sharing Route*** *(On Link Exposure):* Exposure to individuals sharing the right-of-way
- ***Stops*** *(Stops):* Exposure to individuals while shipments are at rest stops
- ***Occupational*** *(Occupational Exposure):* Exposure to vehicle crews

Radiation doses from the first three were summed to obtain the total nonoccupational radiation dose. Additional analyses, discussed later in this section, were conducted to identify the potential impact to the individual likely to receive the greatest radiation dose, such as the state shipment inspector or an individual living along the route.

The TI represents the radiation dose rate at 1 meter (3.3 feet) from the surface of the shipping package and is dependent on the waste density, distribution of radionuclides, quantity of radionuclides per shipment, mix of waste types, self-shielding provided by the waste, and shielding provided by the package. The TI is, therefore, very sensitive to small quantities of gamma-emitting radionuclides such as cobalt-60 (Co-60) and cesium-137 (Cs-137).

Typically, the radionuclide composition of the waste is different for each generator site, and the radionuclide composition of the waste is different from one waste stream to another. BIR-3 considers 11 different waste groups, ranging from paper to vitrified waste. For shipments to WIPP, an average radionuclide composition was developed for each site using the BIR-3 database information and information provided in the Integrated Data Base (IDB) (see Appendix A for additional details).

> ### ACCIDENT-FREE EXPOSURES
>
> Conservative assumptions were made to provide a bounding estimate of potential incident-free exposures to individuals along the shipping routes. First, a bounding transportation index (the exposure from the shipping container at two meters) was chosen for CH-TRU waste and RH-TRU waste. This was done because only about 80 percent of the CH-TRU waste stream volumes and about 15 percent of the RH-TRU waste stream volumes have reported radionuclide inventories. Therefore, basing the transportation index on incomplete data may overestimate the accident-free exposures. Any adjustments made due to improved site waste characterization data would not affect the results significantly.
>
> Conservative assumptions were also used for the accident-free MEI scenarios. The scenarios were conservatively chosen to maximize any potential impacts. For example, for the exposure to the individual living along the route it was assumed that this individual lived at a location where every shipment would pass the person's home. It was also assumed that this individual was exposed to every shipment. Results indicate that even with these conservative assumptions, the impacts to MEI would be low, and potential impacts would be considerably lower.

The shielding dimensions of the TRUPACT-II and RH-72B were used in MICROSHIELD (Worku and Negin 1995) to estimate the external dose rates from unit concentrations of the gamma-emitting radionuclides expected to make the greatest exposure impact. The dose rates were then multiplied by the activity of each radionuclide in the TRU waste inventory for each site to determine an external-dose screening value. These values indicate the comparative impact from external radiation for each radionuclide in the TRU waste inventory at each site. The radionuclides of greatest concern throughout the DOE complex would be americium-241 (Am-241), Cs-137, barium-137m (Ba-137m), and Co-60.

Dose rates were tabulated in a spreadsheet for each radionuclide in the TRU waste inventory to determine the TI. In most cases, the calculated TI was much less than 1; however, because the radionuclide inventory information may be highly variable, a TI of 4 was assumed for all CH-TRU waste shipments, and a TI of 10 was assumed for all RH-TRU waste shipments. No calculated TI exceeded these assumed values. Table E-11 presents the TIs calculated for each site for the Proposed Action and each action alternative.

RADTRAN classified those living along the shipment routes as urban, suburban, and rural fractions with respective mean population densities of 3,861, 719, and 6 persons per square kilometer (10,003, 1,863, and 16 per square mile). This classification is based upon the *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes* (NRC 1977) and the population densities are quite typical of urban, suburban, and rural environments. For example, statistics from the Denver Regional Council of Governments show that, along I-25 through Denver, only a small area around downtown Denver has a population density that exceeds the urban figure used in RADTRAN (3,997 persons per square kilometer [10,354 persons per square mile] for Denver versus the 3,861 persons per square kilometer [10,003 persons per square mile] assumed by RADTRAN). Other segments through Denver have much lower population densities than the RADTRAN urban value. Fifteen miles south of downtown Denver, population densities along I-25 approach the rural value of 6 persons per square kilometer (16 persons per square mile).

Table E-11
Transportation Indices for TRU Waste Transported from the 10 Major Sites (millirem per hour at 1 meter [3.3 feet]) [a]

Sites	Proposed Action	Action Alternative 1	Action Alternative 2A	Action Alternative 2B	Action Alternative 2C	Action Alternative 3
CH-TRU Waste						
ANL-E	4.9E-3	4.1E-3	---	---	4.1E-3	---
Hanford	2.2	3.3	2.8	2.8	3.3	2.7
INEEL	0.6	0.7	1.5	1.3	0.7	0.6
LANL	0.3	0.3	0.8	---	1.3	0.2
LLNL	1.1E-3	9.7E-4	---	---	9.7E-4	---
MOUND	3.2E-5	2.7E-5	---	---	3.0E-5	---
NTS	0.05	0.04	---	---	0.04	---
ORNL	0.2	0.2	---	---	0.2	---
RFETS	0.02	0.02	0.09	---	0.05	0.02
SRS	0.07	0.06	0.2	0.2	0.06	0.06
RH-TRU Waste						
Hanford	1.2	1.2	3.2	3.2	3.2	9.2
INEEL	0.6	0.6	---	---	---	---
LANL	0.1	0.1	---	---	---	---
ORNL	0.6	0.4	1.1	1.1	1.1	1.0

[a] Dashes indicate that no TRU waste would be shipped to WIPP.

Note: The actual TI used for CH-TRU waste was 4, and the actual TI used for RH-TRU waste was 10.

Exposure scenarios were determined to assess the potential impacts from occupational and nonoccupational radiation doses. Some individuals would be exposed to only a single shipment while others would be exposed to multiple shipments. Transportation crew members would be monitored with radiation dosimeters to limit exposures. Radiation dose assessments included the following scenarios:

- A person in a traffic jam next to a truck transporting TRU waste. For this assessment, the exposure distance was assumed to be 1 meter (3.3 feet) and the exposure time assumed to be 30 minutes. The person was assumed to be exposed only once.

- An inspector of trucks ready for departure from a site. For this assessment, it was assumed that the inspector would have an exposure distance of 1 meter (3.3 feet) for 30 minutes, the inspector would work at the same job for 10 years, and there would be two shifts working the same job. The number of shipments inspected by and individual would depend on the total number of shipments from a site and the rate at which they were shipped.

- A state safety inspector. For this assessment, it was assumed that the inspector would be involved in 20 percent of the inspections over a 10-year period with an average exposure distance of approximately 1 meter (3.3 feet). Inspections may occur at the origin facility, upon arrival at WIPP, or in the corridor states at ports of entry for trucks. To allow for queues, a truck inspection time of 1 hour was used. To bound the state inspector dose, the route on which the majority of the waste (77 percent) enters New Mexico was chosen.

- A person residing along a shipment route. For this assessment, it was assumed that the individual would be exposed to every waste shipment for 70 years, at a distance of approximately 30 meters (98 feet).

- A rest stop employee. For this assessment, a stop duration of 2 hours and an exposure distance of 20 meters (66 feet) were assumed. It was also assumed that the individual would be exposed to approximately 20 percent of all CH-TRU and RH-TRU waste shipments sent to WIPP over a 10-year period. This assumption was made on the basis that all trucks stopped at the same location, an individual worked for 10 years at the truck stop, and 3 shifts worked at the truck stop.

- A driver of a truck hauling TRU waste. For this assessment, doses were assessed both when the truck was moving and when it was stopped. An exposure distance of 4 meters (13 feet) was assumed. Doses received while the trucks were stopped were assumed to be due to inspections every 100 miles, refueling, and food stops. A truck driver, rather than a service attendant, was assumed to refuel the truck. Depending upon the number of shipments from a facility and the travel time to WIPP, a truck driver may transport all or only a fraction of the shipments from a particular site. It should be noted that no matter what the estimated impacts are, current regulations state that any monitored crew member who receives a radiation dose that approaches 2 rem (the administrative limit for occupational doses) in any given year is to be reassigned to other duties involving no further dose for the remainder of the year.

Table 3-17 in Chapter 3 presents the amount of time required to emplace CH-TRU and RH-TRU waste under the Proposed Action and the action alternatives based upon a throughput rate of 50 TRUPACT-IIs and 8 RH-72Bs per week. This information was used to estimate the maximum number of shipments that the MEI would be exposed to. All doses were determined during a time fewer shipments would be traveling along the routes.

E.4.1.2 Results

Accident-free radiological impacts (per shipment) for CH-TRU and RH-TRU waste shipments from the 10 major generator sites to WIPP are presented in Tables E-12 and E-13. Results are presented as the population dose (person-rem) for occupational and nonoccupational groups for each site. These population doses were used to calculate the aggregate accident-free doses from CH-TRU and RH-TRU waste transportation. Table E-14 summarizes these doses according to alternative and provides the mathematically expected LCFs.

Aggregate MEI doses are presented in Table E-15, as are the corresponding LCFs from CH-TRU and RH-TRU waste transportation.

E.4.2 Radiological Impacts of Transportation Accidents

Radiological impacts due to transportation accidents could be incurred if any radioactive material were released into the environment. The greatest potential impact from such releases would occur when alpha-emitting radionuclides are either inhaled or ingested.

Table E-12
Population Dose per CH-TRU Waste Shipment

Origination Site	Destination Site	Exposure Category	Population Dose (person-rem)	Applicable to the Following
Argonne National Laboratory - East	WIPP	Occupational	0.03	Proposed Action, Action Alternative 1, and Action Alternative 2C
		Nonoccupational		
		Stops	0.2	
		Sharing Route	9.7E-3	
		Along Route	5.3E-3	
		Total [a]	0.2	
	SRS	Occupational	0.02	Action Alternatives 2A and 2B, Action Alternative 3, and No Action Alternatives 1A and 1B
		Nonoccupational		
		Stops	0.08	
		Sharing Route	5.7E-3	
		Along Route	5.2E-3	
		Total [a]	0.1	
Hanford Site	WIPP	Occupational	0.03	Proposed Action, Action Alternative 1, Action Alternatives 2A, 2B, and 2C, and Action Alternative 3
		Nonoccupational		
		Stops	0.2	
		Sharing Route	9.7E-3	
		Along Route	3.3E-3	
		Total [a]	0.2	
Idaho National Engineering and Environmental Laboratory/ANL-W	WIPP	Occupational	0.02	Proposed Action, Action Alternative 1, Action Alternatives 2A, 2B, and 2C, and Action Alternative 3
		Nonoccupational		
		Stops	0.1	
		Sharing Route	7.5E-3	
		Along Route	2.7E-3	
		Total [a]	0.1	
Lawrence Livermore National Laboratory	WIPP	Occupational	0.03	Proposed Action, Action Alternative 1, and Action Alternative 2C
		Nonoccupational		
		Stops	0.1	
		Sharing Route	8.5E-3	
		Along Route	4.8E-3	
		Total [a]	0.2	
	Hanford	Occupational	0.02	Action Alternatives 2A and 2B, Action Alternative 3, and No Action Alternatives 1A and 1B
		Nonoccupational		
		Stops	0.09	
		Sharing Route	5.7E-3	
		Along Route	4.6E-3	
		Total [a]	0.1	
Los Alamos National Laboratory	WIPP	Occupational	6.0E-3	Proposed Action, Action Alternative 1, Action Alternatives 2A and 2C, and Action Alternative 3
		Nonoccupational		
		Stops	0.03	
		Sharing Route	1.8E-3	
		Along Route	4.6E-4	
		Total [a]	0.04	
	INEEL	Occupational	0.02	Action Alternative 2B, and No Action Alternative 1A
		Nonoccupational		
		Stops	0.1	
		Sharing Route	6.3E-3	
		Along Route	2.5E-3	
		Total [a]	0.1	

[a] Nonoccupational doses include exposure to people at stops, sharing the route, and along the route.

Table E-12
Population Dose per CH-TRU Waste Shipment — Continued

Origination Site	Destination Site	Exposure Category	Population Dose (person-rem)	Applicable to the Following
Mound Plant	WIPP	Occupational	0.03	Proposed Action, Action Alternative 1, and Action Alternative 2C
		Nonoccupational		
		Stops	0.2	
		Sharing Route	0.01	
		Along Route	6.7E-3	
		Total [a]	0.2	
	SRS	Occupational	0.01	Action Alternatives 2A and 2B, Action Alternative 3, and No Action Alternatives 1A and 1B
		Nonoccupational		
		Stops	0.06	
		Sharing Route	4.0E-3	
		Along Route	3.4E-3	
		Total [a]	0.07	
Nevada Test Site	WIPP	Occupational	0.02	Proposed Action, Action Alternative 1, and Action Alternative 2C
		Nonoccupational		
		Stops	0.1	
		Sharing Route	6.4E-3	
		Along Route	1.9E-3	
		Total [a]	0.1	
	INEEL	Occupational	0.01	Action Alternatives 2A and 2B, Action Alternative 3, and No Action Alternatives 1A and 1B
		Nonoccupational		
		Stops	0.07	
		Sharing Route	4.3E-3	
		Along Route	2.7E-3	
		Total [a]	0.08	
Oak Ridge National Laboratory	WIPP	Occupational	0.03	Proposed Action, Action Alternative 1, and Action Alternative 2C
		Nonoccupational		
		Stops	0.1	
		Sharing Route	8.2E-3	
		Along Route	4.5E-3	
		Total [a]	0.2	
	SRS	Occupational	6.2E-3	Action Alternatives 2A and 2B, Action Alternative 3, and No Action Alternatives 1A and 1B
		Nonoccupational		
		Stops	0.03	
		Sharing Route	2.2E-3	
		Along Route	1.6E-3	
		Total [a]	0.04	
Rocky Flats Environmental Technology Site	WIPP	Occupational	0.01	Proposed Action; Action Alternative 1, Action Alternatives 2A and 2C, and Action Alternative 3
		Nonoccupational		
		Stops	0.07	
		Sharing Route	4.0E-3	
		Along Route	2.0E-3	
		Total [a]	0.07	
	INEEL	Occupational	0.01	Action Alternative 2B and No Action Alternative 1B
		Nonoccupational		
		Stops	0.07	
		Sharing Route	4.0E-3	
		Along Route	1.4E-3	
		Total [a]	0.08	
Savannah River Site	WIPP	Occupational	0.03	Proposed Action, Action Alternative 1, Action Alternatives 2A, 2B, and 2C, and Action Alternative 3
		Nonoccupational		
		Stops	0.1	
		Sharing Route	8.9E-3	
		Along Route	5.4E-3	
		Total [a]	0.2	

[a] Nonoccupational doses include exposure to people at stops, sharing the route, and along the route.

Table E-13
Population Dose per RH-TRU Waste Shipment

Origination Site	Destination Site	Exposure Category	Population Dose (person-rem)	Applicable to the Following
Hanford (Richland) Site	WIPP	Occupational	0.01	Proposed Action, Action Alternative 1, Action Alternatives 2A, 2B, and 2C, and Action Alternative 3
		Nonoccupational		
		Stops	0.2	
		Sharing Route	0.01	
		Along Route	4.4E-3	
		Total [a]	0.2	
Idaho National Engineering and Environmental Laboratory / ANL-W	WIPP	Occupational	9.0E-3	Proposed Action and Action Alternative 1
		Nonoccupational		
		Stops	0.2	
		Sharing Route	0.01	
		Along Route	3.6E-3	
		Total [a]	0.2	
	Hanford	Occupational	3.9E-3	Action Alternatives 2A, 2B, and 2C, Action Alternative 3, and No Action Alternatives 1A and 1B
		Nonoccupational		
		Stops	0.08	
		Sharing Route	4.2E-3	
		Along Route	1.2E-3	
		Total [a]	0.08	
Los Alamos National Laboratory	WIPP	Occupational	2.2E-3	Proposed Action and Action Alternative 1
		Nonoccupational		
		Stops	0.04	
		Sharing Route	2.4E-3	
		Along Route	6.1E-4	
		Total [a]	0.05	
	Hanford	Occupational	0.01	Action Alternatives 2A, 2B, and 2C, Action Alternative 3, and No Action Alternatives 1A and 1B
		Nonoccupational		
		Stops	0.2	
		Sharing Route	0.01	
		Along Route	4.2E-3	
		Total [a]	0.2	
Oak Ridge National Laboratory	WIPP	Occupational	9.3E-3	Proposed Action, Action Alternative 1, Action Alternatives 2A, 2B, and 2C, and Action Alternative 3
		Nonoccupational		
		Stops	0.2	
		Sharing Route	0.01	
		Along Route	6.0E-3	
		Total [a]	0.2	

[a] Nonoccupational doses include exposure to people at stops, sharing the route, and along the route.

Table E-14
Aggregate Accident-Free Dose (person-rem) from CH-TRU and RH-TRU Waste Shipments [a, b]

Exposure Category	Proposed Action Scaled to WIPP Max	Action Alternative 1			Action Alternative 2A			Action Alternative 2B			Action Alternative 2C		
	Basic Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste													
Occupational	705 (0.3)	600 (0.2)	410 (0.2)	1.0E+3 (0.4)	455 (0.2)	610 (0.2)	1.0E+3 (0.4)	640 (0.3)	680 (0.3)	1.3E+3 (0.6)	600 (0.2)	415 (0.2)	1.0E+3 (0.4)
Nonoccupational													
Stops	3.9E+3 (2.0)	3.3E+3 (1.7)	2.2E+3 (1.1)	5.5E+3 (2.8)	2.5E+3 (1.3)	3.4E+3 (1.7)	5.9E+3 (3.0)	3.5E+3 (1.8)	3.7E+3 (1.9)	7.2E+3 (3.7)	3.3E+3 (1.6)	2.3E+3 (1.1)	5.6E+3 (2.7)
Sharing Route	220 (0.1)	185 (0.09)	125 (0.06)	310 (0.2)	140 (0.07)	190 (0.09)	330 (0.2)	200 (0.1)	210 (0.1)	410 (0.2)	185 (0.09)	130 (0.06)	315 (0.2)
Along Route	85 (0.04)	70 (0.04)	45 (0.02)	115 (0.06)	55 (0.03)	65 (0.03)	120 (0.06)	75 (0.04)	75 (0.04)	150 (0.08)	70 (0.04)	45 (0.02)	115 (0.06)
Nonoccupational Total [c]	4.2E+3 (2.1)	3.5E+3 (1.8)	2.4E+3 (1.2)	5.9E+3 (3.0)	2.7E+3 (1.4)	3.6E+3 (1.8)	6.3E+3 (3.2)	3.8E+3 (1.9)	4.0E+3 (2.0)	7.8E+3 (3.9)	3.5E+3 (1.8)	2.5E+3 (1.2)	6.0E+3 (3.0)
RH-TRU Waste													
Occupational	80 (0.03)	635 (0.3)	70 (0.03)	705 (0.3)	245 (0.1)	40 (0.02)	285 (0.1)	245 (0.1)	40 (0.02)	285 (0.1)	245 (0.1)	40 (0.02)	285 (0.1)
Nonoccupational													
Stops	1.6E+3 (0.8)	1.3E+4 (6.3)	1.4E+3 (0.7)	1.4E+4 (7.0)	4.8E+3 (2.4)	805 (0.4)	5.6E+3 (2.8)	4.8E+3 (2.4)	805 (0.4)	5.6E+3 (2.8)	4.8E+3 (2.4)	805 (0.4)	5.6E+3 (2.8)
Sharing Route	90 (0.05)	710 (0.4)	80 (0.04)	790 (0.4)	270 (0.1)	50 (0.03)	320 (0.1)	270 (0.1)	50 (0.03)	320 (0.1)	270 (0.1)	50 (0.03)	320 (0.1)
Along Route	35 (0.02)	255 (0.1)	45 (0.02)	300 (0.1)	100 (0.05)	30 (0.02)	130 (0.07)	100 (0.05)	30 (0.02)	130 (0.07)	100 (0.05)	30 (0.02)	130 (0.07)
Nonoccupational Total [c]	1.7E+3 (0.9)	1.4E+4 (6.8)	1.5E+3 (0.7)	1.6E+4 (7.5)	5.2E+3 (2.6)	885 (0.5)	6.1E+3 (3.0)	5.2E+3 (2.6)	885 (0.5)	6.1E+3 (3.0)	5.2E+3 (2.6)	885 (0.5)	6.1E+3 (3.0)
Total CH-TRU and RH-TRU Waste													
Occupational	790 (0.3)	1.2E+3 (0.5)	480 (0.2)	1.7E+3 (0.7)	700 (0.3)	650 (0.2)	1.3E+3 (0.5)	885 (0.4)	720 (0.3)	1.6E+3 (0.7)	845 (0.3)	455 (0.2)	1.3E+3 (0.5)
Nonoccupational													
Stops	5.5E+3 (2.8)	1.6E+4 (8.0)	3.6E+3 (1.8)	2.0E+4 (9.8)	7.3E+3 (3.7)	4.2E+3 (2.1)	1.2E+4 (5.8)	8.3E+3 (4.2)	4.5E+3 (2.3)	1.3E+4 (6.5)	8.1E+3 (4.0)	3.1E+3 (1.5)	1.1E+4 (5.5)
Sharing Route	310 (0.2)	895 (0.5)	205 (0.1)	1.1E+3 (0.6)	410 (0.2)	240 (0.1)	650 (0.3)	470 (0.2)	260 (0.1)	730 (0.3)	455 (0.2)	180 (0.09)	635 (0.3)
Along Route	115 (0.06)	325 (0.1)	90 (0.04)	415 (0.2)	155 (0.08)	95 (0.05)	250 (0.1)	175 (0.09)	105 (0.06)	280 (0.2)	170 (0.09)	75 (0.04)	245 (0.1)
Nonoccupational Total [c]	5.9E+3 (3.0)	1.8E+4 (8.6)	3.9E+3 (1.9)	2.2E+4 (10.5)	7.9E+3 (4.0)	4.5E+3 (2.3)	1.2E+4 (6.2)	9.0E+3 (4.5)	4.9E+3 (2.5)	1.4E+4 (6.9)	8.7E+3 (4.4)	3.4E+3 (1.7)	1.2E+4 (6.0)

[a] Number in parentheses equals LCFs.
[b] Total Inventory numbers may differ due to rounding.
[c] Total nonoccupational exposure is the sum of exposures to persons at stops, sharing the route, and along the route.

Table E-14
Aggregate Accident-Free Dose (person-rem) from CH-TRU and RH-TRU Waste Shipments — Continued [a, b]

Exposure Category	Action Alternative 3			No Action Alternative 1A			No Action Alternative 1B		
	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste									
Occupational	1.2E+3 (0.5)	490 (0.2)	1.7E+3 (0.7)	5 (2.2E-3)	0.5 (1.9E-4)	6 (2.4E-3)	115 (0.05)	30 (0.01)	150 (0.06)
Nonoccupational									
Stops	6.5E+3 (3.3)	2.7E+3 (1.4)	9.3E+3 (4.7)	30 (0.02)	3 (1.3E-3)	33 (0.02)	645 (0.3)	175 (0.09)	820 (0.4)
Sharing Route	370 (0.2)	150 (0.08)	520 (0.3)	2 (9.7E-4)	0.2 (8.2E-5)	2 (1.1E-3)	40 (0.02)	10 (5.0E-3)	50 (0.03)
Along Route	135 (0.07)	55 (0.03)	190 (0.1)	2 (7.6E-4)	0.1 (6.2E-5)	2 (8.2E-4)	15 (7.6E-3)	4 (2.0E-3)	19 (9.6E-3)
Nonoccupational Total [c]	7.1E+3 (3.6)	2.9E+3 (1.5)	1.0E+4 (5.1)	35 (0.02)	3 (1.5E-3)	38 (0.02)	695 (0.4)	190 (0.1)	885 (0.5)
RH-TRU Waste									
Occupational	790 (0.3)	90 (0.04)	880 (0.4)	20 (7.3E-3)	20 (7.5E-3)	40 (0.01)	20 (7.3E-3)	20 (7.5E-3)	40 (0.01)
Nonoccupational									
Stops	1.6E+4 (7.8)	1.8E+3 (0.9)	1.7E+4 (8.7)	365 (0.2)	370 (0.2)	735 (0.4)	365 (0.2)	370 (0.2)	735 (0.4)
Sharing Route	880 (0.4)	105 (0.05)	985 (0.5)	20 (0.01)	25 (0.01)	45 (0.02)	20 (0.01)	25 (0.01)	45 (0.02)
Along Route	320 (0.2)	55 (0.03)	375 (0.2)	10 (4.1E-3)	20 (0.01)	30 (0.01)	10 (4.1E-3)	20 (0.01)	30 (0.01)
Nonoccupational Total [c]	1.7E+4 (8.4)	1.9E+3 (1.0)	1.9E+4 (9.4)	390 (0.2)	415 (0.2)	805 (0.4)	390 (0.2)	415 (0.2)	805 (0.4)
Total CH-TRU and RH-TRU Waste									
Occupational	2.0E+3 (0.8)	580 (0.2)	2.6E+3 (1.1)	25 (9.5E-3)	21 (7.7E-3)	46 (0.02)	135 (0.06)	50 (0.02)	190 (0.07)
Nonoccupational									
Stops	2.3E+4 (11.1)	4.5E+3 (2.3)	2.6E+4 (13.4)	395 (0.2)	373 (0.2)	768 (0.4)	1.0E+3 (0.5)	545 (0.3)	1.6E+3 (0.8)
Sharing Route	1.3E+3 (0.6)	255 (0.1)	1.5E+3 (0.8)	22 (0.01)	25 (0.01)	47 (0.02)	60 (0.03)	35 (0.02)	95 (0.05)
Along Route	455 (0.3)	110 (0.06)	565 (0.3)	12 (4.9E-3)	20 (0.01)	32 (0.01)	25 (0.01)	24 (0.01)	49 (0.02)
Nonoccupational Total [c]	2.4E+4 (12.0)	4.8E+3 (2.5)	2.9E+4 (14.5)	425 (0.2)	418 (0.2)	843 (0.4)	1.1E+3 (0.6)	605 (0.3)	1.7E+3 (0.9)

[a] Number in parentheses equals LCFs.
[b] Total Inventory numbers may differ due to rounding.
[c] Total nonoccupational exposure is the sum of exposures to persons at stops, sharing the route, and along the route.

Table E-15
Aggregate Dose (rem) and LCFs (values in parentheses) to MEIs from CH-TRU and RH-TRU Waste Shipments

MEI	Proposed Action	Action Alternative 1	Action Alternative 2A	Action Alternative 2B	Action Alternative 2C	Action Alternative 3
CH-TRU Waste						
Person in traffic jam next to shipment	5.0E-3 (2.5E-6)	5.0E-3 (2.5E-6)	5.0E-3 (2.5E-6)	5.0E-3 (2.5E-6)	5.0E-3 (2.5E-6)	5.0E-3 (2.5E-6)
Departure Inspectors	5.9 (2.4E-3)	1.1 (4.4E-4)	1.7 (6.8E-4)	1.7 (6.8E-4)	1.9 (7.6E-4)	2.6 (1.0E-3)
State Inspector	11.6 (4.6E-3)	3.7 (1.5E-3)	4.3 (1.7E-3)	4.9 (2.0E-3)	3.9 (1.6E-3)	5.2 (2.1E-3)
Individual residing adjacent to access route	0.5 (2.5E-4)	0.3 (1.5E-4)	0.3 (1.5E-4)	0.3 (1.5E-4)	0.3 (1.5E-4)	0.4 (2.0E-4)
Rest stop employee	0.3 (1.5E-4)	0.2 (1.0E-4)	0.2 (1.0E-4)	0.2 (1.0E-4)	0.2 (1.0E-4)	0.2 (1.0E-4)
RH-TRU Waste						
Person in traffic jam next to shipment	0.01 (5.0E-6)	0.01 (5.0E-6)	0.01 (5.0E-6)	0.01 (5.0E-6)	0.01 (5.0E-6)	0.01 (5.0E-6)
Departure Inspector	4.0 (1.6E-3)	13.7 (5.5E-3)	5.6 (2.2E-3)	5.6 (2.2E-3)	5.6 (2.2E-3)	15.1 (6.0E-3)
State Inspectors	9.7 (3.9E-3)	17.7 (7.1E-3)	7.3 (2.9E-3)	7.3 (2.9E-3)	7.3 (2.9E-3)	19.4 (7.8E-3)
Individual residing adjacent to access route	0.3 (1.5E-4)	1.2 (6.0E-4)	0.4 (2.0E-4)	0.4 (2.0E-4)	0.4 (2.0E-4)	1.2 (6.0E-4)
Rest stop employee	0.7 (3.5E-4)	0.6 (3.0E-4)	0.2 (1.0E-4)	0.2 (1.0E-4)	0.2 (1.0E-4)	0.6 (3.0E-4)

As previously discussed, NRC-certified Type B packages (TRUPACT-II and RH-72B cask) used to ship CH-TRU and RH-TRU waste must undergo a series of performance tests which simulate accident conditions. These tests include drops, punctures, exposure to fire, and water immersion. The packages are passed if no radioactive material is released as a result of the tests.

A 1987 NRC study (Fischer et al. 1987) estimated that only 0.6 percent of truck and rail accidents involving Type B containers or casks could cause a radiation hazard to the public. The highest number of potential accidents (nonradiological), as calculated in Section E.3, would be expected under Action Alternative 3 and would be about 331. Only half of these accidents would be expected to occur when the canister or cask is loaded; the other half would occur when the trucks were transporting empty canisters or casks back to sites. Therefore, of the 166 potential accidents occurring from transportation of TRU waste under Action Alternative 3, only one accident would be expected to result in damage to the package.

An earlier NRC study (NRC 1977) conservatively estimated that 91 percent of truck and 80 percent of rail accidents are category I and II accidents (packaging must survive without a release). Therefore, 9 percent of truck and 20 percent of rail accidents involving Type B containers or casks could result in radioactive material releases and could be more severe than the test conditions. Therefore, approximately 15 truck-related accidents could occur in which the loaded Type B container is subjected to conditions beyond those associated with severity categories I or II. Some of the low-probability events could result in a release from a Type B package. In order to assure conservative (bounding) impact estimates, the more conservative statistics from the older 1977 NRC study were used in these analyses.

BOUNDING CASE ACCIDENTS

The following assumptions were made to conservatively estimate the potential impacts of a low-probability/high-consequence event:

- Accidents occurred in the urban portion of a nonspecific, large metropolitan area (population greater than one million) with a mean population density of 3,861 persons per square kilometer. This is conservative, since typically 80 to 95 percent of travel occurs in rural or suburban population zone. Including the probability of the accident occurring in an urban population zone would reduce the frequency of occurrence.

- Thermal release fractions (Table E-19) represent the range of thermal exposures to which a TRUPACT-II or RH-72B may be subjected. As stated in SEIS-II, most thermal events result in no release of radioactive material. However, the intent of the accident analysis in SEIS-II was to estimate potential impacts from highly unlikely events.

- Accidents occurred during very stable meteorological conditions, limiting dispersion of the radioactive material plume and maximizing radiation doses. It is difficult to incorporate actual meteorological data for the bounding case accident, since it could occur anywhere along the transportation route. Therefore, given the uncertainty, meteorological conditions were chosen to limit dispersion. Other meteorological conditions that could result in higher doses are unlikely to be encountered along the transportation route.

- In the accident scenarios involving truck transportation, one TRUPACT-II or RH-72B was breached and was subsequently engulfed in fire for two hours. It is unlikely that a breach of the shipping cask would occur in combination with an all-engulfing fire.

- Doses were calculated using a resuspension particle half-life of 365 days. DOE must decrease contamination to levels below which unrestricted public access would be allowable. Continued exposure to residual radioactive contamination on or near a highway would cease following cleanup of the accident site and removal of residual contamination. Therefore, doses from resuspension would be much lower.

- The TRUPACT-II or RH-72B inventory was increased to the maximum allowed under the WIPP WAC to bound the inventory modeled in an accident. No waste shipments would have the radionuclide concentrations modeled, so any impacts from a transportation accident would be much less. In fact, average site inventories indicate that the bounding inventory represents an increase in concentration by a factor 4 to 10 for some radionuclides.

Two analyses were conducted for radiological impacts due to transportation accidents. The first analysis assessed the radiological impact due to transportation accidents occurring from each of the 10 major treatment sites to WIPP. For this analysis, a conservative radiological inventory, which assumed that every TRU waste package would be filled with waste containing the highest level of radionuclides and hazardous material allowed by the planning-basis WAC. The total accident impact for each of the 10 sites was obtained by summing the calculated risks of each severity class.

The second analysis assessed four bounding accidents. Two accidents were assumed to involve the breach of a TRUPACT-II, and two accidents involved the breach of an RH-72B. The accidents were assumed to occur under conditions which maximized, within reasonable bounds, the impacts to exposed populations. The probability that such an accident would occur is less than 7.5×10^{-7} for a truck shipment of severity category VIII.

E.4.2.1 Methodology

This section discusses the processes used to calculate the radiological impact due to transportation accidents. Among the elements which are important to such calculations are the severity calculations, release fractions, and the dispersal of released material.

Severity Categories

Most transportation accidents are unlikely to cause any radioactive material release, but very severe accidents may. Thus, the distribution of accidents according to severity must be determined in addition to the overall accident rate.

Accident severity categories define the seriousness of an accident in terms of mechanical and thermal loads. Relevant mechanical parameters include impact speed, force, location and orientation, surface hardness, and puncture characteristics. Thermal characteristics include flame temperature, fire duration, fire source size and orientation with respect to the container, and heat transfer properties (such as flame emissivity and convection coefficients).

NRC defined eight accident severity categories for each mode of transportation in a study assessing the adequacy of regulations for radioactive material transportation (NRC 1977). Severity category I and II accidents are equivalent to the regulatory accident tests required for Type A and Type B packages. By definition, category I and II accidents do not result in any environmental releases because the shipping containers or casks are designed to withstand the accidents. RADTRAN severity categories III through VIII represent accidents that are more severe than the regulatory accident tests.

PROTECTION OF THE TRU WASTE SHIPMENTS FROM SABOTAGE/TERRORIST ACTIVITIES

The Department considers the probability of sabotage or terrorist activities to be very small because TRU waste contains only small, hard-to-recover amounts of plutonium and other materials used to manufacture nuclear weapons. In addition, any act of terrorism or sabotage in which a TRUPACT-II containing the maximum inventory permitted would be breached in an urban area would be unlikely to cause greater impacts than those presented for the accident scenarios in this appendix.

The mass and integrity of the TRUPACT-II and proposed RH-72B packages makes TRU waste shipments unattractive targets for terrorism. The packaging would withstand all but the most extreme efforts to release contaminants. The 1980 FEIS discusses the difficulty of scattering enough material to create a major health hazard. The analysis concluded that more damage would be done by the explosives used to breach the waste packages than by any radioactive materials released.

Although escorts might reduce the already small threat of terrorism, DOE does not believe such escorts are warranted. All nuclear materials are afforded some level of protection, but the level of security provided to a shipment of TRU waste, which contains small amounts of hard-to-recover plutonium, would be considerably less than the level of protection provided to a shipment of material from which nuclear weapons can be made. In the case of TRU waste shipments, current safeguards include the following: the TRANSCOM satellite tracking system would continuously monitor the position and status of shipments en route to WIPP; each vehicle would be equipped with mobile phone communications; the drivers would be required to maintain visual contact with the shipment at all times, even during rest stops; and the drivers would receive specialized training on how to respond to sabotage and terrorism.

Releases from crush impacts were not expected below accident severity category V. Releases from the TRUPACT-II were assumed to be possible during accidents involving fires in category III or above. The large majority of truck (99.90 percent) and rail (99.83 percent) accidents that involve fires last less than 30 minutes (Wolff 1984). The release fractions were combined with the accident rates (probability of fire or impact) for each severity category, the travel distance per shipment, and the fraction of travel through each population density zone to determine an aggregate, probability-weighted consequence for each shipment in terms of radiation dose and LCFs. The probability that an accident will occur diminishes as the accident severity increases. SEIS-II retains the severity classification scheme used by the NRC (1977). The fractional occurrence of truck accidents in each of the eight severity categories is presented in Table E-16.

Table E-16
Fractional Truck Accident Occurrences
by Accident Severity Category and Population Density Zone

Accident Severity Category	Fractional Occurrences	Rural	Suburban	Urban
I	0.55	0.1	0.1	0.8
II	0.36	0.1	0.1	0.8
III	0.07	0.3	0.4	0.3
IV	0.016	0.3	0.4	0.3
V	0.0028	0.5	0.3	0.2
VI	0.0011	0.7	0.2	0.1
VII	8.5E-5	0.8	0.1	0.1
VIII	1.5E-5	0.9	0.05	0.05

Release Fractions

Radionuclide release fractions were assumed to be the same from one alternative to another, which would be a conservative assumption because the inhalation of dispersed radionuclides would be the principal exposure pathway, and the fractions of respirable particles would be greatly reduced for waste treated thermally (Action Alternative 2). More realistic release fractions for thermally treated waste would have resulted in estimated accident impacts that were 1,000 times lower. Radionuclide release fraction analysis in SEIS-I determined how much radioactive material could be released as respirable particulates after a very severe accident that affected the containment capabilities of the shipping containers or casks. Because inhalation is the primary exposure pathway for TRU radionuclides, a mean aerodynamic diameter of less than 10 microns was used in the analysis. Larger particles would be trapped in mucus membranes, filtered, and expelled from the body before they could reach the lungs. This approach was consistent with existing NRC impact assessments (NRC 1977, and Fischer et al. 1987).

The following steps were used to calculate radionuclide release fractions for very severe accidents:

- Characterize the radioactive material being transported

- Identify and quantify breach of the shipping containers or casks due to accident conditions
- Identify and quantify the mechanisms by which radioactive material was released to the environment

SEIS-II analysis used representative parameter values where published data and test results are applicable and reasonable and conservative estimates where uncertainties exist.

To characterize the radioactive material being transported, the radionuclides, quantities, and concentrations used in the analyses were based on waste inventory data and projections presented in Appendix A. DOE has established criteria and procedures that govern the physical, radiological, and chemical composition of TRU waste (DOE 1996a). Physical restrictions require that the waste not be in a free-liquid form. To estimate the impacts for radiological releases in a transportation accident, two radionuclide inventories were determined for both CH-TRU and RH-TRU waste shipments. One inventory was based on maximizing the radionuclide concentrations to the limit allowed in planning-basis WAC, the other was based on average radionuclide concentrations.

For the radiological inventory based on the planning-basis WAC, the PE-Ci activity was maximized using radionuclide-specific weighting factors. To obtain this correlation, the 50-year committed effective dose equivalent (CEDE) or dose conversion factor (DCF) for a unit intake of each radionuclide was used. DCFs were determined by the methodology described in International Commission on Radiological Protection (ICRP) Publications 26 and 30 (ICRP 1977 and ICRP 1978). The radionuclide inventory was obtained by searching through BIR-3 site radionuclide inventories to find the limiting set of radionuclides for CH-TRU and RH-TRU waste.

RELEASE FRACTIONS FOR THERMALLY TREATED WASTE

The products of plasma processing are vitrified glasses and solid metals, which are predicted to withstand severe temperatures. Respirable impact-related release fractions were determined using impact test data for vitrified materials (PNL 1975). The amount of material fractured at an impact velocity of 20 meters per second (66 feet per second) ranged from 0.013 to 0.15 percent. The upper value for this range represented the amount of material released for accident severity category VIII. RADTRAN default values for an immobile material for the aerosol fraction and the respirable fraction were applied to the estimated material released to quantify the respirable impact-related release. Under thermal accident conditions, vitrified materials would be expected to behave like a refractory brick. The primary release mechanism is expected to be the aerosolization of material from contaminated surfaces, and any such releases are expected to occur only at the more severe accident categories involving a prolonged fire (category IV through VIII). The *Nuclear Fuel Cycle Facility Accident Analysis Handbook* (Ayer et al. 1988) recommends a thermal suspension factor of 2.5×10^{-5} per second. This analysis assumed that there would be an effective thermal suspension duration of one hour and that 10 percent of the material fractured would be available for release under accident severity category VIII conditions. Additionally, a decontamination factor of 5×10^{-2} was used for releases from the package cavity to the environment. This would be consistent with values used in Transportation Accident Scenarios for Commercial Spent Nuclear Fuel (Wilmot 1981) and takes credit for mitigation processes that would reduce radioactive material releases such as particulate settlement, plateout, and filtration effects along the leak path. The resulting respirable thermal release fraction was conservatively applied to accident severity categories IV through VIII. The total respirable release fraction was determined by summing the impact and thermal release components.

To determine which site would have average radionuclide inventory, the individual radionuclide concentrations for each site were multiplied by 2.8 cubic meters (100 cubic feet) to obtain the radionuclide inventory per TRUPACT-II. This value was then multiplied by the radionuclide-specific inhalation DCF, expressed in rem per curie (Ci). The resulting doses were summed to get a measure of the relative inhalation hazard associated with each radionuclide inventory. The average radionuclide inventories were found to be at SRS (for CH-TRU waste) and Hanford (for RH-TRU waste).

The bounding (maximized) radionuclide inventory was determined by increasing the average inventories until one of the planning-basis WAC limits (DOE 1996a) was reached. The goal was to increase the radionuclide activity up to the planning-basis WAC PE-Ci limit of 80 PE-Ci grams per drum while not exceeding the fissile-gram limit of 200 grams per drum and 325 grams per TRUPACT-II or the thermal power limit of 40 watts per TRUPACT-II for untreated waste. For treated waste, the PE-Ci limit is 1,800 grams per drum, and the fissile-gram and thermal power limits are the same as for untreated waste. The average inventories for SRS and Hanford were increased by approximately a factor of four. The bounding radionuclide inventories for CH-TRU and RH-TRU waste are shown in Table E-17. When estimating a bounding case inventory for treated waste (Action Alternatives 2 and 3), however, increasing the PE-Ci amount beyond 80 grams per drum would violate planning-basis WAC thermal power or fissile-gram equivalent limits. Therefore, the bounding radionuclide inventory for treated waste was not analyzed.

Table E-17
Bounding Case Radionuclide Inventories for CH-TRU and RH-TRU Waste Accidents [a]

Radionuclide	Adjusted Inventory CH-TRU Waste (Curies per TRUPACT-II)	Adjusted Inventory RH-TRU Waste (Curies per RH-72B)
Co-60	6.4E-4	2.5
Sr-90	0.01	49
Cs-137	0.01	49
U-233	---	0.03
U-235	---	1.0E-3
U-238	---	7.1E-5
Pu-238	990	1,000
Pu-239	16	20
Pu-240	4.2	10
Am-241	3.6	12
Pu-241	200	10
Pu-242	6.8E-4	---

[a] Dashes indicate radionuclides not found in the inventory.

For SEIS-II accident analysis, it was assumed that only one of the TRUPACT-IIs in the shipment would fail in an accident. This assumption was based on Fischer et al. (1987), which found that impact with a hard target, such as a bridge abutment, could potentially breach one TRUPACT-II container per shipment. While a fire of long duration engulfing three TRUPACT-IIs could result in the failure of one or more TRUPACT-IIs, the release fraction from three failed TRUPACT-IIs in such a fire is less than the release fraction from a single package from impact; therefore, the single failure from impact was assumed to be bounding.

Although catastrophic failures would be extremely unlikely, the accident analyses are consistent with the NRC's position (Fischer et al. 1987) and did not take credit for any processes that would reduce radioactive material releases (e.g., particle settlement, vapor plate-out on interior or exterior surfaces, filtration effects along leak path). The immediate release of radioactive material from impact events and the delayed release of radioactive material from fire were assumed; however, TRU waste containers were not assumed to fail from impact until an accident of severity category V was reached. A failure threshold corresponding to severity category III (an accident with conditions slightly exceeding the NRC's test requirements) was conservatively assumed for a fire.

Normally, any release of radioactive material due to a transportation accident would progress in two stages; release inside the shipping containers or casks followed by release to the environment. It was assumed, without regard to waste form or type, that all failed waste containers release an average amount of material for each accident severity category.

In assessing releases from impact events, the following steps were used:

- Identification of the fraction of failed waste containers inside the shipping container or cask
- Determination of the fraction of radioactive material released from failed waste containers
- Calculation of the fraction of radioactive material released from failed waste containers that would aerosolize in a respirable form by the mechanical stress of impact
- Calculation of the fraction of radioactive material released from failed waste containers that becomes aerodynamically entrained in a respirable form after the loss of containment and any subsequent depressurization (e.g., TRUPACT-II design pressure of 50 pounds per square inch gauge [psig])

The algorithm used to calculate the release fraction of respirable radioactive material from impact stresses is presented as Equation E-1:

$$\mathrm{IRF} = (\mathrm{FFC})(\mathrm{FMRC})(\mathrm{FMRPI})(\mathrm{FMAI} + \mathrm{FMEI}) \qquad \text{(Equation E-1)}$$

where

IRF	=	Impact release fraction
FFC	=	Fraction of failed waste containers
FMRC	=	Fraction of material released from failed containers into the package cavity
FMAI	=	Fraction of material aerosolized from impact
FMEI	=	Fraction of material entrained to environment during an impact event
FMRPI	=	Fraction of material released from package cavity during an impact event

Values for specific algorithm parameters are presented in Table E-18.

Fischer et al. (1987) estimated that 1.7 percent of truck accidents would involve fires. For fire, the following method was used for each accident severity category:

- Identification of the fraction of radioactive material subject to thermal release mechanisms
- Calculation of the fraction of radioactive material released in a respirable form by combustion
- Calculation of the fraction of radioactive material released in a respirable form by gases and the heating of contaminated surfaces
- Determination of the fraction of radioactive material released in a respirable form by volatilization of radionuclides

In the absence of detailed knowledge about the responses of shipping and waste containers to fires that are more severe than those specified in Type B packaging requirements, it was conservatively assumed that all radioactive material was available for release, for all accidents exceeding severity category II, as limited by the specific release mechanisms.

For combustion related releases, it was assumed that combustible materials could be ignited in all accident severity categories exceeding category II. The amount of oxygen present to support combustion was calculated by assuming a loaded shipping container has an 85 percent void volume

Table E-18
Impact Release Algorithm Parameters for CH-TRU and RH-TRU Waste Shipments

Parameters	Value	Basis/Reference
FFC	0.2728 lnF -2.814	(DOE 1990) where F is NRC (1977) accident severity breach force (Newtons)
FMRC	From columns below	(DOE 1990) and NRC (1977) used as guidance
FMAI	From columns below	(DOE 1990) resuspension factor of 2.00×10^{-2} m^{-1} used (mechanical stress of vigorous sweeping)
FMEI	1.50×10^{-4}	(DOE 1990) average entrainment value for 4 surfaces used with airflow of 2.5 mph for 30 minutes
FMRPI	Accident severity 1-4: 0.0 Accident severity 5-8: 1.0	Type B package design and NRC (1977) used as guidance

Severity Category	FMRC	FMAI	FMEI	FMRPI	TRUPACT-II		RH-72B Cask	
					FFC	IRF	FFC	IRF
I	0	0	0	0	0	0	0	0
II	0	0	0	0	0	0	0	0
III	0.1	8×10^{-5}	0	0	0.3	0	0.3	0
IV	0.3	8×10^{-5}	0	0	0.5	0	0.7	0
V	0.5	8×10^{-5}	1.5×10^{-4}	1	0.7	8×10^{-5}	1	1×10^{-4}
VI	0.7	8×10^{-5}	1.5×10^{-4}	1	1	2×10^{-4}	1	1×10^{-4}
VII	1	8×10^{-5}	1.5×10^{-4}	1	1	2×10^{-4}	1	2×10^{-4}
VIII	1	8×10^{-5}	1.5×10^{-4}	1	1	2×10^{-4}	1	2×10^{-4}

and that there would be no external sources of air or oxygen (no major breach of container). The results of experiments conducted by Mishima and Schwendiman (DOE 1990) were used to assess the fraction of radioactive material released in a respirable form by the burning of combustible material.

Under accident severity categories IV through VIII, the fire may last longer than 1.5 hours. It was assumed, therefore, that more radioactive material could be converted to an aerosol because of the release of gases from the waste at elevated temperatures. Vaporization was considered as another potential release mechanism during a fire. Alexander et al. (DOE 1990) reports that volatile releases of TRU radionuclides are not of any significance until temperatures of 1,727 Celsius (°C) (3,140°Fahrenheit [°F]) are reached. It was concluded that potential accidents involving CH-TRU waste shipments cannot result in radioactive material releases in a vapor form.

The volatilization of uranium oxide becomes measurable at approximately 1,627°C (2,960°F). Flame temperatures pursuant to the open burning of hydrocarbon fuels (e.g., JP-4, gasoline, diesel) range from 760°C to 1,316°C (1,400°F to 2,400°F), with a median temperature of approximately 982°C (1,800°F). Consequently, a volatile release of material containing TRU radionuclides or uranium oxide would not be reasonably foreseeable for a transportation accident.

RH-TRU waste contains activation/fission products that may volatilize at elevated temperatures. Because the proportion of these products is uncertain, it was assumed that their release would be based on the values estimated for respirable particulate releases. The algorithm for estimating the respirable release fraction of radioactive material from thermal accident events is presented as follows:

$$\mathrm{TRF} = (\mathrm{FAT})(\mathrm{FMRPT})[(\mathrm{FMC})(\mathrm{FMAC}) + (\mathrm{FMAT})] \qquad \text{(Equation E-2)}$$

where

TRF	=	Thermal release fraction
FAT	=	Fraction of accidents involving a thermal event
FMC	=	Fraction of material consumed by combustion
FMAC	=	Fraction of material aerosolized by combustion
FMAT	=	Fraction of material aerosolized by thermal event
FMRPT	=	Fraction of material released from the package cavity during a thermal event

Values for specific algorithm parameters are summarized in Table E-19.

The calculated release fractions from impact (Table E-18) and fires (Table E-19) were added to determine the total respirable release fractions due to very severe transportation accidents and are summarized in Tables E-20 and E-21 for CH-TRU and RH-TRU waste, respectively. A maximum release fraction of 2×10^{-4} was estimated for accidents involving both CH-TRU and RH-TRU waste shipments. This is consistent with or bounds previous transportation impact studies such as SEIS-I (DOE 1990) and the NRC modal study (Fischer et al. 1987), which estimated particulate releases of 2×10^{-6} and vapor (C_s) releases of 2×10^{-4} due to spent fuel shipments.

Table E-19
Thermal Release Algorithm Parameters for CH-TRU and RH-TRU Waste Shipments

Parameter	Value	Basis/Reference
FAT	1.7×10^{-2} (Truck)	Fischer et al. (1987)
FMC	Accident severity I and II: 0	No internal combustion
	Accident severity III and IV: 9×10^{-4} (TRUPACT-II) 7×10^{-4} (RH-72B Cask)	Type B package design Limited internal oxygen source: 3.95 lb O_2 (TRUPACT-II) 0.73 lb O_2 (RH-72B)
FMAC	Accident severity I and II: 0 Accident severity III - VIII: 5×10^{-4}	Type B package design Mishima and Schwendiman (DOE 1990)
FMAT	Accident severity I and II: 0	Type B package design
	Accident severity III: 2×10^{-8}	Only combustion assumed to occur, with attendant off-gas (combustion) products
	Accident severity IV - VIII: 1×10^{-5} (TRUPACT-II) 9×10^{-6} (RH-72B Cask)	Off-gassing assuming steam/graphite reaction and resuspension factor of 5.0×10^{-6} meter^{-1} corresponding to a surface stress from walking (DOE 1990)
FMRPT	Accident severity I and II: 0 Accident severity III - VIII: 1	Type B package design NRC (1977) used as guidance

Severity Category	FMC	FMAC	FMAT	FMRPT	FAT	TRF
TRUPACT-II						
I	0	0	0	0	0.02	0
II	0	0	0	0	0.02	0
III	9×10^{-4}	5×10^{-4}	2×10^{-8}	1	0.02	8×10^{-9}
IV	9×10^{-4}	5×10^{-4}	1×10^{-5}	1	0.02	2×10^{-7}
V	9×10^{-4}	5×10^{-4}	1×10^{-5}	1	0.02	2×10^{-7}
VI	9×10^{-4}	5×10^{-4}	1×10^{-5}	1	0.02	2×10^{-7}
VII	9×10^{-4}	5×10^{-4}	1×10^{-5}	1	0.02	2×10^{-7}
VIII	9×10^{-4}	5×10^{-4}	1×10^{-5}	1	0.02	2×10^{-7}
RH-72B Cask						
I	0	0	0	0	0.02	0
II	0	0	0	0	0.02	0
III	7×10^{-4}	5×10^{-4}	2×10^{-8}	1	0.02	6×10^{-9}
IV	7×10^{-4}	5×10^{-4}	9×10^{-5}	1	0.02	2×10^{-7}
V	7×10^{-4}	5×10^{-4}	9×10^{-5}	1	0.02	2×10^{-7}
VI	7×10^{-4}	5×10^{-4}	9×10^{-5}	1	0.02	2×10^{-7}
VII	7×10^{-4}	5×10^{-4}	9×10^{-5}	1	0.02	2×10^{-7}
VIII	7×10^{-4}	5×10^{-4}	9×10^{-5}	1	0.02	2×10^{-7}

Table E-20
CH-TRU Waste Truck Transportation Release Fractions

Total Respirable Release Fraction (TRRF) = Impact Release Fraction (IRF) + Thermal Release Fraction (TRF)			
Accident Severity Category	**IRF [a]**	**TRF [b]**	**TRRF**
I	---	---	---
II	---	---	---
III	---	8×10^{-9}	8×10^{-9}
IV	---	2×10^{-7}	2×10^{-7}
V	8×10^{-5}	2×10^{-7}	8×10^{-5}
VI	2×10^{-4}	2×10^{-7}	2×10^{-4}
VII	2×10^{-4}	2×10^{-7}	2×10^{-4}
VIII	2×10^{-4}	2×10^{-7}	2×10^{-4}

[a] From Table E-18
[b] From Table E-19

Table E-21
RH-TRU Waste Truck Transportation Release Fractions

Total Respirable Release Fraction (TRRF) = Impact Release Fraction (IRF) + Thermal Release Fraction (TRF)			
Accident Severity Category	**IRF [a]**	**TRF [b]**	**TRRF**
I	---	---	---
II	---	---	---
III	---	6×10^{-9}	6×10^{-9}
IV	---	2×10^{-7}	2×10^{-7}
V	1×10^{-4}	2×10^{-7}	1×10^{-4}
VI	1×10^{-4}	2×10^{-7}	1×10^{-4}
VII	2×10^{-4}	2×10^{-7}	2×10^{-4}
VIII	2×10^{-4}	2×10^{-7}	2×10^{-4}

[a] From Table E-18
[b] From Table E-19

Atmospheric Dispersion and Exposure Pathways

The dispersion of airborne radioactive material during an accident is dependent on meteorologic conditions at the time of the accident. Airborne radioactive material moves downwind from the scene of the accident; its dispersal and transport are affected by the degree of atmospheric turbulence. Large areas may be affected. The degree of dispersion is influenced by factors, such as the season (which influences atmospheric turbulence), time of day, degree of cloud cover, wind speed, land surface features and characteristics, and other meteorologic parameters.

RADTRAN or similar analytical tools can be used to evaluate the impacts of radioactive material released under transportation-accident conditions. Exposure pathways must be identified and the size of exposed populations must be estimated for input to RADTRAN. Transportation accidents may be divided into two categories: those accidents in which the shipping containers maintain their integrity (no release of radioactive material) and those accidents in which the integrity of the shipping containers is compromised. The exposure pathways and the exposed population subgroups are discussed below.

In an accident in which shipping containment is not breached, the only exposure pathway would be direct, external radiation from the intact package. The radiation dose to any member of an exposed population would be evaluated in the same manner as the exposure from normal (accident-free) transportation; adjustments would be made for the duration of exposure and the distance between the shipment and the exposed individuals. Potentially exposed populations include the truck crew, the occupants of the other vehicle(s) involved in the accident, bystanders/pedestrians, the occupants of nearby buildings, and emergency response crews.

In an accident that results in a failure of the shipping containers and the possible release of radioactive material, radiation exposures may result from both nondispersible and dispersible material. The exposure pathway from accidents involving shipping containers with nondispersible material would be direct, external radiation. Certain radioactive material forms are not dispersible because of their chemical or physical form (e.g., irradiated steel hardware); these materials may nevertheless expose individuals to penetrating radiation. The radiation doses received by exposed individuals would be evaluated in the same manner as other direct exposures. Adjustments would be made for the increased dose rates that result from a loss of shielding, as well as exposure time and distances. The exposed populations would be the same as those identified above.

According to SEIS-I (DOE 1990), there are four potential exposure pathways from accidents that could cause a release of dispersible radioactive materials:

- Cloudshine would be the pathway of direct *external* dose from the passing cloud of dispersed radioactive material. Dispersion depends on the meteorologic conditions at the accident scene, the fraction of failed shipping containers, and the fraction of released material that becomes airborne.

- Groundshine would be the pathway of direct *external* dose from material that has deposited on the ground after being dispersed from the accident site.

- Inhalation would be the pathway of intake of respirable radioactive material that may result in internal radiation doses. Doses from inhalation depend on the fraction of failed shipping containers, the fraction of airborne material, the aerosol fraction of respirable size, the dilution factor for radioactive material in the surrounding air, the breathing rate of the exposed individual, and the radiation dose per curie of radionuclide inhaled.

- Resuspension would be a secondary inhalation pathway that exists when radioactive material that was dispersed and deposited is disturbed, becomes airborne, and is inhaled. Radiation dose assessment of this pathway would require combining the mechanisms of dispersion, deposition, and inhalation, as well as estimating the fraction of deposited material that is resuspended. Resuspension may result from changes in wind speed or direction or from disturbing deposited material by other means, such as traffic through a deposition area.

The estimated population doses were calculated using a resuspension particle half-life of 365 days. The resuspension half-life is the time required for half of the initially deposited material to be unavailable for resuspension. A resuspension half-life of 365 days would be extremely conservative given the washing (rain) and weathering (wind) processes that would serve to remove contaminants from the accessible environment. Because inhalation of resuspended particles would be a primary contributor to the estimated population dose, this conservative value was chosen. The assumed population density would also affect the total calculated dose and estimated health effects.

Exposure by ingestion was not included in evaluation of the radiological impacts of accidents. It was assumed that emergency response and governmental authorities would intervene to impound contaminated foodstuffs, provide an alternative water supply, and clean up contaminated land. The bounding accident was assumed to occur in an urban area to maximize the exposed population, which eliminates the ingestion pathway.

The population subgroups that would be exposed by an accident that results in the dispersion of radioactive material include the individuals directly exposed at the scene of the accident and individuals present in the areas over which dispersion would occur.

E.4.2.2 Results

The radiological consequences of truck transportation accidents were apportioned among the eight severity categories, each of which is associated with a release fraction and a likelihood of occurrence and calculated for truck transportation. No release was assumed for accidents assigned to severity category I or II.

No releases from crush impacts were assumed for accidents below severity category VI. Releases from the TRUPACT-II were assumed to be possible during accidents involving fires in category III or above. Release fractions for each severity category were combined with the accident rates for each category, the probability of a fire or impact, the travel distance per shipment, the total number of shipments, and the travel fraction through each population density zone to determine an aggregate, probability-weighted consequence.

The results of the analyses of aggregate radiological impacts due to severe transportation accidents are presented by origination site in Table E-22. The sum of the consequences was multiplied by the likelihood of occurrence for the eight accident severity categories to estimate the impacts. Site-specific information and the bounding case radionuclide inventory for untreated waste were used. Although there are different release fractions for untreated, thermally treated, and shred and grout treated waste, the untreated release fractions were used to determine population doses in order to bound the impacts.

E.4.3 Radiological Impacts of Bounding-Case Transportation Accidents

Bounding-case transportation accident scenarios were used to calculate the impact of very severe accidents (category VIII) in higher population areas (urban) along designated transportation routes. Accidents were postulated and analyzed for both CH-TRU and RH-TRU waste shipments. Both the average and the bounding radionuclide inventories were used in these analyses.

Table E-22
Aggregate Radiological Impacts from Severe Accidents (person-rem)[a, b, c]

Impacts from Accidents	Proposed Action-Scaled to WIPP Max	Action Alternative 1			Action Alternative 2A			Action Alternative 2B			Action Alternative 2C		
	Basic Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste													
Argonne National Laboratory-East	1 (5.0E-4)	0.8 (4.0E-4)	---	0.8 (4.0E-4)	---	---	---	---	---	---	0.7 (3.5E-4)	---	0.7 (3.5E-4)
Hanford Site	590 (0.3)	500 (0.3)	310 (0.2)	810 (0.5)	355 (0.2)	380 (0.2)	735 (0.4)	355 (0.2)	380 (0.2)	735 (0.4)	500 (0.3)	310 (0.2)	810 (0.5)
Idaho National Engineering and Environmental Laboratory/ANL-W	155 (0.08)	130 (0.07)	175 (0.09)	305 (0.2)	110 (0.06)	330 (0.2)	440 (0.3)	230 (0.1)	385 (0.2)	615 (0.3)	130 (0.07)	180 (0.09)	310 (0.2)
Lawrence Livermore National Laboratory	3 (1.5E-3)	3 (1.5E-3)	---	3 (1.5E-3)	---	---	---	---	---	---	3 (1.5E-3)	---	3 (1.5E-3)
Los Alamos National Laboratory	7 (3.5E-3)	6 (3.0E-3)	2 (1.0E-3)	8 (4.0E-3)	4 (2.0E-3)	3 (1.5E-3)	7 (3.5E-3)	---	---	---	6 (3.0E-3)	2 (1.0E-3)	8 (4.0E-3)
Mound Plant	2 (1.0E-3)	2 (1.0E-3)	---	2 (1.0E-3)	---	---	---	---	---	---	2 (1.0E-3)	0.1 (5.0E-5)	2 (1.0E-3)
Nevada Test Site	1 (5.0E-4)	1 (5.0E-4)	---	1 (5.0E-4)	---	---	---	---	---	---	1 (5.0E-4)	---	1 (5.0E-4)
Oak Ridge National Laboratory	6 (3.0E-3)	5 (2.5E-3)	1.0 (5.0E-4)	6 (3.0E-3)	---	---	---	---	---	---	5 (2.5E-3)	0.2 (1.0E-4)	5 (2.5E-3)
Rocky Flats Environmental Technology Site	16 (8.0E-3)	14 (7.0E-3)	---	14 (7.0E-3)	10 (5.0E-3)	---	10 (5.0E-3)	---	---	---	14 (7.0E-3)	---	14 (7.0E-3)
Savannah River Site	48 (0.02)	41 (0.02)	12 (6.0E-3)	53 (0.03)	44 (0.02)	16 (8.0E-3)	60 (0.03)	44 (0.02)	16 (8.0E-3)	60 (0.03)	41 (0.02)	12 (6.0E-3)	53 (0.03)
RH-TRU Waste													
Hanford Site	8 (4.0E-3)	125 (0.06)	4 (2.0E-3)	130 (0.07)	47 (0.02)	3 (1.5E-3)	50 (0.03)	47 (0.02)	3 (1.5E-3)	50 (0.03)	47 (0.02)	3 (1.5E-3)	50 (0.03)
Idaho National Engineering Laboratory/ANL-W	5 (2.6E-3)	5 (2.5E-3)	1 (5.0E-4)	6 (3.0E-3)	---	---	---	---	---	---	---	---	---
Los Alamos National Laboratory	0.03 (1.5E-5)	0.03 (1.5E-5)	0.01 (5.0E-6)	0.04 (2.0E-5)	---	---	---	---	---	---	---	---	---
Oak Ridge National Laboratory	2 (1.0E-3)	9 (4.5E-3)	5 (2.5E-3)	14 (7.0E-3)	3 (1.5E-3)	2 (1.0E-3)	5 (2.5E-3)	3 (1.5E-3)	2 (1.0E-3)	5 (2.5E-3)	3 (1.5E-3)	2 (1.0E-3)	5 (2.5E-3)
Total CH-TRU and RH-TRU Waste	850	840	510	1.4E+3	570	735	1.3E+3	680	785	1.5E+3	760	510	1.3E+3
Total LCFs	0.4	0.5	0.3	0.8	0.3	0.4	0.7	0.3	0.4	0.7	0.4	0.3	0.7

[a] Mathematically expected LCFs are in parentheses.
[b] Total values may differ due to rounding.
[c] Dashes indicate that there would be no shipments to WIPP.

Table E-22
Aggregate Radiological Impacts from Severe Accidents (person-rem) —Continued [a, b, c]

Impacts from Accidents	Action Alternative 3			No Action Alternative 1A			No Action Alternative 1B		
	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
CH-TRU Waste									
Argonne National Laboratory-East	---	---	---	0.7 (3.5E-4)	---	0.7 (3.5E-4)	0.7 (3.5E-4)	---	0.7 (3.5E-4)
Hanford Site	1.1E+3 (0.6)	370 (0.2)	1.5E+3 (0.8)	---	---	---	---	---	---
Idaho National Engineering and Environmental Laboratory/ANL-W	280 (0.1)	210 (0.1)	490 (0.2)	---	---	---	---	---	---
Lawrence Livermore National Laboratory	---	---	---	2.7 (1.3E-3)	---	2.7 (1.3E-3)	2.7 (1.3E-3)	---	2.7 (1.3E-3)
Los Alamos National Laboratory	10 (5.0E-3)	3 (1.5E-3)	13 (6.5E-3)	---	---	---	5.7 (2.9E-3)	2.1 (1.1E-3)	7.8 (4.0E-3)
Mound Plant	---	---	---	1.8 (9.0E-4)	0.1 (5.4E-5)	1.9 (9.5E-4)	1.8 (9.0E-4)	0.1 (5.4E-5)	1.9 (9.5E-4)
Nevada Test Site	---	---	---	1 (5.0E-4)		1 (5.0E-4)	1 (5.0E-4)	---	1 (5.0E-4)
Oak Ridge National Laboratory	---	---	---	4.7 (2.4E-3)	0.2 (1.0E-4)	4.9 (2.5E-3)	4.7 (2.4E-3)	0.2 (1.0E-4)	4.9 (2.5E-3)
Rocky Flats Environmental Technology Site	19 (0.01)	---	19 (0.01)	---	---	---	13.6 (6.8E-3)	---	13.6 (6.8E-3)
Savannah River Site	62 (0.03)	15 (7.5E-3)	77 (0.04)	---	---	---	---	---	---
RH-TRU Waste									
Hanford Site	160 (0.08)	8 (4.0E-3)	170 (0.09)	---	---	---	---	---	---
Idaho National Engineering Laboratory/ANL-W	6 (3.0E-3)	1 (5.0E-4)	7 (3.5E-3)	1.8 (9.0E-4)	0.6 (3.0E-4)	2.4 (1.2E-3)	1.8 (9.0E-4)	0.6 (3.0E-4)	2.4 (1.2E-3)
Los Alamos National Laboratory	0.04 (2.0E-5)	0.02 (1.0E-5)	0.06 (3.0E-5)	0.01 (5.0E-6)	5.4E-3 (2.7E-6)	0.02 (7.7E-6)	0.01 (5.0E-6)	5.4E-3 (2.7E-6)	0.02 (7.7E-6)
Oak Ridge National Laboratory	10 (5.0E-3)	6 (3.0E-3)	16 (8.0E-3)	---	---	---	---	---	---
Total CH-TRU and RH-TRU Waste	1.7E+3	610	2.3E+3	12.7	0.9	13.6	32	3	35
Total LCFs	0.9	0.3	1.2	6.3E-3	4.5E-4	6.8E-3	0.02	1.5E-3	0.02

[a] Mathematically expected LCFs are in parentheses.
[b] Total values may differ due to rounding.
[c] Dashes indicate that there would be no shipments to WIPP.

Note: Shipments for No Action Alternative 1A and 1B are the number of shipments to consolidate TRU waste. No TRU waste would be shipped to WIPP.

E.4.3.1 Assumptions

The assumptions made regarding the bounding-case transportation accident scenarios are as follows:

- Impacts were analyzed without regard to the likelihood of the accident actually occurring.
- The waste shipment would be three fully-loaded TRUPACT-IIs on a combination tractor-trailer truck. Two types of inventories were considered for the bounding case accident. One was based on the average concentrations of radionuclides. The second was chosen to maximally bound the impacts by loading the TRUPACT-II up to the planning-basis WAC limit.
- Accidents were severity category VIII.
- All waste was packaged in Type A drums.
- A minor breach occurred, limiting external oxygen sources.
- A 0.02 percent fraction of the radioactive waste material was released to the environment in a respirable form (less than 10 microns in diameter).
- Radioactive material was evenly distributed throughout the waste volume.
- Accidents occurred in the urban portion of a nonspecific, large metropolitan area (population greater than one million) with a mean population density of 3,861 persons per square kilometer (10,003 persons per square mile).
- Accidents occurred during very stable meteorologic conditions, limiting dispersion of the radioactive material plume and maximizing radiation doses.
- In the accident scenarios involving truck transportation, one TRUPACT-II or RH-72B was breached and subsequently engulfed in fire for two hours.
- In the accident scenarios involving rail transportation, two TRUPACT-IIs or RH-72Bs were breached and subsequently engulfed in fire for two hours.
- Doses were calculated using a resuspension particle half-life of 365 days.

For the CH-TRU waste bounding accident, the radionuclide content (curies per TRUPACT-II) was increased to reach the planning-basis WAC limit for both PE-Ci and fissile-gram equivalents. This increased the amount of Pu-238 in a TRUPACT-II by approximately an order of magnitude. Due to the low likelihood of encountering very stable atmospheric conditions, a bounding radionuclide inventory, and the large population densities, radiological doses are likely to be considerably lower than the calculated values.

For the RH-TRU waste bounding accident, the amount of Pu-238 and Pu-239 was increased by factors of 1,000 and 10, respectively. This was done to maximize the consequences of an accident to both the population and the MEI. Based on reported radionuclide inventories from each site, actual doses from an accident would be expected to be an order of magnitude lower.

E.4.3.2 Results

Because it was assumed that the accidents occurred in an urban area, impacts from the ingestion of contaminated agricultural products were not applicable. Population and MEI doses were calculated by RADTRAN.

No early fatalities or morbidities were estimated. Inhalation (initial or resuspension) was the dominant contributor to radiation doses. The impacts associated with the bounding-case accidents would be as follows:

- For the bounding inventory in a breached TRUPACT-II, the total population dose was estimated to be 31,800 person-rem, resulting in 16 LCFs in the exposed population. The estimated dose to the MEI would be 123 rem total effective dose equivalent (TEDE), resulting in a 0.06 probability of an LCF.

- For the bounding inventory in a breached RH-72B, the total population dose was estimated to be 32,500 person-rem, resulting in 16 LCFs in the exposure population. The MEI dose would be 125 rem (TEDE), resulting in a 0.06 probability of an LCF.

- For the average radionuclide inventory in a breached TRUPACT-II, a total population dose of 6,370 person-rem was estimated. This would result in approximately 3 LCFs in the exposed population. The estimated MEI dose would be 80 rem (TEDE), resulting in a 0.04 probability of an LCF.

- For the average radionuclide inventory in a breached RH-72B, the total population dose would be 72 person-rem, resulting in an expectation of 0.04 LCF in the population. The estimated MEI dose would be 1.4 rem (TEDE), resulting in a 7×10^{-4} probability of an LCF.

The same bounding case accidents were analyzed for thermally treated TRU waste. The release fraction would be reduced by a factor of 1,000, however, thermal treatment increases the concentration of radionuclides by approximately a factor of 2.8. The combination of these two factors reduces the overall radiological impacts from bounding case accidents for thermally treated waste. The impacts would be as follows:

- For the bounding radionuclide inventory in a breached TRUPACT-II, the total population dose resulted in 0.09 LCFs in the exposed population. The estimated MEI dose resulted in a 3×10^{-4} probability of an LCF.

- For the bounding radionuclide inventory in a breached RH-72B, the total population dose resulted in 0.09 LCFs in the exposure population. The estimated MEI dose resulted in a 3×10^{-4} probability of an LCF.

- For the average radionuclide inventory in a breached TRUPACT-II, approximately 0.02 LCFs in the exposed population would be expected. The estimated MEI dose resulted in a 2×10^{-4} probability of an LCF.

- For the average radionuclide inventory in a breached RH-72B, the total population dose resulted in an expectation of 2×10^{-4} LCFs in the population. The estimated MEI dose resulted in a 4×10^{-6} probability of an LCF.

E.5 HAZARDOUS CHEMICAL IMPACTS FROM TRUCK TRANSPORTATION ACCIDENTS

This section evaluates the impacts associated with exposures to hazardous chemicals during the transportation of TRU waste to WIPP. Hazardous chemicals in TRU mixed waste occur as volatile organic compounds (VOC) and metals. Accidents involving hazardous chemicals and metals are evaluated as acute-release events with respect to potential exposures and associated impacts.

During accident-free transportation, exposure to hazardous chemicals and metals would be unlikely because the hazardous chemical components in the waste are completely contained in the transportation container/cask. Thus, no impacts to human health are posed by the hazardous chemical components under accident-free transportation.

The analyses used to assess impacts from hazardous chemicals exposures during transportation accidents was based on those in SEIS-I (DOE 1990).

E.5.1 Methodology

Hazardous material inventories were developed as described in Appendix A. For those VOCs where maximum levels are stipulated in the planning-basis WAC, those levels were assumed to be in the containers/casks being transported during an accident. Where no maximum level was specified in planning-basis WAC, the highest level found during waste drum sampling was selected to ensure that the typical concentration would be bounded. Details can be found in Appendix A.

Because a TRUPACT-II is likely to hold nearly three times the waste volume with hazardous chemicals than an RH-72B would hold, it was assumed that CH-TRU waste hazardous chemical accident scenarios would bound RH-TRU waste accidents. Therefore, no hazardous chemical accidents were analyzed for RH-TRU waste shipments. For the purposes of analyses, it was assumed that the RH-TRU hazardous chemical inventory would be the same as the CH-TRU hazardous chemical inventory.

Although the likelihood that a TRUPACT-II would be breached is low, such an accident would be foreseeable and constitute a source of potential hazardous chemical exposures. The hazardous chemical assessment was conservatively based on a very severe transportation accident. It was assumed that an accident would result in the breach of one of three TRUPACT-IIs and in a fire engulfing all three. This bounding-case accident scenario was also based on the assumption that the entire releasable fraction of each chemical considered was used. The assumptions used in the radiological accident assessment provide the basis for the impacts of accidents involving the chemical components of the waste. The hazardous chemical impact was compared to the maximum airborne chemical concentrations for a member of the public and the immediately dangerous to life or health (IDLH) values. Ratios smaller than one were considered to have no impact.

Based on a 30-minute exposure period and an individual who inhales 10 cubic meters (353 cubic feet) of contaminated air, the IDLHs were originally developed by the National Institute of Occupational Safety and Health (NIOSH) for emergency response purposes. The IDLH-equivalent intake level is the quantity of material inhaled during 30 minutes of exposure. The hazardous constituents analyzed for these accident scenarios and the IDLH values and IDLH-equivalent intake values are shown in Table E-23.

Table E-23
Chemical Constituents Analyzed in CH-TRU Waste

Chemical Name	IDLH (parts per million)	IDLH (milligrams per cubic meter)
Carbon Tetrachloride	200	1,278
Chloroform	500	2,480
Methylene Chloride	2,300	8,119
1,1,2,2-Tetrachloroethane	100	700
Chlorobenzene	1,000	4,680
Methyl Ethyl Ketone	3,000	9,000
Toluene	500	1,915
1,2 Dichloroethane	50	206
Beryllium	N/A	4
Cadmium	N/A	9
Lead	N/A	100
Mercury	N/A	10

N/A = Not Applicable

The following assumptions were used to maximize the hazardous chemical concentrations within the breached TRUPACT-II:

- Nonflammable VOCs with planning-basis WAC-prescribed limits (carbon tetrachloride, chloroform, and methylene chloride) were assumed to be at those limits.

- Flammable VOCs were assumed to have a maximum concentration of 500 parts per million because they are limited to that in planning-basis WAC.

- VOCs without planning-basis WAC limits were assumed to be present at the maximum concentrations identified to date during sampling of CH-TRU waste.

- Heavy metals (lead, mercury, beryllium, and cadmium) were assumed to be uniformly mixed in the waste container and found in the containers in average amounts (see Appendix A). Metals would be released as particulates; therefore, the calculations used to determine the radioactive material particulate releases were applied to these as well.

Carbon tetrachloride, trichloroethylene, and methylene chloride are considered potential carcinogens by the EPA, and 1,1,1-trichloroethane may produce adverse somatic effects. Lead is the most abundant metal found in the waste by both weight and volume (DOE 1996a).

Despite the fact that a fire would destroy virtually all waste VOCs, it was conservatively assumed that VOCs were released from the breached TRUPACT-II in their entirety. The air concentrations of each hazardous chemical for the maximally exposed member of the public at the scene of the accident were determined using the Gaussian dispersion plume equation of Pasquill, as modified (DOE 1995b). Ground-level concentrations were calculated at the centerline of the plume.

Plume depletion effects from particulate settlement (by gravitational or chemical effects) were not considered. Therefore, air concentrations and the resulting intakes by inhalation were overestimated for particulate metals but not for VOCs. Additionally, each accident was postulated to occur during a period of very stable meteorologic conditions (Pasquill Stability Class F, wind speed of 1 meter per second) to introduce additional conservatism into the analyses.

The effective height of the plume from the accident was estimated to be approximately 21 meters (69 feet), which would account for the buoyancy rise associated with the thermal effects from the accident. The maximum airborne chemical concentration inhaled by a member of the public was calculated as the following:

$$Ar = \left(\frac{E}{Q}\right) R \qquad \text{(Equation E-3)}$$

where

Ar = Air concentration at the location of the least plume dispersion (maximum impact)

E/Q = Dispersion estimate, 1.13×10^{-4} seconds per cubic meter (3.2×10^{-6} seconds per cubic foot)

R = Hazardous chemical release rate over the assumed 7,200-second release period in milligrams per second

E.5.2 Results

Hazardous chemical impacts were evaluated for a bounding, severity category VIII accident (likelihood of occurrence of 1×10^{-6}). The MEI (receptor) was assumed to be located 1,000 meters (3,300 feet) downwind from the accident, exposed at the centerline of the plume for two hours under very stable meteorologic conditions and low wind speed. The hazardous chemicals analyzed and the impacts to the MEI as a fraction of the chemical-specific IDLH value are presented in Table E-24. For all chemicals analyzed, the concentration to which the MEI would be exposed would be no more than approximately 1.4×10^{-3} (for beryllium) of the chemical's IDLH value. Therefore, no human health effects would be expected from acute exposure to hazardous chemicals released from a severe transportation accident.

E.6 IMPACTS OF CONSOLIDATION

The preceding sections address the impacts of transporting TRU waste from the consolidation or treatment sites to WIPP. These impacts were used to estimate the impacts of shipping TRU waste from a small quantity site to one of the consolidation or treatment sites based on the total miles traveled. Also included in this section are the impacts of transporting waste from the 10 major generator-storage sites under Action Alternative 2 and Action Alternative 3, and No Action Alternatives 1A and 1B (Table E-25) which do not treat their waste. Table E-26 presents the total one-way miles for the Proposed Action and alternatives for shipment of TRU waste from large quantity sites. The waste is consolidated and treated at one of the sites from the list of the 10 major generator-storage sites and then the treated waste is shipped to WIPP.

Table E-24
Chemical Airborne Releases for a Severity Category VIII Accident
(CH-TRU Waste Truck Shipment)

Chemical	Quantity (milligrams per cubic meter)	Quantity (milligrams)	Release Rate (milligrams per second) [a]	Maximum Receptor Air Concentration (milligrams per cubic meter) [b, c]	Concentration per IDLH Value
Carbon Tetrachloride	1.2E+04	3.4E+04	2.4	5.4E-04	4.2E-07
Chloroform	6.8E+02	2.0E+03	1.4E-01	3.1E-05	1.3E-08
Methylene Chloride	2.3E+04	6.8E+04	4.7	1.1E-03	1.3E-07
1,1,2,2-Tetrachloroethane	3.1E+04	8.9E+04	6.2	1.4E-03	2.0E-06
Chlorobenzene	4.5E+03	1.3E+04	9.0E-01	2.0E-04	4.4E-08
Methyl Ethyl Ketone	8.8E+03	2.6E+04	1.8	4.0E-04	4.5E-08
Toluene	2.9E+03	8.5E+03	5.9E-01	1.3E-04	7.0E-08
1,2 Dichloroethane	2.3E+03	6.7E+03	4.7E-01	1.1E-04	5.1E-07
Beryllium	1.2E+05	3.5E+05	9.7E-03	5.5E-03	1.4E-03
Cadmium	1.9E+03	5.6E+03	1.6E-04	8.8E-05	9.8E-06
Lead	4.8E+06	1.4E+07	3.9E-01	2.2E-01	2.2E-03
Mercury	2.1E+06	6.0E+06	1.7E-01	9.4E-02	9.4E-03

[a] Release rate = Release fraction x quantity of hazardous constituent present in a single TRUPACT-II x 1/7200 seconds x Quantity Released
[b] The receptor is the public MEI
[c] Receptor Concentration = E/Q' (max individual) x release rate (milligrams per second); = 1.13 x 10^{-4} (seconds/cubic meter) x Release Quantity (milligrams) / 7200 seconds; assumes a two hour release.

Table E-25
Transportation Impacts for Consolidation of Waste from Major Generator-Storage Sites

Impact Category	Action Alternative 2A	Action Alternative 2B	Action Alternative 2C	Action Alternative 3	No Action Alternative 1A	No Action Alternative 1B
Nonradiological Impacts						
Emission-related LCFs	9.7E-3	0.04	7.3E-3	9.8E-3	9.7E-3	0.04
Vehicle Related Fatalities	0.2	0.8	0.2	0.2	0.2	0.8
Accident-free Radiological Impacts						
Occupational (person-rem)	26	170	20	26	26	170
Nonoccupational (person-rem)	465	1.3E+3	435	465	465	1.3E+3
Occupational (LCFs)	0.01	0.07	8.2E-3	0.01	0.01	0.07
Nonoccupational (LCFs)	0.2	0.7	0.2	0.2	0.2	0.7

Table E-27 presents the total one-way miles under the Proposed Action and the alternatives for the shipment of TRU waste from the small quantity sites to the consolidation sites. Chapter 3 provides the details of the consolidation. The mileage in this table along with the total miles for shipments to WIPP were used to adjust the impact results estimated for the consolidation/treatment sites. As shown in Table E-28, the impacts from the small quantity sites is a small fraction of the impacts of transportation to WIPP.

E.7 IMPACTS FROM RAIL TRANSPORTATION

This section presents a summary of transportation impacts for both regular rail and dedicated rail. Transportation by rail could be conducted from eight of the 10 major generator-storage sites. Truck shipments would be used from the sites with only small amounts of waste and from those sites without rail spurs.

Table E-26
Truck Mileages for the Consolidation of CH-TRU and RH-TRU Waste from Major Generator Sites to Treatment Sites

	CH-TRU Waste	RH-TRU Waste
Alternatives	Total One-way Mileage	Total One-way Mileage
Proposed Action	---	---
Action Alternative 1	---	---
Action Alternative 2A	250,000	1,800,000
Action Alternative 2B	8,500,000	1,800,000
Action Alternative 2C	---	---
Action Alternative 3	300,000	3,200,000
No Action Alternative 1A	250,000	1,800,000
No Action Alternative 1B	8,500,000	1,800,000

Table E-27
Truck Mileages for the Consolidation of CH-TRU and RH-TRU Waste from Small Quantity Sites

	CH-TRU Waste	RH-TRU Waste
Alternatives	Total One-way Mileage	Total One-way Mileage
Proposed Action	17,000	380,000
Action Alternative 1	34,000	2,600,000
Action Alternative 2A	42,000	3,100,000
Action Alternative 2B	45,000	3,100,000
Action Alternative 2C	103,000	3,100,000
Action Alternative 3	42,000	2,570,000
No Action Alternative 1A	42,000	3,100,000
No Action Alternative 1B	45,000	3,100,000

Table E-28
Small Quantity Sites Transportation Impacts Relative to Major Generator-Storage Site Transportation Risk

Impact Category	Proposed Action	Action Alternative 1	Action Alternative 2A	Action Alternative 2B	Action Alternative 2C	Action Alternative 3	No Action Alternative 1A	No Action Alternative 1B	No Action Alternative 2
Nonradiological Risk									
Emisson-related LCFs	3.0E-3	0.02	0.02	0.02	0.02	0.02	0.02	0.02	3.0E-3
Vehicle-related Fatalities	0.03	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.03
Incident-free Radiological Risk									
Occupational	2.8	17	17	17	17	17	17	17	2.8
Nonoccupational	58	375	376	376	371	454	376	376	58
Occupational (LCFs)	1.1E-3	6.9E-3	6.9E-3	6.9E-3	6.7E-3	6.9E-3	6.9E-3	6.9E-3	1.1E-3
Nonoccupational (LCFs)	0.03	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.03

The impacts presented in this section were determined by adjusting the transportation impacts from truck shipments. This approach was feasible because many previous studies have performed detailed truck and detailed rail analyses (DOE 1994). These studies provided a firm basis for adjusting the rail impacts.

DOE would need to address several issues in conjunction with a decision to use rail transportation. These issues would include the following:

- The DOE trucking contractor has hired and trained drivers and would be responsible for procuring and maintaining vehicles and packagings for transporting the TRU waste. Over the last several years, all communities on highway transportation routes specified in SEIS-I were offered emergency response training. Similar development or planning has not been accomplished for rail.

- Regular rail is traditionally slower than highway; therefore, the NRC may need to allow a longer total shipping period than the 60-day truck limit. This could result in a reduction of the allowable TRUPACT-II loading to meet gas generation limits. Discussions between DOE, the NRC, and railroad industry personnel have not taken place.

- Discussions with railroad personnel regarding the use of regular train service or the safety requirements on dedicated trains have not been finalized. Thus, there is greater uncertainty surrounding rail transportation of TRU waste.

- Although projections of rail costs can be made, they are not based on a negotiated contract.

- Currently, there are no agreements in place for the tariffs for transporting TRU waste by rail from the major generator sites. This will require the coordination of a number of rail lines, depending upon the route traveled, and the coordination of the states and cities through which TRU waste would travel.

- In the event of an accident involving a derailment, a rail line could be disabled during the accident investigation with the possibility of no alternative routing for both WIPP and non-WIPP related rail shipments. It is recognized that highways are not immune to traffic delays, but the potential for a detour around an accident on a highway incident would be expected to be higher than that for rail. However, in the event of an unlikely accident resulting in a release of radioactive material, an exclusion area of typically 46 meters (150 feet) would need to be established for both truck or train transportation.

- The emergency response preparedness and training would need to be addressed in those areas possibly affected by rail transportation.

E.7.1 Fatal Accident Rates

Rail accident statistics typically include the number of rail car miles and the number of accidents and fatalities but not information concerning the number of accidents or fatalities per rail car. This omission makes apportioning accidents or fatalities to additional railcars containing TRU waste difficult. As with truck transportation, the probability of a rail car being involved in an accident is independent of the cargo being hauled. The average number of rail cars for a commercial train is 70; therefore, when a train of 70 cars is in an accident that results in one death, that fatality would be represented as 1/70th of a fatality for each rail car. If that train included three rail cars of TRU waste, the fatality would result in 3/70ths of a fatality for rail cars carrying TRU waste.

There are no statistics for the average number of cars in dedicated trains. It was assumed, therefore, that each dedicated train would have only three rail cars, one locomotive, and one caboose, and each rail car would be carrying TRU waste. Should a dedicated train be involved in an accident that results in a fatality, that fatality would be statistically apportioned as one-fifth a fatality for each car in the train. This number could be changed a great deal, though, by increasing the number of rail cars per train.

Differences in the impacts between regular and dedicated rail service would be primarily due to the differences in apportioning fatalities to rail cars. For SEIS-II, it was estimated that the fatality rate for dedicated rail would be fourteen times greater than that for regular rail. In actuality, if only three rail cars of TRU waste would be included in both cases, the same number of trains would be needed, and the same number of accidents and fatalities would be expected to occur.

E.7.2 Radiological Impacts of Accident-Free Rail Transportation

As under truck accident-free radiological impact analyses, the impacts to three groups of individuals were assessed: those exposed because they would live along a route, those exposed because they share the transportation corridor, and those exposed during rail stops (inspectors and those at rail stops). The impact to the exposed population would be proportional to the number of TRUPACT-IIs and RH-72Bs being shipped and the individual package TIs. Because the number of packages to be shipped and the TIs are the same for both truck and rail, these terms were considered to be constants in the impact analyses.

E.7.2.1 Radiation Exposure to Individuals Residing Close to the Transportation Corridor

Using RADTRAN, the overall exposure would be proportional to the TI, the number of packages shipped, the population density, and would be inversely proportional to the speed of the truck or train. The faster the TRU waste passes by a given location, the lower the exposure. The population density can be somewhat different for rail and truck transportation; however, it would be a second order effect, speed being the most important parameter.

For truck transportation analyses, an average highway speed of 55 miles per hour was assumed for rural population areas. For rail, a speed of 40 miles per hour through rural zones was assumed. As with truck transportation, 25 miles per hour through suburban zones and 15 miles per hour through urban zones was used. Assuming that the total miles would be approximately 25 percent greater for rail transportation and the fractions through rural, suburban, and urban zones would be 89, 10, and 1 percent respectively, a 1.5 increase in the impacts would occur under rail transportation. A more accurate estimate may be obtained by using the ratios of the total miles in each population zone, multiplied by the inverse ratio of the speeds, and adding these to get the impact increase for rail. The result, in most cases, would be close to 1.5.

E.7.2.2 Radiation Exposure to Individuals Sharing the Transportation Corridor

Trains do not typically encounter other traffic of the number and vicinity of that encountered on our nation’s highways. Therefore, it was conservatively assumed that the number of individuals sharing the rail transportation route corridor is at least two orders of magnitude less than for truck transportation.

E.7.2.3 Radiation Exposure During Stops

The biggest difference in impacts between truck and rail transportation would be due to exposures during stops. The model for rail analyses assumes that the people in the vicinity of the stopped train are uniformly distributed with a population density equivalent to the suburban density, 719 individuals per square kilometer (1,863 individuals per square mile). In addition, it was assumed that the other freight cars would effectively shield most of the individuals such that the effective exposure would be 1/10th of the exposure without shielding. The effective exposure distance was assumed to range from 10 to 400 meters (33 to 1,312 feet).

The stop duration was calculated for rail using two components, stops en route and stops at classification yards. For regular rail, the stop time en route is equal to 0.033d where "d" is the distance traveled. The time spent in classification yards was assumed to be 32N where "N" is the number of classification yards that must be used to interchange with other railroads. It was assumed that the average time in such a yard would be 32 hours. For dedicated rail, the stop time en route was assumed to equal 0.004d, and the time associated with interchange was 4N. Dedicated rail would, therefore, have exposures approximately eight times smaller than those from regular rail.

The exposure at stops for regular rail would be 1/8 of those for truck transportation. For dedicated rail, exposures would be 1/64 of those for truck.

The basic equation used to model the impact from rail stops is the following:

$$D = TI \times M \times 2\pi \times \ln\left(\frac{400}{10}\right) \times S \times Pop \times T \times \left[1 + N\left(\frac{970}{M}\right)\right] \times 10^{-6} \times 10^{-3} \qquad \text{(Equation E-4)}$$

where

D = Radiation dose in person-rem

TI = Transportation index in mrem per hour

M = Distance traveled in kilometers

S = Radiation shielding factor based on the congestion of rail cars in a rail yard, 0.1

T = Stop time (hours) per kilometer, 0.033

Pop = Population density, 719 persons per square kilometer (1,863 individuals per square mile)

N = Number of rail shipment transfers, a minimum of 2 (beginning and end of route)

10^{-6} = Conversion of kilometers to meters

10^{-3} = Conversion of rem to millirem

ln = natural log

Therefore, the dose would be:

Dose (mrem) = 1.7E-4 (TI) (M) (Equation E-5)

For dedicated rail shipments, the assumed stop duration was 0.004 hours per kilometer (0.006 per mile), a factor of 8.25 lower than for regular rail shipments. The time at interchanges was a factor of 8 lower for dedicated rail. Thus, dedicated rail service would have a stop time exposure about 8 times lower than for regular rail, and the estimated dose from rail stops would be 64 times lower than the estimated dose from the truck stops.

E.7.3 Impacts from Severe Rail Accidents

The aggregate radiological consequences of rail transportation accidents were estimated from the aggregate radiological impacts of truck transportation accidents. Since the frequency of rail accidents would be less than that for truck, the aggregate radiological impacts for rail transportation were conservatively assumed to be the same as those reported for truck in Table E-22.

Assuming that there would be three rail cars, with six TRUPACT-IIs per rail car (regular or dedicated), it was estimated that two TRUPACT-IIs could be breached in an impact accident. This bounding-case accident was assumed to be a derailment where two rail cars impact bridge abutments on either side of the tracks. Although other TRUPACT-IIs may experience impact forces, it would not be expected that the forces would be sufficient to breach the container. For the scenarios analyzed, no early fatalities or morbidities were estimated. The estimated population doses were dominated by the inhalation pathway (initial or resuspension). Based upon the failure of two TRUPACT-IIs or RH-72Bs, the impacts from the rail bounding-case accident scenarios for Action Alternatives 1 and 3 would be as follows:

- For the bounding radionuclide inventory in two breached TRUPACT-II's, the total population dose was estimated to be 63,600 person-rem, resulting in 32 LCFs in the exposed population. The estimated MEI dose would be 246 rem (TEDE), resulting in a 12 percent chance of a cancer fatality.
- For the bounding radionuclide inventory in two breached RH-72B's, the total population dose was estimated to be 65,000 person-rem, resulting in 32 LCFs in the exposure population. The estimated MEI dose would be 250 rem (TEDE), resulting in a 12 percent chance of a cancer fatality.
- For the average radionuclide inventory in two breached TRUPACT-II's, a total population dose of 12,740 person-rem was estimated. This would result in approximately 6 LCFs in the exposed population. The estimated MEI dose would be 160 rem (TEDE), resulting in a 8 percent chance of a cancer fatality.
- For the average radionuclide inventory in two breached RH-72B's, the total population dose would be 144 person-rem, resulting in an expectation of 0.08 LCF in the population. The estimated MEI dose would be 2.8 rem (TEDE), resulting in a 0.14 percent chance of a cancer fatality.

The impacts due to rail accidents under Action Alternative 2 would be the following:

- For the bounding radionuclide inventory in two breached TRUPACT-II's, the total population dose was estimated to be 178 person-rem, resulting in 0.09 LCFs in the exposed population. The estimated MEI dose would be 0.7 rem (TEDE), resulting in a 0.03 percent chance of a cancer fatality.

- For the bounding radionuclide inventory in two breached RH-72B's, the total population dose was estimated to be 182 person-rem, resulting in 0.09 LCFs in the exposure population. The estimated MEI dose would be 0.7 rem (TEDE), resulting in a 0.03 percent chance of a cancer fatality.

- For the average radionuclide inventory in two breached TRUPACT-II's, a total population dose of 35.6 person-rem was estimated. This would result in approximately 0.02 LCFs in the exposed population. The estimated MEI dose would be 0.5 rem (TEDE), resulting in a 0.02 percent chance of a cancer fatality.

- For the average radionuclide inventory in two breached RH-72B's, the total population dose would be 0.4 person-rem, resulting in an expectation of 2×10^{-4} LCFs in the population. The estimated MEI dose would be 0.008 rem (TEDE), resulting in a 4×10^{-4} percent chance of a cancer fatality.

There would be negligible impacts from hazardous chemical releases in a rail transportation severe accident for all alternatives. Because there would be half as many shipments under rail transportation, the aggregate occupational exposure to the train crew would be a factor of two lower. More importantly, the crew of the train would typically be hundreds of feet from the waste. Any exposures would be minimal, and therefore, were not considered. The only radiation exposures occur during train inspections. Very conservatively, the exposure from such activities would be less than 10 percent of the exposure to the crew doing a similar inspection for truck. The train crew would spend its time equally on 70 cars, whereas the truck crew would spend their entire time in close proximity to a radioactive shipping cask. Thus, it can be conservatively estimated that the exposure to the crew of the train would be 0.05 times the occupational exposure to the truck crew.

The impacts from rail (regular and dedicated) transportation are presented in Tables E-29 through E-32. Only the fatal accident projections would be notably different between truck and rail transportation.

E.8 RETRIEVAL AND RECOVERY

The WIPP Land Withdrawal Act states that if the EPA makes a determination of noncompliance during the disposal or decommissioning phase of WIPP, DOE may be required to retrieve (or recover), to the extent practicable, any TRU waste and contaminated material from the WIPP underground disposal facility.

Table E-29
Rail (Regular and Dedicated) Transportation Impacts for Action Alternative 1

Aggregate Traffic Related Fatalities	
Regular Rail	8
Dedicated Rail	112
Aggregate Radiological Accident Impacts (LCFs)	
CH-TRU Waste	0.7
RH-TRU Waste	0.08
Total CH-TRU and RH-TRU Waste	0.8
Aggregate Accident-Free Radiological Impacts (person-rem) [a]	
CH-TRU Waste	
Occupational	50 (0.02)
Nonoccupational	
Stops	665 (0.3)
Sharing Route	3 (1.6E-3)
Along Route	170 (0.09)
Nonoccupational Total [b]	840 (0.4)
RH-TRU Waste	
Occupational	35 (0.01)
Nonoccupational	
Stops	1.7E+3 (0.8)
Sharing Route	8 (3.9E-3)
Along Route	455 (0.2)
Nonoccupational Total [b]	2.1E+3 (1.1)
Total CH-TRU and RH-TRU Waste	
Occupational	85 (0.03)
Nonoccupational	
Stops	2.3E+3 (1.2)
Sharing Route	11 (5.5E-3)
Along Route	625 (0.3)
Nonoccupational Total [b]	3.0E+3 (1.5)

[a] Number in parentheses equals LCFs.
[b] Total nonoccupational doses include exposure to people at stops, sharing the route, and along the route. Totals may differ from the sums due to rounding.

Table E-30
Rail (Regular and Dedicated) Transportation Impacts for Action Alternative 2

	Action Alternative 2A	Action Alternative 2B	Action Alternative 2C
Aggregate Traffic Related Fatalities			
Regular Rail	5	6	6
Dedicated Rail	70	84	84
Aggregate Radiological Accident Impacts (LCFs)			
CH-TRU Waste	0.7	0.7	0.7
RH-TRU Waste	0.03	0.03	0.03
Total CH-TRU and RH-TRU Waste	0.7	0.7	0.7
Aggregate Accident-Free Radiological Impacts (person-rem) [a]			
CH-TRU Waste			
Occupational	55 (0.02)	65 (0.03)	50 (0.02)
Nonoccupational			
Stops	705 (0.4)	870 (0.4)	670 (0.3)
Sharing Route	3 (1.7E-3)	4 (2.0E-3)	3 (1.6E-3)
Along Route	185 (0.09)	230 (0.1)	175 (0.09)
Nonoccupational Total [b]	890 (0.5)	1.1E+3 (0.6)	845 (0.4)
RH-TRU Waste			
Occupational	15 (5.7E-3)	15 (5.7E-3)	15 (5.7E-3)
Nonoccupational			
Stops	675 (0.3)	675 (0.3)	675 (0.3)
Sharing Route	3 (1.6E-3)	3 (1.6E-3)	3 (1.6E-3)
Along Route	195 (0.1)	195 (0.1)	195 (0.1)
Nonoccupational Total [b]	875 (0.4)	875 (0.4)	875 (0.4)
Total CH-TRU and RH-TRU Waste			
Occupational	70 (0.03)	80 (0.03)	65 (0.03)
Nonoccupational			
Stops	1.4E+3 (0.7)	1.6E+3 (0.8)	1.3E+3 (0.7)
Sharing Route	7 (3.3E-3)	7 (3.7E-3)	6 (3.2E-3)
Along Route	380 (0.2)	425 (0.2)	370 (0.2)
Nonoccupational Total [b]	1.8E+3 (0.9)	2.0E+3 (1.0)	1.7E+3 (0.9)

[a] Number in parentheses equals LCFs.
[b] Total nonoccupational exposure is the sum of exposures to persons at stops, sharing the route, and along the route. Totals may differ due to rounding.

Table E-31
Rail (Regular and Dedicated) Transportation Impacts for Action Alternative 3

Aggregate Traffic Related Fatalities	
Commercial Rail	11
Dedicated Rail	154
Aggregate Radiological Accident Impacts (LCFs)	
CH-TRU Waste	1.1
RH-TRU Waste	0.1
Total CH-TRU and RH-TRU Waste	1.2
Aggregate Accident-Free Radiological Impacts (person—rem) [a]	
CH-TRU Waste	
Occupational	85 (0.03)
Nonoccupational	
Stops	1.1E+3 (0.6)
Sharing Route	5 (2.6E-3)
Along Route	285 (0.1)
Nonoccupational Total [b]	1.4E+3 (0.7)
RH-TRU Waste	
Occupational	45 (0.02)
Nonoccupational	
Stops	2.1E+3 (1.1)
Sharing Route	10 (4.9E-3)
Along Route	560 (0.3)
Nonoccupational Total [b]	2.7E+3 (1.3)
Total CH-TRU and RH-TRU Waste	
Occupational	130 (0.05)
Nonoccupational	
Stops	3.2E+3 (1.6)
Sharing Route	15 (7.5E-3)
Along Route	845 (0.4)
Nonoccupational Total	4.1E+3 (2.0)

[a] Number in parentheses equals LCFs.
[b] Total nonoccupational exposure is the sum of exposures to persons at stops, sharing the route, and along the route. Totals may differ due to rounding.

Table E-32
Rail (Regular and Dedicated) Transportation Impacts for No Action Alternative 1

	No Action Alternative 1A	No Action Alternative 1B
Aggregate Traffic Related Fatalities		
Regular Rail	0.2	0.5
Dedicated Rail	2.8	7.0
Aggregate Radiological Accident Impacts (LCFs)		
CH-TRU Waste	5.6E-3	0.02
RH-TRU Waste	1.2E-3	1.2E-3
Total CH-TRU and RH-TRU Waste	6.8E-3	0.02
Aggregate Accident-Free Radiological Impacts (person—rem) [a]		
Total CH-TRU and RH-TRU Waste		
Occupational	2 (8.6E-4)	9 (3.7E-3)
Nonoccupational		
Stops	90 (0.05)	190 (0.09)
Sharing Route	0.5 (2.3E-4)	0.9 (4.6E-4)
Along Route	45 (0.02)	70 (0.04)
Nonoccupational Total [b]	140 (0.07)	260 (0.1)

[a] Number in parentheses equals LCFs.
[b] Total nonoccupational exposure is the sum of exposures to persons at stops, sharing the route, and along the route. Totals may differ due to rounding.

E.8.1 Waste Retrieval

Retrieval consists of the removal of TRU waste from WIPP prior to the time at which the salt creep would begin to crush the waste drums and canisters. The retrieval volumes were assumed to be the same as the emplacement volume of one panel (17,560 cubic meters [620,125 cubic feet]). The waste would be shipped back to the originating generator-storage site. Transportation impacts were based on the number of shipments required to transport a designated volume of TRU waste to WIPP. With no additional waste to transport, the number of shipments required to transport the waste back would be the same as the number required to ship the waste to WIPP. Transportation impacts for retrieval would be identical to the transportation impacts associated with TRU waste emplacement. The impacts for transporting the TRU waste retrieved from one panel at WIPP are presented in Table E-33.

Table E-33
Waste Retrieval Impacts

Nonradiological Impacts	
Vehicle Emission-related Fatalities	0.02
Vehicle Related Fatalities	0.5
Accident-Free Radiological Impacts	
Occupational (person-rem)	79
Nonoccupational (person-rem)	590
Occupational (LCFs)	0.03
Nonoccupational (LCFs)	0.3

E.8.2 Waste Recovery

Recovery is defined as the removal of TRU waste, waste containers, and any material contaminated by such waste after the time at which the salt creep has crushed some of the containers. As a result, waste would escape into the panels resulting in an additional contamination volume. For the purposes of analyses, it was assumed that recovery would occur following the closure of WIPP and the active institutional control (100-year) period, at which time all containers were assumed to be breached. Analyses were based on the waste volumes of Action Alternative 3 because this alternative has the greatest volume of waste. The total volume of waste would include the amount of contaminated salt from the breached waste drums. This volume was assumed to be 3.4×10^6 cubic meters (1.2×10^8 cubic feet).

It was assumed that, upon removal from WIPP, the waste would be transported to the consolidation sites proposed in the WM PEIS Regionalized 1 Alternative. Each site would receive more waste than its original volume because of the additional volume of contaminated salt. This additional volume of waste would be divided among the sites, commensurate with their original volume. Impacts were based on adjusting the results of the Action Alternative 3 analysis to the waste volume estimates.

Due to the increase in volume, the number of shipments would increase by a factor of approximately 8.5. Due to the decay of the radionuclides with high gamma activity, the accident-free exposures will be lower than the exposures estimated for shipments to WIPP. Therefore, it was conservatively assumed that accident-free exposures for TRU waste recovery would be the same as for shipments to WIPP. Also, given that the half-lives of the radionuclides which contribute most to internal doses (alpha emitters) are relatively long compared to the recovery time frame, it was conservatively assumed that the maximum bounding accident exposures would be the same. Because the mode of transportation would be the same and the accident statistics would remain relatively constant, the number of accidents, injuries, fatalities, and LCFs from pollution effects would also increase by a factor of approximately 8.5. As in the original shipment of the waste, the transportation impact will be dominated by nonradiological impacts. Table E-34 presents a summary of the estimated transportation impacts associated with transporting TRU waste recovered from WIPP.

Table E-34
Waste Recovery Impacts

Nonradiological Impacts	
Vehicle Emission-related Fatalities	7
Vehicle Related Fatalities	185
Accident-Free Radiological Impacts	
Occupational (person-rem)	2.6E+3
Nonoccupational (person-rem)	2.9E+4
Occupational (LCFs)	1
Nonoccupational (LCFs)	15

E.9 COMPARISON OF ALTERNATIVES

This section presents a comparison of the impacts across the alternatives. As shown in Table E-35, the impacts from transportation are directly related to the number of shipments and the miles traveled.

The increase in transportation impacts under the Proposed Action as compared to the other alternatives is primarily based upon an increase in the waste inventory shipped to WIPP. The difference in impacts between the action alternatives is based upon the treatment process and the consolidation of TRU waste. All of the action alternatives begin with the same volume of waste to be emplaced at WIPP, except for the relatively small volume of PCB-commingled waste considered under Action Alternative 2. Action Alternative 1 considers the same TRU waste treatment as the Proposed Action, but would emplace nearly twice the volume. Action Alternative 2 considers thermal treatment of the TRU waste, which would result in an approximate volume reduction of sixty-five percent and thereby reduce the shipment numbers and impacts under Action Alternative 2A and 2B (as compared to Action Alternative 1). Action Alternative 2C has essentially the same transportation impacts as Action Alternative 1 because all of the waste (less the PCB-commingled waste) would be shipped to WIPP before thermal treatment occurs. The largest relative impacts occur under Action Alternative 3. This alternative starts with the same TRU waste volume as Action Alternative 1, but the waste would be treated using a shred and grout process which would result in an increase in waste volume of approximately 20 percent. However, due to the consolidation scheme used under Action Alternative 3, the difference in impacts between Action Alternative 1 and 3 is greater than twenty percent. Under Action Alternative 1, all of the waste is shipped directly to WIPP from the 10 major generator-storage sites. Under Action Alternative 3, the TRU waste is consolidated (CH-TRU waste at Hanford, LANL, RFETS, SRS, and INEEL/ANL-W; RH-TRU waste at Hanford and ORNL) and treated before it is shipped to WIPP.

The impacts under no Action Alternatives 1A and 1B would be the same as those under Action Alternatives 2A and 2B except for those impacts resulting from the shipments to WIPP. No Action Alternatives 1A and 1B use the same consolidation and treatment processes as Action Alternatives 2A and 2B. No Action 1A would have fewer impacts than 1B since there would be less waste consolidation and more shipments under No Action Alternative 1B. Under No Action Alternative 2, TRU waste would be left where it was generated; therefore, transportation was not analyzed.

Table E-35
Comparison of Impacts Across Alternatives (CH-TRU and RH-TRU Waste) [a]

Impacts	Proposed Action Scaled to WIPP Max	Action Alternative 1			Action Alternative 2A			Action Alternative 2B			Action Alternative 2C		
	Basic Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
Nonradiological Impacts													
Accidents	55	136	32	168	63	41	104	67	43	110	72	29	100
Injuries	39	94	23	116	43	28	71	45	29	74	50	19	69
Fatalities	5	13	3	16	6	4	10	6	4	10	7	3	10
Vehicle Pollution (LCF)	0.2	0.4	0.08	0.5	0.2	0.1	0.3	0.2	0.1	0.3	0.2	0.09	0.3
Radiological Impacts													
Accident-free Impacts													
Occupational (person-rem)	785 (0.3)	1.2E+3 (0.5)	465 (0.2)	1.7E+3 (0.7)	675 (0.3)	630 (0.3)	1.3E+3 (0.5)	750 (0.3)	670 (0.3)	1.4E+3 (0.6)	825 (0.3)	435 (0.2)	1.3E+3 (0.5)
Non-occupational (person-rem)	5.8E+3 (2.9)	1.7E+4 (8.5)	3.6E+3 (1.8)	2.1E+4 (10.3)	7.5E+3 (3.7)	4.1E+3 (2.0)	1.2E+4 (5.8)	7.9E+3 (4.0)	4.3E+3 (2.1)	1.2E+4 (6.1)	8.3E+3 (4.2)	2.9E+3 (1.5)	1.1E+4 (5.6)
Aggregate Accident Impacts													
Total (person-rem)	850 (0.4)	840 (0.4)	510 (0.3)	1.4E+3 (0.7)	570 (0.3)	220 (0.1)	790 (0.4)	670 (0.3)	230 (0.1)	900 (0.4)	760 (0.4)	510 (0.3)	1.3E+3 (0.7)
Number of Shipments	37,723	81,717	21,448	103,165	38,691	25,979	64,670	38,690	25,979	64,669	44,946	18,100	63,046
Total Number of Miles	1.05E+8	2.65E+8	6.38E+7	3.29E+8	1.22E+8	7.67E+7	1.99E+8	1.30E+8	8.08E+7	2.11E+8	1.39E+8	5.42E+7	1.93E+8

[a] Total Inventory numbers may differ due to rounding.

Note 1: Number in parenthesis equals LCFs (Latent Cancer Fatalities)
Note 2: Impacts for No Action Alternatives 1A and 1B reflect the number of shipments to consolidate TRU waste from generator-storage sites to treatment sites. No TRU waste would be shipped to WIPP.

Table E-35
Comparison of Impacts Across Alternatives (CH-TRU and RH-TRU Waste) — Continued [a]

Impacts	Action Alternative 3			No Action Alternative 1A			No Action Alternative 1B		
	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory	Basic Inventory	Additional Inventory	Total Inventory
Nonradiological Impacts									
Accidents	194	39	234	2	3	5	9	5	13
Injuries	133	28	160	2	2	4	8	4	12
Fatalities	18	4	22	0.2	0.2	0.3	0.7	0.3	1
Vehicle Pollution (LCF)	0.6	0.1	0.7	9.0E-3	0.02	0.03	0.04	0.03	0.07
Radiological Impacts									
Accident-free Impacts									
Occupational (person-rem)	2.0E+3 (0.8)	560 (0.2)	2.5E+3 (1.0)	25 (9.5E-3)	21 (7.7E-3)	46 (0.02)	135 (0.06)	50 (0.02)	190 (0.07)
Non-occupational (person-rem)	2.4E+4 (11.8)	4.4E+3 (2.2)	2.8E+4 (13.9)	425 (0.2)	418 (0.2)	843 (0.4)	1.1E+3 (0.6)	605 (0.3)	1.7E+3 (0.9)
Aggregate Accident Impacts									
Total (person-rem)	1.7E+3 (0.9)	610 (0.3)	2.3E+3 (1.2)	13 (6.3E-3)	1 (4.5E-4)	14 (6.8E-3)	32 (0.02)	3 (1.5E-3)	35 (0.02)
Number of Shipments	116,166	25,749	141,915	3,977	912	4,889	10,315	2,502	12,817
Total Number of Miles	3.76E+8	7.79E+7	4.54E+8	2.31E+6	6.84E+5	2.99E+6	8.17E+6	1.77E+6	9.94E+6

[a] Total Inventory numbers may differ due to rounding.

Note 1: Number in parenthesis equals LCFs (Latent Cancer Fatalities)

Note 2: Impacts for No Action Alternatives 1A and 1B reflect the number of shipments to consolidate TRU waste from generator-storage sites to treatment sites. No TRU waste would be shipped to WIPP.

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APPENDIX F
HUMAN HEALTH

This appendix describes the methods used to estimate human health impacts that may result from exposure to radioactive material and hazardous chemicals during routine storage operations at waste storage sites and during routine disposal operations at the Waste Isolation Pilot Plant (WIPP). During routine operations, members of the public and workers may be exposed to small quantities of hazardous volatile organic compounds (VOC) and radioactive gases released from vented waste containers. Routine releases of particulates are not expected because container vents have filters to prevent such a release. VOCs and gaseous radionuclides may be released to the environment from exhaust ventilation system emissions at the WIPP Waste Handling Building (WHB) and the underground area. At waste storage sites, there may be releases to the environment from the ventilation exhaust of the facilities; therefore, environmental releases would be the source of potential impacts for the population within 80 kilometers (50 miles) of the facilities, the maximally exposed individual (MEI), the noninvolved worker population (employees not directly involved in waste handling), and the maximally exposed noninvolved worker. Involved workers, those directly involved with waste storage and disposal operations, could also be exposed to direct radiation from radioactive material in the containers as well as the waste container emissions. Human health impacts were calculated for all of these potential receptors from storage and waste disposal activities. Waste treatment impacts from routine operations were based upon information presented in the *Final Waste Management Programmatic Environmental Impact Statement* (WM PEIS) (DOE 1997b). Information on the adjustment of the WM PEIS results is presented in Appendix B.

Based on new information and revised assumptions, these analyses update the impacts reported in the 1990 *Final Supplement Environmental Impact Statement for the Waste Isolation Pilot Plant* (SEIS-I) (DOE 1990). The total transuranic (TRU) waste inventory, individual waste container inventories, and different types of treatment for the waste are among the more notable updates.

F.1 TECHNICAL APPROACH

To estimate human health impacts from exposure to radioactive material and VOCs, spreadsheet calculations and three computer codes were used. Spreadsheets were used to estimate the radiological impacts from routine releases of radioactive gases. The GENII computer code was used to calculate the atmospheric dispersion factors for radioactive gas emissions at waste storage sites and WIPP for input into the spreadsheets (Napier et al. 1988). Version 1.95 of the ISO-PC computer code was used to estimate external radiation doses to involved workers (Rittman 1995). The MEPAS® code was used to estimate hazardous chemical carcinogenic and noncarcinogenic (toxicological) impacts to the public and noninvolved workers from releases of VOCs (Droppo et al. 1989; Strenge and Peterson 1989; Buck et al. 1995; Strenge and Chamberlain 1995; Droppo and Buck 1996). None of these three computer codes could be used to estimate the near-field impacts to involved workers from routine releases of radioactive gases and hazardous chemicals from TRU waste containers. Therefore, for impacts to involved workers from routine operations, computer spreadsheets were used.

F.1.1 Radiological Impacts

Computer spreadsheets were used to estimate the radiological impacts to the population, MEI, noninvolved workers, and involved workers from routine releases of radioactive gases. The radionuclide inventories detailed in Appendix A include the radioactive gases radon-222, radon-220, and carbon-14. These gases may slowly escape through the composite carbon filter, which is on each waste container. No routine release of particulate radioactive material would occur at storage sites or at WIPP during routine operations because of the filters. GENII was used to calculate the average annual atmospheric dispersion factors that were used to estimate far-field radiation doses to the population, MEI, and noninvolved workers from the release of radioactive gases. Atmospheric dispersion was calculated by GENII using a straight-line Gaussian-plume model with site-specific meteorologic data. Because the GENII atmospheric dispersion model is not valid for the near-field scenario, computer spreadsheets were used to estimate the near-field radiological impacts to involved workers from releases of radioactive gases from waste containers.

The major contributor to potential radiological impacts for involved workers is external radiation emitted from the radionuclides present in TRU waste. The ISO-PC code was used to estimate the average surface and 1-meter (3.3-foot) dose rates of the TRU waste drums. The surface and 1-meter (3.3-foot) dose rates of the contact-handled (CH) TRU waste drums were used to estimate the radiological impacts to involved workers from waste handling, storage, disposal, and management activities.

The ISO-PC code, which is based on the ISOSHLD-II code, calculates dose rates from radioactive sources using user-defined geometries. Container dose rates vary with radionuclide concentration and waste density. For a given radionuclide concentration, a more dense waste would have more self-shielding and, thereby, would have a lower external dose rate. Scattering properties also differ for different materials. The density of waste is closely tied to the level of waste treatment.

Average dose rates were calculated using the geometry and wall-thickness of the TRU waste containers, the alternative- and consolidation-site-specific average TRU waste density, and the site-specific average TRU waste radionuclide concentrations. The CH-TRU waste drum was modeled as being 82 centimeters (32 inches) high and 56 centimeters (22 inches) in diameter with an iron wall 0.16 centimeters (0.06 inches) thick. It was assumed, for all alternatives, that the radionuclides and waste were uniformly dispersed within the drum. The radionuclide inventories in Appendix A were used.

Radiological impacts were calculated as the total effective dose equivalent (TEDE), which is the sum of the external effective dose equivalent and the 50-year committed effective dose equivalent from radionuclides taken into the body. The TEDE was converted into the risk of a latent cancer fatality (LCF) using dose-to-risk factors published in *Recommendations of the International Commission on Radiological Protection*, ICRP Publication 60 (ICRP 60) (ICRP 1991), which presents risk estimates based on data from populations that received relatively high radiation doses at high dose rates. When the dose to an individual is below 20 rads per year, as is typically the case with doses resulting from radioactive gas releases, a reduction factor of two is used. The dose-to-risk conversion factors used for estimating cancer deaths from radiation exposures less than 20 rads per year are: (1) 500 LCFs per million person-rem effective dose equivalent or 5.0×10^{-4} LCFs per person-rem for the general population; and (2) 400 LCFs per million person-rem or 4.0×10^{-4} LCFs per person-rem for workers. The difference between the two conversion factors is attributable to an increased sensitivity to radiation among the very young and very old within the general population. Potential radiological impacts to populations (public, noninvolved worker, and involved worker) are reported as the expected number of LCFs that may occur. Radiological impacts to the MEI and the maximally exposed noninvolved worker are reported as the probability of an LCF occurring. No estimate of the impact to the maximally exposed involved worker was made because of the high degree of variability.

The ICRP 60 risk factors used in SEIS-II are approximately twice those values in SEIS-I, which used *The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980* (BEIR-III) risk factors that were current at the time (BEIR 1980). ICRP 60 risk factors are higher due to an improved reconstruction of the exposures on which the dose-to-risk factors were based and information obtained during an extended population follow-up time. By including the incidence of nonfatal cancers and severe hereditary effects from radiation exposure, a total detriment 1.4 to 1.5 times greater than the LCF risk estimate would result (ICRP 1991). These were not included in SEIS-II.

F.1.2 Hazardous Chemical Impacts

Hazardous chemical impacts may result from the routine release of VOCs from waste containers at waste storage sites and at WIPP. No routine releases of particulate hazardous chemicals (metals) would occur at storage sites or at WIPP because the waste containers are filtered. Atmospheric dispersion was estimated by MEPAS® using a straight-line Gaussian-plume model and site-specific meteorological data. MEPAS® cannot be used for the near-field exposure scenarios of involved workers, so computer spreadsheets were used to estimate potential hazardous chemical impacts.

Impacts from exposure to carcinogenic VOCs are presented as the risk of cancer incidence. The U.S. Environmental Protection Agency (EPA) has published slope factors that are based on chronic exposures to specific hazardous chemicals (EPA 1996). Slope factors are used to estimate the mathematical expectation that an individual will contract cancer in his/her lifetime from chronic exposure to a specific hazardous chemical. Estimates of carcinogenic risk to a population indicate the number of cancers expected within a population as a result of a chronic hazardous chemical exposure. Slope factors for the VOCs evaluated in SEIS-II are found in Table F-1.

Noncarcinogenic health effects from chronic exposure to routine releases of VOCs are presented as a Hazard Index (HI). Noncarcinogenic health effects are nonprobabilistic and have an occurrence threshold. The HI is equal to the individual's estimated exposure divided by the EPA chemical-specific "reference dose" (EPA 1996). The EPA reference dose is ideally based on the exposure level at which a deleterious effect is noted following chronic exposure over a year. Table F-2 presents the reference doses used in SEIS-II analyses and the effect upon which each is based.

Toxicological effects are not expected if the estimated HI is less than one. In some cases, initial conservative screening estimates of involved worker exposures resulted in HI values greater than one. In these cases, the estimated air concentrations to which the worker was exposed were compared to the Occupational Safety and Health Administration (OSHA) permissible exposure limit (PEL) for that VOC (NIOSH 1996) (see Table F-3). Unless otherwise noted, PELs are time-weighted averages that must not be exceeded during any 8-hour shift of a 40-hour workweek. Methylene chloride and chloroform have PEL ceiling values that must not be exceeded over any

Table F-1
Carcinogenic Risk Factors for VOCs

VOC	Slope Factor [risk per (milligram per kilogram per day)] [a]	Comment
Carbon tetrachloride	Inhalation = 0.053 Ingestion = 0.13	Probable human carcinogen.
Chlorobenzene	Inhalation = not available Ingestion = not available	Classification as a carcinogen is not possible due to poor database of carcinogenic effects upon which to base a determination.
Chloroform	Inhalation = 0.081 Ingestion = 6.1E-3	Probable human carcinogen.
Methyl Ethyl Ketone	Inhalation = 0.0 Ingestion = 0.0	Not carcinogenic.
Methylene Chloride	Inhalation = 1.6E-3 Ingestion = 7.5E-3	Probable human carcinogen.
1,1,2,2-Tetrachloroethane	Inhalation = 0.2 Ingestion = 0.2	Possible human carcinogen.
Toluene	Inhalation = not available Ingestion = not available	Classification as a carcinogen is not possible due to the poor database of carcinogenic effects upon which to base such a determination.
1,1-Dichloroethylene	Inhalation = 0.18 Ingestion = 0.6	Possible human carcinogen.
1,2-Dichloroethane	Inhalation = 0.091 Ingestion = 0.091	Probable human carcinogen.
Benzene	Inhalation = 0.029 Ingestion = 0.029	Human carcinogen (leukemia).
Xylene	Inhalation = 0 Ingestion = 0	Carcinogenicity is under review.
Tetrachloroethene	Inhalation = not available Ingestion = not available	Carcinogenicity is under review. SEIS-II assumes an inhalation slope factor of 1.8E-3 and an ingestion slope factor of 0.051, as cited in Strenge and Peterson (1989).
Ethyl Benzene	Inhalation = not available Ingestion = not available	Classification as a carcinogen is not possible due to the poor database of carcinogenic effects upon which to base such a determination.

[a] The inhalation slope factor from Integrated Risk Information System (IRIS) was converted to risk per milligram per kilogram per day by assuming an inhalation rate of 20 cubic meters (706 cubic feet) per day and an individual body mass of 70 kilograms (154 pounds) (EPA 1996).

period of time. Because a PEL does not exist for 1,1-dichloroethylene, air concentrations for this VOC were compared to the American Conference of Governmental Industrial Hygienists (ACGIH) short-term exposure limit (STEL) (NIOSH 1996). Overall, the general public is more sensitive to hazardous chemicals than are workers; therefore, an HI of greater than one for a worker signifies the need for further evaluation and perhaps mitigation rather than the expectation of the noncarcinogenic impact.

Because the noncarcinogenic health impacts are nonprobabilistic, the results of calculations for the MEI represent the bounding indicator of noncarcinogenic risks to the 80-kilometer (50-mile) population.

Table F-2
Noncarcinogenic Health Effect Measures for VOCs

VOC	Reference Dose (RfD) (milligrams per kilogram per day) [a]	Comment/Assumption
Carbon tetrachloride	Inhalation, not available Ingestion = 7E-4	Assumption: Inhalation RfD is equivalent to ingestion RfD. Medium confidence level for ingestion RfD, based on liver lesions.
Chlorobenzene	Inhalation, under review Ingestion = 0.02	Medium confidence level for ingestion RfD, based on histopathologic changes in the liver. Assumption: Inhalation RfD = 5.7E-3, from EPA HEAST values (Strenge and Petersen 1989).
Chloroform	Inhalation, under review Ingestion = 0.01	Assumption: Inhalation RfD is equivalent to ingestion RfD. Medium confidence level for ingestion RfD, based on fatty cyst formation in the liver.
Methyl ethyl ketone	Inhalation = 0.29 Ingestion = 0.61	Low confidence level for both inhalation and ingestion RfDs, based on decreased fetal birth weight.
Methylene chloride	Inhalation, under review Ingestion = 0.06	Assumption: Inhalation RfD is equivalent to ingestion RfD. Medium confidence level for ingestion RfD, based on liver toxicity.
1,1,2,2-tetrachloroethane	Inhalation, not available Ingestion, not available	No details available. Assumption: RfDs are zero.
Toluene	Inhalation = 0.11 Ingestion = 0.2	Medium confidence level for inhalation and ingestion RfDs. Inhalation RfD based on neurological effects. Ingestion RfD based on changes in liver and kidney weight.
1,1-dichloroethylene	Inhalation, under review Ingestion = 9E-3	Assumption: Inhalation RfD is equivalent to ingestion RfD. Medium confidence level for ingestion RfD, is based on the growth of hepatic lesions.
1,2-dichloroethane	Inhalation, not available Ingestion, not available	No details available. Assumption: RfDs are zero.
Benzene	Inhalation, not available Ingestion, not available	No details available. Assumption: RfDs are zero.
Xylene	Inhalation, under review Ingestion = 2.0	Assumption: Inhalation RfD = 0.086 based on previous EPA data (Strenge and Peterson 1989). Medium confidence level for ingestion RfD, based on hyperactivity, decreased body weight, and increased mortality (males only).
Tetrachloroethene	Inhalation, not available Ingestion = 0.01	Assumption: Inhalation RfD is equivalent to ingestion RfD. Medium confidence level for ingestion RfD, based on hepatotoxicity and weight gain.
Ethyl benzene	Inhalation = 0.29 Ingestion = 0.1	Low confidence levels for inhalation and ingestion RfDs. Inhalation RfD based on developmental toxicity. Ingestion RfD based on liver and kidney toxicity.

[a] Inhalation reference concentrations listed in IRIS are converted to RfDs by assuming an inhalation rate of 20 cubic meters per day and an individual body mass of 70 kilograms (EPA 1996).

Table F-3
Involved Worker Supplemental Noncarcinogenic Health Effect Measures for VOCs

VOC	Permissible Exposure Level (PEL) [a]
Carbon Tetrachloride	2 parts per million
Chlorobenzene	75 parts per million
Chloroform	Not to exceed 2 parts per million
1,1-Dichloroethylene	None [b]
1,2-Dichloroethane	50 parts per million
Methyl Ethyl Ketone	200 parts per million
Methylene Chloride	500 parts per million [c]
1,1,2,2-Tetrachloroethane	5 parts per million
Toluene	100 parts per million [d]

[a] Time-weighted concentration over an 8-hour exposure period unless otherwise noted.
[b] ACGIH recommends a 5 ppm over an 8-hour limit, with a short-term exposure limit (STEL) of 20 ppm.
[c] Maximum 5-minute concentration of 2000 ppm, not to be exceeded.
[d] Maximum 15-minute concentration of 150 ppm for any 2-hour period.

F.2 ANALYSIS METHODS AND ASSUMPTIONS

This section describes the background data used in the routine exposure impact analyses. This information is separated into three general categories: radionuclide and VOC source terms, atmospheric transport, and exposure scenarios. Radionuclides and VOCs would be routinely released only to the atmosphere. There is no surface water or groundwater near WIPP, no water is used in the disposal process, and no mechanisms exist for direct soil contamination. Atmospheric transport modeling was used to determine the locations where maximum exposures could occur. Individuals were assumed to be present at these locations, providing a bounding exposure estimate for any individual in the region. Exposure scenarios include assumptions for both individuals and populations for the primary exposure pathways of inhalation, dermal absorption, and external exposure and for the secondary exposure pathways of contaminated food and incidental soil ingestion. Radionuclide inventories detailed in Appendix A were used. Likewise, VOC source terms were developed as described in Appendix A. To conservatively estimate the impacts to the WHB worker, VOC inventories based on the WIPP planning-basis Waste Acceptance Criteria (WAC) (DOE 1996b) limits or headspace sampling data were used.

F.2.1 Source Terms

Impacts from routine releases of gaseous radionuclides and VOCs were estimated for the Total Inventory of TRU waste. The impacts were then assigned to either the Basic Inventory or the Additional Inventory based on the fractional volume of each. The radionuclide and VOC content of the Additional Inventory is unknown; therefore, the radionuclide and VOC headspace concentrations of the Basic Inventory were used for the Additional and Total Inventories.

F.2.1.1 Radionuclide Source Term

The impact estimates from routine operations evaluated in SEIS-I have been refined in SEIS-II. Detailed radionuclide inventory information (Appendix A) was taken from the *Transuranic Baseline Inventory Report* (BIR-3) (DOE 1996d), whereas SEIS-I inventories were limited to those pertaining to plutonium-239 equivalent curies (PE-Ci). Routine radiological impacts would result from the release of gaseous radionuclides through the composite filters of the waste containers and, for involved workers only, external radiation doses from waste handling operations. Releases of gaseous radionuclides are discussed below; additional information on radionuclides contributing to involved worker external dose is presented in Section F.2.3.3.

Radiological impacts from releases of gaseous radionuclides were estimated for the major consolidation sites. Radionuclides included in the analyses were radon-222 and carbon-14. While radon-220 is listed in some of the site radionuclide inventories, it has a half-life of only 56 seconds. Very little of the gas, if any, would escape the waste container before it radioactively decays to polonium-216, a radionuclide that would be trapped as a particulate within the waste or the container filter. Small quantities of other gaseous radionuclides may also be present in TRU waste, but they would be negligible contributors to either worker or population doses.

For carbon-14, the entire WIPP or consolidation site radionuclide inventory was assumed to be released as an organic carbon (e.g., methane). Organic carbon-14 forms have a dose factor two and three orders of magnitude greater than the dose factors of carbon-14 in the form of carbon dioxide or carbon monoxide, respectively (DOE 1988b). Therefore, by assuming this chemical form, the radiological impact estimate was maximized.

Radon-222 is continuously generated in the waste from the decay of radium-226, which has a half-life of 1,600 years. For every curie (Ci) of radium-226 in the waste inventory, 2.1×10^{-6} Ci of radon-222 are produced each second. Only radon-222 is released from the waste containers, however, because particulate radon progeny are trapped by the composite filters on each container. Therefore, the ICRP 65 dose factor (ICRP 1993) for radon-222 alone, 273.8 rem-cubic meter per Ci-hour, was used to determine the radiological impacts. For waste treated to meet WAC, all of the radon-222 that would be generated over a lifetime exposure period (70 years for the MEI, 35 years for workers and populations) in the waste inventory at WIPP or the consolidation sites was assumed to escape the waste container. For waste treated to meet the WIPP Land Disposal Restrictions (LDR), only a small fraction of the radon would be able to escape the monolithic mass before it decays. It was assumed, therefore, that 25 percent of the radon generated over the lifetime exposure period was released from the container. Waste treated by shred and grout would be more porous than LDR waste and less porous than WAC waste. In this case, 50 percent of the radon-222 generated over the lifetime exposure period was assumed to escape the waste container. The estimated release fractions account for the amount of gas captured within the waste and not readily escaping the container. This same methodology was applied to determine the radiological impacts to the public and maximally exposed noninvolved workers at Hanford Site (Hanford) and Oak Ridge National Laboratory (ORNL) for the excess remote-handled (RH) TRU waste remaining at the sites under the Proposed Action.

F.2.1.2 VOC Source Term

VOCs may be released routinely from the TRU waste containers through the composite filters because the purpose of the filter is to prevent gas build-up within the container. However, no hazardous metal particulates would be released through the filter. The release rate of the VOCs would be linearly related to the headspace concentration in the waste container. Routine exposures were assessed using average waste headspace concentrations because exposures may vary over short-term periods but will average out over the long-term periods evaluated in SEIS-II.

No quantitative information on the hazardous chemical inventory of TRU waste is available. Headspace sampling data on 31 VOCs in 900 CH-TRU waste containers are provided in the Resource Conservation and Recovery Act (RCRA) Part B Permit Application (DOE 1996a, Table C8-2). This list of VOCs incorporates sampling data from TRU waste containers at Idaho National Engineering and Environmental Laboratory (INEEL) and Rocky Flats Environmental Technology Site (RFETS), whereas only RFETS data was available for SEIS-I. Information on both the average and the maximum waste container inventories is available for a number of VOCs present in TRU waste.

Headspace concentrations of each VOC vary according to the waste matrix. No VOCs would be present in waste after treatment under Action Alternative 2 or No Action Alternative 1 because the thermal treatment destroys VOCs in the waste. Waste treated to meet WAC under the Proposed Action, Action Alternative 1, and the No Action Alternative 2 would contain a variety of waste matrices. A weighted average headspace concentration for waste at WIPP and each of the major waste consolidation sites was calculated for each alternative (see Appendix A). The matrix composition varies by consolidation site, so weighted average headspace concentrations for each consolidation site under Action Alternative 1 and No Action Alternative 2 were determined. The matrix composition under Action Alternative 3 would be a uniform solidified inorganic matrix, so the headspace concentrations of all Action Alternative 3 consolidation sites were assumed to be the same. Headspace concentrations of waste disposed of at WIPP are shown in Table F-4. The average and weighted average concentrations were used for routine release evaluations except for impacts to WIPP involved workers in the WHB, where higher concentrations were used.

Container VOC release rates were based on the diffusion rate of the VOC through the composite filter (DOE 1996c). Release rates are VOC-specific and are proportional to the headspace concentration. Filter diffusion rates and VOC release rates for a single CH-TRU drum are shown in Table F-5. Impacts for routine releases were determined using this information. The calculation of VOC impacts from excess RH-TRU waste remaining at Hanford and ORNL under the Proposed Action used the same methodology. The CH-TRU waste drum release rate of three drums were assumed to be equivalent to the release from an RH-TRU canister.

Table F-4
TRU Waste VOC Headspace Concentrations Used in Health Impact Analyses for WIPP

VOC (milligrams per cubic meter per ppmv)	Planning-Basis WAC Headspace Limit (ppmv) [a]	Treated to Meet Planning-Basis WAC (ppmv)		Treated by Shred and Grout (ppmv)	
		Weighted Average	Maximum [b]	Weighted Average	Maximum [c]
Carbon tetrachloride (6.39)	7,510	184.5	---	316.5	---
Chloroform (4.96)	6,325	13.7	---	1.2	---
1,1-dichloroethylene (4.03)	500 [d]	8.4	---	2.5	---
1,2-dichloroethane (4.11)	9,100	5.6	---	1.1	---
Methylene chloride (3.53)	368,500	662.1	---	8.1	---
Chlorobenzene (4.68)	No Limit	8.4	956.4 4,368	1.3	260
Methyl ethyl ketone (3.0)	No Limit	40.4	2,946 39,311	6.8	130
1,1,2,2-tetrachloroethane (7.0)	No Limit	335.7	537.6 4,368	125.1	270
Toluene (3.83)	No Limit	5.7	764.8 6,992	1.3	320
Benzene (3.25)	500 [d]	7.6	---	22.4	---
Ethyl benzene (4.41)	500 [d]	8.5	---	31.6	---
Tetrachloroethene (6.89)	No Limit	7.1	229.7 2184	21.6	600
Xylene (4.41)	500 [d]	25.6	---	113.9	---

[a] DOE 1996b.
[b] Used for those VOCs without a maximum planning-basis WAC limit. Top values are weighted maximums determined by the maximum headspace concentration of each matrix and were used in routine WHB worker bounding scenarios. Bottom values are maximum concentrations found in any solidified organic sample and were used for the bounding accident scenarios.
[c] Maximum concentration found in any solidified inorganic matrix for those VOCs without a planning-basis WAC limit.
[d] Flammable VOC limit.

Table F-5
Average Single Drum Release Rate of Headspace VOCs (grams per second)

VOC	Filter Diffusion Rates (mol/s/molfrac) [a]	Action Alternative 3	Proposed Action, Action Alternative 1, and No Action Alternative 2						
		All sites	Hanford	INEEL	LANL	RFETS	ORNL	SRS	WIPP Under-ground
Carbon tetrachloride	1.21E-6	5.9E-8	1.1E-8	6.9E-8	5.1E-8	5.1E-8	1.7E-8	2.3E-8	3.4E-8
Chlorobenzene	1.16E-6	1.7E-10	1.3E-9	1.2E-9	8.1E-10	3.5E-10	1.2E-9	1.1E-9	1.1E-9
Chloroform	1.34E-6	1.8E-10	1.9E-9	2.8E-9	2.2E-9	1.4E-9	3.0E-9	2.6E-9	2.2E-9
Methyl ethyl ketone	1.30E-6	6.4E-10	3.8E-9	5.1E-9	2.0E-9	1.8E-9	5.8E-9	5.0E-9	3.8E-9
Methylene chloride	1.48E-6	1.0E-9	1.2E-7	6.0E-8	9.5E-8	1.2E-8	1.8E-8	2.0E-8	8.3E-8
1,1,2,2- tetrachloroethane	1.21E-6	2.5E-8	6.3E-8	9.2E-8	2.6E-8	2.9E-8	1.4E-7	1.3E-7	6.8E-8
Toluene	1.20E-6	1.4E-10	6.7E-10	7.8E-10	4.2E-10	2.5E-10	8.4E-10	7.4E-10	6.3E-10
1,1-dichloroethene	1.40E-6	3.3E-10	1.2E-9	1.4E-9	6.1E-10	4.4E-10	1.9E-9	1.7E-9	1.1E-9
1,2-dichloroethane	1.40E-6	1.5E-10	8.3E-10	9.6E-10	5.0E-10	3.0E-10	1.1E-9	9.2E-10	7.8E-10
Benzene	1.32E-6	2.3E-9	6.0E-10	8.7E-10	1.3E-9	8.5E-10	3.0E-10	6.2E-10	7.9E-10
Ethyl benzene	1.10E-6	3.7E-9	6.8E-10	1.1E-9	1.9E-9	1.1E-9	1.9E-10	7.3E-10	9.9E-10
Tetrachloroethene	1.13E-6	4.1E-9	1.0E-9	1.5E-9	2.3E-9	1.5E-9	3.9E-10	9.6E-10	1.3E-9
Xylene (mixed)	1.11E-6	1.3E-8	1.7E-9	3.6E-9	6.3E-9	3.7E-9	4.0E-10	2.4E-9	3.0E-9

[a] Filter diffusion rates for most constituents are from Table D9-3 (DOE 1996a). For methyl ethyl ketone, benzene, ethyl benzene, tetrachloroethene, and xylene, the values were calculated from Equation D9-1 and D9-2 (DOE 1996a) assuming the respective $P_{c,\,VOC}$: 41.52, 48.34, 35.53, 44.57, 36.81; respective $T_{c,\,VOC}$: 536.8, 562.2, 617.2, 620.2, 616.2; and respective molecular weights: 72.1, 78.1, 106.2, 165.8, 106.2.

Routine hazardous chemical impacts at WIPP would be dependent on the release of VOCs from TRU waste containers in the WHB and the underground area for the Proposed Action, Action Alternative 1, and Action Alternative 3. Containers in the WHB were assumed to have VOC headspace concentrations at the planning-basis WAC limit. For VOCs without planning-basis WAC concentration limits, the weighted maximum headspace concentration was used. These concentrations were determined by weighting the *maximum* matrix-specific headspace concentrations rather than the *average* matrix-specific headspace concentrations (see Appendix A).

The resulting impact estimates bound any impacts for workers in the WHB. Containers in the underground area were assumed to have the average weighted VOC headspace concentrations shown in Table F-4. No RH-TRU waste VOC headspace concentration data are available, so concentrations were assumed to be identical to those in CH-TRU waste.

A limited number of CH-TRU drums (42) and RH-TRU containers (3) can be present in the WHB at any one time. In contrast, the WIPP underground disposal area would have a much larger number of containers at any one time. A maximum of approximately 80,000 CH-TRU drum-equivalents and approximately 1,200 RH-TRU canisters could be placed in a panel. The number of CH-TRU and RH-TRU drum-equivalents are shown in Table F-6. Total VOC releases from the underground area were calculated using the average single drum release rates in Table F-5 and multiplying them by the number of VOC releasing drum-equivalents in Table F-6.

The number of drum-equivalents of CH-TRU waste in the underground area were calculated by dividing the total volume of waste to be emplaced by 0.208 cubic meters (7 cubic feet), which is the volume of a single drum. The number of drum-equivalents of RH-TRU waste in the underground area were calculated by dividing the total RH-TRU waste volume by the waste volume of an RH-TRU canister, 0.89 cubic meters (31 cubic feet) and then multiplying by three, the number of individual drums in an RH-TRU canister. Releases from RH-TRU canisters were modeled as from the three drums contained in a canister.

Table F-6
Number of Drum-Equivalents Assumed for the Evaluation of Routine VOC Releases

	Proposed Action		Action Alternative 1		Action Alternative 3		No Action Alternative 2	
Site	CH-TRU Waste	RH-TRU Waste	CH-TRU Waste	RH-TRU Waste	CH-TRU Waste	RH-TRU Waste	CH-TRU Waste	RH-TRU Waste
WIPP [a]	80,814	3,620	80,815	3,621	81,170	4,345	N/A	N/A
Hanford-East [b]	N/A	N/A	288,683	73,208	699,731	191,548	138,199	49,593
Hanford-West [b]	N/A	N/A	288,603	73,208	699,731	191,548	138,199	49,593
INEEL	N/A	N/A	417,280	11,538	503,101	0	135,314	744
LANL	N/A	N/A	169,320	1,668	201,524	0	101,109	770
RFETS	N/A	N/A	82,695	0	89,869	0	52,221	0
ORNL	N/A	N/A	10,143	26,852	0	32,220	8,027	1,036
SRS	N/A	N/A	81,433	0	112,929	0	58,003	0

[a] Drum-equivalents in a single panel.
[b] Hanford's waste volume was equally distributed between the Hanford-East and Hanford-West locations.

N/A = Not Applicable

A full panel of drum-equivalent containers was assumed to continuously emit VOCs, which would be released by the underground ventilation system throughout each year of operations. Approximately 80,000 drum-equivalents were assumed for the CH-TRU source term. More likely, assuming that approximately two years would be required to fill a typical panel, there would be an average of 20,000 CH-TRU drums emitting VOCs over the first year and an average of 60,000 drums emitting VOCs over the second year. Release from RH-TRU waste does not consider any reduction from the canister containment or from use of the RH-TRU shielding plug.

Conservative VOC release estimates were also made for the waste consolidation sites. The VOCs were assumed to be released at the same rate over a lifetime exposure period, which does not consider the off-site transport of the waste from any site to WIPP for disposal.

F.2.2 Atmospheric Transport

The GENII and MEPAS® codes were used to estimate the atmospheric dispersion and transport of radionuclides and hazardous chemicals, respectively. Atmospheric transport was modeled using site-specific meteorologic data. Routine release impact analyses used annual average meteorologic data for the MEI, the maximally exposed noninvolved worker, and the noninvolved worker population and sector-averaged air concentrations (i.e., averaged across sixteen 22.5-degree sectors) for the population within 80 kilometers (50 miles) of WIPP or the storage facility.

The meteorologic data used in MEPAS® and GENII are joint frequencies of wind speed, wind direction, and atmospheric stability class. Average, multi-year data are most appropriate to use when modeling future releases because the site-specific probability of occurrence of dispersion values for any particular location in the region is recognized. Meteorologic data sets used for the major consolidation sites were the same as those used in the WM PEIS (DOE 1997b). The meteorologic joint frequency data file used for WIPP is shown in Table F-7. Meteorologic data have been recorded at the WIPP site for several years, but the data recording system has captured the data only about half of the time. These data do not adequately characterize the site-specific conditions; therefore, more complete data from the airport in Carlsbad, New Mexico, were used (NOAA 1995).

Table F-7
Meteorologic Joint Frequency Data Used
for WIPP Atmospheric Releases (Percent of Time)

Average Wind Speed (m/s)	Atmospheric Stability Class	Direction (Wind Toward)															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
1.49	A	0.03	0.03	0.03	0.04	0.04	0.03	0.05	0.03	0.04	0.01	0.02	0.01	0.01	0.01	0.01	0.01
	B	0.12	0.09	0.09	0.12	0.15	0.10	0.13	0.16	0.14	0.04	0.02	0.03	0.05	0.03	0.03	0.05
	C	0.07	0.05	0.04	0.05	0.05	0.05	0.05	0.06	0.05	0.02	0.02	0.02	0.04	0.05	0.05	0.03
	D	0.07	0.04	0.04	0.05	0.08	0.05	0.06	0.08	0.07	0.02	0.04	0.03	0.03	0.04	0.04	0.04
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0.35	0.20	0.14	0.14	0.23	0.17	0.28	0.49	0.53	0.32	0.23	0.33	0.47	0.44	0.33	0.24
2.63	A	0.07	0.07	0.07	0.06	0.12	0.09	0.09	0.10	0.07	0.03	0.03	0.02	0.02	0.02	0.02	0.03
	B	0.25	0.25	0.24	0.30	0.44	0.37	0.55	0.57	0.40	0.06	0.08	0.07	0.06	0.04	0.08	0.13
	C	0.34	0.28	0.26	0.27	0.33	0.30	0.41	0.45	0.40	0.09	0.06	0.06	0.12	0.09	0.11	0.18
	D	0.55	0.33	0.31	0.27	0.42	0.36	0.51	0.54	0.55	0.17	0.13	0.15	0.22	0.25	0.26	0.31
	E	0.39	0.19	0.17	0.17	0.28	0.38	0.55	0.76	0.93	0.40	0.35	0.42	0.55	0.47	0.39	0.25
	F	0.94	0.46	0.26	0.23	0.46	0.50	0.79	1.56	2.03	0.97	0.69	1.01	1.51	1.39	0.95	0.69
4.27	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0.20	0.11	0.11	0.10	0.19	0.25	0.39	0.41	0.35	0.06	0.02	0.03	0.05	0.02	0.03	0.08
	C	0.72	0.31	0.17	0.17	0.42	0.49	1.08	1.25	0.98	0.12	0.11	0.14	0.20	0.11	0.17	0.31
	D	1.17	0.42	0.39	0.39	0.57	0.74	1.46	1.88	1.68	0.43	0.39	0.53	0.64	0.41	0.48	0.79
	E	0.55	0.16	0.12	0.15	0.21	0.33	0.87	1.37	2.22	0.98	0.63	1.07	1.52	1.14	0.56	0.42
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6.64	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0.15	0.05	0.02	0.02	0.09	0.17	0.40	0.31	0.29	0.04	0.04	0.10	0.20	0.05	0.05	0.10
	D	1.48	0.43	0.30	0.26	0.48	0.64	1.98	1.76	1.30	0.39	0.40	1.32	2.32	0.77	0.61	1.29
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
9.53	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0.02	0	0	0	0.01	0.01	0.02	0.02	0.02	0.01	0.02	0.04	0.15	0.01	0	0.02
	D	0.29	0.07	0.02	0.02	0.04	0.07	0.23	0.11	0.09	0.10	0.12	0.51	1.29	0.23	0.14	0.35
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
12.88	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0.02	0.05	0.01	0	0
	D	0.07	0.01	0.01	0	0	0.01	0.03	0.02	0.02	0.02	0.02	0.21	0.67	0.08	0.03	0.05
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Source: NOAA 1995. Carlsbad, New Mexico, airport meteorological data (1990-1994)

Meteorologic data were used with effluent release information to determine the dispersion characteristics of WIPP releases. Releases from WIPP to the environment could occur from the stacks at either the WHB or the Exhaust Filter Building, the latter handling the underground ventilation. For each stack release, a stack height and diameter, release velocity, and effluent temperature (Table F-8) was characterized in MEPAS® and GENII. The dispersion estimates used for individual and population impacts were slightly more conservative (i.e., less dispersion of the release) using the Exhaust Filter Building release parameters. Therefore, the Exhaust Filter Building release parameters were used to conservatively model both the WHB and the underground releases at WIPP. Effluent stack releases were assumed to have a 10-meter (33-foot) effective release height at all treatment/consolidation sites.

Routine releases of hazardous chemicals would occur at the consolidation sites during the lag storage period. Therefore, impacts were estimated for the major consolidation sites under all action alternatives. The locations of the MEI and noninvolved worker were determined based on dispersion estimates. Lag storage facilities under the action alternatives were assumed to be at the same locations as the No Action Alternative 2 storage facilities. The locations of the MEI and the maximally exposed noninvolved worker at the major consolidation sites are shown in Table F-9.

Table F-8
WIPP Stack Release Parameters

Parameter [a]	Waste Handling Building	Exhaust Filter Building
Stack height (meters)	14.9	8.2
Stack diameter (meters)	2.4	4.4
Release velocity (cubic meters per second)	170	200
Effluent temperature (degrees Fahrenheit) (DOE 1995b)	61	61
Number of stacks	1	2

[a] DOE 1990, unless otherwise indicated.

Table F-9
MEI and Noninvolved Worker Locations for Lag Storage Operations

Site	Noninvolved Worker Location	MEI Location
Hanford 200-East	100 meters south	16,000 meters east
Hanford 200-West	100 meters south	24,000 meters east
INEEL	100 meters south	12,000 meters south-southeast
LANL	100 meters west	2,400 meters east-southeast
RFETS	300 meters north	5,000 meters north
ORNL	100 meters southwest	4,000 meters southwest
SRS	100 meters west-southwest	12,000 meters west-southwest

F.2.3 Exposure Scenarios

Exposure scenarios define a set of conditions and assumptions under which populations and/or individuals may be impacted by radioactive material or hazardous chemicals. Exposures scenarios that may result from the routine release of radionuclides and VOCs at the storage sites and at WIPP are described below.

Human health impacts from routine, chronic exposures to VOCs and gaseous radionuclides were evaluated for the public, noninvolved workers, and involved workers. Inhalation would be the primary exposure pathway for such releases; however, impacts to involved workers from external radiation during waste handling were also evaluated. Thermal treatment of waste to meet LDRs under Action Alternative 2 and No Action Alternative 1 would destroy the VOCs in the waste, so releases of hazardous chemicals would not occur during routine operations for these alternatives. It was assumed that members of the public were exposed continuously (8,766 hours per year) while noninvolved workers were exposed for 2,000 hours per year.

Human health impacts were evaluated over a lifetime of exposure for members of the public, workers, and the MEI. The MEI was assumed to be exposed at the same location for the full 35-year operational period of the Proposed Action and No Action Alternative 2 and for a 70-year lifetime for Action Alternatives 1, 2, and 3. The average exposure period for the population was assumed to be 35 years for all alternatives to account for population turnover. Maximum working lifetimes for involved and noninvolved workers were assumed to be 35 years for all alternatives.

F.2.3.1 Storage Site Exposure Scenarios

Storage site impact estimates were based on impacts at the six major storage sites: Hanford, INEEL, Los Alamos National Laboratory (LANL), ORNL, RFETS, and Savannah River Site (SRS), where at least 98 percent of the waste would be stored under all alternatives. To estimate human health impacts to the population, MEI, noninvolved worker population, and the maximally exposed noninvolved worker from gaseous radionuclide and VOC emissions from storage facilities at consolidation sites, it was conservatively assumed that all drum-equivalents of treated waste were in storage beginning the first year of operations. It was assumed that the total number of drum-equivalents would emit VOCs over the lifetime of the exposed receptors. The population data for the consolidation sites were the same as those used for the WM PEIS (DOE 1997b), which used data from the 1990 U.S. Census to estimate the population in 80-kilometer (50-mile) areas around the major storage sites. It was assumed that the populations surrounding all sites would remain at the 1990 levels. The noninvolved worker population numbers for the storage sites were the same as those presented in the WM PEIS (DOE 1997b).

Radiological impacts to the involved worker would primarily result from external radiation; exposure to gaseous radionuclides would be a negligible contributor. Specific information on estimating involved worker external dose impacts is presented in Section F.2.3.3 below.

An estimate of the impacts to the involved worker from VOC exposure during routine operations was made by assuming that each VOC would be present at its highest allowable concentration. This concentration was based on the EPA limits of 1×10^{-6} annual probability of cancer incidence for a Class B hazardous chemicals and 1×10^{-5} annual probability of cancer incidence for a Class C hazardous chemicals. Workers at each site were assumed to be exposed for a 35-year occupational lifetime.

F.2.3.2 WIPP Disposal Exposure Scenarios

Impacts to the public and noninvolved workers were evaluated for routine releases of gaseous radionuclides and VOCs from TRU waste at WIPP. For the exposure scenarios, it was assumed that all waste was contained in drums, using “drum-equivalents” to recognize that some waste may be in standard waste boxes (SWB). The releases from the underground area were estimated on a panel basis. Only one panel would be fully ventilated and unsealed at a time. It was assumed to require 2.5 years to fill a single panel with a total of approximately 81,000 drum-equivalents of CH-TRU waste and approximately 4,000 drum-equivalents of RH-TRU waste (DOE 1997a). Impacts were conservatively estimated by assuming that a full panel would continuously emit gaseous radionuclides and VOCs each year of operations. This assumption approximately doubles the actual emission estimate of VOCs from CH-TRU waste drums. A maximum number of drums (42 CH-TRU and 3 RH-TRU drum-equivalents) were assumed to be in the WHB at all times. As noted above, there would be no release of VOCs under Action Alternative 2 or No Action Alternative 1.

1990 U.S. Census data were used to determine the population inhabiting the 80-kilometer (50-mile) region surrounding WIPP. Approximately one-third of this population is located at Carlsbad, 42 kilometers (26 miles) west of WIPP, and one-third of the population is located at Hobbs, 72 kilometers (45 miles) east-northeast of WIPP (Table F-10). Population distributions would change unpredictably over the operations time frames evaluated in SEIS-II. For the purposes of impact analyses, it was assumed that the population surrounding WIPP would remain at 1990 levels.

Table F-10
1990 Population Distribution Within 80 Kilometers (50 Miles) of WIPP

Direction	Distance (miles)										Total
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50	
S	0	0	0	0	0	0	3	2	46	20	**71**
SSW	9	0	0	0	0	0	0	2	43	8	**62**
SW	0	0	0	0	0	0	37	57	0	5	**99**
WSW	0	0	0	0	0	0	1,622	191	57	62	**1,932**
W	0	0	0	0	0	0	138	25,291	197	3	**25,629**
WNW	0	0	0	0	0	0	38	5,765	242	63	**6,108**
NW	0	0	0	0	0	0	6	7	14	12,401	**12,428**
NNW	0	0	0	0	0	9	4	66	104	56	**239**
N	0	0	0	0	0	0	3	0	63	12	**78**
NNE	0	0	0	0	0	0	4	3	122	7,353	**7,482**
NE	0	0	0	0	0	0	0	11	37	9,115	**9,163**
ENE	0	0	0	0	0	0	0	10	282	30,877	**31,169**
E	0	0	0	0	0	0	6	5	2,982	19	**3,012**
ESE	0	0	0	0	0	0	0	16	2,173	97	**2,286**
SE	0	0	0	0	0	0	0	0	15	20	**35**
SSE	0	0	0	0	0	1	3	14	5	73	**96**
Total	**9**	**0**	**0**	**0**	**0**	**10**	**1,864**	**31,440**	**6,382**	**60,184**	**99,889**

Source: DOE 1995b

The location of the MEI was set at the point on the WIPP site boundary where an individual could establish a residence, with the least amount of atmospheric dispersion. This location was determined to be 3,000 meters (9,840 feet) north of the Exhaust Filter Building.

Similarly, the location of the maximally exposed noninvolved worker at WIPP was determined by identifying the on-site location (i.e., within the Exclusive Use Area) with the minimum amount of atmospheric dispersion. This location was 200 meters (660 feet) east of the Exhaust Filter Building, outside of the property protection zone and north of the railroad spur. It is unlikely that an individual would be located at this site, but the impacts here would bound any on-site noninvolved worker impacts. For impacts to the noninvolved worker population at WIPP, it was conservatively assumed that all 1,095 WIPP workers would be exposed at the same level as the maximally exposed noninvolved worker.

Involved workers at WIPP were evaluated according to two different waste-handling groups, those in the WHB and those in the underground area. This approach was necessary to estimate VOC impacts because of the considerable differences in the number of containers emitting VOCs to the ventilation air.

CH-TRU waste is handled in the WHB in a large, open area. The WHB exhaust operates at 43 cubic meters (1,505 cubic feet) per second (DOE 1990). WHB air exchange rates range from 1.5 to 12 per hour. Assuming that an air exchange rate of 12 per hour applies to the 43-cubic-meters (1,505-cubic-feet)-per-second flow rate, an air exchange rate of 1.5 would correlate to a flow rate of 5.4 cubic meters (190 cubic feet) per hour. The minimum flow rate, 5.4 cubic meters (190 cubic feet) per second, was conservatively used to estimate impacts to the WHB worker. The underground flow rate was assumed to be 100 cubic meters (3,500 cubic feet) per second, which is the flow rate of only one of the two exhaust fans that drive the underground ventilation. The external radiation dose received by involved workers is considerably greater than the internal radiation dose from gaseous releases of radionuclides. Therefore, only the external doses were calculated.

F.2.3.3 External Radiation Dose of Involved Workers

The primary source of radiological impacts to involved workers at storage sites and at WIPP would be from external radiation exposures. During routine operations, only involved workers would be exposed to external radiation emitted from the waste. The results presented in the *Waste Isolation Pilot Plant Safety Analysis Report* (DOE 1995b) indicate that the expected annual external dose to RH-TRU waste workers would be approximately 20 percent of that to CH-TRU waste workers during the time when both CH-TRU and RH-TRU waste is being disposed of. Based on this analysis, only the dose to CH-TRU workers was evaluated in detail. To determine the comparative radiological impact to workers for each alternative, screening calculations were performed to determine which radionuclides would be the primary contributors to external radiation dose. Alternative-specific variables included the volume of the Basic Inventory and Additional Inventory considered for storage or disposal and the waste density. WIPP worker radiological impact calculations were performed by using the radionuclide inventories of Appendix A. Dose rate reduction over time due to radioactive decay was not considered in screening.

To identify the primary radionuclides contributing to external dose, the radionuclide-specific air immersion dose-rate factor (DOE 1988a) was multiplied by the radionuclide inventory for each site to determine an *external dose screening value*. These values do not have meaning as an expression of absolute impact, but do indicate the comparative impact from external dose for each radionuclide in the TRU waste. For example, if radionuclide A has a higher screening value than radionuclide B, an individual would receive a greater external dose from radionuclide A, all exposure conditions being equal.

External dose screening values were calculated for the entire CH-TRU waste radionuclide inventory to be sent to WIPP and for each consolidation site. The radionuclides that contributed at least 90 percent of the site's total screening value for the Proposed Action and each alternative are tabulated in Tables F-11 through F-15. The percentage of the site's total screening values contributed by the listed radionuclides, the Department of Energy (DOE or the Department) (1988a) radionuclide-specific external dose factor, and the total screening value of all the radionuclides in each site's consolidated waste are shown in the tables.

Eight radionuclides were found to contribute the majority of the worker external dose for the sites evaluated. Americium-241 and barium-137m (the short-lived progeny of cesium-137) were found to be the largest individual contributors. Under most of the alternatives, waste consolidated at INEEL and RFETS contains the majority of the americium-241, and waste consolidated at Hanford contains the majority of the barium-137m.

Table F-11
Proposed Action External Dose Screening Values for CH-TRU Waste [a, b]

Radionuclide	Ext DF [c]	Hanford	INEEL	LANL	LLNL	Mound	NTS	ORNL	ANL-E	RFETS	SRS	All Sites
Cobalt-60	1.3E+4	---	9.7E+5	---	---	---	---	---	---	---	---	---
Barium-137m	3.1E+3	1.1E+7	---	3.2E+5	---	---	---	---	---	---	3.4E+4	1.2E+7
Europium-152	5.9E+3	---	---	---	---	---	7.5E+3	---	---	---	---	---
Europium-154	6.5E+3	---	---	---	---	---	3.4E+3	---	---	---	---	---
Bismuth-214	8.1E+3	---	---	---	---	---	2.4E+3	8.0E+4	---	---	---	---
Neptunium-239	840	---	---	---	---	---	---	1.9E+4	1.7E+3	---	---	---
Plutonium-238	0.44	---	---	---	---	280	1.7E+4	---	---	---	3.8E+5	---
Americium-241	95	2.5E+6	1.0E+7	2.5E+6	8.4E+4	---	3.3E+4	2.6E+5	1.2E+4	2.2E+7	3.0E+5	3.8E+7
Total		1.4E+7	1.2E+7	3.0E+6	8.5E+4	290	6.8E+4	4.0E+5	1.4E+4	2.2E+7	7.6E+5	5.2E+7
Percent Total Site Value		98	94	93	98	99	93	90	94	99	92	95

[a] Dashed lines indicate a negligible contributor to the external dose from a site's average waste.
[b] Screening values are unitless and are useful for comparative purposes.
[c] External dose conversion factor.

Table F-12
Action Alternative 1 External Dose Screening Values for CH-TRU Waste [a, b]

Radionuclide	Ext DF [c]	Hanford	INEEL	LANL	LLNL	Mound	NTS	ORNL	ANL-E	RFETS	SRS	All Sites
Cobalt-60	1.3E+4	---	2.5E+6	---	---	---	---	---	---	---	---	---
Barium-137m	3.1E+3	1.9E+7	---	4.4E+5	---	---	---	1.7E+4	---	---	4.0E+4	2.0E+7
Europium-152	5.9E+3	---	---	---	---	---	6.4E+3	---	---	---	---	---
Europium-154	6.5E+3	---	---	---	---	---	2.8E+3	---	---	---	---	---
Bismuth-214	8.1E+3	---	---	---	---	---	2.1E+3	7.0E+4	---	---	---	---
Neptunium-239	840	---	---	---	---	---	---	1.7E+4	870	---	---	---
Plutonium-238	0.44	---	---	---	---	240	1.4E+4	---	---	---	4.5E+5	---
Americium-241	95	4.4E+6	2.6E+7	3.5E+6	7.1E+4	---	2.8E+4	2.6E+5	6.3E+3	2.0E+7	3.5E+5	5.5E+7
Total		2.4E+7	3.0E+7	4.3E+6	7.2E+4	240	5.7E+4	4.0E+5	1.2E+4	2.0E+7	9.1E+5	8.0E+7
Percent Total Site Value		98	95	93	98	99	93	91	94	99	92	94

[a] Dashed lines indicate a negligible contributor to the external dose from a site's average waste.
[b] Scoping values are unitless and are useful for comparative purposes.
[c] External dose conversion factor.

Table F-13
Action Alternative 2A, Action Alternative 3, and No Action Alternative 1A External Dose Screening Values for CH-TRU Waste [a, b, c]

Radionuclide	Ext DF [d]	Hanford	INEEL	LANL	RFETS	SRS	All Sites
Cobalt-60	1.3E+4	---	2.5E+6	---	---	---	---
Barium-137m	3.1E+3	1.9E+7	---	4.4E+5	---	5.7E+4	2.0E+7
Bismuth-214	8.1E+3	---	---	---	---	7.0E+4	---
Plutonium-238	0.44	---	---	---	---	4.5E+5	---
Americium-241	95	4.5E+6	2.6E+7	3.5E+6	2.0E+7	6.2E+5	5.5E+7
Total		2.4E+7	3.0E+7	4.3E+6	2.0E+7	1.3E+6	8.0E+7
Percent Total Site Value		98	95	93	99	91	94

[a] Dashed lines indicate a negligible contributor to the external dose from a sites average waste.
[b] Screening values are unitless and are useful for comparative purposes.
[c] Action Alternative 2C screening values are equivalent to the "All Sites" values.
[d] External dose conversion factor.

Table F-14
Action Alternative 2B and No Action Alternative 1B External Dose Screening Values for CH-TRU Waste [a, b]

Radionuclide	Ext DF [c]	Hanford	INEEL	SRS	All Sites
Bismuth-214	8.1E+3	---	---	7.0E+4	---
Barium-137m	3.1E+3	1.9E+7	---	5.7E+4	2.0E+7
Americium-241	95	4.5E+6	5.0E+7	6.2E+5	5.5E+7
Plutonium-238	0.44	---	---	4.5E+5	---
Total		2.4E+7	5.5E+7	1.3E+6	8.0E+7
Percent Total Site Value		98	91	91	94

[a] Dashed lines indicate a negligible contributor to the external dose from a site's average waste.
[b] Screening values are unitless and are useful for comparative purposes.
[c] External dose conversion factor.

Table F-15
No Action Alternative 2 External Dose Screening Values for CH-TRU Waste [a, b]

Radionuclide	Ext DF [c]	Hanford	INEEL	LANL	LLNL	Mound	NTS	ORNL	ANL-E	RFETS	SRS	All Sites
Cobalt-60	1.3E+4	---	8.2E+5	---	---	---	---	---	---	---	---	---
Barium-137m	3.1E+3	9.3E+6	---	2.7E+5	---	---	---	1.0E+4	---	---	4.3E+4	9.8E+6
Europium-152	5.9E+3	---	---	---	---	---	6.4E+3	---	---	---	---	---
Europium-154	6.5E+3	---	---	---	---	---	2.8E+3	---	---	---	---	---
Bismuth-214	8.1E+3	---	---	---	---	---	2.1E+3	6.8E+4	---	---	---	---
Neptunium-239	840	---	---	---	---	---	---	1.6E+4	1.4E+3	---	---	---
Plutonium-238	0.44	---	---	---	---	---	1.4E+4	---	---	---	4.9E+5	---
Americium-241	95	2.1E+6	8.7E+6	2.1E+6	7.1E+4	240	2.8E+4	2.2E+5	9.8E+3	2.0E+7	3.8E+5	3.4E+7
Total		1.2E+7	1.0E+7	2.6E+6	7.2E+4	240	5.7E+4	3.5E+5	1.2E+4	2.0E+7	9.8E+5	4.6E+7
Percent Total Site Value		98	94	93	98	99	93	92	94	99	92	95

[a] Dashed lines indicate a negligible contributor to the external dose from a site's average waste.
[b] Screening values are unitless and are useful for comparative purposes.
[c] External dose conversion factor.

External dose estimates were made for involved workers using the radionuclides indicated by the screening calculations. Although americium-241 and barium-137m would be responsible for the majority of the potential external dose, cobalt-60 would be an important potential contributor for INEEL-consolidated waste; bismuth-214 and neptunium-239 would be important for ORNL-consolidated waste; and plutonium-238 and bismuth-214 would be notable for SRS-consolidated waste. Therefore, external dose impacts were calculated by including americium-241, barium-137m, cobalt-60, bismuth-214, neptunium-239, and plutonium-238.

Involved workers may receive radiation doses either during WIPP disposal operations or during storage operations at consolidation sites. During the years of maximum impact (for the action alternatives), when both CH-TRU and RH-TRU waste would be disposed of, a greater impact would be expected for CH-TRU waste handling than for RH-TRU waste handling because RH-TRU waste would be managed remotely, and the exposure conditions would be limited by distance, shielding, and additional administrative controls contained in the *WIPP Radiological Control Manual* (DOE 1995a). Although the RH-TRU waste container external dose rates would be higher than those for CH-TRU waste, worker exposures from the routine disposal of RH-TRU waste would be expected to be less because of remote handling operations and the smaller volume of waste. Even so, there are no differences in the maximum annual dose limit for CH-TRU and RH-TRU workers.

The external radiation dose received from handling CH-TRU waste containers would be a function of the radionuclides present, the quantity of each radionuclide, and the waste density. The quantity of radionuclides in a container may be reduced as a result of packaging in order to meet WAC or TRUPACT-II acceptance criteria prior to being sent to WIPP. The packaging of individual waste containers would be done to meet dose-rate, weight, thermal power, or PE-Ci limitations (see Table F-16). These limits result from the design criteria for WIPP operations and TRUPACT-II acceptance (DOE 1996b). Thermal power and weight limits were taken into account when estimates of shipment volumes and emplacement volumes were determined.

Table F-16
Specific Planning-Basis WAC Requirements Limiting Waste Container Radionuclide Contents

Administrative Control	Limit	Comment
Container external dose rate	CH-TRU waste surface: 200 mrem/hour RH-TRU waste surface: no more than 5 percent of the RH-TRU waste canisters are allowed dose rates of greater than 100 rem/hour	Dose rate 2 meters from the TRUPACT-II must be less than 10 mrem/hour
Container PE-Ci content	80 PE-Ci for CH-TRU waste drum 130 PE-Ci for CH-TRU waste Standard Waste Box 1,800 PE-Ci for solidified or vitrified waste or overpacked drum 1,000 PE-Ci for RH-TRU waste canister	TRUPACT-II transport container limits restrict the fissile gram equivalents (FGEs) to 325 FGEs per TRUPACT-II, with a maximum of 200 FGEs/drum. These restrictions further limit the PE-Ci content of the container.
Container thermal power limit	Thermal power limit for the contents of a TRUPACT-II is 40 watts.	- - -

The external radiation dose to a worker can also be reduced through radiation protection practices such as radiation shielding within the drum or placing the highest external-dose-rate container in a seven-drum bundle in the interior bundle location. Waste manifests require record of the maximum package dose rate, so external doses can be knowledgeably limited by administrative controls. When practiced, these procedures would lead to small annual reductions in external exposures that could accumulate to substantial reductions over time. For the purposes of SEIS-II impact analyses, however, these practices were not considered when estimating worker doses.

The WIPP annual occupational dose limit is currently 1.0 rem per year TEDE (DOE 1995a). Doses to extremities (i.e., hands) are limited to 50 rem per year. In addition, the *WIPP Radiological Control Manual* (DOE 1995a) contains the WIPP as low as reasonably achievable (ALARA) radiation protection policy, which is an administrative check of worker exposures.

Surface and 1-meter (3.3-foot) dose rates for average CH-TRU waste drums are presented in Table F-17 for the major storage sites and for WIPP. The 1995 activity of the waste was used for dose rate estimates, and drum surface dose rates were calculated according to the following equation:

Table F-17
Average Surface and 1-Meter (3.3-Foot) Dose Rates of CH-TRU Waste Drums (1995 activities)

Alternative	Waste Treatment	Site	Waste Package Density (grams/cubic meter)	Average Surface Dose Rate (mrem/hour)	Average 1-meter Dose Rate (mrem/hour)
Proposed Action	WAC	WIPP	0.582	40	2.9
Action Alternative 1	WAC	Lag-Hanford	0.592	34	2
		Lag-INEEL	0.598	31	3
		Lag-LANL	0.720	9	0.7
		Lag-RFETS	0.153	153	13
		Lag-ORNL	0.259	33	3
		Lag-SRS	0.476	4	0.3
		WIPP	0.582	32	2.3
Action Alternative 2A & No Action Alternative 1A	LDR	Lag-Hanford	1.966	42	2
		Lag-INEEL	1.966	13	1
		Lag-LANL	1.966	6	0.5
		Lag-RFETS	1.966	57	5
		Lag-SRS	1.966	3	0.3
		WIPP	1.966	25	1.6
Action Alternative 2B & No Action Alternative 1B	LDR	Lag-Hanford	1.966	42	2
		Lag-INEEL	1.966	15	1
		Lag-SRS	1.966	3	0.3
		WIPP	1.966	25	1.6
Action Alternative 2C	LDR	WIPP	1.966	25	1.6
Action Alternative 3	Shred and grout	Lag-Hanford	1.375	17	0.9
		Lag-INEEL	1.281	8	0.6
		Lag-LANL	1.224	3	0.2
		Lag-RFETS	0.631	36	3
		Lag-SRS	1.058	2	0.1
		WIPP	1.256	12	0.7
No Action Alternative 2	WAC	Hanford	0.592	34	2
		INEEL	0.621	30	3
		LANL	0.730	9	0.8
		RFETS	0.369	200 [a]	21
		ORNL	0.259	33	2
		SRS	0.476	6	0.4

[a] Calculated surface dose rate was 242 mrem per hour for RFETS. Individual waste containers exceeding limits would be repackaged or internally shielded to meet the planning-basis WAC limit of 200 mrem per hour surface dose rate for CH-TRU waste. This packaging would subsequently reduce the 1-meter (3.3-foot) dose rate.

$$S_{icd} = D_{id} \times C_{ic} \qquad \text{(Equation F-1)}$$

where

S_{icd} = surface dose rate from a drum with radionuclide "i" for waste at consolidation site "c" with a waste density "d" in units of rem per hour

D_{id} = surface dose rate of a drum with 1 Ci of radionuclide "i" and a waste density "d" in units of rem per hour per Ci per drum (see Table F-18)

C_{ic} = activity of radionuclide "i" in Ci for an average drum at consolidation site "c" (see Table F-19)

The total surface dose rate of consolidation site waste was calculated by summing the S_{icd} values for the radionuclides of concern. The 1-meter (3.3-foot) dose rates were calculated in a similar manner, using the 1-meter (3.3-foot) dose rate value (see Table F-18) for the D_{id} variable.

Table F-18
Radionuclide-Specific Dose Rates for Various Waste Densities (rem/h per Ci/drum-equivalent)

Density [a] (g/cm^3)	Am-241		Ba-137m		Co-60		Pu-238		Bi-214		Np-239	
	surface	1-meter	surface	1-meter	surface	1-meter	surface	1-meter	surface	1-meter	surface	1-meter
0.260-W	7.90E-2	6.56E-3	3.994	2.03E-1	12.48	6.33E-1	1.35E-4	8.5E-6	7.832	0.4	1.17	6.22E-2
0.370-W	5.97E-2	5.09E-3	3.566	1.85E-11	11.35	5.85E-1	1.1E-4	7.0E-6	N/A	N/A	N/A	N/A
0.480-W	4.75E-2	4.12E-3	3.154	1.66E-1	10.25	5.35E-1	9.08E-5	5.86E-6	N/A	N/A	N/A	N/A
0.580-W	3.98E-2	3.5E-3	2.822	1.5E-1	9.337	4.92E-1	7.79E-5	5.06E-6	N/A	N/A	N/A	N/A
0.595-W	3.89E-2	3.42E-3	2.776	1.48E-1	9.208	4.86E-1	7.62E-5	4.96E-6	N/A	N/A	N/A	N/A
0.625-W 0.625-G	3.71E-2 1.53E-2	3.27E-3 1.39E-3	2.687 2.992	1.44E-1 1.57E-1	8.956 10.61	4.74E-1 5.50E-1	7.31E-5 6.58E-5	4.76E-6 3.90E-6	N/A	N/A	N/A	N/A
0.725-W	3.21E-2	2.86E-3	2.42	1.31E-1	8.179	4.36E-1	6.41E-5	4.20E-6	N/A	N/A	N/A	N/A
1.060-G	9.11E-3	8.36E-4	2.193	1.19E-1	8.211	4.37E-1	4.26E-5	2.59E-6	3.891	0.211	N/A	N/A
1.250-G	7.73E-3	7.11E-4	1.938	1.06E-1	7.38	3.96E-1	3.667E-5	2.24E-6	N/A	N/A	N/A	N/A
1.370-G	7.05E-3	6.49E-4	1.801	9.86E-2	6.918	3.72E-1	3.36E-5	2.06E-6	N/A	N/A	N/A	N/A
1.966-L	4.9E-3	4.5E-4	1.32	7.31E-2	5.195	2.84E-1	2.38E-5	1.47E-6	2.260	0.125	N/A	N/A

[a] W = WAC waste, G = grouted waste, L = waste treated to meet LDRs.

N/A = Not Applicable

Table F-19
Average Density and Activity of Site-Specific CH-TRU Wastes

Waste form / site	Density (g/cm^3)	Radionuclide concentration (average Ci/drum-equivalent)											
		Am-241		Ba137m		Co-60		Pu-238		Bi-214		Np-239	
WAC		**AA1**	**NA2**	**AA1**	**NA2**	**AA1**	**NA2**	**AA1**	**NA2**	**AA1**	**NA2**	**AA1**	**NA2**
Hanford	0.592	8.02E-2	8.00E-2	1.09E-2	1.09E-2	0	0	1.36E+0	1.36E+0	N/A	N/A	N/A	N/A
INEEL	0.598	6.54E-1	6.66E-1	5.13E-4	4.22E-4	4.53E-4	4.61E-4	4.60E-1	4.42E-1	N/A	N/A	N/A	N/A
LANL	0.720	2.18E-1	2.20E-1	8.50E-4	8.56E-4	9.51E-8	1.49E-10	2.14E+0	2.16E+0	N/A	N/A	N/A	N/A
RFETS	0.369	2.56E+0	4.07E+0	0	0	0	0	1.38E-1	2.18E-1	N/A	N/A	N/A	N/A
ORNL	0.259	2.73E-1	2.54E-1	5.46E-4	1.98E-4	3.15E-5	2.89E-10	9.75E-1	5.50E-1	9.57E-4	1.02E-3	2.20E-3	2.34E-3
SRS	0.476	4.92E-2	6.91E-2	1.74E-4	2.45E-4	8.73E-6	1.23E-5	1.35E+1	1.90E+1	N/A	N/A	N/A	N/A
WIPP	0.582	4.27E-1	N/A	4.93E-3	N/A	1.41E-4	N/A	1.79E+0	N/A	N/A	N/A	N/A	N/A
LDR		**AA2A**	**AA2B**	**AA2A**	**AA2B**	**AA2A**	**AA2B**	**AA2A**	**AA2B**	**AA2A**	**AA2B**	**AA2A**	**AA2B**
Hanford	1.966	2.30E-1	2.30E-1	3.10E-2	3.10E-2	0	0	3.86E+0	3.86E+0	N/A	N/A	N/A	N/A
INEEL	1.966	1.38E+0	1.90E+0	1.08E-3	1.31E-3	9.56E-4	6.89E-4	1.13E+0	2.17E+0	N/A	N/A	N/A	N/A
LANL	1.966	6.28E-1	N/A	2.45E-3	N/A	2.74E-7	N/A	6.16E+0	N/A	N/A	2.60E-4	N/A	N/A
RFETS	1.966	1.16E+1	N/A	0	N/A	0	N/A	6.24E-1	N/A	N/A	2.60E-4	N/A	N/A
SRS	1.966	2.41E-1	2.41E-1	7.26E-4	7.26E-4	3.80E-5	3.80E-5	4.28E+1	4.28E+1	N/A	N/A	N/A	N/A
WIPP	1.966	1.13E+0	1.13E+0	1.30E-2	1.30E-2	3.72E-4	3.72E-4	4.71E+0	4.71E+0	N/A	N/A	N/A	N/A
Grout		**AA3**		**AA3**		**AA3**		**AA3**		**AA3**		**AA3**	
Hanford	1.375	6.72E-2		9.03E-3		0		1.13E+0		N/A		N/A	
INEEL	1.281	5.43E-1		4.25E-4		3.76E-4		4.45E-1		N/A		N/A	
LANL	1.224	1.83E-1		7.15E-4		7.99E-8		1.80E+0		N/A		N/A	
RFETS	0.631	2.36E+0		0		0		1.27E-1		N/A		N/A	
SRS	1.058	7.02E-2		2.12E-4		1.11E-5		1.25E+1		7.58E-5		N/A	
WIPP	1.256	3.58E-1		4.15E-3		1.18E-4		1.50E+0		N/A		N/A	

N/A = Not Applicable

WIPP Involved Workers

Individual exposures to WIPP involved workers were calculated for each alternative based on an annual 400-hour exposure time at 1 meter (3.3 feet) from the CH-TRU waste. A total of 36 involved workers would be involved in waste handling operations annually. Aggregate worker population doses were calculated to account for the total CH-TRU operations period required to dispose of all CH-TRU waste under each alternative.

Annual Exposures

Each involved worker could be exposed to waste for a maximum of 400 hours annually (2 hours per day, 4 days per week, 50 weeks per year), unless this exposure time would result in an annual dose of more than the occupational dose limit of 1 rem per year. The annual doses to involved workers based on a 400-hour annual exposure time are indicated in Table F-20 for each action alternative. Annual doses are decay-corrected each year. The values are conservative because the calculated CH-TRU dose rates were not decay-corrected when the 1995 radionuclide inventories were used to determine the indicated doses for 1998.

Annual doses for WIPP involved workers exposed for 400 hours were calculated using Equation F-2.

$$D_A = (\Sigma\, d_{A,r}) \times 400/1{,}000 \qquad \text{(Equation F-2)}$$

where

D_A = worker annual occupational external dose from CH-TRU operations for alternative "A", in rem

$d_{A,r}$ = decay-corrected average 1-meter (3.3-foot) dose rate from CH-TRU waste for radionuclide "r" under alternative "A", in millirem per hour summed over all radionuclides

400 = exposure time at 1 meter (3.3 feet) from the CH-TRU waste drums, in hours per year

1,000 = millirem per rem

The individual worker doses under the Proposed Action in Table F-20 are the calculated doses based on a 400-hour annual exposure time. For the first 20 years of disposal operations, the doses are expected to be greater than 1 rem per year, which is above the administrative limit and would not be permitted. All Proposed Action workers would not be permitted to work the maximum exposure time of 400 hours annually; they would be permitted to work until exposures were the maximum 1 rem per year. WIPP involved worker exposure times under the Proposed Action would start out at 340 hours annually for the first year of disposal operations, then would increase over the first 20 years of disposal operations as radiological decay reduced the average 1-meter (3.3-foot) dose rate. In other words, individuals would work 356 hours per year through the fifth year of disposal operations, 368 hours per year through the tenth year, 384 hours per year through the fifteenth year, and 396 hours per year through the twentieth year of disposal operations. Given the expectation that 2.5 years would be required to fill a single panel with CH-TRU waste at a disposal efficiency of 75 percent (DOE 1997a) and that 10 panels and 35 years of disposal

Table F-20
WIPP Involved Worker Annual Doses (rem/y per worker)

Calendar Year	Years of Disposal Operations	WIPP Worker Annual Dose (rem/year per worker)				Aggregate h/y per worker [b]	Aggregate WIPP Involved Worker Population Dose (person-rem/y)				
		Proposed Action [a]	Action Alternative 1	Action Alternative 2	Action Alternative 3		Proposed Action [c]	Action Alternative 1	Action Alternative 2	Action Alternative 3	Dose Limit (1rem/y per worker)
1998	1	**1.17**	0.92	0.63	0.30	400	**36**	33	23	11	36
1999	2	**1.16**	0.91	0.61	0.29	800	**72**	66	45	21	72
2000	3	**1.15**	0.90	0.60	0.29	1200	**108**	99	66	31	108
2001	4	**1.14**	0.89	0.59	0.28	1600	**144**	131	87	42	144
2002	5	**1.13**	0.88	0.58	0.27	2000	**180**	163	108	51	180
2003	6	**1.12**	0.87	0.56	0.27	2400	**216**	194	129	61	216
2004	7	**1.11**	0.87	0.55	0.26	2800	**252**	225	149	71	252
2005	8	**1.10**	0.86	0.54	0.26	3200	**288**	256	168	80	288
2006	9	**1.09**	0.85	0.53	0.25	3600	**324**	287	187	89	324
2007	10	**1.08**	0.84	0.52	0.25	4000	**360**	317	206	98	360
2008	11	**1.08**	0.83	0.52	0.25	4400	**396**	347	225	107	396
2009	12	**1.07**	0.83	0.51	0.24	4800	**432**	377	243	116	432
2010	13	**1.06**	0.82	0.50	0.24	5200	**468**	406	261	124	468
2011	14	**1.05**	0.81	0.49	0.23	5600	**504**	436	279	133	504
2012	15	**1.04**	0.81	0.48	0.23	6000	**540**	465	296	141	540
2013	16	**1.04**	0.80	0.48	0.23	6400	**576**	493	313	149	576
2014	17	**1.03**	0.79	0.47	0.22	6800	**612**	522	330	157	612
2015	18	**1.02**	0.79	0.46	0.22	7200	**648**	550	347	165	648
2016	19	**1.02**	0.78	0.45	0.22	7600	**684**	579	363	173	684
2017	20	**1.01**	0.78	0.45	0.22	8000	**720**	606	379	181	720
2018	21	1.00	0.77	0.44	0.21	8400	756	634	395	189	756
2019	22	1.00	0.77	0.44	0.21	8800	792	662	411	196	792
2020	23	0.99	0.76	0.43	0.21	9200	828	689	426	204	828
2021	24	0.99	0.75	0.42	0.20	9600	863	716	441	211	864
2022	25	0.98	0.75	0.42	0.20	10000	898	743	457	218	900
2023	26	0.97	0.74	0.41	0.20	10400	934	770	471	225	936
2024	27	0.97	0.74	0.41	0.20	10800	968	797	486	232	972
2025	28	0.96	0.74	0.40	0.19	11200	1003	823	501	239	1008
2026	29	0.96	0.73	0.40	0.19	11600	1038	849	515	246	1044
2027	30	0.95	0.73	0.39	0.19	12000	1072	876	529	253	1080
2028	31	0.95	0.72	0.39	0.19	12400	1106	902	543	260	1116
2029	32	0.94	0.72	0.38	0.18	12800	1140	927	557	266	1152
2030	33	0.94	0.71	0.38	0.18	13200	1174	953	570	273	1188
2031	34	0.93	0.71	0.37	0.18	13600	1207	979	584	279	1224

[a] Bolded doses indicated are those calculated. Exposure during actual operations would be limited to 1 rem/y.

[b] Aggregate hrs per year assuming 400 h/y per worker for all alternatives except the Proposed Action. See text for Proposed Action Annual exposure times.

[c] Bolded values reflect an administratively limited 1 rem/y dose per worker.

Table F-20
WIPP Involved Worker Annual Doses (rem/y per worker) - Continued

		WIPP Worker Annual Dose (rem/year per worker)					Aggregate for WIPP Involved Worker Population Dose (person-rem/y)				
Calendar Year	Years of Disposal Operations	Proposed Action [a]	Action Alternative 1	Action Alternative 2	Action Alternative 3	Aggregate h/y per worker [b]	Proposed Action [c]	Action Alternative 1	Action Alternative 2	Action Alternative 3	Dose Limit (1rem/y per worker)
2032	35	0.93	0.70	0.37	0.18	14000	1240	1004	597	286	1260
2033	36	0.92	0.70	0.36	0.18	14400		1029	610	292	1296
2034	37	0.92	0.70	0.36	0.17	14800		1054	623	298	1332
2035	38	0.91	0.69	0.36	0.17	15200		1079	636	305	1368
2036	39	0.91	0.69	0.35	0.17	15600		1104	648	311	1404
2037	40	0.90	0.68	0.35	0.17	16000		1129	661	317	1440
2038	41	0.90	0.68	0.34	0.17	16400		1153	673	323	1476
2039	42	0.90	0.68	0.34	0.17	16800		1177	686	329	1512
2040	43	0.89	0.67	0.34	0.16	17200		1202	698	335	1548
2041	44	0.89	0.67	0.33	0.16	17600		1226	710	340	1584
2042	45	0.88	0.67	0.33	0.16	18000		1250	722	346	1620
2043	46	0.88	0.66	0.33	0.16	18400		1274	733	352	1656
2044	47	0.88	0.66	0.32	0.16	18800		1298	745	358	1692
2045	48	0.87	0.66	0.32	0.16	19200		1321	756	363	1728
2046	49	0.87	0.65	0.32	0.15	19600		1345	768	369	1764
2047	50	0.86	0.65	0.31	0.15	20000		1368	779	374	1800
2048	51	0.86	0.65	0.31	0.15	20400		1391	790	380	1836
2049	52	0.86	0.64	0.31	0.15	20800		1415	801	385	1872
2050	53	0.85	0.64	0.30	0.15	21200		1438	812	390	1908
2051	54	0.85	0.64	0.30	0.15	21600		1461	823	396	1944
2052	55	0.85	0.64	0.30	0.15	22000		1484	834	401	1980
2053	56	0.84	0.63	0.29	0.14	22400		1506	844	406	2016
2054	57	0.84	0.63	0.29	0.14	22800		1529	855	411	2052
2055	58	0.84	0.63	0.29	0.14	23200		1552	865	416	2088
2056	59	0.83	0.62	0.29	0.14	23600		1574	875	421	2124
2057	60	0.83	0.62	0.28	0.14	24000		1597	886	426	2160
2058	61	0.83	0.62	0.28	0.14	24400		1619	896	431	2196
2059	62	0.82	0.62	0.28	0.14	24800		1641	906	436	2232
2060	63	0.82	0.61	0.28	0.14	25200		1663	916	441	2268
2061	64	0.82	0.61	0.27	0.13	25600		1685	926	446	2304
2062	65	0.81	0.61	0.27	0.13	26000		1707	936	451	2340
2063	66	0.81	0.61	0.27	0.13	26400		1729	945	455	2376
2064	67	0.81	0.60	0.27	0.13	26800		1751	955	460	2412
2065	68	0.81	0.60	0.27	0.13	27200		1772	964	465	2448
2066	69	0.80	0.60	0.26	0.13	27600		1794	974	469	2484
2067	70	0.80	0.60	0.26	0.13	28000		1816	983	474	2520

[a] Bolded doses indicated are those calculated. Exposure during actual operations would be limited to 1 rem/y.

[b] Aggregate hrs per year assuming 400 h/y per worker for all alternatives except the Proposed Action. See text for Proposed Action Annual exposure times.

[c] Bolded values reflect an administratively limited 1 rem/y dose per worker.

operations are assumed under the Proposed Action, the loss of a small portion of worker hours during the first 20 years of operations would not reduce the overall volume of CH-TRU waste disposed of under the Proposed Action.

Aggregate Dose

To calculate the aggregate dose for the WIPP involved worker population over the entire time period of each alternative, the annual population dose to 36 workers was summed over the years of CH-TRU disposal operations. The time required to completely fill a CH-TRU panel is estimated to be 2.5 years at a 75 percent disposal efficiency rate (DOE 1997a). Under Action Alternative 2, each panel is filled with half the CH-TRU volume of the other alternatives; therefore, 1.25 years of worker exposure time are assumed. Using these operational time periods and the number of CH-TRU panels required for disposal, the expected population doses of each alternative was calculated. Table F-20 indicates the aggregate WIPP involved worker population doses under each alternative. Doses are calculated out to 70 years, indicative of the upper-bound impacts to two generations of involved workers. For comparison purposes, the far right column of Table F-20 indicates the aggregate dose to 36 workers exposed annually to the administrative limit of 1 rem.

Storage Site Involved Workers - Routine Handling and Monitoring

Radiological impacts to involved workers at CH-TRU waste storage facilities would result from the handling and monitoring of CH-TRU waste. Under No Action Alternative 1, however, radiological impacts would result mainly from routine overpacking of CH-TRU waste and were evaluated separately. The radiological impacts to individual workers were assumed to be bounded by site-specific occupational dose limits. The TEDE received by an individual worker cannot exceed 5 rem per year (Title 10 of the Code of Federal Regulations [CFR] Part 835) and may be more restrictive at any of the DOE sites.

The amount of CH-TRU waste in storage will vary over time as a result of the rate of waste characterization and treatment and the rate that waste is shipped from the lag storage site. The number of workers required to handle and monitor the waste varies according to the volume of waste in storage. Other factors that can affect the dose received by storage workers are the design of the storage facilities, the amount of waste handling automation, monitoring activities, and waste handling procedures.

Maximum Annual Storage Worker Populations

To develop radiological impact estimates for storage facility workers, uniform assumptions were applied across all CH-TRU waste storage operations. Individual worker impacts were determined by assuming that exposures occurred at 1 meter (3.3 feet) from CH-TRU waste for 2 hours per day, 5 days per week, 50 weeks per year over the entire 35-year career of the worker. The radionuclides identified as the primary contributors to external doses in the CH-TRU inventory were used to determine the 1-meter (3.3-foot) dose rates (see Tables F-17 and F-18). These radionuclides were decayed over the 35-year exposure period.

To determine maximum storage site worker population doses, the entire site storage worker population was assumed to be exposed in the same manner as the individual worker. Worker populations were estimated by assuming that 20 percent of the total waste volume would be in the storage facility each year and that each worker could manage 1,000 cubic meters (35,300 cubic feet) of waste per year (see Equation F-3). Table F-21 presents involved worker population estimates at storage sites under Action Alternatives 1, 2, and 3 and No Action Alternative 2. The results are extremely conservative because the total waste volume of any site is not expected to be in the storage facility for any extended length of time (see discussion of aggregate storage worker impacts for further discussion of expected waste volumes in storage).

$$W_s = \frac{0.2 \times V_{CH,s}}{T} \qquad \text{(Equation F-3)}$$

where

W_s = worker population at site "s"

$V_{CH,s}$ = site-specific (site "s") post-treatment CH-TRU waste volume from Chapter 3 or Appendix A

T = a constant of 1,000 cubic meters (35,000 cubic feet) of stored waste handled per worker per year

0.2 = fraction of total waste at site "s" handled per year

The number of years of operation for the lag storage sites was assumed to be the maximum occupation exposure period of 35 years. Under the action alternatives, this creates very conservative impact estimates but includes none of the highly uncertain assumptions regarding when each storage site will send waste to WIPP. The results provide an upper-bound case for storage impacts to involved workers. In the following section, the calculation of the aggregate impacts to storage site workers reasonably estimates the amount of waste in storage according to assumed site-specific treatment and shipment rates.

Aggregate Impacts to Storage Site Workers

The volume of waste and length of time that waste is located at a lag storage facility is directly related to the impacts received by storage workers at each site. Assumptions regarding waste treatment and shipment rate are required to determine estimates of the volume and length of time waste is located in lag storage. These assumptions were also made in order to present more realistic estimates of expected impacts to storage workers that can be used for comparison across alternatives and that can be used to estimate aggregate storage worker impacts.

Waste Treatment Facility Throughput

Estimates were made of the CH-TRU waste treatment throughput rates to determine the extent of lag storage under the action alternatives (no lag storage would be required under the Proposed Action). Waste was assumed to be treated at a constant annual rate over the 35-year waste treatment period under each alternative. All sites were assumed to be able to characterize waste to meet the planning-basis WAC in 1998. Design and construction of waste treatment facilities to thermally treat waste and to shred and grout waste was assumed to last 12 years before treatment operations could begin (see Chapter 3).

It was assumed that a small amount of waste could meet the thermal or grouted waste treatment criteria and would be sent to WIPP during the design and construction period. For thermal treatment (Action Alternative 2), it was assumed that enough waste could be shipped to WIPP to

Table F-21
Maximum Storage Site Involved Worker Population Estimates

Site	Alternative	Total Worker Population	
		Basic Inventory	Total Inventory
Total	Action Alternative 1	28	28
	Action Alternative 2A	10	21
	Action Alternative 2B	9	21
	Action Alternative 3	33	67
	No Action Alternative 2	27	N/A
Hanford	Action Alternative 1	12	12
	Action Alternative 2A	4	9
	Action Alternative 2B	4	9
	Action Alternative 3	14	29
	No Action Alternative 2	12	N/A
INEEL	Action Alternative 1	6	6
	Action Alternative 2A	2	8
	Action Alternative 2B	4	11
	Action Alternative 3	7	21
	No Action Alternative 2	6	N/A
LANL	Action Alternative 1	4	4
	Action Alternative 2A	1	2
	Action Alternative 2B	N/A	N/A
	Action Alternative 3	5	8
	No Action Alternative 2	4	N/A
RFETS	Action Alternative 1	4	4
	Action Alternative 2A	1 [a]	1 [a]
	Action Alternative 2B	N/A	N/A
	Action Alternative 3	4	4
	No Action Alternative 2	5	N/A
ORNL	Action Alternative 1	1 [a]	1 [a]
	Action Alternative 2A	N/A	N/A
	Action Alternative 2B	N/A	N/A
	Action Alternative 3	N/A	N/A
	No Action Alternative 2	1 [a]	N/A
SRS	Action Alternative 1	2	2
	Action Alternative 2A	1	1
	Action Alternative 2B	1	1
	Action Alternative 3	3	5
	No Action Alternative 2	2	N/A

[a] Involved workers were assumed to not be involved in waste handling full-time.

N/A = Not Applicable

operate the waste receiving and emplacement operations at 25 percent throughput. For shred and grout treatment (Action Alternative 3), it was assumed that enough waste could be shipped to WIPP to operate the waste receiving and emplacement operations at 40 percent throughput capacity. A result of this assumption is that no site would require the full 35-year treatment period to treat the waste volume under Action Alternative 2 or Action Alternative 3.

To estimate site-specific waste treatment throughput, a "standard facility" was determined for each treatment type (i.e., WAC, thermal treatment, or shred and grout) and for treatment of the Basic Inventory or Total Inventory. The standard facility was sized to be able to treat one-half of the waste at the site with the largest volume of CH-TRU waste within the 35-year waste treatment period for each action alternative. The site with the largest CH-TRU waste volume would have standard-sized treatment facilities, while sites with less waste volume would have correspondingly smaller waste treatment facilities. For Action Alternative 2C, where CH-TRU waste is treated only at WIPP, a single standard facility was sized to process the entire volume of waste over

35 years. The number of standard-sized treatment facilities at each site under each action alternative is presented in Table F-22. Standard facility throughput rates are presented in the footnote of Table F-22.

Waste Shipment Rates

The WIPP CH-TRU waste receiving docks can handle a maximum of 50 TRUPACT-IIs per week. This rate is considered a WIPP shipment receiving efficiency of 100 percent. For this aggregate storage worker impact analysis, it is assumed that the receipt of WIPP waste can occur at a maximum efficiency of 80 percent (13 to 14 trucks per week) when waste treatment facilities are operational. Prior to treatment facility operations under Action Alternatives 2 and 3, the number of trucks received at WIPP would be consistent with operating efficiencies of 25 percent (4 trucks per week) and 42 percent (7 trucks per week), respectively.

Under Action Alternatives 1 and 3, waste trucks sent from a site to WIPP are assumed to bear three TRUPACT-IIs, each carrying 14 drum-equivalents of waste, for a total volume per truck of 8.74 cubic meters (308.65 cubic feet). Transportation impact assumptions for Action Alternative 3 indicate that about 86 percent of the trucks would consist of two TRUPACT-IIs instead of three (see Section A.3.9). This lack of consistency was investigated and found to not result in a significant difference in the time required to send all waste to WIPP. For the dense waste of Action Alternative 2, each of the three TRUPACT-IIs on a truck were assumed to contain seven drums of waste and seven drums of dunnage. This Action Alternative 2 assumption is consistent with transportation impact assumptions which indicate that, under Action Alternative 2, shipments would consist of two full TRUPACT-IIs per truck (see Section A.3.9).

The number of trucks transporting waste from each specific lag storage site are indicated in Table F-23. The number of trucks sending waste to WIPP from each lag storage site was determined according to the relative total waste volumes at each site destined for WIPP and the treatment throughput rates. Sites with smaller waste volumes destined for WIPP will complete shipping before the larger sites. When trucks finish shipping waste to WIPP from the smaller sites, they would be sent to the larger sites to maintain the 80 percent shipment receiving operation efficiency at WIPP. Table F-24 indicates how and when available trucks would be distributed to the larger sites.

Storage Worker Impacts

The annual amount of waste in lag storage is calculated by assuming alternative- and site-specific treatment and shipment rates. The amount of waste in storage can be used with site-specific dose rates at 1 meter (3.3 feet) and the number of workers who routinely monitor the waste to estimate worker population impacts over the entire period of lag storage. The data is then used to determine the expected maximum fraction of the total waste volume that can be found in lag storage, the number of years each site ships waste to WIPP, the total number of worker-years required to monitor waste in lag storage, and the aggregate worker population consequences of the lag storage (i.e., aggregate population dose and LCF estimates) over the entire disposal operations period under each alternative. Section F.3.1 includes the results of this analysis.

Table F-22
Site-Specific Treatment Facility Throughputs

Alternative	Lag Storage Site	Inventory (m^3)	Treatment Facility size (n x standard plant size) [a]	Period of Waste Treatment (in years)
Action Alternative 1	**Hanford**	Basic 57,000	2 x B	35
		Total 120,000	2 x T	35
	INEEL	Basic 30,000	1.25 x B	30
		Total 87,000	1.5 x T	34
	LANL	Basic 21,000	1 x B	26
		Total 35,000	0.75 x T	28
	RFETS	Basic 17,000	1 x B	21
		Total 17,000	0.5 x T	20
	SRS	Basic 12,000	0.75 x B	20
		Total 17,000	0.5 x T	20
	ORNL	Basic 1,800	0.25 x B	9
		Total 2,100	0.25 x T	5
Action Alternative 2A	**Hanford**	Basic 21,000	2 x B	29
		Total 43,000	2 x T	32
	INEEL	Basic 10,000	1 x B	25
		Total 41,000	2 x T	32
	LANL	Basic 7,400	1 x B	21
		Total 12,000	0.75 x T	24
	RFETS	Basic 3,800	0.5 x B	17
		Total 3,800	0.5 x T	9
	SRS	Basic 5,000	0.5 x B	25
		Total 6,800	0.5 x T	19
Action Alternative 2B	**Hanford**	Basic 21,000	2 x B	29
		Total 43,000	2 x T	24
	INEEL	Basic 22,000	2 x B	29
		Total 57,000	2 x T	32
	SRS	Basic 5,000	1 x B	13
		Total 6,900	1 x T	7
Action Alternative 2C	**WIPP**	Basic 47,000	1 x B	28
		Total 107,000	1 x T	32
Action Alternative 3	**Hanford**	Basic 70,000	2 x B	29
		Total 146,000	2 x T	32
	INEEL	Basic 37,000	1.5 x B	20
		Total 105,000	2 x T	24
	LANL	Basic 25,000	1 x B	20
		Total 42,000	1 x T	18
	RFETS	Basic 19,000	1 x B	14
		Total 19,000	0.5 x T	14
	SRS	Basic 17,000	1 x B	12
		Total 23,000	0.5 x T	18

[a] Standard Plant sizes for Basic Inventory (B) and Total Inventory (T) in drum-equivalents per day (dr-eq/d) are:

AA1: B = 21 dr-eq/d, T = 44 dr-eq/d.
AA2A: B = 9.5 dr-eq/d, T = 19.5 dr-eq/d.
AA2B: B = 9.5 dr-eq/d, T = 26.5 dr-eq/d.
AA2C: B = 43 dr-eq/d, T = 98 dr-eq/d.
AA3: B = 38.5 dr-eq/d, T = 80 dr-eq/d.

Table F-23
Shipment Rates During the Design-and-Construction Phases and the Fully Operational Treatment Phases

Sites	AA1 All years	AA2A Disposal years 1-12	AA2A Disposal years 13+	AA2B Disposal years 1-12	AA2B Disposal years 13+	AA2C Disposal years 1-12	AA2C Disposal years 13+	AA3 Disposal years 1-12	AA3 Disposal years 13+
Hanford	5	1.5	5	1.5	5.5	N/A	N/A	2.5	4
INEEL	3.5	1	4.5	2	6	N/A	N/A	1.5	3
LANL	2	0.5	2	N/A	N/A	N/A	N/A	1	2
RFETS	1.5	0.5	1.5	N/A	N/A	N/A	N/A	1	2
SRS	1.5	0.5	1	0.5	2	N/A	N/A	1	2
ORNL	0.5	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
WIPP	N/A	N/A	N/A	N/A	N/A	4	13	N/A	N/A

N/A = Not Applicable

Table F-24
Assumed Truck Transfer Schedule

	Basic Inventory				Total Inventory			
	Disposal Year	***n* Trucks Added per Week**	**To Site**	**From Site**	**Disposal Year**	***n* Trucks Added per Week**	**To Site**	**From Site**
Action Alternative 1	10	0.5	RFETS	ORNL	11	0.5	Hanford	ORNL
					27	1	LANL	RFETS
					27	1	Hanford	0.5
					27	1	INEEL	RFETS + 0.5
					37	2	INEEL	SRS
					37	1	Hanford	
Action Alternative 2A		No Changes [a]			22	1	INEEL	RFETS
					22	0.5	Hanford	RFETS
					38	2	INEEL	LANL
					39	1	Hanford	SRS
Action Alternative 2B		No Changes [a]			26	1	Hanford	SRS
					26	1	INEEL	SRS + 0.5 new
Action Alternative 2C		No Changes				No Changes		
Action Alternative 3	17	0.5	Hanford	RFETS	17	1	Hanford	RFETS
	17	0.5	INEEL	RFETS	17	1	INEEL	RFETS
					34	1	LANL	SRS
					34	1	Hanford	SRS
					49	2	INEEL	LANL
					49	1	Hanford	LANL

[a] The WIPP waste receipt efficiency cannot reach 80 percent because waste volume resulting from treatment limits the amount annually available for shipment. If waste treatment rates were increased, the efficiency of the WIPP waste receiving operations could reach 80 percent.

Overpacking Operations for No Action Alternative 1

Despite the variation in operations at each packaging facility, the impacts from overpackaging were calculated to provide a rough idea of how radiological impacts to the involved worker would compare across the consolidation sites under No Action Alternative 1. Overpacking impact estimates were determined by assuming identical waste handling at all consolidation sites. For impacts to the worker population, it was assumed that the average worker would be 1 meter (3.3 feet) from the waste for 2 hours per day, 5 days per week, 50 weeks per year, and all individuals in the worker population would be involved in these operations over 35 years. The maximally exposed involved worker was assumed to be exposed for twice as long (4 hours per day). The average 1-meter (3.3-foot) dose rates were adjusted annually for radioactive decay; the greatest external dose rates would occur early in the operations period. Therefore, the maximum impacts to workers were estimated from year 20 to year 55 for storage operations because CH-TRU waste containers have a design life of 20 years. The annual occupational dose limit (TEDE) as regulated by 10 CFR Part 835 is 5 rem and was assumed to apply to all packaging workers at the various sites.

The involved worker populations under No Action Alternative 1 were determined by assuming that 4.2 workers would be required to overpackage 12 drums a day. This level of overpackaging maximizes the worker population because worker personnel protection (e.g., workers have self-contained breathing apparatus and are fully suited-up) is maximized. The involved worker population required at each site for overpackaging operations, based on site-specific volumes of waste, are as follows for No Action Alternative 1A: 35 total workers with 14 at Hanford, 14 at INEEL, 4 at LANL, 1 at RFETS, and 2 at SRS; and for No Action Alternative 1B: 35 total workers with 14 at Hanford, 19 at INEEL, and 2 at SRS.

F.3 HUMAN HEALTH IMPACTS

The radiological and hazardous chemical impacts from TRU waste storage at the major consolidation sites and from WIPP disposal operations are presented in this section.

F.3.1 Radiological Impacts

Radiological impacts to the public and noninvolved workers at the storage sites and WIPP may result from the releases of gaseous radionuclides present in TRU waste containers. Radionuclides contributing to these impacts are carbon-14 and radon-222. The radiation doses resulting from gaseous releases are presented in Table F-25 and are well below background radiation levels. The radiological impacts of these doses, presented in Chapter 5, were estimated using ICRP 60 dose-to-risk factors.

Radiological impacts to involved workers at WIPP and at storage sites are presented in Tables F-26 and F-27. Table F-26 indicates the WIPP involved worker dose estimates under each alternative, the individual involved worker and population dose for a 35-year occupational exposure period, the individual involved worker dose for the time required to dispose of the CH-TRU waste under each alternative (bounded at 35 years), and the aggregate involved worker population dose for the time required to dispose of CH-TRU waste in the 2.5-year timeframe required to fill a panel. For comparison, the WIPP administrative limit (1 rem per year per person) for a 35-year exposure period is also indicated in Table F-26.

Radiological presents to the involved workers at lag storage sites are presented in Table F-28. These are upper-bound occupational exposures, assuming routine handling and monitoring of a site’s total waste volume for 35 years. No Action Alternative 1 requires periodic repackaging operations, and Table F-29 indicates the radiological impacts of these repackaging operations.

Table F-27 presents the results of the more realistic assessment of storage worker impacts, where site-specific shipment and treatment rates were considered in the analysis. These impacts may better reflect the comparative radiological impacts to the population of involved workers across the alternatives. The maximum fraction of a site’s total expected waste volume in storage is indicated in the table.

Table F-25
Radiation Doses from Routine Releases
of Gaseous Radionuclides for the Public and Non-Involved Workers [a, b]

Alternative	Hanford C-14	INEEL Rn-222	LANL Rn-222	RFETS	ORNL Rn-222	SRS Rn-222	WIPP Rn-222
Proposed Action [c]							
MEI (rem per 35 years)	0E+0	N/A	N/A	N/A	2E-6	N/A	5E-4
Population (person-rem per 35 years)	0E+0				4E-2		0.6
Noninvolved Worker (rem per 35 years)	0E+0				2E-5		1E-3
Action Alternative 1							
MEI (rem per 70 years)	2E-5	3E-6	1E-3	No known inventory for Rn-222 or C-14	3E-3	2E-10	9E-4
Population (person-rem per 35 years)	2E-1	1E-2	4E+0		2E+1	8E-6	6E-1
Noninvolved Worker (rem per 35 years)	3E-4	1E-4	2E-3		2E-2	2E-8	1E-3
Action Alternative 2A							
MEI (rem per 70 years)	7E-6	1E-6	3E-4	No known inventory for Rn-222 or C-14	N/A	2E-5	2E-4
Population (person-rem per 35 years)	1E-1	5E-3	1E+0			1E+0	1E-1
Noninvolved Worker (rem per 35 years)	2E-4	6E-5	6E-4			3E-3	4E-4
Action Alternative 2B							
MEI (rem per 70 years)	7E-6	9E-6	N/A	N/A	N/A	2E-5	2E-4
Population (person-rem per 35 years)	1E-1	3E-2				1E+0	1E-1
Noninvolved Worker (rem per 35 years)	2E-4	4E-4				3E-3	4E-4
Action Alternative 2C							
MEI (rem per 70 years)	N/A	N/A	N/A	N/A	N/A	N/A	2E-4
Population (person-rem per 35 years)							1E-1
Noninvolved Worker (rem per 35 years)							4E-4
Action Alternative 3							
MEI (rem per 70 years)	1E-5	3E-6	6E-4	No known inventory of Rn-222 or C-14	N/A	5E-5	5E-4
Population (person-rem per 35 years)	1E-1	1E-2	2E+0			2E+0	3E-1
Noninvolved Worker (rem per 35 years)	2E-4	1E-4	1E-3			5E-3	7E-4
No Action Alternative 2							
MEI (rem per 70 years)	2E-6	5E-7	7E-4	No known inventory of Rn-222 or C-14	3E-3	2E-10	N/A
Population (person-rem per 35 years)	3E-2	2E-3	3E+0		2E+0	7E-6	
Noninvolved Worker (rem per 35 years)	5E-5	2E-5	1E-3		2E-2	2E-8	

[a] The predominant gaseous radionuclide is carbon-14 (C-14) or radon-222 (Rn-222), as indicated below site heading.
[b] N/A indicates the site is not a consolidation site under the alternative.
[c] The Hanford and ORNL doses are from the excess RH-TRU waste remaining at these sites under the Proposed Action.

Table F-26
Aggregate Dose to WIPP Involved Worker Populations

Alternative	Maximum Involved Worker Dose (total rem over 35 years)	Maximum Involved Worker Population Dose (person-rem over 35 years)	Inventory	Number of Panels	Years of CH-TRU Waste Disposal Operations	Involved Worker (rem) [a]	Aggregate Worker Population Dose (person-rem) [b]
Proposed Action	35	1240	Basic Basic	10 10	25 (min.) 35 (max.)	25 35	898 1240
Action Alternative 1	28	1010	Basic Total	8.5 16.7	22 43	18 28	662 1202
Action Alternative 2	17	612	Basic Total	5.7 12.8	7 15	4 8	149 296
Action Alternative 3	8	288	Basic Total	10.0 19.8	25 50	6 8	218 374
Administrative Limit	35	1260	N/A N/A	N/A N/A	35 70	35 N/A	N/A N/A

[a] Exposure period is equal to the lesser of the years of CH-TRU waste operations or 35 years.
[b] Exposure period is equal to the years of CH-TRU waste disposal operations.

N/A = Not Applicable

F.3.2 Impacts from VOCs

The impacts of routine releases of VOCs from TRU waste containers at the major storage sites and WIPP were evaluated for the Proposed Action, Action Alternative 1, Action Alternative 3, and No Action Alternative 2. Waste would not be stored at sites under the Proposed Action, and no VOCs would be emitted under Action Alternative 2 and No Action Alternative 1 because of the thermal waste treatment.

F.3.2.1 Proposed Action

Forty-two CH-TRU and three RH-TRU waste drum-equivalents were assumed to be constantly present in the WHB. All 45 drums were assumed to contain VOCs at the planning-basis WAC concentration limit or, where no specific limit was established, the weighted maximum VOC concentration. From the underground, a full single panel containing approximately 84,000 drum-equivalents of TRU waste (alternative-specific values indicated in Table F-6) was assumed to be continuously releasing VOCs every year. All drums in the underground area were assumed to contain the weighted average concentrations of VOCs. Panels would be sealed upon filling; therefore, no more than one panel would be open at any one time. All VOCs were assumed to be released to the atmosphere from the underground exhaust ventilation stack of the Exhaust Filter Building. WHB workers and the underground workers would be exposed to VOCs in the ventilation air; low ventilation rates were assumed to bound potential exposures.

The impacts, dominated by the releases from the underground area rather than from the WHB, are presented in Table F-30. The maximum HI resulting from the stack releases would be 6×10^{-4} (noninvolved worker) and 7×10^{-5} (MEI) from the carbon tetrachloride releases. An HI of one or greater to a member of the public would predict a noncarcinogenic health effect. WHB and underground area workers would have HIs greater than one for carbon tetrachloride (see Table F-30). The HI for methylene chloride is 0.9 for the WHB workers.

Table F-27
Aggregate Lag Storage Worker Population Impacts

Alternative	Storage Sites and Inventories [a]	Maximum Fraction	Time to Ship (in years)	Worker-years	Aggregate Population Dose (person-rem)	LCFs
Action Alternative 1	Basic					
	Hanford	0.00	35	0	0	0
	INEEL	0.00	30	0	0	0
	LANL	0.00	26	0	0	0
	RFETS	0.09	22	25	160	6.4E-2
	SRS	0.00	20	0	0	0
	ORNL	0.00	9	0	0	0
	Total			25	160	6.4E-2
	Total					
	Hanford	0.28	46	193	117	4.7E-2
	INEEL	0.36	46	179	199	7.9E-2
	LANL	0.31	36	57	20	7.8E-3
	RFETS	0.23	26	25	160	6.4E-2
	SRS	0.23	26	25	3.4	1.4E-3
	ORNL	0.48	10	9	11	4.2E-3
	Total			488	509	2.0E-1
Action Alternative 2A	Basic					
	Hanford	0.00	41	0	0	0
	INEEL	0.00	37	0	0	0
	LANL	0.00	33	0	0	0
	RFETS	0.00	29	0	0	0
	SRS	0.00	37	0	0	0
	Total			0	0	0
	Total					
	Hanford	0.03	49	27	21	8.5E-3
	INEEL	0.10	52	32	10	4.2E-3
	LANL	0.04	37	0	0	0
	RFETS	0.00	21	0	0	0
	SRS	0.23	38	25	1.9	7.5E-3
	Total			84	34	1.3E-2
Action Alternative 2B	Basic					
	Hanford	0	41	0	0	0
	INEEL	0	41	0	0	0
	SRS	0	25	0	0	0
	Total			0	0	0
	Total					
	Hanford	0.19	43	43	29	1.1E-2
	INEEL	0.12	49	50	22	8.7E-3
	SRS	0.37	25	12	0.9	3.7E-4
	Total			105	51	2.1E-2
Action Alternative 2C	Basic					
	WIPP	0	40	0	0	0
	Total					
	WIPP	0	44	0	0	0
Action Alternative 3	Basic					
	Hanford	0.03	42	29	8	3.1E-3
	INEEL	0.02	32	19	4	1.6E-3
	LANL	0.10	35	22	1.9	7.7E-4
	RFETS	0.09	16	15	24	9.5E-3
	SRS	0.08	26	13	0.5	2.1E4
	Total			98	38	1.5E-2
	Total					
	Hanford	0.41	65	359	76	3.0E-2
	INEEL	0.55	62	336	65	2.6E-2
	LANL	0.50	48	95	8	3.2E-3
	RFETS	0.12	16	15	24	9.5E-3
	SRS	0.13	33	20	0.8	3.2E4
	Total			825	174	6.9E-2

Table F-28
Storage Site Involved Worker Lifetime Radiological Impacts (35-year Exposures)

Site	Alternative	Individual		Population			
				Dose (person-rem)		Number of LCFs	
		Dose (rem)	Probability of an LCF	Basic Inventory	Total Inventory	Basic Inventory	Total Inventory
Hanford	Action Alternative 1	24	0.01	279	279	0.1	0.1
	Action Alternative 2A	29	0.01	119	247	0.05	0.1
	Action Alternative 2B	29	0.01	119	247	0.05	0.1
	Action Alternative 3	11	4E-03	162	334	0.06	0.1
	No Action Alternative 2	24	0.01	279	N/A	0.1	N/A
INEEL	Action Alternative 1	40	0.02	241	241	0.1	0.1
	Action Alternative 2A	13	5E-03	26	104	0.01	0.04
	Action Alternative 2B	16	6E-03	71	189	0.03	0.08
	Action Alternative 3	8	3E-03	57	161	0.02	0.06
	No Action Alternative 2	39	0.02	225	N/A	0.09	N/A
LANL	Action Alternative 1	12	5E-03	52	52	0.02	0.02
	Action Alternative 2A	7	3E-03	10	17	4E-03	7E-03
	Action Alternative 2B	N/A	N/A	N/A	N/A	N/A	N/A
	Action Alternative 3	3	1E-03	16	27	6E-03	0.01
	No Action Alternative 2	12	5E-03	51	N/A	0.02	N/A
RFETS	Action Alternative 1	191	0.08	764	764	0.3	0.3
	Action Alternative 2A	89	0.04	68	68	0.03	0.03
	Action Alternative 2B	N/A	N/A	N/A	N/A	N/A	N/A
	Action Alternative 3	56	0.02	209	209	0.08	0.08
	No Action Alternative 2	153	0.06	764	N/A	0.3	N/A
ORNL	Action Alternative 1	32	0.01	12	12	5E-03	5E-03
	Action Alternative 2A	N/A	N/A	N/A	N/A	N/A	N/A
	Action Alternative 2B	N/A	N/A	N/A	N/A	N/A	N/A
	Action Alternative 3	N/A	N/A	N/A	N/A	N/A	N/A
	No Action Alternative 2	7	3E-03	3	N/A	1E-03	N/A
SRS	Action Alternative 1	5	2E-03	11	11	4E-03	4E-03
	Action Alternative 2A	3	1E-03	3	4	1E-03	2E-03
	Action Alternative 2B	3	1E-03	3	4	1E-03	2E-03
	Action Alternative 3	1	4E-04	5	7	2E-03	3E-03
	No Action Alternative 2	2	8E-04	4	N/A	2E-03	N/A

N/A = Not Applicable

Table F-29
Overpackaging Involved Worker Radiological Impacts for No Action Alternative 1 (35-year Exposures)

Site	No Action Alternative 1A				No Action Alternative 1B			
	Maximally Exposed Worker		Worker Population		Maximally Exposed Worker		Worker Population	
	Dose (rem)	Probability of an LCF	Dose (person-rem)	LCFs	Dose (rem)	Probability of an LCF	Dose (person-rem)	LCFs
Hanford	38	0.02	266	0.1	38	0.02	266	0.1
INEEL	22	9E-3	154	0.06	30	0.01	285	0.1
LANL	12	5E-3	24	0.01	N/A	N/A	N/A	N/A
RFETS	172	0.07	86	0.03	N/A	N/A	N/A	N/A
SRS	5	2E-3	5	2E-3	5	2E-3	5	2E-3

N/A = Not Applicable

Table F-30
Proposed Action Human Health Impacts from Routine Releases of VOCs at WIPP and Excess RH-TRU Waste Storage Impacts at Hanford and ORNL

Individual/Population	Lifetime Cancer Risk	Major Contributor (percent of total)	Maximum HI	VOC with Maximum HI
Noninvolved Worker	1E-7	1,1,2,2-tetrachloroethane (86)	6E-4	Carbon tetrachloride
35-year MEI	3E-8	1,1,2,2-tetrachloroethane (96)	7E-5	Carbon tetrachloride
50-mile Population	2E-5	1,1,2,2-tetrachloroethane (89)	N/A	N/A
WHB Worker	2E-4	1,2-dichloroethane (30)	2	Carbon tetrachloride
Underground Worker	9E-4	1,1,2,2-tetrachloroethane (86)	6	Carbon tetrachloride
Sites	**Excess RH-TRU Waste Storage Impacts**			
Noninvolved Worker –				
Hanford	4E-8	1,1,2,2 – tetrachloroethane (91)	1E-4	Carbon tetrachloride
ORNL	9E-9	1,1,2,2 – tetrachloroethane (97)	1E-5	Carbon tetrachloride
70-year MEI –				
Hanford	2E-8	1,1,2,2 – tetrachloroethane (72)	8E-6	Carbon tetrachloride
ORNL	4E-8	1,1,2,2 – tetrachloroethane (98)	1E-5	Carbon tetrachloride
50-mile Population –				
Hanford	1E-4	1,1,2,2 – tetrachloroethane (92)	N/A	N/A
ORNL	2E-4	1,1,2,2 – tetrachloroethane (95)	N/A	N/A

N/A = Not Applicable

For these VOCs, the air concentration was compared to the OSHA PEL (see Table F-3). The air concentration for the workers would be at least three orders of magnitude less than the PELs; therefore, no involved worker noncarcinogenic health effects would be expected at WIPP from routine releases under the Proposed Action. A lifetime probability of cancer incidence of 1×10^{-7} (noninvolved worker) and 3×10^{-8} (MEI) may result from the routine releases of VOCs over the 35-year operating period. Annual risk would be 1/35th of the total values. No cancers would occur in the 80-kilometer (50-mile) population (2×10^{-5} cancers).

F.3.2.2 Action Alternative 1

The same assumptions used for the Proposed Action to analyze the impacts at WIPP were used for Action Alternative 1 impact analyses.

The impacts to the noninvolved worker, MEI, and populations at WIPP and the lag storage sites under Action Alternative 1 are presented in Table F-31. The maximum annual HI from stack releases of carbon tetrachloride would be 4×10^{-3} (noninvolved worker at INEEL) and 6×10^{-4} (MEI at LANL or Hanford). No noncarcinogenic effects would be expected from the routine VOC releases under Action Alternative 1. Impacts to the involved workers in the WHB and the underground area would be the same under Action Alternative 1 as under the Proposed Action. The highest individual probabilities of cancer incidence estimated for the routine releases of hazardous chemicals were 5×10^{-7} (noninvolved worker at INEEL) and 1×10^{-7} (MEI at Hanford, LANL, or ORNL). This would be the lifetime carcinogenic risk for the 35 years (noninvolved worker) and 70 years (MEI) to which the individual was conservatively assumed to be exposed. The maximum annual impact would be equivalent to 1/35th or 1/70th of the total noninvolved worker or MEI value, respectively. No cancers would be expected in the 80-kilometer (50-mile) population (maximum expected cancers over each 35-year period would be 1×10^{-3} for RFETS).

Table F-31
Action Alternative 1 Human Health Impacts from Routine Releases of VOCs

Site	Lifetime Cancer Risk	Major Contributor (percent of total)	Maximum HI	VOC with Maximum HI
WIPP				
Noninvolved Worker	1E-7	1,1,2,2-tetrachloroethane (86)	6E-4	Carbon tetrachloride
70-year MEI	5E-8	1,1,2,2-tetrachloroethane (94)	7E-5	Carbon tetrachloride
50-mile Population	2E-5	1,1,2,2-tetrachloroethane (89)	N/A	N/A
WHB Worker	2E-4	1,2-dichloroethane (30)	2	Carbon tetrachloride
Underground Worker	9E-4	1,1,2,2-tetrachloroethane (86)	6	Carbon tetrachloride
Lag storage - Hanford East & West				
Noninvolved Worker	2E-7	1,1,2,2-tetrachloroethane (92)	5E-4	Carbon tetrachloride
70-year MEI	1E-7	1,1,2,2-tetrachloroethane (95)	6E-4	Carbon tetrachloride
50-mile Population	3E-4	1,1,2,2-tetrachloroethane (91)	N/A	N/A
Lag storage - INEEL				
Noninvolved Worker	5E-7	1,1,2,2-tetrachloroethane (84)	4E-3	Carbon tetrachloride
70-year MEI	4E-8	1,1,2,2-tetrachloroethane (90)	3E-5	Carbon tetrachloride
50-mile Population	1E-4	1,1,2,2-tetrachloroethane (80)	N/A	N/A
Lag storage - LANL				
Noninvolved Worker	4E-8	1,1,2,2-tetrachloroethane (64)	7E-4	Carbon tetrachloride
70-year MEI	1E-7	1,1,2,2-tetrachloroethane (79)	6E-4	Carbon tetrachloride
50-mile Population	3E-4	1,1,2,2-tetrachloroethane (66)	N/A	N/A
Lag storage - RFETS				
Noninvolved Worker	1E-8	1,1,2,2-tetrachloroethane (72)	2E-4	Carbon tetrachloride
70-year MEI	2E-8	1,1,2,2-tetrachloroethane (88)	6E-5	Carbon tetrachloride
50-mile Population	1E-3	1,1,2,2-tetrachloroethane (65)	N/A	N/A
Lag storage - ORNL				
Noninvolved Worker	3E-8	1,1,2,2-tetrachloroethane (96)	3E-5	Carbon tetrachloride
70-year MEI	1E-7	1,1,2,2-tetrachloroethane (99+)	3E-5	Carbon tetrachloride
50-mile Population	6E-4	1,1,2,2-tetrachloroethane (95)	N/A	N/A
Lag storage - SRS				
Noninvolved Worker	1E-7	1,1,2,2-tetrachloroethane (92)	2E-4	Carbon tetrachloride
70-year MEI	3E-8	1,1,2,2-tetrachloroethane (96)	2E-5	Carbon tetrachloride
50-mile Population	5E-4	1,1,2,2-tetrachloroethane (93)	N/A	N/A

N/A = Not Applicable

F.3.2.3 Action Alternative 2

No routine releases of VOCs would occur because VOCs in the waste would be destroyed during thermal treatment to meet the LDRs.

F.3.2.4 Action Alternative 3

Action Alternative 3 considers the treatment of the Basic and Additional Inventories of TRU waste by shred and grout. The VOC headspace concentrations are driven by the waste matrix, and the matrix of all Action Alternative 3 waste is a uniform solidified inorganic. As a result, all headspace concentrations for the waste at all consolidation sites were assumed to be identical.

The increased waste volume (as compared to WAC-packaged waste) resulting from the shred and grout process would require that more time be given to dispose of the waste. There is lag storage of waste at consolidation sites. Impacts at WIPP would be greatest in the early years of operations, when both the CH-TRU waste and RH-TRU waste are being disposed of. The maximum annual impacts were estimated using methods similar to those described for Action Alternative 1. It was conservatively assumed that the total number of drum-equivalents that a consolidation site would eventually send to WIPP would release VOCs into the atmosphere at the lag storage sites at the beginning of the operations period. It was also assumed that the MEI would be exposed to this release over a 70-year period, and the noninvolved worker and noninvolved worker population would be exposed over a 35-year period.

The impacts to the noninvolved worker, MEI, and populations at WIPP and the lag storage sites are presented in Table F-32. The maximum HIs for the noninvolved worker and the MEI would be 7 x 10^{-3} (noninvolved worker at Hanford) and 9 x 10^{-4} (MEI at LANL) for carbon tetrachloride. Therefore, no noncarcinogenic effects would be expected from routine stack releases under Action Alternative 3. The Action Alternative 3 WHB worker and underground worker have HIs greater than one for carbon tetrachloride (2 [WHB] and 10 [underground]). Air concentrations for involved workers would be at least two orders of magnitude below the PEL. Individual

Table F-32
Action Alternative 3 Human Health Impacts from Routine Releases of VOCs

Site	Lifetime Cancer Risk	Major Contributor (percent of total)	Maximum HI	VOC with Maximum HI
WIPP				
Noninvolved Worker	5E-8	1,1,2,2-tetrachloroethane (62)	1E-3	Carbon tetrachloride
70-year MEI	2E-8	1,1,2,2-tetrachloroethane (77)	1E-4	Carbon tetrachloride
50-mile Population	1E-5	1,1,2,2-tetrachloroethane (60)	N/A	N/A
WHB Worker	6E-5	Carbon tetrachloride (72)	2	Carbon tetrachloride
Underground Worker	5E-4	1,1,2,2-tetrachloroethane (60)	10	Carbon tetrachloride
Lag storage - Hanford East & West				
Noninvolved Worker	4E-7	1,1,2,2-tetrachloroethane (63)	7E-3	Carbon tetrachloride
70-year MEI	2E-7	1,1,2,2-tetrachloroethane (76)	7E-4	Carbon tetrachloride
50-mile Population	6E-4	1,1,2,2-tetrachloroethane (58)	N/A	N/A
Lag storage - INEEL				
Noninvolved Worker	2E-7	1,1,2,2-tetrachloroethane (67)	4E-3	Carbon tetrachloride
70-year MEI	3E-8	1,1,2,2-tetrachloroethane (83)	9E-5	Carbon tetrachloride
50-mile Population	6E-5	1,1,2,2-tetrachloroethane (59)	N/A	N/A
Lag storage - LANL				
Noninvolved Worker	5E-8	1,1,2,2-tetrachloroethane (63)	9E-4	Carbon tetrachloride
70-year MEI	2E-7	1,1,2,2-tetrachloroethane (81)	9E-4	Carbon tetrachloride
50-mile Population	4E-4	1,1,2,2-tetrachloroethane (60)	N/A	N/A
Lag storage - RFETS				
Noninvolved Worker	1E-8	1,1,2,2-tetrachloroethane (63)	2E-4	Carbon tetrachloride
70-year MEI	2E-8	1,1,2,2-tetrachloroethane (81)	7E-5	Carbon tetrachloride
50-mile Population	1E-3	1,1,2,2-tetrachloroethane (62)	N/A	N/A
Lag storage - ORNL				
Noninvolved Worker	5E-9	1,1,2,2-tetrachloroethane (70)	8E-5	Carbon tetrachloride
70-year MEI	2E-8	1,1,2,2-tetrachloroethane (81)	1E-4	Carbon tetrachloride
50-mile Population	1E-4	1,1,2,2-tetrachloroethane (59)	N/A	N/A
Lag storage - SRS				
Noninvolved Worker	5E-8	1,1,2,2-tetrachloroethane (76)	6E-4	Carbon tetrachloride
70-year MEI	1E-8	1,1,2,2-tetrachloroethane (76)	6E-5	Carbon tetrachloride
50-mile Population	2E-4	1,1,2,2-tetrachloroethane (58)	N/A	N/A

N/A = Not Applicable

probabilities of cancer incidence of 4 x 10^{-7} (noninvolved worker at Hanford) and 2 x 10^{-7} (MEI at LANL) were estimated from exposure to routine releases of VOCs. This would be the lifetime carcinogenic risk for the 35 years (noninvolved worker) and 70 years (MEI) that the individual was conservatively assumed to be exposed. No cancers would occur in the 80-kilometer (50-mile) population (maximum expected number of cancers over each 35-year period would be 1 x 10^{-3} for RFETS).

F.3.2.5 No Action Alternative 1

No routine releases of VOCs would occur because VOCs in the waste would be destroyed during thermal treatment to meet the LDRs.

F.3.2.6 No Action Alternative 2

No Action Alternative 2 considers treating newly generated, Basic Inventory TRU waste to meet WAC. To estimate impacts due to chronic exposure to VOCs, it was assumed that the total number of drum-equivalents of waste for each major storage site would be present in the facility at the beginning of the operations period. All exposed individuals were assumed to be exposed for the entire 35-year operating period.

Impacts to the noninvolved worker, MEI, and populations are presented in Table F-33. The maximum HI would be 1 x 10^{-3} (noninvolved worker at INEEL) and 4 x 10^{-4} (MEI at LANL), for carbon tetrachloride. No noncarcinogenic effects would be expected for routine releases for No

Table F-33
No Action Alternative 2 Human Health Impacts from Routine Releases of VOCs

Site	Lifetime Cancer Risk	Major Contributor (percent of total)	Maximum HI	VOC with Maximum HI
Hanford East & West				
Noninvolved Worker	1E-7	1,1,2,2-tetrachloroethane (92)	3E-4	Carbon tetrachloride
35-year MEI	3E-8	1,1,2,2-tetrachloroethane (83)	3E-5	Carbon tetrachloride
50-mile Population	4E-4	1,1,2,2-tetrachloroethane (94)	N/A	N/A
INEEL				
Noninvolved Worker	2E-7	1,1,2,2-tetrachloroethane (81)	1E-3	Carbon tetrachloride
35-year MEI	5E-9	1,1,2,2-tetrachloroethane (64)	1E-5	Carbon tetrachloride
50-mile Population	5E-5	1,1,2,2-tetrachloroethane (83)	N/A	N/A
LANL				
Noninvolved Worker	2E-8	1,1,2,2-tetrachloroethane (64)	4E-4	Carbon tetrachloride
35-year MEI	4E-8	1,1,2,2-tetrachloroethane (80)	4E-4	Carbon tetrachloride
50-mile Population	2E-4	1,1,2,2-tetrachloroethane (65)	N/A	N/A
RFETS				
Noninvolved Worker	7E-9	1,1,2,2-tetrachloroethane (70)	1E-4	Carbon tetrachloride
35-year MEI	5E-9	1,1,2,2-tetrachloroethane (85)	4E-5	Carbon tetrachloride
50-mile Population	7E-4	1,1,2,2-tetrachloroethane (65)	N/A	N/A
ORNL				
Noninvolved Worker	1E-8	1,1,2,2-tetrachloroethane (92)	1E-5	Carbon tetrachloride
35-year MEI	3E-8	1,1,2,2-tetrachloroethane (97+)	2E-5	Carbon tetrachloride
50-mile Population	3E-4	1,1,2,2-tetrachloroethane (93)	N/A	N/A
SRS				
Noninvolved Worker	9E-8	1,1,2,2-tetrachloroethane (97)	1E-4	Carbon tetrachloride
35-year MEI	1E-8	1,1,2,2-tetrachloroethane (92)	1E-5	Carbon tetrachloride
50-mile Population	4E-4	1,1,2,2-tetrachloroethane (92)	N/A	N/A

N/A = Not Applicable

Action Alternative 2. An individual probability of cancer incidence of 2×10^{-7} (noninvolved worker at INEEL) and 4×10^{-8} (MEI at LANL) would result from the routine releases of VOCs. The annual risk would be equivalent to 1/35th of the total value. No cancers are expected in the 80-kilometer (50-mile) population (maximum number of cancers would be 7×10^{-4} for RFETS).

VOCs would presumably be emitted from the storage area stack after the waste disposition operations end. Impacts to the population and noninvolved worker would be no greater than that estimated during the operations period. Conservatively assuming constant emission rates, the maximum impact to an MEI exposed for 70 years would be twice that of the 35-year carcinogenic impact to the MEI (see Table F-33) for the operations period. The HI for the 70-year MEI would not be any greater than that of the 35-year MEI exposed during the operations period, because HIs are determined on an annual basis.

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APPENDIX G
FACILITY ACCIDENTS

This appendix describes the methods used to estimate the health consequences that may result from exposure to radioactive materials and hazardous chemicals from postulated facility accident scenarios during (1) treatment at various U.S. Department of Energy (DOE or the Department) facilities, (2) storage of treated waste at these facilities, and (3) disposal of treated waste at the Waste Isolation Pilot Plant (WIPP). The health consequences analyzed include the carcinogenic and noncarcinogenic effects that may result from the release of radionuclides, volatile organic compounds (VOC), and heavy metals.

The potential consequences of an accident depend on both the type of treatment and the treated waste form, which vary by alternative. Waste could be minimally treated to meet the requirements of the planning-basis Waste Acceptance Criteria (WAC) (as under the Proposed Action, Action Alternative 1, and No Action Alternative 2), thermally treated to meet the Resource Conservation and Recovery Act (RCRA) land disposal restrictions (LDR) (as under all of the Action Alternative 2 subalternatives and No Action Alternative 1 subalternatives), or treated by a shred and grout process (as under Action Alternative 3). Descriptions of the waste treatment methods are included in Chapter 2. Generalized characteristics of the final waste forms from each treatment method are as follows:

- *Waste treated to planning-basis WAC* contains a variety of combustible and noncombustible transuranic (TRU) waste materials in various unconsolidated and consolidated forms.

- *Waste treated thermally* is immobilized in a glass-like, noncombustible, uniformly mixed mass and contains no VOCs, which are removed during the treatment process.

- *Waste treated by a shred and grout process* solidifies liquid waste and small pieces of solid TRU waste into a uniform, concrete-like matrix. For the purposes of analysis, it was assumed that this waste would contain the same headspace volume and concentration of VOCs as waste treated to planning-basis WAC.

G.1 TECHNICAL APPROACH

The following sections describe the technical approach used to calculate potential consequences to human health from exposure to radionuclides and hazardous chemicals.

G.1.1 Radionuclide Impacts

The health consequences from acute exposures to radionuclides from accidental releases were calculated. Total effective dose equivalents (TEDE) were calculated and converted to estimates of latent cancer fatalities (LCF) using dose conversion factors recommended by the International Commission on Radiological Protection (ICRP) and endorsed by the National Council on Radiation Protection and Measurements (NCRP) and federal regulatory bodies. For populations, the number of estimated LCFs is reported. For individuals, the estimated probability of an LCF occurring is reported for the maximally exposed individual (MEI), the maximally exposed noninvolved worker (worker who would not directly handle waste), and the maximally exposed involved worker (worker who would directly handle waste).

The nominal values of lifetime cancer risk for low dose or low-dose rate exposure used in this supplemental environmental impact statement (SEIS-II) are 5 x 10^{-4} per person-rem for a population of all ages and 4 x 10^{-4} per person-rem for a working population. These values are based on recommendations of the ICRP (1991) and endorsed by the NCRP (1993). The ICRP concluded that it would be appropriate to use a nominal value of 1 x 10^{-3} per person-rem effective dose for the lifetime risk of fatal cancer for a population of all ages and a nominal value of 8 x 10^{-4} per person-rem for a working population for high dose or high-dose rate exposure (ICRP 1991). The ICRP also recommended a Dose and Dose-Rate Effectiveness Factor (DDREF) of two to convert risk estimates after high dose and high-dose rate exposure to those expected after low dose or low-dose rate exposure. For the purposes of estimating radiological consequences from acute exposures due to accidental releases, analyses in SEIS-II do not include the DDREF if the annual effective dose equivalent in any year is greater than 20 rem. This assumption is applied to dose equivalents from low and high linear energy transfer (LET) radiation and may result in overestimating the number of LCFs from a given radiation dose by a factor of two.

Exposure Pathway, Radionuclide, and Waste-Type Screening

Consequences of accidents involving TRU waste are caused mainly by inhalation intakes during the period of plume passage. Consequences from the external dose pathway were also considered, but were determined to be five or more orders of magnitude less than inhalation impacts; therefore, they are not included in the consequences reported here. The ingestion pathway was not considered in the accident analyses because, in the event of an accident, it was assumed that DOE would take action to mitigate potential consequences from ingestion of crops and animal products raised within the area potentially affected by accident releases. Were they to be considered, radiological consequences from the ingestion pathway would only account for 10 percent of the contact-handled (CH) TRU waste consequences from inhalation and about 15 to 30 percent of the remote-handled (RH) TRU waste consequences from inhalation.

The radionuclides most likely to pose the greatest overall risk in the event of a treatment accident were identified by screening. Screening calculations for the inhalation pathway were determined in a manner similar to that of the external dose pathway in Appendix F. Radionuclide-specific inhalation dose factors (DF) were multiplied by the waste volumes and the radionuclide and hazardous chemical inventories (1995 activity) at each treatment waste consolidation site to create an inhalation screening value. Inhalation DFs were obtained from DOE (1988). Five TRU radionuclides – plutonium-238 (Pu-238), plutonium-239 (Pu-239), plutonium-240 (Pu-240), plutonium-241 (Pu-241), and americium-241 (Am-241) – were found to be the largest dose contributors via the inhalation pathway. The plutonium-239 equivalent curies (PE-Ci) limits in the planning-basis WAC were used as a bounding waste radionuclide inventory for evaluating radiological impacts from waste storage and WIPP disposal operations and accidents.

The number of accident scenarios that potentially involve RH-TRU waste container breaches is limited. The main difference between the RH-TRU waste and CH-TRU waste is the presence of larger amounts of fission and activation products (e.g., cesium-137 [Cs-137]/barium-137m [Ba-137m] and cobalt-60 [Co-60]) in RH-TRU waste. These fission and activation products emit penetrating X- and gamma radiation and are more likely to enter human food chains. However, RH-TRU waste containers are constructed to prevent breaching in the event of a severe consequence accident and, therefore, would be expected to release only small amounts of particulates. Also, RH-TRU waste packages were assumed to be doubly contained in drums and waste canisters.

The average PE-Ci levels of RH-TRU waste are typically less than those of CH-TRU waste. As a result, the radiological consequences of CH-TRU waste accidents will be greater than (and, therefore, will bound) RH-TRU waste accidents.

G.1.2 Hazardous Chemical Consequences

The following sections describe how carcinogenic and noncarcinogenic consequences from VOCs and heavy metals were calculated.

G.1.2.1 Carcinogenic Consequences from Hazardous Chemicals

Carcinogenic consequences from hazardous chemicals are presented as the number of cancers that may occur in an exposed population and as the probability of a cancer occurring in a MEI. Slope factors have been developed by the U.S. Environmental Protection Agency (EPA) to assist in estimating the potential for cancer incidence from a lifetime (estimated to be 70 years) of exposure to a specific hazardous chemical (EPA 1996). Table G-1 presents the slope factors for heavy metals included in SEIS-II. Only cadmium and beryllium were included in SEIS-II analyses for carcinogenic consequences, because slope factors are not available for lead and mercury. Slope factors for VOCs analyzed in SEIS-II are found in Appendix F.

No standard method exists to calculate the carcinogenic risk from an acute (one-time or short-term) intake. SEIS-II analyses used the chronic-exposure slope factors and assumed the total acute intake was averaged over the 70-year lifetime. In practice, then, slope factors were used but specified as the risk per total acute intake.

Table G-1
Carcinogenic Risk Factors for Heavy Metals

Hazardous Chemical	Inhalation Slope Factor [risk per milligram per kilogram per day] [a]	Comment
Cadmium	6.3	Probable human carcinogen. SEIS-II assumes the ingestion slope factor equals the inhalation slope factor.
Beryllium	8.4	Probable human carcinogen.
Lead	Not Available	Probable human carcinogen. Age, health, nutritional state, body burden, and exposure duration influence the absorption, release, and excretion of lead. As a result, development of carcinogenic risk factors using standard methods was not believed to be appropriate.
Mercury	Not Available	Classification as a carcinogen is not possible due to poor database of carcinogenic effects upon which to base a determination.

[a] Inhalation slope factor from Integrated Risk Information System converted risk per microgram per cubic meter to risk per milligram per kilogram per day by assuming a 20 cubic meter per day inhalation rate and an individual body mass of 70 kilograms (EPA 1996).

G.1.2.2 Noncarcinogenic Consequences from Hazardous Chemicals

Two methods were used to estimate the noncarcinogenic consequences from postulated hazardous chemical releases. The first method was to compare the intake estimate to the "equivalent intakes" of the National Institute for Occupational Safety and Health (NIOSH) immediately dangerous to life or health (IDLH) values (NIOSH 1996). The IDLH values, originally developed by NIOSH for emergency response purposes, are air concentrations based on a 30-minute exposure period in which an individual is assumed to inhale 10 cubic meters (353 cubic feet) of contaminated air. The IDLH-equivalent intake level is the quantity of material inhaled during the 30 minutes of exposure at the IDLH concentration. The exposure time of the individuals in the SEIS-II accident scenarios is much shorter than 30 minutes, given the assumptions that involved workers would immediately exit the accident site and that the air concentration is the result of a 1-second release. As a result of the short exposure time, the IDLH-equivalent intake was used as the impact measure reference value rather than the IDLH concentration. IDLH and IDLH-equivalent intake values are presented in Table G-2.

The second method of evaluating noncarcinogenic accident consequences was to calculate and compare the air concentrations to which workers would be exposed to Emergency Response Planning Guidelines (ERPG) developed by the American Industrial Hygiene Association (AIHA). The ERPGs are air concentrations that may be tolerated by an individual for a 60-minute period, defined for three levels of health impacts:

Table G-2
IDLH Values for Facility Accident Noncarcinogenic Consequence Analysis

Hazardous Chemical [a]	IDLH Values [b]	IDLH-Equivalent Intake Values (milligrams)
Benzene (3.25)	500 ppm	16,250
Carbon Tetrachloride (6.39)	200 ppm	12,780
Chlorobenzene (4.68)	1,000 ppm	46,800
Chloroform (4.96)	500 ppm	24,800
1,1-Dichloroethylene (4.03)	Not Determined	----
1,2-Dichloroethane (4.11)	50 ppm	2,060
Ethyl Benzene (4.41)	800 ppm	35,280
Methyl Ethyl Ketone (3.0)	3,000 ppm	90,000
Methylene Chloride (3.53)	2,300 ppm	81,190
1,1,2,2-Tetrachloroethane (7.0)	100 ppm	7,000
Tetrachloroethene (6.89)	150 ppm	10,335
Toluene (3.83)	500 ppm	19,150
Xylene (4.41)	900 ppm	39,690
Beryllium	4 milligrams/cubic meter	40
Cadmium	9 milligrams/cubic meter	90
Lead	100 milligrams/cubic meter	1,000
Mercury	10 milligrams/cubic meter	100

[a] Values in parentheses for VOCs are milligrams per cubic meter per ppm conversion factors for the calculation of the IDLH-equivalent intake.
[b] NIOSH 1996.

- The ERPG-1 air concentration is the “low” health impact level and is defined as the maximum air concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing anything other than mild transient adverse health effects or without perceiving a clearly defined objectionable odor.

- ERPG-2 air concentrations are slightly more hazardous. The ERPG-2 level is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individual’s ability to take protective action.

- ERPG-3 air concentrations indicate a high impact from the exposure. The ERPG-3 level is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening effects. Above ERPG-3 values, an individual may experience or develop a life-threatening effect as a result of a 1-hour exposure.

Analyses using ERPGs are useful in determining if individuals could encounter levels of VOCs that could potentially lead to transient or serious health effects.

The difficulty with using the ERPG approach to evaluate potential consequences from accidental releases is that values have not been defined for all of the hazardous chemicals evaluated in SEIS-II. When no AIHA ERPG value was available, a substitute value was used (Craig et al. 1994). These substitute methods use resources such as time-weighted average (TWA) concentrations and short-term exposure limits (STEL) from the American Conference of Governmental Industrial Hygienists (ACGIH) and IDLHs from NIOSH. Table G-3 presents the ERPG values used in SEIS-II; where no AIHA value was available, the substitute method used is indicated.

The ERPG values were compared to the air concentrations to which an individual would be acutely exposed for the accident scenarios evaluated under treatment, storage, or disposal operations. If the ratio of the air concentration to the ERPG was greater than one, an adverse impact would be expected.

G.1.2.3 Volatile Organic Compound Screening

The 13 VOCs that may be found in TRU waste (see Tables F-1 or F-2 in Appendix F) were screened to determine which were major contributors to carcinogenic and noncarcinogenic impacts. To develop a VOC screening value, VOC impact measures were compared to the VOC container headspace concentration. For example, screening values for carcinogenic impact were determined by multiplying the VOC slope factor by the headspace concentration, while two noncarcinogenic impact screening values were determined by dividing the headspace concentration by the IDLH-equivalent intake and the ERPG-2 values. Those VOCs with the greatest individual impact contribution were included for more detailed accident consequence analyses, with an estimated summed impact of at least 90 percent of the overall VOC impact. The VOCs included in the SEIS-II accident analyses are presented in Sections G.2 (Treatment Accident Scenarios) and G.3 (Storage Accident Scenarios). All 13 VOCs were evaluated for the WIPP disposal accident scenarios (Section G.4).

Table G-3
ERPG Values for Facility Accident Noncarcinogenic Impact Analysis

Chemical Name	ERPG-1 (milligrams/ cubic meter)	ERPG-2 (milligrams/ cubic meter)	ERPG-3 (milligrams/ cubic meter)	Reference
Carbon tetrachloride	130	640	4800	HEHF 1995
Chloroform	500	5000	25,000	HEHF 1995
1,1-dichloroethylene	60 [a]	100 [c]	---	ACGIH 1995
1,2-dichloroethane	120 [a]	200 [c]	200 [d]	ACGIH/NIOSH 1995
Methylene chloride	---	1400	3500	HEHF 1995
Chlorobenzene	140 [a]	230 [c]	4700 [d]	ACGIH/NIOSH 1995
Methyl ethyl ketone	885 [b]	3000 [c]	9000 [d]	ACGIH/NIOSH 1995
1,1,2,2-tetrachloroethane	21 [a]	35 [c]	700 [d]	ACGIH/NIOSH 1995
Toluene	580	770	7700	Craig et al. 1994
Benzene	80	800	3300	HEHF 1995
Ethyl benzene	543 [b]	2200 [c]	3500 [d]	ACGIH/NIOSH 1995
Tetrachloroethene	690	1380	3500	Craig et al. 1994
Xylene	660	880	4400	Craig et al. 1994
Beryllium	0.006 [a]	0.025	0.1	HEHF 1995
Cadmium	0.2	1	10	HEHF 1995
Lead	0.15 [a]	0.25 [c]	100 [d]	ACGIH/NIOSH 1995
Mercury	0.075	0.1	28	Craig et al. 1994

Footnotes indicate the method of calculation of a substitute value: [a] TWA x 3; [b] STEL; [c] TWA x 5; [d] IDLH

G.1.3 Selection of Accident Scenarios

Three scenarios were selected for analysis of treatment and storage accidents: one high-frequency/low-consequence accident, one low-frequency/high-consequence accident, and one natural disaster. These scenarios were selected to offer a wide span of possible accidents, while allowing comparability between alternatives. Though the scenarios change among alternatives, each includes a waste spill, a waste fire or explosion, and an earthquake that would collapse the storage or treatment building. An earthquake was selected as the beyond-design-basis accident for analysis (rather than a plane crash or tornado, for instance), but the risk would be comparable to other beyond-design-basis accidents. The estimated accident frequencies among alternatives and sites were assumed to be identical except for some storage and WIPP disposal accidents involving thermally treated waste, which were lower because of the final waste form. Seismic design guidelines for DOE facilities are based on facility usage categories. For each category, an earthquake hazard level is specified using site-specific seismic hazard data. This process ensures that facilities are designed on a uniform basis for the effects of seismic events, regardless of their locations. A beyond-design-basis earthquake, regardless of accident frequency, must be assumed to defeat all building confinement functions. Buildings are typically constructed to withstand earthquakes. Therefore, the frequency of the beyond-design-basis earthquake scenario was leveled across the country by the assumption that the building would collapse.

Only three scenarios are presented for treatment and storage accidents because no actual facility design or specific facility location has been selected or would be selected as a result of this supplemental environmental impact statement. Additional site-specific National Environmental Policy Act (NEPA) reviews and safety analyses would be conducted before operations of such facilities would be begun.

At WIPP, no future NEPA reviews are planned before a decision is made on whether to begin disposal operations. For that reason, eight accident scenarios were assessed for WIPP disposal accidents. These accidents include both CH-TRU and RH-TRU waste accidents in the Waste Handling Building and underground.

G.2 TREATMENT ACCIDENT SCENARIOS

This section presents the evaluation of the potential consequences of treatment facility accidents for each of three types of treatment: treatment to planning-basis WAC (under the Proposed Action, Action Alternative 1, and No Action Alternative 2); thermal treatment (under the Action Alternative 2 subalternatives and No Action Alternative 1 subalternatives); and treatment by a shred and grout process (under Action Alternative 3). Accident scenarios were evaluated for each of the treatment alternatives at the major CH-TRU waste treatment sites – Hanford Site (Hanford), Idaho National Engineering and Environmental Laboratory (INEEL), Los Alamos National Laboratories (LANL), Rocky Flats Environmental Technology Site (RFETS), Savannah River Site (SRS), and, for Action Alternative 2C only, WIPP. Ninety-five percent of the existing CH-TRU waste would be treated at the first five sites over a longer period of time than other sites under analysis; thus, these five sites would have the greater risk of accidents across all alternatives. Accident scenarios were also evaluated for the RH-TRU waste treatment sites, Oak Ridge National Laboratory (ORNL) and Hanford under all alternatives and INEEL and LANL under the Proposed Action and Action Alternative 1.

Because no VOC sampling has been conducted for RH-TRU waste, CH-TRU waste and RH-TRU waste VOC headspace concentrations were assumed to be the same. Estimated heavy metal concentrations presented in Appendix A are also the same for all metals except lead, where RH-TRU waste quantities were estimated to be about 100 times those of CH-TRU waste quantities. Heavy metal impacts, however, were estimated to be quite low in all cases, even under conservative accident assumptions. (RH-TRU waste heavy metal impacts could be higher than CH-TRU waste heavy metal impacts as a result of the lead content of RH-TRU waste. It is likely, though, that they would be less than the factor of 100 difference in lead content.)

G.2.1 Inventory

The inventory of materials to be treated that could potentially cause human health impacts includes both radionuclides and hazardous chemicals. Hazardous chemicals that may be present in TRU waste include VOCs and heavy metals. Appendix A presents the description of the radionuclide and hazardous chemical inventories in greater detail.

G.2.1.1 Radionuclide Inventory

Screening calculations were performed to determine which radionuclides were significant contributors to dose at each site. The total inventory of each radionuclide (see Appendix A) was multiplied by the radionuclide dose conversion factor for inhalation. These screening values were then summed over all radionuclides and ranked according to their contribution to the total value. Those radionuclides that cumulatively contributed to more than 90 percent of the total screening value were selected for further evaluation in the consequence analyses. Nine radionuclides were selected across all of the sites. The mix and inventory of radionuclides varied from site to site; therefore, in order to provide a reasonable upper limit on the TRU waste radionuclide inventory, the radionuclide content of each drum was assumed to be ten times the overall average concentration of that radionuclide at all DOE sites. Only those radionuclides present at a site were evaluated for that site (see Table G-4).

Table G-4
Radionuclide Activity per Drum for Treatment Accident Analyses [a, b]

Radionuclide	Sites Where Radionuclide is a Major Dose Contributor [c]	CH-TRU Waste (Ci/drum) [d]	RH-TRU Waste (Ci/drum)	CH-TRU/RH-TRU Waste Activity Ratio
Pu-238	H, I, L, O, S, W	27.7	0.11	260
Pu-239	H, I, L, O, R, W	11.2	0.35	32
Pu-240	H, R, O	2.2	0.12	18
Pu-241	O	46.2	3.0	16
Am-241	H, I, L, R, O, S, W	7.5	0.3	25
Np-239	O	6.9E-04	-----	-----
Bi-214	O	2.5E-04	-----	-----
Co-60	I	2.0E-03	0.6	3E-3
Ba-137m	H, L	0.025	11.2	2E-3

[a] Maximum estimated drum activities are ten times the average drum activity of all sites.
[b] Analyses were based on 1995 stored volumes and integrated data base radionuclide inventories (DOE 1994b), also cited in Appendix A.
[c] H = Hanford, I = INEEL, L = LANL, R = RFETS, O = ORNL, S = SRS, W = WIPP. Waste treated at ORNL is all RH-TRU waste.
[d] Includes plutonium residue radionuclide inventories.

G.2.1.2 Hazardous Chemical Inventory

Hazardous chemicals that may be present in TRU waste are VOCs and heavy metals. Six VOCs and four metals were evaluated for potential human health consequences from treatment facility accidents.

Volatile Organic Compounds

VOCs were assumed to be present in all TRU waste entering the treatment facilities; therefore, screening calculations (see Section G.1.2.3) were used to determine those VOCs with the greatest potential consequence from treatment facility accidents (see Table G-5). VOC headspace concentration estimates were made using INEEL and RFETS sampling data from CH-TRU waste and applied to the various waste matrix categories at the different sites (Appendix A). The average concentration across all sites was then calculated and multiplied by 10 to account for the presence of maximum concentrations in individual drums. VOCs were assumed to be present in all waste treated to planning-basis WAC and the shred and grout process; no VOCs, however, would be present in waste after thermal treatment (Action Alternative 2). The presence of a maximum concentration of a specific VOC has no bearing on whether another VOC will be found in the same container at a high concentration. The VOC headspace concentrations assumed to be present for treatment accident analyses are presented in Table G-5.

Table G-5
VOC Headspace Concentrations for Treatment Accident Analyses [a]

VOC	Headspace Concentration (ppmv)
Carbon tetrachloride	1,849
Methylene chloride	6,621
1,1,2,2-tetrachloroethane	3,357
Benzene	76
Tetrachloroethene	71
Xylene	256

[a] VOC concentrations are ten times the overall average VOC concentrations of all sites.

Heavy Metals

The inventory of metals in TRU waste was derived from estimates developed for the WIPP Safety Analysis Report (SAR) (DOE 1995b). The SAR assumed a conservatively high concentration of metals in its analysis of a waste fire, and the waste forms used were the result of treatment to planning-basis WAC. These values were used as the basis for estimating the heavy metal inventories from thermally treated waste and waste that had been treated by the shred and grout process (see Appendix A.5.1). Inventories of heavy metals that were assumed to be uniformly mixed in each drum of TRU waste are presented in Table G-6.

G.2.2 Treatment Accident Analysis

Three accident scenarios were analyzed for each of the three treatment methods: two operational accidents and one natural disaster accident. Operational accidents were chosen to include a high-frequency/low-consequence scenario and a low-frequency/high-consequence scenario. The estimated annual frequencies of occurrence for treatment accidents were taken from source documents and are presented in Table G-7. When a frequency range was identified, the highest value (i.e., the greater probability) was used in the analysis.

Criticality concerns were not addressed by a formal analysis (see the text box on criticality in Chapter 5), because of the low concentration of radionuclides within these waste materials. A criticality event would require the possibility of a chain reaction for neutron generation, which can only occur if fissionable material in a critical geometry becomes available. With the waste materials at hand, there is not enough fissionable material in existence to achieve a critical

Table G-6
Heavy Metal Concentrations in TRU Waste for Treatment Accident Analyses [a]

Hazardous Metals	CH-TRU Waste (kilograms/drum)	RH-TRU Waste (kilograms/drum) [a]
Lead	1.0	97
Beryllium	0.025	0.025
Cadmium	4E-4	4E-4
Mercury	0.43	0.43

[a] Based on information presented in Table A-45. One drum-equivalent contains 0.208 cubic meters of waste.

Table G-7
Annual Frequencies of Occurrence for Treatment Accidents

SEIS-II Accident Scenario	Accident Description	Estimated Frequency
Waste Treated to Planning-Basis WAC		
T1	Waste Spill	1E-2
T2	Waste Drum Fire	1E-4
T3	Earthquake	1E-5 or less
Waste Thermally Treated		
T4	Waste Drum Failure	1E-2
T5	Steam Explosion in Glass Melter	1E-4
T6	Earthquake	1E-5 or less
Waste Treated by Shred and Grout		
T7	Waste Spill	1E-2
T8	Fire in Shredder	1E-4
T9	Earthquake	1E-5 or less

configuration. The criticality question has been previously addressed (DOE 1987) in the context of an environmental impact statement discussing disposal of high-level wastes, TRU wastes, and tank wastes at Hanford. It was concluded that there is no credible basis for a criticality potential. Earlier evaluations (Wallace et al. 1980) have come to the same conclusion.

G.2.2.1 Accidents During Treatment to Planning-Basis WAC

Treatment of TRU waste to planning-basis WAC is the treatment option for the Department's Proposed Action, Action Alternative 1, and the newly generated waste in No Action Alternative 2. Treatment to planning-basis WAC, under each of these alternatives, would be conducted at the site where the waste is currently stored or would be generated. Each site was assumed to have its own on-site waste treatment facility.

Treatment to planning-basis WAC may include some or all of the following waste management unit operations: receiving; monitoring and sampling waste for radioactive and hazardous chemicals; opening of drums or boxes; transfer of contents to a conveyor belt for sorting; cutting large objects to fit the shredder; shredding; transfer of contents to a conveyor belt for sorting; assaying; filling a drum; closing of a drum with a vented lid; labeling; frisking; and certifying that applicable packaging and transportation criteria have been met. Subsequently, the treated and repackaged material would be consolidated at the 10 largest generator-storage sites to await shipment to WIPP, or, in the case of No Action Alternative 2, would remain in storage at the 10 largest generator-storage sites.

As stated above, larger nonmetallic objects might be shredded during the treatment process so that a reduced volume could be stored in a drum. Thus, if shredding is performed, the sorting area might receive two feed streams: one directly from the drum emptying station and the other from the shredder. The sorting process would probably include coarse separation of metals from nonmetals. Magnets on the front end of the sorter would remove ferrous materials from the waste stream, reducing the risk of sparking in the shredder. Nonmetallic materials might be compressed in the final package drum while metallic parts, if not decontaminated and recycled as scrap, might be packaged without further volume reduction. For CH-TRU waste, the facility would probably use gloveboxes with HEPA (high efficiency particulate air) filters around the drum opening station, the conveyor belt, the shredder, and the sorting area. Although numerous options for treatment processes exist (DOE 1995a), no decision on the process has been made.

The following discussion describes treatment accidents involving CH-TRU waste. Accident analyses for RH-TRU waste would be similar to those for CH-TRU waste except that treatment would be performed in a hotcell with an extra HEPA filter in place, thereby reducing the impact by a factor of 1×10^{-3}. This is a conservative estimate of releases for RH-TRU waste releases, because hotcells would be designed with multiple HEPA filters and enhanced structural integrity and thus would likely release far fewer contaminants to the environment. RH-TRU waste-specific concentrations were used in the RH-TRU waste analysis.

Accident Scenario T1 - Waste Spill

A waste spill is an anticipated event during the lifetime of the waste treatment facility, with an estimated annual occurrence frequency of about 0.01. In this accident scenario, a drum about to be filled with TRU waste would be mispositioned, resulting in a spill of dry, sorted waste materials from the conveyor belt onto the operating floor. The spill volume was assumed to be an entire drum volume (DOE 1990).

The airborne fraction of solids and metals was estimated at 1×10^{-3}, with a respirable fraction of 0.1. The respirable fraction would be carried into the ventilation system, for which a filter transmission factor of 1×10^{-3} was assumed. Credit for only one HEPA filter is taken. In the design, it is likely that there would be several levels of HEPA filters, which would decrease the release by several orders of magnitude (i.e., 1×10^{-6} protection factor).

VOCs and other gaseous components previously attached to the material would have largely outgassed into the gloveboxes during the sorting process; still, a gaseous release fraction of 1.0 was assumed. All VOCs were assumed to be respirable and to completely pass through the HEPA filter.

The involved workers, positioned outside of the process enclosure, were assumed to exit the facility immediately and thus would escape impact.

Accident Scenario T2 - Waste Drum Fire

A waste drum fire is not an anticipated event during the lifetime of any of the waste treatment facilities, with an estimated annual occurrence frequency of about 1×10^{-4}. In this accident scenario, a waste drum was postulated to spontaneously erupt into flames as it was opened but before it was emptied onto the conveyor belt for sorting.

It was postulated that 10 percent of the contents of the waste container would be combustible because all sites have average waste combustible fractions ranging from 4 to 12 percent. Of the combustible fraction, which conservatively includes all of the metals, 5×10^{-4} is assumed to become airborne, all of which is respirable. Of the noncombustible fraction (0.9), 6×10^{-3} becomes airborne, 0.01 of which is respirable (DOE 1994a). All of the metal was conservatively assumed to be combustible. A HEPA filter transmission factor of 1×10^{-3} was assumed. All VOCs were assumed to be consumed by the fire.

The involved workers, positioned outside of the process enclosure, were assumed to exit the facility immediately and thus would escape impact.

Accident Scenario T3 - Earthquake

An earthquake was postulated to be a beyond-design-basis natural event, with an estimated annual frequency of 1 x 10^{-5} or less. In this accident scenario, an earthquake would cause the collapse of the waste treatment facility and the loss of electrical power.

The total material at risk for this accident scenario was the contents of 21 drums, half of one day's process inventory. Although some of the volatile gaseous components would have outgassed into the gloveboxes as the material moved through the treatment process facility, an in-facility release fraction of 1.0 was assumed for the volatile components. The solids and metals fraction of the entire in-process inventory becoming airborne was estimated at 1 x 10^{-3}, of which the respirable fraction is assumed to be 0.1 (DOE 1994a). A building removal factor of 0.5 was assumed (PNNL 1996), so that releases from the collapsed facility to the environment would be 0.5 of the airborne portion, based on the above described total in-process inventory. During the events of this scenario, some involved workers would probably be killed during the collapse of the building.

A summary of the parameter values for the three accident scenarios during treatment to planning-basis WAC is shown in Table G-8.

Table G-8
Accident Analysis Parameters for Waste Treatment to Planning-Basis WAC for the Proposed Action, Action Alternative 1, and No Action Alternative 2

SEIS-II Accident Scenario	Accident Description	Number of Drums (N)	Airborne Release Fraction (f_{rel})	Respirable Fraction (f_{resp})	Filter Transmission Factor (f_{HEPA})	Other Factors (R)
T1	Waste Spill	1	0.001 VOCs: 1.0	0.1 VOCs:1.0	0.001 VOCs: 1.0	N/A
T2	Waste Drum Fire (c) combustible (nc) non-combustible	1	5E-4 (c) [a] 6E-3 (nc) [a] VOCs: 0	1.0 (c) 0.01 (nc)	0.001	0.1 (c) 0.9 (nc)
T3	Earthquake	21	0.001 VOCs: 1.0	0.1 VOCs:1.0	N/A	0.5 VOCs: 1.0

[a] (c) = combustible fraction; (nc) = noncombustible fraction

N/A = Not Applicable

G.2.2.2 Accidents During Thermal Treatment

This treatment option applies to all of the subalternatives of Action Alternative 2 and both of the subalternatives of No Action Alternative 1, in which all CH-TRU and RH-TRU waste materials would be thermally treated to meet the LDRs.

The process temperature for thermal treatment would probably range from 2,000 to 3,000 degrees Celsius. At these temperatures, all organic constituents would be disassembled and reoxidized, and thereby converted into mineral (nonorganic) substances. Depending on how much glass frit is added to this process, the resulting material might be predominantly glass or predominantly metal slag. If the glass configuration is selected, contiguous glass logs containing 25 percent by mass of waste oxides (Mishima et al. 1986) can be produced by pouring the melt directly into cylindrical carbon steel canisters, typically of 0.61-meter (2 feet) diameter and 3-meter (10 feet) length. A comprehensive compilation of the different vitrification processes (DOE 1995a) has been published, including vitrification with combustion melting; continuous vitrification; vitrification with electric arc melting; vitrification with electric resistance melter; vitrification with fossil fuel fired melting, induction melting, joule heated melting, or microwave melting; plasma arc furnace; and others.

As the feed stream enters the glass pool, evaporation renders the feed materials dry. In a calcining phase, all organic materials decompose to form oxides and enter the glass pool. The relatively cool blanket of freshly formed oxides and unreacted materials that cover the melt is expected to condense most of the escaping volatile radionuclides and refluxes them to the melt. The resulting gaseous effluent will contain all of the remaining moisture, oxides of nitrogen (NO_X), carbon dioxide (CO_2), and some of the oxides of sulfur (SO_X) in the melter feed and, during infrequent periods of abnormal blanket distribution, up to 5 percent of the cesium (Mishima et al. 1986).

The thermal treatment option involves the following waste management unit operations: cutting larger objects to fit the shredder; shredding, if required; mixing the glass frit into the feed stream; routing the feed stream into the glass melter; pouring the final glass melt into the log forms; letting the logs cool to ambient temperature; packaging; labeling; frisking; and certifying that applicable packaging and transportation criteria are being met. Subsequently, the treated materials would remain at the treatment sites to await shipment to WIPP or, in the case of No Action Alternative 1, would remain indefinitely stored at the treatment sites.

The following discussion describes treatment accidents involving CH-TRU waste. Accident analyses for RH-TRU waste would be similar to those for CH-TRU waste except that treatment would be performed in a hotcell with an extra HEPA filter in place, thereby reducing the impact by a factor of 1×10^{-3}. This is a conservative estimate of RH-TRU waste releases, because hotcells would be designed with multiple HEPA filters and enhanced structural integrity and thus would likely release far fewer contaminants to the environment. RH-TRU waste–specific concentrations were used in the RH-TRU waste analysis.

Accident Scenario T4 - Waste Drum Failure

A waste drum failure is an anticipated event during the lifetime of the thermal treatment facility, with an estimated annual occurrence frequency of about 0.01. In this accident scenario, a waste drum would be breached (i.e., accidentally dropped in handling operations) before encapsulation (DOE 1982). DOE guidance (1994a) indicates that the potential release from vitrified waste would be negligible. For the purpose of this analysis, it was assumed that the rupture would be equivalent to a breach caused by the overpressurization of heated gases in the freespace of a container with cooled, vitrified waste.

In this scenario, a 14-drum equivalent batch of thermally treated waste was assumed to be placed in a container and, through operator error, ruptured by a concussive impact. A fraction (0.035 percent) of the fines surrounding the waste could be expelled during the breach. Of this fraction, 0.1 becomes airborne and 0.7 is respirable. A filter factor of 1×10^{-3} was assumed for the ventilation system. The fines could be made up of radioactive or metal waste. No VOCs were assumed to be released in this scenario because they would have been consumed in the thermal treatment process.

The involved workers, positioned outside of the process enclosure, were assumed to exit the facility immediately and thus would escape impact.

Accident Scenario T5 - Steam Explosion in Glass Melter

A steam explosion in a glass melter is not an anticipated event during the lifetime of the waste treatment facility, with an estimated annual occurrence frequency of about 1×10^{-4}. In this accident scenario, failure of a cooling system or human error would cause water to become entrapped in a space where molten glass is poured, resulting in a steam explosion. Thus, placing liquid water in contact with molten glass causes flashing of trapped water into vapor. This phenomenon might occur in the melter or when the glass melt is poured into the log form. The potential release from vitrified waste is close to negligible; however, for the purpose of this analysis, it was conservatively assumed that thermal stress on reactive substances would apply to this scenario (DOE 1994a).

In a scenario developed by Mishima et al. (1986) that had been evaluated previously (DOE 1982), a steam explosion was postulated to occur in the glass melter that was assumed to contain about 1,000 liters (5.6 drum equivalents) of molten product. The shock of the explosion would fragment the molten glass into a large number of small particles that would scatter throughout the operating area. It was assumed that a mass fraction of approximately 0.01 of the fragmented glass would become airborne, all be respirable, and be carried into the ventilation system, for which a filter factor of 1×10^{-3} was assumed. Although it is uncertain how volatile metals such as hot mercury and lead would be handled in this treatment train, 80 percent of the mercury and 10 percent of the lead were postulated to volatilize in the explosion. All VOCs were assumed to be consumed in the melting process.

Because of the serious nature of the accident, any involved workers present at the time of the accident were assumed to be fatally injured. No other type of consequences were calculated for involved workers.

Accident Scenario T6 - Earthquake

An earthquake was postulated to be a beyond-design-basis natural event, with an estimated annual frequency of 1×10^{-5} or less. In this accident scenario, an earthquake causes the collapse of the waste treatment facility and the loss of electrical power. The total material at risk was postulated to be the contents of 125 drums, which constitutes one half of the day's assumed process inventory. Of the 125 drums, 35 are assumed to be in a molten phase and 90 are assumed to be cooling in their canisters. Although some of the volatile gaseous components would have outgassed into the gloveboxes as the material moved through the treatment process facility, an in-facility release fraction of 1.0 was assumed for the volatile components in 63 of the 125 drums. Although the release would likely be negligible for molten glass, the airborne and respirable fractions were developed for the disturbed molten metal with high surface turbulence (DOE 1994a). The airborne solids fraction of the entire in-process inventory was estimated to be 0.01, of which the respirable fraction was assumed to be 1.0. The cooling fraction (90 drums) is not likely to release any glass solids; however, a fraction (0.035 percent) of the fines surrounding the waste was assumed to be expelled as a result of the earthquake. Of this fraction, 0.1 becomes airborne and 0.7 is respirable.

A building removal factor of 0.5 was assumed (PNNL 1996), so that releases from the collapsed facility to the environment would be 0.5 of the airborne portion. During the events of this scenario, involved workers would probably be killed by falling debris as the building collapses.

A summary of the parameter values for the three accident scenarios during thermal treatment is shown in Table G-9.

Table G-9
Accident Analysis Parameters for Thermal Treatment of Waste for the Action Alternative 2 and No Action Alternative 1 Subalternatives

SEIS-II Accident Scenario	Accident Description	Number of Drums (N)	Airborne Release Fraction (f_{rel})	Respirable Fraction (f_{resp})	Filter Transmission Factor (f_{HEPA})	Other Factors (R)
T4	Waste Drum Failure	14.0	0.1 VOCs: 0	0.7	0.001	3.5E-4
T5	Steam Explosion in Glass Melter	5.6	0.01 VOCs: 0	1.0	0.001	Lead: 0.1 Mercury: 0.8
T6	Earthquake	35 molten 90 cooled [a]	0.01 molten 0.1 cooled	1.0 molten 0.7 cooled	0.5	N/A molten 3.5E-4 cooled

[a] Half of the drums at risk were assumed to be unprocessed and awaiting treatment (still containing VOCs).

N/A = Not Applicable

G.2.2.3 Accidents During Shred and Grout Treatment

This treatment option requires shredding of the waste materials into relatively uniform small pieces, (about 4 centimeters [1.6 inches]) to ensure a reasonable measure of structural integrity of the grout blocks. Any particulates and free liquids in the waste material are immobilized in this process, and any pyrophoric or corrosive characteristics of the TRU waste are eliminated. Compared to the treatment to planning-basis WAC option, the shred and grout process offers the advantage of immobilizing the waste materials and significantly reducing the gas generation rate (as does the thermal treatment process) but without having to apply the energy intensive vitrification process. However, the disadvantage of this option is that the waste volume is significantly increased by adding the grout.

A site-specific NEPA review would be performed for each site if a specific shred and grout process were selected for TRU waste treatment. The accident analyses assumed that the process design is comparable to the commercially available technology which has been successfully demonstrated at Hanford.

The following discussion describes treatment accidents involving CH-TRU waste. Accident analyses for RH-TRU waste would be similar to those for CH-TRU waste except that treatment would be performed in a hotcell with an extra HEPA filter in place, thereby reducing the impact by a factor of 1×10^{-3}. This is a conservative estimate of RH-TRU waste releases because hotcells would be designed with multiple HEPA filters and enhanced structural integrity and thus would likely release far fewer contaminants to the environment. RH-TRU waste–specific concentrations were used in the RH-TRU waste analysis.

Accident Scenario T7 - Waste Spill

A waste spill is an anticipated event during the lifetime of the waste treatment facility, with an estimated annual occurrence frequency of about 0.01. In this accident scenario, a malfunction of the automatic equipment causes a drum about to be filled to be mispositioned, resulting in a spill of wet grouted waste materials onto the operating floor. The involved workers, positioned outside of the process enclosure, were assumed to exit the facility immediately and thus would escape impact.

This scenario is described by parameters for a slurry of 40 percent solids (DOE 1994a). The fraction of solids and metals becoming airborne in this scenario was estimated at 5×10^{-5}, with a respirable fraction of 0.8. The respirable fraction would be carried into the ventilation system, for which a filter factor of 1×10^{-3} was assumed. For VOCs and other gaseous components, a gaseous release fraction of 1.0 was postulated.

Accident Scenario T8 - Fire in the Shredder

In this accident scenario, a fire would be initiated by either an explosion of fine particulates generated by the shredding process or a spontaneous combustion of occluded pyrophoric material exposed to air by the shredder action. Because sparks generated during shredding of metal would be anticipated to cause small fires in the shredder, a fire suppression system was assumed to be in place. Failure of the fire suppression system and subsequent fire involving all of the contents in the shredder was estimated to be an extremely unlikely event, with an estimated annual occurrence frequency of about 1×10^{-4}.

It was postulated that 20 drums of material would be in the shredder or in the hopper and that the fire would spread to include all 20 drums and their contents. Ten percent of the contents of the waste container were assumed to be combustible because all sites have average waste combustibles fractions ranging from 4 to 12 percent. Of the combustible fraction, 5×10^{-4} particulates would become airborne and respirable, while 6×10^{-3} of the noncombustible particulates would become airborne, 0.01 of which would be respirable (DOE 1994a). A filter transmission factor of 1×10^{-3} was assumed. All VOCs were assumed to be consumed by the fire.

The involved workers, positioned outside of the process enclosure, were assumed to exit the facility immediately and thus would escape impact.

Accident Scenario T9 - Earthquake

An earthquake was postulated to be a beyond design-basis-natural event, with an estimated annual frequency of 1×10^{-5} or less. In this accident scenario, an earthquake would cause the collapse of the waste treatment facility and the loss of electrical power. The total material at risk for this accident scenario would be the contents of 66 drums, which constitutes the assumed process inventory, half of which is in the shredder and half is in a slurry form. The shredded fraction of the in-process inventory becoming airborne was estimated at 1×10^{-3}, of which the respirable fraction was assumed to be 0.1. The slurry fraction of the in-process inventory becoming airborne was estimated at 5×10^{-5}, of which the respirable fraction was assumed to be 0.8. A building removal factor of 0.5 was assumed (PNNL 1996), so that releases from the collapsed facility to the environment would be 0.5 of the airborne portion, based on the above described total in-process inventory. All the metals were conservatively assumed to be in the shredder. Although some of the volatile gaseous components would have outgassed into the process enclosures as the material moved through the treatment process facility, a release fraction of 1.0 was assumed for the volatile components. During the events of this scenario, it would be expected that some involved workers would be killed by falling debris.

A summary of the parameter values for the three accident scenarios during shred and grout treatment is shown in Table G-10.

Table G-10
Accident Analysis Parameters for Shred and Grout Treatment of Waste for Action Alternative 3

SEIS-II Accident Scenario	Accident Description	Number of Drums (N)	Airborne Release Fraction (f_{rel})	Respirable Fraction (f_{resp})	Filter Transmission Factor (f_{HEPA})	Other Factors (R)
T7	Waste Spill	1	5E-05 VOCs: 1.0	0.8 VOCs: 1.0	0.001 VOCs: 1.0	N/A
T8	Fire in the Shredder	20	5E-04 (c) [a] 6E-03 (nc) [a] VOCs: 0	1.0 (c) [a] 1E-02 (nc) [a]	0.001	0.1 (c) [a] 0.9 (nc) [a]
T9	Earthquake	66	0.001 (shredder) 5E-5 (slurry) VOCs: 1.0	0.1 (shredder) 0.8 (slurry) VOCs: 1.0	0.5	0.5 (shredder) 0.5 (slurry) VOCs: 1.0

[a] (c) = combustible fraction; (nc) = noncombustible fraction.

N/A = Not Applicable

G.2.2.4 Source Term Analysis

The radionuclide and hazardous chemical source terms were estimated based on Equation G-1 shown below:

$$S = (N)(Q)(R)(f_{rel})(f_{resp})(f_{HEPA}) \qquad \text{(Equation G-1)}$$

where

S = source term (Curies [Ci] or kilograms)

N = number of containers involved

Q = radionuclide or heavy metal inventory of a waste container (from Appendix A)

R = factors accounting for other removal mechanisms, such as fraction of drum(s) at risk or volatilization of metals

f_{rel} = fraction of the contents released from the container(s)

f_{resp} = fraction of the spilled or resuspended contents that are respirable-sized particles

f_{HEPA} = fraction of material that passes through the treatment facility HEPA filters and is released to the environment

The number of waste drums or containers at risk during an accident (N) is specific to each scenario and is easily determined for the higher frequency accidents. In most cases, N equals 1 drum. Special consideration was given to the earthquake analyses because of the catastrophic nature of the disaster and the potentially large inventories at risk. It was necessary to find the alternative-specific site that would process the maximum inventory over the course of the 35-year treatment period. In all cases but Action Alternative 2C, the site with the maximum inventory would be Hanford. In Action Alternative 2C, WIPP would have the highest processing inventory. This inventory was then analyzed on a daily basis over the course of 35 years with assumed operational efficiencies of 75 percent for treatment to planning-basis WAC, 60 percent for thermal treatment, and 50 percent for the shred and grout process. It was then postulated that workers could on average open five drums an hour and add them to the treatment batches. From these assumptions, it could be determined that thermal treatment would require the most process lines and treatment to planning-basis WAC would require the least. Assuming further that two batches were treated each day for thermal and shred and grout treatments, that treatment to planning-basis WAC was continuous, and that an earthquake would only put one batch at a time at risk, it was possible to determine the number of drums in each earthquake scenario.

G.2.3 Exposure Analysis

As noted earlier, the potential inhalation doses would be much greater than the potential external doses, so only the doses from inhalation are presented here. Radiation doses were estimated at each site for radionuclides determined to be important dose contributors. Potential radiation doses to receptors were calculated using Equation G-2, then converted to the estimated number of LCFs in exposed populations or the estimated probability of an LCF for an individual.

$$Dose_{inh} = \sum_i \left(DCF_i \times S_i\right) \times \frac{E}{Q} \times 20 \times 1.0 \text{ x } 10^6 \times \frac{1}{86{,}400} \qquad \text{(Equation G-2)}$$

where

$Dose_{inh}$ = total inhalation dose to a receptor from all radionuclides (rem)

DCF_i = inhalation dose conversion factor for radionuclide i (rem/microcurie [µCi] inhaled)

S_i = source term of radionuclide i released to the environment (Ci)

E/Q = atmospheric dispersion at a point downwind where the receptor is located (seconds/cubic meter)

20 = breathing rate (cubic meters/day)

1.0×10^6 = conversion factor (μCi/Ci).

1/86,400 = conversion factor (day/second).

Dose conversion factors (DCF) were taken from *Internal Dose Conversion Factors for Calculation of Dose to the Public* (DOE 1988) for internal dose rates. Where there was a choice, as in the case of inhalation DCFs, the highest value was used in the analysis.

Atmospheric dispersion (E/Q) values for acute elevated point-source releases were calculated using the GENII computer code (Napier et al. 1988). Near-field dispersion from acute diffuse ground-level releases (following earthquakes) was calculated using computer spreadsheets. E/Q values were calculated for the nearest public access point for the MEI, the maximum population-weighted sector for populations, and the point of highest concentration for the maximally exposed noninvolved worker. The consequences to exposed populations are presented in Table G-11, and the consequences to the MEIs and noninvolved workers are presented in Table G-12. Release heights from all facilities were assumed to be 10 meters (33 feet) from the ground surface for all accidents except the catastrophic earthquakes, where releases were assumed to be at ground level for the MEI and noninvolved worker receptors. The population weighted E/Q for the catastrophic accident was based on a release height of 10 meters (33 feet) because it was slightly more conservative. It was further assumed that the location of the facility did not depend on the treatment technology.

For consequences from an acute, diffuse, ground-level release following an earthquake to the maximally exposed noninvolved worker, the near-field atmospheric dispersion was calculated using a computer spreadsheet. Gaussian plume models sometimes used for such calculations assume a point source release and are not designed for such near-field (< 1 kilometer) calculations. Meteorological data were available only for elevated releases. Use of the Gaussian plume to calculate E/Q for these cases would result in unrealistically conservative (low) atmospheric dispersion, significantly overestimating consequences to the maximally exposed noninvolved worker.

Instead, contaminants were assumed to be released from the earthquake into an area representative of a small building cross-section (in this case, 30 meters [100 feet] wide by 10 meters [33 feet] high). The cross-sectional area of the contaminant plume was assumed to remain unchanged from these dimensions as it passed over the noninvolved worker at a windspeed of 1 meter (3.3 feet) per second. The worker was assumed to be present for the entire period of plume passage, breathing at

Table G-11
Exposed Population Locations, Sizes, and Atmospheric Dispersion Factors for Treatment and Storage Accident Analysis

Site	Population Sector	Population-Weighted E/Q (seconds/cubic meter)	80-kilometer (50-mile) Sector Population
Hanford	SE	1.1	98,865
INEEL	E	0.055	75,162
LANL	ENE	0.48	10,381
RFETS	SE	2.2	180,867
ORNL	E	1.6	214,419
SRS	WNW or NW	0.16	194,597 (WNW), 111,899 (NW)
WIPP	W	0.061	25,629

Table G-12
MEI and Noninvolved Worker Locations and Atmospheric Dispersion Factors for Treatment and Storage Accident Analysis

Site	Facility Location	MEI Location	Ground-Level E/Q [a]	10-meter E/Q [a]	Noninvolved Worker Location [b]	Ground-Level E/Q [a]	10-meter E/Q [a]
Hanford	200 W Area	4,200 meters WSW	6.2E-4	2.8E-4	100 meters S	3.3E-3	1.8E-3
INEEL	RWMC	4,000 meters ENE	2.9E-5	2.3E-5	100 meters S	3.3E-3	4.0E-4
LANL	TA54	500 meters NNE	9.2E-4	6.5E-4	100 meters W	3.3E-3	1.4E-3
RFETS	Center of Site	2,600 meters NNW	1.4E-4	1.3E-4	300 meters N	3.3E-3	6.4E-4
ORNL	Y-12	720 meters NNW	1.6E-3	1.2E-3	100 meters SW	3.3E-3	1.4E-3
SRS	E Area	12,000 meters NNW	1.8E-5	8.9E-6	100 meters WSW	3.3E-3	8.7E-4
WIPP	WIPP	300 meters S	2.7E-3	6.5E-4	300/100 m S (10 m/GL) m	3.3E-3	6.5E-4

[a] Units are in seconds per cubic meter.
[b] Location of the noninvolved worker changes with release height.

a rate of 3.33 x 10^{-4} cubic meters per second (1.18 x 10^{-2} cubic feet per second), the rate for light activity in ICRP 23 (ICRP 1975). The calculated E/Q is 3.3 x 10^{-3} seconds per cubic meter (0.12 seconds per cubic foot), as shown in Table G-12. The source term was determined as discussed in Sections G.2.1.1 and G.2.2.

Exposure to hazardous chemicals and metals was evaluated on both a carcinogenic and noncarcinogenic basis. The total risk summed over all carcinogens was analyzed for both chemicals and metals across populations or individuals. For noncarcinogenic impacts, the intake of a specific hazardous chemical was compared to the IDLH-equivalent intake and the ERPG-2 value.

For hazardous chemicals, both VOCs and heavy metals impacts were evaluated using Equation G-3 below for carcinogens and Equations G-4 and G-5 for noncarcinogens.

$$\text{CarcRisk} = \frac{E}{Q} \times S \times \text{SlopFctr} \times 20 \times \frac{1}{86{,}400} \times \frac{1}{(70 \times 70 \times 365.25)} \qquad \text{(Equation G-3)}$$

where

CarcRisk = Risk of contracting cancer due to exposure to carcinogenic materials

S = Source term or the total release of hazardous chemical (milligrams)

SlopFctr = Carcinogenic slope factor or the cancer risk per unit intake (kilograms-day/milligrams)

20 = breathing rate (cubic meters/day)

1/86,400 = conversion factor (day/second)

70 = mass of reference adult (kilograms)

70 = lifespan of reference individual (years)

365.25 = (days/year)

$$IDLHF = \frac{\left(\frac{E}{Q} \times S \times 20 \times \frac{1}{86{,}400}\right)}{IDLH_{eq}} \qquad \text{(Equation G-4)}$$

where

IDLHF = the comparison of a worker's intake to the level at which a hazardous material is immediately dangerous to life or health if exposed for 30 minutes

$IDLH_{eq}$ = the equivalent amount that a worker would inhale if exposed for the entire 30-minute period at the IDLH level

20 = breathing rate (cubic meters/day)

1/86,400 = conversion factor (days/second)

$$ERPGF = \frac{\left(\frac{E}{Q} \times S\right)}{ERPG\text{-}X} \qquad \text{(Equation G-5)}$$

where

ERPGF = comparison of the air concentration at the worker's location to the level at which emergency response protection guides take effect

ERPG-X = the emergency response protection guide levels where X = 1, 2, or 3, and

where

ERPG-1 = the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing other than mild transient adverse health effects or without perceiving a clearly defined objectionable odor

ERPG-2 = the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individual's ability to take protective action

ERPG-3 = the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to one hour without experiencing or developing life-threatening health effects

Evaluation of the noncarcinogenic chemical impacts results in a value that is compared to 1.0. If the impact value is less than 1.0, then a noncarcinogenic impact is unlikely to occur. If the value is greater than 1.0, then a noncarcinogenic impact may occur and additional investigation or mitigation measures may be necessary.

G.2.4 Consequences of Treatment Accidents

Consequences of treatment accidents were calculated for the exposed off-site population, the MEI, and the maximally exposed noninvolved worker for each of the potentially affected CH-TRU and RH-TRU waste treatment sites (Hanford, INEEL, LANL, RFETS, ORNL, SRS, and, for Action Alternative 2C only, WIPP). Consequences to the maximally exposed involved worker were addressed qualitatively. As noted earlier, inhalation would be the dominant exposure pathway and was considered for consequences from exposure to radionuclides, VOCs, and heavy metals. Acute releases were assumed to be dispersed in one direction, so population impacts were estimated for a single, maximally exposed, 22.5-degree sector (out to 80 kilometers [50 miles]) and not for the entire 80-kilometer (50-mile) region population. Population-weighted atmospheric dispersion values were calculated and used to determine the maximally-impacted sector, considering both the change in air concentration over distance and the population distribution in the sector.

Radiological impacts would be greater than impacts from VOCs or heavy metals for all types of treatment accidents evaluated, due to increased potential for radiation-related LCFs. No cancer incidence would be expected from exposure to hazardous chemicals under any of the alternatives. Some life-threatening toxicological effects could occur to MEIs during the postulated earthquake events.

G.2.4.1 Treatment Accident Consequences for the Proposed Action, Action Alternative 1, and No Action Alternative 2

The consequences of treatment accident scenarios for waste treated to meet the planning-basis WAC are shown in Tables G-13 to G-15 and are discussed below for the population, MEI, noninvolved worker, and involved worker. These accident analyses apply to the Proposed Action, Action Alternative 1, and No Action Alternative 2. Radiological consequences from accidents during treatment to planning-basis WAC would be lower than consequences from the other two types of waste treatment. Potential consequences from hazardous chemicals, both carcinogenic and noncarcinogenic, would be low in all cases.

The potential radiological consequences from RH-TRU waste treatment accident scenarios would be the greatest at the ORNL site for all cases; however, they would be 4 to 5 orders of magnitude less than the consequences from CH-TRU waste treatment accidents.

Table G-13
Radiological Consequences from Treatment Accidents for Waste Treated to the Planning-Basis WAC

Site		Accident Scenario T1 (Waste Spill)		Accident Scenario T2 (Drum Fire)		Accident Scenario T3 (Earthquake)	
CH-TRU Waste	**Population**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**
	Hanford	0.5	2E-4	0.5	3E-4	5,200	3
	INEEL	0.03	1E-5	0.03	1E-5	300	0.1
	LANL	0.2	1E-4	0.2	1E-4	2,200	1
	RFETS	0.5	3E-4	0.6	3E-4	5,700	3
	SRS	0.05	2E-5	0.05	2E-5	500	0.2
	MEI	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**
	Hanford	1E-4	6E-8	1E-4	7E-8	3	1E-3
	INEEL	1E-5	6E-9	1E-5	6E-9	0.2	8E-5
	LANL	3E-4	1E-7	3E-4	1E-7	4	2E-3
	RFETS	3E-5	2E-8	3E-5	2E-8	0.4	2E-4
	SRS	3E-6	1E-9	3E-6	1E-9	0.06	3E-5
	Maximally Exposed Noninvolved Worker						
	Hanford	1E-3	4E-7	1E-3	4E-7	20	9E-3
	INEEL	2E-4	1E-7	3E-4	1E-7	30	0.01
	LANL	7E-4	3E-7	7E-4	3E-7	20	9E-3
	RFETS	2E-4	8E-8	2E-4	8E-8	10	5E-3
	SRS	3E-4	1E-7	3E-4	1E-7	10	6E-3
	Maximally Exposed Involved Worker						
	Any Site	0	0	0	0	See Text	See Text
RH-TRU Waste	**Population**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**
	Hanford	1E-5	5E-9	1E-3	5E-7	0.1	6E-5
	INEEL	5E-7	2E-10	2E-3	1E-6	5E-3	3E-6
	LANL	3E-6	1E-9	4E-6	2E-9	3E-2	2E-5
	ORNL	3E-5	1E-8	5E-3	2E-6	0.3	1E-4
	MEI	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**
	Hanford	3E-9	1E-12	3E-7	1E-10	6E-5	3E-8
	INEEL	2E-10	1E-13	9E-7	5E-10	3E-6	1E-9
	LANL	4E-9	2E-12	5E-9	4E-12	6E-5	3E-8
	ORNL	2E-8	1E-11	3E-6	2E-9	3E-4	1E-7
	Maximally Exposed Noninvolved Worker						
	Hanford	2E-8	9E-12	2E-6	7E-10	5E-4	2E-7
	INEEL	4E-9	2E-12	2E-5	6E-9	4E-4	2E-7
	LANL	1E-8	4E-12	1E-8	5E-12	3E-4	1E-7
	ORNL	3E-8	1E-11	4E-6	2E-9	8E-4	3E-7
	Maximally Exposed Involved Worker						
	Any Site	0	0	0	0	See Text	See Text

Table G-14
Hazardous Chemical Carcinogenic Consequences from Treatment Accidents for Waste Treated to the Planning-Basis WAC

Site	Accident Scenario T1 (Waste Spill) VOCs	Accident Scenario T1 (Waste Spill) Metals	Accident Scenario T2 (Drum Fire) VOCs	Accident Scenario T2 (Drum Fire) Metals	Accident Scenario T3 (Earthquake) VOCs	Accident Scenario T3 (Earthquake) Metals
Population	**Number of Cancers**		**Number of Cancers**		**Number of Cancers**	
Hanford	1E-7	3E-12	0	2E-11	2E-6	3E-8
INEEL	6E-9	2E-13	0	8E-13	1E-7	2E-9
LANL	5E-8	1E-12	0	7E-12	1E-6	1E-8
RFETS	2E-7	6E-12	0	3E-11	5E-6	6E-8
SRS	2E-8	4E-13	0	2E-12	3E-7	5E-9
MEI	**Probability of Cancer**		**Probability of Cancer**		**Probability of Cancer**	
Hanford	3E-11	8E-16	0	4E-15	1E-9	2E-11
INEEL	2E-12	6E-17	0	3E-16	6E-11	8E-13
LANL	7E-11	2E-15	0	9E-15	2E-9	3E-11
RFETS	1E-11	4E-16	0	2E-15	3E-10	4E-12
SRS	9E-13	2E-17	0	1E-16	4E-11	5E-13
Maximally Exposed Noninvolved Worker						
Hanford	2E-10	7E-15	0	4E-14	7E-9	1E-12
INEEL	4E-11	2E-15	0	8E-15	7E-9	1E-12
LANL	1E-10	6E-15	0	3E-14	7E-9	1E-12
RFETS	7E-11	3E-15	0	1E-14	7E-9	1E-12
SRS	9E-11	3E-15	0	2E-14	7E-9	1E-12
Maximally Exposed Involved Worker						
Any Site	0	0	0	0	See Text	See Text

Table G-15
Hazardous Chemical Noncarcinogenic Consequences from Treatment Accidents for Waste Treated to the Planning-Basis WAC

Site	Accident Scenario T1 (Waste Spill)				Accident Scenario T2 (Drum Fire)				Accident Scenario T3 (Earthquake)			
	VOCs		Metals		VOCs		Metals		VOCs		Metals	
	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]/3 [c]	IDLH [a]	ERPG-2 [b]/3 [c]
MEI												
Hanford	3E-8	0.03	3E-11	1E-4	0	0	1E-10	6E-4	7E-7	0.6	6E-7	3/0.2
INEEL	3E-9	2E-3	2E-12	1E-5	0	0	1E-11	5E-5	6E-8	0.05	3E-8	0.1
LANL	7E-8	0.06	6E-11	3E-4	0	0	3E-10	1E-3	2E-6	1/0.5	1E-6	4/0.2
RFETS	1E-8	0.01	1E-11	6E-5	0	0	6E-11	3E-4	3E-7	0.3	1E-7	0.6
SRS	1E-9	9E-4	9E-13	4E-6	0	0	4E-12	2E-5	2E-8	0.02	2E-8	0.08
Maximally Exposed Noninvolved Worker												
Hanford	2E-7	0.2	3E-10	8E-4	0	0	1E-9	4E-3	8E-6	6/0.3	5E-6	20/0.9
INEEL	5E-8	0.04	6E-11	2E-4	0	0	3E-10	9E-4	8E-6	6/0.3	5E-6	20/0.9
LANL	2E-7	0.1	2E-10	6E-4	0	0	1E-9	3E-3	8E-6	6/0.3	5E-6	20/0.9
RFETS	7E-8	0.06	9E-11	3E-4	0	0	5E-10	1E-3	8E-6	6/0.3	5E-6	20/0.9
SRS	1E-7	0.09	1E-10	4E-4	0	0	6E-10	2E-3	8E-6	6/0.3	5E-6	20/0.9
Maximally Exposed Involved Worker												
Any Site	0	0	0	0	0	0	0	0	See Text	See Text	See Text	See Text

[a] The highest ratio of the receptor hazardous chemical intake to the IDLH-equivalent value.
[b] The highest ratio of the receptor hazardous chemical air concentration to the ERPG-2 value.
[c] The highest ERPG-3 ratio is listed for entries where the ERPG-2 ratio is greater than 1.

Impacts to Population

The potential radiological impacts to the populations surrounding any site would be greatest for RFETS (see Table G-13). This is due to a combination of the radionuclides determined to be significant at the RFETS site through screening calculations and the population-weighted dispersion factor for the population east of RFETS. No cancer fatalities would be expected from the operational accident scenarios T1 or T2 (a waste spill or waste drum fire during treatment to planning-basis WAC) at ORNL or any other site. Population consequences of Accident Scenarios T1 and T2 would range from 1 x 10^{-5} to 3 x 10^{-4} LCFs. For Accident Scenario T3 (an earthquake during treatment to planning-basis WAC), consequences would range from 0.1 to 3.0 LCFs at both RFETS and Hanford.

The potential hazardous chemical impacts to the population would be very small. No cancer fatalities would be expected in the populations surrounding the five treatment sites as a result of the hazardous chemical or metal releases from any analyzed accident scenario. Consequences would range from 2 x 10^{-13} to 2 x 10^{-7} cancers for Accident Scenario T1, 8 x 10^{-13} to 3 x 10^{-11} cancers for Accident Scenario T2, and 2 x 10^{-9} to 5 x 10^{-6} cancers for Accident Scenario T3 (see Table G-14).

Impacts to Maximally Exposed Individual

The potential radiological impacts to the MEI would be greatest for LANL for all three accidents scenarios analyzed for treatment to planning-basis WAC (see Table G-13). The magnitude of the consequences is due to a combination of the radionuclides determined to be significant at the LANL site through the screening calculations discussed above and to the MEI air dispersion factor. In Accident Scenario T1 (waste spill), radiation doses to the MEI would range from 3 x 10^{-6} to 3 x 10^{-4} rem TEDE, with a calculated probability of an LCF of 1 x 10^{-9} to 1 x 10^{-7}. In Accident Scenario T2 (waste drum fire), radiation doses to the MEI would range from 3 x 10^{-6} to 3 x 10^{-4} rem TEDE, with a calculated probability of an LCF of 1 x 10^{-9} to 1 x 10^{-7}. In Accident Scenario T3 (earthquake), radiation doses to the MEI would range from 0.06 to 4 rem TEDE, with an associated probability of an LCF of 3 x 10^{-5} to 2 x 10^{-3}.

The potential hazardous chemical consequences would be greatest at LANL for all three accident scenarios, because the MEI air dispersion factor is greater (less dispersion) at LANL than at any of the other sites and because the inventories of hazardous chemicals and metals are assumed to be the same at all of the sites.

Overall, however, the carcinogenic consequences from the accidents would be very small. Carcinogenic consequences would be no greater than a 7 x 10^{-11}, 9 x 10^{-15}, and 2 x 10^{-9} probability of contracting cancer for Accident Scenarios T1, T2, and T3, respectively (see Table G-14). Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for a noncarcinogenic impact. For Accident Scenario T3, the maximum ERPG-2 ratio would be 4.0 at LANL for both mercury and lead (see Table G-15). Therefore, some irreversible impacts may be expected, but no life-threatening effects would occur (the maximum ERPG-3 ratio is 0.2). The maximum IDLH-equivalent ratio for Accident Scenario T3 is 2 x 10^{-6} for the LANL MEI. For Accident Scenario T1, the maximum ERPG-2 ratio for the MEI is 0.06 and the maximum IDLH-equivalent ratio is 7 x 10^{-8}. Finally, for Accident Scenario T2, the maximum ERPG-2 ratio would be

1 x 10^{-3} and the IDLH-equivalent ratio would be 3 x 10^{-10}. Therefore, the only serious impacts would be expected under the beyond-design-basis accident involving the earthquake.

Impacts to Maximally Exposed Noninvolved Worker

The potential radiological impacts to the noninvolved worker would be greatest for Hanford for Accident Scenarios T1 and T2 and at INEEL for Accident Scenario T3 (see Table G-13). The noninvolved worker consequences are driven by the site-specific dispersion factor for this individual and by the radionuclides with the greatest consequence found to be at this site during the screening process. Under Accident Scenario T1 (waste spill), the radiation doses to the noninvolved worker would range from 2 x 10^{-4} to 1 x 10^{-3} rem TEDE, with an associated probability of an LCF of 8 x 10^{-8} to 4 x 10^{-7}. Under Accident Scenario T2 (waste drum fire), the radiation doses to the noninvolved worker would range from 2 x 10^{-4} to 1 x 10^{-3} rem TEDE, with an associated probability of an LCF of 8 x 10^{-8} to 4 x 10^{-7}. Under Accident Scenario T3 (earthquake), the radiation doses to the noninvolved worker would range from 10 to 30 rem TEDE to the noninvolved worker. The potential Accident Scenario T3 consequences to the LANL, INEEL, and Hanford noninvolved worker would be the most serious; the probability of an LCF for the individuals at these sites would range from 9 x 10^{-3} to 0.01.

The potential hazardous chemical consequences to the noninvolved worker would be greatest for the Hanford noninvolved worker for all accident scenarios evaluated (see Table G-14). However, the consequences would be very small. Hazardous chemical carcinogenic consequences would be no greater than a 2 x 10^{-10}, 4 x 10^{-14}, and 7 x 10^{-9} probability of contracting cancer for Accident Scenarios T1, T2, and T3, respectively. Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for a noncarcinogenic impact. For Accident Scenario T3, the maximum ERPG-2 ratio would be 20 at all sites for mercury and lead and 6 for 1,1,2,2-tetrachloroethane (see Table G-15). Some life-threatening effects may be expected (ERPG-3 ratios of 0.3 for 1,1,2,2-tetrachloroethane and 0.9 for beryllium at all sites). The maximum IDLH-equivalent ratio for Accident Scenario T3 would be 8 x 10^{-6} for the noninvolved worker. For Accident Scenario T1, the maximum ERPG-2 ratio for the noninvolved worker is 0.2 and the maximum IDLH-equivalent ratio would be 2 x 10^{-7}. For Accident Scenario T2, the maximum ERPG-2 ratio would be 4 x 10^{-3} and the maximum IDLH-equivalent ratio would be 1 x 10^{-9}. Hanford shows the maximum values for Accident Scenarios T1 and T2. Therefore, the only serious impacts would be expected under the beyond-design-basis earthquake accident scenario.

Impacts to Maximally Exposed Involved Worker

No radiological or chemical impacts to the maximally exposed involved worker would be anticipated from either the waste spill (Accident Scenario T1) or the drum fire (Accident Scenario T2). These accidents are such that involved workers would be able to evacuate immediately or would not be affected by the event. Substantial radiological and chemical consequences would be possible from a beyond-design-basis earthquake (Accident Scenario T3), ranging from workers killed by debris from the collapsing treatment facilities to high external radiation doses from RH-TRU waste treatment and intakes of radionuclides, VOCs, and heavy metals. The involved worker would not be expected to survive the catastrophic earthquake: if not killed by falling debris, the involved worker could inhale high levels of radionuclides or hazardous materials.

G.2.4.2 Treatment Accident Consequences for Action Alternative 2 and No Action Alternative 1

The consequences of thermal treatment accidents are shown in Tables G-16 to G-18 and are discussed below for the population, MEI, maximally exposed noninvolved worker, and maximally exposed involved worker. These accident analyses apply to all of the subalternatives of Action Alternative 2 and No Action Alternative 1. Radiological consequences from thermal treatment accidents would be the highest of the three treatment methods examined. There would be no impacts from VOCs because they are destroyed during the treatment process. Carcinogenic impacts from heavy metals would be low in all cases, but some life-threatening toxicological effects could occur to MEIs during the postulated earthquake events.

The potential radiological consequences from RH-TRU waste treatment accidents would be greatest at the ORNL site for all accident scenarios and receptors; however, they are 4 to 5 orders of magnitude less than the consequences from CH-TRU waste treatment accidents.

Impacts to Population

Under Action Alternatives 2A and 2B, the potential radiation impacts to the populations surrounding any site would be greatest for RFETS and Hanford, respectively (see Table G-16). This is due to a combination of the radionuclides determined to be significant at the sites through screening calculations and to the population-weighted dispersion factor for the population near RFETS and Hanford. No cancer fatalities would be expected from the operational Accident Scenarios T4 or T5 (drum failure or steam explosion) at RFETS or any other facility. Accident Scenario T4 population consequences would range from 5×10^{-5} to 1×10^{-3} LCFs, and Accident Scenario T5 population consequences would range from 8×10^{-3} to 0.2 LCFs. For Accident Scenario T6 (earthquake during thermal treatment of waste), the consequences for all sites would range from 25 to 480 LCFs, with the greatest number of LCFs at RFETS.

Under Action Alternative 2C, where all of the CH-TRU waste would be treated at WIPP, the expected number of LCFs in the population for Accident Scenarios T4, T5, and T6 would be 5×10^{-5}, 9×10^{-3} and 28, respectively.

The potential hazardous chemical and heavy metal consequences to the population would be very small; therefore, no cancers would be expected in the populations surrounding the five treatment sites or at WIPP as a result of releases from any analyzed accident (see Table G-17). Consequences would range from 5×10^{-13} to 2×10^{-11} cancers for Accident Scenario T4, 8×10^{-11} to 3×10^{-9} cancers for Accident Scenario T5, and 3×10^{-7} to 1×10^{-5} cancers for Accident Scenario T6. The RFETS site would have the highest potential cancer incidence (1×10^{-5} cancers) under Action Alternative 2A, the Hanford site would have the highest potential cancer incidence (3×10^{-7} cancers) under Action Alternative 2B, and the WIPP site would have the highest potential cancer incidence (3×10^{-7} cancers) under Action Alternative 2C.

Table G-16
Radiological Consequences from Treatment Accidents for Thermally Treated Waste

Site		Accident Scenario T4 (Drum Failure)		Accident Scenario T5 (Steam Explosion)		Accident Scenario T6 (Earthquake)	
CH-TRU Waste	**Population**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**
	Hanford	2	9E-4	280	0.1	880,000	440
	INEEL	0.1	5E-5	20	8E-3	50,000	25
	LANL	0.7	4E-4	110	0.06	360,000	180
	RFETS	2	1E-3	310	0.2	960,000	480
	SRS	0.2	8E-5	30	0.01	83,000	42
	WIPP	0.1	5E-5	20	9E-3	56,000	28
	MEI	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**
	Hanford	4E-4	2E-7	7E-2	4E-5	500	0.4
	INEEL	4E-5	2E-8	7E-3	3E-6	30	0.01
	LANL	1E-3	5E-7	0.2	8E-5	690	0.6
	RFETS	1E-4	6E-8	0.02	9E-6	60	0.03
	SRS	9E-6	5E-9	1E-3	7E-7	9	5E-3
	WIPP	2E-3	8E-7	0.3	1E-4	2,500	1
	Maximally Exposed Noninvolved Worker						
	Hanford	3E-3	1E-6	0.5	2E-4	3,800	1
	INEEL	9E-4	3E-7	0.1	6E-5	4,300	1
	LANL	2E-3	1E-6	0.4	2E-4	3,600	1
	RFETS	7E-4	3E-7	0.1	4E-5	2,100	1
	SRS	1E-3	4E-7	0.2	7E-5	2,500	1
	WIPP	2E-3	8E-7	0.3	1E-4	4,300	1
	Maximally Exposed Involved Worker						
	Any Site	0	0	0	0	See Text	See Text
Site		**Accident Scenario T4 (Drum Failure)**		**Accident Scenario T5 (Steam Explosion)**		**Accident Scenario T6 (Earthquake)**	
RH-TRU Waste	**Population**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**
	Population						
	Hanford	4E-5	2E-8	6E-3	3E-6	20	0.01
	ORNL	9E-5	4E-8	1E-2	7E-6	40	0.02
	MEI	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**
	Hanford	1E-8	5E-12	2E-6	8E-10	1E-2	5E-6
	ORNL	6E-8	3E-11	1E-5	5E-9	4E-2	2E-5
	Maximally Exposed Noninvolved Worker						
	Hanford	8E-8	3E-11	1E-5	5E-9	8E-2	3E-5
	ORNL	9E-8	4E-11	1E-5	6E-9	0.1	5E-5
	Maximally Exposed Involved Worker						
	Any Site	0	0	0	0	See Text	See Text

Note: Annual doses above 20 rem TEDE to individuals do not include a DDREF (see Section G.1.1).

Table G-17
Hazardous Chemical Carcinogenic Consequences from Treatment Accidents for Thermally Treated Waste

Site	Accident Scenario T4 (Drum Failure)		Accident Scenario T5 (Steam Explosion)		Accident Scenario T6 (Earthquake)	
	VOCs	Metals	VOCs	Metals	VOCs	Metals
Population	**Number of Cancers**		**Number of Cancers**		**Number of Cancers**	
Hanford	0	1E-11	0	2E-9	0	5E-6
INEEL	0	5E-13	0	8E-11	0	3E-7
LANL	0	5E-12	0	7E-10	0	2E-6
RFETS	0	2E-11	0	3E-9	0	1E-5
SRS	0	2E-12	0	2E-10	0	8E-7
WIPP	0	6E-13	0	9E-11	0	3E-7
MEI	**Cancer Incidence**		**Cancer Incidence**		**Cancer Incidence**	
Hanford	0	3E-15	0	4E-13	0	3E-9
INEEL	0	2E-16	0	4E-14	0	1E-10
LANL	0	6E-15	0	1E-12	0	4E-9
RFETS	0	1E-15	0	2E-13	0	7E-10
SRS	0	8E-17	0	1E-14	0	9E-11
WIPP	0	8E-15	0	1E-12	0	2E-7
Maximally Exposed Noninvolved Worker						
Hanford	0	2E-14	0	4E-12	0	2E-8
INEEL	0	5E-15	0	9E-13	0	2E-8
LANL	0	2E-14	0	3E-12	0	2E-8
RFETS	0	9E-15	0	1E-12	0	2E-8
SRS	0	1E-14	0	2E-12	0	2E-8
WIPP	0	1E-14	0	2E-12	0	2E-8
Maximally Exposed Involved Worker						
Any Site	0	0	0	0	See Text	See Text

Table G-18
Hazardous Chemical Noncarcinogenic Consequences from Treatment Accidents for Thermally Treated Waste

Site	Accident Scenario T4 (Drum Failure)				Accident Scenario T5 (Steam Explosion)				Accident Scenario T6 (Earthquake)			
	VOCs		Metals		VOCs		Metals		VOCs		Metals	
	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]/3 [c]
MEI												
Hanford	0	0	1E-10	4E-4	0	0	1E-8	0.05	0	0	1E-4	470/27
INEEL	0	0	8E-12	3E-5	0	0	1E-9	4E-3	0	0	5E-6	22/1
LANL	0	0	2E-10	1E-3	0	0	3E-8	0.1	0	0	2E-4	690/40
RFETS	0	0	4E-11	2E-4	0	0	6E-9	0.03	0	0	2E-5	110/6
SRS	0	0	3E-12	1E-5	0	0	4E-10	2E-3	0	0	3E-6	14/0.8
WIPP	0	0	3E-10	1E-3	0	0	4E-8	0.2	0	0	5E-4	2000/120
Maximally Exposed Noninvolved Worker												
Hanford	0	0	9E-10	3E-3	0	0	1E-7	0.3	0	0	8E-4	2500/150
INEEL	0	0	2E-10	6E-4	0	0	3E-8	0.08	0	0	8E-4	2500/150
LANL	0	0	7E-10	2E-3	0	0	9E-8	0.3	0	0	8E-4	2500/150
RFETS	0	0	3E-10	9E-4	0	0	4E-8	0.1	0	0	8E-4	2500/150
SRS	0	0	4E-10	1E-3	0	0	6E-8	0.2	0	0	8E-4	2500/150
WIPP	0	0	4E-10	1E-3	0	0	6E-8	0.2	0	0	8E-4	2500/150
Maximally Exposed Involved Worker												
Any Site			0	0	0	0	0	0	0	0	See Text	See Text

[a] The highest ratio of the receptor hazardous chemical intake to the IDLH-equivalent value.
[b] The highest ratio of the receptor hazardous chemical air concentration to the ERPG-2 value.
[c] The highest ERPG-3 ratio is listed for entries where the ERPG-2 ratio is greater than 1.

Impacts to Maximally Exposed Individual

Under Action Alternative 2A, the potential radiological consequences for the MEI would be greatest for LANL (see Table G-16). Under Action Alternative 2B, the radiological consequences would be greatest at Hanford. This is a combination of the radionuclides determined to be significant at the sites through screening calculations and the MEI air dispersion factors. In a waste drum failure (Accident Scenario T4), radiation doses to the MEI would range from 9 x 10^{-6} to 1 x 10^{-3} rem TEDE, with an associated probability of an LCF of 5 x 10^{-9} to 5 x 10^{-7}. In a steam explosion (Accident Scenario T5), radiation doses to the MEI would range from 1 x 10^{-3} to 0.2 rem TEDE, with an associated probability of an LCF of 7 x 10^{-7} to 8 x 10^{-5}. In an earthquake (Accident Scenario T6), radiation doses to the MEI would range from 9 to 690 rem TEDE, with an associated probability of an LCF of 5 x 10^{-3} to 0.6.

WIPP would be the only treatment site for CH-TRU waste under Action Alternative 2C. In WIPP WIPP would be the only treatment site for CH-TRU waste under Action Alternative 2C. In Accident Scenario T4, radiation dose to the WIPP MEI would be 2 x 10^{-3} rem TEDE, with an associated probability of an LCF of 8 x 10^{-7}. In Accident Scenario T5, radiation dose to the WIPP MEI would be 0.3 rem TEDE, with an associated probability of an LCF of 1 x 10^{-4}. In Accident Scenario T6, radiation dose to the WIPP MEI would be 2,500 rem TEDE, with an associated probability of an LCF of 1.0.

Under Action Alternatives 2A and 2B, the potential hazardous chemical consequences for the MEI are greatest at LANL and Hanford, respectively, for Accident Scenarios T4, T5, and T6. This is because of the MEI air dispersion factors and because the inventories of hazardous chemicals and metals were assumed to be the same at all the sites. Overall, however, the carcinogenic consequences from the accidents would be very small. Carcinogenic consequences are no greater than a 6 x 10^{-15}, 1 x 10^{-12}, and 4 x 10^{-9} probability of contracting cancer for Accident Scenarios T4, T5, and T6, respectively (see Table G-17). Under Action Alternative 2C, all the consequences are to the WIPP MEI, but would be very low. The probability of contracting cancer from Accident Scenarios T4, T5, and T6 would be 8 x 10^{-15}, 1 x 10^{-12}, and 2 x 10^{-7}, respectively.

Ratios of 1.0 or greater would indicate the potential for a noncarcinogenic impact. For Accident Scenario T6 under Action Alternative 2A, the maximum ERPG-2 ratio was estimated to be 690 at LANL for both mercury and lead and 160 for beryllium (see Table G-18). The maximum ERPG-3 ratio is 40 for beryllium and 2 for mercury and lead; thus, these ratios indicate that life-threatening effects may be expected. The maximum IDLH-equivalent ratio for Accident Scenario T6 would be 2 x 10^{-4} for the LANL MEI. For Accident Scenario T4, the maximum ERPG-2 ratio for the MEI would be 1 x 10^{-3} and the maximum IDLH-equivalent ratio would be 2 x 10^{-10}, and for Accident Scenario T5, the maximum ERPG-2 ratio would be 0.1 and the IDLH-equivalent ratio would be 3 x 10^{-8}. Under Accident Scenario 2B, the maximum consequences would be at the Hanford site, but would be less than those described for Accident Scenario 2A. Therefore, for both Accident Scenarios 2A and 2B, the only serious consequences would be expected under the beyond-design-basis accident. The same holds true for Action Alternative 2C where all the consequences would be at WIPP. For Accident Scenario T4, the maximum ERPG-2 ratio for the WIPP MEI would be 1 x 10^{-3} and the maximum IDLH-equivalent ratio would be 3 x 10^{-10}. For Accident Scenario T5, the maximum ERPG-2 ratio would be 0.2 and the IDLH-equivalent ratio would be 4 x 10^{-8}. For Accident Scenario T6, the maximum ERPG-2 ratio is 2,000 for both mercury and lead and 50 for beryllium, while the maximum ERPG-3 ratio is 120 for beryllium and 7 and 5 for mercury and lead, respectively. Thus, life-threatening effects would be expected to the MEI for Accident Scenario T6 at WIPP.

Impacts to Maximally Exposed Noninvolved Worker

Under Action Alternatives 2A and 2B, the potential radiological consequences for the noninvolved worker would be greatest at Hanford for Accident Scenarios T4 and T5, and would be greatest at INEEL for accident Scenario T6 (see Table G-16). The noninvolved worker consequences are driven by the site-specific dispersion factor for this individual and by the site-specific mix of radionuclides chosen for the evaluation. Under Accident Scenario T4, the radiation doses to the noninvolved worker would range from 7×10^{-4} to 3×10^{-3} rem TEDE, with an associated probability of an LCF of 3×10^{-7} to 1×10^{-6}. Under Accident Scenario T5, the radiation doses to the noninvolved worker would range from 0.1 to 0.5 rem TEDE, with a calculated probability of an LCF of 4×10^{-5} to 2×10^{-4}. Under Accident Scenario T6, the radiation doses to the noninvolved worker would range from 2,100 to 4,300 rem TEDE, with calculated probabilities of an LCF of 1.

Under Action Alternative 2C, all the consequences would be to the WIPP noninvolved worker. Under Accident Scenario T4, the radiation dose to the noninvolved worker would be 2×10^{-3} rem TEDE, with an associated probability of an LCF of 8×10^{-7} (see Table G-16). Under Accident Scenario T5, the radiation dose to the noninvolved worker would be 0.3 rem TEDE, with an associated probability of an LCF of 1×10^{-4}. Under Accident Scenario T6, the radiation dose to the noninvolved worker would be 4,300 rem TEDE. The calculated probability of an LCF for the WIPP noninvolved worker is 1.

Under Action Alternatives 2A and 2B, the potential hazardous chemical consequences would be greatest for the Hanford noninvolved worker for Accident Scenarios T4 and T5. The consequences for the noninvolved worker are the same across all sites for Accident Scenario T6. Hazardous chemical carcinogenic consequences would be no greater than a 2×10^{-14}, 4×10^{-12}, and 2×10^{-8} probability of contracting cancer for Accidents T4, T5, and T6, respectively (see Table G-17).

Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for the noncarcinogenic impact (see Table G-18). For Accident Scenario T6, the maximum ERPG-2 ratio is equal to 2,500 at all sites for both mercury and lead. The ERPG-3 ratios are 150, 9, and 6 for beryllium, mercury, and lead, respectively; therefore, these ratios indicate that life-threatening effects would be expected for Accident Scenario T6. The maximum IDLH-equivalent ratio for Accident Scenario T6 is 8×10^{-4} for the noninvolved worker. For Accident Scenario T4, the maximum ERPG-2 ratio for the noninvolved worker is 3×10^{-3} with a maximum IDLH-equivalent ratio of 9×10^{-10} and for Accident Scenario T5, the maximum ERPG-2 ratio for the noninvolved worker is 0.3 and the maximum IDLH-equivalent ratio is 1×10^{-7}. Therefore, the only serious consequences would be expected under the beyond-design-basis accident.

Under Action Alternative 2C, the potential hazardous chemical consequences would be for the WIPP noninvolved worker for all accidents evaluated. Hazardous chemical carcinogenic consequences would be no greater than a 1×10^{-14}, 2×10^{-12}, and 2×10^{-8} probability of contracting cancer for accidents T4, T5, and T6, respectively (see Table G-17). Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for the noncarcinogenic consequence. For Accident

Scenario T6, the maximum ERPG-2 ratio is equal to 2,500 for both mercury and lead and the maximum ERPG-3 ratio is 150 for the WIPP site (see Table G-18). Therefore, these ratios indicate that life-threatening effects would be expected for Accident Scenario T6. The maximum IDLH-equivalent ratio for Accident Scenario T6 would be 8 x 10^{-4} for the WIPP noninvolved worker. For Accident Scenario T4, the maximum ERPG-2 ratio for the noninvolved worker would be 1 x 10^{-3} with a maximum IDLH-equivalent ratio of 4 x 10^{-10}. For Accident Scenario T5, the maximum ERPG-2 ratio for the noninvolved worker would be 0.2 and the maximum IDLH-equivalent ratio would be 6 x 10^{-8}. Therefore, the only serious consequences would be expected under the beyond-design-basis accident.

Impacts to Maximally Exposed Involved Worker

No impacts to the maximally exposed involved worker would be anticipated from the waste spill (Accident Scenario T4). This accident is such that involved workers would be able to evacuate immediately or would not be affected by the event. The nature of the steam explosion (Accident Scenario T5) is such that an involved worker would be killed. Substantial consequences would be possible from a beyond-design-basis earthquake (Accident Scenario T6), ranging from workers killed by debris from the collapsing treatment facilities to high external radiation doses from RH-TRU waste being treated and intakes of radionuclides, VOCs, and heavy metals. If not killed by falling debris, the involved worker could inhale high levels of radionuclides or hazardous chemicals.

G.2.4.3 Treatment Accident Consequences for Action Alternative 3

The accident consequences of waste treatment by a shred and grout process are shown in Tables G-19 to G-21 and are discussed below for the population, MEI, maximally exposed noninvolved worker, and maximally exposed involved worker. These accident analyses apply only to Action Alternative 3. Radiological consequences from shred and grout treatment accidents are somewhat higher than those for treatment to planning-basis WAC accidents and substantially lower than thermal treatment accidents. Carcinogenic consequences from hazardous chemicals are low in all cases, but some life-threatening toxicological effects could occur in maximally exposed individuals during the postulated earthquake events.

The potential radiological consequences from RH-TRU waste treatment accidents are greatest at the ORNL site for all accident scenarios and receptors; however, they are 4 to 5 orders of magnitude less than the consequences from CH-TRU waste treatment accidents.

Impacts to Population

The potential radiological impacts to the populations surrounding any site would be greatest for RFETS (see Table G-19). This is due to a combination of the radionuclides determined to be significant at the RFETS site through screening calculations and the population-weighted dispersion factor for the population east of RFETS. No cancer fatalities would be expected from operational accident T7; population consequences from the waste spill would range from 6 x 10^{-6} to 1 x 10^{-4} LCFs. No LCFs would be expected under operational accident T8, a fire in the shredder; population consequences would range from 3 x 10^{-4} to 6 x 10^{-3} LCFs. Because of the severity of

Table G-19
Radiological Consequences from Treatment Accidents for Waste Treatment by Shred and Grout

Site		Accident Scenario T7 (Waste Spill)		Accident Scenario T8 (Fire in the Shredder)		Accident Scenario T9 (Earthquake)	
CH-TRU Waste	**Population**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**
	Hanford	0.2	1E-4	10	5E-3	12,000	6
	INEEL	0.01	6E-6	0.6	3E-4	660	0.3
	LANL	0.08	4E-5	4	2E-3	4,700	2
	RFETS	0.2	1E-4	10	6E-3	13,000	6
	SRS	0.02	9E-6	1	5E-4	1,100	0.5
	MEI	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**
	Hanford	5E-5	3E-8	3E-3	1E-6	6.0	3E-3
	INEEL	5E-6	2E-9	2E-4	1E-7	0.3	2E-4
	LANL	1E-4	6E-8	6E-3	3E-6	9.0	5E-3
	RFETS	1E-5	6E-9	7E-4	3E-7	0.8	4E-4
	SRS	1E-6	5E-10	5E-5	3E-8	0.1	6E-5
	Maximally Exposed Noninvolved Worker						
	Hanford	4E-4	2E-7	0.02	8E-6	50	0.02
	INEEL	1E-4	4E-8	5E-3	2E-6	60	0.02
	LANL	3E-4	1E-7	0.01	6E-6	50	0.02
	RFETS	8E-5	3E-8	4E-3	2E-6	30	0.01
	SRS	1E-4	5E-8	6E-3	3E-6	30	0.01
	Maximally Exposed Involved Worker						
	Any Site	0	0	0	0	See Text	See Text
RH-TRU Waste	**Population**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**
	Hanford	4E-6	2E-9	2E-4	1E-7	0.3	1E-4
	ORNL	1E-5	5E-9	5E-4	3E-7	0.6	3E-4
	MEI	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**
	Hanford	1E-9	6E-13	6E-8	3E-11	1E-4	7E-8
	ORNL	7E-9	4E-12	4E-7	2E-10	6E-4	3E-7
	Maximally Exposed Noninvolved Worker						
	Hanford	9E-9	3E-12	4E-7	2E-10	1E-3	4E-7
	ORNL	1E-8	4E-12	5E-7	2E-10	2E-3	7E-7
	Maximally Exposed Involved Worker						
	Any Site	0	0	0	0	See Text	See Text

the earthquake scenario (Accident Scenario T9), Hanford, LANL, and RFETS would have some LCFs. The greatest number of LCFs would be 6 at RFETS or Hanford; for all sites, Accident Scenario T9 consequences could range from 0.3 to 6 LCFs.

The potential hazardous chemical impacts to the population would be very small. No cancers would be expected in the populations surrounding the five treatment sites as a result of the hazardous chemical or metal releases from any analyzed accident. Consequences would range from 6×10^{-14} to 2×10^{-7} cancers for Accident Scenario T7, 2×10^{-11} to 6×10^{-10} cancers for Accident Scenario T8, and 5×10^{-9} to 1×10^{-5} cancers for Accident Scenario T9 (see Table G-20).

Impacts to Maximally Exposed Individual

The potential radiological consequences for the MEI would be greatest for LANL for all the accident scenarios analyzed (see Table G-19), due to the combination of radionuclides determined

Table G-20
Hazardous Chemical Carcinogenic Consequences from Treatment Accidents for Waste Treatment by Shred and Grout

Site	Accident Scenario T7 (Waste Spill)		Accident Scenario T8 (Fire in the Shredder)		Accident Scenario T9 (Earthquake)	
	VOCs	Metals	VOCs	Metals	VOCs	Metals
Population	**Number of Cancers**		**Number of Cancers**		**Number of Cancers**	
Hanford	1E-7	1E-12	0	3E-10	7E-6	1E-7
INEEL	6E-9	6E-14	0	2E-11	4E-7	5E-9
LANL	5E-8	5E-13	0	1E-10	3E-6	4E-8
RFETS	2E-7	2E-12	0	6E-10	1E-5	2E-7
SRS	2E-8	2E-13	0	4E-11	1E-6	1E-8
MEI	**Probability of Cancer Incidence**		**Probability of Cancer Incidence**		**Probability of Cancer Incidence**	
Hanford	3E-11	3E-16	0	8E-14	4E-9	6E-11
INEEL	2E-12	3E-17	0	6E-15	2E-10	3E-12
LANL	7E-11	7E-16	0	2E-13	6E-9	8E-11
RFETS	1E-11	1E-17	0	4E-14	9E-10	1E-11
SRS	9E-13	1E-17	0	2E-15	1E-10	2E-12
Maximally Exposed Noninvolved Worker						
Hanford	2E-10	3E-15	0	7E-13	2E-8	4E-10
INEEL	4E-11	6E-16	0	2E-13	2E-8	4E-10
LANL	1E-10	2E-15	0	6E-13	2E-8	4E-10
RFETS	7E-11	1E-15	0	3E-13	2E-8	4E-10
SRS	9E-11	1E-15	0	3E-13	2E-8	4E-10
Maximally Exposed Involved Worker						
Any Site	0	0	0	0	See Text	See Text

Table G-21
Hazardous Chemical Noncarcinogenic Consequences from Treatment Accidents for Waste Treatment by Shred and Grout

Site	Accident Scenario T7 (Waste Spill)				Accident Scenario T8 (Fire in the Shredder)				Accident Scenario T9 (Earthquake)			
	VOCs		Metals		VOCs		Metals		VOCs		Metals	
MEI	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]	IDLH [a]	ERPG-2 [b]/3 [c]	IDLH [a]	ERPG-2 [b]/3 [c]
Hanford	3E-8	0.03	1E-11	5E-5	0	0	3E-9	0.01	5E-6	4/0.2	2E-6	9/0.5
INEEL	3E-9	2E-3	9E-13	4E-6	0	0	2E-10	1E-3	2E-7	0.2	1E-7	0.4
LANL	7E-8	0.06	3E-11	1E-4	0	0	6E-9	0.03	7E-6	6/0.3	3E-6	10/0.8
RFETS	1E-8	0.01	5E-12	2E-5	0	0	1E-9	6E-3	1E-6	0.9	5E-7	2/0.1
SRS	1E-9	9E-4	4E-13	2E-6	0	0	9E-11	4E-4	1E-7	0.1	6E-8	0.3
Maximally Exposed Noninvolved Worker												
Hanford	2E-7	0.2	1E-10	3E-4	0	0	3E-7	0.08	3E-5	20/1	2E-5	50/3
INEEL	5E-8	0.04	2E-11	7E-5	0	0	6E-9	0.02	3E-5	20/1	2E-5	50/3
LANL	2E-7	0.1	8E-11	2E-4	0	0	2E-8	0.06	3E-5	20/1	2E-5	50/3
RFETS	7E-8	0.06	4E-11	1E-4	0	0	9E-9	0.03	3E-5	20/1	2E-5	50/3
SRS	1E-7	0.09	5E-11	1E-4	0	0	1E-8	0.04	3E-5	20/1	2E-5	50/3
Maximally Exposed Involved Worker												
Any Site			N/A	N/A	N/A	N/A	N/A	N/A	See Text	See Text	See Text	See Text

[a] The highest ratio of the receptor hazardous chemical intake to the IDLH-equivalent value.
[b] The highest ratio of the receptor hazardous chemical air concentration to the ERPG-2 value.
[c] The highest ERPG-3 ratio is listed for entries where the ERPG-2 ratio is greater than 1.

N/A = Not Applicable

to be significant at the LANL site through screening calculations and the MEI air dispersion factor. In Accident Scenario T7, radiation doses to the MEI would range from 1 x 10^{-6} to 1 x 10^{-4} rem TEDE, with an associated probability of an LCF of 5 x 10^{-10} to 6 x 10^{-8}. In Accident Scenario T8, radiation doses to the MEI would range from 5 x 10^{-5} to 6 x 10^{-3} rem TEDE, with an associated probability of an LCF of 3 x 10^{-8} to 3 x 10^{-6}. In Accident Scenario T9, radiation doses to the MEI would range from 0.1 to 9 rem TEDE with an associated probability of an LCF of 6 x 10^{-5} to 5 x 10^{-3}.

The potential hazardous chemical consequences would be greatest at LANL for Accident Scenarios T7, T8, and T9, because the MEI air dispersion factor would be larger at LANL than at any of the other sites and the inventories of hazardous chemicals and metals are assumed to be the same at all the sites. Overall, however, the carcinogenic consequences from the accidents would be very small, with a 7 x 10^{-11}, 2 x 10^{-13}, and 6 x 10^{-9} probability of contracting cancer for Accidents Scenarios T7, T8, and T9, respectively (see Table G-20). Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for the noncarcinogenic consequence. For Accident Scenario T9, the maximum ERPG-2 ratio would be 10 at LANL for both mercury and lead (see Table G-21). Therefore, some irreversible consequences would be expected, but no life-threatening effects would occur (maximum ERPG-3 ratio of 0.8). The maximum IDLH-equivalent ratio for Accident Scenario T9 would be 3 x 10^{-6} for the LANL MEI. For Accident Scenario T7, the maximum ERPG-2 ratio for the MEI would be 0.06 and the maximum IDLH-equivalent ratio would be 7 x 10^{-8}. For Accident Scenario T8, the maximum ERPG-2 ratio would be 0.03 and the IDLH-equivalent ratio would be 6 x 10^{-9}. Therefore, the only serious consequences would be expected under the beyond-design-basis accident.

Impacts to Maximally Exposed Noninvolved Worker

The potential radiological consequences would be greatest for Accident Scenarios T7 and T8 for the Hanford maximally exposed noninvolved worker and would be highest for the INEEL noninvolved worker for Accident Scenario T9 (see Table G-19). The noninvolved worker consequences are driven by the site-specific dispersion factor for this individual and by the site-specific mix of radionuclides chosen for the evaluation. Under Accident Scenario T7, the radiation doses to the maximally exposed noninvolved worker would range from 8 x 10^{-5} to 4 x 10^{-4} rem TEDE, with an associated probability of an LCF of 3 x 10^{-8} to 2 x 10^{-7}. Under Accident Scenario T8, the radiation doses to the noninvolved worker would range from 4 x 10^{-3} to 0.02 rem TEDE, with an associated probability of an LCF of 2 x 10^{-6} to 8 x 10^{-6}. Under Accident Scenario T9, the radiation doses to the noninvolved worker would range from 30 to 60 rem TEDE to the noninvolved worker. The potential Accident Scenario T9 consequences for the LANL, ORNL, and Hanford noninvolved worker would be the most serious. The probability of an LCF for the individuals at these sites would range from 0.01 to 0.02.

The potential hazardous chemical consequences would be greatest for the Hanford maximally exposed noninvolved worker for all accidents evaluated (see Table G-20). Hazardous chemical carcinogenic consequences would be no greater than a 2 x 10^{-10}, 7 x 10^{-13}, and 2 x 10^{-8} probability of contracting cancer for Accident Scenarios T7, T8, and T9, respectively. Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for the noncarcinogenic consequence. For Accident Scenario T9, the maximum ERPG-2 ratio would be equal to 50 at all sites for both mercury and lead (see Table G-21). Therefore, some irreversible consequences would be expected as well as life-threatening effects (ERPG-3 ratios of 3 at all sites). The maximum IDLH-equivalent ratio for Accident Scenario T9 would be 3×10^{-5} for the noninvolved worker. For Accident Scenario T7, the maximum ERPG-2 ratio for the noninvolved worker would be 0.2 with a maximum IDLH-equivalent ratio of 2×10^{-7}. For Accident Scenario T8, the maximum ERPG-2 ratio for the noninvolved worker would be 0.08 and the maximum IDLH-equivalent ratio would be 3×10^{-7}. Therefore, the only serious impacts would be expected under the beyond-design-basis accident.

Impacts to Maximally Exposed Involved Worker

No consequences for the maximally exposed involved worker would be anticipated from either the waste spill (Accident Scenario T7) or the fire in the shredder (Accident Scenario T8) (see Table G-19). These accidents are such that involved workers would be able to evacuate immediately or would not be affected by the event. Substantial consequences would be possible from a beyond-design-basis accident, including workers killed by debris from the collapsing treatment facilities to high external radiation doses from RH-TRU waste being treated and intakes of radionuclides, VOCs, and heavy metals.

G.3 STORAGE ACCIDENT SCENARIOS

The storage accident analyses were conducted to evaluate the potential accident consequences of storing TRU waste. Generic CH-TRU and RH-TRU waste storage facilities were assumed for all waste forms under all alternatives because none of the facilities have yet been designed or constructed. Site-specific NEPA evaluations and safety assessments would be considered prior to facility construction and operation.

Under Action Alternative 1; Action Alternatives 2A, 2B, and 2C; Action Alternative 3; and No Action Alternative 2, waste storage activities would include moving the waste containers into storage, monitoring the waste for contamination on the outside of the containers, and relocating the waste while in storage to support monitoring and maintenance activities. Monitoring and maintenance activities would require workers to be near stored waste: monitoring activities would include visual inspections and swiping the outside of containers for contamination, and building maintenance activities would include maintaining roofs and preventing or removing build-up of combustibles at locations that could endanger the stored TRU waste. Accidents assessed during lag or indefinite storage included waste container breaches, a waste container fire, and a catastrophic event. All three accidents were evaluated for CH-TRU waste at each storage location under each of these alternatives.

In addition to the above activities, CH-TRU waste would be overpacked at 20-year intervals under No Action Alternatives 1A and 1B. No accidents were specifically evaluated for overpacking operations because consequences would be bounded by other storage accident analyses. For example, consequences for the maximally exposed involved worker would probably be lower than under other alternatives because this worker was assumed to wear a HEPA-filtered mask and, therefore, would not inhale particulate radionuclides in the event of an overpacking accident.

The accident analyses concentrated on the consequences of events involving CH-TRU waste. RH-TRU waste canisters were assumed to be substantially more robust (cylinders with 6-millimeter [0.24-inch] thick carbon-steel walls) than CH-TRU waste containers because of both the greater handling hazard (high external radiation dose rates) of RH-TRU waste and the reduced likelihood that the canister would be breached during a minor consequence event. RH-TRU waste storage facilities were assumed to withstand severe conditions introduced by humans or a natural disaster. It was assumed there would be remote monitoring of the RH-TRU waste and that there would be no overpacking of RH-TRU waste canisters under No Action Alternatives 1A and 1B. Therefore, except for the beyond-design-basis natural disaster, no conditions were believed to exist that would lead to the environmental release of RH-TRU waste during storage operations. The most severe accidental exposure involving RH-TRU waste during storage would be that of an involved worker who remained for an extended period of time in an elevated radiation area near RH-TRU waste canisters.

Under the Proposed Action only RH-TRU waste would be stored. The RH-TRU waste storage facilities would be constructed to maintain waste containment during all but the most severe circumstances. The current RH-TRU waste containers are designed to result in only a minor breach during a significant consequence event. It was assumed that, at a minimum, a similar design would be used at Hanford and ORNL. Site-specific NEPA review and safety analyses conducted prior to facility construction and operation would require detailed accident risk calculations for the storage facility.

Accident Scenario S1, a drum puncture and lid failure, would be a relatively high-frequency incident. The estimated annual frequency of occurrence would be 1×10^{-2} to 1×10^{-4}. This estimated frequency, taken from DOE (1997), was developed for the handling of waste containers during WIPP disposal operations. Storage operations would require handling of waste during the year it enters storage, the year it leaves storage, and on an infrequent basis during the monitoring phase. Therefore, for waste containers that are in the storage facility for more than two years, the average frequency of occurrence would be expected to decrease somewhat over time. Accident Scenario S2, a drum fire, is not expected to occur, but was evaluated to address public concerns. The WIPP disposal operations frequency of occurrence estimate for a drum fire is 1×10^{-4} to 1×10^{-6} for planning-basis WAC waste for drums with less than 8 PE-Ci and less than 1×10^{-6} for drums with more than 8 PE-Ci (DOE 1997). This same estimate is applied to planning-basis WAC waste in interim storage. The accident is not applicable to thermally treated waste, which contains no combustible materials, and is likely not applicable to grouted waste, which has limited combustibility.

Accident Scenario S3, a beyond-design-basis earthquake, has an estimated annual frequency of occurrence of 1×10^{-5} or less. This accident frequency does not vary by waste treatment level.

G.3.1 Inventory

The inventory of materials in stored waste that could potentially cause human health consequences includes both radionuclides and hazardous chemicals. One purpose in developing the planning-basis WAC (DOE 1996b) was to ensure that all DOE sites package TRU waste to meet a minimum standard that assures the safe transport, handling, and disposal of TRU waste. Some key planning-basis WAC requirements that limit the inventory of high-risk constituents packaged in waste containers and, thereby, reduce the accident consequences are presented in Table G-22.

Accident consequences were estimated using bounding container inventories of radioactive and hazardous chemicals, when these inventories were defined by the planning-basis WAC. For some VOCs and all the heavy metals, container limits were not defined. In such cases, the container inventories were estimated as described below.

Table G-22
Planning-Basis WAC Requirements that Reduce Accident Risk

Requirement	Comment
Liquids- Waste shall contain as little residual liquid as is reasonably achievable.	This requirement reduces the risk of exposure to hazardous constituents in the event of a container breach and reduces the uncertainty associated with the long-term performance of the WIPP facility.
Pyrophoric materials - No nonradionuclide pyrophorics are permitted. Radionuclides in pyrophoric form are limited to < 1 percent by weight in each waste package.	This requirement increases the stability of the waste.
Explosives and Compressed Gases - No explosives or compressed gases are permitted.	This requirement increases the stability of the waste.
Ignitable, Corrosive, and Reactive Hazardous Materials – EPA-defined characteristic ignitable, corrosive, or reactive wastes are not permitted.	This requirement increases the stability of the waste.
Criticality - Acceptable package limits are less than 200 fissile-gram-equivalents (FGE) per drum and less than 325 FGEs per standard waste box.	This requirement eliminates the risk of a nuclear criticality during transport and storage.
PE-Ci Limits - CH-TRU waste packages: Drum: 80 PE-Ci Standard Waste Box: 130 PE-Ci Drum overpacked in a Standard Waste Box or a Ten-Drum-Overpack: 1,800 PE-Ci Solidified/vitrified waste: 1,800 PE-Ci RH-TRU waste packages: 1,000 PE-Ci	Since the major exposure pathway from a container breach is inhalation and the major constituents of concern are transuranics, this requirement limits the potential radiological consequences in the event of a container breach.
Gas generation - Numerous requirements limit the amount of explosive gases that could accumulate in the interior of a waste package, including the requirement to vent all TRU waste packages.	This requirement increases the stability of the waste.

G.3.1.1 Radionuclide Inventory

The quantity of radionuclides in waste containers was assumed to be the PE-Ci limits of the planning-basis WAC (DOE 1996b) for all accident scenarios, including those of the no action alternatives. Nonsolidified CH-TRU waste drums are limited to 80 PE-Ci, CH-TRU waste standard waste boxes are limited to 130 PE-Ci, and RH-TRU waste canisters are limited to 1,000 PE-Ci. Radionuclide activities at these limits were assumed to be present in all waste containers evaluated. Vitrified and solidified final waste forms under Action Alternatives 2A, 2B, and 2C and Action Alternative 3 are limited to 1,800 PE-Ci. Current TRUPACT-II thermal power limits, however, do not allow transport of waste drums with 1,800 PE-Ci so the nonsolidified limits were applied to waste under these alternatives.

Estimated average drum PE-Ci contents for the existing TRU waste inventory are much lower than the current planning-basis WAC limits. The average radionuclide PE-Ci contents at the major treatment sites and WIPP are presented in Tables G-23 and G-24 for CH-TRU and RH-TRU waste, respectively. Because the accident analyses are intended to evaluate reasonably bounding events, however, the planning-basis WAC inventory limit was used for events involving a small number of containers. Accidents involving a large number of containers (i.e., an earthquake) were evaluated using the site-specific PE-Ci container averages.

Table G-23
Average Radionuclide Content of CH-TRU Waste
(PE-Ci per drum-equivalent) at the Major Storage Sites and WIPP

Site	Proposed Action	Action Alternative 1	Action Alternative 2A & No Action Alternative 1A	Action Alternative 2B & No Action lternative 1B	Action Alternative 3	No Action Alternative 2
Hanford	2.2	1.9	5.3	5.3	1.6	1.9
INEEL	1.8	1.5	3.3	6.9	1.3	1.5
LANL	4.3	3.6	10.0	N/A	3.1	3.7
ORNL	2.0	2.0	N/A	N/A	N/A	1.7
RFETS	13.0	7.6	35	N/A	7.0	12
SRS	21.0	16.0	40.0	40.0	12.0	18.0
WIPP	5.0	3.2	8.4 [a]	8.4 [a]	2.5	4.3

[a] WIPP is the only storage site for CH-TRU waste, so this is the CH-TRU value used for Action Alternative 2C accidents at WIPP.

N/A = Not Applicable

Table G-24
Average Radionuclide Content of RH-TRU Waste
(PE-Ci per canister) at the Major Storage Sites and WIPP

Site	Proposed Action	Action Alternative 1	Action Alternative 2 & No Action Alternative 1	Action Alternative 3	No Action Alternative 2
Hanford	0.3	3.7	9.9	2.9	3.7
INEEL	0.6	0.6	N/A	N/A	0.6
LANL	0.9	0.9	N/A	N/A	0.9
ORNL	0.1	0.3	0.9	0.3	0.4
WIPP	0.3	3.0	8.6	2.5	N/A

N/A = Not Applicable

Comparison of Tables G-23 and G-24 shows that the average PE-Ci content of RH-TRU waste is typically lower than that of CH-TRU waste. RH-TRU waste contains greater quantities of fission and activation product radionuclides that emit penetrating radiation like X- or gamma radiation which produces the high external dose rates and requires the waste to be remotely handled. However, most of the potential radiation dose and radiological consequences from radionuclide releases would result from inhalation, not from external dose, so fission and activation product radionuclides are small contributors to the PE-Ci inventory compared to transuranic radionuclides. The fission and activation product contributing the most to external dose rates were Cs-137/ Ba-137m and Co-60, with smaller but important contributions from europium-152 (Eu-152) and europium-154 (Eu-154) at some consolidation sites.

G.3.1.2 Hazardous Chemical Inventory

Hazardous chemicals that may be present in TRU waste are volatile organic compounds and heavy metals, including lead, mercury, beryllium, and cadmium. The planning-basis WAC includes limits for several VOCs based on their flammability and their health risk, as determined by the RCRA Part B Application health impact analyses (DOE 1996a). No limits are included in the planning-basis WAC for the four heavy metals of concern.

Volatile Organic Compounds

VOCs were assumed to be present in all waste treated to planning-basis WAC and the shred and grout process; however, VOCs would not be present in waste after thermal treatment under Action Alternative 2 and No Action Alternative 1. Maximum headspace concentrations of VOCs were assumed for accidents involving relatively small numbers of waste containers. This is conservative since the presence of a maximum headspace concentration of a specific VOC would have no bearing on whether another VOC would be found in the same container at a high concentration. VOCs with specific planning-basis WAC limits (carbon tetrachloride, chloroform, and methylene chloride) were assumed to be present at the maximum allowable headspace concentrations. Flammable VOCs (1,1-dichloroethylene, 1,2-dichloroethane, benzene, ethyl benzene, and xylene) were assumed to be present at a maximum concentration of 500 parts per million (ppm). Other VOCs likely to be present in the waste were assumed to be at concentrations equivalent to the maximum headspace concentration found in solidified organic samples. Solidified organics have the highest headspace concentrations of all waste matrix categories. Average VOC headspace concentrations were assumed for accidents involving a large number of waste containers, such as natural disasters. VOC headspace concentrations are presented in Table G-25. VOCs may be present in RH-TRU waste as well as CH-TRU waste. Because of the lack of sampling information, VOC concentrations in RH-TRU waste were assumed to be identical to those in CH-TRU waste.

Table G-25
VOC Headspace Concentrations in Treated TRU Waste

VOC [a]	Planning-Basis WAC Limit for Headspace Concentration (ppmv) [b]	Waste Treated to Meet Planning-Basis WAC (ppmv)		Treated by Shred and Grout (ppmv)	
		Weighted Average	Maximum [c]	Average	Maximum [d]
Carbon tetrachloride (6.39)	7,510	184.5	---	316.5	---
Chloroform (4.96)	6,325	13.7	---	1.2	---
1,1-dichloroethene (4.03)	500 [e]	8.4	---	2.5	---
1,2-dichloroethane (4.11)	500 [e]	5.6	---	1.1	---
Methylene chloride (3.53)	368,500	662.1	---	8.1	---
Chlorobenzene (4.68)	No Limit	8.4	4,368	1.3	260
Methyl ethyl ketone (3.0)	No Limit	40.4	39,311	6.8	130
1,1,2,2-tetrachloroethane (7.0)	No Limit	335.7	4,368	125.1	270
Toluene (3.83)	No Limit	5.7	6,992	1.3	320
Benzene (3.25)	500 [e]	7.6	---	22.4	---
Ethyl benzene (4.41)	500 [e]	8.5	---	31.6	---
Tetrachloroethene (6.89)	No Limit	7.1	2184	21.6	600
Xylene (4.41)	500 [e]	25.6	---	113.9	---

[a] Values in parentheses are milligram/cubic meter per ppm conversion factors.
[b] DOE 1996b.
[c] Maximum values for those VOCs without a maximum planning-basis WAC limit. Values are maximum concentrations found in any solidified organic sampled and were used for the bounding accident scenarios.
[d] Maximum concentration found in any solidified inorganic matrix for those VOCs without a planning-basis WAC limit.
[e] Flammable VOC limit.

A screening calculation was performed to identify VOCs with the greatest potential impacts from an acute exposure event. Impacts from acute hazardous chemical exposures were evaluated as potentially both carcinogenic and noncarcinogenic. As described in Section G.1.2.3, screening for carcinogenic impact was performed by multiplying the VOC headspace concentration by the EPA slope factor; noncarcinogenic impact screening was performed by comparing two different impacts measures, IDLH-equivalent intakes, and ERPG-2 values. Only the VOCs that would contribute to the major potential consequences from an accidental release (greater than 90 percent) were included in the detailed evaluation (see Table G-26). For waste treated to planning-basis WAC (Proposed Action, Action Alternative 1, No Action Alternative 2), methylene chloride would be the major contributor to toxicological consequences and 1,1,2,2-tetrachloroethane would be the main contributor to carcinogenic consequences. For grouted waste (Action Alternative 3), the major contributor to both toxicological and carcinogenic consequences would be methylene chloride. The scoping measure for which these VOCs would be the major contributor is also indicated in Table G-26.

Heavy Metals

The inventory of metals in the waste was derived from estimates developed for the WIPP SAR (DOE 1995b). The SAR assumed a conservatively high concentration of metals in the waste for analysis of a waste fire. The waste form used for the SAR was the planning-basis WAC. These values were used as the basis for estimating heavy metal inventories of thermally treated waste and waste treated by a shred and grout process (see Appendix A). Waste form and alternative-specific heavy metal inventories of lead, beryllium, cadmium, and mercury are presented in Table G-27. The metals were assumed to be uniformly mixed in the waste containers (see Appendix A).

Table G-26
VOCs with Major Consequence Contribution for Storage Accidents [a]

	Planning-Basis WAC Waste	Grouted Waste
VOCs	Methylene chloride (IDLH, ERPG-2)	Methylene chloride (slope factor, IDLH, ERPG-2)
	1,1,2,2-tetrachloroethane (slope factor)	Carbon tetrachloride
	Carbon tetrachloride	Chloroform
	Chloroform	1,1-dichloroethene
	Chlorobenzene	Benzene
	Benzene	Ethyl benzene
	Ethyl benzene	Tetrachloroethene
	Tetrachloroethene	

[a] The risk measure in parentheses identifies the VOC with the greatest screening value for that measure.

Table G-27
Heavy Metal Concentrations in Treated TRU Waste

Heavy Metals	Concentration (kilograms per drum-equivalent) [a]			
	Waste Treated to Planning-Basis WAC		Thermally Treated Waste	Waste Treated by Shred and Grout Process
	Proposed Action	Action Alternative 1	Action Alternative 2	Action Alternative 3
Lead	1.0	0.97	2.6	0.81
Beryllium	0.025	0.025	0.065	0.021
Cadmium	3.6E-4	3.5E-4	9.3E-4	3.0E-4
Mercury	0.43	0.42	1.1	0.35

[a] One drum-equivalent contains 0.208 cubic meters of waste.

G.3.2 Storage Accident Analyses

This section describes the three storage accident scenarios for waste treated to meet the planning-basis WAC, with descriptions of the scenarios as they apply to thermally treated waste or waste that has been treated by a shred and grout process. Table G-28 summarizes the parameters used in the storage accident analyses.

Accident Scenario S1 - Container Puncture, Drop, and Lid Failure

In this accident scenario, operator error causes a forklift to strike and puncture two drums during placement or relocation of waste containers. As a result, a third drum falls from the stack and its lid is knocked off upon impact with the floor. It was assumed that 10 percent of the waste spills out of the punctured containers and 25 percent of the waste spills out of the lidless container; thus, 0.001 of the spilled fraction resuspends in the room air, 0.1 of which is respirable. Particulate releases from the facility were assumed to be HEPA-filtered, with a transmission factor of 1×10^{-3}. All headspace VOCs were assumed to be released.

For waste that has been thermally treated and waste that has been treated by a shred and grout process, no particulate release would be expected from the punctured drums. The lidless drum analyses assumed brittle fracture of the uniformly mixed mass as a result of the fall. Twenty-five percent of the fractured waste was assumed to spill from the container with a respirable resuspension fraction value of 1×10^{-5}, an appropriate value for fractured brittle material (DOE 1994a). VOCs are not present in the thermally treated waste, but all headspace VOCs in the waste treated by a shred and grout process were assumed to be released from all three involved containers.

Accident Scenario S2 - Container Fire

In this scenario, it was assumed that the contents of a single container undergo spontaneous combustion and the mechanism initiating this event is unknown. Past instances of spontaneous combustion at DOE sites have included pyrophoric or incompatible materials reacting in a container, static electricity, and nitric acid reactions (Silva 1991). Packaging activities should

Table G-28
Summary of Storage Accident Scenario Parameters

Scenario	Waste Treatment	*n* Drums Breached	Particulate Release Fraction	VOC Release Fraction	Airborne Respirable Particulate Resuspension Fraction	Other
S1	WAC	3	2 @ 0.1 1@0.25	1	1E-4	1E-3
	LDR	3	1@0.25 2 @ 0.00	0	1E-5	1E-3
	Grout	3	1@0.25 2 @ 0.00	1	1E-5	1E-3
S2	WAC	1	1	0	5E-4	1E-3
	Grout	1	0.22	0	6E-5	1E-3
S3	WAC CH-TRU Waste RH-TRU Waste	 V[a] V[a]	 0.25 0.05	 1 1	 2.5E-5 5E-6	 0.5 0.25
	LDR	V[a]	0.25	0	1E-6	0.5
	Grout	V[a]	0.25	1	1E-6	0.5

[a] V is the total site-specific waste volume (drum-equivalent for CH-TRU waste canisters for RH-TRU waste).

preclude this scenario, but an analysis was conducted on a "what-if" basis without regard to likelihood of initiation. It was assumed that 5×10^{-4} of the radioactive and heavy metals were resuspended as respirable particles, which is the more conservative value of the fraction of respirable particles expected from the burning of either combustible or noncombustible materials (DOE 1994a). Plateout of airborne particulates onto the cool building interior surfaces would be expected but was not considered in the analysis. Particulates released from the facility were assumed to be HEPA-filtered, with a transmission factor of 1×10^{-3}. VOCs were assumed to be consumed by the fire.

This scenario does not apply to thermally treated waste because it is not combustible. Waste treated by a shred and grout process is unlikely to combust, but calculations were performed to bound the accident case. It was assumed that 22 percent of the grouted waste would be transformed to a powder as a result of a 650+ degree Celsius fire (DOE 1994a). A respirable resuspension fraction of 6×10^{-5} was used and reflects the fraction of a heated powder resuspended by the vapor flux generated by an open fire. As with the accident analysis for waste treated to planning-basis WAC, all VOCs in the grouted waste headspace were assumed to be consumed by the fire.

Accident Scenario S3 - Earthquake

The storage facilities were assumed to withstand a certain magnitude of earthquake (i.e., a design-basis earthquake). This scenario, however, assumes a beyond-design-basis earthquake for each storage facility, resulting in loss of confinement capability and structural failure. Although the total site-specific volume of treated waste could be in the storage facility at the time of the seismic event, it was assumed the structure collapses on the waste containers and breaches 25 percent of the drums. Except for the thermally treated waste under No Action Alternative 1, it is unlikely that any site would have the total volume of treated waste in storage at any one point in time. In Appendix F, the cumulative impact analyses estimate that the amount of waste expected in lag storage would range from 0 to 10 percent for the Basic Inventory and 0 to 55 percent for the Total Inventory.

Release estimates were based on the site-specific average PE-Ci inventories (see Table G-23). Breached CH-TRU waste drums were assumed to release all headspace VOCs, and an average of 25 percent of the contents were assumed to spill from the breached drums. The respirable airborne fraction was assumed to be 2.5×10^{-5} of the particulates, half of which were assumed to escape the collapsed structural debris and be released to the environment. For the thermally treated waste and waste treated by a shred and grout process, a portion in the containers was assumed to brittle-fracture as a result of the building collapse. Again, 25 percent of the drum contents was assumed to spill out of the breached containers and a lower 1×10^{-6} airborne respirable fraction was assumed to be available for release.

Consequence estimates for this scenario are more limited than those of the other storage accident scenarios. For the hazardous chemical consequences, only carcinogenic consequences and the IDLH-equivalent ratio impacts were estimated. No ERPG ratios were calculated because of the catastrophic nature of the accident: physical injury or fatalities would be expected for those in the facility as a result of the structural failure, and significant amounts of particulates would be suspended in the air, making suspect any estimated ERPG-2 ratios.

This same scenario was applied to accident consequence analyses for the excess RH-TRU waste stored at Hanford and ORNL under the Proposed Action. This waste meets the planning-basis WAC and is stored in the more robust RH-72B canisters. For the analyses involving an RH-TRU waste storage facility during a beyond-design-basis earthquake analyses, 5 percent of the waste canisters were assumed to breach, an airborne respirable release fraction of 5×10^{-6} was assumed, and a smaller fraction of the airborne particulates was assumed to be released from the debris of the collapsed facility (0.25 for the earthquake involving RH-TRU waste as compared to the 0.5 assumed for the earthquake involving CH-TRU waste).

G.3.3 Consequences of Storage Accidents

Consequences of waste storage accidents were estimated for the MEI, the maximally exposed noninvolved worker, the maximally exposed involved worker, and the exposed off-site population around each of the possible storage sites. Inhalation was the only exposure pathway considered for radionuclides, heavy metals, and VOCs. Acute releases were assumed to be dispersed in one direction, so population consequences were estimated for a single, maximally exposed 22.5-degree sector (out to 80 kilometers [50 miles]) and not for the entire 80-kilometer (50-mile) region population. Population-weighted atmospheric dispersion values were calculated and used to determine the maximally impacted sector, considering both the change in air concentration over distance and the population distribution in the sector. Radiological consequences are reported as the TEDE, the number of LCFs in an exposed population, and the probability of an LCF in an individual. Carcinogenic hazardous chemical consequences are similarly reported as the number of cancers occurring in the exposed population and as the probability of cancer occurring in an individual. Noncarcinogenic consequences are presented as the ratio of receptor air concentration to ERPG-2 value for VOC exposures and as the ratio of receptor intake to IDLH-equivalent intake for both VOCs and heavy metals exposures. Ratios of 1.0 or higher are significant for a noncarcinogenic impact. Accident consequences were estimated using conservative or bounding input and parameter values and, thus, will likely overestimate the consequences that could occur.

G.3.3.1 Storage Accident Consequences for Action Alternative 1 and No Action Alternative 2

The consequences of storage accidents for waste treated to meet the planning-basis WAC are presented in Tables G-29 to G-31. Table G-29 indicates the radiological consequences. Table G-30 indicates the carcinogenic consequences as a result of the VOC and metals exposures. Table G-31 indicates the noncarcinogenic consequences, presented as the maximum IDLH-equivalent ratio and the maximum ERPG-2 ratio, for maximally exposed workers under Action Alternative 1 and No Action Alternative 2, assuming that newly generated waste under the latter would be packaged to meet the planning-basis WAC. Consequences presented in this section for Action Alternative 1 and No Action Alternative 2 would bound potential consequences of RH-TRU waste storage accidents under the Proposed Action.

Impacts to Population

The greatest potential radiological consequences to the population surrounding any site would be for RFETS (see Table G-29). This result is driven by the population-weighted dispersion factor of the area southeast of the RFETS site, which considers the distance from the population and plume dispersion at various distances. Locations with higher populations at closer distances would result in higher population-weighted dispersion factors. Containers with bounding radionuclide inventories involved in container breach (Accident Scenario S1) or container fire (Accident Scenario S2) accidents would not be expected to result in any LCFs at RFETS or any other site. Accident Scenario S1 population consequences could range from 1 x 10-5 to 5 x 10-4 LCFs, and

Table G-29
Radiological Consequences from Storage Accidents for Waste Treated to Planning-Basis WAC (Action Alternative 1 and No Action Alternative 2)

Site	Accident Scenario S1 (Drum Puncture and Lid Failure)		Accident Scenario S2 (Drum Fire)		Accident Scenario S3 (Earthquake)			
					CH-TRU Waste		RH-TRU Waste	
Population	**Dose (person- rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**	**Dose (person-rem)**	**Number of LCFs**
Hanford	0.47	2E-4	2.1	1E-3	450,000	200	1900	0.9
INEEL	0.023	1E-5	0.10	5E-5	13,000	6	1.2	6E-4
LANL	0.20	1E-4	0.91	5E-4	110,000	50	2.3	1E-3
RFETS	0.94	5E-4	4.2	2E-3	510,000	300	N/A	N/A
ORNL	0.68	3E-4	3.0	2E-3	12,000	6	5.1	3E-3
SRS	0.068	3E-5	0.30	2E-4	75,400	38	N/A	N/A
MEI	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**	**Dose (rem)**	**Probability of an LCF**
Hanford	1.2E-4	6E-8	5.3E-4	3E-7	250	0.1	1.1	5E-4
INEEL	1.0E-5	5E-9	4.3E-5	2E-8	6.7	3E-3	6.3E-4	3E-7
LANL	2.8E-4	1E-7	1.2E-3	6E-7	210	0.1	4.3E-3	2E-6
RFETS	5.5E-5	3E-8	2.5E-4	1E-7	32	0.02	N/A	N/A
ORNL	5.1E-4	3E-7	2.3E-3	1E-6	12	6E-3	5.1E-3	3E-6
SRS	4.0E-6	2E-9	1.7E-5	8E-9	8.5	4E-3	N/A	N/A
Maximally Exposed Noninvolved Worker								
Hanford	1.1E-3	4E-7	4.9E-3	2E-6	1,900	1.0	8.1	3E-3
INEEL	2.4E-4	1E-7	1.1E-3	4E-7	1,100	0.6	0.1	4E-5
LANL	8.6E-4	3E-7	3.8E-3	2E-6	1,100	0.6	0.022	9E-6
RFETS	3.9E-4	2E-7	1.7E-3	7E-7	1,100	0.6	N/A	N/A
ORNL	8.6E-4	3E-7	3.8E-3	2E-6	36	0.01	0.015	6E-6
SRS	5.3E-4	2E-7	2.4E-3	9E-7	2,200	1.0	N/A	N/A
Maximally Exposed Involved Worker								
Any Site	140	0.6	See Text	See Text	See Text	See Text	See Text	See Text

Note: Annual doses above 20 rem TEDE to individuals do not include a DDREF (see Section G.1.1).

N/A = Not Applicable

Table G-30
Hazardous Chemical Carcinogenic Consequences from Storage Accidents for Waste Treated to Planning-Basis WAC (Action Alternative 1 and No Action Alternative 2)

Site	Accident Scenario S1 (Drum Puncture and Lid Failure)		Accident Scenario S2 (Drum Fire)		Accident Scenario S3 (Earthquake)			
					CH-TRU Waste		RH-TRU Waste	
	VOCs	Metals	VOCs [a]	Metals	VOCs	Metals	VOCs	Metals
Population	Number of Cancers		Number of Cancers		Number of Cancers		Number of Cancers	
Hanford	9E-7	1E-12	N/A	6E-12	2E-3	5E-5	1E-4	4E-7
INEEL	4E-8	7E-14		3E-13	6E-5	2E-6	5E-7	2E-9
LANL	4E-7	6E-13		3E-12	2E-4	7E-6	6E-7	2E-9
RFETS	2E-6	3E-12		1E-11	5E-4	2E-5	N/A	N/A
ORNL	1E-6	2E-12		9E-12	4E-5	1E-6	3E-5	1E-7
SRS	1E-7	2E-13		9E-13	3E-5	1E-6	N/A	N/A
MEI	Cancer Incidence		Cancer Incidence		Cancer Incidence		Cancer Incidence	
Hanford	2E-10	3E-16	N/A	2E-15	9E-7	3E-8	7E-8	2E-10
INEEL	2E-11	3E-17		1E-16	3E-8	1E-9	2E-10	8E-13
LANL	5E-10	8E-16		4E-15	4E-7	1E-8	1E-9	4E-12
RFETS	2E-10	2E-16		7E-16	3E-8	1E-9	N/A	N/A
ORNL	9E-10	1E-15		7E-15	4E-8	1E-9	3E-8	1E-10
SRS	7E-12	1E-17		5E-17	4E-9	1E-10	N/A	N/A
Maximally Exposed Noninvolved Worker								
Hanford	1E-9	3E-15	N/A	1E-14	5E-6	2E-7	4E-7	2E-9
INEEL	3E-10	7E-16		3E-15	4E-6	2E-7	3E-8	1E-10
LANL	1E-9	2E-15		1E-14	1E-6	7E-8	4E-9	2E-11
RFETS	5E-10	1E-15		5E-15	7E-7	3E-8	N/A	N/A
ORNL	1E-9	2E-15		1E-14	9E-8	4E-9	7E-8	3E-10
SRS	7E-10	2E-15		7E-15	7E-7	3E-8	N/A	N/A
Maximally Exposed Involved Worker								
Any Site	7E-8	4E-10	N/A	See Text	See Text	See Text	See Text	See Text

[a] VOCs are assumed to be consumed by the fire under Accident Scenario S2.

N/A = Not Applicable

Table G-31
Hazardous Chemical Noncarcinogenic Consequences from Storage Accidents for Waste Treated to Planning-Basis WAC (Action Alternative 1 and No Action Alternative 2)

Site	Accident Scenario S1 (Drum Puncture and Lid Failure)		Accident Scenario S2 (Drum Fire)		Accident Scenario S3 (Earthquake)	
					CH-TRU Waste	RH-TRU Waste
	Maximum IDLH-Equivalent Ratio	Maximum ERPG-2 Ratio	Maximum IDLH-Equivalent Ratio	Maximum ERPG-2 Ratio	Maximum IDLH-Equivalent Ratio	
MEI						
Hanford	5E-7	1E-1	5E-11	2E-4	1E-3	7E-5
INEEL	4E-8	9E-3	4E-12	2E-5	4E-5	3E-7
LANL	1E-6	3E-1	1E-10	5E-4	5E-4	1E-6
RFETS	2E-7	5E-2	3E-11	1E-4	4E-5	N/A
ORNL	2E-6	5E-1	2E-10	1E-3	5E-5	4E-5
SRS	1E-8	4E-3	2E-12	7E-6	4E-6	N/A
Maximally Exposed Noninvolved Worker						
Hanford	3E-6	7E-1	5E-10	2E-3	8E-3	4E-4
INEEL	7E-7	2E-1	1E-10	3E-4	6E-3	3E-5
LANL	2E-6	6E-1	4E-10	1E-3	2E-3	4E-6
RFETS	1E-6	3E-1	2E-10	5E-4	1E-3	N/A
ORNL	2E-6	6E-1	4E-10	1E-3	1E-4	7E-5
SRS	1E-6	6E-1	2E-10	7E-4	1E-3	N/A
Maximally Exposed Involved Worker						
Any Site	2E-4	2	See Text	See Text	See Text	See Text

N/A = Not Applicable

Accident Scenario S2 population consequences could range from 5×10^{-5} to 2×10^{-3} LCFs. The catastrophic earthquake scenario (Accident Scenario S3) could result in up to 300 LCFs in the RFETS population. For all sites, Accident Scenario S3 consequences could range from 6 to 300 LCFs. The potential hazardous chemical consequences to the population would be very small: 7×10^{-14} to 2×10^{-6} cancers under Accident Scenario S1, 3×10^{-13} to 1×10^{-11} cancers under Accident Scenario S2, 1×10^{-6} to 2×10^{-3} cancers under Accident Scenario S3 with CH-TRU waste, and 2×10^{-9} to 1×10^{-4} cancers under Accident Scenario S3 with RH-TRU waste (see Table G-30). Therefore, no cancers would be expected in the populations surrounding the five major sites as a result of the hazardous chemical releases of any analyzed accident.

Impacts to Maximally Exposed Individual

The greatest potential radiological consequences for the MEI for Accident Scenarios S1 and S2 would be for ORNL, and the greatest potential radiological consequences for the MEI for Accident Scenario S3 would be for Hanford (see Table G-29). The site of maximum consequence changes for Accident Scenario S3 because these consequences are dependent on site-specific volumes of stored waste and average PE-Ci levels. Accidents Scenarios S1 and S2 assume the same waste volume and PE-Ci levels for each site-specific accident. In contrast to the population consequences discussed above, the change in the MEI dispersion factors for each site is less than the site-specific waste characteristics. Therefore, the MEI consequences are driven more by site-specific waste characteristic changes than by the dispersion factors. Under Accident Scenario S1, radiation doses to the MEI would range from 4×10^{-6} to 5×10^{-4} rem TEDE, with an associated probability of an LCF of 2×10^{-9} to 3×10^{-7}. Under Accident Scenario S2, radiation doses to the MEI would range from 2×10^{-5} to 2×10^{-3} rem TEDE, with an associated probability of an LCF of 8×10^{-9} to 1×10^{-6}. Catastrophic accident Scenario S3 with CH-TRU waste results in an estimated 7 to 250 rem TEDE to the MEI and a 3×10^{-3} to 0.1 probability of an LCF. Accident Scenario S3 with RH-TRU waste results in an estimated 6.3×10^{-4} to 1.1 rem TEDE to the MEI and a 3×10^{-7} to 5×10^{-4} probability of an LCF.

The greatest potential hazardous chemical consequences to the MEI would be for ORNL under Accident Scenarios S1 and S2, and the greatest potential chemical consequencesfor the MEI under Accident Scenario S3 would be for Hanford. The site of maximum hazardous chemical consequence changes for Accident Scenario S3 because of the site-specific parameters of the Accident Scenario S3 analyses. The hazardous chemical inventory of each site is assumed to be constant, but the affected volume changes. The waste volume affected at Hanford drives the Accident Scenario S3 consequence results.

The potential hazardous chemical consequences to the MEI would be very small: carcinogenic consequences would be no greater than a 9×10^{-10}, 7×10^{-15}, and 4×10^{-7} probability of contracting cancer for Accident Scenarios S1, S2, and S3 with CH-TRU waste, respectively (see Table G-30). Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for a noncarcinogenic consequence. For Accident Scenario S1 noncarcinogenic consequences, the maximum ERPG-2 ratio would be 0.5 for ORNL for both methylene chloride and 1,1,2,2-tetrachloroethane. Therefore, no serious or irreversible consequences would be expected. Also, no mild transient health consequences or objectionable odors would be expected for this ORNL MEI (maximum ERPG-1 ratio of 0.8 for 1,1,2,2-tetrachloroethane). The maximum IDLH-equivalent ratio for Accident Scenario S1 would be 2×10^{-6} (see Table G-31). Under Accident Scenario S2, the maximum ERPG-2 ratio for the MEI would be 1 x 10^{-3} and the maximum IDLH-equivalent ratio would be 2 x 10^{-10}. Only ERPGs for metals are considered under the Accident Scenario S2 analyses. Under Accident Scenario S3, the maximum IDLH-equivalent ratio would be 1 x 10^{-3} for the Hanford MEI. Therefore, no serious noncarcinogenic consequences would be expected from hazardous chemical exposures under any of the three accident scenarios evaluated.

Impacts to Maximally Exposed Noninvolved Worker

The potential radiological consequences would be greatest for all accidents evaluated for the Hanford noninvolved worker (see Table G-29). Under Accident Scenario S1, radiological doses to the maximally exposed noninvolved worker would range from 2 x 10^{-4} to 1 x 10^{-3} rem TEDE, with an associated probability of an LCF of 1 x 10^{-7} to 4 x 10^{-7}. Under Accident Scenario S2, radiological doses to the noninvolved worker would range from 1 x 10^{-3} to 5 x 10^{3} rem TEDE with an associated probability of an LCF of 4 x 10^{-7} to 2 x 10^{-6}. Accident Scenario S3 with CH-TRU waste would result in an estimated 36 to 1,900 rem TEDE to the noninvolved worker. Potential consequences of Accident Scenario S3 from CH-TRU waste to Hanford, SRS, INEEL, LANL, and RFETS would be severe. The first-year dose at these sites was estimated to be greater than 20 rem (88 rem for SRS, 76 for Hanford, and 44 rem for INEEL, LANL and RFETS), which increases their LCF risk for that year by a factor of two (see Section G.1.1). The probability of an LCF for the individuals at these sites is 1.0 for Hanford and SRS and 0.6 for INEEL, LANL, and RFETS. Consequences of Accident Scenario S3 with RH-TRU waste for the noninvolved worker would be greatest at Hanford. Accident Scenario S3 with RH-TRU waste results in an estimated 0.02 to 8.1 rem TEDE with an associated probability of an LCF of 6 x 10^{-6} to 3 x 10^{-3}.

The potential carcinogenic hazardous chemical consequences would be greatest for the Hanford maximally exposed noninvolved worker for all accidents (see Table G-30). Rounding of Accident Scenarios S1 and S2 consequences for the LANL and ORNL noninvolved workers also gives carcinogenic consequences that are equivalent to those of the Hanford noninvolved worker. The potential hazardous chemical consequences to the noninvolved worker would be very small, with a 1 x 10^{-9}, 1 x 10^{-14}, and 5 x 10^{-6} probability of contracting cancer for Accident Scenarios S1, S2, and S3, respectively. The Accident Scenario S3 consequences are attributable to the 1,1,2,2-tetracholorethane releases (see Table G-30). Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for a noncarcinogenic consequence. For Accident Scenario S1 noncarcinogenic consequences, the maximum ERPG-2 ratio would be 0.7 for Hanford for both methylene chloride and 1,1,2,2-tetrachloroethane releases (see Table G-30). Therefore, no serious or irreversible consequences would be expected. Mild transient health consequences or objectionable odors would be expected for this Hanford noninvolved worker (maximum ERPG-1 ratio of 1.2 for 1,1,2,2-tetrachloroethane). The maximum IDLH-equivalent ratio for Accident Scenario S1 would be 3 x 10^{-6}. Under Accident Scenario S2, the maximum ERPG-2 ratio for the noninvolved worker would be 2 x 10^{-3} and the maximum IDLH-equivalent ratio would be 5 x 10^{-10}. Under Accident Scenario S3, the maximum IDLH-equivalent ratio would be 8 x 10^{-3} for the Hanford MEI. No serious noncarcinogenic consequences are expected from hazardous chemical exposures under any of the three accident scenarios evaluated.

Impacts to Maximally Exposed Involved Workers

Accident consequences were estimated quantitatively for the maximally exposed involved worker only for Accident Scenario S1. The drum drop was assumed to occur 3 meters (10 feet) from the worker and the estimated release of the three involved containers was assumed to expand in a uniform 5-meter (17-foot) radius hemisphere. The worker was assumed to inhale air at the concentration of the hemisphere for 60 seconds prior to exiting to a fresh air source (DOE 1997). For waste packaged to meet the planning-basis WAC, the worker would receive a dose of 140 rem TEDE with an associated probability of an LCF of 0.06. The maximum first-year dose from such an intake would be 6 rem. Carcinogenic consequences from VOC and heavy metal intakes would be a 7×10^{-8} and 4×10^{-10} probability of a cancer incidence, respectively. Releases could have irreversible, non-life-threatening consequences from exposure to 1,1,2,2-tetrachloroethane and methylene chloride (ERPG-2 ratio of 2 for both) (see Table G-31). Mild transient health consequences would be expected from lead and mercury exposures (ERPG-1 ratios of 1 for both) (see Table G-31). Noncarcinogenic consequences evaluated according to IDLH-equivalent intake ratios do not indicate that any consequence would result (maximum ratio of 2×10^{-4}).

If the maximally exposed involved worker were located next to a drum fire (Accident Scenario S2) when it erupted, the consequences would be severe. However, the probability that this scenario would occur is minute. The smoke would be apparent and the worker can be presumed to exit the facility immediately.

If the maximally exposed involved worker were present when the storage facility collapsed (Accident Scenario S3), consequences would be severe. If not killed by falling accident debris, the worker could inhale high levels of radionuclides and hazardous chemicals.

G.3.3.2 Storage Accident Consequences for Action Alternative 2 and No Action Alternative 1

Waste which has been thermally treated to a final waste form that meets land disposal restrictions is safer to store than waste packaged to meet the planning-basis WAC. Thermally treated waste is a feature of Action Alternative 2 and No Action Alternative 1, has no VOCs, and is solidified such that radioactive material and heavy metals are unlikely to disperse in the event of a containment breach. Radiological consequences are presented first, followed by hazardous chemical consequences. Consequence estimates for thermally treated waste are presented in Tables G-32 and G-33. The subalternatives under Action Alternative 2 and No Action Alternative 1 are not presented separately; however, the highest consequence that would occur under any of the subalternatives is presented in Tables G-32 and G-33. Accident Scenario S2 is not expected for waste treated to meet land disposal restictions.

Impacts to Population

The potential radiological consequences to the population surrounding any site would be highest around RFETS. Releases of radioactive materials from a container breach (Accident Scenario S1) would not be expected to result in any LCFs in the population (6×10^{-7} to 3×10^{-5}) (see Table G-32). The consequences from the catastrophic CH-TRU waste storage facility failure of Accident Scenario S3 would result in an estimated 0.8 to 10 LCFs. All major storage facility sites except INEEL would be expected to have at least 1 LCF from such an event. No LCFs would be expected from catastrophic failure of an RH-TRU waste storage facility at ORNL or Hanford (4×10^{-4} and 2×10^{-2} LCFs).

Table G-32
Radiological Consequences from Storage Accidents for Thermally Treated Waste
(Action Alternative 2 and No Action Alternative 1)

Site	Accident Scenario S1 (Drum Puncture and Lid Failure)		Accident Scenario S3 (Earthquake)			
			CH-TRU Waste		RH-TRU Waste	
	Dose (person-rem or rem)	LCF	Dose (person-rem or rem)	LCF	Dose (person-rem or rem)	LCF
Population						
Hanford	2.6E-2	1E-5	18,000	9	39	2E-2
INEEL [a]	1.3E-3	6E-7	1,500	0.8	N/A	N/A
LANL [b]	1.1E-2	6E-6	4,200	2	N/A	N/A
RFETS [b]	5.2E-2	3E-5	21,000	10	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	0.86	4E-4
SRS	3.8E-3	2E-6	3,200	2	N/A	N/A
WIPP [c]	1.4E-3	7E-7	3,600	2	N/A	N/A
MEI		**Probability of an LCF**		**Probability of an LCF**		**Probability of an LCF**
Hanford	6.6E-6	3E-9	9.9	5E-3	2.2E-2	1E-5
INEEL [a]	5.4E-7	3E-10	0.81	4E-4	N/A	N/A
LANL [b]	1.5E-5	8E-9	8.0	4E-3	N/A	N/A
RFETS [b]	3.1E-6	2E-9	1.3	7E-4	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	8.6E-4	4E-7
SRS	2.1E-7	1E-10	0.35	2E-4	N/A	N/A
WIPP [c]	2.6E-5	1E-8	160	0.08	N/A	N/A
Maximally Exposed Noninvolved Worker		**Probability of an LCF**		**Probability of an LCF**		**Probability of an LCF**
Hanford	6.1E-5	2E-8	76	3E-2	0.17	7E-5
INEEL [a]	1.4E-5	5E-9	130	5E-2	N/A	N/A
LANL [b]	4.8E-5	2E-8	41	2E-2	N/A	N/A
RFETS [b]	2.2E-5	9E-9	45	2E-2	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	2.6E-3	1E-5
SRS	3.0E-5	1E-8	94	4E-2	N/A	N/A
WIPP [c]	2.6E-5	1E-8	280	0.1	N/A	N/A
Maximally Exposed Involved Worker		**Probability of an LCF**		**Probability of an LCF**		**Probability of an LCF**
Any Site	7.8	3E-3	See Text	See Text	See Text	See Text

[a] Consequences are greatest at INEEL under Action Alternative 2B.
[b] CH-TRU waste is stored at LANL and RFETS only under Action Alternative 2A.
[c] CH-TRU waste is stored WIPP only under Action Alternative 2C.

N/A = Not Applicable

The potential hazardous chemical carcinogenic consequences to populations would be very small. The population around RFETS and Hanford were estimated to be the most affected of any of the exposed populations. Only heavy metals would contribute any hazardous chemical consequences, and only cadmium and beryllium would be carcinogenic. Maximum potential carcinogenic consequences would range from 1×10^{-14} to 4×10^{-13} cancers for Accident Scenario S1, 5×10^{-8} to 2×10^{-6} cancers for Accident Scenario S3 with CH-TRU waste, and 2×10^{-9} to 8×10^{-9} cancers for Accident Scenario S3 with RH-TRU waste (see Table G-33).

Impacts to Maximally Exposed Individuals

The greatest potential radiological consequences for the MEI would be at WIPP for Accident Scenarios S1 and S3 (Action Alternative 2C only) (see Table G-32). The MEI doses for Accident Scenario S1 would range from 2×10^{-7} to 3×10^{-5} rem TEDE, with a 1×10^{-10} to 1×10^{-8}

Table G-33
Hazardous Chemical Carcinogenic and Noncarcinogenic Consequences from Storage Accidents for Thermally Treated Waste (Action Alternative 2 and No Action Alternative 1)

Site	Accident Scenario S1 (Drum Puncture and Lid Failure)			Accident Scenario S3 (Earthquake)			
				CH-TRU Waste		RH-TRU Waste	
	Cancer Incidence	Maximum IDLH-equivalent Ratio	Maximum ERPG-2 Ratio	Cancer Incidence	Maximum IDLH-equivalent Ratio	Cancer Incidence	Maximum IDLH-equivalent Ratio
Population							
Hanford	2E-13	N/A	N/A	2E-6	N/A	8E-9	N/A
INEEL	1E-14			1E-7		N/A	
LANL	9E-14			3E-7		N/A	
RFETS	4E-13			4E-7		N/A	
ORNL	N/A			N/A		2E-9	
SRS	3E-14			5E-8		N/A	
WIPP	1E-14			3E-7		N/A	
MEI	**Probability of an LCF**			**Probability of an LCF**		**Probability of an LCF**	
Hanford	5E-17	2E-12	8E-6	1E-9	4E-5	5E-12	2E-7
INEEL	4E-18	1E-13	6E-7	7E-11	3E-6	N/A	N/A
LANL	1E-16	4E-12	2E-5	5E-10	2E-5	N/A	N/A
RFETS	2E-17	8E-13	4E-6	2E-11	8E-7	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	2E-12	7E-8
SRS	2E-18	6E-14	2E-7	5E-12	2E-7	N/A	N/A
WIPP	2E-16	6E-12	2E-5	2E-7	7E-3	N/A	N/A
Maximally Exposed Noninvolved Worker	**Probability of an LCF**			**Probability of an LCF**		**Probability of an LCF**	
Hanford	5E-16	2E-11	5E-5	9E-9	3E-4	3E-11	1E-6
INEEL	1E-16	4E-12	1E-5	1E-8	4E-4	N/A	N/A
LANL	4E-16	1E-11	4E-5	2E-9	9E-5	N/A	N/A
RFETS	2E-16	6E-12	2E-5	8E-10	3E-5	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	6E-12	2E-7
SRS	2E-16	8E-12	2E-5	1E-9	5E-5	N/A	N/A
WIPP	2E-16	6E-12	2E-5	2E-8	8E-4	N/A	N/A
Maximally Exposed Involved Worker	**Probability of an LCF**			**Probability of an LCF**		**Probability of an LCF**	
Any Site	6E-11	0.5	See Text	See Text	See Text	See Text	See Text

N/A = Not Applicable

probability of an LCF. For Accident Scenario S3 with CH-TRU waste, MEI doses would range from 0.35 to 160 rem TEDE, with a 2×10^{-4} to 0.08 probability of an LCF. For Accident Scenario S3 with RH-TRU waste, MEI doses would be 9×10^{-4} to 2×10^{-2} rem TEDE, with a 4×10^{-7} and 1×10^{-5} probability of an LCF.

The hazardous chemical impacts to the MEI would be extremely small. Carcinogenic consequences from exposure to hazardous chemicals would range from a 2×10^{-18} to 2×10^{-16} probability of a cancer and a 5×10^{-12} to 2×10^{-7} probability of a cancer, and 2×10^{-12} to 5×10^{-12} probability of cancer for Accident Scenarios S1 and S3 with CH-TRU waste with RH-TRU waste.

Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for a noncarcinogenic consequence. For Accident Scenario S1, the maximum ERPG-2 ratio would be 2×10^{-5} and the maximum IDLH-equivalent ratio would be even smaller (6×10^{-12}). For Accident Scenario S3 with CH-TRU waste and RH-TRU waste, the maximum IDLH-equivalent ratio would be 7×10^{-3}. Therefore, no noncarcinogenic consequences would be expected under either of the two accidents evaluated.

Impacts to Maximally Exposed Noninvolved Worker

The potential radiological consequences would be greatest for the Hanford maximally exposed noninvolved worker for Accident Scenario S1 and for the WIPP maximally exposed noninvolved worker for Accident Scenario S3 (see Table G-32). Under Accident Scenario S1, radiological doses to the noninvolved worker would range from 1 x 10^{-5} to 6 x 10^{-5} rem TEDE, with a 5 x 10^{-9} to 2 x 10^{-8} probability of an LCF. Under Accident Scenario S3 with CH-TRU waste, the assumed storage of all waste at WIPP under Action Alternative 2C resulted in the bounding consequences. Doses under Accident Scenario S3 with CH-TRU waste would range from 41 to 280 rem TEDE, with a 0.02 to 0.1 probability of an LCF. Doses for Accident Scenario S3 with RH-TRU waste would be 3 x 10^{-3} and 0.2 rem TEDE with a 1 x 10^{-6} and 7 x 10^{-5} probability of an LCF.

The potential consequences to the maximally exposed noninvolved worker from exposures to hazardous chemicals as a result of the two accidents evaluated would be extremely small. Under Accident Scenario S1, carcinogenic consequences would be no greater than a 5 x 10^{-16} probability of cancer. Under Accident Scenario S3 for CH-TRU and RH-TRU waste, carcinogenic consequences would be no greater than a 2 x 10^{-8} probability of cancer. Accident Scenario S1 has a maximum ERPG-2 ratio of 5 x 10^{-5} and a maximum IDLH-equivalent ratio of 2 x 10^{-11}. Accident Scenario S3 has a maximum IDLH-equivalent ratio of 8 x 10^{-4}. Therefore, no noncarcinogenic consequences would be expected.

Impacts to Maximally Exposed Involved Worker

Accident consequences were estimated quantitatively for the maximally exposed involved worker only under Accident Scenario S1. The drum drop was assumed to occur 3 meters (10 feet) from the worker, and the estimated release of the three involved containers was assumed to expand in a uniform 5-meter (17-foot) radius hemisphere. The worker was assumed to inhale air at the concentration of the hemisphere for approximately 60 seconds prior to exiting to a fresh air source (DOE 1997). For thermally treated waste, the worker would receive a dose of 7.8 rem TEDE, with a 3 x 10^{-3} probability of an LCF. The maximum first-year dose from such an intake would be 0.3 rem TEDE. A carcinogenic consequence of a 6 x 10^{-11} probability of cancer would be expected from heavy metal intakes. No noncarcinogenic consequences would be expected.

If an involved worker were present when the storage facility collapsed (Accident Scenarios S3 with CH-TRU and RH-TRU waste), consequences would be severe. If not killed by falling accident debris, surviving workers could inhale high levels of transuranic radionuclides and hazardous chemicals.

G.3.3.3 Storage Accident Consequences for Action Alternative 3

The consequences from the potential accidental releases of waste treated by a shred and grout process are indicated in Tables G-34 through G-36. Because of the solidified waste form, these accident analyses, which apply only to Action Alternative 3, indicate fewer consequences than those calculated for the Proposed Action. Radiological consequences are discussed first, followed by hazardous chemical consequences. The Accident Scenario S1 radiological consequences are identical to those reported for the thermally treated waste. This is a consequence of the evaluation

Table G-34
Radiological Consequences from Storage Accidents for Waste Treatment by Shred and Grout (Action Alternative 3)

Site	Accident Scenario S1 (Drum Puncture and Lid Failure)		Accident Scenario S2 (Drum Fire)		Accident Scenario S3 (Earthquake)			
					CH-TRU Waste		RH-TRU Waste	
	Rem	LCF	Rem	LCF	Rem	LCF	Rem	LCF
Population								
Hanford	2.6E-2	1E-5	5.5E-2	3E-5	18,000	9	39	0.02
INEEL	1.3E-3	6E-7	2.7E-3	1E-6	530	0.3	N/A	N/A
LANL	1.1E-2	6E-6	2.4E-2	1E-5	4,400	2	N/A	N/A
RFETS	5.2E-2	3E-5	1.1E-1	5E-5	20,000	10	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	N/A	0.98	5E-4
SRS	3.8E-3	2E-6	8.0E-3	4E-6	3,200	2	N/A	N/A
MEI								
Hanford	6.6E-6	3E-9	1.4E-5	7E-9	10	5E-3	0.022	1E-5
INEEL	5.4E-7	3E-10	1.1E-6	6E-10	0.28	1E-4	N/A	N/A
LANL	1.5E-5	8E-9	3.2E-5	2E-8	8.5	4E-3	N/A	N/A
RFETS	3.1E-6	2E-9	6.5E-6	3E-9	1.3	6E-4	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	N/A	9.8E-4	5E-7
SRS	2.1E-7	1E-10	4.4E-7	2E-10	0.36	2E-4	N/A	N/A
Maximally Exposed Noninvolved Worker								
Hanford	6.1E-5	2E-8	1.3E-4	5E-8	79	3E-2	0.17	7E-5
INEEL	1.4E-5	5E-9	2.9E-5	1E-8	46	2E-2	N/A	N/A
LANL	4.8E-5	2E-8	1.0E-4	4E-8	44	2E-2	N/A	N/A
RFETS	2.2E-5	9E-9	4.6E-5	2E-8	44	2E-2	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	N/A	2.9E-3	1E-6
SRS	3.0E-5	1E-8	6.2E-5	2E-8	95	4E-2	N/A	N/A
Maximally Exposed Involved Worker								
Any Site	7.8	3E-3	See Text	See Text	See Text	See Text	See Text	See Text

N/A = Not Applicable

Table G-35
Hazardous Chemical Carcinogenic Consequences from Storage Accidents for Waste Treatment by Shred and Grout (Action Alternative 3)

Site	Accident Scenario S1 (Drum Puncture and Lid Failure)		Accident Scenario S2 (Drum Fire)	Accident Scenario S3 (Earthquake)			
				CH-TRU Waste		RH-TRU Waste	
	VOCs	Metals	Metals	VOCs	Metals	VOCs	Metals
Population							
Hanford	5E-7	6E-14	1E-13	1E-3	2E-6	8E-5	9E-9
INEEL	3E-8	3E-15	7E-15	4E-5	8E-8	N/A	N/A
LANL	2E-7	3E-14	6E-14	1E-4	3E-7	N/A	N/A
RFETS	1E-6	1E-13	3E-13	3E-4	6E-7	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	2E-5	2E-9
SRS	7E-8	9E-15	2E-14	2E-5	5E-8	N/A	N/A
MEI							
Hanford	1E-10	2E-17	3E-17	6E-7	1E-9	5E-8	5E-12
INEEL	1E-11	1E-18	3E-18	2E-8	4E-11	N/A	N/A
LANL	3E-10	4E-17	8E-17	3E-7	5E-10	N/A	N/A
RFETS	6E-11	7E-18	2E-17	2E-8	4E-11	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	2E-8	2E-12
SRS	4E-12	5E-19	1E-18	3E-9	6E-12	N/A	N/A
Maximally Exposed Noninvolved Worker							
Hanford	8E-10	1E-16	3E-16	3E-6	1E-8	2E-7	4E-11
INEEL	2E-10	3E-17	7E-17	2E-6	7E-9	N/A	N/A
LANL	6E-10	1E-16	2E-16	9E-7	3E-9	N/A	N/A
RFETS	3E-10	5E-17	1E-16	4E-7	1E-9	N/A	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	4E-8	6E-12
SRS	4E-10	7E-17	2E-16	5E-7	2E-9	N/A	N/A
Maximally Exposed Involved Worker							
Any Site	5E-8	2E-11	See Text	See Text	See Text	See Text	See Text

N/A = Not Applicable

Table G-36
Hazardous Chemical Noncarcinogenic Consequences from Storage Accidents for Waste Treatment by Shred and Grout (Action Alternative 3)

Site	Accident Scenario S1 (Drum Puncture and Lid Failure)		Accident Scenario S2 (Drum Fire)		Accident Scenario S3 (Earthquake)	
					CH-TRU Waste	RH-TRU Waste
	Maximum IDLH-equivalent Ratio	Maximum ERPG-2 Ratio	Maximum IDLH-equivalent Ratio	Maximum ERPG-2 Ratio	Maximum IDLH-equivalent Ratio	Maximum IDLH-equivalent Ratio
MEI						
Hanford	5E-7	1E-1	1E-12	5E-6	6E-4	5E-5
INEEL	4E-8	9E-3	1E-13	4E-7	2E-5	N/A
LANL	1E-6	3E-1	3E-12	1E-5	2E-4	N/A
RFETS	2E-7	5E-2	6E-13	2E-6	2E-5	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	2E-5
SRS	1E-8	4E-3	4E-14	2E-7	3E-6	N/A
Maximally Exposed Noninvolved Worker						
Hanford	3E-6	7E-1	1E-11	3E-5	3E-3	2E-4
INEEL	7E-7	2E-1	2E-12	7E-6	2E-3	N/A
LANL	2E-6	6E-1	9E-12	3E-5	9E-4	N/A
RFETS	1E-6	3E-1	4E-12	1E-5	4E-4	N/A
ORNL	N/A	N/A	N/A	N/A	N/A	4E-5
SRS	1E-6	1E-1	5E-12	2E-5	5E-4	N/A
Maximally Exposed Involved Worker						
Any Site	2E-4	2	See Text	See Text	See Text	See Text

N/A = Not Applicable

of 80 PE-Ci containers for each solidified waste form. It is more likely, however, that consequences from randomly selected drums under the grouted waste alternative would be less than those of randomly selected drums under thermally treated waste alternatives because actual average PE-Ci levels of grouted waste are less than those of thermally treated waste (see Table G-23).

Impacts to Population

The potential radiological consequence to the population surrounding the waste storage sites would be greatest at RFETS. This result is driven by the population-weighted dispersion factor for the population southeast of the site. No LCFs would be anticipated for the RFETS or any other population as a result of Accident Scenario S1. Consequence estimates range from 6 x 10^{-7} to 3 x 10^{-5} LCFs. A waste container fire (Accident Scenario S2) would also not be expected to result in any LCFs in the surrounding population of any site (1 x 10^{-6} to 5 x 10^{-5} LCFs). The catastrophic CH-TRU waste storage facility failure (Accident Scenario S3 with CH-TRU waste) would potentially result in up to 10 LCFs in the RFETS population. Consequences from Accident Scenario S3 with CH-TRU and RH-TRU waste would range from 5.0 x 10^{-4} to 10 LCFs.

The potential hazardous chemical impacts would be very small; no incidences of cancer would be expected in the populations as a result of the hazardous chemical releases of the accidents analyzed. Estimates ranged from a 3 x 10^{-15} to 1 x 10^{-6} cancer incidences for Accident Scenario S1, 7 x 10^{-15} to 3 x 10^{-13} cancers for Accident Scenario S2, and 2 x 10^{-9} to 1 x 10^{-3} cancers for Accident Scenario S3 with CH-TRU and RH-TRU waste.

Impacts to Maximally Exposed Individual

The potential radiological consequences for the MEI would be greatest for LANL and Hanford. For Accident Scenario S1, estimated doses to the MEI would range from 2 x 10^{-7} to 2 x 10^{-5} rem

TEDE, with a 1×10^{-10} to 8×10^{-9} probability of an LCF. Under Accident Scenario S2, doses would range from 4×10^{-7} to 3×10^{-5} rem TEDE, with a 2×10^{-10} to 2×10^{-8} probability of an LCF. Accident Scenario S3 with CH-TRU waste would result in an estimated 0.3 to 10 rem TEDE, with a 1×10^{-4} to 5×10^{-3} probability of an LCF. Accident Scenario S3 with RH-TRU waste would result in an estimated 1×10^{-3} to 0.02 rem TEDE, with a 5×10^{-7} and 1×10^{-5} probability of an LCF, respectively.

The greatest potential hazardous chemical consequences for the MEI would be at LANL under Accident Scenarios S1 and S2 and at Hanford under Accident Scenario S3. Yet, the potential consequences overall would be very small: carcinogenic consequences would be no greater than a 3×10^{-10}, 8×10^{-17}, 6×10^{-7}, and 5×10^{-8} cancer incidence for Accident Scenarios S1, S2, and S3 with CH-TRU waste, and S3 with RH-TRU waste, respectively. Noncarcinogenic impact ratios of 1.0 or greater would indicate the potential for the noncarcinogenic consequence. For Accident Scenario S1, the maximum ERPG-2 ratios to the MEI would be 0.3 at LANL and 0.1 at Hanford, and the maximum IDLH-equivalent ratio would be 1×10^{-6} (see Table G-36). These consequences result from bounding methylene chloride inventories in the involved waste containers. For Accident Scenario S2, the maximum ERPG-2 ratio would be 1×10^{-5}, and the maximum IDLH-equivalent ratio would be 3×10^{-12}. Accident Scenario S3 with CH-TRU and RH-TRU waste calculations show a maximum IDLH-equivalent ratio of 6×10^{-4}. No noncarcinogenic consequences to the MEI would be expected for any of the accident scenarios evaluated.

Impacts to Maximally Exposed Noninvolved Worker

The potential radiological consequences were estimated to be greatest for the Hanford maximally exposed noninvolved worker. Under Accident Scenario S1, doses would range from 1×10^{-5} to 6×10^{-5} rem TEDE, with a 5×10^{-9} to 2×10^{-8} probability of an LCF. Under Accident Scenario S2, doses would range from 3×10^{-5} to 1×10^{-4} rem TEDE, with a 1×10^{-8} to 5×10^{-8} probability of an LCF. Accident Scenario S3 with CH-TRU waste would result in the most serious radiological consequences to the noninvolved worker, with doses ranging from 44 to 95 rem TEDE and a 2×10^{-2} to 4×10^{-2} probability of an LCF.

The potential hazardous chemical consequences to the noninvolved worker would be small: Carcinogenic consequences would be no greater than a 8×10^{-10}, 3×10^{-16}, and 3×10^{-6} cancer incidence for Accident Scenarios S1, S2, and S3 with CH-TRU waste, respectively. Noncarcinogenic consequences for Accident Scenario S1 would have a maximum ERPG-2 ratio of 0.7 at Hanford and 0.6 at LANL, which is attributable to bounding methylene chloride releases. The maximum IDLH-equivalent ratio for Accident Scenario S1 was estimated to be 3×10^{-6}. For Accident Scenario S2, the maximum ERPG-2 ratio is 3×10^{-5}, and the maximum IDLH-equivalent ratio would be 1×10^{-11}. For Accident Scenario S3 with CH-TRU waste, the maximum IDLH-equivalent ratio would be 3×10^{-3}. For Accident Scenario S3 with RH-TRU waste, the maximum IDLH-equivalent ratio would be 2×10^{-4}. Therefore, no noncarcinogenic consequences to the noninvolved worker would be expected from the hazardous chemical releases for the accidents evaluated.

Impacts to Maximally Exposed Involved Worker

Accident consequences were estimated quantitatively for the maximally exposed involved worker only for Accident Scenario S1. The drum drop was assumed to occur 3 meters (10 feet) from the worker and the estimated release of the three involved containers was assumed to expand in a uniform 5-meter (17- foot) radius hemisphere. The worker was assumed to inhale air at the concentrations of the hemisphere for approximately 60 seconds prior to exiting to a fresh air source (DOE 1997). For waste treated by a shred and grout process, the worker would receive a dose of 7.8 rem TEDE, with a 3×10^{-3} probability of an LCF. Carcinogenic consequences from VOC and heavy metal intakes would be a 5×10^{-8} and 2×10^{-11} probability of a cancer incidence, respectively. Releases could have irreversible, non-life-threatening consequences from bounding methylene chloride releases (ERPG-2 ratio of 1.6).

If an involved worker were located next to a drum fire when it erupted (Accident Scenario S2), consequences would be great. However, the probability that this scenario would occur is minute because an action by the worker does not initiate the accident. The smoke would be apparent and the worker can be presumed to exit the facility immediately.

If an involved worker were present when the storage facility collapsed (Accident Scenario S3 with CH-TRU and RH-TRU waste), consequences would be severe. If not killed by falling accident debris, surviving workers could inhale high levels of transuranic radionuclides and hazardous chemicals.

G.4 WIPP DISPOSAL ACCIDENT SCENARIOS

This section describes the accident scenarios evaluated for disposal operations at the WIPP site. Scenarios include accidents previously evaluated in *Final Supplement Environmental Impact Statement for the Waste Isolation Pilot Plant* (SEIS-I) (DOE 1990), and, for the roof fall (Accident Scenario W7), in both SARs (DOE 1995b; DOE 1997). Table G-37 presents the accident scenarios evaluated in this section with their specified SEIS-II identifying numbers and, for those scenarios that were also evaluated in SEIS-I, the corresponding SEIS-I identifier.

Table G-37
WIPP Disposal Accident Scenarios

SEIS-II Accident Scenario	Accident Description	Waste Type	Prior Analysis Reference
W1	Container Drop and Lid Seal Failure in the Waste Handling Building	CH-TRU	SEIS-I C2
W2	Container Puncture, Drop and Lid Seal Failure in the Waste Handling Building	CH-TRU	SEIS-I C3
W3	Container Drop and Lid Seal Failure in the Underground	CH-TRU	SEIS-I C4
W4	Container Puncture, Drop and Lid Seal Failure in the Underground	CH-TRU	SEIS-I C6
W5	Container Fire in the Underground	CH-TRU	SEIS-I C10
W6	Failure of the Waste Shaft Hoist	CH-TRU RH-TRU	SEIS-I C8 none [a]
W7	Roof Fall in a Disposal Room in the Underground	CH-TRU	SAR CH 11
W8	RH-TRU Waste Canister Breach in the Waste Handling Building	RH-TRU	SEIS-I R4

[a] Accident R5 in SEIS-I but no analysis was conducted.

Additional information on the accident scenarios is provided below. The base case descriptions assume that the waste is packaged to meet the minimum planning-basis WAC requirements. Additional cases are added to describe accidents for thermally treated waste (Action Alternative 2 and No Action Alternative 1) and waste treated by the shred and grout process (Action Alternative 3). The consequences of accident scenarios involving CH-TRU waste are reported for waste drum releases. For waste treated to planning-basis WAC, source-term information is also provided for waste contained in standard waste boxes (SWBs). The consequences of an SWB accident can be scaled from the reported consequences for a drum accident (under Proposed Action and Action Alternative 1), using the ratio of the source terms. The only scenario for which an SWB scenario results in a greater release than a drum scenario is the container fire; the SWB release is 160 percent of the drum release. Accident evaluations for all RH-TRU waste assumed that waste would be packaged in drums inside RH-TRU waste canisters.

G.4.1 Inventory

Waste container inventories for radionuclides, heavy metals, and VOCs are the same as the stored waste inventories in Section G.3.1. The quantity of any potentially hazardous constituents in waste to be disposed of at WIPP is limited by the planning-basis WAC. Radionuclides have greater potential consequence than hazardous chemicals; therefore, CH-TRU waste drums may contain no more than 80 PE-Ci, CH-TRU waste standard waste boxes may contain no more than 130 PE-Ci, and RH-TRU waste canisters may contain no more than 1,000 PE-Ci. Radionuclide activities at these limits were assumed to be present in waste containers evaluated for all accidents. Table G-38 shows the average radionuclide content of waste at WIPP, which are the same values shown in Tables G-23 and G-24. Drum headspace concentrations of VOCs would be the same as those shown in Table G-25, and drum heavy metal contents would remain the same as those shown in Table G-27.

Table G-38
Average WIPP Radionuclide Content of CH-TRU Waste and RH-TRU Waste (PE-Ci per waste container) [a]

Proposed Action		Action Alternative 1		Action Alternative 2		Action Alternative 3	
CH-TRU Waste	RH-TRU Waste	CH-TRU Waste	RH-TRU Waste	CH-TRU Waste	RH-TRU Waste	CH-TRU Waste	RH-TRU Waste
3.5	0.3	2.9	3.0	7.7	8.6	2.5	2.5

[a] CH-TRU waste content is measured by PE-Ci per drum-equivalent (or 0.208 cubic meters). RH-TRU waste content is measured by PE-Ci per canister, each containing three drum-equivalents.

G.4.2 WIPP Disposal Accident Analysis

This section presents the accident analyses and estimated source terms for each of the eight WIPP disposal accident scenarios. TRU waste may be stored in either drums or standard waste boxes when treated to the planning-basis WAC. For thermally treated waste or waste treated by a shred and grout process, which has a greater density and mass, these waste forms would only be in drums. Initial analyses showed that potential consequences from accidents involving standard waste boxes could be about 60 percent higher than the same accident involving drums. However, only consequences from waste in drums was analyzed in detail because drums were assumed to contain the majority of waste disposed of underground and are also expected to contain higher concentrations of radionuclides.

Estimates of particulate radionuclides and heavy metals released to the atmosphere outside a facility (the accident source term for members of the public and maximally exposed noninvolved workers) do not consider HEPA filtration of exhaust streams. The Waste Handling Building has continuous HEPA filtration. Therefore, consequences reported for the Waste Handling Building particulate releases (i.e., radioactive materials and heavy metals) may be overestimated by a factor of one thousand or more, assuming a filtration efficiency of at least 99.9 percent. The underground effluent is not normally HEPA-filtered; salt accumulation on the HEPA filter from filtration of the ambient underground air would degrade the filter. A system is in place that would allow air to be routed through HEPA filter banks in the event that radioactive material is detected in the underground exhaust effluent. Recent investigations by the Environmental Evaluation Group (EEG) (Bartlett 1993 and 1996) have questioned the ability of the detection system to operate appropriately because of salt build-up on and degradation of the detection probe. Therefore, excluding consideration of HEPA filtration for underground accidents would encompass potential consequences in the event that the detector fails to function properly. If the underground releases are HEPA-filtered, the consequence estimates from particulate releases would be reduced by a factor of as much as one million.

The entire VOC headspace volume was assumed to be released as a result of container failures for each accident scenario. An average of 0.147 cubic meters (5 cubic feet) of headspace was assumed per drum (DOE 1990). The same headspace volume was assumed for waste treated to planning-basis WAC and for grouted waste. Because the packing efficiency should improve as a result of the packaging and uniform waste matrix, this would be a conservative assumption for grouted waste under Action Alternative 3. For releases from the underground through the Exhaust Filter Building to the atmosphere, it was assumed that none of the VOCs would deposit on interior walls, exhaust ventilation walls, or the HEPA filtration system.

Descriptions of each of the WIPP accident scenarios for each of the three waste forms that may be disposed of underground are presented below. Scenario parameters, summarized in Tables G-39 and G-40, follow the scenario descriptions.

Accident Scenario W1 - Container Drop and Lid Seal Failure in the Waste Handling Building

Under this scenario, a package is dropped from a forklift (either a seven-pack of CH-TRU waste drums or a standard waste box) while being handled in the Waste Handling Building. Because the waste containers are Type A packages, per U.S. Nuclear Regulatory Commission (NRC) requirements, they are designed to withstand a 1-meter (4-foot) drop onto an unyielding surface without damage. However, because the vertical lift can exceed this designed rating, it was assumed that the container drop and subsequent crushing causes the lid of a single container to be knocked off. No inner plastic liner was assumed to be present. A fraction of the respirable-sized particulates in the drum were assumed to be suspended inside the drum during the fall and released when a lid failed. Spilled contents would be released and the respirable particles resuspended from this material. Facility HEPA filtration was not considered for releases to the atmosphere. This method was applied to the quantities of particulate radionuclides and heavy metals (lead, mercury, beryllium, and cadmium) in the waste.

Table G-39
Accident Analysis Parameters for Waste Treated to Planning-Basis WAC for the Proposed Action and Action Alternative 1

SEIS-II Accident Scenario	Accident Description	Parameter Values[a]		
		N	f_{rel}	f_{resp}
W1	Drop, Lid Failure in Waste Handling Building	D: 1 S: 1	0.25	1E-4
W2	Drop, Puncture, Lid Failure in Waste Handling Building	D: 3 S: 2	D: 2 @ 0.1 1 @ 0.25 S: 1 @ 0.1 1 @ 0.25	1E-4
W3	Drop, Lid Failure in Underground	D: 1 S: 1	0.25	1E-4
W4	Drop, Puncture, Lid Failure in Underground	D: 3 S: 2	D: 2 @ 0.1 1 @ 0.25 S: 1 @ 0.1 1 @ 0.25	1E-4
W5	Container Fire in the Underground	D: 1 S: 1	1	5E-4
W6	Hoist Failure CH-TRU Waste	D: 28 S: 4	D: 14 @ 1.0 14 @ 0.1 S: 2 @ 1.0 2 @ 0.1	1E-3
	RH-TRU Waste	1	0.7	5.5E-4
W7	Roof Fall	D: 18 S: 5	0.25	1E-4
W8	RH-TRU Waste Canister Breach	RH: 1	0.01	1E-4

[a] D = drum, S = standard waste box, RH = RH-TRU waste canister

Table G-40
Accident Analysis Parameters for Thermally Treated and Grouted Waste for the Action Alternative 2 Subalternatives and Action Alternative 3

SEIS-II Accident Scenario	Accident Description	Parameter		
		N	f_{rel}	f_{resp}
W1	Drop, Lid Failure in Waste Handling Building	1	0.25	1E-5
W2	Drop, Puncture, Lid Failure in Waste Handling Building	3	1 @ 0.25 2 @ 0	1E-5
W3	Drop, Lid Failure in Underground	1	0.25	1E-5
W4	Drop, Puncture, Lid Failure in Underground	3	1 @ 0.25 2 @ 0	1E-5
W5[a]	Container Fire	1	0.22	6E-5
W6	Hoist Failure CH-TRU Waste RH-TRU Waste	 28 1	 1 0.7	 3E-3 3E-3
W7	Roof Fall	18	0.25	1E-5
W8	RH-TRU Waste Canister Breach	1	0	N/A

[a] Only applies to grouted waste under Action Alternative 3.

N/A = Not Applicable

All VOCs present in the container headspace were assumed to be released, with a total release fraction equal to one. Calculation of material released to the environment assumed that no adsorption, absorption, or plateout onto internal building surfaces would occur.

Workers in the immediate vicinity of the accident were assumed to notice the container breach and exit the work area. The concentration of radionuclide and hazardous chemicals in the air that the worker inhales was determined by assuming that the worker is located 3 meters (10 feet) from the airborne release and the release expands in a uniform 5-meter (17-foot) radius hemisphere from ground level. The involved worker is assumed to breathe at a rate of 3.33×10^{-4} cubic meters per second (1.18×10^{-2} cubic feet per second) (ICRP 1975), which is the male light activity rate, for a period of 60 seconds. The 60-second period represents the length of time it is assumed for the worker to take to stop waste handling (10 seconds), examine the situation and note that containment has been breached (20 seconds), and exit the Waste Handling Building (30 seconds) (DOE 1997).

Case W1a: Waste Treated to Planning-Basis WAC

Either a seven-pack of CH-TRU waste drums or a standard waste box would be handled in the Waste Handling Building. Twenty-five percent of a single container contents were assumed to spill (DOE 1990). The fraction of the spilled contents that would become airborne was assumed to be 0.001 and the respirable fraction was assumed to be 0.1, based on material packaged in a drum that opens due to impact with the floor or falling debris (DOE 1994a). This value applies to the suspension of powder in a can due to debris impact. The total respirable release fraction for particulates would be 2.5×10^{-5}.

Case W1b: Thermally Treated Waste

Only drums would be handled in this accident scenario. The waste was assumed to be a solid vitrified mass, so that only a small amount of particulate material would be released. A respirable resuspension fraction of 1×10^{-5} was used (DOE 1994a), assuming a brittle fracture of the waste mass during the fall. Twenty-five percent of the resuspended material was assumed to be released from the container, so the total respirable release fraction would be 2.5×10^{-6} of the radionuclides and heavy metals in the container. No VOCs would be present in the thermally treated waste.

Case W1c: Waste Treated by a Shred and Grout Process

Only drums would be handled in this accident scenario. The grouted waste is a solid concrete-like mass. A respirable resuspension fraction value of 1×10^{-5} was used (DOE 1994a), assuming brittle fracture of the waste mass during the fall. Twenty-five percent of the respirable resuspended material was assumed to be released from the container, resulting in a total respirable release fraction of 2.5×10^{-6} (the same as for thermally treated waste).

Accident Scenario W2 - Container Puncture, Drop, and Lid Failure in the Waste Handling Building

A Waste Handling Building forklift operator error causes a forklift to strike and puncture either drums or a standard waste box. An additional drum or standard waste box is knocked off and the lid fails. Because the waste containers are Type A packages, per NRC requirements, they are designed to withstand a 1-meter (3.3-foot) drop onto an unyielding surface without damage.

However, because the vertical lift can exceed this designed rating, it was assumed that the container drop and subsequent crushing causes the lid of a single container to be knocked off. No inner plastic liner was assumed to be present. A fraction of the respirable-sized particulates in the drum was assumed to be suspended inside the drum during the fall. A fraction of these would then be released when the lid failed, or the contents may be released and respirable particles resuspended from this material. Facility HEPA filtration was not considered for releases to the atmosphere. This method was applied to the quantities of particulate radionuclides and heavy metals (lead, mercury, beryllium, and cadmium) in the waste.

All VOCs present in the container headspace were assumed to be released, with total release fraction equal to one. Calculation of material released to the environment assumed that no adsorption, absorption, or plateout onto internal building surfaces would occur.

Involved worker intakes were calculated in the same manner as Accident Scenario W1.

Case W2a: Waste Treated to Planning-Basis WAC

Both waste drum and standard waste box accident scenarios were evaluated. For the drum scenario, an error by a Waste Handling Building forklift operator causes a forklift to strike and puncture two drums. As a result of the impact, a third drum falls from the stack and its lid is knocked off upon impact with the floor. It was assumed that 10 percent of the waste spills out of the two punctured drums and 25 percent spills out of the lidless drum. For the standard waste box scenario, one standard waste box is punctured, and the one stacked above it falls to the ground. The lid seal of the fallen standard waste box is assumed to fail as a result of the impact and 25 percent of the contents are released. As in Accident Scenario W1 for both drums and standard waste boxes, 0.001 of the spilled fraction was assumed to be resuspended in the room air and 0.1 of this resuspended fraction was assumed to be respirable.

Case W2b: Thermally Treated Waste

Only drums were assumed to be used under this treatment option. Consequences from particulate releases of Case W2b would be identical to those of Case W1b, with material released only from the drum with lid failure. No releases of particulates from the punctured drums would occur. No VOCs would be present in thermally treated waste.

Case W2c: Waste Treated by Shred and Grout Process

Only drums were assumed to be used under this treatment option. Consequences from particulate releases for Case W2c would be identical to those of Case W1c, with material released only from the drum with lid failure. No particulate releases from the punctured containers would occur. Headspace VOCs from all three of the damaged drums were assumed to be released, so VOC-related consequences would be three times higher than for Case W1c.

Accident Scenario W3 - Container Drop and Lid Seal Failure in the Underground

Accident Scenario W3 and its container releases are identical to those described for Accident Scenario W1 except that the accident occurs underground. Particulate radionuclide and heavy metal releases to the atmosphere from the underground ventilation system would likely be reduced compared to similar Waste Handling Building releases because of particle depletion and plateout over the long distance between the underground location of the incident and the aboveground exhaust point to release at the Exhaust Filter Building. However, plateout was not considered.

All VOCs present in the container headspace were assumed to be released, with a total release fraction equal to one. Calculation of material released to the environment assumed that no adsorption, absorption or plateout onto internal building surfaces would occur.

Involved worker intakes underground were calculated in the same manner as for Accident Scenario W1. The worker is assumed to require 60 seconds to move into the upstream ventilation air. This underground worker exposure is extremely conservative given the high ventilation rate of the underground air.

Case W3a: Waste Treated to Planning-Basis WAC

Case W3a is identical to Case W1a. Depletion of the source term would be expected but was not considered. As a result, releases from Case W3a are identical to those of Case W1a.

Case W3b: Thermally Treated Waste

Case W3b is identical to Case W1b. Depletion of the source term would be expected but was not considered. As a result, releases from Case W3b are identical to those of Case W1b.

Case W3c: Waste Treated by a Shred and Grout Process

Case W3c is identical to Case W1c. Depletion of the source term would be expected but was not considered. As a result, releases from Case W3c are identical to those of Case W1c.

Accident Scenario W4 - Container Puncture, Drop, and Lid Seal Failure in the Underground

Accident Scenario W4 and its container releases are identical to those described for Accident Scenario W2 except that the accident occurs underground. All VOCs present in the container headspace were assumed to be released with a total release fraction equal to one. Calculation of VOCs released to the environment assumed that no adsorption, absorption, or plateout onto internal building surfaces would occur.

Involved worker intakes underground were calculated in the same manner as for Accident Scenario W3.

Case W4a: Waste Treated to Planning-Basis WAC

Case W4a is identical to Case W2a. Depletion of the source term would be expected but was not considered. As a result, releases from Case W4a are identical to those of Case W2a.

Case W4b: Thermally Treated Waste

Case W4b is identical to Case W2b. Depletion of the source term would be expected but was not considered. As a result, releases from Case W4a are identical to those of Case W2b.

Case W4c: Grouted Waste

Case W4c is identical to Case W2c. Depletion of the source term would be expected but was not considered. As a result, releases from Case W4c are identical to those of Case W2c.

Accident Scenario W5 - Container Fire in the Underground

A fire was assumed to start inside a closed waste container and involve only the single container. Only a fire in the underground was evaluated because of the relatively short period of time any one drum would be present in the Waste Handling Building. Released particulates would likely be subject to a high amount of deposition due to the heated aerosol reacting with the relatively cool surfaces within the facility (DOE 1990). However, depletion was not considered in the analysis. The VOCs in the waste container were assumed to be consumed by the fire. Because an individual worker's actions do not initiate the accident, involved worker exposures were not calculated.

The EEG investigated the history of hazardous waste drum fires, explosions, and other pressurizations at DOE facilities (Silva 1991). The incidents were attributed to the discharge of static electricity, spontaneous ignition of pyrophoric materials, and reactions involving nitric acid. Other contributing circumstances included the drums being painted black, exposure to direct sunlight, and improper packaging material. At WIPP, such incidents would be limited by the fact that containers would be painted white, would not be exposed to direct sunlight, and would be certified to the planning-basis WAC, which limits combustion initiators in the waste containers.

Case W5a: Waste Treated to Planning-Basis WAC

Waste drum and standard waste box accident scenarios were evaluated. It was assumed that 5×10^{-4} of the radioactive and heavy metals would be resuspended as respirable particles, a conservative value of the fraction of respirable particles expected from the burning of either combustible or noncombustible materials (DOE 1994a).

Case W5b: Thermally Treated Waste

There would be no combustible materials in the waste after thermal treatment, so a fire would be an unlikely scenario.

Case W5c: Waste Treated by a Shred and Grout Process

Only drums were assumed to be used under this treatment option. Although the waste would be uniformly mixed within a noncombustible grout matrix, accident consequences were calculated for this scenario. Twenty-two percent of a concrete mass would be transformed to a powder as a result of fire temperatures of at least 650 degrees Celsius (1,200 degrees Fahrenheit) (DOE 1994a); therefore, it was assumed that 22 percent of the grouted waste would be transformed to a powder as a result of a container fire. A respirable resuspension fraction of 6.0×10^{-5} for particulates was used (DOE 1994a), which reflects the fraction of heated powder resuspended by the vapor flux generated by an open fire.

Accident Scenario W6 - Failure of the Waste Shaft Hoist

The waste hoist braking system was assumed to fail when the hoist was fully loaded and at the top of the shaft. The hoist was assumed to fall 655 meters (2,150 feet) to the bottom of the waste hoist shaft. Depending on its physical form, material dropped from great heights will generally exhibit plastic properties or will shatter on impact. The respirable fraction differs for each type of material; for example, material that shatters produces the greatest quantity of respirable-sized articles. The maximum estimated respirable resuspended particulate fraction for a brittle material that drops 655 meters (2,150 feet) is 3×10^{-3} (DOE 1994a). The waste hoist was assumed to be fully loaded with either 28 drums or four standard waste boxes containing CH-TRU waste or one RH-TRU waste canister.

Releases of particulate radionuclides and heavy metals to the atmosphere from the underground ventilation system would likely be reduced because of particle depletion and plateout while traveling over the long distance between the underground location of the incident and the aboveground exhaust point to release at the Exhaust Filter Building (DOE 1990). However, plateout was not considered in the analysis. All headspace VOCs in CH-TRU and RH-TRU waste were assumed to be released, with a total release fraction equal to one.

This accident would require simultaneous failure of six hoisting cables or loss of power and failure of the hoist braking system. The probability of failure of the waste shaft hoist has been investigated by EEG (Greenfield and Sargent 1995). The most critical element for this accident was determined to be failure of the hoist hydraulic brake system. The 95th percentile annual probability of this incident occurring is 4.5×10^{-7}, updating the previous 1×10^{-4} to 1×10^{-6} annual probability estimate, which resulted from the absence of preoperational checks of the hoist system at the start of each shift in WIPP operating procedures. Because of this low probability, this accident scenario is comparable to the beyond-design-basis earthquakes evaluated for treatment and storage.

Case W6a: Waste Treated to Planning-Basis WAC

The source terms resulting from both waste drums and standard waste boxes containing CH-TRU waste were evaluated. All radionuclides and heavy metals were assumed to be released from half of the waste containers upon impact and to completely escape the accident debris for a release fraction of one. Material in the other half of the CH-TRU waste containers involved in the accident was assumed to have an overall release fraction of 0.1, with 90 percent of the material either not released or contained within the accident debris. The particulate resuspension fraction for all of the released radionuclides and heavy metals was assumed to be 1×10^{-3}, and all of this material was assumed to be of respirable size.

Under the same accident scenario, 70 percent of the radionuclides and heavy metals in RH-TRU waste would be released, with 30 percent contained by the accident debris at the bottom of the shaft. The released radionuclides and heavy metals were assumed to have a resuspension fraction of 5.5×10^{-4} (half of the waste having a resuspension fraction of 1×10^{-4} and the other half of the waste having a resuspension fraction of 1×10^{-3}).

Case W6b: Thermally Treated Waste

Only waste drums were assumed to be used under this treatment option. Vitrified CH-TRU waste would shatter on impact and all radionuclides and heavy metals would be released. The bounding respirable resuspension fraction of 3×10^{-3} for a shattering material was assumed. No VOCs would be present in thermally treated waste.

Under the same accident scenario, 70 percent of radionuclides and heavy metals in RH-TRU waste would be released, with 30 percent contained by the accident debris at the bottom of the shaft. The released radionuclides and heavy metals were assumed to have the bounding respirable resuspension fraction of 3×10^{-3}. No VOCs would be present in thermally treated waste.

Case W6c: Waste Treated by a Shred and Grout Process

For particulate radionuclides and heavy metals in CH-TRU and RH-TRU waste, the analysis would be identical to that of Case W6b, assuming the grouted waste would shatter on impact. Grouted waste containing VOCs and the entire headspace volume were assumed to be released.

Accident Scenario W7 - Roof Fall in a Disposal Room in the Underground

A portion of the roof in a disposal room of a waste panel was assumed to fall during waste emplacement. Roofs are more likely to fall when panels have been open a long time. This accident would have a higher probability of occurring in Panel 1, because it would have been open the longest. The roof fall scenario was based on the same scenario evaluated in the WIPP SAR (DOE 1995b). Although DOE subsequently updated this analysis (DOE 1997), the results of the SEIS-II analyses are more conservative and have been retained.

The CH-TRU waste containers in disposal rooms would be stacked three high with a maximum of five groups of seven-packs across the width of the room. The drums directly under the fallen roof section were assumed to be crushed. The crushed drums shift and deform adjoining waste stacks, and several of the stacks at the working end of the emplacement operations would fall. The roof collapse was assumed to occur when a disposal room was more than half full, and the roof section that falls was assumed to crush the equivalent of half of a disposal room of drums (approximately 2,100). All five seven-packs (35 drums total) on the upper level of the working end of the stack were assumed to fall to the panel floor. Lid failure was assumed for half of the drums in the seven-packs (18 drums) as a result of the drop. The seven-packs in the bottom two levels of the stack were assumed to shift and deform, but not fall or breach. The RH-TRU waste, placed in the panel walls, would not be affected by a roof fall. The particulates in crushed drums were assumed to be contained by the fallen roof section and not be released. The fallen roof section was assumed not to be thick enough to halt the ventilation flow through the disposal room and panel. No facility HEPA filtration of particulate releases was considered. All VOCs in the crushed container headspace were assumed to be released, with a total release fraction equal to one.

Case W7a: Treatment of Waste to Planning-Basis WAC

Drum and standard waste box accident scenarios were evaluated, with either 18 drums or five standard waste boxes breached. Twenty-five percent of the waste was assumed to spill out of the fallen drums, with a respirable resuspension fraction of 1×10^{-4} of the radioactive materials and heavy metals.

Case W7b: Thermally Treated Waste

Only drums would be emplaced, and only about half of the volume of CH-TRU waste of the other alternatives can be disposed of in each panel because of thermal power limitations. Therefore, a smaller total volume of waste containers would be impacted by the roof fall. This assumption does not affect accident consequences because the number of breached containers at the working end of the stack was assumed to be the same as described above, with 18 drums breached. A respirable resuspension fraction of 1×10^{-5} was used (DOE 1994a), assuming a brittle fracture of the waste mass during the fall. Twenty-five percent of the resuspended material was assumed to be released from the container, so the total respirable release fraction would be 2.5×10^{-6} of the radionuclides and heavy metals in the container. No VOCs would be present in the thermally treated waste.

Case W7c: Waste Treated by a Shred and Grout Process

Only drums would be emplaced, and 18 drums were assumed to breach. A respirable resuspension fraction value of 1×10^{-5} was used (DOE 1994a), assuming brittle fracture of the waste mass during the fall. Twenty-five percent of the respirable resuspended material was assumed to be released from the container, resulting in a total respirable release fraction of 2.5×10^{-6}, the same as for thermally treated waste. All container headspace VOCs were assumed to be released.

Accident Scenario W8 - RH-TRU Waste Canister Breach in the Waste Handling Building

This scenario is similar to one evaluated in SEIS-I (DOE 1990). An RH-TRU waste canister is dropped into the transfer cell from the hot cell (a distance of 11 meters [36 feet]) when a grapple fails. It was assumed that the canister is breached in the fall. No facility HEPA filtration was considered.

All VOCs in the container headspace were assumed to be released, with a total release fraction of 1.0.

No consequences to involved workers were calculated because the accident is assumed to occur inside the shielded transfer cell.

Case W8a: Waste Treated to Planning-Basis WAC

One percent of the waste spills from the canister and 1×10^{-4} of the spilled material is resuspended in the room air as respirable particulates.

Case W8b: Thermally Treated Waste

No particulates would be spilled from the breached container because of the solidified waste form. No VOCs would be present in the thermally treated waste.

Case W8c: Waste Treated by a Shred and Grout Process

Accident Scenario Parameter Summary

The radionuclide and heavy metal source term for each accident was calculated using Equation G-6, shown below. The source term, S, was specified in terms of the number of respirable PE-Ci that would be released for radionuclides and as the number of kilograms released. No particulates would be spilled from the breached container because of the solidified waste form. All of the headspace VOCs would be released.

For many of the accidents, use of this equation and these parameters may be conservative for lead or mercury, which can be highly malleable and less subject to particulate rupture as a result of accident impact shocks.

$$S = (N)(Q)(f_{rel})(f_{resp}) \quad \text{(Equation G-6)}$$

where

S = source term (PE-Ci or kilograms)

N = number of containers involved

Q = radionuclide or hazardous material inventory of a waste container (PE-Ci or kilograms)

f_{rel} = fraction of the contents released from the container

f_{resp} = fraction of released contents that becomes airborne as a respirable-sized particle

Parameter values are described in the accident scenario descriptions. The parameter values used to estimate the radionuclide and heavy metal source terms are presented in Table G-39 for the Proposed Action and Action Alternative 1 and in Table G-40 for Action Alternatives 2 and 3. In Table G-39, parameter values for both drums and standard waste boxes are presented although consequence estimates are presented in Section G.4.3 only for waste in drums. Consequences may be slightly higher for standard waste box accident scenarios, but consequences for waste stored in drums are a more appropriate comparison for consequences of Action Alternatives 2 and 3.

The radionuclide accident source term (in PE-Ci) following release from the stack is presented in Table G-41. The quantity of radionuclides that would be inhaled by an involved worker, also in PE-Ci, is presented in Table G-42. No analyses were performed to estimate consequences to involved workers for Accident Scenarios W5, W6, W7, and W8, because consequences could range from negligible to catastrophic. The range of consequences is discussed further in Section G.4.3.

G.4.3 Consequences of WIPP Disposal Accidents

Consequences from WIPP disposal accidents were calculated for the exposed off-site population around WIPP, the MEI, the maximally exposed noninvolved worker, and the maximally exposed involved worker. Acute releases were assumed to be dispersed in one direction, so population consequences were estimated for a single, maximally exposed 22.5-degree sector (out to 80 kilometers [50 miles]) and not for the entire 80-kilometer (50-mile) region population. Population-weighted atmospheric dispersion values were calculated and used to determine the maximally impacted sector, considering both the change in air concentration over distance and the population distribution in the sectors of various distances. The population west of the WIPP site, including the nearest population center at Carlsbad, New Mexico, would receive the greatest

Table G-41
Source Terms for Off-site Releases from WIPP Disposal Accidents (PE-Ci) [a]

SEIS-II Accident Scenario	Accident Description	Proposed Action	Action Alternative 1	Action Alternative 2 Subalternatives	Action Alternative 3
W1 or W3	Drop, Lid Failure in Waste Handling Building or Underground	2E-3 3.25E-3 (S)	2E-3 3.25E-3 (S)	2E-4	2E-4
W2 or W4	Drop, Puncture, Lid Failure in Waste Handling Building or Underground	3.6E-3 4.55E-3 (S)	3.6E-3 4.55E-3 (S)	2E-4	2E-4
W5	Container Fire	4E-2 6.5E-2 (S)	4E-2 6.5E-2 (S)	N/A	1.1E-3
W6	Hoist Failure CH-TRU Waste RH-TRU Waste	 1.2 0.5 (S) 0.3	 1.2 0.5 (S) 0.3	 6.7 2.1	 6.7 2.1
W7	Roof Fall	0.036 0.016 (S)	0.036 0.016 (S)	3.6E-3	3.6E-3
W8	RH-TRU Waste Canister Breach	0.001	0.001	0	0

[a] Releases are those resulting from scenarios involving drums unless otherwise noted with an "(S)", indicating SWB scenarios.

N/A = Not Applicable

Table G-42
Involved Worker Intakes from WIPP Disposal Accidents (PE-Ci)

SEIS-II Accident Scenario	Accident Description	Proposed Action	Action Alternative 1	Action Alternative 2 Subalternatives	Action Alternative 3
W1	Drop, Lid Failure in Waste Handling Building	1.5E-7	1.5E-7	1.5E-8	1.5E-8
W2	Drop, Puncture, Lid Failure in Waste Handling Building	2.7E-7	2.7E-7	1.5E-8	1.5E-8
W3	Drop, Lid Failure in Underground	1.5E-7	1.5E-7	1.5E-8	1.5E-8
W4	Drop, Puncture, Lid Failure in Underground	2.7E-7	2.7E-7	1.5E-8	1.5E-8

potential consequence from an accidental WIPP release. The MEI was assumed to be at the WIPP fenceline location where the atmospheric dispersion would be minimized, resulting in maximum air concentrations. This location is 300 meters (1,000 feet) south of the Exhaust Filter Building at the Exclusive Use Area boundary.

The same location was used to calculate consequences to the maximally exposed noninvolved worker because this would be the point of highest consequence of any on-site location. The maximally exposed worker is a worker who handles the waste directly. All accident evaluations assumed the noninvolved worker, MEI, and population remained in the plume during its entire time of passage. Impacts from the inhalation pathway dominate those of all other exposure pathways.

The estimated annual frequencies of occurrence for WIPP disposal accidents (Accident Scenarios W1 through W8) are presented in Table G-43. When a frequency range was identified, the highest value (i.e., the greater frequency) is presented in the table. Occurrence frequencies for the Action Alternative 2 subalternatives were assumed to be half of the values for waste treated to planning-basis WAC and waste treated by a shred and grout process because waste would not need to be stacked as high in the underground, thereby reducing the probability of lid failure in the event of a container drop. The mass of waste containers under the Action Alternative 2 subalternatives is potentially at the maximum drum limit of 450 kilograms (1,000 pounds). This may affect the likelihood of an accident, but this effect was not considered.

Table G-43
Annual Frequencies of Occurrence for WIPP Disposal Accidents

SEIS-II Accident Scenario	Accident Description	Proposed Action[a]	Action Alternative 1[b]	Action Alternative 2 Subalternatives[b]	Action Alternative 3[b]
W1	Drop, Lid Failure in Waste Handling Building	0.01	0.01	5E-3	0.01
W2	Drop, Puncture, Lid Failure in Waste Handling Building	0.01	0.01	5E-3	0.01
W3	Drop, Lid Failure in Underground	0.01	0.01	5E-3	0.01
W4	Drop, Puncture, Lid Failure in Underground	0.01	0.01[a]	5E-3[a]	0.01[a]
W5	Container Fire: < 8PE-Ci Container Fire: > 8PE-Ci	1E-4 < 1E-6	1E-4 < 1E-6	N/A	1E-4 < 1E-6
W6	Hoist Failure	< 1E-6[c]	< 1E-6	< 1E-6	< 1E-6
W7	Roof Fall: Panel 1 Roof Fall: other panels	0.01[d] < 1E-6	0.01 < 1E-6	5E-3 < 1E-6	0.01 < 1E-6
W8	RH-TRU Waste Canister Breach	1E-4 to 1E-6 [d]	1E-4 to 1E-6	5E-5 to 5E-7	1E-4 to 1E-6

[a] Taken from the1996 Final WIPP SAR (DOE 1997) except where noted.
[b] The same as or extrapolated from the Proposed Action value.
[c] Taken from EEG-59 (Greenfield and Sargent 1995).
[d] Taken from the WIPP SAR (DOE 1995b).

N/A = Not Applicable

G.4.3.1 Accident Consequences for Proposed Action and Action Alternative 1

Consequences of accidents under the Proposed Action and Action Alternative 1 would be the same because waste is treated to planning-basis WAC. There would be no expected cancer incidence in the exposed population and very low probabilities of cancer to exposed individuals from VOCs and heavy metals for all accidents. Only one accident, the hoist failure (Accident Scenario W6), would result in radiation-related LCFs in the exposed population around WIPP. Up to five LCFs could occur in an exposed sector-population from a hoist failure involving CH-TRU waste, and two LCFs could occur from a hoist failure involving RH-TRU waste. The probability of an LCF to the MEI would be 0.08 for the CH-TRU waste accident scenario, and 0.03 for the RH-TRU waste accident scenario. Table G-44 presents the radiological consequences from WIPP disposal accidents under the Proposed Action and Action Alternative 1.

Table G-44
Radiological Consequences of WIPP Disposal Accidents for the Proposed Action and Action Alternative 1

SEIS-II Accident Scenario	Accident Description	Population (person-rem, LCFs)		MEI (rem, probability of LCF)		Maximally Exposed Noninvolved Worker (rem, probability of LCF)		Maximally Exposed Involved Worker (rem, probability of LCF)
W1	Drop, Lid Failure in Waste Handling Building	17 9E-3		0.26 1E-4		0.26 1E-4		7.8 0.03
W2	Drop, Puncture, Lid Failure in Waste Handling Building	31 0.02		0.47 2E-4		0.47 2E-4		140 0.06
W3	Drop, Lid Failure in Underground	17 9E-3		0.26 1E-4		0.26 1E-4		78 0.03
W4	Drop, Puncture, Lid Failure in Underground	31 0.02		0.47 2E-4		0.47 2E-4		140 0.06
W5	Container Fire	565 0.3		8.5 4E-3		8.5 3E-3		N/A
W6	Hoist Failure	CH-TRU 11,000 5	RH-TRU 3,400 2	CH-TRU 160 0.08	RH-TRU 50 0.03	CH-TRU 160 0.06	RH-TRU 50 0.02	See Text
W7	Roof Fall	310 0.2		4.7 2E-3		4.7 2E-3		See Text
W8	RH-TRU Waste Canister Breach	9 5E-3		0.13 7E-5		0.13 5E-5		0.0 0

N/A = Not Applicable

Potential carcinogenic consequences from exposure to VOCs and heavy metals released during WIPP disposal accidents are presented in Table G-45. As noted above, no incidences of cancer would be expected in the exposed population, and consequences for individuals would be very low.

Potential noncarcinogenic consequences from exposure to VOCs and heavy metals released during an accident are presented in Tables G-46 and G-47. No noncarcinogenic health effects would be expected. The majority of the small noncarcinogenic impact is from tetrachloroethene and 1,1,2,2-tetrachloroethane releases. No life-threatening impacts are expected based on the IDLH-equivalent intake analyses. Some impacts are expected based on the ERPG analyses.

The highest air concentration to which the MEI, maximally exposed noninvolved worker, and maximally exposed involved worker could be exposed were estimated and compared to ERPG values (see Table G-3). The ratios of the air concentration and the ERPG-2 value are presented in Table G-47. Heavy metal and VOC releases of hoist failure and roof fall scenarios potentially adversely affect the MEI and noninvolved workers based on the ERPG analyses. Some VOC releases from scenarios W2 and W4 could adversely affect the involved worker.

Bounding consequence estimates to the MEI and noninvolved worker for the CH hoist failure scenario (W6-CH) could be severe. Bounding methylene chloride and beryllium releases could seriously threaten these individuals (ERPG-3 ratios of 1 and 3, respectively). Releases of 1,1,2,2-tetrachloroethane, lead, and mercury could have irreversible, non-life-threatening consequences (ERPG-2 ratios of 3, 54, and 58, respectively). These consequence estimates reflect upper bound consequences. It is unlikely that the individuals would be exposed to the high air concentrations for the length of time required to experience such severe effects. Mild transient consequences might be experienced from carbon tetrachloride releases (ERPG-1 ratio of 1).

Table G-45
Hazardous Chemical Carcinogenic Consequences of WIPP Disposal Accidents for the Proposed Action and Action Alternative 1

SEIS-II Accident Scenario	Accident Description	Hazardous Constituents	Population (cancers)	MEI (probability of cancer)	Maximally Exposed Noninvolved Worker (probability of cancer)	Maximally Exposed Involved Worker (probability of cancer)
W1	Drop, Lid Failure in Waste Handling Building	VOCs Metals	2E-8 4E-11	2E-10 6E-13	2E-10 6E-13	9E-8 2E-10
W2	Drop, Puncture, Lid Failure in Waste Handling Building	VOCs Metals	5E-8 8E-11	7E-10 1E-12	7E-10 1E-12	3E-7 4E-10
W3	Drop, Lid Failure in Underground	VOCs Metals	2E-8 4E-11	2E-10 6E-13	2E-10 6E-13	9E-8 2E-10
W4	Drop, Puncture, Lid Failure in Underground	VOCs Metals	5E-8 8E-11	7E-10 1E-12	7E-10 1E-12	3E-7 4E-10
W5	Container Fire	VOCs Metals	0.0 2E-10	0.0 2E-12	0.0 2E-12	N/A
W6	Hoist Failure (CH-TRU Waste)	VOCs Metals	54E-7 3E-8	6E-9 4E-10	6E-9 4E-10	See Text
W6	Hoist Failure (RH-TRU Waste)	VOCs Metals	5E-8 4E-9	7E-10 6E-11	7E-10 6E-11	See Text
W7	Roof Fall	VOCs Metals	1E-6 8E-10	2E-8 1E-11	2E-8 1E-11	See Text
W8	RH-TRU Waste Canister Breach	VOCs Metals	5E-8 5E-12	7E-10 7E-14	7E-10 7E-14	3E-7 0

N/A = Not Applicable

Table G-46
Hazardous Chemical Noncarcinogenic Consequences of WIPP Disposal Accidents for the Proposed Action and Action Alternative 1

SEIS-II Accident Scenario	Accident Description	Hazardous Constituents	MEI (maximum IDLH-equivalent ratio)	Maximally Exposed Noninvolved Worker (maximum IDLH-equivalent ratio)	Maximally Exposed Involved Worker (maximum IDLH-equivalent ratio)
W1	Drop, Lid Failure in Waste Handling Building	VOCs Metals	5E-7 2E-8	5E-7 2E-8	2E-4 8E-6
W2	Drop, Puncture Lid Failure in Waste Handling Building	VOCs Metals	1E-6 4E-8	1E-6 4E-8	5E-4 1E-5
W3	Drop, Lid Failure in Underground	VOCs Metals	5E-7 2E-8	5E-7 2E-8	2E-4 8E-6
W4	Drop, Puncture, Lid Failure in Underground	VOCs Metals	1E-6 4E-8	1E-6 4E-8	5E-4 E-5
W5	Container Fire	VOCs Metals	0.0 9E-8	0.0 9E-8	N/A
W6	Hoist Failure (CH-TRU Waste)	VOCs Metals	1E-5 1E-5	1E-5 1E-5	See Text
W6	Hoist Failure (RH-TRU Waste)	VOCs Metals	1E-6 2E-6	1E-6 2E-6	See Text
W7	Roof Fall	VOCs Metals	2E-5 4E-7	2E-5 4E-7	See Text
W8	RH-TRU Waste Canister Breach	VOCs Metals	1E-6 3E-9	1E-6 3E-9	0 0

N/A = Not Applicable

Table G-47
Ratios of Exposure Air Concentrations to ERPG-2 Values for WIPP Disposal Accidents for the Proposed Action and Action Alternative 1 [a]

SEIS-II Accident Scenario	Accident Description	Hazardous Constituents	MEI and Maximally Exposed Noninvolved Worker: Air Concentration to ERPG-2 Ratio	Maximally Exposed Involved Worker: Air Concentration to ERPG-2 Ratio
W1	Container Drop and Lid Seal Failure	VOC Metals	0.1 0.1	0.5 0.01
W2	Container Puncture, Crop and Lid Seal Failure	VOC Metals	0.4 0.2	2 0.7
W3	Container and Lid Seal Failure	VOC Metals	0.1 0.1	0.5 0.01
W4	Container Puncture, Drop, and Lid Seal Failure	VOC Metals	0.4 0.2	2 0.7
W5	Container Fire	Metals	2	N/A
W6	Failure of the Waste Shaft Hoist (CH-TRU Waste)	VOC Metals	3 58	See Text
W6	Failure of the Waste Shaft Hoist (RH-TRU Waste)	VOC Metals	0.4 9	See Text
W7	Roof Fall	VOC Metals	18 2	See Text
W8	RH-TRU Waste Canister Breach	VOC Metals	0.4 0.01	0 0

[a] These exposure air concentration to ERPG-2 ratios are the highest of any individual hazardous chemical.

N/A = Not Applicable

The RH hoist failure scenario (W6-RH) consequences to the MEI and noninvolved worker could result in irreversible, non-life-threatening consequences from bounding beryllium, lead, and mercury releases (ERPG-2 ratios of 2, 8, and 9, respectively).

Consequences to the MEI and noninvolved worker could result from the roof fall scenario (W7). Irreversible, non-life-threatening consequences from 1,1,2,2-tetrachloroethane, lead, and mercury releases could result (ERPG-2 ratios of 18, 2, and 2). Mild transient consequences could result from carbon tetrachloride releases (ERPG-1 ratio of 2).

The hoist failure and roof fall scenarios could have consequences ranging from negligible to lethal effects for any involved worker in the underground disposal area. Typically, four underground workers are involved in emplacement operations. Some or all of these workers could be killed if they were in the immediate area when the accidents occurred. ERPG ratios were estimated for involved workers for Accident Scenarios W1 through W4. Accident Scenarios W2 and W4 result in the same exposures. Irreversible, non-life-threatening consequences from bounding methylene chloride and 1,1,2,2-tetrachloroethane releases could result from these accidents (ERPG-2 ratios of 2 for both).

G.4.3.2 Accident Consequences for the Action Alternative 2 Subalternatives

The thermally treated waste form in the Action Alternative 2 subalternatives would reduce the consequences of WIPP disposal accidents. There would be no expected cancer incidence in the exposed population and very low probabilities of cancer to exposed individuals from heavy metals for all accidents (there would be no VOCs in thermally treated waste). As for the Proposed Action and Action Alternative 1, only the hoist failure (Accident Scenario W6) would result in radiation-related LCFs in the exposed population around WIPP. The radiological consequence is greater for the Action Alternative 2 subalternatives than for the Proposed Action and Action Alternative 1 because the thermally treated waste is more likely to completely brittle-fracture upon impact and create a larger quantity of respirable particles than is the waste treated to planning-basis WAC. Up to 29 LCFs could occur in an exposed sector-population from a hoist failure involving CH-TRU waste, and 9 LCFs could occur from a hoist failure involving RH-TRU waste. The probability of an LCF to the MEI would be 0.6 for the CH-TRU waste accident scenario and 0.1 for the RH-TRU waste accident scenario. Table G-48 presents the radiological consequences from WIPP disposal accidents under Action Alternative 2.

Potential carcinogenic consequences from exposure to VOCs and heavy metals released during WIPP disposal accidents are presented in Table G-49. As noted above, no cancers would be expected in the exposed population, and consequences to individuals would be very low.

Potential noncarcinogenic consequences from exposure to heavy metals released during an accident are presented in Tables G-50 and G-51. No life-threatening consequences are expected based on the IDLH-equivalent intake analyses, although some consequences are expected based on the ERPG analyses.

Table G-48
Radiological Consequences of WIPP Disposal Accidents for the Action Alternative 2 Subalternatives

SEIS-II Accident Scenario	Accident Description	Population (person-rem, LCFs)		MEI (rem, probability of LCF)		Maximally Exposed Noninvolved Worker (rem, probability of LCF)		Maximally Exposed Involved Worker (rem, probability of LCF)
W1	Drop, Lid Failure in WHB	1.7 9E-4		0.026 1E-5		0.026 1E-5		7.8 3E-3
W2	Drop, Puncture, Lid Failure in WHB	1.7 9E-4		0.026 1E-5		0.026 1E-5		7.8 3E-3
W3	Drop, Lid Failure in Underground	1.7 9E-4		0.026 1E-5		0.026 1E-5		7.8 3E-3
W4	Drop, Puncture, Lid Failure in Underground	1.7 9E-4		0.026 1E-5		0.026 1E-5		7.8 3E-3
W5	Container Fire	N/A		N/A		N/A		N/A
W6	Hoist Failure	CH-TRU 58,000 29	RH-TRU 18,000 9	CH-TRU 870 0.6 [a]	RH-TRU 270 0.1	CH-TRU 870 0.5 [a]	RH-TRU 270 0.1	See Text
W7	Roof Fall	31 0.02		0.47 2E-4		0.47 2E-4		See Text
W8	RH-TRU Waste Canister Breach	0		0		0		0

[a] Maximum annual dose is 37 rem and remains above 20 rem per year for the first 11 years; therefore, annual carcinogenic risk for the first 11 years is calculated without using the low dose-rate dose-reduction factor of two.

N/A = Not Applicable

Table G-49
Heavy Metal Carcinogenic Consequences of WIPP Disposal Accidents for the Action Alternative 2 Subalternatives

SEIS-II Accident Scenario	Accident Description	Population (cancers)	MEI (probability of cancer)	Maximally Exposed Noninvolved Worker (probability of cancer)	Maximally Exposed Involved Worker (probability of cancer)
W1	Drop, Lid Failure in WHB	1E-11	2E-13	2E-13	6E-11
W2	Drop, Puncture, Lid Failure in WHB	1E-11	2E-13	2E-13	6E-11
W3	Drop, Lid Failure in Underground	1E-11	2E-13	2E-13	6E-11
W4	Drop, Puncture, Lid Failure in Underground	1E-11	2E-13	2E-13	6E-11
W5	Container Fire	N/A	N/A	N/A	N/A
W6	Hoist Failure (CH-TRU Waste)	4E-7	5E-9	5E-9	See Text
W6	Hoist Failure (RH-TRU Waste)	3E-8	4E-10	4E-10	See Text
W7	Roof Fall	2E-10	3E-12	3E-12	See Text
W8	RH-TRU Waste Canister Breach	0	0	0	0

N/A = Not Applicable

Table G-50
Heavy Metal Noncarcinogenic Consequences of WIPP Disposal Accidents for the Action Alternative 2 Subalternatives

SEIS-II Accident Scenario	Accident Description	MEI (maximum IDLH-equivalent ratio)	Maximally Exposed Noninvolved Worker (maximum IDLH equivalent ratio)	Maximally Exposed Involved Worker (maximum IDLH-equivalent ratio)
W1	Drop, Lid Failure in WHB	6E-9	6E-9	2E-6
W2	Drop, Puncture, Lid Failure in WHB	6E-9	6E-9	2E-6
W3	Drop, Lid Failure in Underground	6E-9	6E-9	2E-6
W4	Drop, Puncture, Lid Failure in Underground	6E-9	6E-9	2E-6
W5	Container Fire	N/A	N/A	N/A
W6	Hoist Failure (CH-TRU Waste)	2E-4	2E-4	See Text
W6	Hoist Failure (RH-TRU Waste)	1E-5	1E-5	See Text
W7	Roof Fall	1E-7	1E-7	See Text
W8	RH-TRU Waste Canister Breach	0	0	0

N/A = Not Applicable

Table G-51
Ratios of Exposure Air Concentrations
to ERPG-2 Values for the Action Alternative 2 Subalternatives [a]

SEIS-II Accident Scenario	Hazardous Constituents	MEI and Maximally Exposed Noninvolved Worker: Air Concentration to ERPG-2 Ratio	Maximally Exposed Involved Worker: Air Concentration to ERPG-2 Ratio
W1	Metals	0.1	0.1
W2	Metals	0.02	0.1
W3	Metals	0.1	0.1
W4	Metals	0.02	0.1
W5	Metals	N/A	N/A
W6 (CH-TRU Waste)	Metals	810	See Text
W6 (RH-TRU Waste)	Metals	61	See Text
W7	Metals	0.4	See Text
W8	Metals	0.01	0

[a] These exposure air concentration to ERPG-2 ratios are the highest of any individual hazardous chemical.

N/A = Not Applicable

The highest air concentrations to which the MEI, the maximally exposed noninvolved worker, and the maximally exposed involved worker could be exposed were estimated and compared to ERPG values (see Table G-3). Ratios of exposure air concentrations to ERPG-2 values are presented in Table G-51. Heavy metal releases from a hoist failure (Accident Scenario W6) could result in severe consequences to the MEI and the noninvolved worker.

Bounding releases of beryllium, lead, and mercury from the hoist failure scenario involving CH-TRU waste (W6-CH) could seriously threaten the MEI and the maximally exposed noninvolved worker (ERPG-3 ratios of 48, 2, and 3, respectively). Bounding releases of beryllium from the hoist failure scenario involving RH-TRU waste (W6-RH) could seriously threaten these individuals (ERPG-3 ratio of 4). Irreversible, non-life-threatening consequences from bounding lead and mercury releases could result from W6-RH releases (ERPG-2 ratios of 58 and 61, respectively).

The hoist failure and roof fall scenarios could have consequences ranging from negligible to lethal effects for any involved worker in the underground disposal area. Typically, four underground workers are involved in emplacement operations. Some or all of these workers could be killed if they were in the immediate area when the accidents occurred. ERPG ratios were estimated for involved workers for Accident Scenarios W1 through W4. The ERPG analyses do not indicate any consequence to the involved workers for these four accident scenarios.

G.4.3.3 Accident Consequences for Action Alternative 3

Compared to accidents involving waste packaged to meet the planning-basis WAC, the consolidated waste form in Action Alternative 3 would reduce the consequences of most WIPP disposal accident scenarios. There would be no expected cancer incidence in the exposed population and very low probabilities of cancer to exposed individuals from VOCs and heavy metals for all accidents. As with the Proposed Action, Action Alternative 1, and Action Alternative 2 only, the hoist failure (Accident Scenario W6) would result in radiation-related LCFs in the exposed population around

WIPP. The radiological consequence is identical to that of Action Alternative 2 and greater than for the Proposed Action and Action Alternative 1 because the grouted waste form also is assumed to brittle-fracture upon impact and create a larger quantity of respirable particles than is the waste treated to planning-basis WAC. Up to 29 LCFs could occur in an exposed sector-population from a hoist failure involving CH-TRU waste, and 9 LCFs could occur from a hoist failure involving RH-TRU waste. The probability of an LCF to the MEI would be 0.6 for the CH-TRU waste accident scenario and 0.1 for the RH-TRU waste accident scenario. Table G-52 presents the radiological consequences from WIPP disposal accidents under Action Alternative 3.

Table G-52
Radiological Consequences of WIPP Disposal Accidents for Action Alternative 3

SEIS-II Accident Scenario	Accident Description	Population (person-rem, LCFs)		MEI (rem, probability of LCF)		Maximally Exposed Noninvolved Worker (rem, probability of LCF)		Maximally Exposed Involved Worker (rem, probability of LCF)
W1	Drop, Lid Failure in WHB	1.7 9E-4		0.026 1E-5		0.026 1E-5		7.8 3E-3
W2	Drop, Puncture Lid Failure in WHB	1.7 9E-4		0.026 1E-5		0.026 1E-5		7.8 3E-3
W3	Drop, Lid Failure in Underground	1.7 9E-4		0.026 1E-5		0.026 1E-5		7.8 3E-3
W4	Drop, Puncture, Lid Failure in Underground	1.7 9E-4		0.026 1E-5		0.026 1E-5		7.8 3E-3
W5	Container Fire	9.2 5E-3		0.14 7E-5		0.14 6E-5		N/A
W6	Hoist Failure	CH-TRU 58,000 29	RH-TRU 18,000 9	CH-TRU 870 0.6 [a]	RH-TRU 270 0.1	CH-TRU 870 0.5 [a]	RH-TRU 270 0.1	See Text
W7	Roof Fall	31 0.02		0.47 2E-4		0.47 2E-4		See Text
W8	RH-TRU Waste Canister Breach	0		0		0		0

[a] Maximum annual dose is 37 rem and remains above 20 rem per year for the first 11 years. Therefore, annual carcinogenic risk for the first 11 years was calculated without using the low dose rate dose reduction factor of 2.

N/A = Not Applicable

Potential carcinogenic consequences from exposure to VOCs and heavy metals released during WIPP disposal accidents are presented in Table G-53. As noted above, no cancer incidence would be expected in the exposed population, and consequences to individuals would be very low.

Potential noncarcinogenic consequences from exposure to VOCs and heavy metals released during an accident are presented in Table G-54. No life-threatening consequences are expected based on the IDLH-equivalent intake analyses; however, some consequences are expected based on the ERPG analyses.

The highest air concentration to which the MEI, maximally exposed noninvolved worker, and maximally exposed involved worker could be exposed were estimated and compared to ERPG values (see Table G-3). Ratios of exposure air concentrations to ERPG-2 values are presented in Table G-55. Heavy metal and VOC releases from a hoist failure and a roof fall could adversely affect the MEI and noninvolved workers based on the ERPG analyses. Some VOC releases from scenarios W2 and W4 could adversely affect the involved worker.

Table G-53
Hazardous Chemical Carcinogenic Consequences of WIPP Disposal Accidents for Action Alternative 3

SEIS-II Accident Scenario	Accident Description	Hazardous Constituents	Population (cancers)	MEI (probability of cancer)	Maximally Exposed Noninvolved Worker (probability of cancer)	Maximally Exposed Involved Worker (probability of cancer)
W1	Drop, Lid Failure in WHB	VOCs Metals	9E-9 4E-12	1E-10 5E-14	1E-10 5E-14	5E-8 2E-11
W2	Drop, Puncture, Lid Failure in WHB	VOCs Metals	3E-8 4E-12	4E-10 5E-14	4E-10 5E-14	2E-7 2E-11
W3	Drop, Lid Failure in Underground	VOCs Metals	9E-9 4E-12	1E-10 5E-14	1E-10 5E-14	5E-8 2E-11
W4	Drop, Puncture, Lid Failure in Underground	VOCs Metals	3E-8 4E-12	4E-10 5E-14	4E-10 5E-14	2E-7 2E-11
W5	Container Fire	VOCs Metals	0.0 4E-12	0.0 5E-14	0.0 5E-14	N/A
W6	Hoist Failure (CH-TRU Waste)	VOCs Metals	3E-7 1E-7	4E-9 2E-9	4E-9 2E-9	See Text
W6	Hoist Failure (RH-TRU Waste)	VOCs Metals	3E-8 9E-9	4E-10 1E-10	4E-10 1E-10	See Text
W7	Roof Fall	VOCs Metals	7E-7 6E-11	1E-8 9E-13	1E-8 9E-13	See Text
W8	RH-TRU Waste Canister Breach	VOCs Metals	3E-8 0	4E-10 0	4E-10 0	0 0

N/A = Not Applicable

Table G-54
Hazardous Chemical Noncarcinogenic Consequences of WIPP Disposal Accidents for Action Alternative 3

Accident Scenarios	Accident Description	Hazardous Constituents	MEI (maximum IDLH-equivalent ratio)	Maximally Exposed Noninvolved Worker (maximum IDLH-equivalent ratio)	Maximally Exposed Involved Worker (maximum IDLH-equivalent ratio)
W1	Drop, Lid Failure in WHB	VOCs Metals	5E-7 2E-9	5E-7 2E-9	2E-4 7E-7
W2	Drop, Puncture, Lid Failure in WHB	VOCs Metals	1E-6 2E-9	1E-6 2E-9	5E-4 7E-7
W3	Drop, Lid Failure in Underground	VOCs Metals	5E-7 2E-9	5E-7 2E-9	2E-4 7E-7
W4	Drop, Puncture, Lid Failure in Underground	VOCs Metals	1E-6 2E-9	1E-6 2E-9	2E-4 7E-7
W5	Container Fire	VOCs Metals	0.0 2E-9	0.0 2E-9	N/A
W6	Hoist Failure (CH-TRU Waste)	VOCs Metals	1E-5 6E-5	1E-5 6E-5	See Text
W6	Hoist Failure (RH-TRU Waste)	VOCs Metals	1E-6 4E-6	1E-6 4E-6	See Text
W7	Roof Fall	VOCs Metals	1E-5 3E-8	1E-5 3E-8	See Text
W8	RH-TRU Waste Canister Breach	VOCs Metals	1E-6 0	1E-6 0	0 0

N/A = Not Applicable

Table G-55
Ratios of Exposure Air Concentrations
to ERPG-2 Values for WIPP Disposal Accidents for Action Alternative 3 [a]

SEIS-II Accident Scenario	Hazardous Constituents	MEI and Maximally Exposed Noninvolved Worker	Maximally Exposed Involved Worker
		Air Concentration to ERPG-2 Ratio	Air Concentration to ERPG-2 Ratio
W1	VOC Metals	0.1 0.01	0.5 0.03
W2	VOC Metals	0.4 0.01	1.6 0.03
W3	VOC Metals	0.1 0.01	0.5 0.03
W4	VOC Metals	0.4 0.01	1.6 0.03
W5	Metals	0.04	N/A
W6 (CH-TRU Waste)	VOC Metals	3 260	See Text
W6 (RH-TRU Waste)	VOC Metals	0.4 19	See Text
W7	VOC Metals	7 0.1	See Text
W8	VOC Metals	0.4 0.01	0 0

[a] These exposure air concentration to ERPG-2 ratios are the highest of any individual hazardous chemical.

N/A = Not Applicable

Bounding consequence estimates for the MEI and noninvolved worker for the hoist failure scenario involving CH-TRU waste (W6-CH) could be severe. Releases of bounding methylene chloride concentrations and beryllium could seriously threaten these individuals (ERPG-3 ratios of 1 and 16, respectively). Irreversible, non-life-threatening consequences from releases of lead and mercury (ERPG-2 ratios of 240 and 260) could also result from this accident scenario. Transient consequences could result from carbon tetrachloride releases (ERPG-1 ratio of 1).

The hoist failure scenario involving RH-TRU waste (W6-RH) could also result in consequences to the MEI and maximally-exposed noninvolved worker under bounding release and exposure conditions, including life-threatening consequences from beryllium (ERPG-ratio of 3). Releases of lead and mercury could also result in irreversible, non-life-threatening consequences to these individuals (ERPG-2 ratios of 58 and 61, respectively).

The roof fall scenario (W7) could result in irreversible, non-life-threatening consequences to the MEI and noninvolved worker from 1,1,2,2-tetrachloroethane releases (ERPG-2 ratio of 7). Transient consequences to these individuals could result from carbon tetrachloride releases (ERPG-1 ratio of 4).

The hoist failure and roof fall scenarios could have consequences ranging from negligible to lethal effects for any involved worker in the underground disposal area. Typically, four underground workers are involved in emplacement operations. Some or all of these workers could be killed if in the immediate area when the accidents occurred. ERPG ratios were estimated for involved workers for Accident Scenarios W1 through W4. Irreversible, non-life-threatening consequences to the maximally-exposed involved worker could result from the Accident Scenarios W2 and W4, with releases of methylene chloride and 1,1,2,2-tetrachloroethane (ERPG-2 ratios of 2 for both).

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APPENDIX H
LONG-TERM CONSEQUENCE ANALYSIS FOR PROPOSED ACTION AND ACTION ALTERNATIVES

H.1 INTRODUCTION

This appendix describes the methods used in the performance assessment to estimate the potential for long-term migration of radionuclides, heavy metals, and volatile organic compounds (VOCs) from the Waste Isolation Pilot Plant (WIPP) repository disposal system (including natural and engineered barriers), transport to the accessible environment, and human health impacts from exposure to these materials. As described in Chapter 3, transuranic (TRU) waste would be disposed of at WIPP under the Proposed Action and Action Alternatives 1, 2, and 3. The potential for contaminant migration and human health impacts was evaluated for 10,000 years after closure for each of these alternatives.

This appendix presents an overview of the changes from earlier performance assessment analyses, summary descriptions of conceptual models used for the disposal system and waste source-term release, the computer codes used to estimate the extent and rate of contaminant transport, descriptions of the exposure scenarios and methods used to estimate the exposures, intakes, and potential impacts to human health from radionuclides and heavy metals. Detailed background information on the WIPP repository disposal system, detailed descriptions of the conceptual release models, and selected data input parameters may be found in supporting regulatory compliance documents completed or under development by the U.S. Department of Energy (DOE or the Department).

H.2 CHANGES IN PERFORMANCE ASSESSMENT ANALYSIS SINCE THE 1990 *FINAL SUPPLEMENT ENVIRONMENTAL IMPACT STATEMENT FOR THE WASTE ISOLATION PILOT PLANT* (SEIS-I)

The potential long-term impacts from the release of contaminants from the WIPP repository disposal system were analyzed in SEIS-I. However, new analyses were deemed necessary for this *Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement* (SEIS-II) because of (1) different alternatives and their associated waste volumes and inventories, (2) new data gathered on the repository environment and disposal system performance, and (3) substantial changes in WIPP performance assessment approaches and computational tools. A discussion of these new developments is summarized below. The SEIS-II alternatives and associated waste volumes and inventories are discussed in detail in Chapter 3 and Appendix A.

H.2.1 Pre-Disposal and Disposal Phase Experimental Programs

Since the WIPP site was selected and developed in the late 1970s and early 1980s, DOE has been conducting experiments and investigations to provide technical justification for major components of the conceptual model at the WIPP repository disposal system. During the pre-disposal phase, which began in 1990, experiments and studies at the WIPP site have evolved into major programs in the areas of rock mechanics and seal system performance, disposal room interactions, fluid flow and transport, and system response to human-initiated activities. The following is a brief description of studies being conducted at the WIPP site in each of these areas.

Rock Mechanics and Seal System Performance

The WIPP program has conducted experimental measurements of creep and fracture properties to gain a better understanding of the fundamental processes of salt creep closure of underground openings, the development of a disturbed rock zone (DRZ) and underground openings, and the healing of the DRZ by creep closure around emplaced wastes and shaft seal systems.

Investigations have shown that creep closure begins immediately after excavation because of excavation-induced deviatoric stresses. The effect of the creep closure phenomena on the two-phase flow of brine and gases through waste is addressed in the *Title 40 Code of Federal Regulations* (CFR) *Part 191 Compliance Certification Application for the Waste Isolation Pilot Plant* (CCA) performance assessment, as described in Appendix PORSURF (DOE 1996f). This appendix explains the mathematical basis of the halite creep model, waste consolidation, and the constitutive model used to simulate the inelastic behavior of anhydrite marker beds (See Appendix PORSURF, Attachment 1).

Gas generation in the repository caused by decomposing wastes can result in expansion of the excavated region and possibly the fracturing of anhydrite beds. Performance assessment models incorporate the effect of gas-driven hydrofracture through changes in porosity and permeability in marker beds intersecting the repository DRZ at pressure levels consistent with in-situ experiments, laboratory experiments, and the Linear Elastic Fracture Model (DOE 1996f, Appendix MASS, Attachment 13.2).

The conceptual design of the shaft seals, which evolved during the 1980s, provides the long-term ability to isolate the repository from overlying units. The key design feature included a salt component that would be emplaced throughout the Salado, consolidate under the pressure of salt creep over a period of a hundred to several hundred years, and help the shaft seal system develop properties similar to intact salt. Concerns have been raised that brine flow to the salt seal from upper water-bearing units could delay and even prevent creep consolidation of the long-term seal components of the crushed salt. Early concepts of the shaft seal system, which used concrete and concrete-grout plugs to protect the salt component, were not thought to be robust enough to control brine flow. These concerns led to the development of the current shaft seal system concept, which is based on the principle that multiple components and materials can provide a demonstrable level of protection of key components from downward brine flow.

The seal system program has included measurements of shaft seal component properties and performance, studies of shaft seal designs, and characterization of seal behavior, including measurements of creep closure and DRZ development around the shaft. The results of these studies and the current understanding of the shaft seal system have been incorporated into current performance assessment models and calculations. Details of the shaft seal performance studies are described in Appendix SEAL of the CCA (DOE 1996f).

During the disposal phase of the WIPP site, which would begin when the WIPP site opens, rock mechanics and seal system performance studies will continue to be used to enhance the disposal system operations and maintain certification compliance. According to the *Disposal Phase Experimental Program Plan* (DOE 1997), planned activities include:

- Geomechanical and subsidence monitoring of the development and healing of the DRZ surrounding the air intake shaft and disposal horizon

- Optimizing the component specification of the shaft seal system to enhance compaction of crushed salt, evaluate construction of clay components, tailor specifications of selected clay mixtures to field application, evaluate the longevity of concrete under WIPP conditions, and examine emplacement methodologies for asphaltic seal components

Disposal Room Interactions

To support its understanding of disposal room interactions, the Department has been investigating a variety of processes including creep closure in the disposal rooms, waste consolidation, and chemical interactions of the wastes with the host rock and brine that may flow into the disposal facilities from the surrounding Salado or Castile Formations. These units are a potential source of brine to the repository horizon. Actinide solubilities, which were not considered important in the early 1980s with the initial conceptual model of dry salt beds, became increasingly important in release pathways involving brine flow. Transport of actinides in colloidal form was also considered important. Chemical interactions under investigation include studies of actinide mobility such as the determination of actinide solubilities and oxidation states in brine at the WIPP horizon, and the quantities of actinide that could be mobilized in stable colloid forms found at WIPP. The results of these investigations provide the basis of the transport models and parameters being used in the current performance assessment calculations. Details of how actinide studies are used in the CCA are described in Appendix SOTERM of the CCA (DOE 1996f).

The Department has also conducted numerous experiments and investigations aimed at assessing the gas generation rates from anoxic corrosion of metals, anaerobic microbial decomposition of cellulosic materials found in disposed wastes, and radiolysis of brine coming into contact with wastes. These studies led to the development of gas generation models and relevant parameters currently used in performance assessment models and calculations to evaluate the effects of gas generation. Details of the gas generation studies used in the CCA are described in Appendices SOTERM and MASS of the CCA (DOE 1996f).

Studies of chemical interactions also led to the concept of using a magnesium oxide backfill in the disposal rooms to react with carbon dioxide created by microbial degradation, buffer the acidity of brine present in the repository, and control the solubility of actinides. Details of the basis for the selection of the magnesium oxide backfill as a chemical control component of the disposal system are described in Chapter 3 and Appendix BACK of the CCA (DOE 1996f).

During the disposal phase period, studies on disposal room interactions will be used to support certification compliance and system operations. Planned activities, as outlined in the *Disposal Phase Experimental Program Plan* (DOE 1997), include:

- Studies of actinide solubilities and geochemistry using experiments and tests with actual waste

- Continued gas generation experiments to evaluate the impact of major waste constituents that contribute to gas generation

- Additional laboratory tests to enhance the effectiveness of a magnesium oxide backfill, to examine the optimal amount of magnesium oxide required for the backfill, and to evaluate the cost-effectiveness and performance of backfill packaging
- Continued waste characterization studies to reduce the uncertainty in waste components and characteristics significant to long-term performance

Fluid Flow and Transport

Studies of fluid flow and transport at the WIPP site have focused on the presence and flow of brines in the Salado Formation, the overlying Rustler Formation, and the underlying Castile Formation.

During the early excavations of the WIPP site, observations of brine seeping into the repository from boreholes and freshly excavated surfaces led to significant changes in the conceptual models of WIPP performance. However, information about the rate of brine inflow showed that concerns about the waste becoming fluidized were unrealistic because consolidation to a sufficiently low porosity would occur before significant amounts of brine could accumulate. A series of tests, including the small-brine inflow, Brine Sampling and Evaluation Program (BSEP), Room Q, in-situ permeability tests, and laboratory flow tests were instituted to help validate the understanding of the interaction of brine inflow and the hydraulic properties of the Salado Formation.

Investigations of the Salado Formation have also provided an extensive set of measurements of the Salado's hydraulic properties within and adjacent to the WIPP site. These measurements have been incorporated into current performance assessment models and calculations and have been used to estimate the amount of brine that would enter into the repository, contribute to gas-generating reactions, and provide a medium for transport actinides released from wastes. Details of the Salado Formation studies are described in Chapter 2 and Appendix MASS of the CCA (DOE 1996f).

Investigations in the Rustler Formation have focused on the flow and transport properties of the Culebra Dolomite. Field measurements (hydraulic and tracer tests) have been used to characterize diffusion and fracture flow properties of the Culebra at multiple locations near the repository, which provide the basis for flow and transport models and related parameters used in the current performance assessment calculations. In the 1980s and 1990s, the interest has focused on a variety of field and laboratory investigations designed to increase the Department's understanding of the flow and transport characteristics of the Culebra. These investigations have included multi-well tracer tests and regional pumping tests, resulting in the recent seven-well tracer tests conducted at H-19, multi-well retesting at H-11, and single-well injection and withdrawal tests at both H-19 and H-11. Details of the Culebra Dolomite studies are described in Chapter 2 and in Appendices MASS and HYDRO of the CCA (DOE 1996f).

Properties of the Castile Formation have been primarily derived from hydraulic measurements obtained at boreholes that have penetrated pressurized brine reservoirs found in the Castile. Geophysical studies have helped resolve whether brine reservoirs encountered in WIPP-12 might exist under waste panels. Investigations in the late 1980s indicated that a zone of lower resistivity, which can be interpreted as brine, exists under a portion of the panels. Details of the Castile Formation studies are described in Chapter 2 and in Appendices MASS and HYDRO in the CCA (DOE 1996f).

During the disposal phase period, monitoring of fluid flow in all major hydrogeologic units of concern in the vicinity of WIPP will continue to support certification compliance and system operations. Planned activities, as outlined in the *Disposal Phase Experimental Program Plan* (DOE 1997), include groundwater surveillance of the existing monitoring network for routine water-level and water quality monitoring, and maintenance of a database on the location and properties of brine reservoirs in the Castile Formation.

System Response to Human-Related Activities

As part of the studies to examine system response to human-related activities, the WIPP program has developed passive institutional controls designed to reduce the possibility of human intrusion into the disposal facility during the 10,000-year regulatory time frame. Components of this control system include: (1) physical markers that would warn of the presence of buried nuclear wastes and identify the boundary of the disposal area and the controlled area, (2) external records about the WIPP repository, and (3) continued federal control. Details of the passive institutional control system are described in Section 7.1.3.2 and Appendices EPIC and PIC of the CCA (DOE 1996f).

The Department has also conducted a number of studies to evaluate the critical processes associated with the direct release of waste materials and brine as the result of human intrusion by exploratory boreholes into the repository. Early analyses of inadvertent penetration of the repository by deep drilling were initially treated deterministically until 40 CFR Part 194 required the analysis of multiple borehole intrusions and their interactions to be considered. The combination of postulated brine saturation and high pressures in the repository led to a need for the development of gas spallings and for direct brine release models to be considered with the cuttings and caving releases that were previously modeled.

Additional studies examined the potential for releases to occur from drill cuttings, material eroded from a borehole wall from circulation of drilling fluid, material spalled or forced into the borehole by pressurized fluid, and release of radionuclides dissolved in pressurized brine. Calculations and measurements have also examined the heterogeneity and uncertainty in the properties of consolidated wastes and how they could contribute to direct releases from intrusion. Modeling of cuttings, cavings, and direct brine releases is described in Appendices MASS and CUTTINGS_S of the CCA (DOE 1996f).

The Department has also conducted studies to evaluate the hydraulic characteristics of plugs used in standard borehole plugging practices and their potential for degradation after abandonment. The data collected during these studies, combined with studies of pressurized brine reservoirs in the Castile Formation and flow and transport properties of the Culebra Dolomite, provided the technical basis for the development of performance assessment codes and conceptual models of disturbed conditions described in Appendix DEL of the CCA (DOE 1996f).

During the disposal phase period, studies of system response to human-related activities will continue to support certification compliance and system operations. Planned activities, as outlined in the *Disposal Phase Experimental Program Plan* (DOE 1997), include:

- Continued development and enhancement of the passive institutional control system

- Maintenance of a database on oil and gas drilling and potash mining activities in the Delaware Basin, and the location of pressurized brine reservoirs in the Castile Formation

- Additional laboratory and modeling studies to investigate and reduce the uncertainty of sensitive parameters used to estimate direct releases from the repository as the result of human intrusion. Gas generation, backfill, and waste characterization studies will provide critical information for improving estimates of releases from borehole intrusions

H.2.2 Computational Tools and Codes

The computer codes and databases used to assess the long-term performance of the WIPP repository disposal system have evolved substantially since the SEIS-I analysis was completed. A major development effort accompanied the preliminary performance assessment of WIPP in 1992. The primary areas of enhancement include the development of models to simulate coupled gas generation, brine migration, and salt creep (SNL 1992). Conceptual models and databases supporting performance assessment calculations were modified and enhanced for the Systems Prioritization Methodology (SPM) process conducted in 1994 and 1995. Further code development and enhancement have continued to support more recent regulatory compliance efforts associated with the development of the CCA (DOE 1996f). The following is a description of key developments since SEIS-I.

In the SEIS-I analysis, long-term radiological and hazardous chemical impacts of the WIPP repository disposal system were determined by accepted conceptual models and computer codes implemented by DOE in performance assessment programs. The two principal codes in use at that time were the NEFTRAN and SWIFT II codes.

NEFTRAN, a groundwater flow and radionuclide transport code developed by Sandia National Laboratories (SNL) for the U.S. Nuclear Regulatory Commission, was used to calculate radionuclide releases from an undisturbed repository (Longsine et al. 1987). NEFTRAN was designed with the assumption that all substantial groundwater flow and radionuclide transport would progress along discrete one-dimensional legs or paths. A flow field is represented by a network of these legs. The solution of the flow equations in NEFTRAN requires pressure boundary conditions specified in the input data.

SWIFT II is a fully transient, three-dimensional code that solves equations for groundwater flow and radionuclide transport in both porous and fractured media. In SEIS-I (DOE 1990), SWIFT II was used to calculate releases from a disturbed repository. Also, SWIFT II was used to establish the groundwater flow field in the Culebra Dolomite and to simulate the injection of pressurized brine from the Castile Formation into the Culebra Dolomite.

The *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992) used a suite of new codes and modules that substantially improved simulation capabilities from those in SEIS-I. Improvements were made in the areas of simulating two-phase flow and salt creep in the vicinity of the repository within the Salado Formation and in the representation of transmissivity fields, flow fields, and transport calculations within the Culebra Dolomite of the Rustler Formation. Table H-1 briefly describes the key modules that were used for the first time in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* calculations and are relevant to this analysis.

Table H-1
Summary of Computer Codes Used in the 1992 WIPP Performance Assessment

Computer Code	Description
BRAGFLO	Module used to simulate two-phase flow of gas and brine through a porous, heterogeneous reservoir. This module is used to simulate two-phase (gas and brine) flow through the repository, shaft seals, and surrounding environment.
PANEL	Module used to calculate radionuclide concentrations in the brine phase with an equilibrium-mixing cell approach. This module calculates the rate of discharge and cumulative discharge of radionuclides from a repository panel through an intrusion borehole. Discharge is a function of the fluid flow rate, nuclide solubility, and remaining inventory.
SANTOS	Module used to simulate quasi-static, large-deformation, inelastic response of halite. These simulations are used in the calculation of waste porosity as a function of time and moles of gas generated.
CUTTINGS_S	Module used to calculate the quantity of radioactive material brought to the surface as cuttings and cavings as a result of an exploratory drilling operation that penetrates a waste panel.
SECOFL2D **SECOTP2D**	Modules are groundwater flow and transport models used to calculate subsurface transport through the Culebra Dolomite of the Rustler Formation to the Land Withdrawal Boundary. Flow calculations assumed a single matrix, porous medium (dolomite). Transport calculations modeled single- or dual-porosity transport through an idealized fractured medium. Retardation in the dolomite matrix and in the fracture-lining clay could be included simultaneously or separately.
GRASP-INV	Module used to generate multiple, plausible transmissivity fields for use by SECOFL2D. This module was an improvement, in that it produced calibrated transmissivity fields that reproduced measured values at well locations.
GENII-S	Module used to estimate potential radiation doses to humans from radionuclides in the environment.

The SPM, initiated by DOE in March 1994, was the result of an effort to define the most viable combinations of scientific investigation, engineering alternatives, and the planning-basis Waste Acceptance Criteria (WAC) (DOE 1996c) for supporting WIPP compliance applications. As a part of this SPM development process, DOE established a technical baseline by which to summarize the conceptual models of disposal system performance and to assemble new information from technical position papers for use in the SPM process.

To the extent possible, the SPM process implemented the computer codes used in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992) with modifications for conceptual model enhancements and new information identified in the development of the technical baseline. Details of the modifications and new information are described in *The Second Iteration of the Systems Prioritization Method: A Systems Prioritization and Decision-Aiding Tool for the Waste Isolation Pilot Plant: Final Report Revision 1* (SPM-2) (SNL 1995).

With a few exceptions, the same codes used in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992) are being used in the current regulatory performance assessment application and related documentation. Key new developments included the following: (1) the SANTOS (formerly the SANCHO) module that was used to simulate quasi-static, large-deformation, inelastic response of halite has now been incorporated and integrated in the computational framework of the BRAGFLO module (described in Section H.3.3.1), and (2) the NUTS module is currently being used to simulate the transport and decay of multiple radioactive components in three dimensions through the fracture and matrix continuum. The NUTS module (described in Section H.3.3.2) is now used to simulate long-term transport of contaminants in brine within the repository to the surrounding rock and up intrusion boreholes, providing a source-term for the SECOTP2D code. The PANEL code is now used only to simulate scenarios involving two intrusion boreholes. These scenarios were not analyzed in SEIS-II.

H.3 DESCRIPTION OF APPROACH

The purpose of the SEIS-II performance assessment was to estimate the potential for long-term migration of radionuclides, heavy metals, and VOCs from the WIPP repository disposal system, evaluate transport to the accessible environment, and assess potential human health impacts from exposure to those contaminants released. SEIS-II analyses of the Proposed Action and Action Alternatives 1, 2, and 3 evaluated the potential impacts for 10,000 years beginning with the end of active institutional control, which is assumed to occur 100 years after the closure of WIPP.

SEIS-I analyses (DOE 1990) evaluated the performance of the WIPP repository for both undisturbed and disturbed (human intrusion by exploratory drilling) scenarios. For undisturbed and disturbed performance cases, SEIS-I evaluated the impacts of variations in selected parameters that would lead to expected (realistic) and less than expected (degraded) repository performance. In an undisturbed, degraded performance scenario (Case IB), SEIS-I analyzed the impact of changing some parameters, including the solubilities of key radionuclides, by factors of 100 and reducing the resistance to flow in the shaft and panel seals by a factor of 100. In a second undisturbed, degraded performance scenario (Case IC), leakage through marker bed 139 (MB 139) via accessway seals was assumed to increase marker bed permeability by a factor of 10 beyond those used in Case IB. The permeability of the lower shaft seal was also increased by a factor of 100 above what was assumed in Case IB. Disturbed scenarios examined the impacts of a hypothetical intrusion into the repository by a borehole drilled through the repository into a pressurized brine reservoir below. Disturbed, degraded repository performance scenarios (IIB, IIC, and IID) included parameter adjustments similar to those for undisturbed cases but also examined other parameter changes such as the amount of waste compaction. Descriptions of specific parameter changes considered in SEIS-I are provided in Section 5.4.2.2 and Appendix I of SEIS-I.

The SEIS-II analyses of long-term performance of the WIPP repository used an approach similar to SEIS-I but incorporated current computer codes and data developed since SEIS-I (DOE 1996f). Key features of the SEIS-II performance assessment approach were as follows:

- Detailed mathematical models to simulate key physical and chemical processes beginning with repository closure

- Deterministic analyses using current computational codes

- Updated parameter databases

- Performance evaluated for undisturbed and disturbed (human intrusion by exploratory drilling) scenarios over the 10,000-year period following site closure. Disturbed scenarios were assumed not to take place until after the end of the active institutional control period 100 years after repository closure.

- Analyses of repository releases and transport performed using cases of median and 75th percentile values for those parameters where statistical distributions were available. Median values were considered to be realistic estimates of performance and 75th percentile values were considered indicative of degraded or pessimistic repository performance that would result in greater contaminant release and transport. Values for some parameters, such as Kd's in the Culebra, were chosen at the 25th rather than 75th percentile in order to simulate faster transit times. However, the general terminology "75th percentile" is used to describe this parameter selection process.

H.3.1 Data Sources and Parameter Selection

Data used in the long-term performance analysis were derived primarily from the CCA (DOE 1996f). The electronic database used by the CCA and documented in Appendix PAR of the CCA was also used to derive the parameters for this analysis.

Values for individual parameters in the computer codes were selected using the following approach:

- Median values chosen from the statistical distribution defining the parameter. These values were selected to represent expected repository performance.

- Seventy-fifth percentile values chosen from the statistical distribution defining the parameter. The value of the parameter was chosen to represent degraded performance, such that it would lead to higher releases than when the median value of the parameter was used.

Although many parameters in the models were employed for the long-term performance assessment, few have a substantial impact on the amount of material released. Some of the important parameters include the solubility of contaminants in the waste form, sorption of contaminants on the host salt, and hydraulic conductivity of the salt units near the repository. Relative to a probabilistic analysis, if (1) these parameters account for most of the variability in the computed release (or risk) values, and (2) values for these sensitive parameters were chosen at the 75th percentile of their respective distributions, the resulting single output realization would be expected to approach the maximum release expected if 100 realizations were run.

H.3.2 Release Scenarios Analyzed

Two conditions were considered in this study: the undisturbed repository performance and disturbed performance as affected by human intrusion. The undisturbed condition considers the performance of WIPP after closure for a period of 10,000 years without human contact. Two scenarios were considered for the disturbed condition.

- A drilling event that breaches the repository

- A drilling event that breaches the repository and penetrates a hypothetical pressurized brine reservoir in the Castile Formation below the repository horizon

Analyses of the impacts of each intrusion event are reported as the consequence of a single event and not as the combined impacts from a probabilistic set of drilling events over 10,000 years. The uncertainty inherent in some of the physical parameters that control flow and transport calculations was treated by considering the median and 75th percentile values of the parameters for both undisturbed cases and human-intrusion scenarios.

A total of sixteen cases, outlined in Table H-2, were used to estimate the impacts of undisturbed and disturbed performance for the Proposed Action and Action Alternatives 1, 2, and 3. The notable differences among the Proposed Action and Action Alternatives 1, 2, and 3 include their differing inventory loadings, changes in disposal room concentrations of radionuclides and heavy metals given thermal treatment of the waste, and inventory porosity and permeability changes. The main difference between alternatives is related to the total radionuclide and heavy metal inventory and the repository area (number of panels). The pathways for release and associated models used to quantify flow and transport in the subsurface environment for the Proposed Action and all action alternatives are illustrated in Figure H-1. Brine and gas flow depicted in the repository and disposal system was modeled using the BRAGFLO computer code. The distribution of fluid flow velocities and other quantities obtained from BRAGFLO simulations were then used to develop the necessary input parameters for the NUTS computer code, a subsurface contaminant transport simulator code.

Table H-2
Cases Considered in Long-term Performance Analysis of the Proposed Action and Action Alternatives

Alternative	Case Number	Case/Scenario	Data Selection
Proposed Action	1	Undisturbed	Median values
	2	Disturbed (Borehole Intrusion)	Median values
	3	Undisturbed	75th percentile values
	4	Disturbed (Borehole Intrusion)	75th percentile values
Action Alternative 1	6	Undisturbed	Median values
	7	Disturbed (Borehole Intrusion)	Median values
	8	Undisturbed	75th percentile values
	9	Disturbed (Borehole Intrusion)	75th percentile values
Action Alternative 2	11	Undisturbed	Median values
	12	Disturbed (Borehole Intrusion)	Median values
	13	Undisturbed	75th percentile values
	14	Disturbed (Borehole Intrusion)	75th percentile values
Action Alternative 3	16	Undisturbed	Median values
	17	Disturbed (Borehole Intrusion)	Median values
	18	Undisturbed	75th percentile values
	19	Disturbed (Borehole Intrusion)	75th percentile values

Note: Cases 5, 10, 15, and 20 were dropped because backfill was incorporated into the repository design rather than modeled as a mitigation measure.

The SECOFL2D and SECOTP2D computer codes were used in SEIS-II to simulate the migration of radionuclides and heavy metals for disturbed conditions where releases to the Culebra Dolomite were predicted. The SECOTP2D computer code was used to model contaminant transport in the Culebra. This model of the Culebra Dolomite is a two-dimensional system that relies on flow fields calculated by the SECOFL2D code. Additional information is provided in Section H.8.

H.3.2.1 Undisturbed Conditions

For undisturbed conditions (i.e., no human intrusion), the release of radionuclides, heavy metals, and VOCs would occur only through dilute aqueous-phase and gas-phase transport from the WIPP repository into the Salado Formation and up the shaft seal system. The probabilistic analysis

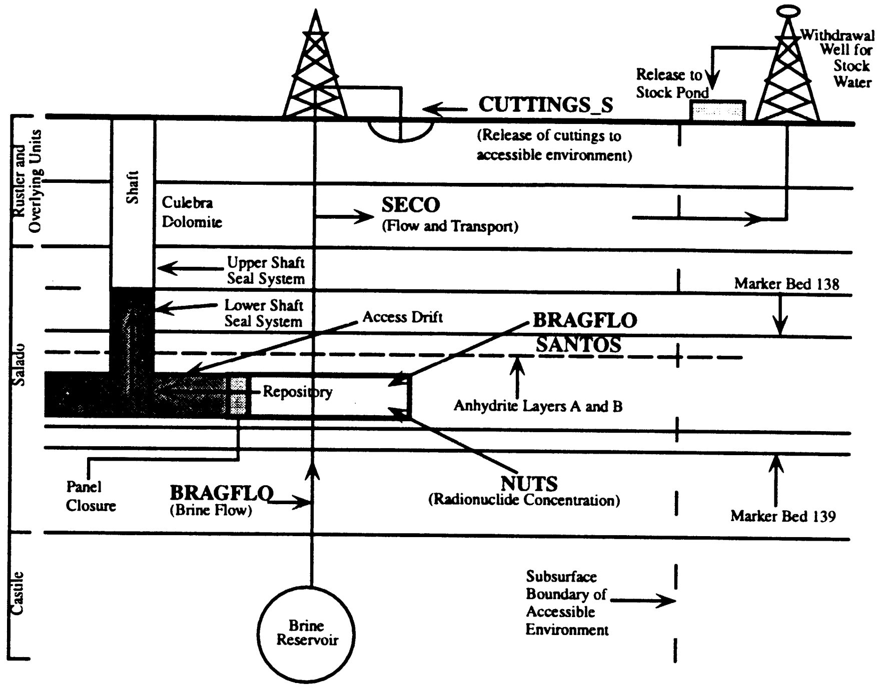


Figure H-1
Schematic Side View of the Repository and Disposal System
Associating Major Performance Assessment Codes with Principal Components

conducted in the CCA (DOE 1996f) shows rapid reduction of waste panel porosity during the first 300 to 500 years, at which point the waste panels are near their final state (SNL 1996). The waste is expected to compact to an estimated final porosity of 1 to 5 percent, depending on the amount of gas generation (SNL 1996).

H.3.2.2 Disturbed Conditions

This section describes two release scenarios for disturbed conditions.

Surface Release Caused by Drilling into the Repository

In this disturbed condition, a hypothetical exploratory drilling operation inadvertently penetrates a waste panel in the repository. As a result, the drilling brings waste originating in the repository to the land surface and exposes individuals involved in the drilling operation to radionuclides and heavy metals. Three separate physical processes (Cole and Simmons 1995) are assumed to influence the quantity of repository waste brought to the ground surface as a result of a drilling intrusion (Figure H-2):

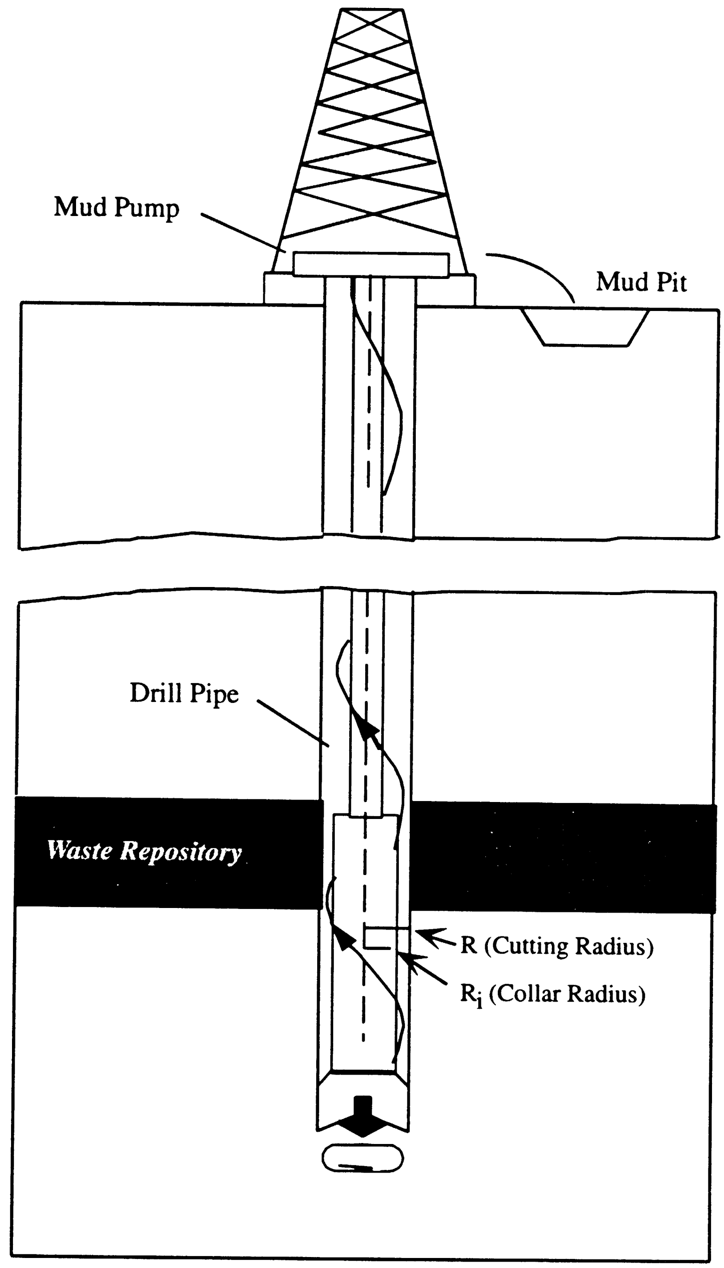


Figure H-2
Illustration of Exploratory Drilling Equipment for Human-Intrusion Scenario

- Cuttings - waste in the cylindrical volume created by the cutting action of the drill bit passing through the emplaced waste

- Cavings - waste that erodes from the borehole in the response to the upward-flowing drilling fluid within the annulus

- Spallings - release of solid materials into the drilling fluid as a result of the release of waste-generated gas escaping to the lower-pressure borehole. This process requires a repository gas pressure in excess of the hydrostatic pressure of the drilling mud to contribute to the releases.

A fourth release model, termed “direct brine release,” accounts for the release of brine fractions of material introduced in a spallings event. Radionuclides and heavy metals released at the surface by cuttings, cavings, and spallings are modeled in this document. Direct brine releases are not considered in the SEIS-II analyses due to their relatively minor contribution to surface releases in the CCA (DOE 1996f).

The relationship of repository pressure to the release processes has been quantified and provides the basis for calculations of direct releases of wastes for the CUTTINGS_S computer code, as depicted in Figure H-3. The spallings release mode includes a blowout, gas erosion, and stuck pipe release mode. Because of the higher waste permeabilities considered in the SEIS-II analyses, only the blowout mode was relevant. More detailed information about these release modes are provided in Appendix CUTTINGS_S of the CCA (DOE 1996f). The values of brine and gas pressure in the repository and the permeability of the waste used to determine the release process under this model were obtained from the fluid flow simulations performed with the BRAGFLO computer code. Simulated brine and gas pressures were derived for undisturbed conditions calculated for expected (median values) and degraded (75th percentile values) cases. Disturbed and undisturbed repository pressure histories are considered identical until the time of intrusion, with the undisturbed cases being the basis on which pressure is estimated at any given intrusion time. The calculations of direct releases for the simulated BRAGFLO conditions were carried out using the CUTTING_S code.

Drilling Through the Repository into a Pressurized Brine Reservoir

Under this scenario a borehole is drilled through the repository and penetrates a pressurized brine reservoir in the Castile Formation below the repository horizon. Brine in the reservoir is assumed to come into contact with wastes in the repository and move further up the borehole to more permeable units lying above the repository horizon, like the Culebra Dolomite in the Rustler Formation. Should it occur, a release to the Culebra Dolomite could then be transported downgradient and become available for withdrawal through a well.

Future conditions of groundwater flow used in the transport analysis were based on the assumption that mining of potash reserves near the WIPP site would occur. Regulatory guidance, provided in 40 CFR Part 194 on the assessment of potash mining, directs that the effect of mining can be considered in evaluating off-site impacts by increasing the hydraulic conductivities of key hydrogeologic units, e.g., the Culebra Dolomite. These increases in hydraulic conductivity (to approximate the hydraulic effect of potash mining) can affect the transport of contaminants released from the repository if and only if these contaminants reach the Culebra Dolomite.

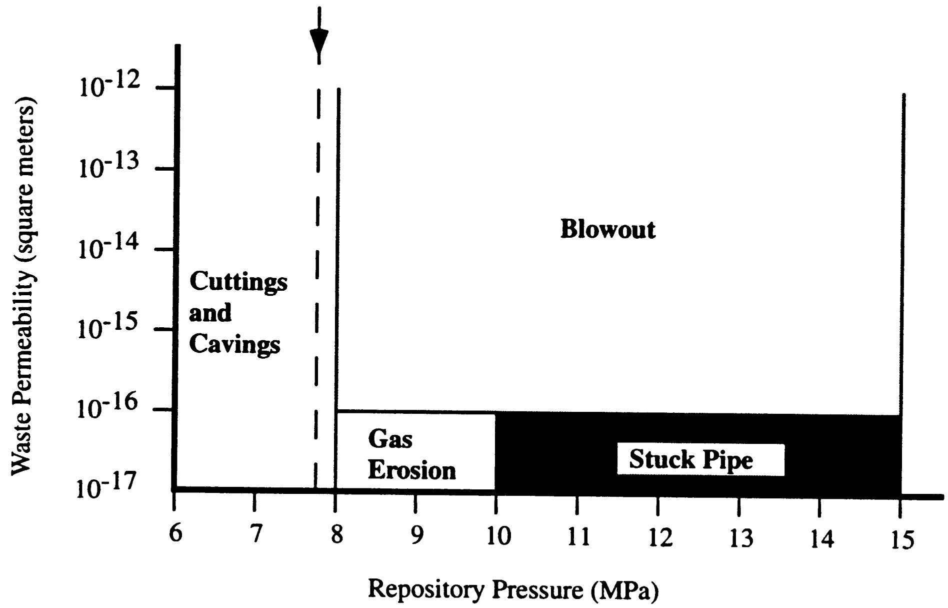


Figure H-3
Release Processes for Inadvertent Drilling Intrusion at Varying Repository Pressure and Waste Permeability

Following guidance provided in 40 CFR Part 194, hydraulic conductivities of the Culebra Dolomite in the areas impacted by mining were changed by a factor ranging from 1 to 1,000. SEIS-II analyses use the flow field results developed for the CCA (DOE 1996f): one with mining outside the Land Withdrawal Area boundary (partial mining) and one with all regions mined (full mining). Additional information on how these flow fields are used in modeling transport of contaminants in the Culebra Dolomite is provided in Section H.7.

H.3.3 Source-Term Release and Transport Codes

Computer codes used in this analysis include those used by DOE in the CCA (DOE 1996f). A schematic cross-section of the repository and the disposal system, associating major performance assessment codes with the principal components of the disposal system they simulate, is presented in Figure H-1. A brief description of each code used is provided below.

H.3.3.1 BRAGFLO

The BRAGFLO code in the CCA (Appendix BRAGFLO of DOE 1996f) was used to quantify the effects of gas and brine flow on radionuclide transport for undisturbed conditions. The code incorporates the effects of disposal room consolidation and closure, gas generation, and interbed fracturing in response to gas pressurization of the repository. BRAGFLO simultaneously solves partial differential equations that describe the mass conservation of mobile gas and brine components, using appropriate initial conditions, boundary conditions, and other constraints. The important features of BRAGFLO include the following:

- Uses a finite difference approach to simultaneously solve partial differential equations that describe mass and energy conservation of mobile components of gas and brine with appropriate constraint equations, initial conditions, and boundary conditions

- Simulates a porous medium that can be occupied by brine, gas, or both brine and gas where the brine and gas are assumed to be immiscible

- Considers formation permeability simulations to be anisotropic

- Uses relative permeability and capillary pressure equation models available, including van Genuchten-Parker, original Brooks-Corey, and modified Brooks-Corey (Appendix BRAGFLO in DOE 1996f)

- Calculates the overall movement of gas and brine in the disposal unit and surrounding formations and defines the flow fields for contaminant migration postprocessing codes

- Contains the submodels for estimating gas generation in the repository, disposal room closure and consolidation, and interbed fracturing

- Simulates gas generation by creating gas in the waste disposal panels from the corrosion of waste containers and by microbial degradation of cellulose materials in the disposed wastes

- Changes in permeability and gas-storage volume of the waste resulting from creep closure are coupled to BRAGFLO through SANTOS, a code that provides a “porosity surface” used as a reference to track changes in room volume. SANTOS results are included in BRAGFLO through a series of tables that provide data to BRAGFLO describing dynamic changes in porosity as a function of time and pressure

- Allows fracture treatment for pressure-induced alterations to material porosity by introducing a pressure-dependent compressibility using a piecewise linear rock compressibility function

- Includes boundary conditions such as (1) Dirichlet (constant pressure), (2) inhomogeneous Neuman (fixed-pressure gradient), and (3) mixed boundaries (mix of Dirichlet and Neuman). For this analysis, a no-flow boundary condition is used for all exterior grid boundaries except at the far-field boundaries of the Culebra and Magenta Members and the top of the grid (i.e., the surface ground). The boundaries of the Culebra and Magenta are assigned constant pressure conditions used in those Members. The ground surface elements are maintained at one atmosphere (Appendix BRAGFLO in DOE 1996f).

For detailed information of the governing equations, initial and boundary conditions, and submodels used by BRAGFLO, refer to the CCA (Appendix BRAGFLO in DOE 1996f).

H.3.3.2 NUTS

The NUTS code in the CCA (Appendix NUTS of DOE 1996f) was used to track brine that has been in contact with waste in the repository. NUTS uses the calculated gas and brine flow fields computed by BRAGFLO to transport the radionuclides in solution in the brine from the repository into the surrounding halite and anhydrite beds of the Salado Formation. These calculations also include the transport of radionuclides up a borehole or repository shaft to determine the quantity of radionuclides that could potentially reach the overlying Culebra Dolomite of the Rustler Formation. The important features of NUTS include the following:

- Uses finite difference techniques to simulate the decay and transport of multiple radionuclide components in three dimensions in fractured or unfractured media. Simulations can be performed using single-porosity, dual-porosity, and dual-permeability models.

- Simulates transport of both radioactive and nonradioactive contaminants

- Considers transport of radionuclides with chain decay

- Simulates sorption using three different sorption isotherms: linear, Freundlich, and Langmuir equilibrium isotherms

- Considers transport with solubility limits of individual contaminants and their precipitation. The precipitate can be decayed or redissolved in calculated concentrations that drop below solubility limits.

- Considers transport with multiradioactive site representations, a variety of source and sink terms, and the implementation of temperature dependency of certain parameters (temperature-dependent solubility, molecular diffusion, and sorption)

For more information regarding the underlying theory, governing equations, initial and boundary conditions, and submodels used by NUTS, refer to the CCA (Appendix NUTS of DOE 1996f).

H.3.3.3 CUTTINGS_S

The CUTTINGS_S code was used with the results calculated by BRAGFLO to determine radionuclide releases to the land surface by inadvertent repository intrusion by an exploratory borehole. CUTTINGS_S estimates the effect of separate physical processes that can influence the quantity of wastes brought to the ground surface by an exploratory borehole. These processes are:

- Generation of cuttings - wastes in the cylindrical volume created by the cutting action of the drill bit

- Cavings - wastes that erode from the borehole wall in response to the upward-flowing drilling fluid within the borehole annulus

- Spallings – solid particulate materials introduced into the drilling fluid by the release of waste-generated gas escaping into the lower-pressure borehole

For more information regarding the underlying theory, governing equations, and utilities used by CUTTINGS_S, refer to the CCA (Appendix CUTTINGS_S of DOE 1996f).

H.3.3.4 SECOFL2D and SECOTP2D

The SECOFL2D and SECOTP2D computer codes were used to simulate the migration of radionuclides and metals released into the Culebra Dolomite for selected cases of disturbed conditions. These codes are part of the suite of codes used to simulate groundwater flow, particle tracking, and solute transport in the Culebra Dolomite in support of the CCA (DOE 1996a). The SECOFL2D code is a two-dimensional, finite difference groundwater flow model capable of simulating transient or steady-state flow in saturated or unsaturated porous media. The SECOTP2D code is a two-dimensional, dual porosity transport model developed to simulate radionuclide transport in fractured porous media. The SECOTP2D code is also capable of simulating non-radioactive solutes such as the suite of heavy metals being considered in this analysis. The code assumes parallel plate type fracturing where fluid flow is restricted to the advective component of the flow system (i.e. in the fractures), and mass is transferred between the advective and diffusive parts of the flow system by molecular diffusion. SECOTP2D assumes linear equilibrium sorption isotherms in modeling retardation between the advective and diffusive components of the flow system. Radioactive decay is accounted for in the model through the use of multiple straight decay chains. Additional information on both codes and their governing equations is provided in SNL (1996) and Appendices SECOFL2D and SECOTP2D of the CCA (DOE 1996a).

H.4 CONCEPTUAL MODEL OF THE REPOSITORY DISPOSAL SYSTEM

Existing conceptual models of the WIPP repository disposal system developed for regulatory compliance analysis in the CCA (DOE 1996f) provided the basis for the SEIS-II analysis of long-term performance. The following is a synopsis of key conceptual models of the repository disposal system drawn from the CCA.

The disposal system is defined as the combination of engineered and natural barriers that isolate disposed waste from events and processes that are capable of affecting isolation of the waste. The key feature of the disposal system is the Salado Formation which provides a critical natural barrier to contaminant migration from the repository. The engineered barrier system includes materials emplaced as backfill and seal closures installed in drifts, shafts, and boreholes. The following overview of the conceptual model covers some of the principal aspects of the disposal system assumed for this analysis. A summary of critical assumptions regarding brine and gas migration taken from the CCA (Appendix MASS, DOE 1996f) is provided in Table H-3.

H.4.1 Repository System

The repository system contains a number of key elements that contribute to its overall long-term performance. Brief descriptions of these elements are described in Sections H.4.1.1 to H.4.1.7.

H.4.1.1 Salt Creep

Salt creep is an important process in the conceptual model of the disposal system. It occurs naturally in Salado Formation halite in response to deviatoric stress created by the excavation of the repository. Closure of the waste disposal panels by salt creep will eventually consolidate waste in the disposal areas until an equilibrium with the surrounding rock is reached. The shaft and repository excavation have resulted in a system of fractures caused by stress relief within the salt.

Table H-3
Brine and Gas Migration Modeling Assumptions

MODELING AREA	MODELING ASSUMPTIONS
General Assumptions	1. Flow is governed by mass conservation and Darcy's Law in porous media. Flow is laminar and fluids are Newtonian. 2. Two-phase flow in the porous media is by simultaneous immiscible displacement. 3. The Brooks-Corey or Van Genuchten/Parker equations represent interaction between brine and gas. 4. The Klinkenberg effect is included for flow of gases at low pressures. 5. Threshold displacement pressure for flow of gas into brine is constant. 6. Fluid composition and compressibility are constant. 7. The gas phase is assigned the density and viscosity properties of hydrogen. 8. All liquid physical properties are assigned the properties of Salado brine.
Model Geometries	The disposal system is represented by a two-dimensional, north-south, vertical cross section.
Disposal System Geometry	Flow in the disposal system is radially convergent or divergently centered on the repository, shaft, and borehole for disturbed performance. Variable dip in the Salado is approximated by a 1 degree dip to the south. Stratigraphical layers are parallel. The stratigraphy consists of units above the Dewey Lake, the Dewey Lake, the Forty-niner, the Magenta, the Tamarisk, the Culebra, the Unnamed lower member, and the Salado (comprising impure halite, MB 138, anhydrites a and b [lumped together], and MB 139). The dimensions of these units are constant. A Castile brine reservoir is included in all scenarios.
Culebra Geometry	The Culebra is represented by a two-dimensional, horizontal geometry for groundwater flow and radionuclide transport simulation. Transmissivity varies spatially. There is no vertical flow to or from the Culebra. The regional flow field provides boundary conditions for local transport calculations.
The Repository	The repository comprises five regions: a waste panel, the panel closures, the remainder of the panels and the access drifts, the operations region, and the experimental region. Also, a single shaft region is modeled, and a borehole region is included for a borehole that intersects the separate waste panel. The dimensions of these regions are constant. Long-term flow up plugged and abandoned boreholes is modeled as if all intrusions occur into a down-dip (southern) panel. For each repository region the model geometry preserves design volume. Pillars and individual drifts and rooms and panel closures in the nine lumped panels are not modeled for long-term performance, and containers provide no barrier to fluid flow. The distance from the south end of the modeled waste panel to the modeled shaft is the true distance from the south end of the waste disposal region to the waste handling shaft. Long-term flow is radial to and from the borehole that intersects the waste disposal panel during disturbed performance. Panel closures are modeled with the same properties as the surrounding DRZ.
Creep Closure	Creep closure is modeled using a two-dimensional model of a single room. Room interactions are insignificant. Creep closure causes a decrease in room volume which decreases waste porosity. The amount of creep closure is a function of time, gas pressure, and waste matrix strength. Porosity of operations and experimental areas is fixed at a value representative of consolidated material.

Table H-3
Brine and Gas Migration Modeling Assumptions — Continued

MODELING AREA	MODELING ASSUMPTIONS
Repository Fluid Flow	General Assumptions 1 through 8. The waste disposal region is assigned a constant permeability representative of average consolidated waste without backfill. The experimental and operations regions are assigned a constant permeability representative of unconsolidated material and a constant porosity representative of consolidated material. For gas generation calculations, the effects of wicking are accounted for by assuming that brine in the repository contacts waste to an extent greater than that calculated by the Darcy flow model used.
Gas Generation	Gas generation occurs by anoxic corrosion of steel containers and Fe and Fe-base alloys in the waste, giving H_2, and microbial degradation of cellulosics and, perhaps, plastics and rubbers, giving mainly CO_2 and CH_4. Radiolysis, toxic reactions, and other gas generation mechanisms are insignificant. Gas generation is calculated using the Average Stoichiometry model and is dependent on brine availability. The anoxic corrosion rate is dependent on liquid saturation. Anoxic corrosion of steel continues until all the steel is consumed. Steel corrosion will not be passivated by microbially generated gases CO_2 or $H2_S$. Brine is consumed by the corrosion reaction. Laboratory-scale experimental measurements of gas generation rates at expected room temperatures are used to account for the effects of biofilms and chemical reactions. The rate of biodegradation is dependent on the amount of liquid present. It is assumed that biodegradation neither produces nor consumes water. Gas generation by microbial degradation takes place in half the simulations. In half of the simulations with microbial gas generation, microbes consume all of the cellulosics but none of the plastics and rubbers. In the other half of the simulations with microbial gas generation, microbes consume all of the cellulosics and all of the plastics and rubbers. Microbial gas generation will continue until all biodegradable organic materials are consumed if brine is present. The MgO backfill will react with CO_2 and remove it from the gas phase. Gas dissolution in brine is of negligible consequence. The gas phase is assigned the properties of hydrogen.
Chemical Conditions in the Repository [a]	Chemical conditions in the repository will be constant. Chemical equilibrium is assumed for all reactions that occur between brine in the repository, waste, and abundant minerals, with the exceptions of gas generation and redox reactions. Brine and waste in the repository will contain a uniform mixture of dissolved and solid-state species. No microenvironments that influence the overall chemical environment will persist. For the undisturbed performance and exploratory borehole scenarios, brine in the waste panels has the composition of Salado brine. For scenarios penetrating the repository and pressurized brine reservoir, all brine in the waste panel intersected by the borehole has the composition of Castile brine. Chemical conditions in the waste panels will be reducing; however, a condition of redox disequilibrium will exist between the possible oxidation states of the actinide elements. The pmH and pCO_2 in the waste panels will be controlled by the equilibrium between brucite and magnesite. (A result of this assumption is low pCO_2 and alkaline conditions.)
Dissolved Actinide Source Term [b, c]	Radionuclide dissolution to solubility limits is instantaneous. Six actinides (Th, U, Np, Pu, Cm, and Am) are considered for calculations of radionuclide transport of brine. Choice of radionuclides is discussed in Appendix H. The reducing conditions in the repository will eliminate significant concentrations of Am(V), Pu(V), Pu(VI), and NP(VI) species. Am and Cm will exist predominantly in the III oxidation state, Th in the IV oxidation state. It is assumed that the solubilities and Kds of Pu, Np, and U will be dominated by one of the remaining oxidation states: Pu(III) or Pu(IV), Np(IV) or NP(V), U(IV) or U(VI). For a given oxidation state, the different actinides exhibit similar chemical behavior and thus have similar solubilities. Organic ligands will not significantly affect solubility. For undisturbed performance and for all aspects of disturbed performance except for cuttings and cavings releases, radionuclide-bearing compounds are distributed evenly throughout the disposal panel. Mobilization of actinides in the gas phase is negligible. Actinide concentrations in the repository will be inventory limited when the mass of an actinide becomes depleted such that the predicted solubilities cannot be achieved.

Table H-3
Brine and Gas Migration Modeling Assumptions — Continued

MODELING AREA	MODELING ASSUMPTIONS
Source Term for Colloidal Actinides	Four types of colloids comprise the source term for colloidal actinides: microbes, humic substances, intrinsic colloids, and mineral fragments. The only intrinsic colloids that will form are those of the plutonium Pu(IV) polymer. Concentrations of intrinsic colloids and mineral fragment colloids are modeled as constants that were based on experimental observations. Humic and microbe colloid actinide concentrations are modeled as proportional to dissolved actinide concentrations. The maximum concentration of each actinide associated with each colloid type is constant.
Shafts and Shaft Seals	General Assumptions 1 to 8. The four shafts connecting the repository to the surface are represented by a single shaft with a cross section and volume equal to the total volume of the four real shafts and separated from the waste by the distance of the nearest real shaft. The seal system is represented by nine materials occupying eleven model regions. The shaft is surrounded by a DRZ which heals with time. The DRZ is represented through the permeabilities of the shaft system itself, rather than as a discrete zone. The effective permeability of shaft salt, clay, and concrete seals are adjusted several times after closure to reflect consolidation and possible degradation. Permeabilities are constant for asphalt and earthen-fill components. Concrete shaft components are modeled as if they degrade 400 years after emplacement. Radionuclides are not retarded by the seals.
Salado Formation	General Assumptions 1 to 8.
Impure Halite	Rock and hydrologic properties are constant.
Salado Interbeds	Interbeds have a fracture-initiation pressure above which local fracturing and changes in porosity and permeability occur in response to changes in pore pressure. A power function relates the permeability increase to the porosity increase. A pressure is specified above which porosity and permeability do not change. Interbeds have identical physical properties; they differ only in position, thickness, and some fracture parameters.
Disturbed Rock Zone	The permeability of the DRZ is constant and higher than intact Salado. The DRZ porosity is equal to the porosity of impure halite to +0.29 percent.
Actinide Transport in the Salado	Dissolved actinides and colloidal actinides are transported by advection in the Salado. Diffusion and dispersion are assumed negligible. Sorption of actinides in the anhydrite interbeds, colloid retardation, colloid transport at higher than average velocities, co-precipitation of minerals containing actinides, channeled flow, and viscous fingering are not modeled. Sorption of actinides in the borehole is not modeled.
Units Above the Salado	Above the Salado, lateral actinide transport to the accessible environment can occur only through the Culebra.
Unnamed Lower Member	The Unnamed lower member is assumed to be impermeable.

Table H-3
Brine and Gas Migration Modeling Assumptions — Continued

MODELING AREA	MODELING ASSUMPTIONS
Culebra Dolomite Member	General Assumptions 1, 6, and 8. For fluid flow, the Culebra is modeled as a uniform (single-porosity) porous medium. For radionuclide transport, a double-porosity model is used (advection in high permeability features and diffusion and sorption in low-permeability features). The Culebra flow field is determined from the observed hydraulic conditions and estimates of the effects of potash mining outside the controlled area and does not change with time unless mining is predicted to occur in the disposal system in the future. The Culebra is assigned a single permeability to calculate brine flow into the unit from an intrusion borehole. Gas flow in the Culebra is not modeled. Gas from the repository does not affect fluid flow in the Culebra. Different thicknesses of the Culebra are assumed for BRAGFLO, SECOFL-2D, and SECOTP-2D calculations, although the transmissivities are consistent. Uncertainty in the spatial variability of the Culebra transmissivity is accounted for by statistically generating many transmissivity fields. Potentiometric heads are set on the edges of the regional grid to represent flow in a portion of a much larger hydrologic system.
- *Transport of Dissolved Actinides in the Culebra*	Dissolved actinides are transported by advection in high-permeability features and diffusion in low permeability features. Sorption occurs on dolomite in the matrix. Sorption on clays present in the Culebra is not modeled. Sorption is represented using a linear isotherm model. The possible effects on sorption of the injection of brines from the Castile and Salado into the Culebra are accounted for in the distribution of actinide Kds. Hydraulically-significant fractures are assumed to be present everywhere in the Culebra.
- *Transport of Colloidal Actinides in the Culebra*	Humic actinides are chemically retarded identically to dissolved actinides and are treated as dissolved actinides. The concentration of intrinsic colloids is sufficiently low to justify elimination from performance assessment transport calculations. Microbial colloids and mineral fragments are too large to undergo matrix diffusion. Filtration of these colloids occurs in high permeability features (which is modeled using a decay approach). Attenuation is so effective that associated actinides are assumed to be retained within the disposal system and are not transported in SECOTP-2D
- *Potash Mining*	Subsidence due to potash mining increases the hydraulic conductivity in the Culebra by a factor from 1 to 1,000.
Tamarisk Member	The Tamarisk is assumed to be impermeable.
Magenta Member	General Assumptions 1 to 8. The Magenta permeability is set to the lowest value measured near to the center of the WIPP site. This increases the flow into the Culebra. No radionuclides entering the Magenta will reach the accessible environment. However, the volumes of brine and actinides entering and stored in the Magenta are modeled.
Forty-niner Member	The Forty-niner is assumed to be impermeable.

Table H-3
Brine and Gas Migration Modeling Assumptions — Continued

MODELING AREA	MODELING ASSUMPTIONS
Dewey Lake Member	General Assumptions 1 to 8. The sorptive capacity of the Dewey Lake is large enough to prevent any release over 10,000 years.
Supra-Dewey Lake Units	General Assumptions 1 to 8. The units above the Dewey Lake (the Gatuno and Santa Rosa Formations) are a single hydrostratigraphic unit. The units are thin and predominantly unsaturated.
The Intrusion Borehole	Any actinides that enter the borehole are assumed to reach the surface.
Cuttings, Cavings, and Spall Releases during Drilling	Future drilling practices will be the same as they are at present. Releases of particulate waste material are modeled (cuttings, cavings, and spallings). Releases are corrected for radioactive decay until the time of intrusion. Particle waste shear based on properties of marine clay, considered a worst case.
Long-Term Releases Following Drilling	Plugging and abandonment of future boreholes are assumed to be consistent with practices in the Delaware Basin. A two-plug configuration was assumed. A lower plug is located between the Castile brine reservoir and underlying formations. A second plug is located immediately above the Salado. The brine reservoir and waste panel are in direct communication though an open-cased hole. The casing and upper concrete plug are assumed to fail after 200 years, and the borehole is assumed to be filled with silty sand-like material. At 1,200 years after abandonment, the permeability of the borehole below the waste panel is decreased by one order of magnitude as a result of salt creep. The diameter of the intrusion borehole is constant at 12.25 inches (31.12 centimeters).
Castile Brine Reservoir	The Castile region is assigned a low permeability, which prevents fluid flow. Brine occurrences in the Castile are bounded systems. Brine reservoirs under the waste panels are assumed to have limited extent and interconnectivity, with effective radii on the order of several hundred meters.
Initial and Boundary Conditions for Disposal System Modeling	There are no gradients for flow in the far-field of the Salado, and pressures are above hydrostatic but below lithostatic. Excavation and waste emplacement result in partial drainage of the DRZ. Initial and boundary conditions for other computational models were interpolated from previously executed BRAGFLO calculations.
Disposal System Flow and Transport Modeling	An initial water table surface is set in the Dewey Lake at an elevation of 3,215 feet (980 meters) above mean sea level. The initial pressures in the Salado are extrapolated from a sampled pressure in MB 139 at the shaft and are in hydrostatic equilibrium. The excavated region is assigned an initial pressure of one atmosphere. The liquid saturation of the waste-disposal region is consistent with the liquid saturation of emplaced waste. Other excavated regions are assigned zero liquid saturation, except the shaft which is fully saturated. Molecular transport boundary conditions are no diffusion or dispersion in the normal direction across far-field boundaries. Initial actinide concentrations are zero everywhere except in the waste.
Culebra Flow and Transport Modeling	Constant head boundary conditions are set on the far-field boundaries of the regional flow model. Constant head boundary conditions are also set on the boundaries of the local domain, and are derived by interpolating the solution of the regional domain. Initial actinide concentrations in the Culebra are zero.

[a] The terms pmH and pCO_2 represent the $\log_{10}$ of the molality of the hydrogen ion and the partial pressure of carbon dioxide, respectively.
[b] Roman numerals represent different oxidation states for the radionuclides.
[c] The term "Kds" represents the distribution coefficients.

These fractures, which surround the shaft and excavation, create what is referred to as a "disturbed" rock zone. The DRZ will develop within the Salado Formation around shafts connecting the repository to the surface. The process of salt creep will partially heal fractures in the Salado Formation halites, leading to a general reduction in the overall permeability and porosity within the DRZ over time.

H.4.1.2 Brine Flow

Pressure gradients created by the excavation of the repository will cause brine in the surrounding rocks to flow into the waste disposal panels. Brine flow into the repository decreases as the repository pressure increases, as a result of the generation of gas from waste degradation. Conceptually, brine could be expelled from the repository should pressure in the repository exceed the brine pressure in the surrounding rock.

H.4.1.3 Gas Generation

Gases such as hydrogen gas, carbon dioxide (CO_2), and methane will be generated as waste stored in the repository comes into contact with inflowing brine and degrades via a variety of chemical and microbial processes. These processes are expected to degrade metals, cellulose and similar materials (cellulosics), and plastics and rubber materials contained within the disposed waste. The dominant gas-generating processes are anoxic corrosion of ferrous metals in the waste and waste containers and the microbial degradation of cellulosics, plastic, and rubber in the waste. In general, as gas pressure rises as a result of repository closure and gas generation, increased pressure will impede creep closure and consolidation of the waste region. Gas generation is expected to cause fracturing or increase the porosity of existing fractures of anhydrites beyond the DRZ as repository pressures approach lithostatic levels.

H.4.1.4 Source-Term Release Mechanisms

As rooms and access drifts are closed by the process of salt creep, waste containers will be crushed and breached. In the absence of backfill which would slow this process, the chemical conditions in the after closure environment would rapidly become reduced (anoxic) as oxygen is consumed by initially toxic reactions, as gas is generated by waste degradation, and as brine fills the void volume in the waste disposal region. Radioactive and hazardous constituents would be released as waste drums are breached and waste comes into contact with brine and gas. For liquid-phase contaminants to be generated, sufficient brine inflow must occur to dissolve the waste constituents in the solid phase or serve as a medium for partitioning of vapor-phase organics into the brine. Furthermore, repository conditions (pressure and temperature) and chemical conditions (pH and Eh) must be suitable to dissolve and mobilize metals existing as elemental metals or salts. For this SEIS-II analysis, as in the CCA (DOE 1996f) analyses, it was conservatively assumed that instantaneous dissolution of waste containers and immediate mobility of radioactive and hazardous constituents in the gas and liquid phases at WIPP closure would occur.

H.4.1.5 Dilation and Fracturing of the Anhydrite Interbeds

Gas generated by waste degradation approaching lithostatic pressure is expected to fracture the anhydrite interbeds and dilate the existing fractures in the vicinity of the repository and, thus, enhance flow and mass transport of contaminants. Potential transport processes within the fractured interbeds include advection, diffusion, dispersion, fracture-matrix flow, channeling and fingering, retardation, and sieving. Mass transport in gas and brine flow in unfractured anhydrite beds is possible; however, because of their low permeability, transport in these beds is not likely to be important.

H.4.1.6 Repository Features

The reference design of the repository under the Proposed Action contains 10 panel equivalents. Each panel consists of seven disposal rooms and connecting access drifts. These areas will be sequentially filled with waste and then sealed. The repository size for Action Alternatives 1, 2, and 3 are 68, 75, and 71 panel equivalents, respectively. The repository size is embedded in the grid geometry for BRAGFLO that simulate brine and gas flow in the repository and surrounding region.

H.4.1.7 Engineered Components

Current plans make use of cylindrical seals consisting of salt columns interleaved with concrete plugs, clay, and other engineered materials to seal the repository access drifts and shafts from inflowing groundwater and to reduce the migration of contaminants through the repository and shaft system. The seals will be emplaced in the four shafts connecting the repository to the surface.

Use of magnesium oxide backfill is also planned to provide chemical control of the solubility of radionuclides in the after closure repository environment. Long-term performance calculations for the Proposed Action and action alternatives include the effects of the magnesium oxide backfill. Actinide solubility in the repository is highly dependent on pH conditions and the oxidation state of the actinide. Gas generation resulting from microbial degradation of carbon in waste materials is expected to generate CO_2, lower pH, and generate carbonate species that bind very strongly to actinides, as complexes, to form relatively highly soluble actinide species. The presence of appropriate amounts of magnesium oxide is expected to react with brine that may reach the repository and any CO_2 gas generated as a result of microbial action to maintain a sufficiently high pH and minimize the formation of carbonate complexes that result in higher actinide solubilities. More details on the theory behind the current plans for this backfill are described in Appendix BACK of the CCA (DOE 1996f).

H.4.2 Salado Formation

The Salado Formation is the principal natural barrier to fluid flow between the repository and the accessible environment. For the purpose of this analysis, the Salado is conceptualized as a porous medium composed of several rock types arranged in layers, except in the vicinity of the repository where stress-relief fractures have disrupted the continuous layers. Near the repository, the DRZ in the Salado is conceptualized as a zone of increased permeability and porosity, offering little resistance to flow between the repository and the surrounding rocks. The intact Salado consists of sequences of two rock types, impure halite and anhydrite. These rock types are assumed to be a homogeneous porous medium with spatially constant properties. Specific information and model inputs used in this analysis to represent the major rock types and the DRZ in the numerical models of the repository are summarized in Section H.6.

H.4.3 Units Above and Below the Salado Formation

Elements of the disposal system model conceptualized above and below the Salado Formation include units within the Rustler Formation, the Dewey Lake Formation (also called the Redbeds) and supra-Dewey Lake units, and the Castile Formation with associated brine reservoirs. Brief descriptions of each of these elements are provided below.

H.4.3.1 Rustler Formation

The Rustler Formation is conceptualized as having five recognized members: the Unnamed lower, the Culebra Dolomite, the Tamarisk, the Magenta, and the Forty-Niner. The Unnamed lower member of the Rustler Formation is characterized by relatively low transmissivity and thus is treated as an impermeable unit.

The Culebra Dolomite is conceptualized as the most permeable unit within the Rustler Formation and, therefore, the most notable unit when considering the long-term release from WIPP to the accessible environment. It may be possible for radionuclides and other hazardous constituents to travel up the sealed shafts, through gas or brine flow, into the Culebra as a result of high repository pressure. Gas and/or brine flowing up either exploratory boreholes that have penetrated the repository or deeper pressurized brine pockets below the repository could also be introduced into the Culebra.

Human-intrusion scenarios may introduce gas into the boreholes either from such sources as the Castile Formation brine reservoirs or from gas in the repository generated in the waste. Because of the lower pressures in the units above the Salado Formation, the gas volume will expand over the volume occupied in the reservoir. Gas bubbles could alter the natural flow patterns and velocities of the brine because of flow blockage and density differences, and the gas bubbles could migrate differently than the brine. Fracture-matrix flow could be considerably changed with the introduction of gas as a result of capillary pressure difference between the phases, possibly resulting in gas flow primarily in fractures and brine flow restricted to the matrix. Chemistry could also be changed by the introduction of gas into the non-Salado Formation units. For the purpose of this analysis, the Culebra Dolomite was analyzed using single-phase or fully saturated approaches for flow and transport, assuming that two-phase conditions associated with gas release do not substantially impact transport calculations.

According to the CCA (DOE 1996f), an intrusion borehole connecting the Culebra with the Salado and a Castile brine reservoir could allow brine to flow into the Culebra which, in turn, could cause hydraulic head in the Culebra to increase, locally inducing radial flow from the borehole. The area affected would depend on the transmission and storage capabilities of the Culebra, and on the flow rate into the Culebra. The study by Reeves et al. (1991) indicated that, for much of the range of brine reservoir and breach borehole parameters, the fluid disturbance created in the Culebra by the borehole had minimal impact on the flow field. In addition, transport calculations under these conditions need not include the transient impact of locally increased hydraulic head near the breach borehole. This study also found that, for extreme conditions at the high end of the brine reservoir volume, pressure range, and borehole permeabilities, travel time for a conservative solute can be reduced by as much as 7.4 percent because of increased heads in the vicinity of the borehole. This effect can be implemented in an undisturbed flow field by increasing hydraulic heads in the vicinity of the borehole. The fluid from an intrusion borehole could have the following four effects on flow and transport in the Culebra:

- Increase in hydraulic gradients and flow velocities in the vicinity of the borehole
- Change in the density of the Culebra fluid
- Result in rock/water interactions that locally alter flow and transport properties in the Culebra
- Result in local multiphase flow conditions in the Culebra

For the purpose of this analysis, SEIS-II did not consider any of these specific processes in analyzing the impacts of a borehole intrusion on the Culebra.

If radionuclides reach the Culebra, they may be transported to off-site receptors from the point of introduction by groundwater flowing through the Culebra. Radionuclide transport in the Culebra is represented in this analysis by two-dimensional flow through a horizontal, confined aquifer containing fractures and spatially variant transmissivity.

According to the CCA (DOE 1996f), the Culebra Dolomite is a double-porosity medium at some locations on and around WIPP. Double-porosity simply means that the Culebra has a porosity attributable to its rock matrix and another attributable to fractures. Allowing flow and transport in fractures within the Culebra generally overestimates transport in those areas where the Culebra has a low transmissivity and has been interpreted as a single-porosity, matrix-only medium. In a double-porosity, fractured medium, flow is generally conceptualized as occurring primarily in the fractures, because flow velocities are usually orders of magnitude higher in the fractures than in the matrix. However, the process of diffusive (or advective) transport of radionuclides or contaminants from fractures into the matrix can physically retard these substances.

For the purpose of this analysis, SEIS-II has adopted the conceptual model for Culebra flow and transport used in previous WIPP performance assessments. The double-porosity conceptualization used in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992) and, more recently, in the CCA (DOE 1996f) analyses assume that advective transport occurs only in fractures, with diffusion of radionuclides and other contaminants occurring between the fractures and matrix.

Interactions of brines containing radionuclides and hazardous contaminants with the Culebra Dolomite and, in particular, clay mineral linings on fracture surfaces have been postulated as having the potential to cause chemical retardation. The CCA (DOE 1996f) reports that, for the purpose of estimating contaminant transport in groundwater, the Culebra is best characterized as a double-porosity medium. Groundwater flow and advective transport of dissolved species or colloidal particles occur primarily in a small fraction of the total rock porosity, which corresponds to the open and interconnected fractures and vugs. Diffusion and slower flow occur in the remainder of the porosity that is associated with the low-permeability dolomite matrix. Transported species, including actinides if present, will diffuse into this porosity. Diffusion out of the advective porosity into the dolomite matrix will retard actinide transport. Physical retardation occurs when actinides that diffuse into the matrix are no longer transported with the flowing groundwater, and transport is interrupted until they diffuse back into the advective porosity. In situ tracer tests have been conducted to demonstrate this phenomenon. Chemical retardation occurs within the matrix as actinides are sorbed onto dolomite grains. The relationship between sorbed and liquid concentrations is assumed to be linear, and the distribution coefficients (K_ds) that characterize the extent to which actinides will sorb on dolomite are based on experimental data (Appendix MASS.15, DOE 1996f).

The Tamarisk Member rests between the more transmissive Culebra Dolomite and Magenta Dolomite Members of the Rustler Formation. Like the Unnamed lower member, the Tamarisk Member does not have a high transmissivity. For the purpose of this analysis, this member was treated as impermeable.

Although the Magenta Dolomite Member is transmissive, transport of radionuclides within this unit was not considered because it has been shown that any radionuclides within the Magenta will not reach the site boundary in 10,000 years (Barr 1983).

Because of its low permeability, the Forty-Niner Member is considered to be relatively unimportant for flow and transport analysis (Beauheim [1986] and Beauheim et al. [1991]). For the purpose of this analysis, the Forty-Niner Member is considered impermeable.

H.4.3.2 Dewey Lake (Redbeds) Member and Supra-Dewey Lake Units

The Dewey Lake (Redbeds) Member is conceptualized as having a low permeability compared to that of the Culebra Dolomite Member. Because of the high adsorptive capacity of the redbeds, transport of radionuclides in this member is assumed to be negligible.

For the purpose of this analysis, all units above the Dewey Lake (Redbeds) Member, the Gatuna and Santa Rosa Formations, were assumed to behave as a single hydrogeologic unit of relatively high permeability.

H.4.3.3 Castile Formation and Brine Reservoirs

The Castile Formation is postulated to have a very low permeability because of its evaporite content. However, brine under high pressure has been encountered in the Castile Formation (the WIPP-12 borehole) within the disposal system boundary and the U.S. Energy Research and Development Administration borehole number 6 (ERDA-6 borehole). The connection of a brine reservoir in the Castile with the waste panels at the repository level and overlying units by an exploratory borehole has been postulated as a possible human-intrusion scenario (Appendix MASS.18, DOE 1996f).

H.5 CONCEPTUAL MODEL OF THE SOURCE-TERM RELEASE

The TRU waste to be disposed of at WIPP includes radionuclides, heavy metals, and VOCs. TRU waste is primarily packaged at the generator-storage sites in metal drums or containers. The containers may also include several internal barriers: layers of plastic; plastic, metal, and glass containers; and adsorbents in the void spaces. VOCs and semivolatile organic compounds are present within solidified liquids and sludges and in trace quantities sorbed onto cellulosics and other solid waste materials. A VOC gas/vapor phase dominates void spaces within the container and within the inner layers of confinement. Heavy metals, mainly lead used for radiation shielding, will exist in the solid phase. Other regulated metals may occur as trace contaminants in soil, debris, sludges, and solidified liquids and as components of metal tools, equipment, and machinery. The planning-basis WAC are designed to preclude the presence of free liquids;

therefore, a large initial liquid phase is not a feature of the conceptual model. Conceptually, the liquid phase would result from brine inflow and waste dissolution in the closed repository.

DOE has considered the potential effects of a number of possible chemical and thermal processes in the disposal system environment. These processes include corrosion, microbial activity, radiolysis, dissolution reactions, reactions with cementitious materials, and adsorption/desorption. These processes can either immobilize or enhance the mobility of radionuclides, metals, and VOCs. For practical purposes, not enough information is known about many long-term chemical processes in the WIPP environment to accurately model the potential effects of each of these processes. In cases where information was limited, assumptions were made that increased the mobility of radionuclides and metals, increasing the extent of contaminant migration and the potential for reaching the accessible environment.

H.5.1 Key Assumptions for Source-Term Release

The assumptions presented below on the source-term conceptual model used in the SEIS-II analyses are consistent with many of the assumptions on the source-term release conceptual models used in the CCA (DOE 1996f).

- Waste containers were assumed to lose their capacity to isolate waste at the time of repository closure. In addition, radionuclides and heavy metals were assumed to instanteously dissolve up to the solubility limit for each element in brine. Thus, contaminants in the gas and liquid phases would be mobile at repository closure. This is a conservative assumption because the pressure, temperature, and chemical conditions (such as pH and Eh) may not be sufficient to completely dissolve and mobilize constituents. It is also conservative because a number of time-dependent mechanisms would decrease the initial source-term concentrations of contaminants.

- Some radionuclides are mobilized either by dissolution in brine as intrinsic colloids or by adsorption onto colloidal particles carried by the brine. Such radionuclides exist primarily in the liquid phase, decrease in radioactivity through radioactive decay, and may decay into one or a chain of radioactive progeny. For the purpose of these analyses, solubility-controlled releases of key actinides were considered but adsorption processes were not considered in the transport of radionuclides vertically from the repository to the Culebra. However, adsorption processes were modeled in the horizontal movement of radionuclides through the Culebra. This is a conservative assumption because adsorption will retard the near-field release and transport of radionuclides. Colloids and organic-complexing agents capable of enhancing or inhibiting constituent mobility and transport are not modeled because of a lack of waste-specific and repository-specific information.

- Metals exist in each phase in the waste disposal panels at constant concentration over time. This is a conservative assumption because it assumes an infinite source and complete persistence. In reality, however, some metals and organics will migrate from the source-term region and organics will likely degrade. Gas generation can deplete the metals of cellulose.

- Migration of metals from the repository does not decrease the initial source-term concentration. In reality, however, some of these constituents would likely migrate away from the waste disposal panels and the source term should decrease with time.

- The waste-loading strategy for both contact-handled (CH) and remote-handled (RH) TRU wastes limits the heat-generation rate to less than 24 kilowatts per hectare (10 kilowatts per acre) (DOE 1996b). At this level, heat generation is expected to have inconsequential effects on flow and transport processes and is not considered in this analysis.

For more details of the conceptual model for source-term release, see the Appendix SOTERM of the CCA (DOE 1996f) where model descriptions for the liquid-phase source-term are provided.

H.5.2 Source-Term Constituents

As noted above, the TRU waste source term disposed of at WIPP includes radionuclides, heavy metals, and VOCs. SEIS-II performance assessment analyses evaluated the release, transport, and exposure potential for constituents of each of these categories. The inventory on which these analyses were performed is described in Appendix A. These analyses differed from those performed for the CCA (DOE 1996f).

H.5.2.1 Screening and Selection of Radionuclides

Screening calculations were performed to determine which radionuclides were significant contributors to radiological impacts. A total of 52 radionuclides were evaluated for potential dose from groundwater ingestion. The relative contribution to radiation dose was determined considering the radionuclide-specific ingestion dose factor, the solubility of the element in water, and the inventory of the radionuclide in the WIPP repository. The solubilities for each element were taken from published performance assessments (SNL 1992); if the solubility was not available in published literature, a value of 1 x 10^{-6} molar was used. The amount of water available for dissolution was assumed to be the repository volume adjusted for a waste and backfill porosity of 0.25. The total inventory of each radionuclide was assumed to go into solution. Radionuclides with the highest inventory (in curies [Ci]) for Action Alternative 1 are shown in Table H-4. Because active institutional controls are assumed to be in place for the first 100 years after closure and no disturbance of the repository is anticipated, the radionuclide screening was performed using the inventory at 100 years after closure rather than using the emplaced inventory.

The relative contribution to radiological impacts for key radionuclides at 100 and 1,000 years after closure is presented in Table H-5. These two cases were evaluated to reflect exposure scenarios for human intrusion and long-term performance assessment discussed in Section H.8 and to illustrate the effect radionuclide half-life. For example, cobalt-60 (Co-60) has a 5.26-year half-life. Although over 50,000 Ci of Co-60 may be emplaced in the WIPP repository, it will have decayed to a small amount within 100 years. Therefore, radionuclides with a short half-life, no parent radionuclide in the inventory, and no long-lived radioactive progeny were eliminated from consideration. The radionuclides shown in Table H-5 contribute greater than 99 percent of the total dose from the ingestion pathway. A number of them, particularly for 1,000 years after closure, are radioactive progeny produced by radioactive decay of a parent radionuclide. Because element solubilities were uncertain or unknown and conservative (high solubility) values were used, the entire inventories of radionuclides could be dissolved in small quantities of water. Therefore, the list of contributing radionuclides is more dependent on each radionuclide’s

Table H-4
Radionuclides with Highest Inventories (curies) for Action Alternative 1

CH-TRU Waste		RH-TRU Waste	
Radionuclide	**Inventory (Ci)**	**Radionuclide**	**Inventory (Ci)**
Pu-241	2,623,553	Cs-137	1,085,025
Pu-238	2,413,581	Y-90	1,056,299
Pu-239	923,975	Sr-90	1,056,096
Am-241	575,924	Ba-137m	1,026,566
Pu-240	204,657	Pu-241	706,676
Y-90	8,937	Co-60	52,221
Sr-90	8,935	Pu-239	51,455
Cs-137	7,045	Am-241	29,911
Cm-244	6,843	Pu-240	25,318
Ba-137m	6,664	Pu-238	7,509
U-233	4,019	Eu-152	7,308
Pu-242	1,550	Eu-154	3,550

Table H-5
Radionuclide Screening of Dose Contribution for Long-term Performance Assessment

100 Years After Closure		1,000 Years After Closure	
Radionuclide	**Percent Contribution**	**Radionuclide**	**Percent Contribution**
Sr-90	72.2	Ac-227	32.6
Cs-137	7.7	Pa-231	26.4
U-233	6.5	U-233	21.3
Cm-244	3.0	Pu-238	6.5
Cm-243	2.9	U-234	3.9
Pu-238	2.0	Am-241	2.6
Ac-227	1.1	Ra-225	2.3
Pa-231	1.1	Ra-223	1.3
U-234	0.9	Pb-210	1.0
Am-241	0.8	Ac-225	0.7
Pb-210	0.4	Po-210	0.3
Pu-241	0.3	Ra-226	0.2
U-232	0.2	---	---

inventory rather than on its solubility. Moderate changes in the solubilities would likely yield the same of radionuclides. All radionuclides would be expected to migrate slowly under undisturbed repository conditions.

The initial set of radionuclides considered for performance assessment analysis included nine having the highest inventory: plutonium (Pu)-238, Pu-239, Pu-240, Pu-241, americium (Am)-241, cesium (Cs)-137, barium (Ba)-137m, yttrium (Y)-90, and strontium (Sr)-90. Y-90 and Ba-137m were subsequently dropped from the list because of their short half-life; however, their impacts are included in the reported dose values because the dose factors used for Sr-90 and Ba-137m include the decay energies of Y-90 and Ba-137m. Based on the results of the radionuclide screening, actinium (Ac)-227, protactinium (Pa)-231, uranium (U)-233, U-234, curium (Cm)-243, Cm-244, and lead (Pb)-210 were also added to the list.

The final list of radionuclides for long-term performance assessment analyses consisted of 15 radionuclides: Ac-227, Am-241, Cs-137, Cm-243, Cm-244, Pa-231, Pb-210, Pu-238, Pu-239, Pu-240, Pu-241, U-232, U-233, U-234, and Sr-90. Two of the radionuclides, Cs-137 and Sr-90, are fission products; the remainder are uranium isotopes, transuranic radionuclides, or radioactive progeny from the decay of uranium or transuranic radionuclides. For impacts beyond 100 years, essentially all of the Cm-243 and Cm-244 will have decayed into Pu-239 and Pu-240, respectively. A total of 30 radionuclides were carried in several decay chains in the computer codes NUTS and CUTTINGS_S to maintain correct inventories of the chosen radionuclides.

H.5.2.2 Heavy Metals

Four heavy metals (lead, mercury, cadmium, and beryllium) were included in SEIS-II analyses of WIPP long-term performance. The inventory of these metals for each alternative is shown in Table A-45 of Appendix A. As noted in Section A.5.1, the selection of these metals and their estimated inventory was based on information presented in the *Waste Isolation Pilot Plant Safety Analysis Report* (SAR) (DOE 1995, Table 5.1-2) because the *Transuranic Baseline Inventory Report, Revision 3* (BIR-3) (DOE 1996e) does not contain detailed information on hazardous constituents in TRU waste.

H.5.2.3 VOCs

VOCs were not included in SEIS-II long-term performance assessment calculations for WIPP because these chemicals were not included in parameter databases and analyses performed as part of the WIPP Compliance Certification Application (DOE 1996f). In the *Final No-Migration Variance Petition* (DOE 1996d), the Department evaluated the potential for migration of hazardous constituents of TRU waste, including VOCs, from the disposal system for undisturbed conditions. Simulation of the migration of gaseous compounds based on conservatively high gas generation rates from waste degradation demonstrated zero gas saturation at all subsurface disposal unit boundaries except for the shaft. Less than one cubic meter of potentially contaminated gas was predicted to occur at the unit boundary of the shaft. Calculated bounding soil-based concentrations based on estimated gas-available porosity within the shaft seals and within anhydrite marker beds were found to be orders of magnitude below U.S. Environmental Protection Agency (EPA) approved health based levels. The Department was able to demonstrate that, for undisturbed conditions, there would be no migration of VOCs to the accessible environment. For the purposes of SEIS-II, potential impacts from exposure to VOCs released from the repository were assumed to be bounded by quantitative estimates of impacts to the maximally exposed noninvolved worker or involved worker calculated for WIPP routine operations (Appendix F) and WIPP accident scenarios (Appendix G). Additional information is presented in Section H.8.

H.6 REPOSITORY DISPOSAL SYSTEM NUMERICAL MODEL

The following sections describe the methods, parameters, and data used to model the release and transport of radionuclides and heavy metals from the WIPP repository disposal system.

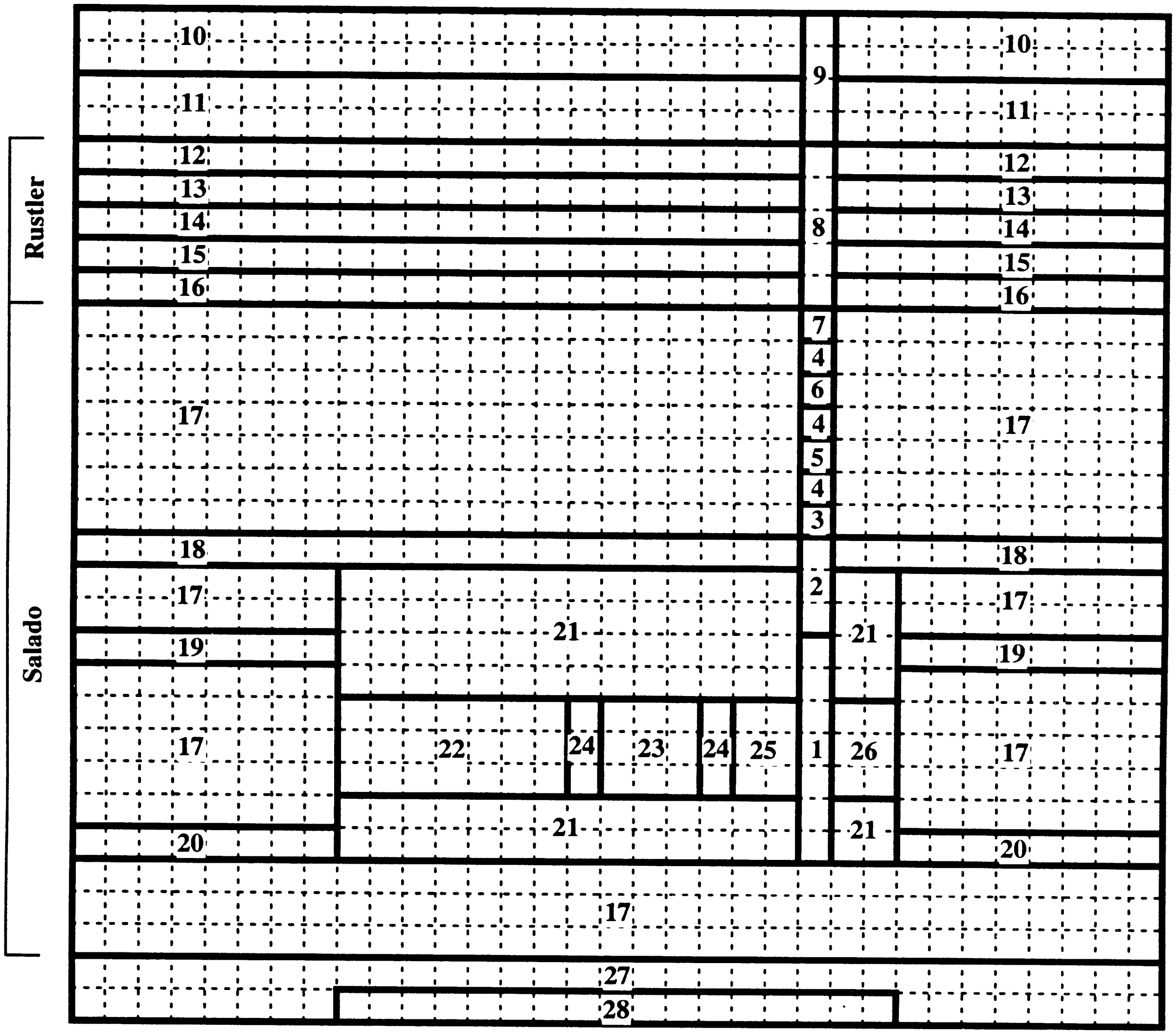
H.6.1 Model Geometry

A quasi-two-dimensional repository grid geometry implemented with the BRAGFLO code for the CCA (DOE 1996f), shown in Figure H-4, was used to represent the three-dimensional geometry of the disposal system for analysis of undisturbed conditions. This grid represents a vertical north-south cross-section through the disposal system and shows the distribution of grid blocks associated with important features of the disposal system and the major hydrostratigraphic units overlying the Salado Formation. Associations between grid blocks and material properties of major features and units are shown in Figure H-4 by pattern and number.

While the equidimensional grid system (Figure H-4) shows the relationship among material regions in the model and how connections are made within the finite-difference scheme of the BRAGFLO code, the grid greatly distorts the volumetric relationship between grid blocks. The grid system measures about 1,000 meters (3,280 feet) in vertical thickness, but the relatively thin waste panel area appears disproportionately thick. The modeled system extends approximately 23.3 kilometers (14.5 miles) to the north and south from the center of the grid system. The same BRAGFLO mesh, shown in Figure H-5, had a slightly different material property distribution to represent an intrusion borehole (material zones 29 to 31) and was used to simulate the disposal system for disturbed conditions.

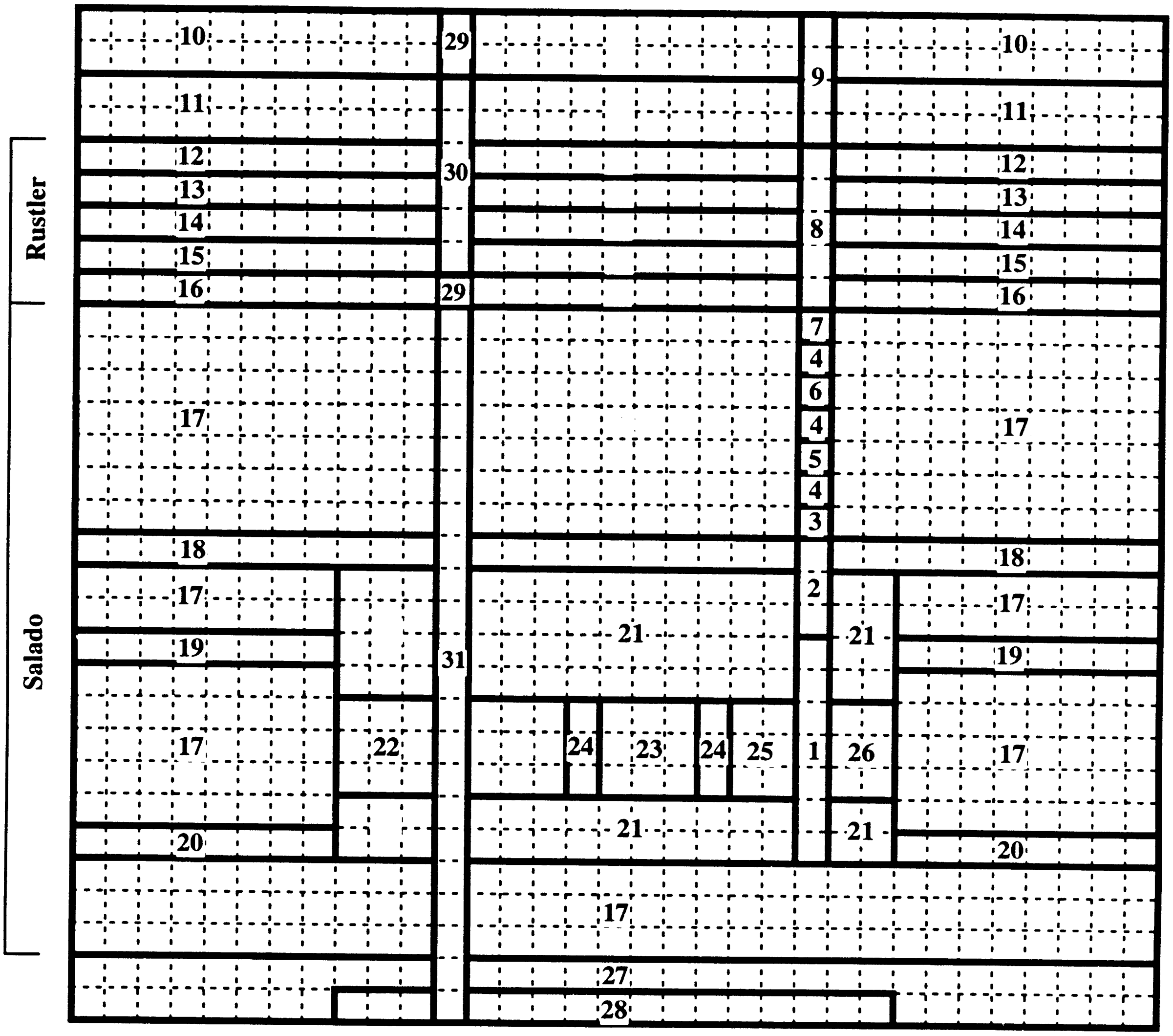
The top-down (plan) view of the model shown in Figure H-6 illustrates the dimension of the grid system in the orthogonal (out-of-plane) direction to the grid depicted in Figures H-4 and H-5. This view shows the approach adopted to simulate radially convergent or divergent flow. Effects of flow in the third (out-of-plane) dimension are approximated with a two-dimensional element configuration that simulates radially convergent or divergent flow in two directions, centered on the repository, in intact rocks, and laterally away from the repository. The effects of the grid assumptions on fluid-flow processes in the Salado Formation are discussed in Section H.6 (Appendix MASS.4, DOE 1996f).

To simulate long-term performance of Action Alternatives 1, 2, and 3, adjustments were made to the model geometry to reflect the changes in waste volumes defined for the action alternatives. Waste volumes are distributed between two sets of grid blocks, the panel and the rest of the repository grid spaces, as shown in Figure H-7. To accommodate the prescribed volumes, only the z dimension of grid spaces, representing the rest of the repository, were adjusted. A summary of these adjustments is provided in Table H-6. The x and y dimensions of these same grid spaces and the x, y, and z dimensions of the other grid spaces, representing the panel and all other elements of the disposal system, remained the same for all simulations.



1. Concrete Monolith
2. Lower Clay Component
3. Lower Salado Compacted Clay
4. Concrete
5. Crushed Salt (Salado Salt Column)
6. Upper Salado Compacted Clay
7. Asphalt
8. Rustler Clay Component
9. Earth Fill
10. Units Above the Dewey Lake
11. Dewey Lake Formation
12. Forty-Niner Member
13. Magenta Dolomite Member
14. Tamarisk Member
15. Culebra Dolomite Member
16. Unnamed Lower Member
17. Salado Halite
18. Marker Bed 138
19. Anhydrite Layer A & B
20. Marker Bed 139
21. Disturbed Rock Zone
22. Waste Panel
23. Rest of Repository
24. Panel Seals
25. Operation Region
26. Experimental Area
27. Castile Formation
28. Hypothetical Castile Brine Reservoir

Figure H-4
Vertical North-South Cross Section of the Model Grid Through the Disposal System for the Undisturbed Cases



1. Concrete Monolith
2. Lower Clay Component
3. Lower Salado Compacted Clay
4. Concrete
5. Crushed Salt (Salado Salt Column)
6. Upper Salado Compacted Clay
7. Asphalt
8. Rustler Clay Component
9. Earth Fill
10. Units Above the Dewey Lake
11. Dewey Lake Formation
12. Forty-Niner Member
13. Magenta Dolomite Member
14. Tamarisk Member
15. Culebra Dolomite Member
16. Unnamed Lower Member
17. Salado Halite
18. Marker Bed 138
19. Anhydrite Layer A & B
20. Marker Bed 139
21. Disturbed Rock Zone
22. Waste Panel
23. Rest of Repository
24. Panel Seals
25. Operation Region
26. Experimental Area
27. Castile Formation
28. Hypothetical Castile Brine Reservoir
29. Borehole Concrete Plug
30. Upper Unrestricted Borehole
31. Lower Unrestricted Borehole

Figure H-5
Vertical North-South Cross Section of the Model Grid Through the Disposal System for the Disturbed Cases

x	cell	Δz
1.50E+04	1	6.13E+04
5.00E+03	2	2.13E+04
1.00E+03	3	8.75E+03
7.50E+02	4	6.56E+03
5.00E+02	5	3.31E+03
1.00E+02	6	2.11E+03
4.00E+01	7	1.83E+03
1.00E+01	8	1.73E+03
4.38E+01	9	1.26E+02
1.00E+01	10	2.88E+01
2.00E+00	11	4.80E+00
4.00E-01	12	4.00E-01
2.00E+00	13	4.80E+00
1.00E+01	14	2.88E+01
4.38E+01	15	1.26E+02
4.00E+01	16	1.00E+01
1.00E+02	17	1.32E+02
4.95E+02	18	1.44E+02
1.00E+02	19	1.42E+02
8.00E+01	20	1.89E+01
2.83E+02	21	1.89E+01
5.00E+01	22	1.89E+01
1.00E+01	23	9.50E+00
6.00E+01	24	2.05E+01
5.00E+02	25	5.31E+01
1.00E+01	26	1.26E+03
4.00E+01	27	1.46E+03
1.00E+02	28	2.02E+03
5.00E+02	29	3.93E+03
7.50E+02	30	6.91E+03
1.00E+03	31	9.21E+03
5.00E+03	32	2.17E+04
1.50E+04	33	6.17E+04

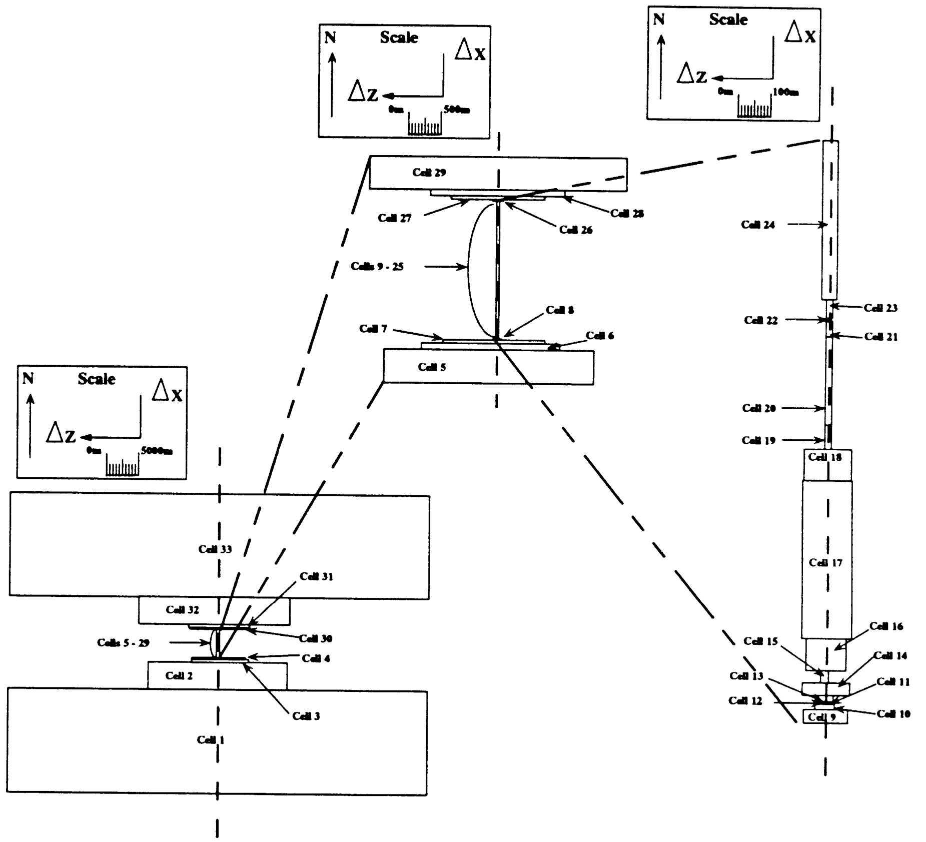


Figure H-6
Top-Down (Plan) View of the Model Through the Disposal System

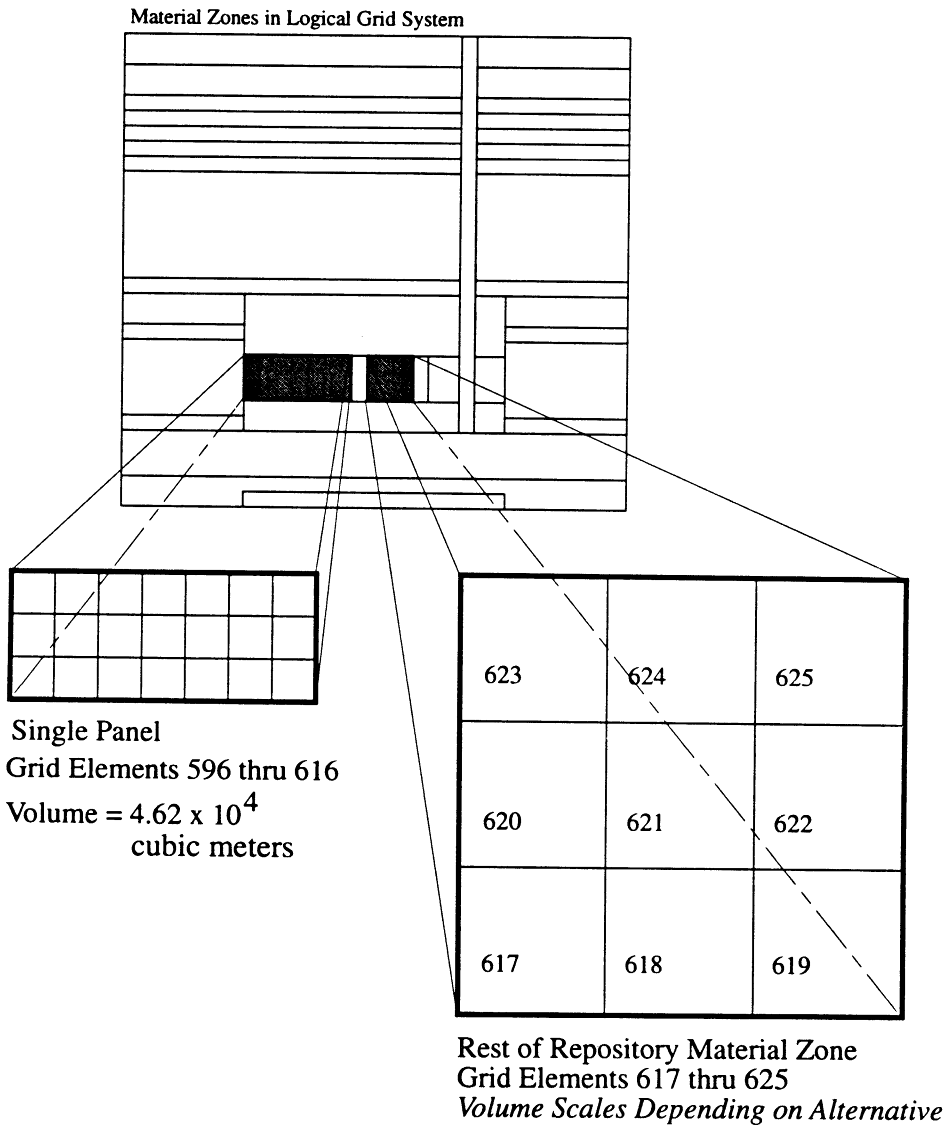


Figure H-7
Grid Spaces Representing the Rest of the Repository
Adjusted for Simulation of Action Alternatives 1, 2, and 3

Table H-6
Dimensional Lengths of Rest-of-Repository Material Zone Grid Elements for the Proposed Action; Action Alternatives 1, 2, 3; and Resulting Grid Element and Material Zone

Grid Element	x and y Element Length		Proposed Action		Action Alternative 1		Action Alternative 2		Action Alternative 3	
Index	x Depth (meters)	y Depth (meters)	z Depth (meters)	Volume (cubic meters)	z Depth (meters)	Volume (cubic meters)	z Depth (meters)	Volume (cubic meters)	z Depth (meters)	Volume (cubic meters)
617	100	1.3208	132.3	1.7E+4	1060	1.4E+5	1171	1.5E+5	1108	1.5E+5
618	495	1.3208	143.5	9.4E+4	1060	6.9E+5	1171	7.7E+5	1108	7.2E+5
619	100	1.3208	141.6	1.9E+4	1060	1.4E+5	1171	1.5E+5	1108	1.5E+5
620	100	1.3208	132.3	1.7E+4	1060	1.4E+5	1171	1.5E+5	1108	1.5E+5
621	495	1.3208	143.5	9.4E+4	1060	6.9E+5	1171	7.7E+5	1108	7.2E+5
622	100	1.3208	141.6	1.9E+4	1060	1.4E+5	1171	1.5E+5	1108	1.5E+5
623	100	1.3208	132.3	1.7E+4	1060	1.4E+5	1171	1.5E+5	1108	1.5E+5
624	495	1.3208	143.5	9.4E+4	1060	6.9E+5	1171	7.7E+5	1108	7.2E+5
625	100	1.3208	141.6	1.9E+4	1060	1.4E+5	1171	1.5E+5	1108	1.5E+5
Rest of Repository Volume (cubic meters)				3.9E+5	- - -	2.9E+6	- - -	3.2E+6	- - -	3.1E+6
Separately Modeled Panel Volume (cubic meters)				4.6E+4	- - -	4.6E+4	- - -	4.6E+4	- - -	4.6E+4
Total Repository Volume (cubic meters)				4.4E+5	- - -	3.0E+6	- - -	3.3E+6	- - -	3.1E+6

The characteristics of the hydrostratigraphic units depicted in Figures H-4 and H-5 at elevations near the repository horizon are based on the observed differences in permeability between anhydrite-rich interbeds and halite-rich intervals. Although not depicted in Figures H-4 and H-5, a 1-degree dip to the south in the BRAGFLO computational mesh has been incorporated to approximate the variable southerly dip observed in the Salado Formation.

H.6.2 Boundary and Initial Conditions

Initial conditions for a number of parameters used in the BRAGFLO model, such as liquid pressure, liquid saturation, ferrous metal, and biodegradable content in the waste disposal region, are assigned at the start of the long-term performance simulations for the modeled regions. Initial conditions for the repository and the Salado Formation used in SEIS-II analysis are consistent with the CCA (DOE 1996f) and include the following:

- No gradients for lateral flow exist in the Salado Formation.
- Assumed pore pressures in the Salado Formation are elevated above hydrostatic from the surface but below lithostatic.
- Assumed permeability and porosity in the Salado Formation are low.

- Excavation and waste emplacement result in partial drainage of the DRZ and subsequent evaporation of drained brine into mine air, which is then removed by air exchanged to the surface.

No-flow boundary conditions are assigned along all of the exterior boundaries of the computational mesh, except at the far-field boundaries of the Culebra and Magenta Members and the top of the grid (i.e., the surface of the ground). The far-field boundaries of the Culebra and Magenta Members are maintained at pressures of 0.822 and 0.917 megapascal, respectively, corresponding to the initial pressure conditions used in the Culebra and Magenta Members. The ground-surface grid blocks are maintained at 1 atmosphere, 0.10 megapascal; liquid saturations in these blocks are held constant at 20 percent.

Initial Conditions in the Salado Formation and DRZ

A five-year initial simulation was performed prior to the long-term simulation of the repository. The purpose of the five-year simulation was to estimate initial conditions for the near-repository, partially drained DRZ conditions. The initial liquid pressures in the Salado are based on marker bed (MB) 139 pressure of 12.5 megapascal at the shaft and adjusted throughout the Salado to account for a 1-degree dip, assuming hydrostatic equilibrium. The DRZ permeability is set to 1×10^{-17} square meters (1×10^{-16} square feet) for the startup simulation and then held constant at 1×10^{-15} square meters (1×10^{-14} square feet) for the rest of the simulation. The porosity of the DRZ is assumed as the value of impure halite. Porosity in all other lithologic units is initially 100 percent liquid-saturated during the initial simulation.

Initial Conditions in the Waste Disposal Region

In this analysis, the individual panels were assumed to remain open for 5 years to allow for waste emplacement. An initial period of 5 years is used to allow depressurization around the excavated regions to atmospheric pressure. After the initial 5-year period, the waste is placed at a liquid saturation of 0.015 and a pressure of 1 atmosphere. The remaining excavations outside the waste disposal area are assigned a gas saturation of 100 percent and an initial pressure of 1 atmosphere. Corrosion and/or biodegradation reactions that produce gas are modeled to begin at time zero, $T = 0$ years. For the purpose of this analysis, waste emplacement is assumed to occur instantaneously throughout the repository. The concentrations of ferrous metals and biodegradables in the waste regions are assigned initial parameter values of 158 and 92.5 kilograms per cubic meter (10 and 6 pounds per cubic foot), respectively.

Initial Conditions in the Shaft

After the initial 5-year period, shaft materials are assumed to be emplaced. The initial pressure in the shaft was set at 1 atmosphere, and the initial liquid saturation of all shaft materials was assumed to be at 100 percent. The exception is the asphalt region, which was set at 0 percent.

H.6.3 Repository and Panel Parameters

The repository is represented by regions 22 to 26 in Figures H-4 and H-5. These regions include an isolated waste disposal panel (22), panel closures (24), panels and access drifts in the rest of the waste disposal region (23), operations region (25), and an experimental region at the north end of the repository (26). The four shafts connecting the repository to the surface are represented by a single shaft in regions 1 through 9 in Figures H-4 and H-5. The lower shaft region (1) intersects the repository between the operations and experimental regions.

As mentioned in the CCA (DOE 1996f), the geometry depicted in the BRAGFLO model is a simplification of reality. The model geometry attempts to preserve the true excavated volume. Lateral dimensions have been defined to approximate the true excavated volume and retain important cross-sectional areas and distances between defined regions, as discussed below. These simplifications are conservative, with respect to fluid contact with waste, and are critical factors in determining the quantity of contaminants dissolved in the aqueous phase. The simplifications are also conservative because (1) all pillars have been removed from panels, resulting in homogeneous waste regions through which fluid can flow directly; (2) panel closures are included to retain the effects of their dimensions on fluid flow and are modeled with a higher permeability than they are expected to have; and (3) panels in the rest of the repository have neither pillars nor closures, resulting in a very large region of homogeneous waste that is assigned transmissive properties.

The panel closure has a cross-sectional area for fluid flow equal to the cross-sectional area of the drifts between panels. The panel closure between the rest of the repository and the operations region has a cross-sectional area for fluid flow equal to the cross-sectional area of the drifts between the north end of the waste disposal region and the operations region. Because two sets of closures exist between the waste disposal region and the shafts in the operations region, the panel closures between the rest of the repository and the operations region have a length equal to two sets of panel closures.

Fluid properties used for the BRAGFLO model are presented in Table H-7. Median and 75th percentile performance values used for repository and panel seal parameters in this analysis are summarized in Table H-8. The values used for the gas-generation model in this analysis are provided in Table H-9. The reader is referred to the CCA (DOE 1996f) for the relationship of these parameters in modeling salt creep, brine inflow, and gas generation in the repository system.

Table H-7
Fluid Property Parameter Values [a, b]

Parameters (units)	Values
Reference Temperature (Kelvin)	300.15
Liquid Density (kilograms per cubic meter) at:	
Atmospheric Pressure	1220.0
8 megapascal	1223.0
15 megapascal	1225.7
Liquid Viscosity (pascal second)	2.1E-3
Liquid Compressibility (1/pascal)	3.1E-10
Gas Density (kilograms per cubic meter) at:	
Atmospheric Pressure	0.0818
8 megapascal	6.17
15 megapascal	11.1
Gas Viscosity (pascal second)	8.93E-6

[a] See Appendix BRAGFLO (DOE 1996f) for equations of state.
[b] These values applied to all fluids in all material regions in BRAGFLO model.

Table H-8
Repository and Panel Seal Parameter Values [a]

Parameter (units)	Median Values	75th Percentile Values
Repository		
Permeability (square meter)	1.70E-13	
Effective Porosity (percent)	84.8	
Threshold Pressure P_t (pascal) [b]	0	
Residual Brine Saturation S_{br} (unitless) [c]	0.276	0.138
Residual Gas Saturation S_{gr} (unitless) [c]	0.075	0.0375
Pore Shape Distribution Parameter (unitless) [c]	2.89	2.165
Maximum Capillary Pressure (pascal)	1.0E+8	
Rock Compressibility (1/pascal)	0	
Panel Seals		
Permeability (square meter) – Panel Seals	1.0E-15	
Effective Porosity (percent) – Panel Seals	7.5	
Threshold Pressure P_t (pascal) [b] – Panel Seals	8.7E+4	
Residual Brine Saturation S_{br} (unitless) [c]	0.20	
Residual Gas Saturation S_{gr} (unitless) [c]	0.20	
Pore Shape Distribution Parameter (unitless) [c]	0.94	
Maximum Capillary Pressure (pascal)	1.0E+8	
Rock Compressibility (1/pascal)	2.64E-9	

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Threshold pressure (P_t) was determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

H.6.4 Shaft and Seal Parameters

The four shafts connecting the repository to the surface are represented with a single shaft in Figures H-4 and H-5. This single shaft has a cross-section and volume equal to the four real shafts it represents, and it is separated from the waste disposal region in the model by the true north-south distance from the waste to the nearest shaft (the Waste Shaft). On closure of the repository, the shafts will be sealed, as described in Chapter 3 of the CCA (DOE 1996f).

Values for shaft component materials and properties used in the simulation are given in Table H-10. From top to bottom, the system is represented in the simulation by the following materials:

- An earthen fill region above the Rustler Formation
- A clay region in the Rustler Formation (designated Rustler clay)
- Three concrete sections (upper, middle, lower) within the Salado, consolidated for modeling purposes into a single concrete region with the same total thickness
- A thick section of compacted crushed salt within the Salado
- An upper compacted clay region within the Salado (designated upper Salado compacted clay)

Table H-9
Parameter Values for the Average Stoichiometry Gas-Generation Model [a]

Parameter (units)	Median Values	75th Percentile Values
Iron Corrosion Rate under Inundated Conditions with MgO Backfill Added (thickness of steel corroded per second, meters/second)	7.937E-15	1.19025E-14
Iron Corrosion Rate under Humid Conditions (meters/second)	0	
Rate of Inundated Cellulosics Degradation (moles of carbon per kilogram of cellulose biodegraded per second)	4.915E-09	7.214E-09
Rate of Humid Cellulosics Biodegradation (moles of carbon per kilogram of cellulose biodegraded per second)	6.342E-10	9.513E-10
Scaling Factor for the Average Stoichiometric Factor Y in the Microbial Reaction (unitless)	0.5	0.75
Stoichiometric Factor for Iron Corrosion (moles of gas generated per mole of iron consumed due to corrosion reaction, moles/moles)	1.0	
Fraction of Plastics and Rubbers that are Biodegradable (unitless)	1	
Average Density of Iron-Based Materials in CH-TRU Waste (kilogram/cubic meter)	170	
Average Density of Iron-Based Materials in RH-TRU Waste (kilogram/cubic meter)	100	
Average Density of Plastics in CH-TRU Waste (kilogram/cubic meter)	34	
Average Density of Plastics in RH-TRU Waste (kilogram/cubic meter)	15	
Average Density of Rubber in CH-TRU Waste (kilogram/cubic meter)	10	
Average Density of Rubber in RH-TRU Waste (kilogram/cubic meter)	3.3	
Average Density of Cellulose in CH-TRU Waste (kilogram/cubic meter)	54	
Average Density of Cellulose in RH-TRU Waste (kilogram/cubic meter)	17	
Average Density of Iron Containers, CH-TRU Waste (kilogram/cubic meter)	280	
Average Density of Iron Containers, RH-TRU Waste (kilograms/cubic meter)	2,650	
Average Density of Plastic in CH-TRU Waste Containers (kilograms/cubic meter)	26	
Average Density of Plastic in RH-TRU Waste Containers (kilograms/cubic meter)	3.1	
Average Density of Iron in CH-TRU Waste Containers (kilograms/cubic meter)	139	
Average Density of Iron in RH-TRU Waste Containers (kilograms/cubic meter)	2,591	
Total Volume of RH-TRU Waste (cubic meters)	7,080	
Total Volume of CH-TRU Waste (cubic meters)	168,500	
Index for Computing Wicking (unitless)	0.50	

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.

- A lower compacted clay region within the Salado (designated lower Salado compacted clay)
- A basal clay component below MB 138 (designated bottom clay)
- A lower concrete section at the repository horizon (designated shaft station concrete monolith)
- An asphalt region at the top of the Salado Formation

Additional documentation of the shaft material parameters and their use in the BRAGFLO analysis are described in the CCA (Chapter 6 of DOE 1996f).

Table H-10
Shaft Materials Parameter Values [a]

Parameters (units)	Median Values	75th Percentile Values
Clay Shaft Materials		
Permeability (square meter), Rustler Clay	5.000E-19	
Permeability (square meter), Upper Salado Compacted Clay (0 to 1 year)	8.598E-17	
Permeability (square meter), Upper Salado Compacted Clay (1 to 3 years)	5.629E-17	
Permeability (square meter), Upper Salado Compacted Clay (3 to 5 years)	3.381E-17	
Permeability (square meter), Upper Salado Compacted Clay (5 to 100 years)	1.297E-17	
Permeability (square meter), Upper Salado Compacted Clay (After 100 years)	5.000E-19	
Permeability (square meter), Lower Salado Compacted Clay (0 to 1 year)	1.048E-16	
Permeability (square meter), Lower Salado Compacted Clay (1 to 3 years)	1.944E-17	
Permeability (square meter), Lower Salado Compacted Clay (3 to 5 years)	7.317E-19	
Permeability (square meter), Lower Salado Compacted Clay (After 5 years)	5.000E-19	
Permeability (square meter), Bottom Clay	5.000E-19	
Thickness (meter), Rustler Clay	94.3	
Thickness (meter), Upper Salado Compacted Clay	104.85	
Thickness (meter), Lower Salado Compacted Clay	23.9	
Thickness (meter), Bottom Clay	0.18	
Effective Porosity (percent), Rustler, Upper, Lower, and Bottom Clays	24	
Residual Brine Saturation S_{br} (unitless) [b], Rustler, Upper, Lower, and Bottom Clays	0.20	
Residual Gas Saturation S_{gr} (unitless) [b], Rustler, Upper, Lower, and Bottom Clays	0.20	
Threshold Pressure P_t (pascal) [c], Rustler, Upper, Lower, and Bottom Clays	0	
Pore Shape Distribution Parameter (unitless) [b], Rustler, Upper, Lower, and Bottom	0.94	
Maximum Capillary Pressure (pascal), Rustler, Upper, Lower, and Bottom Clays	1.0E+8	
Rock Compressibility (1/pascal), Rustler Clay	1.96E-9	
Rock Compressibility (1/pascal), Upper Salado Compacted Clay	1.81E-9	
Rock Compressibility (1/pascal), Lower Salado Compacted Clay	1.59E-9	
Rock Compressibility (1/pascal), Bottom Clay	1.59E-9	
Salt Shaft Materials		
Permeability (square meter), Salt (0 to 1 year)	1.748E-15	
Permeability (square meter), Salt (1 to 3 years)	1.662E-15	
Permeability (square meter), Salt (3 to 5 years)	1.649E-15	
Permeability (square meter), Salt (5 to 100 years)	1.486E-18	
Permeability (square meter), Salt (100 to 200 years)	6.108E-20	
Permeability (square meter), Salt (After 200 years)	5.349E-21	
Thickness (meter), Salt	171.37	
Effective Porosity (percent), Salt	5.0	
Residual Brine Saturation S_{br} (unitless) [b], Salt	0.20	
Threshold Pressure P_t (pascal) [c], Salt	0.0	
Pore Shape Distribution Parameter (unitless) [b], Salt	0.94	0.50
Maximum Capillary Pressure (pascal), Salt	1.0E+8	
Rock Compressibility (1/pascal), Salt	1.60E-9	
Residual Gas Saturation S_{gr} (unitless) [b], Salt	0.20	

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).
[c] Threshold pressure (P_t) was determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.

Table H-10
Shaft Materials Parameter Values — Continued [a]

Parameters (units)	Median Values	75th Percentile Values
Concrete Shaft Materials		
Permeability (square meter), Concrete (0 to 400 year)	1.780E-19	
Permeability (square meter), Concrete (After 400 year)	1.0E-14	
Thickness (meter), Upper, Middle, and Lower Concrete (each)	15.24	
Thickness (meter), Shaft Station Monolith Concrete	9.08	
Effective Porosity (percent), Concrete	5.0	
Residual Brine Pressure S_{br} (unitless) [b], Concrete	0.20	0.10
Residual Gas Pressure S_{gr} (unitless) [b], Concrete	0.20	0.10
Threshold Pressure P_t (pascal) [c], Concrete	0	
Pore Shape Distribution Parameter (unitless) [b], Concrete	0.94	0.50
Maximum Capillary Pressure (pascal), Concrete	1.0E+8	
Rock Compressibility (1/pascal), Concrete	1.20E-9	
Asphalt Shaft Materials		
Permeability (meters squared), Asphalt	1.0E-20	
Thickness (meter), Asphalt	23.9	
Effective Porosity (percent) Asphalt	1	
Residual Brine Saturation S_{br} (unitless) [b], Asphalt	0.20	0.10
Residual Gas Saturation S_{gr} (unitless) [b], Asphalt	0.20	0.10
Threshold Pressure P_t (pascal) [c], Asphalt	0	
Pore Shape Distribution Parameter (unitless) [b], Asphalt	0.94	0.50
Maximum Capillary Pressure (pascal) , Asphalt	1.0E+8	
Rock Compressibility (1/pascal) , Asphalt	2.97E-8	
Earth Fill Shaft Materials		
Permeability (square meter), Earth	1.0E-14	2.62E-14
Thickness (meter), Earth	165.06	
Effective Porosity (percent), Earth	32	
Residual Brine Saturation S_{br} (unitless) [b], Earth	0.20	0.10
Residual Gas Saturation S_{gr} (unitless) [b], Earth	0.20	0.10
Threshold Pressure P_t (pascal) [c], Earth	0	
Pore Shape Distribution Parameter (unitless) [b], Earth	0.94	0.50
Maximum Capillary Pressure (pascal), Earth	1.0E+8	
Rock Compressibility (1/pascal), Earth	3.1E-8	

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).
[c] Threshold pressure (P_t) was determined from the relationship: $P_t = PCT\_A\ k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.

H.6.5 Salado Formation

The Salado Formation is modeled as a porous medium composed of several layered rock types, through which flow takes place according to Darcy's law (see description in Chapter 6 of the CCA [DOE 1996f]). Two rock types, impure halite and anhydrite, are used to represent the intact Salado. The DRZ near the repository is assumed to have increased permeability compared to intact rock and offers little resistance to flow between anhydrite interbeds and the repository. Conceptually, properties for Salado rock are assumed to be constant, based on observations of compositional and structural regularity in layers exposed by the repository. The inference from this assumption is that there is little variation in large-scale averages of rock or flow properties across the disposal system.

H.6.5.1 Impure Halite

In this analysis, a single, porous medium with spatially constant rock and hydrologic properties (see region 17 in Figures H-4 and H-5) is used to represent intact, halite-rich layers in the Salado. Minor interbeds are contained within those layers that are not explicitly represented. Table H-11 shows median and 75th percentile parameter values used in modeling the Salado impure halite. Additional information on the use of these parameter values in BRAGFLO simulations is contained in Appendix PAR of the CCA (DOE 1996f).

Table H-11
Parameter Values for Salado Formation Halite [a]

Parameter (units)	Median Values	75th Percentile Values
Permeability (square meter)	3.16E-23	1.778E-22
Effective Porosity (percent)	1.0	0.55
Specific Storage (1/meter)	1.0E-6	
Threshold Pressure P_t (pascal) [b]	3.4E+7	1.9E+7
Residual Brine Saturation (unitless) [c]	0.30	0.20
Residual Gas Saturation (unitless) [c]	0.20	0.10
Pore Shape Distribution Parameter (unitless) [c]	0.70	0.50
Maximum Capillary Pressure (pascal)	1.0E+8	
Pore Compressibility (1/pascal)	9.75E-9	2.63E-08

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR of DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Threshold pressure (P_t) determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

H.6.5.2 Anhydrite Interbeds

Three distinct anhydrite interbeds are modeled in this BRAGFLO simulation, representing MB 138 (region 18), anhydrite layers a and b (region 19), and MB 139 (region 20), all shown in Figures H-4 and H-5. The three interbeds have the same set of model parameters, which are initially held constant. During a simulation, porosity and permeability can vary spatially with simulated interbed fracturing. The three interbeds are included because they exist in the disturbed region around the repository, within which fluid is expected to flow with relative ease compared to the surrounding formation. MB 139 and anhydrite layers a and b are present within the DRZ that forms around excavations; MB 138 may be above the DRZ but is below the long-term seal components that will be constructed in the shafts. MB 138 is included because of its uncertainty in the long-term isolation from the repository. Median and 75th percentile values used for parameters associated with the interbeds are shown in Table H-12. Table H-13 lists values of parameters used in the model for interbed dilation and fracture. Documentation on the selection and use of these parameters in BRAGFLO simulations are found in the CCA (Appendix PAR in DOE 1996f).

Table H-12
Parameter Values for the Salado Formation
Anhydrite Interbeds A and B and Marker Beds 138 and 139 [a]

Parameter (units)	Median Values	75th Percentile Values
Permeability (square meter)	1.288E-19	3.162E-19
Effective Porosity (percent)	1.1	0.895
Threshold Pressure P_t (pascal) [b]	9.7E+5	7.1E+5
Residual Brine Saturation S_{br} (unitless) [c]	0.08363	0.04986
Residual Gas Saturation S_{gr} (unitless) [c]	0.07711	0.03390
Pore Shape Distribution Parameter (unitless) [c]	0.6436	0.5704
Maximum Capillary Pressure (pascal)	1.0E+8	
Pore Compressibility (1/pascal)	7.512E-9	1.444E-8

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Threshold pressure (P_t) was determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

Table H-13
Parameter Values for the Salado Formation
Anhydrite Interbeds A and B and Marker Beds 138 and 139 Fracture [a]

Parameter (units)	Values
Fracture Initiation Pressure (pascal)	1.4E+6
Increment to Give Full Fracture Porosity (percent)	1.0
Maximum Permeability (square meter)	1.0E-9
Increment of Lithostatic Pressure to Obtain Maximum Fracture Pressure (pascal)	2.5E+6
Brine Far-Field Pore Pressure (pascal)	1.3E+7

[a] Values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f).

H.6.5.3 Disturbed Rock Zone

Near the repository at the Salado Formation, the permeability and porosity of the DRZ salt are expected to increase over that of intact salt. The increases in the permeability and porosity of salt in interbeds are not expected to be completely reversible with creep closure of the disposal rooms. The increase in DRZ permeability affects the ability of fluid to flow from interbeds to the waste disposal region.

The increase in DRZ porosity provides a volume in which some fluid could be retained so that it does not contact waste. DRZ pore volume can also slow radionuclide and hazardous constituent migration.

In this analysis, the permeability of a region around the repository is increased relative to intact Salado rock for the duration of the simulation and the threshold pressure is set to zero. The porosity of this region is left equal to the porosity of Salado halite to prevent reduced fluid retention in the DRZ. The DRZ extends above and below the repository from the base of MB 138 to MB 139. Defining the DRZ in this manner creates a permanent, highly permeable region that does not impede flow between the repository and interbeds. Median values of the parameters used in the representation of the DRZ are summarized in Table H-14.

Table H-14
Parameter Values for the Disturbed Rock Zone [a]

Parameter (units)	Values
Permeability (square meter)	1.0E-15
Effective Porosity (percent)	1.29
Threshold Pressure P_t (pascal) [b]	0
Residual Brine Saturation S_{br} (unitless) [c]	0
Residual Gas Saturation S_{gr} (unitless) [c]	0
Pore Shape Distribution Parameter (unitless) [c]	0.70
Maximum Capillary Pressure (pascal)	1.0E+8
Pore Compressibility (1/pascal)	5.744E-8

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). All parameters in this table were held constant.
[b] Threshold pressure (P_t) determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

H.6.5.4 Units Above the Salado Formation

The BRAGFLO simulations used in this analysis consider the Unnamed Lower, Culebra, Tamarisk, Magenta, and Forty-Niner Members of the Rustler Formation, the Dewey Lake Formation, the Gatuna Formation, and the Santa Rosa Formation. For modeling purposes, the Gatuna and Santa Rosa Formations were combined as units above the Dewey Lake Formation. BRAGFLO separates and calculates flow in these units to establish the pressure gradient in the disposal system. The other three Rustler Formation members are modeled as effectively impermeable. These units are represented in the BRAGFLO element mesh by regions 10 through 16 (shown in Figures H-4 and H-5).

In this analysis, the water table was set equal to 59 meters (194 feet) below the ground surface at an elevation of 980 meters (3,215 feet) within the Dewey Lake Formation. For regions above the water table, the initial liquid saturation of 20 percent (the residual liquid saturation of the Dewey Lake, the Gatuna, and Santa Rosa Formations) was used. For regions above the water table, the initial liquid pressure is assumed to be 1 atmosphere. For the portion of the Dewey Lake Formation below the water table and the Rustler Formation regions (excluding the Culebra and Magenta Members, which are specified at 0.82 and 0.91 megapascal, respectively), a hydrostatic gradient is assumed for specifying the initial pressure conditions. For the time period -5 to 0 years, all regions above the Salado are treated as impermeable. Conceptually, this corresponds to the time periods when liners are emplaced in the shafts. Parameter values for these units are presented in Tables H-15, H-16, H-17, H-18, and H-19. Documentation of the selection and use of these parameters is described in more detail in Appendix PAR of the CCA (DOE 1996f).

Table H-15
Parameter Values for the Culebra Member of the Rustler Formation [a]

Parameter (units)	Values
Permeability (square meter)	2.1E-14
Effective Porosity (percent)	15.1
Threshold Pressure P_t (pascal) [b]	1.50E+4
Residual Brine Saturation S_{br} (unitless) [c]	0.08363
Residual Gas Saturation S_{gr} (unitless) [c]	0.07711
Pore Shape Distribution Parameter (unitless) [c]	0.6436
Maximum Capillary Pressure (pascal)	1.0E+8
Pore Compressibility (1/pascal)	6.6225E-10

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). All parameters in this table were held constant.
[b] Threshold pressure (P_t) determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

Table H-16
Parameter Values for the Magenta Member of the Rustler Formation [a]

Parameter (units)	Median Values	75th Percentile Values
Permeability (square meter)	6.310E-16	
Effective Porosity (percent)	13.8	8.23
Threshold Pressure P_t (pascal) [b]	5.06E+4	
Residual Brine Saturation S_{br} (unitless) [c]	0.08363	
Residual Gas Saturation S_{gr} (unitless) [c]	0.07711	
Pore Shape Distribution Parameter (unitless) [c]	0.6436	
Maximum Capillary Pressure (pascal)	1.0E+8	
Pore Compressibility (1/pascal)	1.916E-9	

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Threshold pressure (P_t) determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

Table H-17
Parameter Values for the Forty-Niner, Tamarisk, and Unnamed Lower Members of the Rustler Formation [a]

Parameter (units)	Median Values	75th Percentile Values
Permeability (square meter)	1.0E-35	
Effective Porosity (percent) - Forty Niner Member	8.2	1.26
Effective Porosity (percent) - Tamarisk Member	6.4	4.3
Effective Porosity (percent) - Unnamed Lower Member	18.1	9.6
Threshold Pressure P_t (pascal) [b]	0	
Residual Brine Saturation S_{br} (unitless) [c]	0.20	
Residual Gas Saturation S_{gr} (unitless) [c]	0.20	
Pore Shape Distribution Parameter (unitless) [c]	0.70	
Maximum Capillary Pressure (pascal)	1.0E+8	
Pore Compressibility (1/pascal)	0	

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Threshold pressure (P_t) was determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

Table H-18
Parameter Values for the Dewey Lake Formation [a]

Parameter (units)	Median Values	75th Percentile Values
Permeability (square meter)	5.0E-17	
Effective Porosity (percent)	14.3	9.1
Threshold Pressure P_t (pascal) [b]	0	
Residual Brine Saturation S_{br} (unitless) [c]	0.08363	
Residual Gas Saturation S_{gr} (unitless) [c]	0.07711	
Pore Shape Distribution Parameter (unitless) [c]	0.6436	
Maximum Capillary Pressure (pascal)	1.0E+8	
Pore Compressibility (1/pascal)	6.993E-8	
Thickness (meter)	149.3	
Initial Pressure (pascal)	Hydrostatic; water table at 980 meters, 43.3 meters below top of formation	
Initial Pressure (atm) 20 percent Liquid Saturation, Above Water Table	1.0	

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Threshold pressure (P_t) was determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

Table H-19
Parameter Values for the Units Above the Dewey Lake Formation [a]

Parameter (units)	Values
Permeability (square meter)	1.0E-10
Effective Porosity (percent)	17.5
Threshold Pressure P_t (pascal) [b]	0
Residual Brine Saturation S_{br} (unitless) [c]	0.08363
Residual Gas Saturation S_{gr} (unitless) [c]	0.07711
Pore Shape Distribution Parameter (unitless) [c]	0.06436
Maximum Capillary Pressure (pascal)	1.0E+8
Pore Compressibility (1/pascal)	5.714E-8
Thickness (meter)	15.76
Initial Pressure (atm) 20 percent Liquid Saturation, Above Water Table	1.0

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). All parameters in this table were held constant.
[b] Threshold pressure (P_t) was determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

H.6.6 The Castile Formation

The BRAGFLO simulations used in both the undisturbed and disturbed analyses considered the Castile Formation to incorporate the effects of brine pocket pressure below the Salado Formation. The Castile Formation is represented in the BRAGFLO element mesh by region 27 (shown in Figures H-4 and H-5).

The brine pocket pressure represented in the BRAGFLO element mesh by region 28 (see Figures H-4 and H-5) was represented in median parameter cases by an initial pressure of 12.5 megapascal and in 75th percentile parameter cases by 14.4 megapascal. For undisturbed cases, this initial pressure has little impact on the predicted pressure field. For the disturbed cases, however, this unit is penetrated by an intrusion borehole, allowing its hydraulic impact to be transmitted to other units within the modeled domain, most notably the repository and the Culebra Dolomite Member.

Median and 75th percentile values of important parameters for the Castile Formation and Brine Reservoir are presented in Tables H-20 and H-21. Documentation of the selection and use of these parameters is described in more detail in Chapter 6 of the CCA (DOE 1996f).

H.6.7 Intrusion Borehole Parameters

For the disturbed cases, BRAGFLO simulations considered an intrusion borehole that penetrates the entire sequence of units in the modeled domain. The borehole is represented in the BRAGFLO element mesh by region 29-31 (see Figure H-5). The intrusion borehole was assumed to be an exploratory borehole that penetrates the WIPP repository and extends through a pressurized brine reservoir in the Castile Formation, creating a potential pathway between the brine reservoirs and

Table H-20
Parameter Values for the Castile Formation [a]

Parameter (units)	Values
Permeability (square meter)	1.0E-35
Effective Porosity (percent)	0.5
Threshold Pressure P_t (pascal) [b]	0
Residual Brine Saturation S_{br} (unitless) [c]	0
Residual Gas Saturation S_{gr} (unitless) [c]	0
Pore Shape Distribution Parameter (unitless) [c]	0.7
Maximum Capillary Pressure (pascal)	1.0E+8
Pore Compressibility (1/pascal)	0

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (Appendix PAR in DOE 1996f). All parameters in this table were held constant.
[b] Threshold pressure (P_t) was determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

Table H-21
Parameter Values for the Brine Reservoir [a]

Parameter (units)	Median Values	75th Percentile Values
Permeability (square meter)	1.585E-12	1.738E-12
Effective Porosity (percent)	0.696	1.40
Threshold Pressure P_t (pascal) [b]	6.776E+3	6.564E+3
Residual Brine Saturation S_{br} (unitless) [c]	0.20	
Residual Gas Saturation S_{gr} (unitless) [c]	0.20	
Pore Shape Distribution Parameter (unitless) [c]	0.70	
Maximum Capillary Pressure (pascal)	1.0E+8	
Pore Compressibility (1/pascal)	1.149E-8	3.710E-9
Initial Brine Pressure (pascal)	1.27E+7	1.44E+7
Volume (cubic meter)	4.0E+6	

[a] Median and 75th percentile values were based on data and parameter distributions contained in CCA (DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.
[b] Threshold pressure (P_t) was determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

the repository. This pathway could inundate the repository and flush water material from the repository to overlying water-bearing units in the Rustler Formation and other near-surface units.

Under this scenario, it was assumed that the borehole would be drilled and plugged at abandonment using standard regulatory requirements and practices. The parameters used to represent this scenario were based on intrusion borehole permeability studies described in Thompson et al. (1996). These studies evaluated two parameters: the long-term estimates of permeabilities in exploratory borehole materials, and the dimensions from the review of current regulatory requirements and drilling and borehole plugging practices. Using models and data for steel corrosion and concrete alteration, estimates of long-term changes in borehole permeabilities for three different borehole plugging configurations were made. The three configurations considered were:

- Borehole with a continuous plug - For this scenario, a single continuous plug is emplaced through the entire sequence of evaporites in the Castile and Salado Formations. In current practices, this approach to plugging a borehole represents the maximum standards for plugging in the Delaware Basin.

- Borehole with two plugs - For this scenario, a cement plug is placed in the Bell Canyon below the depth of the Castile Formation brine pockets, and a second plug is placed in the Rustler Formation between the Culebra Dolomite and the repository. This configuration represents the minimum standards for plugging in the Delaware Basin.

- Borehole with multiple plugs - For this configuration, plugs are emplaced as in the two-plug scenario. An additional plug is emplaced in the Salado Formation between the repository and the Castile formation brine pockets. This multiplugging scheme represents a more typical approach to borehole plugging in the Delaware Basin.

Over the 10,000-year period of interest, properties of boreholes and plugging materials will change and degrade by a variety of degradation mechanisms including (1) iron corrosion of well steel casing, (2) concrete degradation, and (3) salt creep. According to Thompson et al. (1996), borehole plugs would initially be expected to have a permeability of 5×10^{-17} square meters (5×10^{-16} square feet). In the continuous plug scenario, the borehole permeability is expected to remain largely undiminished. For the other plugging configurations, the properties of various sections of the borehole would be expected to change with time. Casing in upper parts of the borehole above the Rustler Formation are estimated to degrade completely within a 200-year period. This degradation would cause the plug in the Rustler to fail over time, and the corroded casing and plug would likely spill into the borehole, filling it with material that would likely have a permeability similar to silty sand (1×10^{-11} to 1×10^{-14} square meters [1×10^{-10} to 1×10^{-13} square feet]). Over time, salt creep would compress this material into the borehole, creating a permeability about one order of magnitude less. For deeper plugs in the borehole, the casing would not corrode as extensively, and the plugs at that depth would not fail for an estimated time of 500 years.

For the purpose of this analysis, the two-plug configuration was selected for the disturbed cases since it is the minimal plugging practice in the Delaware Basin. Parameter values used to represent the intrusion borehole for all disturbed cases are presented in Table H-22. Selected parameters used in the CUTTINGS_S code to model direct releases associated with the hypothetical exploratory drilling operation to the ground surface are provided in Table H-23.

Table H-22
Exploratory Borehole Parameter Values Used After Intrusion

Parameter (units)	Values [a]
Borehole Plug Permeability (square meter) (0 to 200 years after intrusion)	5.0E-17
Borehole Permeability (square meter) (0 to 200 years after intrusion)	1.0E-9
Borehole Permeability (square meter) (after 200 years after intrusion)	3.162E-13
Permeability (square meter), Borehole Permeability Below WIPP (square meter) (After 1200 years after intrusion)	3.162E-14
Effective Porosity (percent)	32
Threshold Pressure P_t (pascal) [b]	0
Residual Brine Saturation S_{br} (unitless) [c]	0
Residual Gas Saturation S_{gr} (unitless) [c]	0
Pore Shape Distribution Parameter (unitless) [c], Borehole Plugs (0 to 100 years after intrusion), Borehole (After 200 years after intrusion), and Borehole below WIPP (After 1200 years after intrusion)	0.70
Pore Shape Distribution Parameter (unitless) [c], Borehole (0 to 200 years after intrusion)	0.94
Maximum Capillary Pressure (pascal)	1.0E+8
Rock Compressibility (1/pascal), Concrete Borehole Plug	1.20E-9
Rock Compressibility (1/pascal), Borehole	0

[a] Values were based on data contained in the CCA (DOE 1996f) and Thompson et al (1996).
[b] Threshold pressure (P_t) determined from the relationship: P_t = PCT_A $k^{(PCT\_EXP)}$, where PCT_A and PCT_EXP are constants and k is the permeability.
[c] Two-phase flow: Brooks-Corey model used (Appendix BRAGFLO in DOE 1996f).

Table H-23
Select Parameter Values Used in CUTTINGS_S Code [a]

Parameter (units)	Median Values	75th Percentile Values
Intrusion Time (years)	400	300
Brine Density (kilograms per cubic meter)	1210	
Viscosity of H_2 gas at 27 degrees Celsius and 0.101325 megapascal (pascal second)	9.17E-3	
Yield Stress Point (pascal)	4.40	
Logarithm of Waste Particle Diameter (meters)	2.8E-3	3.4E-4
Drill string angular velocity (radians per second)	7.80	
Effective Shear Strength for Erosion (pascal)	5.0300	2.5375
Exploratory Borehole Diameter (meters)	0.3115	

[a] Median and 75th percentile values were based on data and parameter distributions contained in CCA (DOE 1996f). If a parameter value was constant, none is shown for the 75th percentile value.

Estimates of median and 75th percentile concentrations of individual nuclides were developed for use in the drilling intrusion into CH-TRU wastes scenarios. Values presented are representative of uncompacted waste from 457 waste streams. The methodology used to develop these estimates is described in Section A.4.8. Only a limited amount of information was available for RH-TRU waste streams so concentrations used for drilling intrusions into RH-TRU wastes used average results with no differentiation between median and 75th percentile values.

H.6.8 Solubility-Controlled Release Model Parameters

Solubility-controlled release models available in the NUTS code were used in the source-term release models for the Proposed Action and Action Alternative 1, 2, and 3 inventories for all undisturbed and disturbed cases. Data used in the analyses were selected from the probabilistic distributions of solubility of key radionuclides, which are available in SNL model databases for performance assessment calculations supporting the *Final No-Migration Variance Petition* and the CCA. Median and 75th percentile values of solubility for selected radionuclides simulated in transport modeling using the NUTS code are presented in Table H-24. The solubilities for metals included in this analysis were set to a value of one. For some radionuclides, the CCA has different solubilities for the Salado and Castile formations. The higher of these two solubilities were assigned to both formations for the SEIS-II.

Table H-24
Log Solubility Values for Elements [a, b]

Element	Median Values	75th Percentile Values
Actinium (Ac)	-4.256	-4.023
Americium (Am) [c]	-5.548	-5.315
Beryllium (Be)	0	
Bismuth (Bi)	0	
Cadmium (Cd)	0	
Cesium (Cs)	0	
Curium (Cu)	-4.256	-4.023
Mercury (Hg)	0	
Neptunium (Np)	-4.256	-4.023
Protactinium (Pa)	-4.256	-4.023
Lead (Pb)	0	
Polonium (Po)	0	
Plutonium (Pu)	-4.866	-4.633
Radium (Ra)	0	
Radon (Rn)	0	
Strontium (Sr)	0	
Thorium (Th)	-4.627	-4.394
Uranium (U)	-4.901	-4.668
Yttrium (Y)	0	

[a] Median and 75th percentile values were based on the data and parameter distributions contained in the CCA (DOE 1996a). If a parameter value was constant, none is shown for the 75th percentile value. Elements without project-specific solubility data –- Cs, Pb, Ra, Sr -- were assigned a value of zero.
[b] Values include the effects of magnesium oxide (MgO) backfill.
[c] Except for Action Alternative 2, in which case the median value for Americium was -5.641 and the 75th value was -5.408.

H.7 CULEBRA DOLOMITE MODEL

In the SEIS-II analyses, contaminants were found to be released to the Culebra Dolomite under 75th percentile parameter value cases of drilling through the repository and into a pressurized brine reservoir. Transport of contaminants was analyzed using the flow and transport model developed for the Culebra Dolomite to support the CCA performance assessment calculations (DOE 1996f). Basic information on this model and its application in the SEIS-II analysis is briefly described in the following sections.

H.7.1 Model Geometry, Boundaries, and Assumed Flow Conditions

The flow and transport model of the Culebra Dolomite used in the SEIS-II analysis was based on the implementation of the SECOFL2D and SECOTP2D flow and transport codes for regional and local hydrogeologic conditions of the WIPP site as described in Chapter 6 of the CCA (DOE 1996f). This model consists of a regional-scale flow model and local-scale transport submodel. The regional-scale model is based on a finite difference grid which covers about 660 square kilometers (266 square miles). The grid is oriented with its long dimension in a southwest to northeast direction and the WIPP site approximately at its center. The grid boundaries extend out to distant topographic and hydrologic features that control the distribution of hydraulic head in the WIPP region. The overall grid dimensions are approximately 22 kilometers (14 miles) in a northwest to southeast direction by 30 kilometers (19 miles) in a southwest to northeast direction.

Evaluation of radionuclide and heavy metal transport was evaluated in the more refined local-scale submodel that covers an area of 49 square kilometers (16 square miles) and is about 7 kilometers (4 miles) on a side. The local-scale model grid is approximately oriented in a north-south direction and is designed to examine the transport of key radionuclides from the repository area in the principal direction of groundwater movement south of the Land Withdrawal Area boundary. Hydraulic boundary conditions used in the local submodel were obtained from the interpolation of predicted hydraulic head distributions defined in the regional flow model. Additional information on the regional-scale flow and local-scale transport model grids is provided in Chapter 6 of the CCA (DOE 1996f).

Different thicknesses of the Culebra Dolomite were assumed in the various flow and transport models used in the SEIS-II analysis. In the repository and disposal system model using BRAGFLO (Section H.6), a thickness of 7.7 meters (25.3 feet) was used, which is representative of the Culebra thickness over the waste disposal panels. For calibrating transmissivity fields (see Appendix TFIELD, Section 4.4.1, DOE 1996f) and calculating flow conditions in the regional flow model of the Culebra, a thickness of 7.75 meters (25.4 feet) was assumed, consistent with the average thickness of the Culebra over the regional model area. For transport calculations using the local submodel of the Culebra, a thickness of 4 meters (13 feet) was used, consistent with observations of the Culebra thickness where transport is active. The details behind these assumptions are discussed at length in Section 6.4.6.2 of the CCA (DOE 1996f).

In the CCA analysis, variation in the hydraulic properties of the Culebra Dolomite were incorporated in the regional model by assigning different transmissivity values to every computational cell in the model grid. Because uncertainty in the estimated value of the Culebra transmissivity exists in areas where measurements have not been made, a large set of transmissivity fields were developed. Each transmissivity field was assumed to be a statistical representation of the natural variation in transmissivity that honors measured data that is equally likely to represent actual conditions. Details of the generation and use of transmissivity fields are described Section 4.1 of Appendix TFIELD of the CCA (DOE 1996f).

Flow conditions assumed in the local transport analysis performed for SEIS-II were based on transmissivity fields that incorporated the hydraulic impacts of future potash mining within the modeled region. Two conditions were considered: a partial mining scenario which considered the impacts of mining all potash reserves within the modeled region but outside of the Land Withdrawal Area, and a full mining scenario which considered the impact of mining all potash reserves within the modeled region including those found within the Land Withdrawal Area. Details of implementation of these scenarios are described in Chapter 6 of the CCA (DOE 1996f).

For purposes of the SEIS-II analysis, a 75th percentile flow field was selected from each set of the one hundred flow fields generated from the calibrated regional two-dimensional flow model developed to approximate partial and full potash mining conditions in the CCA (Appendix TFIELD, DOE 1996f). The ranking of flow fields was based on their predicted travel times from the WIPP to the Land Withdrawal Area boundary. The velocity fields generated from the 75th percentile flow fields for partial and full potash mining conditions provided the basis for transport calculations made using the SECOTP2D model of the Culebra Dolomite to a stock well located 3 kilometers (2 miles) downgradient from the point of intrusion.

H.7.2 Radionuclide and Heavy Metal Transport

Transport and retardation of contaminants introduced into the Culebra Dolomite under the75th percentile parameter values cases of drilling through the repository into a pressurized brine reservoir were simulated for four radionuclides previously considered in the CCA analyses: Am-241, Pu-239, Th-230, and U-234. The SEIS-II analyses used parameter values from the CCA for radionuclide transport in the SECOTP2D model of the Culebra Dolomite. Parameters values are provided in Section MASS.15.2 of Appendix MASS and Appendix PAR (see Parameters 49 through 57) of the CCA (DOE 1996f).

The distribution coefficients (K_d) used in the SEIS-II analyses were based on the linear adsorption isotherm used in the CCA to represent the retardation that occurs as dissolved radionuclides and heavy metals are sorbed onto different minerals (primarily dolomite) lining pore walls and fractures. The linear isotherm uses a single parameter, K_d, to express the relationship between sorbed concentration and liquid concentration. For the SEIS-II analysis, K_ds were selected to conservatively represent the K_d distributions used in the CCA, which were derived from experimental data. K_ds for these radionuclides were as follows: Am-241, 0.14 cubic meters per kilogram; Pu-239, 0.14 cubic meters per kilogram; Th-230, 5.675 cubic meters per kilogram; and U-234, 7.52 x 10^{-3} cubic meters per kilogram. The assumed oxidation states for these radionuclides were +III, +III, +IV, and +VI, respectively.

Transport results for U-234, the most mobile of the four radionuclides evaluated, were assumed to be representative of transport for seven other radionuclides — Ac-227, Np-237, Pa-231, Pa-233, Pb-210, Pu-240, and Ra-226— and four heavy metals— beryllium, cadmium, lead, and mercury. This assumption was made because these radionuclides and elements were not included in CCA analyses; thus, complete sets of WIPP-specific transport parameters were not available for them. Dose adjustment factors were developed for each radionuclide that allowed the calculated U-234 dose to be adjusted based on each radionuclide's specific ingestion dose factor, radioactive half-life, and release rate into the Culebra at the point of intrusion (due to the inventory differences between each of the radionuclides and U-234). Potential health impacts from heavy metals were estimated in a similar manner, using the predicted concentration of U-234 to estimate a concentration for each of the heavy metals. Estimated concentrations were then used with appropriate metal-specific slope factors and reference doses to estimate carcinogenic and noncarcinogenic impacts.

H.8 ANALYSIS AND RESULTS OF LONG-TERM PERFORMANCE ASSESSMENT

Analysis and results of the long-term performance assessment analysis conducted for the Proposed Action and Action Alternatives 1, 2, and 3 are provided in this section.

H.8.1 Exposure Scenarios

This section describes the exposure scenarios used to evaluated the potential impacts of undisturbed and disturbed conditions for long-term performance assessment.

H.8.1.1 Undisturbed Conditions

For all cases of undisturbed conditions evaluated for the Proposed Action and Action Alternatives 1, 2, and 3, radionuclides and metals did not reach the accessible environment. There were no postulated scenarios in which individuals or populations could be exposed. Therefore, results of undisturbed cases are presented as the extent of radionuclide and heavy metal migration away from the repository and were analyzed over the 10,000 years after closure.

H.8.1.2 Disturbed Conditions

Exposure scenarios for disturbed conditions evaluated the potential impacts to individuals who could be directly exposed by drilling an exploratory borehole and indirectly exposed by drilling through the repository into a pressurized brine reservoir. The later exposure scenario could allow contaminants to reach the accessible environment under unfavorable conditions, using 75th percentile parameter values as discussed in Section H.6. There were no postulated scenarios in which populations could be exposed.

Human health impacts from disturbed conditions evaluated for long-term performance assessment use the same metrics as those for individual exposures for human health (Appendix F) and facility accidents (Appendix G). Impacts from radiation exposure or intakes of radionuclides are presented as the probability of a latent cancer fatality (LCF) for an exposed individual. Impacts were based on the calculated radiation dose to the individual (external dose or committed effective dose equivalent). Carcinogenic impacts of exposure to metals are presented as the probability of cancer incidence for an exposed individual. Noncarcinogenic impacts of exposure to metals are presented as a hazard index (HI) for the exposed individual.

As noted in Section H.5.2.3, exposure to VOCs was not included in long-term performance assessment calculations. Impacts from acute exposures from surface releases caused by drilling through the repository were considered to be bounded by impacts from WIPP disposal accidents (Appendix G). A drilling crew member could be exposed to VOCs if the drilling occurred when high gas pressures caused a mud blowout. The concentration and duration of VOC exposure would not be higher than those of WIPP disposal accidents. The probability of cancer incidence would be no more than 3×10^{-7} from exposure to VOCs. Toxicololgical impacts were estimated to

RELEASE SCENARIOS CONSIDERED BUT NOT ANALYZED

During the comment response period for the Draft SEIS-II, the public expressed significant interest in the potential impacts of a number of release scenarios not considered by DOE in the long-term performance assessment analysis of the WIPP site. The following summarizes the reasons why certain release scenarios were not analyzed in SEIS-II.

Fluid Injection: The potential impacts of fluid injection in the form of water flooding and salt water disposal are evaluated in the Compliance Certification Application (CCA). The results provided in Section SCR.3.3.1.3.1 of the CCA indicate that fluid injection would not have a significant impact on repository performance. Even when the least favorable rock properties were specifically considered, the amount of brine reaching the repository over 10,000 years would be well within the range of volumes expected to flow into the repository during normal, undisturbed performance. Therefore, fluid injection was screened out for consideration in the SEIS-II analysis on the basis of a low consequence to the long-term performance of the disposal system.

Karst and Dissolution Processes: For nearly 20 years, DOE has investigated the hydrology of important geologic units overlying the WIPP facility and the importance of karst features and related dissolution processes in defining the surface features in the region surrounding the WIPP site. A description of the current understanding of the extent, timing, and features related to dissolution, including a brief history of past project studies related to karst in the area surrounding WIPP, is presented in Section 2.1.6.2.1 of the CCA. These studies have shown that while there is considerable evidence of dissolution and karst features at shallow depths, no evidence has been collected to date that would suggest that shallow dissolution processes are active within the deeper Salado Formation. Deep dissolution at the WIPP site has been eliminated from the SEIS-II performance assessment calculations on the basis of low probability of occurrence over the next 10,000 years. Additional information supporting this conclusion is provided in Section SCR.1.1.5.1 of Appendix SCR of the CCA (DOE 1996f).

Climate Change: The uncertainties of possible climate changes, including the effects on groundwater flow and potential radionuclide transport in groundwater, have been incorporated into the performance assessment analyses of the CCA (Section 2.5 and Appendix CLI). Overall, the results suggest that if an intrusion were to cause a release of radionuclides to one of the significant water-bearing units (e.g., the Culebra Dolomite) above the repository, radionuclides would be transported at a faster rate to the accessible environment but would be diluted to lower concentrations by increased groundwater flow. Direct effects that do not involve groundwater (e.g., wind) are not likely to affect the long-term performance of WIPP because of its depth below the land surface. Thus, the impact of climatic change on the long-term performance of the repository would be minimal.

Multiple Borehole Intrusions: The single scenario involving borehole intrusion described in SEIS-II was used to bound the impacts from an exploratory borehole into the repository. For a multiple intrusion scenario, the impacts after the first intrusion would be expected to decrease as the result of dissipating gas pressure and the diminishing amount of material that could be released at any one borehole. Thus, the dose from materials released in future intrusions would be well below the dose estimated from the first intrusion because of the overall decay in the available inventory.

Explosions and Criticalities: The potential for explosions (specifically, gas explosions) of hydrogen and methane generated by waste degradation are extremely unlikely in the long term because of the anoxic environment in the repository. Should such explosions occur, the effect would be limited to the disruption of rock units in the immediate vicinity of the disposal region and the possible creation of pathways for fluid migration above and below the disposed of waste. While this type of impact was not explicitly evaluated, SEIS-II simulated conditions representing the highly permeable DRZ (e.g., see Case 3 for the Proposed Action in Section H.8). The planning-basis WAC establish nuclear criticality criteria for TRU waste that define the maximum allowable quantity of fissile material as two times the measurement error when the waste packages are assayed. Because of these limitations, the formation of a critical mass in the geometry necessary to achieve a self-sustaining nuclear chain reaction in the WIPP environment is considered to be an “incredible” event. Additional information is provided in the CCA (DOE 1996f).

Thermal Impacts: The planning-basis WAC include thermal-loading design limits of 10 kilowatts per surface acre, which would preclude any significant thermal impacts as the result of emplaced TRU wastes in WIPP. DOE has found that the average increase in temperature at depth due to radioactive decay of emplaced CH-TRU and RH-TRU waste would be less than 2°C, which is insufficient to induce significant thermal convection and thermal stresses and strain and modify anticipated chemical reactions. Increased temperatures from the heat of geothermal origin and compression of gas as a result of salt creep inward are predicted to be similarly insignificant in the CCA (Section SCR.2) and were not analyzed in detail in SEIS-II.

be a small fraction of the immediately dangerous to life or health (IDLH) values, but some serious but non-life-threatening health impacts could occur based on calculated comparisons to Emergency Response Planning Guidelines (ERPGs). Impacts from chronic exposures were considered to be bounded by impacts from routine WIPP disposal operations (Appendix F), with a lifetime probability of cancer incidence no higher than 9×10^{-8} for any alternative.

Estimates of radiation doses were calculated using the GENII code, and estimates of impacts from heavy metal exposures were calculated using the MEPAS® code. Descriptions of these codes and their uses are provided in Appendix F.

Surface Release Caused by Drilling Into the Repository

For this scenario, a hypothetical exploratory drilling operation inadvertently penetrates a waste panel in the repository. As a result, the drilling brings waste originating in the repository to the land surface and exposes individuals involved in the drilling operation to radionuclides and hazardous chemicals. Impacts from radiation and heavy metal exposure were evaluated using two exposure scenarios for individuals associated with the drilling process. These exposure scenarios were:

- A drill crew member directly involved in the drilling of the exploratory borehole. This individual was assumed to be exposed to external radiation from materials at the drill head and in the drill cutting pond and assumed to inadvertently ingest small amounts of borehole material releases. Ingestion was assumed to be 100 milligrams of cuttings per day at an average concentration derived from all of the heavy metals in the top 15 centimeters (6 inches) of a 10 meter by 10 meter (33 foot by 33 foot) cuttings disposal pile. The drill crew member was assumed to be exposed to the materials for a period of 168 hours (i.e., 21 working days).

- A well-site geologist involved in the periodic examination of cuttings generated by the drilling process. This individual would be exposed to external radiation only through the direct handling of an exhumed fragment of waste. The geologist was assumed to pick up a cylindrical waste fragment 5 centimeters (2 inches) in radius, with a volume of 524 cubic centimeters (32 cubic inches). A maximum exposure time of one hour was assumed.

Drilling was assumed to occur sometime after the end of the active institutional control period and was timed to coincide with the maximum potential health impact to exposed individuals. The concentration of radionuclides in the exhumed waste was assumed to be the emplacement concentration decayed to the time of intrusion. Because results of BRAGFLO calculations showed a steady increase in brine pressures in the repository (approximately 5 to 14 megapascal) over the initial 2,000 years following repository closure, the potential impact of repository pressure conditions on the release of materials up the borehole was examined. Given the range of potential release processes involved in the CUTTINGS_S code (see Figure H-3) for these pressure ranges, the potential impact of the repository pressure on the release of materials through the borehole and its impacts on exposed individuals at various intrusion times were examined.

Calculations with the CUTTINGS_S code were performed using repository pressures simulated with BRAGFLO code calculations and various intrusion times for undisturbed conditions (Cases 1 and 3 described below) of the Proposed Action. These calculations estimated the amount of material and the associated radionuclide activity that would be released through the borehole to the ground surface. The radionuclide activities were then used to calculate impacts to an exposed individual. Details of the random sampling technique used to select the radionuclide inventory for drilling intrusions are provided in Section A.4.8 of Appendix A.

Radionuclide and heavy metal releases were calculated with the CUTTINGS_S code at 100, 200, 300, 400, 500, 800, 1,200 and 2,000 years after repository closure. Results show that, although the amount of material released up the borehole from potential intrusions later increases, the released radionuclide activity would decrease because the radionuclides with short half-lives (which would contribute significantly to early doses) would have decayed. Calculations showed the maximum dose would occur at 400 years after closure for median parameter cases (Proposed Action Case 2) and 300 years after closure for 75th percentile parameter cases (Proposed Action Case 4), when the repository pressure would exceed the 8 megapascal threshold in the CUTTINGS_S model that divides the cuttings and cavings release mode from the more significant spallings release mode. For the Proposed Action (and extrapolated to Action Alternatives 1, 2, and 3), the drilling intrusion scenario was assumed to occur at 400 years after closure for median parameter cases and 300 years after closure for 75th percentile parameter cases. Use of 75th percentile parameter values resulted in material releases approximately 40 percent higher than did the use of median parameter values, with correspondingly higher impacts.

Drilling Through the Repository into a Pressurized Brine Reservoir

A hypothetical drilling event was assumed to breach the repository and penetrate a hypothetical pressurized brine reservoir in the Castile Formation below the repository horizon. For this condition, brine in the reservoir has the potential to come into contact with wastes in the repository and move further up the borehole to more permeable units lying above the repository horizon, such as the Culebra Dolomite or Rustler Formation. Effects of the migration to the accessible environment were evaluated at a well located 3 kilometers (2 miles) downgradient from WIPP. This well was assumed to pump the contaminated water to the stock ponds used by cattle. Direct uses by humans were not considered because of the high salinity of groundwater in the area. Beef from cattle using this water was assumed to be consumed by an individual such as a cattle rancher at a rate of 42 kilograms (93 pounds) of beef annually (approximately 4 ounces a day) over a 70-year lifetime.

H.8.2 Results for the Proposed Action

The four cases below were analyzed for the Proposed Action. The cases considered the following conditions:

- Case 1 considered undisturbed repository performance. Median parameter values were used for all input variables where probability distributions had been defined.

- Case 2 considered an intrusion resulting from exploratory drilling. It was assumed that the repository would be penetrated and the drill would intercept a pressurized brine pocket in the Castile Formation. Median parameter values were used.

- Case 3 considered undisturbed repository performance. Seventy-fifth percentile parameter values were used for all input variables where probability distributions had been defined.

- Case 4 considered the same disturbed conditions as Case 2. Seventy-fifth percentile parameter values were used.

Cases 1 through 4 were simulated with the BRAGFLO code to produce brine and gas pressure fields and flow velocity fields for use in subsequent transport and direct-release calculations. Brine pressures at each time step simulated by BRAGFLO were extracted and averaged, with respect to volume, for a single panel. The average pressures show the time evolution of brine pressure predicted for the repository. The resulting pressure curves, as a function of time after closure, are depicted in Figure H-8. The pressure release of the waste panel, as a result of the exploratory drilling event at 400 years after closure in Case 2 and 300 years after closure in Case 4, are clearly evident. Brine pressures in the panel remain below lithostatic conditions at 10,000 years after closure for the intrusion cases. Biodegradable material is completely consumed in the gas generation process for all four cases, and the iron inventory is completely consumed in the gas generation process for both of the intrusion cases. The gas generation for the undisturbed cases is brine-limited, and the corrosion proceeds slowly enough that not all of the iron is consumed.

H.8.2.1 Impacts of Undisturbed Conditions

The extent of radionuclide migration for undisturbed conditions at 10,000 years after closure for Case 3 (using 75th percentile parameter values) over the model domain is presented in Figure H-9. Case 3 resulted in slightly more extensive migration than Case 1. The figure shows the locations in the modeled region where the total radionuclide activity concentration in brine (summed over

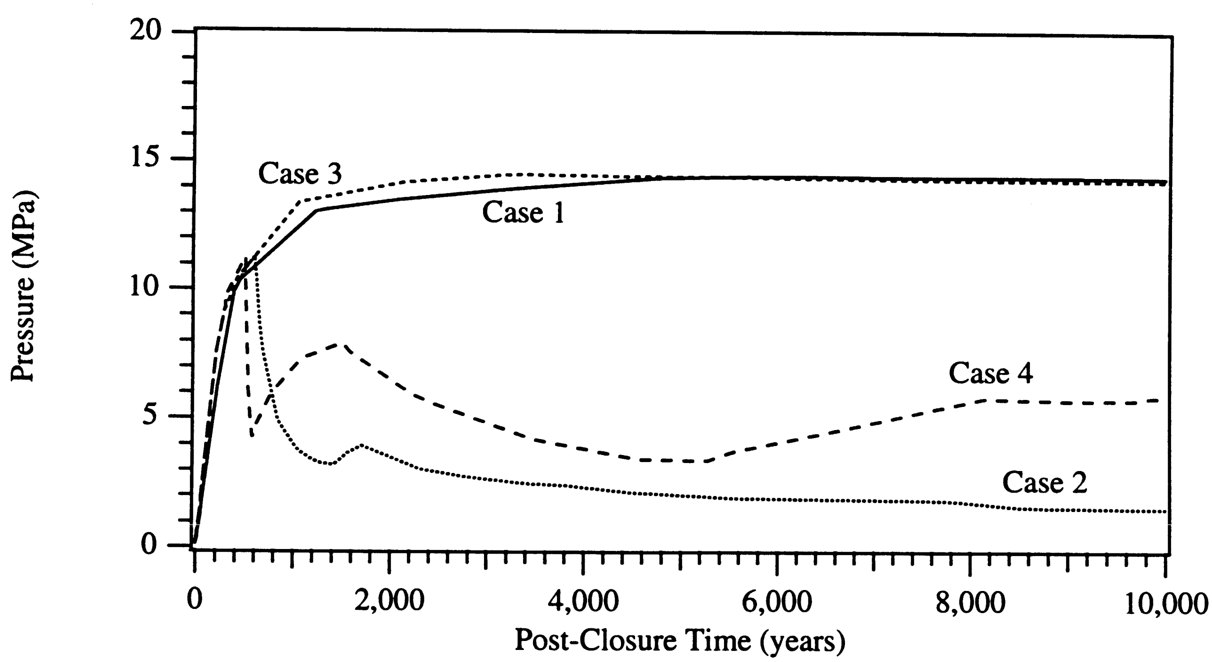


Figure H-8
Volume-Averaged Brine Pressures in a Waste Panel for the Proposed Action

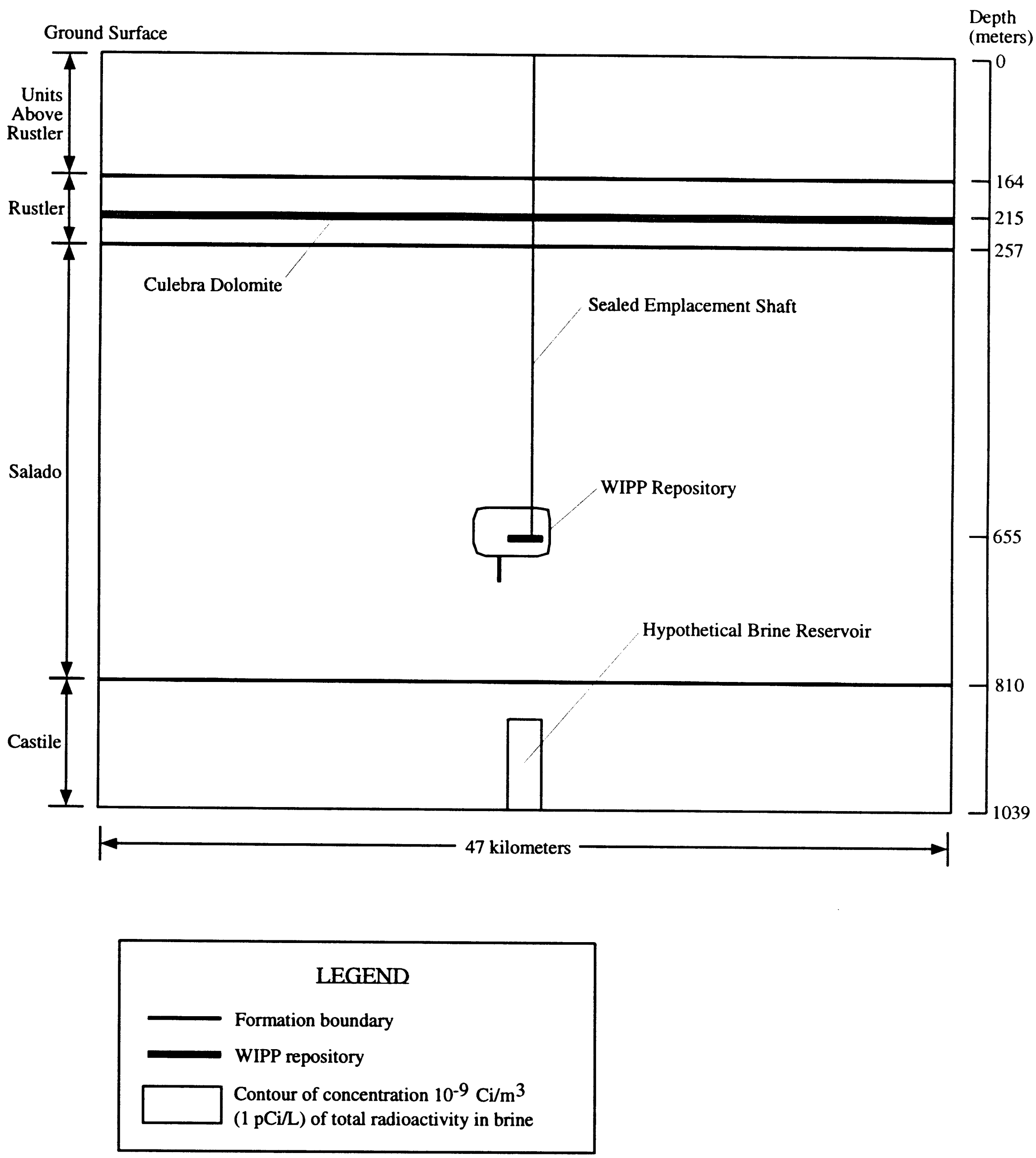


Figure H-9
Extent of Radionuclide Migration at 10,000 Years with Undisturbed Conditions Using 75th Percentile Parameter Values (Case 3) for the Proposed Action

30 radionuclides) is equal to 1 pCi per liter (1×10^{-9} curies per cubic meter). Migration of heavy metals (lead, mercury, beryllium, and cadmium) was also simulated. For Case 3, the extent of total heavy metal concentrations of one part per billion (1×10^{-3} milligrams per cubic meter) of brine is approximately the same as the one pCi per liter level of total radionuclide activity concentration.

Because lead is by far the predominant heavy metal in the analyzed inventory (see Appendix A), the total heavy metal concentration can be interpreted to approximate the predicted concentration of lead only.

The total vertical scale of the modeled region in Figure H-9 is 1,039 meters (3,409 feet), with the horizontal extent approximating 47 kilometers (29 miles). For Case 3, migration of total radionuclide concentrations at one pCi per liter extended vertically below the repository 60 meters (200 feet) and above the repository 40 meters (130 feet). The furthest extent of lateral migration at this same concentration was to 1,900 meters (6,200 feet) from the south (left) edge of the repository. Migration of heavy metals at a concentration of one part per billion were approximately the same as radionuclides at 1 pCi per liter.

H.8.2.2 Impacts of Disturbed Conditions

This section presents the impacts of two exposure scenarios evaluated for disturbed conditions of the Proposed Action.

Surface Release Caused by Drilling into the Repository

Under the Proposed Action, the estimated releases to the ground surface from a drilling intrusion 400 years after closure would be 3.1 curies for Case 2 (median parameter values) and 4.5 curies for a drilling intrusion at 300 years after closure for Case 4 (75th percentile parameter values). In both cases, releases were mainly from Am-241, Pu-239, and Pu-238. Heavy metal releases from an intrusion were estimated at 24 and 2.1 kilograms (53 and 4.6 pounds) of lead and mercury, respectively, for Case 2. For Case 4, 31 and 2.7 kilograms (67 and 6.0 pounds) of lead and mercury, respectively, were released in Case 4. Results of these analyses are provided in Tables H-25 and H-26.

For Case 2 (median parameter values), radiological impacts to the drilling crew member from a drilling intrusion resulted in a 1.8×10^{-4} probability of an LCF. For Case 4 (75th percentile parameter values), the radiological impacts were higher at a 4.4×10^{-4} probability of an LCF. The dominant exposure pathway was by ingestion of drill cuttings, and Am-241 was the most significant radionuclide, contributing approximately 97 percent of the total dose. Results are presented in Table H-27. The drilling crew member may also ingest heavy metals (lead, beryllium, cadmium, and mercury) in the drill cuttings. There would be a 2.2×10^{-8} probability of a cancer incidence to the drill crew member. There would be no noncarcinogenic impacts expected from the ingestion of the metals because all hazard indices would be much less than one. Impacts from the ingestion of heavy metals are shown in Table H-28.

Table H-25
Radionuclide Releases (curies)
to the Ground Surface from Drilling Intrusions for the Proposed Action

Radionuclide	Case 2 (Median Parameters) Intrusion at 400 Years After Closure	Case 4 (75th Percentile Parameters) Intrusion at 300 Years After Closure
Ac-227	8.60E-06	6.25E-06
Am-241	1.04E+00	2.47E+00
Cm-243	3.59E-09	5.89E-07
Cm-244	1.54E-10	5.82E-07
Cs-137	5.34E-06	7.17E-04
Pa-231	9.17E-06	6.97E-06
Pb-210	1.69E-06	1.99E-06
Pu-238	1.53E-01	2.79E-01
Pu-239	1.89E+00	1.69E+00
Pu-240	8.28E-05	1.52E-04
Pu-241	4.45E-10	6.53E-06
Sr-90	4.30E-06	6.58E-04
U-232	5.74E-06	7.45E-05
U-233	2.12E-02	4.01E-02
U-234	3.00E-03	1.22E-03
Y-90	4.30E-06	6.58E-04
Total	3.11	4.48

Table H-26
Releases (kilograms) of Heavy Metals
to the Ground Surface from Drilling Intrusions for the Proposed Action

Heavy Metal	Case 2 (Median Parameters) Intrusion at 400 Years After Closure	Case 4 (75th Percentile Parameters) Intrusion at 300 Years After Closure
Lead	2.4 E+01	3.05E+01
Beryllium	1.3E-01	1.59E-01
Cadmium	1.8E-03	2.27E-03
Mercury	2.1E+00	2.69E+00
Total	2.6E+01	3.33E+01

Radiological impacts to a well-site geologist from external radiation exposure for Case 2 (median parameter values) would result in a 2.8 x 10^{-9} probability of an LCF from CH-TRU waste and a 8.9 x 10^{-10} probability of an LCF from RH-TRU waste. For Case 4 (75th percentile parameter values), the radiological impacts would be a 3.2 x 10^{-9} probability of an LCF from CH-TRU waste and a 3.3 x 10^{-9} probability of an LCF from RH-TRU waste. Results of these analyses are presented in Table H-29.

Drilling Through the Repository into a Pressurized Brine Reservoir

Performance assessment analyses of the Proposed Action indicated that releases of radionuclides and heavy metals from the WIPP repository would not reach the accessible environment or the Culebra Dolomite for Case 2 (median parameter values). However, the results of Case 4 (75th percentile parameter values) analyses indicated the potential for a release to the Culebra and the accessible environment. Radionuclide migration with disturbed conditions at 10,000 years after closure for Case 2 (median parameter values) and Case 4 (75th percentile parameter values)

Table H-27
Radiation Dose to a Member of the Drilling Crew from Drilling Intrusions for the Proposed Action

Case 2 (Median Parameters) Intrusion at 400 Years After Closure				Case 4 (75th Percentile Parameters) Intrusion at 300 Years After Closure			
Radionuclide	**Ingestion Dose (rem)**	**External Dose (rem)**	**Total Dose (rem)**	**Radionuclide**	**Ingestion Dose (rem)**	**External Dose (rem)**	**Total Dose (rem)**
Am-241	3.4E-01	1.0E-02	3.5E-01	Am-241	8.4E-01	2.6E-02	8.6E-01
Pu-239	9.5E-03	2.1E-04	9.7E-03	Pu-239	7.5E-03	1.7E-04	7.6E-03
Pu-238	6.7E-04	7.4E-06	6.8E-04	Pu-238	1.3E-03	1.4E-05	1.3E-03
U-233	5.2E-05	7.8E-06	6.0E-05	Cs-137	3.2E-06	1.0E-03	1.0E-03
Ac-227	1.1E-05	1.7E-09	1.1E-05	U-233	1.0E-04	1.5E-05	1.1E-04
Pa-231	9.1E-06	6.5E-07	9.7E-06	Y-90	6.7E-07	1.0E-05	1.1E-05
Cs-137	2.4E-08	7.8E-06	7.8E-06	Ac-227	8.3E-06	1.2E-09	8.3E-06
U-234	7.2E-06	4.3E-07	7.7E-06	Sr-90	7.5E-06	1.8E-07	7.7E-06
Pb-210	8.6E-07	1.4E-09	8.6E-07	Pa-231	6.9E-06	4.9E-07	7.4E-06
Pu-240	3.9E-07	4.0E-09	3.9E-07	U-234	2.8E-06	1.8E-07	3.0E-06
Y-90	4.4E-09	6.7E-08	7.2E-08	Pb-210	1.0E-06	1.6E-09	1.0E-06
Sr-90	5.0E-08	1.2E-09	5.1E-08	Pu-240	7.0E-07	7.3E-09	7.1E-07
U-232	3.6E-08	1.3E-09	3.7E-08	U-232	4.8E-07	1.8E-08	4.9E-07
Cm-243	8.5E-10	8.2E-10	1.7E-09	Cm-243	1.4E-07	1.3E-07	2.7E-07
Cm-244	2.8E-11	6.0E-15	2.8E-11	Cm-244	1.1E-07	2.3E-11	1.1E-07
Pu-241	3.2E-14	4.0E-20	3.2E-14	Pu-241	4.7E-10	5.9E-16	4.7E-10
Total Dose	3.5E-01	1.0E-02	3.6E-01	Total Dose	8.5E-01	2.7E-02	8.7E-01
Probability of LCF	1.8E-04	5.1E-06	1.8E-04	**Probability of LCF**	4.2E-04	1.4E-05	4.4E-04

Table H-28
Carcinogenic and Noncarcinogenic Impacts from Ingestion of Metals for Drilling Intrusion into the Repository for the Proposed Action

	Case 2 (Median Parameters)		Case 4 (75th Percentile Parameters)	
Heavy Metal	**Probability of Cancer Incidence**	**Hazard Index**	**Probability of Cancer Incidence**	**Hazard Index**
Beryllium	2.2E-08	1.0E-06	2.2E-08	1.0E-06
Cadmium	4.6E-10	1.5E-07	4.6E-10	1.4E-07
Lead	-	7.0E-04	-	6.9E-04
Mercury	-	2.9E-04	-	2.9E-04

Table H-29
Radiation Dose to the Well-Site Geologist
from Drilling Intrusions for the Proposed Action

Case 2 (Median Parameters) Intrusion at 400 Years After Closure				Case 4 (75th Percentile Parameters) Intrusion at 300 Years After Closure			
CH-TRU Waste		RH-TRU Waste		CH-TRU Waste		RH-TRU Waste	
Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)
Am-241	3.7E-06	Am-241	7.3E-07	Am-241	4.5E-06	Sr-90	3.6E-06
U-234	1.8E-06	U-234	5.4E-07	U-234	1.7E-06	Cs-137	1.6E-06
Pu-239	7.4E-08	Sr-90	3.0E-07	Pu-239	7.6E-08	Am-241	8.6E-07
U-233	4.5E-08	Cs-137	1.6E-07	U-233	4.5E-08	U-234	5.4E-07
Pu-240	1.5E-08	U-233	2.6E-08	Pu-238	1.9E-08	U-233	2.6E-08
Pu-238	8.5E-09	Pu-239	1.0E-08	Pu-240	1.5E-08	Pu-239	1.0E-08
Ac-225	7.5E-09	Pu-240	3.6E-09	Sr-90	8.9E-09	Pu-240	3.9E-09
Sr-90	8.3E-10	Ac-225	2.5E-09	Ac-225	5.7E-09	Ac-225	1.9E-09
Cs-137	3.0E-10	Pa-231	3.6E-10	Cs-137	3.0E-09	Y-90	5.2E-10
Pb-210	1.3E-11	Pu-238	6.0E-11	Pb-210	1.3E-11	Pa-231	2.5E-10
Pa-231	1.1E-12	Y-90	4.4E-11	Y-90	1.3E-12	Pu-238	1.3E-10
Y-90	1.2E-13	U-232	8.8E-14	Pa-231	7.8E-13	U-232	2.4E-13
U-232	8.1E-14	Pb-210	7.1E-18	U-232	2.0E-13	Pb-210	2.8E-18
Pu-241	1.8E-17	Pu-241	9.7E-22	Pu-241	1.8E-17	Pu-241	1.2E-19
Total Dose	5.7E-06	Total Dose	1.8E-06	Total Dose	6.4E-06	Total Dose	6.6E-06
Probability of LCF	2.8E-09	**Probability of LCF**	8.9E-10	**Probability of LCF**	3.2E-09	**Probability of LCF**	3.3E-09

showed migration of total radionuclide concentrations of one pCi per liter migrating upward and downward in the exploratory borehole. Results for Case 4 are shown in Figure H-10. Heavy metal concentrations equal to one part per billion showed similar patterns of migration. In both cases, contaminants at these levels penetrated a short distance into the surrounding rock and down into the Castile Formation and the hypothetical brine reservoir. Transport analyses of releases to the Culebra showed that contaminants would be highly sorbed and only small amounts would migrate from the point of intrusion over 10,000 years.

The estimated maximum release rate for Case 4 of 0.65 Ci per year (predominantly Am-241, Pu-239, and Pu-240) and 23.1 kilograms per year (50.9 pounds per year) of heavy metals (predominantly lead) would occur at approximately 1,600 years after closure. The estimated total release to the Culebra Dolomite would be 523 Ci and 20,960 kilograms (46,220 pounds) of heavy metals, with the release occurring from 700 years to 2,200 years after closure.

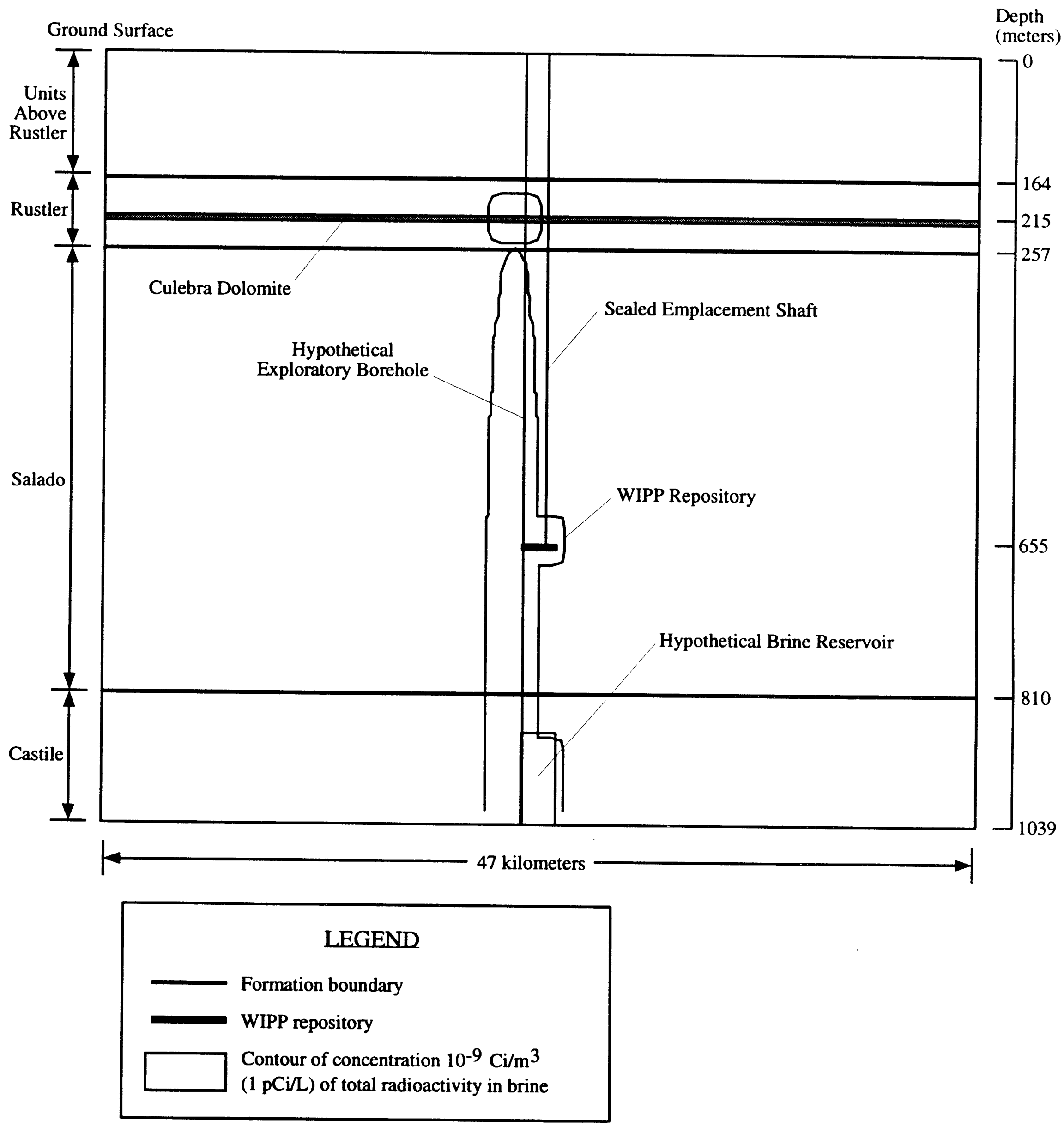


Figure H-10
Extent of Radionuclide Migration at 10,000 Years with Disturbed Conditions Using 75th Percentile Parameter Values (Case 4) for the Proposed Action

The potential for off-site human health impacts due to migration of contaminants through the Culebra was evaluated. Two flow-field conditions were evaluated: one to represent partial mining and one to represent full mining. Changing the hydraulic conductivity of the Culebra to reflect the hydrogeologic impacts of future potash mining would result in an overall increase in travel times to locations downgradient of the WIPP site. This effect has been attributed to the increases in hydraulic conductivity in the postulated mined areas, causing groundwater flow paths to change from areas of higher transmissivity south and east of the WIPP site to a more westerly path into areas of lower transmissivity. Calculated travel times for the full mining scenarios were nearly two orders of magnitude higher than those calculated for the partial mining scenario, resulting in lower potential impacts.

For partial and full potash mining conditions, impacts were calculated to an individual, such as a rancher, who consumed beef from cattle that had consumed water from a stock well located downgradient at the Land Withdrawal Area boundary 3 kilometers (2 miles) from the point of intrusion. Radiological impacts would be higher under partial mining but would be negligible, with a 7×10^{-28} probability of an LCF. The principal dose contributor was Pu-239 with all 10 other radionuclides considered less than 0.02 percent of Pu-239. Ingestion of heavy metals from consumption of contaminated beef was also found to result in negligible impacts, with a probability of cancer incidence of 3×10^{-27}. Impacts under full mining conditions would be significantly lower than those under partial mining, with radiological impacts of a 4×10^{-41} probability of an LCF and ingestion of heavy metals resulting in a probability of cancer incidence of 3×10^{-37}.

H.8.3 Results for Action Alternative 1

Long-term performance assessment analyses were conducted for Action Alternative 1 using the same methods as were used for the Proposed Action. The radionuclide and heavy metal inventories were increased from those used in the Proposed Action to account for the increased inventory destined for WIPP under this alternative. Total radionuclide activities of 7.3×10^{6} Ci in CH-TRU waste and 5.1×10^{6} Ci in RH-TRU waste were included for this alternative. Detailed information on the radionuclide and heavy metal inventories are provided in Appendix A. The repository size was adjusted from 10 panels (for the Proposed Action) to 68 panels. The material properties outside the repository were not changed from the comparable cases analyzed for the Proposed Action.

The four cases below were analyzed for Action Alternative 1. The cases considered the following conditions:

- Case 6 considered undisturbed repository conditions. Median parameter values were used for all input variables where probability distributions had been defined.

- Case 7 considered disturbed conditions where a borehole from exploratory drilling is assumed to breach the repository and penetrate a pressurized brine pocket in the Castile Formation. Median parameters values were used.

- Case 8 considered undisturbed repository conditions. Seventy-fifth percentile parameter values were used for all input variables where probability distributions had been defined.

- Case 9 considered the same disturbed conditions as Case 7. Seventy-fifth percentile parameter values were used.

Cases 6 through 9 were simulated with the BRAGFLO code to produce brine and gas pressure fields and flow velocity fields for use in subsequent transport and direct-release calculations. Brine pressures at each time step simulated by BRAGFLO were extracted and averaged, with respect to volume, for a single panel. The average pressures show the time evolution of brine pressure predicted for the repository. The resulting pressure curves, as a function of time after closure, are depicted in Figure H-11. The pressure release of the waste panel, as a result of the exploratory drilling events at 400 years after closure for Case 7 and 300 years after closure for Case 9, respectively, is clearly evident in this figure. Brine pressures in the panel remain below lithostatic conditions at 10,000 years after closure for the intrusion cases. The biodegradable material is completely consumed in the gas generation process for all four cases. The iron inventory is completely consumed in the gas generation process for both of the intrusion cases and for the median undisturbed case. The gas generation for the undisturbed 75th percentile parameter case is brine-limited, and the corrosion proceeds slowly enough that not all of the iron is consumed.

H.8.3.1 Impacts of Undisturbed Conditions

The extent of radionuclide migration for undisturbed conditions at 10,000 years after closure for Case 8 (75th percentile parameter values) is presented in Figure H-12. Case 8 is presented because it resulted in slightly more extensive migration than Case 6. For undisturbed conditions, results from Action Alternative 1 simulations showed about the same downward and lateral radionuclide and heavy metal migration as the Proposed Action. Migration of total radionuclide concentrations of one pCi per liter extends 20 meters (65 feet) below the repository. The maximum distance of lateral migration of the same concentration levels was about 1,900 meters (6,200 feet). The maximum extent of vertical upward migration was calculated to be about 40 meters (130 feet) above the top of the repository. The extent of total heavy metal migration to a concentration to one part per billion was approximately the same as the one pCi per liter total radionuclide level.

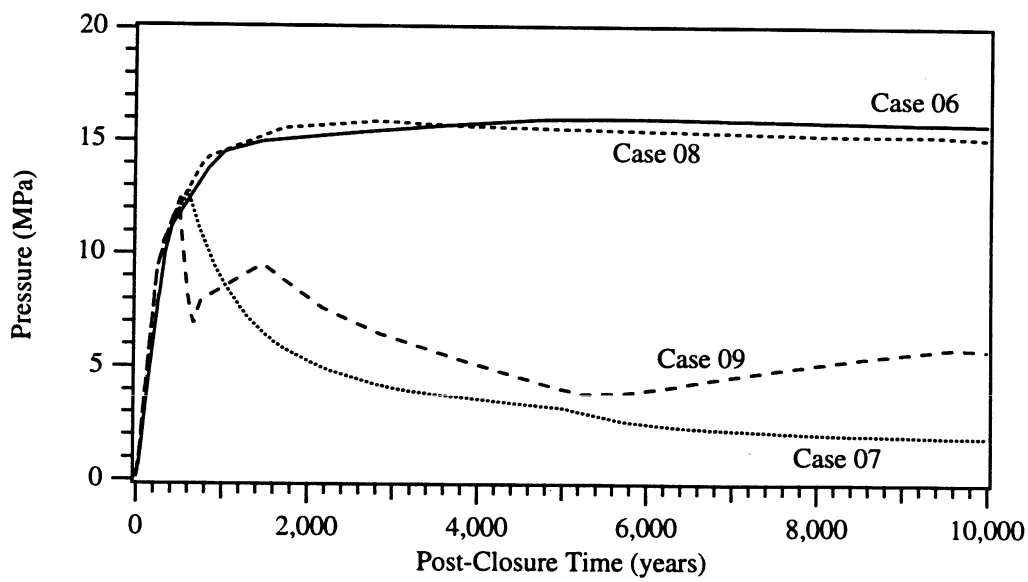


Figure H-11
Volume-Averaged Brine Pressures in a Waste Panel for Action Alternative 1

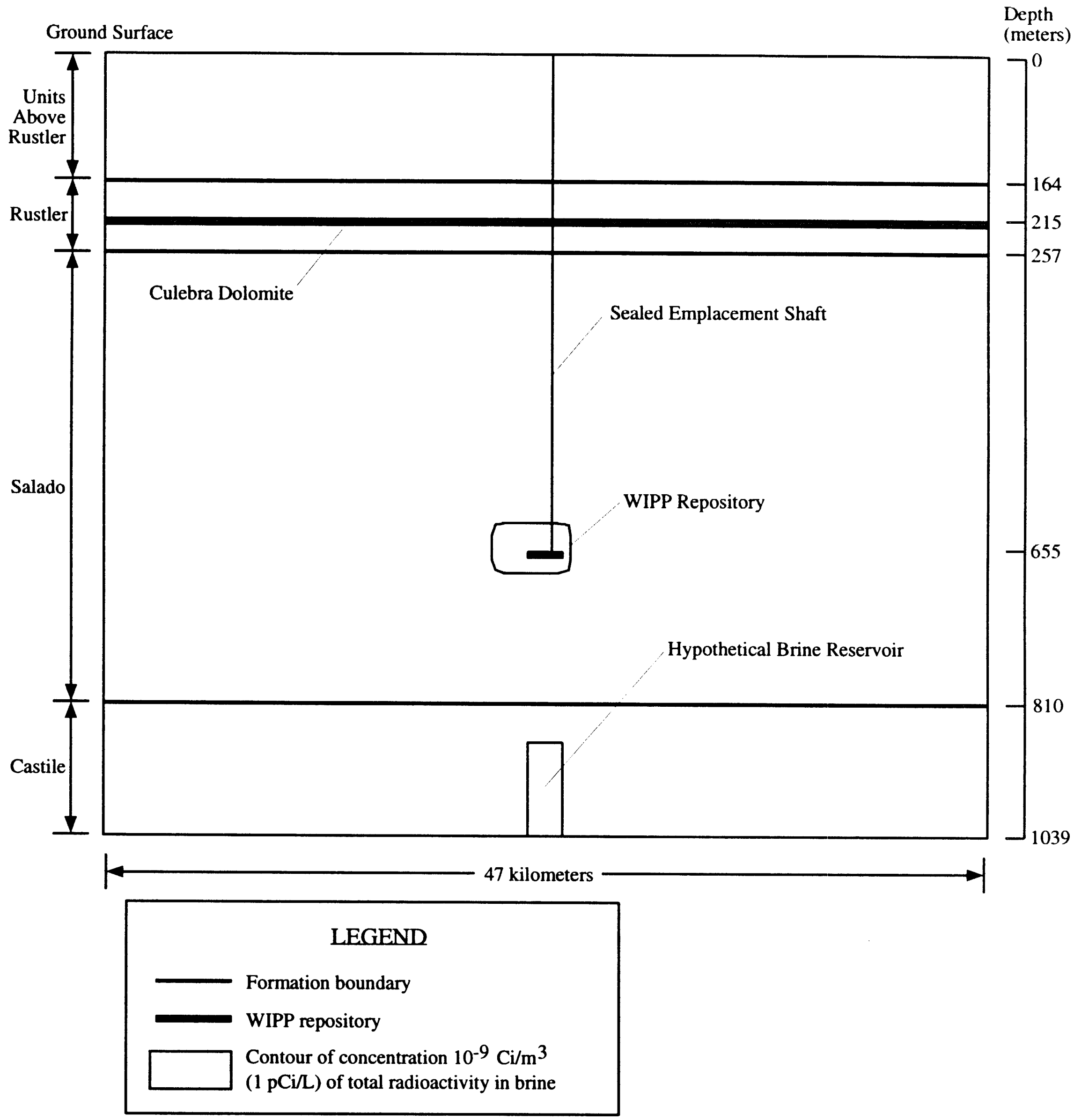


Figure H-12
Extent of Radionuclide Migration at 10,000 Years with Undisturbed Conditions Using 75th Percentile Parameter Values (Case 8) for Action Alternative 1

H.8.3.2 Impacts of Disturbed Conditions

This section presents the impacts of two exposure scenarios evaluated for disturbed conditions of Action Alternative 1.

Surface Release Caused by Drilling into the Repository

Under Action Alternative 1, the estimated release to the ground surface from a drilling intrusion depends on the kind of waste intercepted by the drilling equipment. For Case 7 (median parameter values), the estimated total radioactivity releases would be 2.6, 0.1, and 2.7 curies for drilling into CH-TRU waste, RH-TRU waste, and both CH-TRU and RH-TRU waste, respectively. For Case 9 (75th percentile parameter values), the releases would be 3.5, 0.15, and 3.6 curies for drilling into CH-TRU waste, RH-TRU waste, and both CH-TRU and RH-TRU waste, respectively. Major radionuclides were Am-241, Pu-239, and Pu-238.

Heavy metal releases from an intrusion were estimated at 6, 22, and 23 kilograms (13.2, 48.4, and 50.6 pounds) for drilling into CH-TRU waste, RH-TRU waste, and both CH-TRU and RH-TRU waste, respectively, for Case 7 (median parameter values). For Case 9 (75th percentile parameter values), the releases would be 8, 30, and 31 kilograms (18, 66, and 68 pounds) for drilling into CH-TRU waste, RH-TRU waste, and both CH-TRU and RH-TRU waste, respectively. Results are presented in Tables H-30 and H-31.

Radiological impacts to the drilling crew member would be 1.9×10^{-4}, 6.8×10^{-6}, and 1.7×10^{-4} probability of an LCF, respectively, for drilling into CH-TRU waste, RH-TRU waste, and both CH-TRU and RH-TRU waste for Case 7 (median parameter values). For Case 9 (75th percentile parameter values), the impacts would be somewhat higher at a 3.6×10^{-4}, 1.2×10^{-5}, and 3.6×10^{-4} probability of an LCF, respectively. The dominant exposure pathway was ingestion of drill cuttings, with Am-241 contributing 87 to 98 percent of the total dose. Results are presented in Table H-32. Ingestion of heavy metals for this scenario would result in a 1.7×10^{-8}, 1.0×10^{-9}, and 1.8×10^{-8} probability of cancer incidence, respectively. There would be no noncarcinogenic impacts expected from ingestion of the metals because all hazard indices would be much less than one. Impacts from ingestion of heavy metals are shown in Table H-33.

Radiological impacts to the well-site geologist from external radiation exposure for Case 7 (median parameter values) would result in a 2.4×10^{-9} probability of an LCF from CH-TRU waste and a 1.5×10^{-9} probability of an LCF for RH-TRU waste. For Case 9 (75th percentile parameter values), the radiological impacts would be a 2.7×10^{-9} probability of an LCF from CH-TRU waste and a 4.9×10^{-9} probability of an LCF from RH-TRU waste. Results of these analyses are presented in Table H-34.

Drilling Through the Repository into a Pressurized Brine Reservoir

Radionuclide migration under disturbed conditions at 10,000 years after closure for Case 7 (median parameter values) and Case 9 (75th percentile parameter values) showed migration of total radionuclide concentrations of one pCi per liter migrating upward and downward in the exploratory borehole. Heavy metal concentrations equal to one part per billion showed similar patterns of migration. In both cases, contaminants at these levels penetrated a short distance into the surrounding rock and down into the Castile Formation and the hypothetical brine reservoir. The

Table H-30
Radionuclide Releases (curies)
to the Ground Surface from Drilling Intrusions for Action Alternative 1

Radionuclide	Case 7 (Median Parameters) Intrusion at 400 Years After Closure			Case 9 (75th Percentile Parameters) Intrusion at 300 Years After Closure		
	CH-TRU Waste	RH-TRU Waste	CH-TRU and RH-TRU Waste	CH-TRU Waste	RH-TRU Waste	CH-TRU and RH-TRU Waste
Ac-227	1.70E-08	3.77E-06	2.55E-06	1.29E-08	3.65E-06	2.47E-06
Am-241	9.07E-01	3.53E-02	9.31E-01	1.93E+00	5.51E-02	1.97E+00
Cm-243	1.80E-11	2.18E-08	1.47E-08	4.08E-10	3.29E-07	2.22E-07
Cm-244	4.45E-09	5.53E-10	4.82E-09	4.07E-07	3.38E-08	4.30E-07
Cs-137	4.16E-06	1.30E-04	9.15E-05	7.97E-05	1.74E-03	1.25E-03
Pa-231	1.86E-08	4.09E-06	2.77E-06	1.45E-08	4.08E-06	2.76E-06
Pb-210	1.03E-06	6.14E-08	1.07E-06	1.56E-06	4.42E-08	1.59E-06
Pu-238	2.52E-01	3.94E-04	2.52E-01	2.19E-01	1.15E-03	2.20E-01
Pu-239	1.41E+00	6.28E-02	1.45E+00	1.31E+00	8.38E-02	1.37E+00
Pu-240	5.27E-05	6.57E-06	5.72E-05	1.06E-04	8.83E-06	1.12E-04
Pu-241	4.02E-08	3.79E-09	4.27E-08	5.00E-06	6.21E-07	5.42E-06
Sr-90	2.89E-06	9.56E-05	6.72E-05	5.93E-05	1.37E-03	9.84E-04
U-232	1.11E-05	9.22E-08	1.12E-05	5.83E-05	3.21E-07	5.85E-05
U-233	1.57E-02	2.20E-04	1.59E-02	3.13E-02	2.93E-04	3.15E-02
U-234	2.14E-03	2.45E-04	2.30E-03	8.80E-04	3.26E-04	1.10E-03
Y90	2.89E-06	9.57E-05	6.72E-05	5.93E-05	1.38E-03	9.84E-04
Total	2.58	0.099	2.65	3.49	0.145	3.59

Table H-31
Releases (kilograms) of Heavy Metals
to the Ground Surface from Drilling Intrusions for Action Alternative 1

Heavy Metal	Case 7 (Median Parameters) Intrusion at 400 Years After Closure			Case 9 (75th Percentile Parameters) Intrusion at 300 Years After Closures		
	CH-TRU Waste	RH-TRU Waste	CH-TRU and RH-TRU Waste	CH-TRU Waste	RH-TRU Waste	CH-TRU and RH-TRU Waste
Lead	4.15	22.23	20.98	5.52E+00	2.96E+01	2.79E+01
Beryllium	0.0974	0.0058	0.1013	1.30E-01	7.72E-03	1.35E-01
Cadmium	0.0015	0.0001	0.0016	2.00E-03	1.10E-04	2.08E-03
Mercury	1.7784	0.0984	1.8445	2.37E+00	1.31E-01	2.45E+00
Total	6.03	22.33	22.93	8.02	29.70	30.50

Table H-32
Radiation Dose to a Member of the Drilling Crew for a Drilling Intrusion for Action Alternative 1

Case 7 (Median Parameters) Intrusion at 400 Years After Closure				Case 9 (75th Percentile Parameters) Intrusion at 300 Years After Closure			
Radionuclide	Ingestion Dose (rem)	External Dose (rem)	Total Dose (rem)	Radionuclide	Ingestion Dose (rem)	External Dose (rem)	Total Dose (rem)
CH-TRU Waste							
Am-241	3.1E-01	9.8E-03	3.1E-01	Am-241	6.9E-01	2.2E-02	7.1E-01
Pu-239	6.5E-02	1.5E-03	6.6E-02	Pu-239	6.0E-03	1.4E-04	6.1E-03
Pu-238	1.1E-03	1.2E-05	1.2E-03	Pu-238	1.1E-03	1.1E-05	1.1E-03
U-233	3.7E-05	5.7E-06	4.2E-05	Cs-137	3.6E-07	1.2E-04	1.2E-04
Cs-137	1.8E-08	6.1E-06	6.1E-06	U-233	7.9E-05	1.2E-05	9.0E-05
U-234	5.2E-06	3.0E-07	5.5E-06	U-234	2.1E-06	1.3E-07	2.3E-06
Pb-210	5.0E-06	7.6E-09	5.0E-06	Y-90	6.1E-08	9.2E-07	9.8E-07
Ac-227	2.3E-06	3.4E-10	2.3E-06	Pb-210	7.5E-07	1.2E-09	7.5E-07
Pa-231	1.8E-06	1.3E-07	1.9E-06	Sr-90	6.8E-07	1.6E-08	7.0E-07
Pu-240	2.5E-07	2.5E-09	2.5E-07	Pu-240	5.0E-07	5.0E-09	5.0E-07
U-232	6.8E-08	2.5E-09	7.0E-08	U-232	3.7E-07	1.4E-08	3.9E-07
Y-90	2.9E-09	4.4E-08	4.7E-08	Cm-244	7.6E-08	1.6E-11	7.6E-08
Sr-90	3.3E-08	7.8E-10	3.4E-08	Ac-227	1.7E-08	2.6E-12	1.7E-08
Cm-243	4.3E-10	4.1E-10	8.4E-10	Pa-231	1.5E-08	1.1E-09	1.6E-08
Cm-244	8.4E-11	1.8E-14	8.4E-11	Pu-241	3.6E-10	4.4E-16	3.6E-10
Pu-241	2.9E-14	3.6E-20	2.9E-14	Cm-243	9.5E-11	9.1E-11	1.9E-10
Total Dose (rem)	3.8E-01	1.1E-02	3.8E-01	**Total Dose (rem)**	7.0E-01	2.2E-02	7.2E-01
Probability of LCF	1.9E-04	5.7E-06	1.9E-04	**Probability of LCF**	3.5E-04	1.1E-05	3.6E-04
RH-TRU Waste							
Am-241	1.2E-02	4.0E-04	1.3E-02	Am-241	1.9E-02	6.1E-04	2.0E-02
Pu-239	2.9E-04	6.7E-06	3.0E-04	Cs-137	8.1E-06	2.6E-03	2.6E-03
Cs-137	5.7E-07	1.9E-04	1.9E-04	Pu-239	3.9E-04	9.0E-06	4.0E-04
Ac-227	5.1E-06	7.5E-10	5.1E-06	Y-90	1.4E-06	2.2E-05	2.3E-05
Pa-231	4.0E-06	2.9E-07	4.3E-06	Sr-90	1.6E-05	3.9E-07	1.6E-05
Pu-238	1.8E-06	1.9E-08	1.8E-06	Pu-238	5.3E-06	6.0E-08	5.3E-06
Y-90	9.7E-08	1.5E-06	1.6E-06	Ac-227	4.8E-06	7.2E-10	4.8E-06
Sr-90	1.1E-06	2.6E-08	1.1E-06	Pa-231	4.0E-06	2.9E-07	4.3E-06
U-233	5.8E-07	8.5E-08	6.6E-07	U-234	8.0E-07	4.7E-08	8.5E-07
U-234	5.9E-07	3.5E-08	6.3E-07	U-233	7.3E-07	1.1E-07	8.4E-07
Pb-210	3.0E-08	4.9E-11	3.1E-08	Cm-243	7.8E-08	7.3E-08	1.5E-07
Pu-240	3.1E-08	3.2E-10	3.1E-08	Pu-240	4.1E-08	4.3E-10	4.2E-08
Cm-243	5.5E-09	5.2E-09	1.1E-08	Pb-210	2.2E-08	3.5E-11	2.2E-08
U-232	5.8E-10	2.2E-11	6.1E-10	Cm-244	6.6E-09	1.4E-12	6.6E-09
Cm-244	1.1E-10	2.2E-14	1.1E-10	U-232	2.0E-09	7.5E-11	2.1E-09
Pu-241	2.7E-13	3.3E-19	2.7E-13	Pu-241	4.4E-11	5.5E-17	4.4E-11
Total Dose (rem)	1.2E-02	6.0E-04	1.4E-02	**Total Dose (rem)**	1.9E-02	3.2E-03	2.3E-02
Probability of LCF	6.2E-06	3.0E-07	6.8E-06	**Probability of LCF**	9.7E-06	1.6E-06	1.2E-05
CH-TRU and RH-TRU Waste							
Am-241	3.2E-01	1.0E-02	3.3E-01	Am-241	6.9E-01	2.2E-02	7.1E-01
Pu-239	7.0E-03	1.6E-04	7.1E-03	Pu-239	6.5E-03	1.5E-04	6.6E-03
Pu-238	1.1E-03	1.2E-05	1.2E-03	Cs-137	5.2E-06	1.8E-03	1.8E-03
Cs-137	4.1E-07	1.3E-04	1.3E-04	Pu-238	1.1E-03	1.1E-05	1.1E-03
U-233	4.2E-05	6.4E-06	4.8E-05	U-233	7.9E-05	1.2E-05	9.0E-05
U-234	5.7E-06	3.3E-07	6.0E-06	Y-90	1.0E-06	1.5E-05	1.6E-05
Ac.-227	3.4E-06	5.1E-10	3.4E-06	Sr-90	1.1E-05	2.6E-07	1.2E-05
Pa-231	2.7E-06	1.9E-07	2.9E-06	Ac-227	3.2E-06	4.7E-10	3.2E-06
Y-90	6.9E-08	1.0E-06	1.1E-06	Pa-231	2.7E-06	1.9E-07	2.9E-06
Sr-90	7.8E-07	1.9E-08	7.9E-07	U-234	2.6E-06	1.5E-07	2.7E-06
Pb-210	5.3E-07	8.4E-10	5.4E-07	Pb-210	8.6E-07	1.4E-09	8.6E-07
Pu-240	2.6E-07	2.7E-09	2.7E-07	Pu-240	5.0E-07	5.0E-09	5.0E-07
U-232	6.8E-08	2.5E-09	7.0E-08	U-232	3.7E-07	1.4E-08	3.9E-07
Cm-243	3.5E-09	3.5E-09	7.0E-09	Cm-243	5.3E-08	5.0E-08	1.0E-07
Cm-244	9.0E-10	1.9E-13	9.0E-10	Cm-244	8.2E-08	1.7E-11	8.2E-08
Pu-241	3.0E-12	3.8E-18	3.0E-12	Pu-241	3.9E-10	4.8E-16	3.9E-10
Total Dose (rem)	3.3E-01	1.0E-02	3.4E-01	**Total Dose (rem)**	7.0E-01	2.4E-02	7.2E-01
Probability of LCF	1.6E-04	5.2E-06	1.7E-04	**Probability of LCF**	3.5E-04	1.2E-05	3.6E-04

Table H-33
Carcinogenic and Noncarcinogenic Impacts from Ingestion of Metals for Drilling Intrusion into the Repository for Action Alternative 1

Heavy Metal	Case 7 (Median Parameters) Probability of Cancer Incidence	Case 7 (Median Parameters) Hazard Index	Case 9 (75th Percentile Parameters) Probability of Cancer Incidence	Case 9 (75th Percentile Parameters) Hazard Index
CH-TRU Waste				
Beryllium	1.7E-08	8.0E-07	1.7E-08	8.0E-07
Cadmium	3.9E-10	1.2E-07	3.9E-10	1.2E-07
Lead	0	1.2E-04	0	1.2E-04
Mercury	0	2.4E-04	0	2.4E-04
RH-TRU Waste				
Beryllium	1.0E-09	4.8E-08	1.0E-09	4.8E-08
Cadmium	2.1E-11	6.8E-09	2.1E-11	6.8E-09
Lead	0	6.5E-04	0	6.5E-04
Mercury	0	1.4E-05	0	1.3E-05
CH-TRU and RH-TRU Waste				
Beryllium	1.8E-08	8.3E-07	1.8E-08	8.3E-07
Cadmium	4.0E-10	1.3E-07	4.0E-10	1.3E-07
Lead	0	6.2E-04	0	6.1E-04
Mercury	0	2.5E-04	0	2.5E-04

Table H-34
Radiation Dose to the Well-Site Geologist from Drilling Intrusions for Action Alternative 1

Case 7 (Median Parameters) Intrusion at 400 Years After Closure				Case 9 (75th Percentile Parameters) Intrusion at 300 Years After Closure			
CH-TRU Waste		RH-TRU Waste		CH-TRU Waste		RH-TRU Waste	
Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)
Am-241	3.2E-06	Am-241	1.3E-06	Am-241	3.7E-06	Sr-90	4.7E-06
U-234	1.5E-06	U-234	9.6E-07	U-234	1.5E-06	Cs-137	2.6E-06
Pu-239	6.1E-08	Sr-90	4.4E-07	Pu-239	6.1E-08	Am-241	1.5E-06
U-233	5.2E-08	Cs-137	2.6E-07	U-233	5.2E-08	U-234	9.6E-07
Pu-240	1.1E-08	Pu-239	1.6E-08	Pu-238	1.5E-08	Pu-239	1.6E-08
Ac-225	8.8E-09	U-233	1.2E-08	Pu-240	1.1E-08	U-233	1.2E-08
Pu-238	6.6E-09	Pu-240	6.7E-09	Sr-90	8.0E-09	Pu-240	6.7E-09
Sr-90	7.2E-10	Ac-225	8.8E-10	Ac-225	6.7E-09	Y-90	6.8E-10
Cs-137	3.1E-10	Pa-231	1.2E-10	Cs-137	3.3E-09	Ac-225	6.4E-10
Pb-210	7.5E-12	Pu-238	8.9E-11	Pb-210	7.7E-12	Pu-238	2.1E-10
Pa-231	1.1E-12	Y-90	6.4E-11	Y-90	1.2E-12	Pa-231	8.7E-11
U-232	1.2E-13	U-232	2.9E-14	Pa-231	8.2E-13	U-232	7.4E-14
Y-90	1.0E-13	Pb-210	2.7E-18	U-232	3.1E-13	Pb-210	1.1E-18
Pu-241	1.9E-17	Pu-241	1.8E-21	Pu-241	1.9E-17	Pu-241	2.2E-19
Total Dose (rem)	4.8E-06	**Total Dose (rem)**	3.0E-06	**Total Dose (rem)**	5.4E-06	**Total Dose (rem)**	9.8E-06
Probability of LCF	2.4E-09	**Probability of LCF**	1.5E-09	**Probability of LCF**	2.7E-09	**Probability of LCF**	4.9E-09

migration down to the brine would occur as the initial pressure in the reservoir dissipates and equilibrates with pressures in the repository and the surrounding units penetrated by the borehole.

Radionuclide migration for Case 9, shown in Figure H-13, resulted in greater vertical upward migration than Case 7 and a release of contaminants into the Culebra Dolomite. The estimated maximum release rate of 0.99 Ci per year (predominantly Pu-239, Am-241, and Pu-240) and 50.3 kilograms (111 pounds) per year of heavy metals (predominantly lead) would occur at approximately 1,600 years after closure. The estimated total release to the Culebra Dolomite would be 1,090 Ci and 60,250 kilograms (133,000 pounds) of heavy metals, with the release occurring from 600 years to 3,900 years after closure. Analyses of releases to the Culebra showed that contaminants would be highly sorbed and only small amounts would migrate from the point of intrusion over 10,000 years. Transport calculations for Case 9 releases to the Culebra were made with the SECOTP2D model of the Culebra developed for the CCA (DOE 1996f).

Radiological impacts were calculated to an individual, such as a rancher, who consumed beef from cattle that had consumed water from a stock well located downgradient at the Land Withdrawal Area boundary 3 kilometers (2 miles) from the point of intrusion. Results are presented only for partial mining conditions, since calculated impacts for full mining conditions were significantly lower (see Section H.8.2). Radiological impacts would be negligible, with a 2×10^{-27} probability of an LCF. The principal dose contributor was Pu-239 with the 10 other radionuclides considered no more than 0.1 percent of Pu-239. Ingestion of heavy metals from consumption of contaminated beef, was also found to result in negligible impacts, with a probability of cancer incidence of 3×10^{-27}.

H.8.4 Results for Action Alternative 2

Long-term performance assessment analyses were conducted for Action Alternative 2 using the same methods as were used for other alternatives. The radionuclide and heavy metal inventories were increased slightly over those of Action Alternative 1 to account for the small additional volume of polychlorinated biphenyl (PCB)-commingled TRU waste (see Chapter 3 and Appendix A). Total radionuclide activities of 7.3×10^{6} Ci in CH-TRU waste and 5.1×10^{6} Ci in RH-TRU waste were included for this alternative. Detailed information on the radionuclide and heavy metal inventories is provided in Appendix A. Only radiological impacts would result from disposal operations at WIPP because VOCs would be removed by thermal treatment. The repository size was adjusted from 10 panels (for the Proposed Action) to 75 panels. The material properties outside the repository were not changed from the comparable cases analyzed for the Proposed Action and other alternatives.

The four cases below were analyzed for Action Alternative 2. The cases considered the following conditions:

- Case 11 considered undisturbed repository performance. Median parameter values were used for all input variables where probability distributions had been defined.

- Case 12 considered disturbed conditions where a borehole from exploratory drilling is assumed to breach the repository and penetrate a pressurized brine pocket in the Castile Formation. Median parameter values were used.

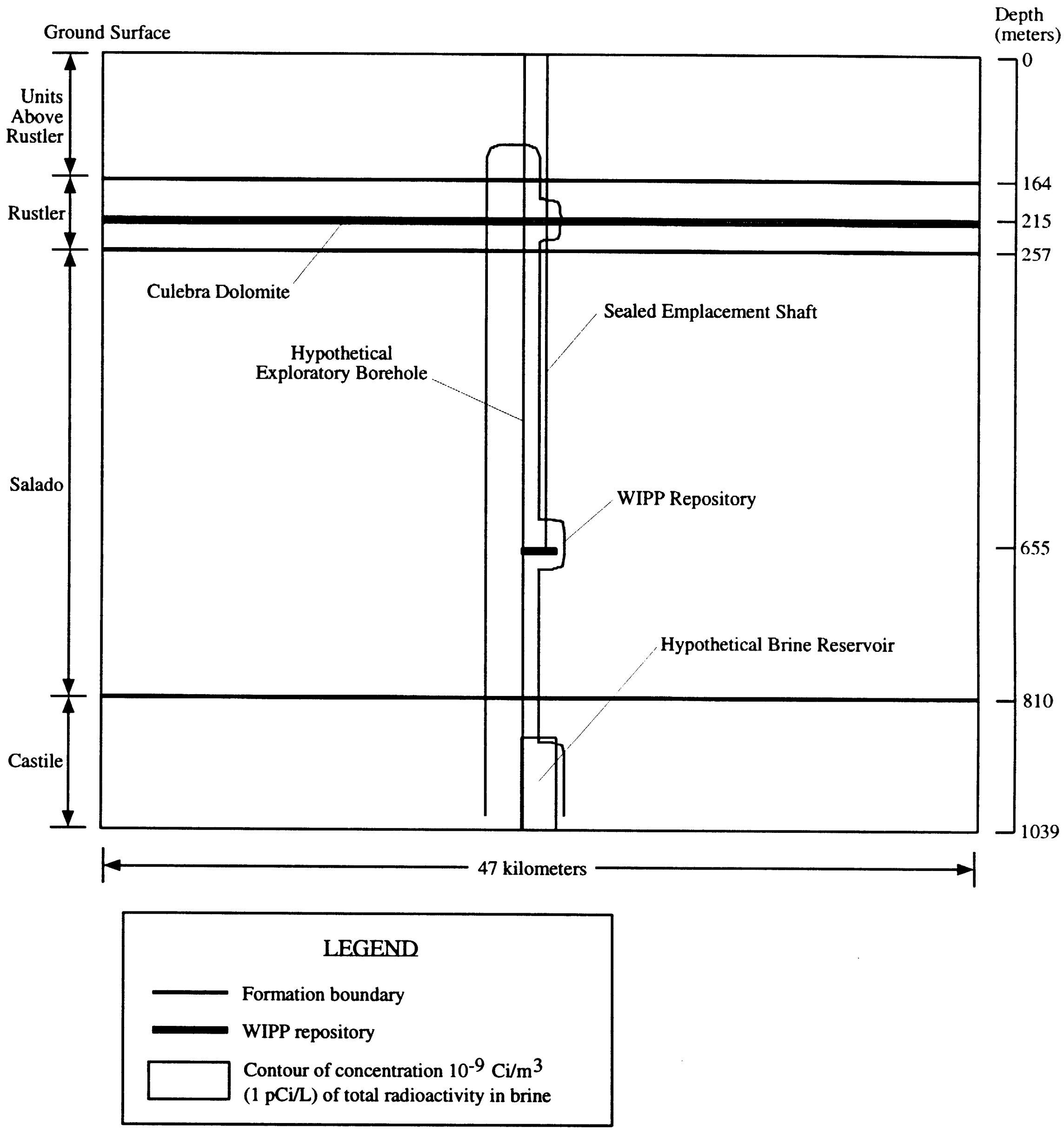


Figure H-13
Extent of Radionuclide Migration at 10,000 Years with Disturbed Conditions Using 75th Percentile Parameter Values (Case 9) for Action Alternative 1

- Case 13 considered undisturbed repository performance. Seventy-fifth percentile parameter values were used for all input variables where probability distributions had been defined.

- Case 14 considered the same disturbed conditions as Case 12. Seventy-fifth percentile parameter values were used.

Cases 11 through 14 were simulated using the BRAGFLO code to produce brine and gas pressure fields and flow velocity fields for use in subsequent transport and direct-release calculations. Brine pressures at each time step simulated by BRAGFLO were extracted and averaged, with respect to volume, for a single panel. The average pressures show the time evolution of brine pressure predicted for the repository. The resulting pressure curves, as a function of time after closure, are depicted in Figure H-14. The pressure release of the waste panel as a result of the exploratory drilling event at 400 years after closure for Case 12 and 300 years after closure for Case 14 is clearly evident in this figure. The secondary perturbations in the pressure curves (approximately 700 years and 1,700 years after closure) in the disturbed cases are due to the material property changes imposed in BRAGFLO that account for the degradation of the exploratory borehole concrete plugs (200 years following intrusion) and backfill (1,200 years following intrusion). Brine pressures in the panel remain below lithostatic conditions at 10,000 years after closure for all cases. The notable change for this alternative relative to all the other alternatives was that the thermal processing was assumed to destroy all of the biodegradable material that contributes to gas generation. The iron corrosion was assumed to still occur. The iron inventory is not completely consumed in the gas generation process for any of the cases (disturbed or undisturbed), and a substantial amount of iron remained at 10,000 years after closure. The gas generation for the undisturbed cases is brine-limited, and the corrosion proceeds slowly enough that not all of the iron is consumed.

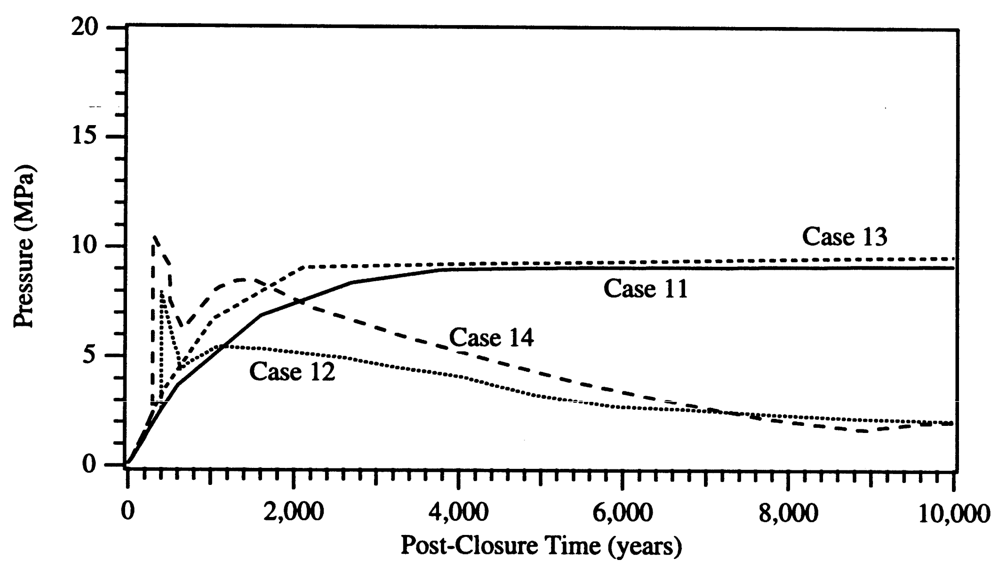


Figure H-14
Volume-Averaged Brine Pressures in a Waste Panel for Action Alternative 2

H.8.4.1 Impacts of Undisturbed Conditions

The extent of radionuclide migration for undisturbed conditions at 10,000 years after closure for Case 13 (using 75th percentile parameters) is presented in Figure H-14. Case 13 is presented because it resulted in slightly more extensive migration than Case 11. Simulation results showed similar migration to that presented for Case 8 of Action Alternative 1. Vertical migration of total radionuclide concentrations of one pCi per liter was estimated at about 40 meters (130 feet) above the top of the repository and about 20 meters (65 feet) below the bottom of the repository. Lateral migration at these concentration levels extended 1,900 meters (6,200 feet) from the repository edge. The extent of total heavy metal migration to a concentration of one part per billion was approximately the same as the one pCi per liter total radionuclide level (see Figure H-15).

H.8.4.2 Impacts of Disturbed Conditions

This section presents the impacts of two exposure scenarios evaluated for disturbed conditions of Action Alternative 2.

Surface Release Caused by Drilling into the Repository

Under Action Alternative 2, the estimated release to the ground surface from a drilling intrusion depends on the kind of waste panel intercepted by the drilling equipment. The predicted release would be 0.8 Ci from CH-TRU waste and 0.02 Ci from RH-TRU waste using the median parameter values of Case 12. Using the 75th percentile parameter values of Case 14, the predicted release would be 1.0 and 0.02 curies from CH-TRU waste and RH-TRU waste, respectively. Releases were mainly of Am-241, Pu-239, and Pu-238. Heavy metal releases from an intrusion were estimated at 1.9 and 3.4 kilograms (4.2 and 7.5 pounds) from CH-TRU waste and RH-TRU waste, respectively, using median parameter values of Case 12 and 2.3 and 4.2 kilograms (5.1 and 9.2 pounds) using 75th percentile parameter values of Case 14. Results of these analyses are presented in Tables H-35 and H-36.

Radiological impacts to the drilling crew member from a drilling intrusion under Case 12 (median parameter values) resulted in a 5.1×10^{-5} and 9.9×10^{-7} probability of an LCF for CH-TRU and RH-TRU waste, respectively. For Case 14 (75th percentile parameter values), the impacts were higher, with a 1.0×10^{-4} and 1.6×10^{-6} probability of an LCF for CH-TRU and RH-TRU waste, respectively. The dominant exposure pathway was by ingestion of drill cuttings, with Am-241 contributing 87 to 99 percent of the total dose. Results are presented in Table H-37. Ingestion of heavy metals for this scenario would result in a 3.2×10^{-8} and 8.5×10^{-10} probability of cancer incidence for intrusions in CH-TRU and RH-TRU waste, respectively. There would be no noncarcinogenic impacts expected from ingestion of the metals because all hazard indices would be much less than one. Impacts from ingestion of heavy metals are shown in Table H-38.

Radiological impacts to the well-site geologist for Case 12 (median parameter values) from external radiation would result in a 6.3×10^{-9} probability of an LCF for CH-TRU waste and a 4.2×10^{-9} probability of an LCF for RH-TRU waste. For Case 14 (75th percentile parameter values), the radiological impacts would be a 7.0×10^{-9} probability of an LCF for CH-TRU waste and a 1.4×10^{-8} probability of an LCF for RH-TRU waste. Results of these analyses are presented in Table H-39.

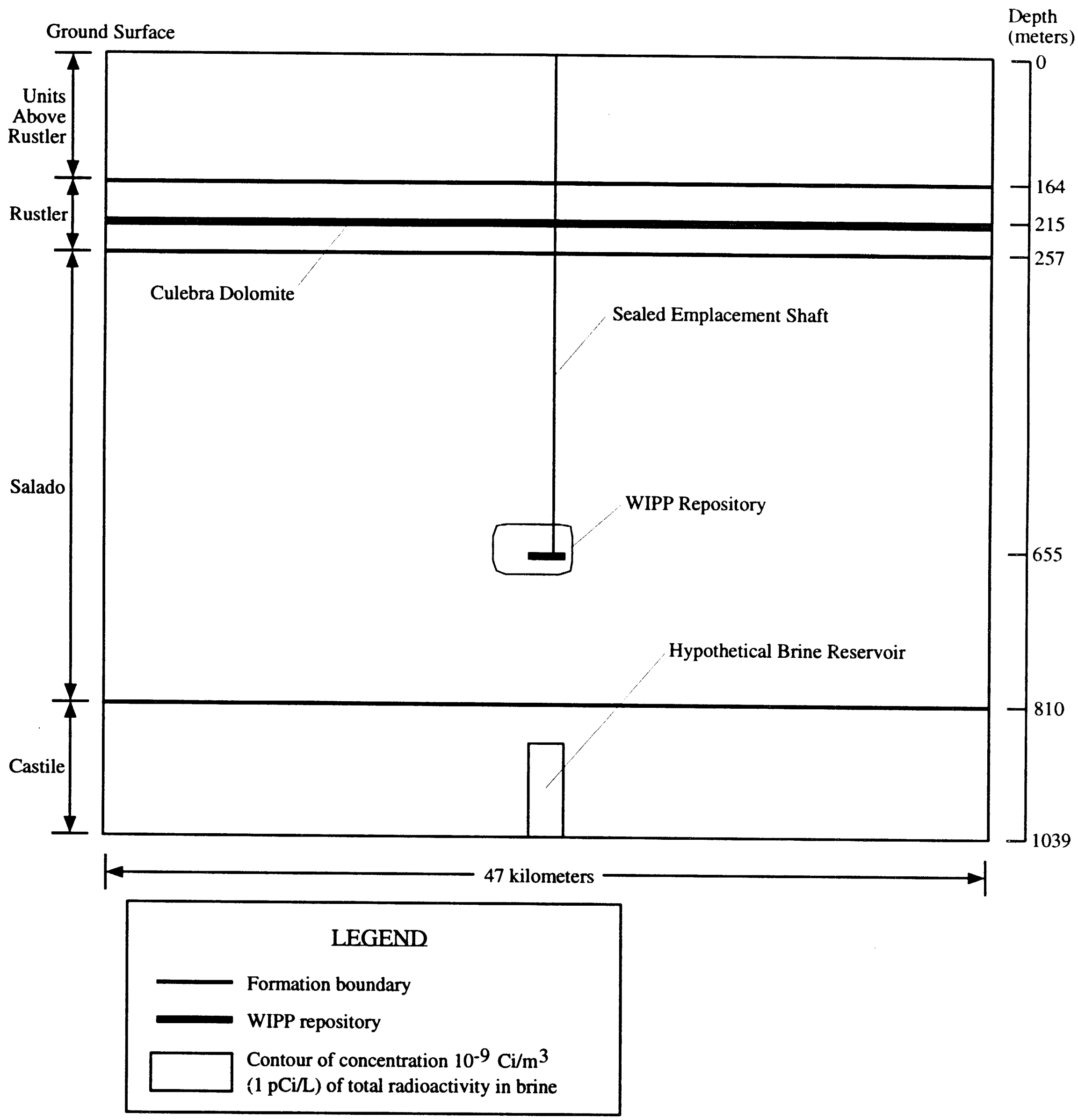


Figure H-15
Extent of Radionuclide Migration at 10,000 Years with Undisturbed Conditions Using 75th Percentile Parameter Values (Case 13) for Action Alternative 2

Table H-35
Radionuclide Releases (curies) to the Ground Surface from Drilling Intrusions for Action Alternative 2

	Case 12 (Median Parameters) Intrusion at 400 Years After Closure		Case 14 (75th Percentile Parameters) Intrusion at 300 Years After Closure	
Radionuclide	**CH-TRU Waste**	**RH-TRU Waste**	**CH-TRU Waste**	**RH-TRU Waste**
Ac-227	5.28E-09	5.70E-07	3.75E-09	5.18E-07
Am-241	2.81E-01	5.34E-03	5.61E-01	7.81E-03
Cm-243	5.57E-12	3.29E-09	1.19E-10	4.67E-08
Cm-244	1.38E-09	8.37E-11	1.18E-07	4.80E-09
Cs-137	1.29E-06	1.96E-05	2.32E-05	2.47E-04
Pa-231	5.78E-09	6.18E-07	4.22E-09	5.79E-07
Pb-210	3.20E-07	9.29E-09	4.53E-07	6.26E-09
Pu-238	7.82E-02	5.96E-05	6.38E-02	1.64E-04
Pu-239	4.36E-01	9.51E-03	3.81E-01	1.19E-02
Pu-240	1.64E-05	9.93E-07	3.09E-05	1.25E-06
Pu-241	1.25E-08	5.73E-10	1.45E-06	8.80E-08
Sr-90	8.96E-07	1.45E-05	1.72E-05	1.95E-04
U-232	3.46E-06	1.40E-08	1.69E-05	4.56E-08
U-233	4.87E-03	3.33E-05	9.10E-03	4.15E-05
U-234	6.63E-04	3.71E-05	2.56E-04	4.62E-05
Y-90	8.96E-07	1.45E-05	1.73E-05	1.95E-04
Total	0.80	0.02	1.02	0.02

Table H-36
Releases (kilograms) of Heavy Metals to the Ground Surface from Drilling Intrusions for Action Alternative 2

	Case 12 (Median Parameters) Intrusion at 400 Years After Closure		Case 14 (75th Percentile Parameters) Intrusion at 300 Years After Closure	
Heavy Metal	**CH-TRU Waste**	**RH-TRU Waste**	**CH-TRU Waste**	**RH-TRU Waste**
Lead	1.29E+00	3.36E+00	1.60E+00	4.19E+00
Beryllium	3.25E-02	8.78E-04	4.06E-02	1.09E-03
Cadmium	4.67E-04	1.25E-05	5.82E-04	1.56E-05
Mercury	5.52E-01	1.49E-02	6.88E-01	1.86E-02
Total	1.87	3.38	2.33	4.21

Table H-37
Radiation Dose to a Member of the Drilling Crew from Drilling Intrusions for Action Alternative 2

Case 12 (Median Parameters) Intrusion at 400 Years After Closure				Case 14 (75th Percentile Parameters) Intrusion at 300 Years After Closure			
Radionuclide	Ingestion Dose (rem)	External Dose (rem)	Total Dose (rem)	Radionuclide	Ingestion Dose (rem)	External Dose (rem)	Total Dose (rem)
CH-TRU Waste							
Am-241	9.8E-02	3.1E-03	1.0E-01	Am-241	1.9E-01	6.1E-03	2.0E-01
Pu-239	2.0E-03	4.7E-05	2.1E-03	Pu-239	1.8E-03	4.1E-05	1.8E-03
Pu-238	3.5E-04	3.7E-06	3.5E-04	Pu-238	2.9E-04	3.1E-06	2.9E-04
U-233	1.2E-05	1.8E-06	1.4E-05	Cs-137	1.0E-07	3.3E-05	3.3E-05
Cs-137	5.7E-09	1.9E-06	1.9E-06	U-233	2.2E-05	3.4E-06	2.6E-05
U-234	1.6E-06	9.6E-08	1.7E-06	U-234	6.2E-07	3.7E-08	6.6E-07
Pb-210	1.6E-07	2.5E-10	1.6E-07	Y-90	1.8E-08	2.8E-07	3.0E-07
Pu-240	8.0E-08	8.3E-10	8.1E-08	Pb-210	2.2E-07	3.6E-10	2.2E-07
U-232	2.2E-08	8.5E-10	2.3E-08	Sr-90	1.9E-07	4.7E-09	2.0E-07
Y-90	9.1E-10	1.4E-08	1.5E-08	Pu-240	1.5E-07	1.5E-09	1.5E-07
Sr-90	1.0E-08	2.5E-10	1.0E-08	U-232	1.1E-07	4.1E-09	1.1E-07
Ac-227	7.0E-09	1.0E-12	7.0E-09	Cm-244	2.2E-08	4.8E-12	2.2E-08
Pa-231	5.9E-09	4.3E-10	6.3E-09	Ac-227	4.9E-09	7.4E-13	4.9E-09
Cm-244	2.6E-10	5.6E-14	2.6E-10	Pa-231	4.2E-09	3.0E-10	4.5E-09
Cm-243	1.3E-12	1.3E-12	2.6E-12	Pu-241	1.1E-10	1.4E-16	1.1E-10
Pu-241	8.4E-13	1.1E-18	8.4E-13	Cm-243	2.8E-11	2.8E-11	5.6E-11
Total Dose (rem)	1.0E-01	3.2E-03	1.0E-01	**Total Dose (rem)**	1.9E-01	6.2E-03	2.0E-01
Probability of LCF	5.0E-05	1.6E-06	5.1E-05	**Probability of LCF**	9.6E-05	3.1E-06	1.0E-04
RH-TRU Waste							
Am-241	1.9E-03	5.9E-05	1.9E-03	Am-241	2.7E-03	8.5E-05	2.7E-03
Pu-239	4.4E-05	1.0E-06	4.5E-05	Cs-137	1.1E-06	3.5E-04	3.5E-04
Cs-137	9.0E-08	2.9E-05	2.9E-05	Pu-239	5.5E-05	1.3E-06	5.6E-05
Ac-227	7.5E-07	1.1E-10	7.5E-07	Y-90	2.1E-07	3.1E-06	3.3E-06
Pa-231	6.2E-07	4.5E-08	6.6E-07	Sr-90	2.3E-06	5.5E-08	2.4E-06
Pu-238	2.7E-07	2.9E-09	2.7E-07	Pu-238	7.6E-07	8.3E-09	7.7E-07
Y-90	1.5E-08	2.3E-07	2.5E-07	Ac-227	6.9E-07	1.0E-10	6.9E-07
Sr-90	1.6E-07	3.9E-09	1.6E-07	Pa-231	5.9E-07	4.3E-08	6.3E-07
U-234	9.3E-08	5.5E-09	9.9E-08	U-234	1.1E-07	6.7E-09	1.2E-07
U-233	8.1E-08	1.2E-08	9.3E-08	U-233	1.0E-07	1.6E-08	1.2E-07
Pb-210	4.7E-09	7.4E-12	4.7E-09	Cm-243	1.1E-08	1.1E-08	2.2E-08
Pu-240	4.6E-09	4.6E-11	4.7E-09	Pu-240	5.5E-09	6.0E-11	5.5E-09
Cm-243	7.8E-10	7.3E-10	1.5E-09	Pb-210	3.2E-09	5.0E-12	3.2E-09
U-232	8.8E-11	3.4E-12	9.2E-11	Cm-244	9.0E-10	1.9E-13	9.0E-10
Cm-244	1.6E-11	3.3E-15	1.6E-11	U-232	2.9E-10	1.1E-11	3.0E-10
Pu-241	4.0E-14	5.0E-20	4.0E-14	Pu-241	6.3E-12	7.9E-18	6.3E-12
Total Dose (rem)	1.9E-03	8.9E-05	2.0E-03	**Total Dose (rem)**	2.8E-03	4.4E-04	3.1E-03
Probability of LCF	9.7E-07	4.5E-08	9.9E-07	**Probability of LCF**	1.4E-06	2.2E-07	1.6E-06

Table H-38
Carcinogenic and Noncarcinogenic Impacts from Ingestion of Metals from Drilling Intrusion into the Repository for Action Alternative 2

Heavy Metals	Case 12 (Median Parameters)		Case 14 (75th Percentile Parameters)	
	Probability of Cancer Incidence	Hazard Index	Probability of Cancer Incidence	Hazard Index
CH-TRU Waste				
Beryllium	3.2E-08	1.5E-06	3.2E-08	1.5E-06
Cadmium	6.7E-10	2.1E-07	6.7E-10	2.1E-07
Lead	0	2.1E-04	0	2.1E-04
Mercury	0	4.2E-04	0	4.2E-04
RH-TRU Waste				
Beryllium	8.5E-10	4.0E-08	8.5E-10	4.0E-08
Cadmium	1.8E-11	5.7E-09	1.8E-11	5.7E-09
Lead	0	5.4E-04	0	5.4E-04
Mercury	0	1.1E-05	0	1.1E-05

Table H-39
Radiation Dose to the Well-Site Geologist from Drilling Intrusions for Action Alternative 2

Case 12 (Median Parameters) Intrusion at 400 Years After Closure				Case 14 (75th Percentile Parameters) Intrusion at 300 Years After Closure			
CH-TRU Waste		RH-TRU Waste		CH-TRU Waste		RH-TRU Waste	
Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)
Am-241	8.2E-06	Am-241	3.7E-06	Am-241	9.5E-06	Sr-90	1.3E-05
U-234	4.0E-06	U-234	2.6E-06	U-234	4.0E-06	Cs-137	7.2E-06
Pu-239	1.6E-07	Sr-90	1.2E-06	Pu-239	1.6E-07	Am-241	4.1E-06
U-233	1.4E-07	Cs-137	7.2E-07	U-233	1.4E-07	U-234	2.6E-06
Pu-240	2.8E-08	Pu-239	4.8E-08	Pu-238	3.7E-08	Pu-239	4.8E-08
Ac-225	2.4E-08	U-233	3.2E-08	Pu-240	3.0E-08	U-233	3.2E-08
Pu-238	1.8E-08	Pu-240	1.9E-08	Sr-90	2.1E-08	Pu-240	2.0E-08
Sr-90	1.9E-09	Ac-225	2.5E-09	Ac-225	1.8E-08	Y-90	1.9E-09
Cs-137	8.4E-10	Pa-231	3.2E-10	Cs-137	8.4E-09	Ac-225	1.8E-09
Pb-210	2.0E-11	Pu-238	2.5E-10	Pb-210	2.0E-11	Pu-238	5.6E-10
Pa-231	2.9E-12	Y-90	1.8E-10	Y-90	3.0E-12	Pa-231	2.5E-10
U-232	3.3E-13	U-232	7.9E-14	Pa-231	2.2E-12	U-232	2.0E-13
Y-90	2.7E-13	Pb-210	7.7E-18	U-232	8.6E-13	Pb-210	3.0E-18
Pu-241	4.8E-17	Pu-241	5.1E-21	Pu-241	4.8E-17	Pu-241	6.3E-19
Total Dose (rem)	1.3E-05	**Total Dose (rem)**	8.3E-06	**Total Dose (rem)**	1.4E-05	**Total Dose (rem)**	2.7E-05
Probability of LCF	6.3E-09	**Probability of LCF**	4.2E-09	**Probability of LCF**	7.0E-09	**Probability of LCF**	1.4E-08

Drilling Through the Repository into a Pressurized Brine Reservoir

Radionuclide migration for disturbed conditions at 10,000 years after closure for Case 12 (median parameter values) and Case 14 (75th percentile parameter values) showed the migration of total radionuclide concentrations of one pCi per liter migrating upward and downward in the exploratory borehole. Heavy metal (predominantly lead) concentrations of one part per billion exhibited a similar pattern of migration. In both cases, contaminants at these levels penetrated a short distance into the surrounding rock and down into the Castile Formation and the hypothetical brine reservoir. The migration down to the brine would occur as the initial pressure in the brine reservoir dissipates and equilibrates with pressures in the repository and the surrounding units penetrated by the borehole.

Radionuclide migration for Case 14, shown in Figure H-16, resulted in a higher vertical upward migration than Case 12 and a release of contaminants into the Culebra Dolomite. The estimated maximum release rate of 2.5 Ci per year (predominantly Am-241, Pu-239 and Pu-240) and 49 kilograms (108 pounds) per year of heavy metals (predominantly lead) would occur at approximately 500 years after closure. The estimated total release to the Culebra Dolomite would be 286 Ci and 20,890 kilograms (46,050 pounds) of heavy metals, with the release occurring from 500 years to 4,500 years after closure. Analyses of releases to the Culebra showed that contaminants would be highly sorbed and only small amounts would migrate from the point of intrusion over 10,000 years. Transport calculations for Case 14 releases to the Culebra were made with the SECOTP2D model of the Culebra developed for the CCA (DOE 1996f).

Radiological impacts were calculated to an individual, such as a rancher, who consumed beef from cattle that had consumed water from a stock well located downgradient at the Land Withdrawal Area boundary 3 kilometers (2 miles) from the point of intrusion. Results are presented only for partial mining conditions, since calculated impacts for full mining conditions were significantly lower. Radiological impacts would be negligible, with a 7×10^{-28} probability of an LCF. The principal dose contributor was Pu-239, with the 10 other radionuclides contributing no more than 0.2 percent of Pu-239. Ingestion of heavy metals from consumption of contaminated beef, was also found to result in negligible impacts, with a probability of cancer incidence of 3×10^{-27}. No noncarcinogenic impacts would be expected.

H.8.5 Results for Action Alternative 3

Long-term performance assessment analyses were conducted for Action Alternative 3 using the same methods as were used for other alternatives. The radionuclide and heavy metal inventories were similar to those of Action Alternatives 1 and 2 (see Chapter 3 and Appendix A). Total radionuclide inventories of 7.3×10^{6} Ci in CH-TRU waste and 5.1×10^{6} Ci in RH-TRU waste were included. Detailed information on the radionuclide and heavy metal inventories is provided in Appendix A. The repository size was adjusted from 10 panels (for the Proposed Action) to 71 panels. The material properties outside the repository were not changed from the comparable cases analyzed for the Proposed Action and other alternatives.

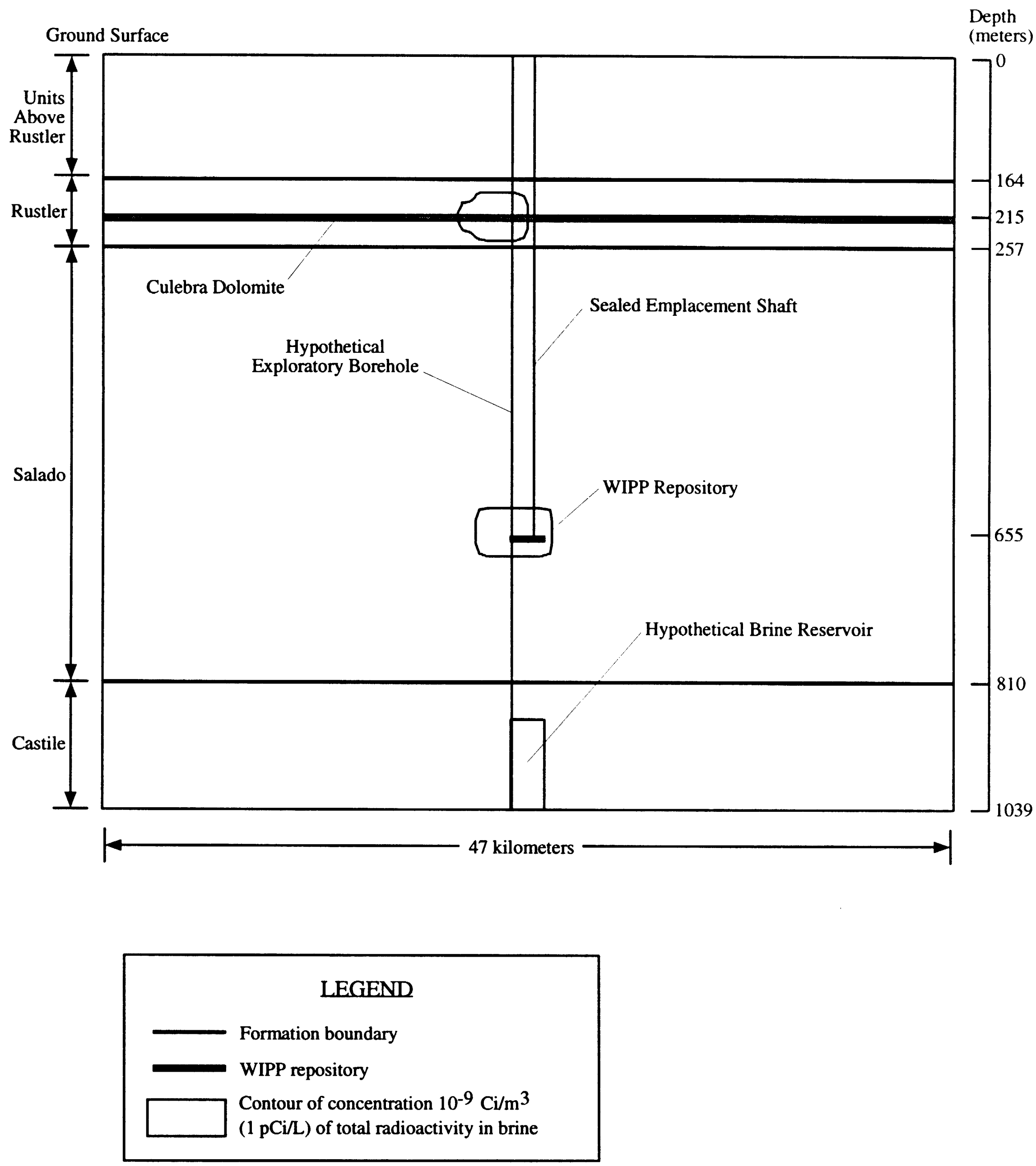


Figure H-16
Extent of Radionuclide Migration at 10,000 Years with Disturbed Conditions Using 75th Percentile Parameter Values (Case 14) for Action Alternative 2

The four cases below were analyzed for Action Alternative 3. The cases considered the following conditions:

- Case 16 considered undisturbed repository conditions. Median parameter values were used for all input variables where probability distributions have been defined.

- Case 17 considered an intrusion resulting from exploratory drilling. It was assumed that the repository would be penetrated and the drill would intercept a pressurized brine pocket in the Castile Formation. Median parameter values were used.

- Case 18 considered undisturbed repository conditions. Seventy-fifth percentile parameter values were used for all input variables where probability distributions had been defined.

- Case 19 considered the same disturbed conditions as Case 17. Seventy-fifth percentile parameter values were used.

Cases 16 through 19 were simulated in BRAGFLO to produce brine and gas pressure fields and flow velocity fields for use in subsequent transport and direct-release calculations. Brine pressures at each time step simulated by BRAGFLO were extracted and averaged, with respect to volume, for a single panel. The average pressures show the time evolution of brine pressure predicted for the repository. The resulting pressure curves, as a function of time after closure, are depicted in Figure H-17. The pressure release of the waste panel as a result of the exploratory drilling event at 400 years after closure for Case 17 and 300 years after closure for Case 19 is clearly evident in this figure. Secondary pertubations in the pressure curves at approximately 700 and 1,700 years after closure are due to material property changes imposed in the BRAGFLO simulation, to account for degradation of the exploratory borehole concrete plug (200 years following intrusion) and the degradation of the borehole backfill (1,200 years after intrusion). Brine pressures in the

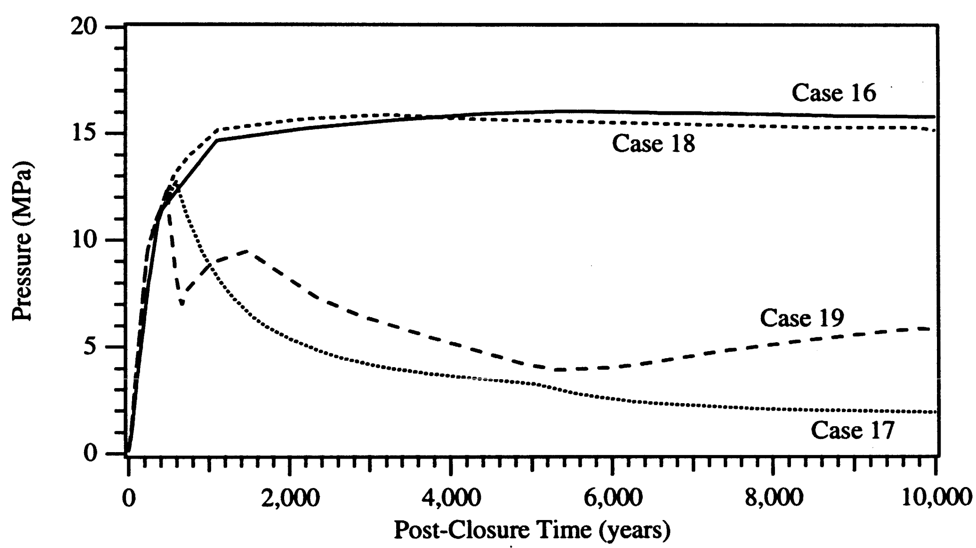


Figure H-17
Volume-Averaged Brine Pressures in a Waste Panel for Action Alternative 3

panel remain below lithostatic conditions at 10,000 years after closure for the intrusion cases. The biodegradable material is completely consumed in the gas generation process for all four cases. The iron inventory is completely consumed in the gas generation process for both of the intrusion cases. The gas generation for the undisturbed cases is brine-limited, and the corrosion proceeds slowly enough that not all of the iron is consumed.

H.8.5.1 Impacts of Undisturbed Conditions

The extent of radionuclide and heavy metal (predominantly lead) migration for undisturbed conditions at 10,000 years after closure for Case 18 (using 75th percentile parameter values) was very similar to that shown for Case 8 of Action Alternative 1. Migration of total radionuclide concentrations of one pCi per liter extend about 20 meters (65 feet) below the repository. The maximum distance of lateral migration at the same concentration level was about 1,900 meters (6,200 feet). The maximum extent of vertical upward migration was calculated to be about 40 meters (130 feet) above the top of the repository. The extent of total heavy metal migration to a concentration to one part per billion was also approximately the same as the one pCi per liter total radionuclide level calculated for Case 8 of Action Alternative 1.

H.8.5.2 Impacts of Disturbed Conditions

This section presents the impacts of two exposure scenarios evaluated for disturbed conditions of Action Alternative 3.

Surface Release Caused by Drilling into the Repository

For Action Alternative 3, the release of radionuclides to the ground surface from a drilling intrusion for Case 17 (median parameter values) would be 2.2, 0.1, and 2.3 Ci, respectively, for intrusions into a panel containing CH-TRU waste, RH-TRU waste, and both CH-TRU and RH-TRU waste. For Case 19 (75th percentile parameter values), the radionuclide release would be 3, 0.15, and 3 Ci for the same respective intrusions. Radionuclides released were mainly Am-241, Pu-239, and Pu-238. Under Case 17, heavy metal releases from an intrusion were estimated at 5, 22, and 18 kilograms (11, 48, and 39 pounds), respectively. Under Case 19, the heavy metal releases were estimated at 7, 30, and 23 kilograms (15, 66, and 51 pounds) for the same respective intrusions. Results of these analyses are presented in Tables H-40 and H-41.

Radiological impacts to the drilling crew member from the drilling intrusion for Case 17 (median parameter values) would result in a 1.4×10^{-4}, 6.8×10^{-6}, and 1.4×10^{-4} probability of an LCF, respectively, for intrusions into CH-TRU waste, RH-TRU waste, and both CH-TRU and RH-TRU waste. For Case 19 (75th percentile parameter values), the impacts would be higher, at a 2.8×10^{-4}, 1.2×10^{-5}, and 2.8×10^{-4} probability of an LCF for the same respective intrusions. The dominant exposure pathway was by ingestion of drill cuttings, with Am-241 contributing 87 to 98 percent of the total dose. Results are presented in Table H-42.

Table H-40
Radionuclide Releases (curies)
to the Ground Surface from Drilling Intrusions for Action Alternative 3

Radionuclide	Case 17 (Median Parameters) Intrusion at 400 Years After Closure			Case 19 (75th Percentile Parameters) Intrusion at 300 Years After Closure		
	CH-TRU Waste	RH-TRU Waste	CH-TRU and RH-TRU Waste	CH-TRU Waste	RH-TRU Waste	CH-TRU and RH-TRU Waste
Ac-227	1.45E-08	3.77E-06	2.12E-06	1.09E-08	3.64E-06	2.05E-06
Am-241	7.72E-01	3.53E-02	7.91E-01	1.64E+00	5.49E-02	1.67E+00
Cm-243	1.53E-11	2.18E-08	1.22E-08	3.46E-10	3.28E-07	1.84E-07
Cm-244	3.78E-09	5.54E-10	4.09E-09	3.45E-07	3.37E-08	3.64E-07
Cs-137	3.53E-06	1.30E-04	7.61E-05	6.76E-05	1.74E-03	1.04E-03
Pa-231	1.58E-08	4.09E-06	2.30E-06	1.23E-08	4.07E-06	2.29E-06
Pb-210	8.77E-07	6.14E-08	9.11E-07	1.32E-06	4.41E-08	1.35E-06
Pu-238	2.14E-01	3.94E-04	2.15E-01	1.86E-01	1.15E-03	1.87E-01
Pu-239	1.20E+00	6.28E-02	1.23E+00	1.11E+00	8.36E-02	1.16E+00
Pu-240	4.48E-05	6.57E-06	4.85E-05	9.01E-05	8.80E-06	9.51E-05
Pu-241	3.41E-08	3.79E-09	3.63E-08	4.24E-06	6.19E-07	4.58E-06
Sr-90	2.46E-06	9.56E-05	5.59E-05	5.03E-05	1.37E-03	8.16E-04
U-232	9.48E-06	9.22E-08	9.53E-06	4.94E-05	3.20E-07	4.96E-05
U-233	1.34E-02	2.20E-04	1.35E-02	2.66E-02	2.92E-04	2.67E-02
U-234	1.82E-03	2.45E-04	1.95E-03	7.46E-04	3.25E-04	9.28E-04
Y-90	2.46E-06	9.57E-05	5.59E-05	5.03E-05	1.37E-03	8.17E-04
Total	2.20	0.099	2.25	2.96	0.145	3.04

Table H-41
Releases (kilograms) of Heavy Metals
to the Ground Surface from Drilling Intrusions for Action Alternative 3

Heavy Metal	Case 17 (Median Parameters) Intrusion at 400 Years After Closure			Case 19 (75th Percentile Parameters) Intrusion at 300 Years After Closure		
	CH-TRU Waste	RH-TRU Waste	CH-TRU and RH-TRU Waste	CH-TRU Waste	RH-TRU Waste	CH-TRU and RH-TRU Waste
Lead	3.53E+00	2.22E+01	1.60E+01	4.68E+00	2.95E+01	2.12E+01
Beryllium	8.29E-02	5.80E-03	8.61E-02	1.10E-01	7.70E-03	1.14E-01
Cadmium	1.28E-03	8.28E-05	1.33E-03	1.70E-03	1.10E-04	1.76E-03
Mercury	1.51E+00	9.84E-02	1.57E+00	2.01E+00	1.31E-01	2.08E+00
Total	5.13	22.33	17.61	6.80	29.61	23.35

Table H-42
Radiation Dose to a Member of the Drilling Crew from Drilling Intrusions for Action Alternative 3

Case 17 (Median Parameters) Intrusion at 400 Years After Closure				Case 19 (75th Percentile Parameters) Intrusion at 300 Years After Closure			
Radionuclide	**Ingestion Dose (rem)**	**External Dose (rem)**	**Total Dose (rem)**	**Radionuclide**	**Ingestion Dose (rem)**	**External Dose (rem)**	**Total Dose (rem)**
CH-TRU Waste							
Am-241	2.6E-01	8.4E-03	2.7E-01	Am-241	5.4E-01	1.8E-02	5.6E-01
Pu-239	5.5E-03	1.3E-04	5.6E-03	Pu-239	5.0E-03	1.1E-04	5.1E-03
Pu-238	9.6E-04	1.0E-05	9.7E-04	Pu-238	8.1E-04	8.8E-06	8.2E-04
U-233	3.1E-05	5.0E-06	3.6E-05	Cs-137	3.0E-07	9.9E-05	9.9E-05
Cs-137	1.6E-08	5.3E-06	5.3E-06	U-233	6.6E-05	9.9E-06	7.6E-05
U-234	4.4E-06	2.6E-07	4.7E-06	U-234	1.8E-06	1.1E-07	1.9E-06
Pb-210	4.4E-07	7.0E-10	4.4E-07	Y-90	5.2E-08	7.7E-07	8.3E-07
Pu-240	2.1E-07	2.2E-09	2.1E-07	Pb-210	6.4E-07	1.1E-09	6.4E-07
U-232	6.0E-08	2.3E-09	6.3E-08	Sr-90	5.8E-07	1.4E-08	6.0E-07
Y-90	2.5E-09	3.7E-08	3.9E-08	Pu-240	4.2E-07	4.3E-09	4.2E-07
Sr-90	2.8E-08	6.5E-10	2.9E-08	U-232	3.1E-07	1.2E-08	3.2E-07
Ac-227	2.0E-08	3.0E-12	2.0E-08	Cm-244	6.6E-08	1.4E-11	6.6E-08
Pa-231	1.6E-08	1.2E-09	1.7E-08	Ac-227	1.4E-08	2.1E-12	1.4E-08
Cm-244	7.0E-10	1.5E-13	7.0E-10	Pa-231	1.2E-08	8.9E-10	1.3E-08
Cm-243	3.5E-12	3.5E-12	7.0E-12	Pu-241	3.0E-10	3.8E-16	3.0E-10
Pu-241	2.5E-12	3.2E-18	2.5E-12	Cm-243	8.3E-11	8.0E-11	1.6E-10
Total Dose (rem)	2.7E-01	8.6E-03	2.8E-01	**Total Dose (rem)**	5.5E-01	1.8E-02	5.7E-01
Probability of LCF	1.3E-04	4.3E-06	1.4E-04	**Probability of LCF**	2.7E-04	9.1E-06	2.8E-04
RH-TRU Waste							
Am-241	1.2E-02	4.0E-04	1.3E-02	Am-241	1.9E-02	6.1E-04	2.0E-02
Pu-239	2.9E-04	6.7E-06	3.0E-04	Cs-137	8.1E-06	2.6E-03	2.6E-03
Cs-137	5.7E-07	1.9E-04	1.9E-04	Pu-239	3.9E-04	9.0E-06	4.0E-04
Ac-227	5.1E-06	7.5E-10	5.1E-06	Y-90	1.4E-06	2.2E-05	2.3E-05
Pa-231	4.0E-06	2.9E-07	4.3E-06	Sr-90	1.6E-05	3.9E-07	1.6E-05
Pu-238	1.8E-06	1.9E-08	1.8E-06	Pu-238	5.3E-06	6.0E-08	5.3E-06
Y-90	9.7E-08	1.5E-06	1.6E-06	Ac-227	4.8E-06	7.2E-10	4.8E-06
Sr-90	1.1E-06	2.6E-08	1.1E-06	Pa-231	4.0E-06	2.9E-07	4.3E-06
U-233	5.8E-07	8.5E-08	6.6E-07	U-234	8.0E-07	4.7E-08	8.5E-07
U-234	5.9E-07	3.5E-08	6.3E-07	U-233	6.8E-07	1.0E-07	7.8E-07
Pb-210	3.0E-08	4.9E-11	3.1E-08	Cm-243	7.8E-08	7.3E-08	1.5E-07
Pu-240	3.1E-08	3.2E-10	3.1E-08	Pu-240	4.1E-08	4.3E-10	4.2E-08
Cm-243	5.5E-09	5.2E-09	1.1E-08	Pb-210	2.2E-08	3.5E-11	2.2E-08
U-232	5.8E-10	2.2E-11	6.1E-10	Cm-244	6.2E-09	1.3E-12	6.2E-09
Cm-244	1.1E-10	2.2E-14	1.1E-10	U-232	2.0E-09	7.5E-11	2.1E-09
Pu-241	2.7E-13	3.3E-19	2.7E-13	Pu-241	4.4E-11	5.5E-17	4.4E-11
Total Dose (rem)	1.2E-02	6.0E-04	1.4E-02	**Total Dose (rem)**	1.9E-02	3.2E-03	2.3E-02
Probability of LCF	6.2E-06	3.0E-07	6.8E-06	**Probability of LCF**	9.7E-06	1.6E-06	1.2E-05
CH-TRU and RH-TRU Waste							
Am-241	2.7E-01	8.8E-03	2.8E-01	Am-241	5.4E-01	1.8E-02	5.6E-01
Pu-239	5.5E-03	1.3E-04	5.6E-03	Pu-239	5.5E-03	1.3E-04	5.6E-03
Pu-238	9.6E-04	1.0E-05	9.7E-04	Cs-137	4.5E-06	1.4E-03	1.4E-03
Cs-137	3.4E-07	1.1E-04	1.1E-04	Pu-238	8.1E-04	8.8E-06	8.2E-04
U-233	3.4E-05	5.3E-06	3.9E-05	U-233	6.6E-05	9.9E-06	7.6E-05
U-234	4.9E-06	2.9E-07	5.2E-06	Y-90	8.2E-07	1.2E-05	1.3E-05
Ac-227	2.8E-06	4.2E-10	2.8E-06	Sr-90	9.2E-06	2.2E-07	9.4E-06
Pa-231	2.3E-06	1.6E-07	2.5E-06	Ac-227	2.8E-06	4.2E-10	2.8E-06
Y-90	5.7E-08	8.5E-07	9.0E-07	Pa-231	2.3E-06	1.6E-07	2.5E-06
Sr-90	6.4E-07	1.5E-08	6.6E-07	U-234	2.3E-06	1.3E-07	2.4E-06
Pb-210	4.6E-07	7.3E-10	4.6E-07	Pb-210	7.0E-07	1.1E-09	7.0E-07
Pu-240	2.2E-07	2.3E-09	2.3E-07	Pu-240	4.4E-07	4.5E-09	4.5E-07
U-232	6.1E-08	2.3E-09	6.3E-08	U-232	3.1E-07	1.2E-08	3.2E-07
Cm-243	2.8E-09	2.8E-09	5.6E-09	Cm-243	4.3E-08	4.1E-08	8.4E-08
Cm-244	7.6E-10	1.6E-13	7.6E-10	Cm-244	6.8E-08	1.4E-11	6.8E-08
Pu-241	2.6E-12	3.2E-18	2.6E-12	Pu-241	3.4E-10	4.2E-16	3.4E-10
Total Dose (rem)	2.8E-01	9.1E-03	2.9E-01	**Total Dose (rem)**	5.5E-01	2.0E-02	5.7E-01
Probability of LCF	1.4E-04	4.5E-06	1.4E-04	**Probability of LCF**	2.7E-04	9.8E-06	2.8E-04

Ingestion of heavy metals under this scenario would result in a 1.5 x 10^{-8}, 1.0 x 10^{-9}, and 1.5 x 10^{-8} probability of cancer incidence, respectively, for intrusions into CH-TRU waste, RH-TRU waste, and both CH-TRU and RH-TRU waste. There would be no noncarcinogenic impacts expected from ingestion of the metals because all hazard indices would be much less than one. Impacts from ingestion of heavy metals are shown in Table H-43.

Radiological impacts to the well-site geologist from external radiation exposure for Case 17 (median parameter values) would be a 2.1 x 10^{-9} probability of an LCF from CH-TRU waste and a 1.2 x 10^{-9} probability of an LCF from RH-TRU waste. For Case 19 (75th percentile parameter values), the radiological impacts would be a 2.2 x 10^{-9} probability of an LCF from CH-TRU waste and a 4.0 x 10^{-9} probability of an LCF from RH-TRU waste. Results of these analyses are presented in Table H-44.

Drilling Through the Repository into a Pressurized Brine Reservoir

Radionuclide migration for disturbed conditions at 10,000 years after closure for Case 17 (median parameter values) and Case 19 (75th percentile parameter values) showed migration of total radionuclide concentrations of one pCi per liter migrating upward and downward in the exploratory borehole. In both cases, contaminants at these levels penetrated a short distance into the surrounding rock and down into the Castile Formation and the hypothetical brine reservoir. The migration down to the brine would occur as the initial pressure in the reservoir dissipates and equilibrates with pressures in the repository and the surrounding units penetrated by the borehole. Heavy metal concentrations of one part per billion were similar to those calculated for Case 9 of Action Alternative 1.

Table H-43
Carcinogenic and Noncarcinogenic Impacts from Ingestion of Metals from Drilling Intrusions into the Repository for Action Alternative 3

Hazardous Metal	Case 17 (Median Values)		Case 19 (75th Percentile Values)	
	Probability of Cancer Incidence	Hazard Index	Probability of Cancer Incidence	Hazard Index
CH-TRU Waste				
Beryllium	1.5E-08	6.8E-07	1.5E-08	6.8E-07
Cadmium	3.3E-10	1.1E-07	3.3E-10	1.1E-07
Lead	0	1.0E-04	0	1.0E-04
Mercury	0	2.1E-04	0	2.1E-04
RH-TRU Waste				
Beryllium	1.0E-09	4.8E-08	1.0E-09	4.8E-08
Cadmium	2.1E-11	6.8E-09	2.1E-11	6.8E-09
Lead	0	6.5E-04	0	6.5E-04
Mercury	0	1.3E-05	0	1.3E-05
CH-TRU and RH-TRU Waste				
Beryllium	1.5E-08	7.1E-07	1.5E-08	7.1E-07
Cadmium	3.4E-10	1.1E-07	3.4E-10	1.1E-07
Lead	0	4.7E-04	0	4.7E-04
Mercury	0	2.1E-04	0	2.1E-04

Table H-44
Radiation Dose to the Well-Site Geologist from Drilling Intrusions for Action Alternative 3

Case 17 (Median Values) Intrusion at 400 Years After Closure				Case 19 (75th Percentile Values) Intrusion at 300 Years After Closure			
CH-TRU Waste		RH-TRU Waste		CH-TRU Waste		RH-TRU Waste	
Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)	Radionuclide	External Dose (rem)
Am-241	2.8E-06	Am-241	1.1E-06	Am-241	3.0E-06	Sr-90	3.9E-06
U-234	1.3E-06	U-234	7.9E-07	U-234	1.3E-06	Cs-137	2.0E-06
Pu-239	5.2E-08	Sr-90	3.6E-07	Pu-239	5.2E-08	Am-241	1.3E-06
U-233	4.5E-08	Cs-137	2.0E-07	U-233	4.5E-08	U-234	7.9E-07
Pu-240	9.0E-09	Pu-239	1.4E-08	Pu-238	1.2E-08	Pu-239	1.4E-08
Ac-225	7.2E-09	U-233	1.0E-08	Pu-240	9.1E-09	U-233	1.0E-08
Pu-238	5.5E-09	Pu-240	5.6E-09	Sr-90	6.7E-09	Pu-240	5.6E-09
Sr-90	6.1E-10	Ac-225	7.2E-10	Ac-225	5.6E-09	Y-90	5.6E-10
Cs-137	2.7E-10	Pa-231	9.7E-11	Cs-137	2.7E-09	Ac-225	5.3E-10
Pb-210	6.2E-12	Pu-238	7.5E-11	Pb-210	6.4E-12	Pu-238	1.8E-10
Pa-231	9.2E-13	Y-90	5.2E-11	Y-90	9.6E-13	Pa-231	7.1E-11
U-232	1.0E-13	U-232	2.4E-14	Pa-231	6.8E-13	U-232	6.3E-14
Y-90	8.8E-14	Pb-210	2.3E-18	U-232	2.8E-13	Pb-210	9.1E-19
Pu-241	1.6E-17	Pu-241	1.5E-21	Pu-241	1.6E-17	Pu-241	1.9E-19
Total Dose (rem)	4.2E-06	**Total Dose (rem)**	2.5E-06	**Total Dose (rem)**	4.4E-06	**Total Dose (rem)**	8.0E-06
Probability of LCF	2.1E-09	**Probability of LCF**	1.2E-09	**Probability of LCF**	2.2E-09	**Probability of LCF**	4.0E-09

Radionuclide and heavy metal migration for Case 19 (75th percentile parameter values) is essentially the same as shown in Figure H-13 for Case 9. Case 19 resulted in higher vertical upward migration than Case 17 and an actual release of contaminants into the Culebra Dolomite. The estimated maximum release rate of 1.3 Ci per year of radionuclides (predominantly Pu-239, Am-241, and Pu-240) and 57 kilograms (127 pounds) per year of heavy metals (predominantly lead) would occur at approximately 1,000 years after closure. The estimated total release to the Culebra Dolomite would be 1,090 Ci and 59,760 kilograms (131,700 pounds) of heavy metals, with the release occurring from 700 years to 3,900 years after closure. Analyses of releases to the Culebra showed that contaminants would be highly sorbed and only small amounts would migrate from the point of intrusion over 10,000 years. Transport calculations for Case 19 releases to the Culebra were made with the SECOTP2D model of the Culebra developed for the CCA (DOE 1996f).

Radiological impacts were calculated to an individual, such as a rancher, who consumed the beef of cattle that had consumed water from a stock well located downgradient at the land withdrawal area boundary 3 kilometers (2 miles) from the point of intrusion. Results are presented only for partial mining conditions, since calculated impacts for full mining conditions were significantly lower. Radiological impacts would be negligible, with a 2×10^{-27} probability of an LCF. The principal dose contributor was Pu-239 with all of the 10 other radionuclides contributing no more than 0.2 percent of Pu-239. Ingestion of heavy metals from consumption of contaminated beef was also found to result in negligible impacts, with a maximum probability of cancer incidence of 3×10^{-25}. No noncarcinogenic impacts would be expected.

H.9 ALTERNATIVE CONCEPTUAL MODELS AND VIEWS OF DISPOSAL PERFORMANCE

Several qualified and respected organizations and individuals, including members and contractors of the EPA, the Environmental Evaluation Group (EEG), the Office of the Attorney General of New Mexico, and others have participated in critical review of major WIPP compliance documents that are an important part of the compliance process. Key compliance documents include the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992), the *Compliance Status Report for the Waste Isolation Pilot Plant* (DOE 1994), and the CCA (DOE 1996f). These document reviews have given rise to a number of alternative conceptual models and views regarding WIPP repository disposal system performance that are relevant to this analysis. This discussion is included to present other viewpoints of WIPP performance assessment and to discuss how SEIS-II analyses consider them in its calculations.

Most of the review comments pertain to the performance assessment process and scenario screening and selection. Primary concerns appear to lie with characteristics and processes of the Culebra Dolomite Member of the Rustler Formation and the Culebra's role in radionuclide transport and WIPP performance under certain human-intrusion scenarios. Less has been said about modeling of the Salado Formation salt and the repository. In light of the existence of natural resources and the region's hydrocarbon extraction and potash-mining history, the probabilities of human intrusion used in calculations have been questioned as well, and several additional scenarios have been suggested for analysis. This section provides a summary of most of the major findings and conclusions presented by these past reviewers. The review comments are organized into two categories: (1) WIPP characteristics and conceptual models and (2) human impacts and intrusion.

H.9.1 WIPP Characteristics and Conceptual Models

This section presents an overview of alternative opinions regarding currently used WIPP site characteristics data, conceptual models, and calculation methods put forth by scientific and engineering review organizations. This overview is organized into the following major subject areas: repository characteristics, Salado Formation, Rustler/Salado Contact, Rustler Formation/Culebra Dolomite Member, and units above the Rustler Formation.

H.9.1.1 Repository Characteristics

Radionuclide Mobility - Partitioning of Actinides Into Oxidation States

Generic actinide solubilities are used in the CCA (DOE 1996f) calculations (i.e., solubilities of actinides of the same oxidation state are assumed to be equal). Actinides with multiple oxidation states are partitioned according to a set of simple formulae. According to Neill et al. (1996), this partitioning approach forces an actinide species to exist at several simultaneous oxidation states in certain proportions, which is not supported by experimental evidence. Also, a large uncertainty was associated with the assumed distribution of aqueous solubilities.

The Department has planned all along that experimentally determined actinide solubilities from its Actinide Solubility Program would be incorporated into calculations as they became available. Now, solubility values for several actinides have been experimentally determined and are being incorporated into WIPP performance assessment calculations. The SEIS-II analyses have incorporated the new actinide solubility values in source-term release calculations.

Radionuclide Mobility - Colloids

The EEG (Neill et al. 1996) point out that colloids are thought to travel no faster than the noncolloidal dissolved contaminants. This assumption negates the concern about colloids and allows calculations of unconservative (i.e., lower-than-expected) transport rates. Therefore, initial colloid concentration is assumed instead of being measured or calculated.

Currently, WIPP performance assessments methods and codes incorporate the concentrations of colloidal radionuclides into source-term release calculations. DOE has recognized the issues related to radionuclides in colloid form. In the CCA, Section 6.4.5.4 (DOE 1996f), the following discussion is presented: "Colloidal activities are subject to retardation by interaction between colloids and solid surfaces and by clogging of small pore throats (that is, sieving). It is expected that there would be some interaction of colloids with solid surfaces in the anhydrite interbeds. Because of the low permeability of intact interbeds, it is expected that pore apertures are small and some sieving will occur. However, colloidal particles, if not retarded, are transported more rapidly than the average velocity of the bulk liquid flow. Because the effects on transport of increased average pore velocity and retarding interactions with solid surfaces and sieving are offsetting, the DOE assumes residual effects of these opposing processes will be either small or beneficial and does not incorporate them in modeling of the transport of actinides in the Salado interbeds." Similarly, SEIS-II analyses include calculated colloid concentrations and do not include consideration of colloid transport processes.

Radionuclide Transport Within Repository

The assumed location(s) of the intrusion borehole(s) penetrating a Castile Formation brine reservoir within the repository would affect calculated release rates of radionuclides from the repository (EPA 1995). The effect of multiple intrusions on flow and transport was analyzed in the E1E2 scenario in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992), but radionuclide transport within the repository and the effect on release rates were not modeled.

The EPA (1995) evaluated the sensitivity of radionuclide release rates to borehole locations for several different geometries. The results of the analysis varied widely with both the number and the locations of the boreholes. In one case, most of the brine flowed quickly up the borehole leaving the radionuclides far behind. Four boreholes located at the corners of the repository enabled greater quantities of radionuclides to flow up the borehole. Ten uniformly arranged boreholes maximize the flow rate and quantity of radionuclides from the repository.

Radionuclide transport within the repository was not explicitly modeled in the CCA (DOE 1996f) or in the SEIS-II analyses. After a single intrusion, the pressure in the repository would decline substantially. For a multiple-intrusion scenario, the maximum impact would be seen in the first intrusion; less material would be released at any one borehole following the first intrusion because gas pressure would likely dissipate after the first intrusion. Therefore, although it is recognized that multiple-borehole intrusion scenarios could result in higher cumulative releases than those from a single-borehole intrusion, the intrusion scenario analyzed for SEIS-II would result in the highest releases from a single event.

Isothermal Calculations With BRAGFLO

Neill et al. (1996) state that BRAGFLO, according to the description and application presented in the CCA (DOE 1996f), is "apparently an isothermal code," and that the justification for isothermal treatment of two-phase flow should be given. Neill et al. (1996) also feel that an assessment of errors introduced by this sort of approximation should be provided.

In the SEIS-II analyses, isothermal conditions were assumed for all simulations. With the thermal power limits imposed by planning-basis WAC, it is not anticipated that thermal loading from TRU waste will be an important factor in source-term release and near-field transport from the repository. The small amount of heat that will be generated by RH-TRU waste will be dissipated by the repository salt.

H.9.1.2 Salado Formation

Salado Halite and Interbed Parameters

The permeability of halite, as modeled in BRAGFLO, is that of "impure" halite and was determined through testing. Neill et al. (1996), in their review of the CCA (DOE 1996f), maintained that it should be demonstrated that halite permeability bounds the overall permeability when interbeds, such as polyhalite and anhydrite, in the halite are considered. In addition, the effects of increased permeability of anhydrite interbeds from elevated gas pressures on the overall impure halite permeability should be incorporated.

Similarly, Neill et al. (1996) stated that if halite specific storage is used for the halite model, it should be demonstrated that the contribution of the interbeds to specific storage can be ignored. (The specific storage of a rock unit is a hydrologic term referring to the amount of water which would be squeezed from the unit's pores if the unit were compressed, as during creep closure, or the amount of water which the unit's pores would absorb if the unit were dilated, as may occur with elevated gas pressure in the repository. An anhydrite interbed would likely have a specific storage that is different from that of an impure halite layer, particularly when dilation of fractures in the interbeds due to elevated gas pressure is considered.)

In the April 30, 1996, draft of the *Final No-Migration Variance Petition*, Section 8.0 (DOE 1996d), which employs median and other appropriate values (not sampled parameter distributions), it is stated that parameter values for impure halite used in the modeling are "supported by four hydraulic tests in the underground repository [and are] believed to represent far-field conditions and stratigraphic variation in the Salado." Further, "except for the DRZ and anhydrite interbeds, under certain circumstances, this simulation [involving fluid flow, gas generation, and volume changes resulting from creep closure] assumes spatially constant properties for Salado rock types based on observations of compositional and structural regularity in layers exposed by the repository. The inference is that there is little variation in large-scale averages of rock or flow properties across the disposal system. Except for anhydrite interbeds, model parameters are also spatially invariant within each material region. At relatively low repository pressures, porosities of all Salado materials vary slightly; however, for interbeds, the model implemented to simulate the effects of interbed fracturing causes large increases in both porosity and permeability above a designated fracture initiation pressure."

Regarding changes in interbed permeability and specific storage due to elevated gas pressures, it is stated in the April 30, 1996 draft of the *Final No-Migration Variance Petition* (DOE 1996d) that, "if high pressure develops in an interbed, its preexisting fractures may dilate, or new fractures may form, altering its porosity and permeability. Pressure-dependent changes in permeability are supported by experiments conducted in the underground repository and in the laboratory. Accordingly, the DOE has implemented in BRAGFLO a porous-media model of interbed dilation and fracturing that causes the porosity and permeability of a computational cell in an interbed to increase as its pore pressure rises above a designated value. There is a trade-off between the effects of permeability and porosity enhancements. Dilation or fracturing of interbeds is expected to increase the transmissivity of interbed intervals. Increased porosity will increase storage, which will retard outward flow." However, because of assumptions incorporated into the calculations, large increases in permeability are accompanied by modest increases in porosity. These concerns are accounted for in the performance assessment calculations.

In SEIS-II analyses, values of Salado Formation halite and interbed parameters were derived from DOE databases supporting the CCA (DOE 1996f). The analyses use both expected and conservative values based on existing distributions of these parameters.

Brine Inflow

The EEG (1994), in its review of the *Compliance Status Report for the Waste Isolation Pilot Plant* (DOE 1994), states that the "project position" on the preferred conceptual model for brine inflow should be "developed and justified." Furthermore, "the EEG does not agree with the strategy of treating various conceptual models to be of equal importance when overwhelming evidence exists that a particular model is far superior than others." The EEG's recommendation is to assume Darcy flow in the salt, impure salt permeabilities, and fractured anhydrite, using measured in situ permeabilities in the marker beds.

The conceptual model for brine inflow, as described in the CCA (DOE 1996f), includes two submodels to approximate far-field flow and redistribution mechanisms. A previously proposed mechanism, clay consolidation, was not considered important and was not included. A 1-degree stratigraphic dip was added to the model, Darcy flow was assumed, and permeability for impure halite was used. Permeability for anhydrite (presumably intact) was based on field and laboratory measurements, and the chosen maximum fractured anhydrite permeability values were "thought to be upper limits."

This approach used in the CCA (DOE 1996f) was adopted for the SEIS-II analyses. For the conservative case, the anhydrite permeability range included values for fractured anhydrite.

H.9.1.3 Rustler/Salado Contact

According to Neill et al. (1996), the contact zone between the Rustler and Salado Formations is characterized by residue left from dissolution of salt and has not been adequately considered as a potential pathway for migration of radionuclides. Several facts suggest that the Rustler/Salado contact merits further study. Chaturvedi and Channell (1985; cited in Neill et al. 1996) indicate that data from hydrologic testing at the WIPP site "shows that the 'brine aquifer' of the pre-WIPP investigators extends east of Nash Draw to the WIPP site." Most boreholes have encountered brine in the Rustler/Salado contact zone, and water-level-recovery rates were more rapid in these wells than in a well in the Culebra Dolomite east of the WIPP site, suggesting that the contact zone exhibits relatively high permeability and transmissivity.

In DOE's responses to EPA comments (DOE 1996a), it is stated that dissolution at the top of the Salado is not expected to reach the edge of the controlled area within the regulatory period of analysis of 10,000 years. Thus, the Rustler/Salado contact was not explicitly considered as a region of enhanced flow in the CCA or SEIS-II analyses.

H.9.1.4 Rustler Formation/Culebra Dolomite Member

Discussion of alternative conceptual models and issues related to the Culebra Dolomite are presented below in three parts: (1) regional flow, (2) hydraulic properties and characteristics, and (3) contaminant transport and retardation characteristics.

Regional Flow

Two-Dimensional versus Three-Dimensional Flow

The *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992) treated flow in the Rustler Formation as occurring primarily in the Culebra Dolomite, which has been modeled as a confined, nonleaky, two-dimensional, horizontal, and heterogeneous water-bearing unit. It is, however, actually part of a three-dimensional flow system. SNL has modeled three-dimensional flow in the Rustler and interim results indicate that, while some flow occurs vertically between units of the Rustler, most flow occurs within the Culebra. Therefore, confining flow to the Culebra in the performance assessment calculations is a justifiable simplification. Konikow (1995) believes that the performance assessment effort has been "overly focused" on the two-dimensional analysis and that the ongoing three-dimensional analyses of the regional groundwater flow system are critical to the performance assessment effort.

Currently, DOE is retaining the concept of confined two-dimensional flow in the Culebra for compliance analysis. DOE is considering the development of a three-dimensional model for possible use in future performance assessment calculations, to evaluate the impact of regional groundwater flow on long-term WIPP performance. The SEIS-II analyses used the two-dimensional flow and transport model based on SECOFL2D and SECOTP2D to estimate the off-site impacts from releases to the Culebra Dolomite for the disturbed cases using 75th percentile parameter values. Releases to the Culebra were predicted to be too small to constitute a significant health risk.

Regional Flow and Anomalous Groundwater Geochemistry

It has been suggested by a number of investigators that the chemistry of the Culebra groundwater is inconsistent with the apparent direction of flow. Based on hydrologic well data, groundwater in the Culebra flows roughly from north to south. Groundwater chemistry along the flow direction is not what would be predicted, based on common flow-chemistry relationships. Total dissolved solids decrease downgradient, and the general chemical nature of the water changes as well (sodium and chloride at the WIPP site to magnesium, calcium, and sulfate south of the site).

According to Neill et al. (1996) and Konikow (1995), this inconsistency reflects an inadequate level of understanding of the entire hydrogeologic system. Axness et al. (1995) state that “the relationship between water chemistry and groundwater flow in the Culebra remains unresolved at this time.”

In Neill et al. (1996), the EEG advocates a “full discussion [in the CCA] with respect to flow directions, vertical seepage, karst, present day recharge and paleo-recharge.” The basis for determining the estimated age of Culebra groundwater as presented in the CCA (i.e., “tens of thousands of years”) and as “a relict of a flow regime of a wetter climate” (Neill et al. 1996), has never been accepted by the EEG (Neill et al. 1996). According to the EEG, the arguments against the use of isotopic data from Carlsbad Caverns pools for Rustler groundwater (Neill et al. 1996) should be presented.

In a recent interpretation by Corbet (1997), the Department has suggested that changes in groundwater chemistry that would be expected for a confined aquifer system are complicated by contributions of vertical leakage and regional groundwater recharge that interact with distinctive rock types originating in different areas surrounding the WIPP site. Corbet (1997) concluded that the distributions of solute chemistry observed in the Culebra were consistent with inferred groundwater flow conditions and reflected a mixing of distinctly different groundwater originating from recharge areas surrounding the site. Additional information on this topic is provided in Section 4.1.3.2 and in Corbet (1997).

Recharge/Discharge

In Neill et al. (1996), the EEG points out that the recharge area for the Rustler has never been identified. At least two areas have been proposed on the basis of potentiometric surfaces, but “existing data are inadequate to determine recharge to the groundwater system in the vicinity of the WIPP site.”

It is accepted that the Culebra probably discharges ultimately into the Pecos River and, perhaps, elsewhere (Neill et al. 1996). However, hydraulically separate water-bearing zones cannot be distinguished within the Rustler Formation at least 3 kilometers (2 miles) east of Livingston Ridge; therefore, water flowing into postulated areas of discharge may not be traced to a particular member of the Rustler Formation (Neill et al. 1996).

For the purpose of SEIS-II analyses, existing models that support the CCA analyses (DOE 1996f) are being used.

Unexplained Recent Changes in Water Levels

Neill et al. (1996) believe that the observed water-level rises in the Culebra are notable, in that they may be related to hydrocarbon and potash activity in the area. The CCA (DOE 1996f) used data through 1991 only in its related discussion, although there have been data collected up to the present. Konikow (1995) believes that the observations are important and maintains that the lack of a satisfactory interpretation is another element in the generally inadequate understanding of the site hydrogeology.

According to Neill et al. (1996), “water level rises in WIPP monitoring wells potentially correlate with brine disposal from the potash industry.” Further, “in 1988, WIPP monitoring wells experienced sharp water level rises which were strongly correlated with a nearby salt water disposal well operated by the oil and gas industry." These instances emphasize that WIPP is located in a resource-rich area, exploration and exploitation of the resources are likely to continue, and activities related to this may influence the regional hydrology apart from the modeled scenarios involving penetration of the repository. Therefore, Neill et al. (1996) maintain that more emphasis must be placed on interpreting the changes in Culebra water levels.

Analysis of water level changes in H-9 and in a number of other observation wells north and south of the WIPP site and near WIPP have suggested that the changes in wells to the south are the result of possible hydraulic impacts of water flooding (water injection to enhance secondary oil and gas recovery) activities south of the WIPP site. No specific evidence is currently available to conclude which specific water flooding operation or specific hydraulic condition is creating the specific water level changes observed near the WIPP site.

In response to these observations and other examples of the impacts of fluid injection, DOE has examined the potential impacts of fluid injection in the form of water flooding and salt water disposal in the CCA. The potential effects of water flooding and salt water disposal were modeled assuming two hypothetical injection wells located at the land withdrawal boundary operating over a 50-year period. The results of the modeling, given in Section SCR.3.3.1.3.1 of the CCA, indicate that fluid injection would not have a significant impact on repository performance. Specifically, even for the least favorable rock properties considered, the amount of brine reaching the repository over 10,000 years would be well within the range of volumes of brine expected to flow into the repository during normal undisturbed performance. On this basis, fluid injection was screened out of the performance assessment calculations in the CCA and SEIS-II analyses.

Hydraulic Properties and Characteristics

Independent Sampling of Parameters

In the probabilistic performance assessment calculations, many parameters are sampled. It has been pointed out (*cf.* EEG 1994; Konikow 1995; Neill et al. 1996) that certain model parameters in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992) are being treated as independent random variables. This is a weakness (Konikow 1995) because, in reality, the variability of some sets of parameters indicates dependence on one another. For example, the values for fracture porosity and fracture spacing used in some flow and transport modeling calculations were sampled randomly from a selected range. This approach implies that there is no dependency between these two parameters. Single fractures could have very small apertures (resulting in low porosity) and multiple closely spaced fractures could have very large apertures (resulting in high porosity). These assumptions result in very large ranges in fracture hydraulic conductivity, which runs contrary to most hydrologists' understanding that fracture spacing and fracture aperture (porosity) generally increase or decrease together.

The EPA (1995) evaluated the relationship between and among fracture spacing, fracture porosity, and hydraulic conductivity. Based on the range of fracture spacings used in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992), the EPA calculations indicated that corresponding fracture porosities are much lower than those used in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992). The performance assessment calculations, therefore, indicate higher flow and transport rates and are very conservative.

In the SEIS-II analyses, expected and conservative parameter values (corresponding to median and 75th percentile, respectively) were used, rather than sampled values.

Fracture Properties

According to the EEG (1994), statements in the *Compliance Status Report for the Waste Isolation Pilot Plant* (DOE 1994) regarding areal distribution and density of fractures in the Culebra are simplistic and ignore existing data, that the "pattern of fracture distribution and corresponding transmissivity values distribution is too complex to be explained away," and that its complexity has become more apparent as additional data are collected. In response, DOE stated that deduced fracture patterns were based on many observations in drill core, as well as outcrops and exposures in the air intake shaft.

Konikow (1995) states that fractures are a dominant control on transmissivity and they represent the highest-velocity channels for migration of contaminants. Further, a better definition of the nature, density, spacing, length, and interconnectedness of fractures and fracture networks is needed. He characterizes this uncertainty regarding Culebra fractures as "the most likely fatal flaw in site integrity."

Accordingly, Konikow (1995) believes that, given the fact that detailed local and regional mapping of fracture traces and lineaments were not completed during the early stages of site characterization, new approaches should be taken that may help clarify the possible existence and spacing of major through-going fracture zones. He indicates that techniques that could accomplish this include computer analysis of digital elevation data and high-resolution three-dimensional geophysical tomographic techniques. This issue remains open.

The EEG (1994) states that more than one conceptual model of the Culebra appears to fit the available data. Other possible interpretations include single-porosity flow (flow through the rock matrix), dual-porosity flow (flow through fractures in the rock, as well as the rock matrix), and channeling (flow through certain fractures only). Of equal importance is the role of matrix diffusion as a retardation process (diffusion of radionuclides through the fracture faces into the rock matrix; see below).

The EEG (1994) points out that the INTRAVAL committee (International Project to Study Validation of Geosphere Transport Models) believes that, while the current model incorporates dual-porosity flow, a model based entirely on channeling also fits the current field data. The channeling model does not incorporate matrix diffusion.

Since 1994, the model of the Culebra Dolomite regional hydrologic characteristics has not changed from those conceptualized in previous performance assessment calculations, although additional large-scale information from pumping at H-19 and small-scale information at Water Quality Sampling Program (WQSSP) wells has been incorporated into the calibration. Existing borehole transmissivity interpretations have been refined on the basis of analysis of new data from H-19 and H-11 and reanalysis of previous tests of H-3, H-11, and H-6.

The Culebra is now conceived of as a fractured porous medium with inherent local variability in the degree and scale of fracturing. Examination of core and shaft exposures has revealed that there are multiple scales of porosity within the Culebra including fractures from microscale to large, vuggy zones, and interparticle and intercrystalline porosity. This variability leads to both lateral and vertical variations in permeability. Advection is believed to occur largely through fractures; however, in some areas it may also occur through vugs connected by small fractures and interparticle porosity. Performance assessment, rather than conceiving of transport in terms of fracture and matrix porosities, conceived the Culebra as being composed of advective and diffusive porosities. Matrix diffusion is still believed to be effective and significant. The effective transport thickness is thought to be less than the total stratigraphic thickness. The available data suggest that the permeability of the upper portion of the Culebra is relatively low. Therefore, the DOE has concluded that the Culebra is adequately represented by the double porosity continuum model on the scale of the performance assessment calculations, and it is not necessary to use the discrete-fracture model on this scale.

For compliance calculations and the purposes of SEIS-II, DOE is incorporating the assumption that flow in the Culebra is confined to a single fracture "in order not to overestimate the amount of diffusion [of radionuclides]" (Axness et al. 1995). Diffusion of radionuclides into fracture walls has the effect of slowing their transport and, thus, increasing retardation. Limiting the Culebra to a single fracture minimizes retardation and is a more conservative approach. In addition, in recognition of the importance of fracture characteristics of the Culebra, the DOE has conducted multiple-well tests and tracer tests to evaluate these characteristics and their influence on the regional groundwater flow field.

Karst Development

Anderson (1994), in his review of the *Compliance Status Report for the Waste Isolation Pilot Plant* (DOE 1994), disagreed that the absence of visible karstic surface features at WIPP implies that important karst processes are not occurring. According to Anderson, the "moderate thickness of halite and gypsum strata in the Rustler Formation precludes the development of large, visible collapse structures until late stages of dissolution." Sand cover at WIPP would tend to obscure smaller-scale features.

Anderson (1994) presents evidence that dissolution from karst processes is active at WIPP: (1) a dissolution front beginning along the Nash Draw axis has moved eastward approximately 16 kilometers (10 miles) to its present position within the WIPP site; (2) this dissolution front has moved in pulses in response to changes in climate; and (3) the northward extension, or finger, of the southeastern lobe of Nash Draw in the southern part of the WIPP site coincides with the main flow path in the Rustler aquifer and with the known localized area of increased transmissivity (Beauheim and Holt 1990; Anderson 1994).

Anderson (1994) cites several lines of evidence that suggest that karst processes at WIPP are at a relatively early stage: (1) the age of Nash Draw has been determined to be less than 600,000 years, which is considerably younger than has been believed (Beauheim and Holt 1990, cited in Anderson 1994); (2) the high-transmissivity zone is characterized by relatively fresh groundwater that is unsaturated with respect to gypsum, and by fractures from which gypsum has been dissolved; (3) fractures in soluble units below the Culebra, visible in one of the WIPP shafts, have been enlarged by dissolution to form flow channels; (4) hydraulic conductivity across the site varies by a factor of one million (specific Rustler lithologic units were not specified); and (5) vertical movements of fluids between units in the Rustler are characteristic.

For nearly 20 years, DOE has investigated the hydrology of important geologic units overlying the WIPP facility and the importance of karst features and related dissolution processes in defining the surface features in the region surrounding the WIPP site. A description of the current understanding of the extent, timing, and features related to dissolution, including a brief history of past project studies related to karst in the area surrounding WIPP, is presented in Section 2.1.6.2.1 of the CCA. These studies have shown that there is considerable evidence of dissolution and karst features at shallow depths, although no evidence has been collected to date that would suggest that shallow dissolution processes are active within the deeper Salado Formation. Deep dissolution at the WIPP site was eliminated from the CCA and the SEIS-II performance assessment calculations on the basis of low probability of occurrence over the next 10,000 years. Additional information supporting this conclusion is provided in Section SCR.1.1.5.1 of Appendix SCR of the CCA (DOE 1996f).

DOE does not specifically address the karst issue in the long-term performance assessment. However, for SEIS-II, a potash mining scenario was analyzed which incorporated a three-order-of-magnitude increase in the hydraulic conductivities (which control permeabilities) in overlying hydrologic units. This analysis allowed the investigation of the potential effects of increased permeability from any cause, including karst development.

Contaminant Transport and Retardation Characteristics

Equilibrium Sorption

During transport of contaminants, several chemical processes may serve to slow down (i.e., retard) the contaminants. These processes include precipitation of the contaminants in a chemical compound, ion-exchange processes, or adsorption onto solid surfaces. These processes are collectively referred to as retardation. They may be temporary, as precipitation may stop in response to chemical conditions, or the solid surfaces may fill up with sorbed contaminants and lose their capacity to sorb further. For many contaminants, groundwater compositions, and rock/soil types, it is possible to estimate by calculation the distribution of contaminants sorbed on the rock and dissolved in the water by determining the distribution coefficient.

Distribution coefficients have been measured for application to WIPP but they do not represent anticipated conditions in the Culebra (Neill et al. 1996) for the following reasons: (1) experiments have used water that was chemically different from Culebra water; (2) the distribution coefficients were determined from single measurements on powdered samples, which have much greater surface area compared to their volume than the actual fractured Culebra rock and, therefore, would tend to show an artificially high degree of sorption; and (3) it was not well demonstrated that equilibrium was achieved in the experiments.

According to Konikow (1995), the use of a single retardation factor, as in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* calculations (SNL 1992), has a “very weak scientific basis” because it cannot represent the sources of variation in the performance assessment model (chemical reactions, reaction rates, heterogeneous mineralogy, and changing aqueous geochemistry). In addition, Konikow (1995) states that the retardation factor model used does not place an upper limit on the amount of contaminant that can be sorbed. This means that the model does not account for the possibility of sorption sites filling up and precluding further sorption. He further states that sorbing tracer tests in the field are needed.

A multiwell tracer test is currently being conducted at the WIPP site and is designed to ascertain, among other things, distribution coefficients for sorbing contaminants but using a nonsorbing tracer. Neill et al. (1996) compared residence times for sorbing and nonsorbing species in a system of porous and fractured rock. Their simple analysis indicates that a nonsorbing tracer test cannot be used to obtain a distribution coefficient for sorbing contaminants such as the radionuclides at WIPP.

In the CCA analysis, clay linings in Culebra fractures are not currently assumed to be present. In the SEIS-II analyses, transport calculations in the Culebra were performed using the CCA models to evaluate the predicted releases to the Culebra for selected disturbed performance cases. Evaluation of a release at a 3-kilometer (2-mile) stock well showed negligible risks.

Repository and Culebra not Coupled in Modeling

In the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992), the quantities of radionuclides calculated to leave the repository and flow up an intrusion borehole were incorporated into the simulation of flow and transport in the Culebra, but they were not introduced hydraulically into the flow regime of the Culebra (i.e., the influence of water flowing up the borehole on the ambient flow field of the Culebra was not calculated). Expressed another way, the processes in the repository were decoupled from processes in the Culebra.

The EPA (1995a) evaluated the effect on transport times of coupling repository and Culebra processes. It was found that travel times to the WIPP boundary decreased (i.e., transport rates increased) by approximately ten times (one order of magnitude). Similarly, radionuclide concentrations calculated at one kilometer (0.6 mile) from the repository after 10,000 years were approximately ten times higher in the coupled-process calculations.

The SEIS-II analyses follow the approach taken in the CCA (DOE 1996f) in which impacts of the borehole intrusion on the Culebra flow field are not considered. Reeves et al. (1991) performed a study that indicated that, for expected conditions and much of the range of brine reservoir and breach borehole parameters, the fluid disturbance created in the Culebra by the breach borehole would have minimal impact on the flow field. It was further stated that, under these conditions, transport calculations need not include the transient impact of locally increased hydraulic head near the breach borehole.

In the intrusion scenario analyzed in SEIS-II, the borehole was designed to reflect current oil field practices. Two concrete plugs are assumed to have a significant effect on long-term flow in the borehole: the lower plug is assumed to be located between the hypothetical Castile brine reservoir and the underlying formations, and a second plug is located within the lower portion of the Rustler and immediately above the Salado. Additional plugs that have little effect on long-term flow are also assumed to be present, both deeper in the hole and at the land surface. The brine reservoir and the repository are assumed to be in direct communication through an open-cased hole immediately following drilling. The plugs are represented in the borehole by material zone 29 of the BRAGFLO mesh in Figure H-4 (a surface plug and a plug in the Lower unnamed member). The plugs located below the brine reservoir are not modeled explicitly. Plugs are assigned initial permeabilities of 5 x 10^{-17} square meters (5.4 x 10^{-16} square feet), which is consistent with the expected properties of intact concrete, and the open segments of the borehole (between the plugs) are assigned an initial permeability of 10^{-9} square meters (1 x 10^{-8} square feet). Steel casing above the Salado Formation is assumed to begin to degrade within decades after abandonment and is assumed to have failed completely after 200 years. The concrete plugs above the Salado are also assumed to fail after 200 years, as a result of chemical degradation by contact with brine. The plug below the Castile brine reservoir is in a less aggressive chemical environment, and its properties remain constant in performance assessment. After the upper plugs and casing have failed, the borehole is assumed to be filled by a silty, sand-like material containing degraded concrete, corrosion products, and material that sloughs into the hole from the walls. Thus, beginning 200 years after the time of intrusion, the entire borehole region in the BRAGFLO model, including the sections previously modeled as concrete plugs, is assigned a permeability corresponding to silty sand. This permeability is sampled from a log-uniform distribution from 10^{-11} square meters (1 x 10^{-10} square feet) to 10^{-14} square meters (1 x 10^{-13} square feet).

One thousand years after the plug at the base of the Rustler Formation has failed (1,200 years after the time of intrusion), the permeability of the borehole region below the waste-disposal panel in the BRAGFLO model used is decreased from its sampled value by one order of magnitude. For the remainder of the 10,000-year period, the borehole is modeled with its sampled permeability value above the repository and the adjusted value below. Conceptually, the decrease in permeability below the panel corresponds to compaction of the silty, sand-like material by partial creep closure of the lower portion of the borehole. As discussed in Appendix MASS (DOE 1996f), creep closure of boreholes is not expected to be significant above the repository horizon but will be effective at greater depths because of the greater lithostatic stress.

For these assumptions, releases were simulated only for the 75th percentile parameter cases. In these cases, the releases were quantitatively small enough to not pose a significant health risk (refer to Section H.8), and additional modeling of transport in the Culebra (which would further dilute the concentrations due to dispersion) was unwarranted.

Realism of Calculated Travel Times

In the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992), the Culebra was modeled by using various transmissivity fields, each of which was divided into regions of different hydraulic conductivity. For the purpose of travel time calculations, however, a mean Culebra hydraulic conductivity of 7 meters (23 feet) per year was used.

The EPA (1995) compared travel times calculated by using the mean, as in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992), with those calculated by using the separate regions of hydraulic conductivity. Their results suggest that using separate hydraulic conductivities yields travel times “far shorter” (less conservative) than using the mean hydraulic conductivity for the entire Culebra.

In the SEIS-II analyses, transport calculations using the SECOTP2D model of the Culebra were performed for contaminant releases to the Culebra from disturbed conditions to a 3-kilometer (2-mile) stock well downgradient of the point of intrusion. Impacts at the stock well were found to be negligible.

Parallel Fracture Model Analysis for Culebra

Neill et al. (1996) state that the basis for the parallel fracture model in the SECO analyses for the Culebra in the CCA (DOE 1996f) is not presented nor is the justification for clay linings on the fracture walls, and that the influence of this assumption on the outcome of the calculations should be described. Channeling of groundwater flow should be considered, because it is recognized as a possibly important phenomenon in the Culebra.

In performance assessment calculations for compliance, a single fracture in the Culebra was assumed to limit the amount of fracture surface and calculated matrix diffusion. Porosity was assumed to be the approximate median of the distribution sampled in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992), and tortuosity was given a “medium low” value. The fracture is assumed to be devoid of clay lining, and the distribution coefficient is taken to be zero.

In the SEIS-II analyses, transport calculations using the SECOTP2D model of the Culebra were performed to estimate the impacts of migration of contaminant releases to the Culebra from selected disturbed conditions to a 3-kilometer (2-mile) stock well downgradient from the point of intrusion. These impacts were found to be negligible.

Presence/Absence of Clay in Fractures

Neill et al. (1996) believe that there is insufficient evidence of clay linings in Culebra Dolomite fractures to assume their presence for purposes of estimation of retardation of radionuclide transport. The assumption of corrensite as the predominant clay mineral present in the fractures is also based on limited data.

According to the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* calculations (SNL 1992), chemical sorption on clay fracture linings is assumed to occur and the clay mineral present is assumed to be corrensite. Neill et al. (1996) provide a discussion of the evidence on which these assumptions are based. SNL has been assuming in their performance assessment calculations that fractures in the Culebra are lined to some degree with clay and that the clay has been determined to be corrensite. The presence of corrensite is based on X-ray diffraction and electron microscopic analysis of core samples taken from clay-rich zones in the Rustler Formation (not necessarily in the Culebra Member), primarily from locations in Nash Draw, several miles west of the WIPP site.

In a study by Sewards et al. (1991; cited in EEG 1994 and Neill et al. 1996), X-ray diffraction determination of corrensite was not corroborated by the electron microscopy, yet they concluded that corrensite is the dominant clay phase in the Culebra (EEG 1994). Also, in Sewards et al. (1991; cited in EEG 1994 and Neill et al. 1996), it was stated that only small amounts of clay could be sampled from the Culebra fracture coatings. As a result, initial laboratory studies of adsorption on WIPP site clays were carried out with material from a black shale layer from the unnamed member of the Rustler Formation. The material was determined from a single sample to be mostly corrensite. Neill et al. (1996) state, in summary, that using a single sample from a shale located in a different part of the Rustler from the Culebra is not appropriate for performance assessment calculations that depend on the presence of sufficient corrensite clay in Culebra fractures to effect notable retardation of radionuclides. Therefore, either more evidence is needed for radionuclide sorption on clay linings or credit should not be taken for chemical retardation in fractures.

Clay can be a medium for sorption but it can also block radionuclides from diffusing into the rock matrix (see discussion below on physical retardation). The *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992) did not include the latter concept in its calculations (Neill et al. 1996). If the clay is not an effective sorber, it serves to inhibit migration of radionuclides into the matrix, thereby increasing the efficiency of channel flow.

For the purpose of performance assessment calculations in the CCA and SEIS-II analyses, clay linings in Culebra fractures are assumed not to be present. In the SEIS-II analyses, transport calculations using the SECOTP2D model of the Culebra were performed to estimate the impacts of migration of contaminant releases to the Culebra from selected disturbed conditions to a 3-kilometer (2-mile) stock well downgradient from the point of intrusion. These impacts were found to be negligible.

Physical Retardation (Matrix Diffusion)

Performance assessment calculations (SNL 1992) suggest that radionuclides will take from 100 to 1,000 years to travel from the repository to the WIPP boundary. This indicates that without matrix diffusion to slow up (retard) the radionuclides, cumulative releases would be greater over the 10,000-year regulatory period. It is believed by some (Neill et al. 1996; Konikow 1995) that there is insufficient evidence to assume that matrix diffusion plays an important role in retarding radionuclides in the Culebra. It is Konikow's (1995) opinion that field tests performed to date are ambiguous, that diffusion parameters have not been adequately characterized in laboratory tests, and that the nature of the fractures in the Culebra are not known sufficiently well to formulate a representative model.

Neill et al. (1996) state that, though performance assessment takes credit for matrix diffusion, there is "no direct experimental evidence for its extent." The EEG (1994) points out that the INTRAVAL committee believes that existing field data support a channeling flow model (i.e., without matrix diffusion) as well as the dual-porosity model. The EPA (1995) performed simple calculations of travel times and distances with and without the retarding effects of matrix diffusion to obtain comparisons. For a set of simplifying conditions (fracture spacing 3.85 meters [12.6 feet], equivalent porous media hydraulic conductivity 7 meters [23 feet] per year, no chemical retardation), the calculation of distance traveled in 10,000 years yielded the following results: (1) with matrix diffusion, radionuclides traveled approximately 5 kilometers (3 miles) and (2) with no matrix diffusion, radionuclides traveled approximately 13,500 kilometers (8,383 miles), or 2,700 times farther.

In an attempt to limit the amount of fracture surface and ensure that matrix diffusion calculations are conservative, it was assumed in the CCA that fracturing in the Culebra is limited to one horizontal fracture. Fracture porosity was taken to be approximately the median of the range sampled in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992), and tortuosity was given a "medium low value."

In the SEIS-II analyses, transport calculations using the SECOTP2D model of the Culebra were performed to estimate the impacts of migration of contaminant releases to the Culebra from selected disturbed conditions to a 3-kilometer (2-mile) stock well downgradient from the point of intrusion. These impacts were found to be negligible.

H.9.1.5 Supra-Rustler Units

The water table in the region of the WIPP has not been defined. Konikow (1995) states that, in part, this reflects the degree of understanding of the site hydrology. He recommends that it either should be defined or a reasonable explanation regarding why it is not important or not technically feasible to define should be documented.

According to Neill et al. (1996), "without an understanding of the basic regional hydrologic parameters of an area, such as the water table and the recharge and discharge areas and amounts, the knowledge about the site is incomplete." They state that it is believed that the water table is in the Dewey Lake Formation, based on observations of water in several wells and one of the repository shafts at the WIPP site. Wells in the Dewey Lake have produced water at rates up to 106 liters (28 gallons) per minute (Neill et al. 1996; Axness et al. 1995).

In the *Final CCA* (DOE 1996f) simulation of the WIPP repository disposal system pressure gradient, it was assumed that the water table is located approximately 59 meters (194 feet) below the ground surface at an elevation of 980 meters (3,215 feet) within the Dewey Lake Formation. The Dewey Lake Formation contains a "productive zone of saturation, probably under water-table conditions" in the southwestern and south-central portion of the WIPP site as well as south of the site. This zone occurs approximately in the middle of the Dewey Lake and appears to derive much of its transmissivity from open fractures. North of the site, open fractures and/or moist (not saturated) conditions have been observed in the Dewey Lake at similar depths. Fractures below the productive zone tend to be filled with gypsum.

The role of the Dewey Lake Formation in repository performance is an issue as yet unresolved. Because the Culebra is assumed to be the principal pathway for contaminant transport above the Salado in the event of a repository breach as a result of human intrusion, the SEIS-II analyses do not specifically address off-site contaminant transport in the Dewey Lake.

H.9.2 Issues Related to Human Impacts and Intrusion

Scenarios analyzed in the *Preliminary Performance Assessment for the Waste Isolation Pilot Plant* (SNL 1992) and assumptions regarding deterrents to human intrusion have come under review, and several scenarios not yet analyzed have been proposed. This section presents discussions on an analyzed scenario and the application of institutional controls.

Presence/Absence of Borehole Casing in Intrusion Borehole

Neill et al. (1996) identified an apparent inconsistency in intrusion scenarios analyzed to date. One set of scenarios, in which contaminated brine from the repository flows up the borehole and through the Culebra, implies that there is no casing in the vicinity of the Culebra. Another scenario analyzes the effect of CUTTINGS_S generated during drilling and brought to the land surface, bypassing the Culebra. This implies, according to Butcher et al. (1995; cited in Neill et al. 1996), that a well casing is present. Current drilling technology in the Delaware Basin calls for steel casing from the surface to within 100 to 200 meters (330-660 feet) of the "top of the salt section" (Butcher et al. 1995; cited in Neill et al. 1996).

According to Neill et al. (1996), two different scenarios should be analyzed: one with casing and the other without, with assignment of probabilities of occurrence to each. This issue was not directly addressed in SEIS-II. In the intrusion scenario analyzed in SEIS-II, the borehole was designed to reflect current oil field practices. No explicit credit is taken for the presence of casing. It was assumed that the intrusion borehole is plugged and thereafter maintains a relatively low permeability. In the scenario, the borehole penetrates the entire sequence of units in the modeled domain. The borehole permeability was set initially to 1×10^{-10} square meters (1×10^{-9} square feet) to represent a relatively high borehole permeability for 100 years after the intrusion. After

200 years, the borehole permeability was decreased to 1 x 10^{-14} square meters (1 x 10^{-13} square feet) to reflect a decrease in permeability consistent with plugging the borehole with concrete.

Credit for Passive Institutional Controls

Passive institutional controls (PICs) were considered to be a sufficiently effective deterrent to human intrusion in the CCA (DOE 1996f), to the extent that the possibility of human intrusion was not included in the cumulative complementary distribution function. Neill et al. (1996) believe that no credit should be taken in the Complementary Cumulative Distribution Function for a reduced future (beyond 100 years) drilling frequency based on PICs. The SEIS-II intrusion scenario was analyzed at 400 years after closure for cases using median parameter values and 300 years after closure for cases using 75th percentile parameter values.

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APPENDIX I
LONG-TERM CONSEQUENCE ANALYSIS FOR NO ACTION ALTERNATIVE 2

I.1 INTRODUCTION

This appendix provides detailed information related to the consequence analysis for No Action Alternative 2, including background information on the scenarios analyzed, descriptions of the conceptual models of releases used, and data input parameters cited. Also provided are the specific analytical methods, computer codes, and exposure calculations used. Methods described include summaries of models and codes used for waste source-term release, contaminant transport, radiation dose, and chemical exposures. The report also provides a summary of human health impacts for the sites considered in the analysis.

I.1.1 Background

Long-term environmental consequence analyses were not performed for the no action alternatives in either the *Final Environmental Impact Statement for the Waste Isolation Pilot Plant* (FEIS) (DOE 1980) or the *Final Supplement Environmental Impact Statement for the Waste Isolation Pilot Plant* (SEIS-I) (DOE 1990). The impact analyses described in those documents focused on expected site operations associated with treatment and storage, with the assumptions that transuranic (TRU) waste would be indefinitely stored at 10 major generator sites and that institutional control would be sufficient to preclude any site impacts. In general, it is estimated that if effective monitoring and maintenance of storage facilities were provided, adverse health effects to the general public would be quite small, and the principal adverse effects, also small, would be related to occupational activity at the facility. Health effects would continue indefinitely at such levels under the hypothesis of the U.S. Department of Energy (DOE or the Department) control.

However, if the DOE were to lose institutional control of the storage facilities, it is estimated that intruders could receive substantial radiation doses, a situation that could persist for the indefinite future. In addition, contaminants in TRU wastes stored in shallow burial trenches and surface storage facilities will eventually be released and would persist in the surrounding environments at the generator-storage sites, exposing on-site and off-site populations to chronic health risks.

In the FEIS (DOE 1980) and SEIS-I (DOE 1990) analyses, DOE referenced completed National Environmental Policy Act documents for some of the major retrievable storage facilities to describe the effects of continued retrievable storage. These sites included the Hanford Site (Hanford), Idaho National Engineering and Environmental Laboratory (INEEL), and the Savannah River Site (SRS).

The Record of Decision on the FEIS, which was published on January 28, 1981, determined, as part of the basis for decision, that the No Action Alternative was “unacceptable.” This determination was made at the time because of the potential impacts of natural, low-probability events and human intrusion at storage facilities after governmental control of the site is lost. In SEIS-I (DOE 1990), a summary of the FEIS analysis was provided and the conclusion was again reached in the Record of Decision, published in June 21, 1990, that the No Action Alternative was unacceptable.

I.1.2 Purpose and Scope

In this *Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement* (SEIS-II), No Action Alternative 1 assumes that DOE would indefinitely maintain institutional control and, therefore, long-term impacts of post-closure intruders and environmental release were not assessed. No Action Alternative 2 assumes that TRU waste would not be emplaced at Waste Isolation Pilot Plant (WIPP) during the disposal phase, and, therefore, no radiological consequences to workers or the public would be realized in and around WIPP in this case. However, exposures would continue to occur at the major treatment facilities.

Under No Action Alternative 2, TRU waste is generated at all sites, including small-quantity sites, over the next 35 years. During this period, waste generated at the small-quantity sites would be consolidated and treated at the 10 major treatment sites, as described for this alternative in Chapter 3. Both consolidated and generated TRU waste will be put into retrievable storage consistent with current practices. Current storage configurations include soil-covered asphalt or concrete pads, shallow trenches, earthen berms, covered enclosures, storage buildings for contact-handled (CH) TRU waste, and buried caissons for remote-handled (RH) TRU waste. TRU waste would remain in these assumed storage configurations for an institutional control period of 100 years, beginning in 2033. During this period of institutional control, effective monitoring, surveillance, and maintenance would be expected to minimize the risk of contaminant release from the storage configurations.

At the end of the 100 years, following a TRU waste-generation period (i.e., 2133), institutional control is assumed to be lost. As facilities begin to degrade, TRU waste would be introduced into the accessible environment.

Calculations of the long-term consequences resulting from environmental releases from the storage facilities were performed for a 10,000-year period after the loss of institutional control. Environmental and human health impacts as a result of storage-facility releases were not evaluated for the period of institutional control.

Because 99 percent of the estimated TRU waste volume and inventory that would be generated can be accounted for at seven of the 10 major treatment sites (see Appendix A), environmental and human health impacts were estimated at these seven sites only: Hanford, INEEL, Lawrence Livermore National Laboratory (LLNL), Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL), Rocky Flats Environmental Technology Site (RFETS), and SRS. The three remaining sites not considered for this analysis were Argonne National Laboratory, the Mound Plant, and Nevada Test Site.

To the extent possible, this long-term consequence analysis for No Action Alternative 2 uses environmental data sets and models developed for low-level waste and low-level mixed waste consequence analyses conducted in the *Final Waste Management Programmatic Environmental Impact Statement* (WM PEIS) (DOE 1997b). The data sets and models were modified for assumed TRU waste inventories, storage site locations, and related environmental transport parameters, as appropriate. Data sources for this analysis include site descriptions and data provided in the following:

- WM PEIS (DOE 1997b)
- referenced contractor reports supporting the WM PEIS analysis (Holdren et al. 1995, Bergenback et al. 1995, and Blaylock et al. 1995)
- key site-specific environmental references
- *Transuranic Waste Baseline Inventory Report, Revision 3* (BIR-3) (DOE 1996b)

I.2 RISK ANALYSIS METHODOLOGY

The human health impacts of TRU waste were estimated for two types of exposures: (1) inadvertent human intrusion into areas of TRU waste storage and (2) source-term releases to surface and subsurface environmental exposure points.

Consistent risk measures were used to facilitate the comparison with the disposal alternatives. For radioactive substances, doses were estimated for the maximally exposed individual (MEI) and exposed populations for a 70-year lifetime period of highest dose and then expressed in terms of latent cancer fatalities (LCFs). For hazardous carcinogens, excess cancer incidence was calculated to the MEI and to the exposed population for a 70-year lifetime of highest exposure. For noncarcinogenic substances, the hazard index for the MEI for the highest period of exposure was estimated.

The following section provides the approaches used for the inadvertent human intrusion and the long-term environmental releases used in this analysis.

I.2.1 Inadvertent Human Intrusion Impacts

Inadvertent human intrusion into waste remaining at the sites may result in human health impacts. Two human intrusion scenarios were considered for both buried waste and surface-stored waste configurations. RFETS and LLNL do not have waste stored in shallow burial configurations, so buried waste intrusion scenarios were not evaluated at these two sites.

Buried waste intrusion scenarios include the driller and gardener scenarios, described below.

- **Driller.** A hypothetical intruder drills a well directly through buried or soil-covered TRU waste to underlying groundwater. As a result of the drilling, contaminated soil is brought to the surface and mixes with the topsoil. The circular drill hole was assumed to be 30 centimeters (12 inches) in diameter and 4 meters (13 feet) in depth, with the volume of waste removed by drilling instantaneously combined with the clean soil in the top 15 centimeters (6 inches) of the soil column. The extent of the contamination was limited to an area of 10 meters by 10 meters (33 feet by 33 feet). The driller would be exposed at the drill site over a five-day work week via external ground radiation for 40 hours, via inhalation of resuspended soil for one hour at a rate of 20 cubic meters (706 cubic feet) per day, and via inadvertent ingestion of soil for five days at a rate of 100 milligrams (3.5×10^{-3} ounces) per day. The soil resuspension was based on an average mass loading factor of 1.0×10^{-4} gram of soil per cubic meter (6.3×10^{-9} pounds per cubic feet) of air.

- **Gardener.** An individual farms a garden on the land containing the contaminated soil (following the driller intrusion) over a period of 30 years. During this time, 25 percent of the individual's yearly vegetable and fruit intake was assumed to be produced from this garden. The area of contamination was limited to that of the drill cuttings, assumed to be 100 square meters (1,090 square feet). In addition to food crops, the individual would be further exposed via inhalation of resuspended contamination, external radiation, and inadvertent ingestion of contaminated soil. The gardener was assumed to spend 12 hours a day working outside, thus exposed to the soil 4,383 hours per year. The gardener's inhalation exposure is 8,766 hours per year (at 24 hours per day) and the soil ingestion rate is 100 milligrams (3.5 x 10^{-3} ounces) per day for 365.25 days per year. Also, the gardener would ingest 14 kilograms (30.8 pounds) per year of leafy vegetables, 55 kilograms (121 pounds) per year of root vegetables, 31 kilograms (68 pounds) per year of fruit, and 73 kilograms (161 pounds) per year of grain, all assumed to have been grown in the contaminated region of his or her yard.

Surface-stored waste intrusion scenarios include the scavenger and farm family scenarios, described below.

- **Scavenger.** A hypothetical scavenger intruder comes into direct contact with surface-stored TRU waste over a 24-hour period. The scavenger is exposed via inhalation of resuspended contamination, external radiation, and inadvertent ingestion of contaminated soil while at the site. The scavenger does not ingest any food but is exposed via inhalation of resuspended soil (waste) and via external radiation. It is also assumed that no clean soil covers the waste, so the dose factors per unit concentration are multiplied by the waste form concentration to get the total dose to the scavenger.

- **Farm Family.** In this scenario, a hypothetical farm family of two adults and two children lives and farms on the land immediately over the former surface-stored TRU waste disposal area. The MEI in the family is exposed via ingestion of contaminated food crops, inhalation of resuspended contamination, external radiation, and inadvertent ingestion of contaminated soil.

Estimates of radiation dose were made using unit dose factors (dose per unit concentration of each contaminant in soil) developed for each intruder scenario and site using the GENII computer code (see Appendix F). Unit dose factors were multiplied by calculated concentration of each contaminant in relocated waste in the top 15 centimeters (6 inches) of soil to produce dose per contaminant. The contaminant concentrations for waste were developed using volume information from Table A-14 and contaminant inventory information from Tables A-40, A-42, A-45, A-49, and A-50. Doses from all of the radionuclides of concern were summed to yield the total effective dose equivalent for the intruders in each scenario. Calculated doses were then converted to LCFs using methods described in Appendix F. All intrusions were assumed to occur at the time institutional control would be lost (i.e., 2133), minimizing reduction of radionuclide activity by radioactive decay.

Impacts from hazardous chemicals were determined by estimating the total intake of each chemical. This intake was then compared to the slope factor for carcinogens and to the permissible exposure limits for noncarcinogens. Methods and reference values used for calculating hazardous chemical impacts are presented in Appendix F and Appendix G.

I.2.2 Impacts of Long-Term Environmental Releases

Populations and individuals living near sites where TRU waste would remain under No Action Alternative 2 may be impacted by long-term environmental releases of contaminants. The following two scenarios were used to evaluate impacts to the MEI from chronic long-term environmental releases.

- Groundwater Exposure. The MEI from a farm family was assumed to live 300 meters (980 feet) downgradient based on average groundwater flow of a TRU waste storage area. The family grows and consumes their own crops and livestock and uses contaminated groundwater as a source of drinking water. Contaminated groundwater is used for watering the crops and animals. This receptor was considered for long-term releases from buried or soil-covered TRU waste and surface-stored TRU waste.

- Air Pathway Exposure. A hypothetical individual was assumed to be exposed to the maximum airborne contaminant concentration released from the stored TRU waste site. This receptor, located at least 100 meters (330 feet) from the site but within an 80-kilometer (50-mile) radius, was only considered for long-term releases from surface-stored TRU waste.

Impacts to the off-site populations within 80 kilometers (50 miles) of the sites were assumed to be exposed via atmospheric transport of contaminants and/or by contamination of surface water (used only for drinking water) from releases to the ground water pathway. Population exposure from the groundwater/surface water pathway is applicable only for Hanford, ORNL, and SRS. Current population distributions were used for all sites. Long-term releases from both buried or soil-covered TRU waste and surface-stored TRU waste were included.

The Modular Risk Analysis (MRA) methodology used in the WM PEIS (DOE 1997b) was used for these analyses. Evaluation of the multimedia transport of radionuclides and chemical contaminants in air, surface water, and groundwater pathways was done using the MEPAS® code. The source-term-release component of the MEPAS® code was used to generate a specified release rate for each assumed TRU waste form. This was done to simulate the release of radionuclides and hazardous chemicals from TRU waste from a storage facility. MEPAS® calculates the annual flux rate of each contaminant released from a storage facility. Output from the MEPAS® source-term-release module was used as input for the MEPAS® transport module calculations.

Output for the hazardous chemical concentrations calculated by the MEPAS® transport module was used by MEPAS® risk components to calculate a cancer incidence and hazard index profile of over 10,000 years for chronic releases of both carcinogenic and noncarcinogenic hazardous substances, respectively.

The MRA methodology was developed by Pacific Northwest National Laboratory and Advanced Sciences, Inc. to facilitate regional-scale risk analysis. This methodology is described in several documents (Strenge and Chamberlain 1995; Whelan et al. 1995) and presentations (Whelan et al. 1994). The MRA methodology was developed for regional- and site-wide risk computations involving a large number of release sites with different TRU waste forms for various environmental settings and transport and exposure pathways.

The MRA methodology is based on the assumptions of linearity between the release-site source, the environmental transport, and the impacts at the receptor. By assuming the linearity of the system, the methodology can be divided into compartments that can be implemented both independently and concurrently. The compartments of the MRA methodology are (1) contaminant mass at the source, (2) determination of contaminant release rate from the source, (3) transport modeling of the contaminant into the environment (environmental concentrations at the receptor location), (4) exposure assessment for dose to receptor (MEI or population), and (5) estimation of impacts at the receptor.

The MRA methodology is based on the following general description for health impact:

$$\text{Health Impact} = P \times RF \times URF \qquad \text{(Equation I-1)}$$

where Health Impact is the estimated probability of adverse effects (carcinogenic risk for radionuclides and chemical carcinogens, and hazard quotient for noncarcinogenic constituents) from a contaminant at a receptor; P is the probability of the release event (unitless); RF is the releasable fraction of the source (unitless); and URF is the health impact associated with a contaminant at a receptor based on a unit quantity at the source. The UTF and UFF require the convolution of time series and those products can combine with the UDF by straight multiplication. URF expresses health impact (cancer incidence for radionuclides and chemical carcinogens, and hazard quotient for chemical noncarcinogens) and is determined as follows:

$$URF = [\,(Q \times UFF) \bullet UTF\,] \times UDF \times UIF \qquad \text{(Equation I-2)}$$

where Q is the estimated quantity of contaminant at the source in grams or curies; UFF is a time series of contaminant release rate fluxes designated as contaminant mass per time divided by unit contaminant mass; and UTF is a time series of environmental concentrations at a receptor produced from the UFF for groundwater, air, surface water, and soil media (expressed as contaminant mass per volume of medium divided by unit contaminant mass per time).

UDF is the dose to an organism from a unit concentration for a given exposure pathway. For chemicals, UDF is expressed as contaminant mass per body mass per time divided by unit contaminant mass per volume of contaminant in the environment at the receptor point. For radionuclides, UDF is expressed as contaminant total dose (rem) divided by unit contaminant mass per volume of contaminant in the environment at the receptor point.

UIF is the unit health impact factor that provides the dose conversion factor for radionuclides, cancer potency factor for chemical carcinogens, or reference dose for noncarcinogenic contaminants. For radionuclides, UIF is expressed as cancer fatalities divided by unit contaminant total dose. For chemical carcinogens, UIF is expressed as cancer incidence divided by unit contaminant mass per body mass per time. For chemical noncarcinogens, UIF is expressed as hazard quotient divided by unit contaminant mass per body mass per time.

The UFF and UTF are time series at different locations. The UFF is the time series of contaminant release rate from the source, while the UTF is the time series of contaminant concentration at the receptor point. The UTF and UFF require the convolution of time series and those products can combine with the UDF by straight multiplication. Equation I-2 provides the convolution method used to combine the series (the convolution operation is represented by the symbol $\bullet$).

Whereas Equation I-2 provides a description of the link between the different unit factors involved in computing the URF, Equations I-3 through I-6 provide a description of each of the unit factors that were developed to compute URFs. Note that the UFF and UTF are time series that must be convoluted together. The source, UDF, and UIF are multipliers.

$$UFF = \frac{F_s}{S_u} \quad \text{(Equation I-3)}$$

where S_u is the unit source mass (grams) or the unit source activity (curies) and F_s is the contaminant flux release rate from the TRU waste form expressed as mass per time. UFF includes the probability of release and the release factor fraction for a given scenario.

$$UTF = \frac{C_f}{F_u} \quad \text{(Equation I-4)}$$

where F_u is the unit contaminant flux rate expressed as mass per time and C_f is the contaminant concentration at the receptor based on transport through the appropriate media expressed as mass per volume.

$$UDF = \frac{D_c}{C_u} \quad \text{(Equation I-5)}$$

where C_u is the unit concentration at the receptor expressed as mass per volume and is based on contaminant transport through the appropriate media and D_c is the dose from the contaminant (for chemicals, D_c is expressed as mass of contaminant per body mass per time; for radionuclides, D_c is expressed as the total dose to a human receptor).

$$UIF = \frac{R_d}{D_u} \quad \text{(Equation I-6)}$$

where D_u is the unit dose to a human (for chemicals, D_u is expressed as mass of contaminant per body mass per time; for radionuclides, D_u is expressed as total dose). R_d is the health impact associated with a unit dose (for chemical carcinogens, R_d is expressed as cancer incidence divided by mass of contaminant per body mass per time; for chemical noncarcinogens, R_d is expressed as hazard quotient divided by mass of contaminant per body mass per time; and for radionuclides, R_d is expressed as cancer fatalities divided by total dose).

Average environmental conditions that are dependent on the TRU waste storage site were selected. Separate URFs and associated factors were developed for the different environmental settings of the seven major generator sites. Environmental settings were assumed to have homogeneous climatological, hydrologic, and geologic characteristics. Therefore, the URF is representative of the risk from a release site within a region and not an actual risk. The local climatological, hydrologic, and geologic characteristics for this analysis were developed using Holdren et al. (1995). The regional climatology (joint frequency distributions) and population for regional air receptors for the seven major generator sites were based on the WM PEIS methods described in detail in Bergenback et al. (1995). Details of the selection and application of these data and information for this analysis are described in Buck et al. (1997).

Conceptual site models were developed for each environmental setting associated with a storage site. These models defined the relationship between the source contaminant at the release sites and the health impacts at the receptors. The important components associated with these relationships were the constituents of interest, waste-source types, release mechanisms, exposure media, and receptor types. For this analysis, the probability of a release or exposure event was assumed to be 1.0. Likewise, it was assumed that sources were in a releasable form such that RF is equal to 1.0.

Once the waste configuration and TRU waste forms for each environmental setting were identified, the release mechanisms were identified. For this analysis, infiltration of contaminants to the vadose and groundwater system was considered to be the primary release mechanism. Volatilization, suspension, and overland flow release mechanisms were also considered. The resulting release rate (contaminant flux) for each release mechanism was also dependent on the TRU waste form. In addition, the solubility of each contaminant in the TRU waste form was an important factor in determining contaminant release rates. TRU waste forms listed in Appendix A were categorized to approximate the release of contaminants from the waste form into the environmental media. It was determined that all waste forms fell into one of two bounding waste form types, soil or cement. An analysis of the solubility limits for the primary TRU waste contaminants at each site was conducted. The results were incorporated into the computations of contaminant fluxes.

UFFs were generated using the MEPAS® model to simulate the release of contaminants from a source term. The model directly considers contaminated soil and solidified (cement) TRU waste forms. Contaminant is removed from the source by simultaneously evaluating degradation or decay, groundwater leaching, atmospheric volatilization, and soil erosion by wind suspension and overland runoff, as appropriate. To verify that contaminant release rates were not higher than the potential solubility associated with the TRU waste forms analyzed, Q (the estimated quantity of contaminant at the source) was included with the UFF to produce total flux factors. These were subsequently convoluted with the UTFs, thereby eliminating the contaminant release versus waste form solubility issue.

The UTF represented the environmental fate and transport component of the unit factor methodology. The UTF value was based on 1 gram or 1 curie of contaminant at the source which, after being transported through a specific environmental medium, ultimately arrives at the receptor exposure point. The receptor exposure point for groundwater analysis was a well 300 meters (984 feet) directly downgradient from the source, assumed centerline of the plume. The atmospheric receptor was located at the point of highest concentration that is at least 100 meters (328 feet) and within a radius of 80 kilometers (50 miles) from the release point. The UTF is media dependent. For example, UTF for air is expressed as milligrams per cubic meter per gram, or picocuries per cubic meter per curie; as milligrams per square meter per gram, or picocuries per square meter per curie for soil; and as grams per milliliter per gram, or curies per milliliter per curie for surface water and groundwater.

The UDF involves an average daily intake in milligrams per kilogram per day for chemicals or a lifetime radiation dose in rem for radionuclides. UIF relates the chemical intake or radiation dose to a risk or hazard index, as appropriate, for each pollutant. Both UIF and UDF are defined for intake or exposure routes of inhalation, ingestion, and external radiation.

Three different human health impact types were estimated, including exposure to carcinogenic radionuclides, carcinogenic chemicals, and noncarcinogenic chemicals. These impacts are directly related to the three types of UIFs computed for this analysis.

MEI receptors influence UDF calculations by defining dose intake factors and UTF calculations by defining the exact location of the receptor; as a result, MEI impacts were calculated and then used in the determination of population impacts. Equation I-7 provides the convolution method used to combine the series (the convolution operation is represented by the symbol $\bullet$).

$$URF = [Q \times UFF] \bullet UTF(\mathrm{t}) = \int_0^{\mathrm{t}} [Q \times UFF(t)] \times UTF(\mathrm{t} - t)dt \qquad \text{(Equation I-7)}$$

Once the convolution is completed, all the factors can be combined (based on Equation I-2) to provide health risk or hazard quotient impact values.

I.3 COMPUTER CODES

The potential health impacts from exposure to radioactive material and hazardous, nonradioactive material releases were evaluated with two computer codes. The MEPAS® code (described in Droppo et al. 1989 and 1991; Whelan et al. 1987; Strenge and Peterson 1989; and Buck et al. 1995) was used to assess contaminant transport and to calculate toxicological impacts and carcinogenic risks from hazardous constituents. GENII, described in Napier et al. (1988a, 1988b, 1988c), was used to calculate radiation dose from atmospheric releases and from radioactive material contamination trapped in soil. A brief discussion of the key components of the MEPAS® and GENII codes used in this analysis is presented in this section.

I.3.1 MEPAS® Code

The MEPAS® code integrates and evaluates transport and exposure pathways for chemicals and radioactive releases according to their potential human health impacts. MEPAS® is a physics-based approach that couples contaminant release, migration, and fate for environmental media with exposure routes and health consequences for radiological and nonradiological carcinogens and noncarcinogens.

Contaminant release from the waste zone was modeled with the source-term release component of MEPAS®. In general, the mass or activity of a contaminant in the source zone decreases over time because of contaminant removal by first-order degradation or radioactive decay, leaching to the groundwater, wind suspension, surface water erosion, and volatilization.

Radioactive and hazardous waterborne and airborne contaminant transport in multiple media were calculated using the transport components of the MEPAS® code. The MEPAS® waterborne transport code consists of: (1) groundwater, (2) surface water, and (3) overland transport models. These three transport models can either be run separately or linked to provide environmental concentrations at specified receptor points. For each waterborne transport pathway, contaminant retardation is described by an equilibrium coefficient, k_d. First-order degradation or decay is assumed for all contaminants that do not result in toxic decay products. For radionuclides in the waterborne transport pathway, parent contaminants are conservatively treated (i.e., not decayed) during transport through intermediate pathways. On reaching the environmental receptor point, radiological decay is corrected using the Bateman equation, and the code subsequently computes the temporal distribution of each decay progeny.

The MEPAS® atmospheric transport code considers the input of suspension and volatilization release rates to compute transport and dilution, washout by cloud droplets and precipitation, and deposition on the underlying surface cover. The atmospheric model uses climatological information on wind speed and direction, precipitation, and atmospheric stability (joint frequency distribution data) to compute average air and surface contamination concentrations. The atmospheric model also accounts for plume depletion from decay and deposition to ensure mass balance for the system. Contaminant transport was assumed to occur quickly enough so that chemical transformation can be omitted.

Results from the different transport pathway models were used as input to the exposure to calculate the human health impacts for each hazardous chemical contaminant. The following exposure routes were considered to determine the potential exposure to the MEI and the surrounding population: (1) dermal contact, (2) external exposure, (3) inhalation, and (4) ingestion. Each exposure route is evaluated to obtain an estimated average daily human exposure from each contaminant. The daily exposure rates are then converted, using mathematical codes, to average individual impact factors for carcinogenic and noncarcinogenic chemicals. Detailed information on the exposure component of MEPAS® can be found in Appendix F.

I.3.2 GENII Code

Although GENII models the environmental transport, contaminant accumulation, and radiation dose to an individual or population, it was used to calculate radiation doses under human intrusion scenarios. Methods for calculating doses with GENII are found in Appendix F.

I.4 WASTE CHARACTERISTICS

To model the health effects associated with TRU waste at the various DOE sites, the waste must be characterized in terms of its volume, contaminant inventory, and waste form. Appendix A presents the site-specific volumes (Table A-14) and contaminant inventories (Tables A-40, A-42, A-45, A-49, and A-50) used for No Action Alternative 2 analyses. The following discussion presents descriptions of TRU waste forms and waste form categories, the quantity of each waste-form category, and the contaminant inventory distribution of each TRU waste-form category used in this analysis.

I.4.1 Description of TRU Waste Forms and TRU Waste-Form Categories

TRU waste form characteristics can have large effects on the rate at which contaminants are released from the waste zone. TRU waste-form characteristics vary widely at the treatment sites but can be classified into one of two general waste-form categories. These waste-form categories and the modeling parameters associated with each are discussed below.

The first general waste-form category is comprised of unconsolidated TRU wastes. Because this waste is unconsolidated, it is assumed to be permeable. Thus, water percolating through a zone containing this category of waste would come into contact with all surfaces of the waste once waste contours are breached. This type of waste zone is also susceptible to wind suspension and water erosion processes. The release of contamination by leaching is regulated by the relative tendency of contaminants to exist in an aqueous phase or sorbed to solid surfaces. This partitioning between the aqueous and adsorbed phases is often expressed in terms of a surface-adsorption coefficient, k_d, and is dependent on the contaminant and the solid adsorbent material. If a contaminant is present in high-enough concentrations, the capacities of the aqueous and sorbed phases to contain the contaminant can be exceeded. Contaminant release, therefore, will be controlled by its solubility.

Unconsolidated TRU waste types in this first waste-form category can be further subdivided into those having either low or high surface-area-to-volume ratios. Examples of TRU waste with low surface-area-to-volume ratios and relatively low surface adsorption coefficients include waste containers, personal protective equipment (PPE), and metal process equipment. Contaminants from this type of TRU waste readily leach into the surrounding soil. Their release from the waste zone is controlled by the sorptive properties of the surrounding soil.

Unconsolidated TRU waste with high surface-area-to-volume ratios can have high surface adsorption coefficients. Contaminant release from these TRU waste types may be controlled by the sorptive properties of the TRU waste form itself. Examples of high surface area wastes include sludge, soil, and spent filters/adsorbents. Although the TRU waste form may control the release rate, physical data available for these waste types are limited, making it difficult to estimate sorption coefficients. For these types of TRU waste it is conservatively assumed, therefore, that the sorptive properties of the surrounding soil can also be used to determine the release characteristics of the high surface area TRU waste forms.

The second general waste-form category is comprised of solidified TRU waste whose permeability is much lower than that of the surrounding soil while sufficiently high to allow contaminant mobility within the TRU waste form. Percolating water tends to move around this category of waste and leach contaminants only from their exterior surface. The most common example of this TRU waste form is cemented waste. As contaminants are removed from the exterior surface, concentration gradients are established and contaminants tend to diffuse from the interior of the waste to the exterior surface. Therefore, the contaminant leaching release rate depends on the internal mobility of the contaminant, which is often expressed as an effective contaminant diffusion coefficient. Wind suspension and water erosion are assumed not to affect this solidified waste-form category until the TRU waste form fails.

In addition to contaminant-specific grout diffusion coefficients (Buck et al. 1997), the surface-area-to-volume ratio of the TRU waste form is required to model the release of contamination. It was assumed that waste will be disposed of in 55-gallon drums approximately 91 centimeters (36 inches) tall and 66 centimeters (26 inches) in diameter and that a cement slurry is poured directly into the drum, completely filling it. Using these assumptions, a surface-area-to-volume ratio of 0.082 cm^{-1} was calculated.

Another important parameter required for modeling contaminant release from this TRU waste-form category is the effective lifetime of the waste form. Cement degrades over time and will crack into small pieces such that the release of contamination is no longer limited to surface diffusion. At this point, contaminant release will be controlled more by surface adsorption and desorption than by diffusion. The effective lifetime of the TRU waste form depends on various properties of the cement, including the type of solidification agent used, solidification agent-to-waste-to-water ratios, curing conditions, waste composition, and storage environment conditions, such as the number of wet/dry and freeze/thaw cycles. An effective lifetime of 500 years was used for cemented TRU waste forms in this analysis.

I.4.2 Quantity of Each TRU Waste-Form Category

The relative quantities of each TRU waste-form category at a site were determined using information available in Appendix A and BIR-3 (DOE 1996b). BIR-3 specifies a waste volume and waste density for each of 10 waste material types (Table I-1). These waste material types were categorized into one of the general TRU waste-form categories modeled in this analysis.

Table I-1
Categorization of BIR-3 TRU Waste Materials into General Waste-Form Categories [a]

TRU Waste Material Type	General TRU Waste-Form Category
Iron-based Metals/Alloys	Soil/Debris
Aluminum-Based Metal/Alloys	Soil/Debris
Other Metals	Soil/Debris
Other Inorganic Materials	Soil/Debris
Cellulosics	Soil/Debris
Rubber	Soil/Debris
Plastics	Soil/Debris
Solidified, Inorganic Matrix	Cement
Solidified, Organic Matrix	Cement
Soils	Soil/Debris

[a] BIR-3 (DOE 1996b)

The total mass of each TRU waste material type at each site was calculated by summing the product of the waste volume and the waste density for each waste stream. The total mass of all TRU waste material was used to determine the weight percent of each type of waste material at the various sites.

CH-TRU and RH-TRU waste masses for the different waste material types within a general category were combined to determine the relative weight percent for each general TRU waste-form category. Because of the lack of supporting data, the densities of the cement and soil TRU waste forms were assumed equal. In this way, weight percent analysis could be applied directly to the total waste volume. This assumption has minor impact on the resulting information.

Under No Action Alternative 2, TRU waste would be located in the 200-East and 200-West areas at Hanford. Because the distribution between the two Hanford areas is not currently known, it was assumed that 50 percent of the total volume would be disposed of at each Hanford location. The final relative percentages for each general TRU waste-form category at each release site are shown in Table I-2. These relative quantities were multiplied by the total TRU waste volumes for the site (see Appendix A) to determine final site volumes for each TRU waste-form category. Volumes are also reported in Table I-2.

Table I-2
Relative Quantities and Volumes of Each TRU Waste-Form Category by Release Site

Release Site	Relative Quantities (percent)		Volumes (cubic meters)	
	Soil/Debris	Cement	Soil/Debris	Cement
Hanford Site - 200 East Area	100.0	0.0	43,500	0
Hanford Site - 200 West Area	100.0	0.0	43,500	0
Idaho National Engineering and Environmental Laboratory [a]	84.3	15.7	26,200	4,890
Lawrence Livermore National Laboratory	97.4	2.6	1,160	31
Los Alamos National Laboratory	63.8	36.2	13,600	7,700
Oak Ridge National Laboratory	76.1	23.9	3,610	1,140
Rocky Flats Environmental Technology Site	77.6	22.4	8,430	2,430
Savannah River Site	85.2	14.8	10,300	1,790

[a] INEEL values include waste volumes from ANL-W.

I.4.3 Contaminant Inventory Distribution of Each TRU Waste-Form Category

The radioactive and hazardous contaminant inventories used in the No Action Alternative 2 analysis, as discussed in Appendix A, are the total inventories present at each site for CH-TRU and RH-TRU waste. The inventories were not broken down by TRU waste material type. For the purpose of this analysis, the CH-TRU and RH-TRU waste inventories are added together and assumed to be distributed over the different TRU waste-form categories with the same relative ratios as the volume fractions. For example, if a site has 75 percent soil/debris and 25 percent cement TRU waste volume, the contaminant inventory was distributed 75 percent to soil/debris and 25 percent to cement.

I.5 WASTE CONFIGURATION AND CONTAMINANT RELEASE SCENARIOS

This section describes the basis for the source-term release analysis, which provided the contaminant flux factors in the MRA methodology described in Section I.2.2. Topics of this section include a general conceptual site storage model for buried or soil-covered TRU waste and surface-stored TRU waste, and assumptions governing the release of contaminants from the TRU waste-form categories.

The overall geometrical configuration of a waste storage zone, the assumed degradation of the waste storage zone, and the distribution of the TRU waste forms within the zone affect the magnitude and areal extent of the contaminant release fluxes from the zone. Each is discussed in this section.

I.5.1 Waste Storage Configuration

The following assumptions were made for the analysis of the buried or soil-covered TRU waste: (1) all TRU waste is contained in 55-gallon drums 91 centimeters (36 inches) tall and 66 centimeters (26 inches) in diameter that are stored together in one shallow burial zone, (2) four layers of drums are stacked on an asphalt or concrete pad with plywood sheets between the layers and on top, and (3) 1.2 meters (4 feet) of contaminant-free soil is used as backfill over the layers of drums. This overall configuration is illustrated in Figure I-1.

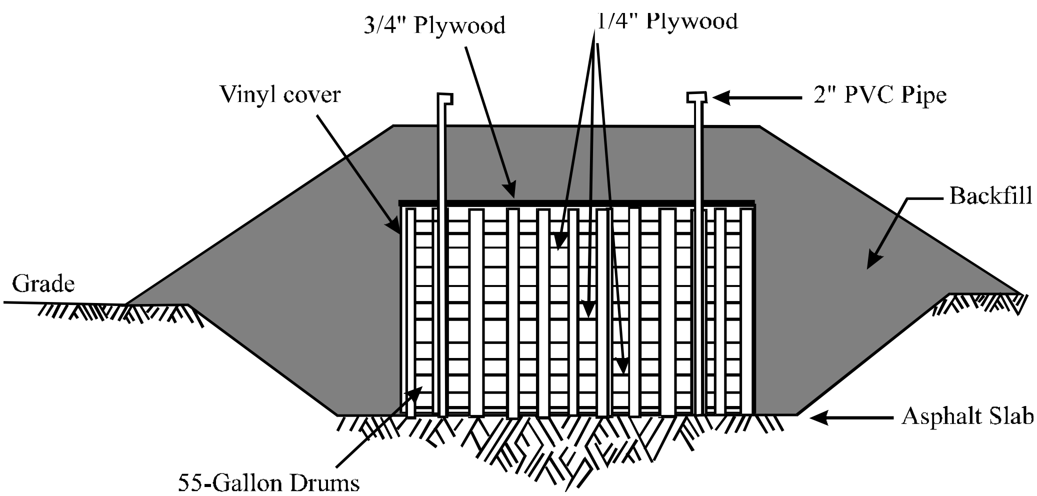

Figure I-1
Buried or Soil-Covered TRU Waste Storage Zone Configuration

For the analysis of surface-stored TRU waste, the initial waste zone configuration is similar to that for buried or soil-covered TRU waste. It is assumed that four layers of drums are stacked on an asphalt or concrete pad with plywood sheets between the layers and on top as illustrated in Figure I-2. However, instead of being placed in a buried or soil-covered configuration, the stacked drums are placed in metal storage buildings or covered storage areas.

The relative amount of material in a surface storage configuration versus a buried storage configuration was determined for each site using the following assumptions. First, the waste that is currently stored in either a surface configuration or buried waste configuration is not moved to a different configuration. Second, newly generated wastes are placed in a surface storage configuration. The relative amounts of waste in each configuration are shown in Table I-3.

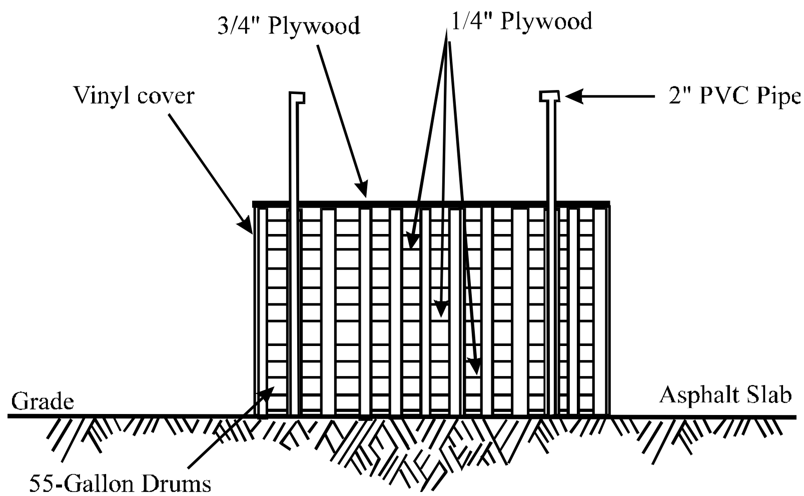

Figure I-2
Surface TRU Waste Storage Zone Configuration

Table I-3
Waste Zone Volumes, Horizontal Areas, and Configurations for Each Waste-Form Category by Release Site

	Waste Zone Volumes (cubic meters)		Waste Zone Horizontal Areas (square meters)		Waste Zone Configuration (Percent)	
Release Site	**Soil/Debris**	**Cement**	**Soil/Debris**	**Cement**	**Buried**	**Surface**
Hanford 200 East	55,300	0	15,200	0	13.6	86.4
Hanford 200 West	55,300	0	15,200	0	13.6	86.4
INEEL	33,400	6,230	9,150	1,710	49.9	50.1
LANL	17,300	9,800	4,740	2,690	37.9	62.1
LLNL	1,480	40	406	11	0	100
ORNL	5,350	1,680	1,470	460	15.2	84.8
RFETS	10,700	3,090	2,940	848	0	100
SRS	13,091	2,280	3,590	624	11.6	88.4

The vertical dimension of the waste zone for buried, soil-covered, and surface-stored configurations is approximately equal to 3.7 meters (12 feet). The drums are placed as close to one another as is possible in a rectangular grid arrangement. Because of this configuration, the volume of the waste zone will be larger than the volume of the TRU waste itself. The ratio of waste zone and waste drum volumes for a rectangular grid arrangement is:

$$\left[\frac{n \times (2r)^2 \times (4h)}{n\pi r^2 \times (4h)}\right] = \frac{4}{\pi} \qquad \text{(Equation I-8)}$$

where n is the number of drums, r is the drum radius, and h is the drum height. Waste zone volumes for each TRU waste-form category are calculated for each site by multiplying the corresponding volumes reported in Table I-2 by $4/\pi$. These overall waste zone volumes are reported in Table I-3. The horizontal cross-sectional areas of each TRU waste-form category for each site can be calculated by dividing the waste zone volumes by 3.7 meters (12 feet). These areas are also reported in Table I-3.

I.5.1.1 Facility and Waste Degradation

The ability of storage buildings, waste configuration components, waste containers, and TRU waste forms to contain contaminants needs to be considered when modeling the long-term release of contaminants. The following discussion provides an overview of each of these considerations in terms of their effect on the long-term release of contaminants from TRU wastes.

Facility Degradation

The surface storage scenario assumes that TRU waste is housed in metal storage buildings or a covered storage area. These buildings or covers will degrade relatively quickly as compared to the 10,000-year evaluation period, due chiefly to the lack of maintenance after the loss of institutional control. Therefore, metal storage buildings, enclosures, and covers are assumed to offer no protection and the surface storage scenario is modeled as if the stacked waste drums were not sheltered for the entire evaluation period. Facility degradation is not applicable to the buried or soil-covered TRU waste.

Waste Configuration Degradation

Components of the TRU waste form configuration are assumed to degrade quickly relative to the 10,000-year evaluation period. Degradation of the plywood or the storage building allows the spaces between the drums to fill with soil from the surface layer of the site. Likewise, once storage buildings degrade, the drums themselves will degrade at an accelerated rate, further altering the waste configuration.

It is also assumed that any asphalt or concrete pad at the base of the waste storage zone will be cracked or otherwise degraded for essentially the entire 10,000-year evaluation period. This allows infiltrating water to percolate through the waste zone, pass through the cracked or degraded pad, and move through the remainder of the vadose zone directly beneath the waste zone.

Waste Container Degradation

The integrity and longevity of the waste drums is also a factor in contaminant release from the waste zone. Both the surface-stored and buried or soil-covered scenarios assume that TRU waste will be contained in mild steel, U.S. Department of Transportation (DOT)-17C 55-gallon drums. Corrosion rates for mild steel drums are quite high even when buried in favorable, dry environments (i.e., drum lives of less than 100 years are expected). For surface storage facilities, once the storage enclosure or building degrades to the point where waste drums are directly exposed to the elements, stored TRU waste drums are expected to degrade more rapidly than TRU waste drums in a buried or soil-covered configuration. Because the expected life of the waste drums is relatively short compared to the 10,000-year evaluation period, no credit was given for the presence of containers during the evaluation period in the analysis.

Cemented TRU Waste Form Degradation

Initially, the cemented TRU waste form is assumed to be a solid block having the same size and shape as a 55-gallon drum. Cemented monoliths are known to crack and degrade into porous material over time. Unfortunately, the theory for modeling the transition from a solid block to porous material and its effect on contaminant release is not well developed. It is assumed, therefore, that the cemented waste blocks remain intact for the first 500 years and then catastrophically fail. After failure, the waste zone is assumed to act as a porous material.

I.5.1.2 Distribution of TRU Waste Forms

The horizontal cross-sectional area of the source zone is a required modeling input parameter. In reality, all drums of a given TRU waste-form category will not be emplaced in a single location within the waste zone. Rather, they will be interspersed with drums containing other TRU waste-form categories. To simplify the analysis, however, it is assumed that any specific area location contains drums of only one TRU waste-form category over the four vertical layers. It is also assumed that the waste zone is composed of a random distribution of “reasonably large” subareas of drums of only one TRU waste-form category. Each subarea is of sufficient size so that contaminant release is controlled by the physics and chemistry of that subarea’s TRU waste-form category alone. Therefore, contaminant release from the waste zone can be modeled in two parts (one for each waste-form category) using the conceptual mathematical models described and the appropriate fractional inventories and areas for each TRU waste-form category.

Contaminant mass flux is the output of source-term calculations for the subarea of each TRU waste-form category. Because the subareas are assumed to be uniformly dispersed throughout the waste zone, the mass flux of any contaminant from the two waste-form categories can be summed to determine the total mass flux of that contaminant over the cross-sectional area. Mass fluxes over the total waste zone area are required inputs for subsequent transport simulations.

I.5.2 Contaminant Release Scenarios

An overview of contaminant release scenarios from the different TRU waste forms for both the surface-stored and buried or soil-covered waste configurations is presented in this section. Geochemical controls that may limit the contaminant release from the overall waste zone are also discussed.

The overall rate of contaminant mass loss from the waste zone is the sum of the mass loss rates of five different loss processes. These processes are (1) decay, (2) leaching, (3) wind erosion, (4) water erosion, and (5) volatilization. The buried or soil-covered-waste scenario under No Action Alternative 2 assumes a 1.2-meter (4-foot) cover layer of soil that considerably reduces TRU waste interaction with surface erosion/dispersion mechanisms. By assuming that contaminant release to these mechanisms is zero, the contaminant inventories available for leaching increase, providing a maximized scenario for groundwater contamination. Leaching and decay, therefore, are the only two loss processes considered for the buried or soil-covered waste scenario.

A “multimedia” scenario was used for surface-stored releases. Water erosion, wind suspension, and volatilization were considered in addition to leaching and decay. This scenario assumes that there is no cover layer and contaminant transport by water erosion and wind suspension begins at the start of the analysis. Thus, the surface-stored scenario maximizes the potential air exposures. The effects of different TRU waste forms on each of the release mechanisms are discussed below.

I.5.2.1 Soil/Debris TRU Waste Form

When the waste zone is comprised of a soil/debris waste form, all five loss processes can occur. Degradation or decay is assumed to be a first-order process. Leaching is either solubility- or desorption-controlled. When there is no solubility-controlled solid phase, as with radionuclides and metals, or an organic liquid phase, as with organic chemicals, contaminant loss via leaching is assumed to occur by desorption-controlled transport. The velocity of the water percolating through the porous TRU waste form dominates this mode of transport. If the aqueous concentration of the contaminant is controlled by solubility, the mass flux is the product of the solubility of the contaminant and the volume of leachate passing through the waste zone. Water erosion and wind suspension are assumed to strip particles from the soil surface at a constant rate. These values were assumed to be zero for the buried or soil-covered waste scenarios to maximize leaching losses. Water erosion and wind suspension rates for the surface-stored scenario were calculated with MEPAS®.

Volatilization losses of organic contaminants were assumed to be zero in the buried or soil covered-waste scenario to establish a bounding case for groundwater contamination. In contrast, holes would develop in waste drums rather readily in the surface-stored scenario, causing most of the volatile organic inventory contaminants to be lost through volatilization. Therefore, the entire organic inventory is assumed to be released through volatilization during the first year, generating a maximized airborne scenario for volatile organic contaminants.

I.5.2.2 Cemented TRU Waste Form

The distribution of contaminants between different phases within the porous cement is not accounted for explicitly in the analysis. Decay and leaching are the only loss processes assumed to occur prior to failure. Decay of the overall contaminant mass is again assumed to be a first-order process. Infiltration water percolating through the waste zone is assumed to not penetrate the cemented TRU waste form. Rather, leaching loss results from percolating water flowing around the surface of the waste form and picking up contaminants as they diffuse through the water-filled pores of the cement. The cemented TRU waste form is assumed to fail after 500 years and, like the surface-stored scenario, it begins to act as a soil/debris TRU waste form and wind erosion, water erosion, and volatilization will begin.

I.5.3 Geochemical Controls on TRU Waste-Form Leaching

When the TRU waste form is cemented, leached contaminants do not immediately move out of the bottom of the waste zone. Under the assumed waste configuration, soil exists between each of the drum-shaped forms. Leached contaminants enter this soil zone before exiting the bottom of the waste zone with the percolating water. If the physical and chemical processes in this soil zone are such that contaminant leaching from the soil is slower than from the TRU waste form itself, this release process is the limiting step. The source-term release module, therefore, compares the leaching mass flux calculated from the cement TRU waste form with the leaching mass flux calculated under the assumption that the waste zone was composed of soil. The leaching mass flux used is the lower value of either the predicted desorption-controlled or the solubility-controlled soil release.

I.6 CONTAMINANTS OF CONCERN (CoC)

Initially, there were 141 radioactive, 47 organic, and 13 nonradioactive inorganic CoCs possible at the various treatment sites. To concentrate data collection efforts and analysis time on those CoCs that would contribute most to associated site hazards, a screening analysis was conducted. This analysis varied for radioactive, organic, and inorganic CoCs because of differing amounts of data available for each group. Data for the radioactive contaminants included site-specific radionuclide inventories. The screening analysis for this group of contaminants was divided into two possible transport pathways of concern: (1) airborne and (2) waterborne. A schematic of the screening process is shown in Figure I-3.

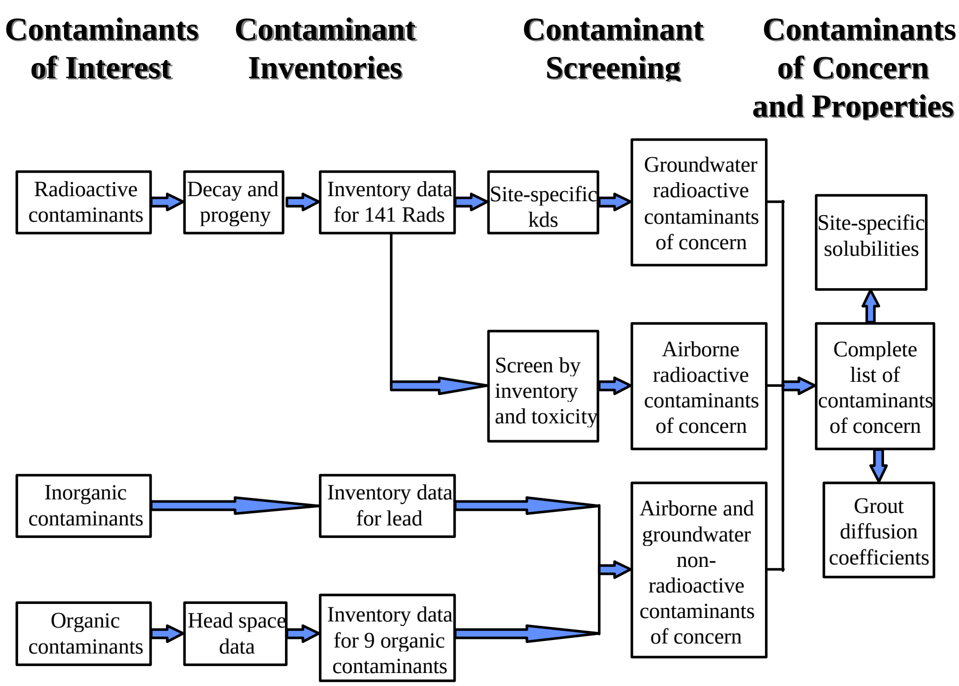

Figure I-3
Schematic of Screening Process for Contaminants of Concern

I.6.1 Radioactive Contaminant Screening Analysis

The first transport pathway to be considered was airborne contamination. It was assumed that unit amounts of waste, soil, or debris, with radioactive contaminant concentrations proportional to their inventories, were suspended by wind and transported through the air to a receptor. The human health impact resulting from this transport is, therefore, a function of the contaminant's inventory and inhalation dose factor. The relative impacts for each contaminant were compared and ranked according to their contribution to human health impact. The radionuclides with a combined risk equal to 90 percent of the total relative risk were designated as the airborne radioactive CoCs (Table I-4).

Table I-4
Contaminants of Concern for No Action Alternative 2

Contaminant	Type			
	Airborne Radioactive	Waterborne Radioactive	Inorganic	Organic
Am-241	x	---	---	---
Am-243	x	---	---	---
C-14	---	x	---	---
Cm-243	---	x	---	---
Cm-244	x	x	---	---
Cs-137	x	---	---	---
Eu-152	x	---	---	---
Np-237	---	x	---	---
Pa-233	x	---	---	---
Pu-238	x	x	---	---
Pu-239	x	x	---	---
Pu-240	x	x	---	---
Pu-241	x	---	---	---
Ra-226	x	---	---	---
Sr-90	x	---	---	---
U-233	---	x	---	---
U-234	---	x	---	---
U-235	---	x	---	---
Lead	---	---	x	---
Beryllium	---	---	x	---
Cadmium	---	---	x	---
Mercury	---	---	x	---
Carbon tetrachloride	---	---	---	x
Chloroform	---	---	---	x
Methylene chloride	---	---	---	x
1,1-Dichloroethylene	---	---	---	x
Methyl ethyl ketone	---	---	---	x
1,1,2,2-Tetrachloroethane	---	---	---	x
Toluene	---	---	---	x
Chlorobenzene	---	---	---	x
1,2-Dichloroethane	---	---	---	x
1,1,1-Trichloroethane	---	---	---	x

Leaching through the vadose zone to the groundwater was the second transport pathway to be considered. Contaminants must be present in sufficient quantities to result in health impacts through this pathway. Also, the site must have the necessary climatological and surface soil characteristics to percolate the amount of water needed to leach the contaminant from the waste zone and transport it through the vadose zone. The contaminant must then transport through the aquifer to a groundwater well, where it must be present in high enough concentrations and with sufficient toxicity to present consequential health impacts. Finally, radioactive contaminants must have long enough half-lives to sustain notable quantities of contaminant during the time required for transport.

To conduct this screening, slightly simplified MEPAS® runs were made. These runs utilized all of the release site data and assumed unit inventories for each contaminant. The release from the waste zone was assumed to be controlled by contaminant k_d values (i.e., the release was from a soil/debris TRU waste form and was not solubility limited). Transport through the environment was also controlled by the contaminant k_d values. Because of the importance of the assumed k_d values, all MEPAS®-generated radioactive contaminant k_d values were reviewed and modified with site-specific data, where available. These MEPAS® runs produced unit impact factors for each contaminant/site pair that was multiplied by the site-specific inventory to establish the estimated impact for each contaminant at each site. Relative impacts for each contaminant were again compared and ranked. The radioactive contaminants whose combined impact contributed 99 percent of the total relative impact were designated as waterborne CoCs (see Table I-4).

I.6.2 Inorganic Contaminant Screening Analysis

Reliable inventory data were generally not available for the inorganic contaminants. As a result, a qualitative screening method was employed.

Lead shielding is used to reduce surface dose rates to acceptable levels for RH-TRU waste containers; therefore, lead becomes a major part of the total waste mass and is included on the CoCs list. Lead from PPE is also a major contaminant in CH-TRU waste. The lead concentrations assumed in RH-TRU and CH-TRU waste are discussed in Appendix A.

Beryllium, cadmium, and mercury were also included on the list of CoCs, based on inventory estimates from the *Waste Isolation Pilot Plant Safety Analysis Report* fire scenario (DOE 1997a). Other inorganic contaminants, such as chromium, were not included on the CoCs list because of the lack of available inventory data. The assumed inorganic concentrations in RH-TRU and CH-TRU waste are discussed in Appendix A.

I.6.3 Organic Contaminant Screening Analysis

As with inorganic contaminants, little reliable inventory data were available for the organic contaminants. Some data reported in the *Final No-Migration Variance Petition* (DOE 1996a), however, could be used as an indirect indication of the volatile organic concentrations. The *No-Migration Variance Petition* summarized the results of a headspace sampling and analysis study conducted on TRU waste from the INEEL and RFETS. This study sampled 930 drums of varying waste types to determine a weighted-average headspace-gas composition that could be used for all TRU waste. The weighted values were screened using the concentration toxicity screening technique presented by the U.S. Environmental Protection Agency in the *Risk Assessment*

Guidance to Superfund, Volume 1, Human Health Evaluation Manual (Part A), (EPA 1989). This screening provided a list of carcinogenic and noncarcinogenic contaminants that account for over 99 percent of the human health impacts resulting from airborne contamination.

I.6.4 Key Contaminants Evaluated in No Action Alternative 2

The complete list of CoCs for No Action Alternative 2 is the combination of the waterborne radioactive, airborne radioactive, inorganic, and organic contaminants shown in Table I-4. The screening analyses resulted in a combined total of 32 CoCs. Once this list of CoCs was developed, an effort was made to obtain improved values for certain contaminant properties at specific sites. The contaminant diffusion coefficient in porous cement (required to model contaminant release from cement TRU waste forms) and contaminant solubility (required to model the solubility bounding case) were evaluated (Buck et al. 1997). Once the updated set of contaminant parameters was developed, actual waste zone contaminant release calculations were performed and flux factors for No Action Alternative 2 were generated.

I.6.5 Flux Factors

Source-term contaminant release calculations were run for all 32 CoCs for each site and for each waste-form category for a 10,000-year time period. The resulting contaminant flux factors were used to compute modular risk, which is represented as the flux factor term in Equation I-6. If a contaminant on the CoCs list was not present at a particular site, an inventory of zero was used for that calculation. Furthermore, if a particular TRU waste-form category was not present at a site, it was not considered in the flux-factor analysis. Table I-5 shows the number of nonzero flux factors produced by site and TRU waste-form category.

Table I-5
Nonzero Contaminant Flux Factors by Site and TRU Waste-Form Category

Release Site	Number of Nonzero Flux Factors	
	Soil/Debris	**Cement**
Hanford Site 200 East Area	30	---
Hanford Site 200 West Area	30	---
Idaho National Engineering and Environmental Laboratory	32	32
Lawrence Livermore National Laboratory	28	28
Los Alamos National Laboratory	31	31
Oak Ridge National Laboratory	32	32
Rocky Flats Environmental Technology Site	25	25
Savannah River Site	29	29

I.7 WATERBORNE AND AIRBORNE TRANSPORT

The transport portion of the impact analysis required specific information related to waterborne transport, airborne transport, and receptor locations. Each area is discussed below.

I.7.1 Waterborne Transport Parameters

Parameters related to the hydrologic and geologic characteristics of each site analyzed were selected from site-specific environmental settings developed for the WM PEIS (DOE 1997b) and Holdren et al. (1995). The environmental settings assumed for each of major treatment site analyzed are summarized in Buck et al. (1997). The references listed above contained the number of vadose zone layers at each site, the thicknesses and hydraulic conductivity properties of the vadose zone and aquifer layers, and a suite of physicochemical properties for all layers.

I.7.2 Calculated Waste Infiltration Rates

Flux factor calculations required additional site-specific parameters, in order to determine the waste infiltration rates needed for contaminant release calculations. Table I-6 shows MEPAS®-calculated water infiltration and soil erosion rate values for each site. The calculated rates are based on local climatology (i.e., precipitation, cloudiness, wind speed, and humidity), surface soil properties, and vegetation cover.

Table I-6
MEPAS®-Calculated Water Infiltration and Soil Erosion Rates for Each Site

Category	Hanford		INEEL	LANL	LLNL	ORNL	RFETS	SRS
	200E	200W						
Water Infiltration (centimeters/year)	1.49	1.49	1.43	0.663	9.28	42.2	0.156	24
Annual Precipitation (centimeters)	16	16	22	45	37	139	39	110
Percent Precipitation to Infiltration	9	9	6	1.4	25	32	0.4	22
Wind Suspension Erosion Rate (centimeters/year)	5.8E-04	5.8E-04	0.045	1.3E-03	6.1E-03	3.5E-05	7.7E-03	2.0E-03
Soil Eroded by Wind in 10,000 Years (feet)	0.19	0.19	1.48	0.44	2.00	0.01	2.53	6.59
Overland Flow Erosion Rate (centimeters/year)	1.9E-04	1.9E-04	3.4E-03	5.2E-03	0.014	0.012	6.5E-04	1.1E-02
Soil Eroded by Water in 10,000 Years (feet)	0.06	0.06	1.11	1.70	4.06	3.79	0.21	3.45
Total Soil Erosion in 10,000 Years (feet)	0.25	0.25	1.59	2.14	6.06	3.81	2.74	4.10

I.7.3 Airborne Transport Parameters

Parameters related to atmospheric release, transport, and deposition analyses required surface soil characteristics and regional climatological information. The surface soil and regional climatological information required to estimate soil suspension rates were obtained from Holdren et al. (1995). The regional meteorological data and atmospheric dispersion data, in the form of a joint frequency distribution of wind direction, wind speed, and atmospheric stability, were obtained from the WM PEIS (DOE 1997b).

I.7.4 Calculated Soil Erosion Rates

Flux factor calculations required additional site-specific parameters in order to determine the soil erosion rates needed for contaminant release calculations. Table I-6 showed MEPAS®-calculated water infiltration and soil erosion rates for each site. Wind suspension and overland waterflow soil erosion rates were computed using site-specific surface soil and local climatological information. Table I-6 also provided an estimate of the amount of surface soil removed over the 10,000-year modeling period to determine whether the 1.2-meter (4-foot) overburden could be removed to expose waste to the surface.

I.7.5 Air and Water Receptor Locations

Population impacts from atmospheric releases were calculated for all No Action Alternative 2 sites using site-specific joint frequency and population data. Population impacts from domestic and agricultural surface water uses were calculated for some of the sites. The atmospheric population distributions were obtained for all the sites from the WM PEIS (DOE 1997b). Results of a review of the annual site reports and specific recommendations regarding population exposure for each site are summarized in Buck et al. (1997).

I.8 CALCULATION OF UNIT EXPOSURE AND UNIT IMPACT FACTORS

The human health impact analysis for SEIS-II requires definition of the unit dose factor (UDF) and the unit impact factor (UIF). The UDF relates average daily intake (milligrams per kilogram per day) for chemicals and lifetime radiation dose (rem) for radionuclides. The UIF relates intake or dose to impact or hazard index for each pollutant, as appropriate. The UIF and UDF are defined for contaminant inhalation and ingestion and for external radiation exposure. The following sections describe the calculation of UDF and UIF from the health impact endpoint values provided in the MEPAS® output files for both individual and population exposures. Parameter arrays can also be calculated from information in the MEPAS® output files. Background information and the scope of the analysis are summarized first.

I.8.1 Background Information and Scope of Analysis

SEIS-II project analysis requires calculation of population health impacts as well as individual health impacts. The UIF values for individual and population exposures are the same and provide conversion from intake or dose to impact. These values are based on slope factors, reference doses, and radiation dose conversion factors. The UDF values differ from UIF values, in that the population UDF is evaluated using average parameter values instead of 90th percentile values. The number of people exposed is not included in UDF or UIF values; however, the population exposed must be included in the final analysis of impact, because populations are defined for each release site and receptor location. The UDF and UIF values are independent of release site and receptor location.

Many combinations of variables must be used to generate the UDF and UIF. These variables are described below.

I.8.1.1 Exposure Scenario

Analyses are performed for an MEI and for local populations. Both scenarios involve the potential exposure to air, soil, and waterborne contamination.

I.8.1.2 Receptor Type

Like the exposure scenario, MEI and local population receptor types are evaluated for No Action Alternative 2. Each receptor type requires the generation of specific UIF/UDF files; they cannot be combined into one calculation.

I.8.1.3 Exposure Media

Each analysis is performed with the appropriate exposure media for the MEI and local population scenarios. Types of exposure media include soil per unit mass, soil per unit area, air, and groundwater.

I.8.1.4 Pollutant Type

Specific output files are generated for each type of pollutant: noncarcinogenic chemicals, carcinogenic chemicals, and radionuclides. As with previous analyses, the list of chemicals in the two chemical file types is identical. All chemicals are analyzed for carcinogenic risk and noncarcinogenic hazard quotient. Chemical types that are not appropriate for a specific chemical will result in a zero health impact result.

I.8.2 Individual UDF and UIF Calculations

The results from individual UDF and UIF calculations provide the cancer incidence risk and hazard index values for the following parameters: each exposure pathway, each pollutant, one scenario, one pollutant type, and one set of up to 20 pollutants. These parameters are referred to as unit exposure factors. The file also contains the slope factors and reference doses used for chemicals in the analysis. MEPAS® assimilates the data and a postprocessor program extracts the UDF and UIF from each output file, as necessary, for subsequent calculations. For each set of results, the UDF values are summed over the specific values for each exposure pathway within an exposure route. The calculation output is a set of UDF values for each pollutant, calculated from the risk/hazard quotient values that are described in more detail in Buck et al. (1997). For total cancer fatalities, the UIF is set to 5 x 10^{-4} LCFs per rem.

I.8.3 Population UDF and UIF Calculations

Population UDF and UIF values are calculated according to the equations for individual UDF and UIF values (Equation I-5 and I-6). Population UDF, risk and hazard quotient values in the output files, however, must be taken from files generated specifically for population exposures. The population UIF value should be numerically equal to the individual UIF value.

I.9 HUMAN HEALTH IMPACTS OF NO ACTION ALTERNATIVE 2

This section provides estimates of human health impacts from stored TRU waste at the major generator-storage sites following loss of institutional control under No Action Alternative 2.

Impacts are presented for scenarios of human intrusion into the waste, and for scenarios of long-term environmental release from the waste via the atmosphere and groundwater/surface water. These scenarios are described in Sections I.2.1 and I.2.2.

The impacts to human health from waste intrusion and long-term environmental release were estimated using methods outlined in this appendix and described in more detail in Buck et al. (1997). This analysis focused on the impacts of waste at the seven major treatment sites, because the majority (99 percent) of the wastes generated would be stored at these sites under No Action Alternative 2. Estimates of impacts from RH-TRU waste were made only for those sites storing RH-TRU waste (i.e., Hanford, INEEL, LANL, and ORNL).

I.9.1 Impacts from Intrusion into Wastes

Human health impacts from waste intrusion were evaluated under scenarios for buried waste storage and surface-stored waste. Buried waste storage scenarios include those for a driller and gardener, while surface-stored waste scenarios include those for a scavenger and a farm family. The following sections present radiological and hazardous chemical impacts to hypothetical intruders from buried and surface-stored wastes.

I.9.1.1 Intrusion into Buried Waste

With the loss of institutional control, an inadvertent intruder could become directly exposed to waste stored in shallow burial facilities. The driller scenario postulates that an individual would drill into the waste and become exposed to waste material brought to the land surface by the drilling process. The gardener scenario assumes that an individual would farm in soil contaminated by the waste materials brought to the surface from the driller scenario and would ingest contaminated materials and eat produce from the garden. The results of analyses performed for these scenarios for both CH-TRU and RH-TRU wastes are presented below.

The estimated maximum dose to a hypothetical driller from exposure to CH-TRU WASTE ranged from about 2.2×10^{-3} to 0.01 rem, corresponding to a maximum probability of an LCF occurring in the intruder of 1.1×10^{-6} to 5.4×10^{-6} for the five sites with buried wastes (Table I-7). The estimated maximum dose to a hypothetical driller from RH-TRU wastes ranged from 2.2×10^{-3} to 0.058 rem, corresponding to a maximum probability of an LCF occurring in the intruder of 1.1×10^{-6} to 2.9×10^{-5} for the four sites that store RH-TRU wastes (Table I-8).

The estimated maximum 30-year dose for the gardener exposed to CH-TRU waste ranged from 19 to 126 rem, corresponding to a maximum probability of an LCF occurring in the intruder of 9.6×10^{-3} to 0.063 LCFs (Table I-7). The estimated maximum 30-year dose to the gardener from RH-TRU wastes ranged from 6.1 to 89 rem, corresponding to a maximum probability of an LCF occurring in the intruder of 3.6×10^{-3} to 0.045 (Table I-8). The highest estimated dose for CH-TRU wastes was calculated for INEEL for the driller scenario and for SRS for the gardener scenario. The highest estimated dose for RH-TRU wastes was calculated for Hanford for both scenarios.

Table I-7
Radiation Dose and Hazardous Chemical Impacts from Buried CH-TRU Waste Intrusion Scenarios

Radiological Impacts					
Radionuclide	**Hanford**	**INEEL**	**LANL**	**ORNL**	**SRS**
Driller Impacts (rem)					
Sr-90	6.1E-7	1.3E-8	4.9E-8	1.1E-5	1.3E-8
Y-90	8.5E-7	1.7E-8	6.1E-8	1.5E-5	1.7E-8
Cs-137	8.2E-5	5.3E-06	6.4E-6	2.4E-3	1.8E-6
Pu-238	2.9E-4	1.1E-4	4.6E-4	1.6E-4	4.1E-3
Pu-239	2.2E-4	1.5E-4	7.5E-4	2.2E-4	1.6E-4
Pu-240	5.2E-5	3.6E-5	9.8E-7	7.2E-5	4.0E-5
Am-241	1.5E-3	0.010	3.3E-3	7.6E-3	3.0E-3
Cm-244	2.4E-7	7.9E-7	6.1E-7	3.2E-5	8.3E-6
Total Dose (rem)	2.2E-3	0.011	4.5E-3	0.010	7.3E-3
LCF Probability	1.1E-6	5.4E-6	2.3E-6	5.2E-6	3.2E-6
Gardener Impacts (rem) - 30 years					
Sr-90	0.022	4.5E-4	1.6E-3	0.40	9.5E-4
Y-90	2.2E-3	4.3E-5	1.6E-4	0.048	4.5E-5
Cs-137	0.094	6.1E-3	7.4E-3	2.7	2.1E-3
Pu-238	8.3	3.1	13	4.4	115
Pu-239	6.7	4.6	23	6.6	4.9
Pu-240	1.6	1.1	30	2.2	1.2
Am-241	2.5	17.2	5.5	12	5.0
Cm-244	2.9E-4	9.8E-4	7.5E-4	0.039	0.010
Total Dose (rem)	19	26	41	29	126
LCF Probability	9.6E-3	0.013	0.021	0.015	0.063

Hazardous Chemical Impacts		
Driller Impacts		
Hazardous Chemical	**PEL**	**Cancer Incidence**
Cadmium	9.8E-2	1.4E-9
Beryllium	17	1.3E-7
Lead	27	
Mercury	12	
Gardener Impacts – 30 years		
Hazardous Chemical	**Hazard Quotient**	**Cancer Incidence**
Cadmium	0.01	2E-5
Beryllium	0.08	1E-4
Lead	CH: 36	
Mercury	77	

LCF = Latent Cancer Fatality

Table I-8
Radiation Dose and Hazardous Chemical Impacts from Buried RH-TRU Waste Intrusion Scenarios

Radiological Impacts				
Radionuclide	**Hanford**	**INEEL**	**LANL**	**ORNL**
Dose to Driller (rem)				
Sr-90	3.5E-04	1.4E-04	1.4E-05	1.5E-04
Y-90	4.8E-04	1.9E-04	2.0E-05	2.1E-04
Cs-137	0.051	8.9E-03	2.1E-03	6.2E-03
Pu-238	1.0E-05	3.4E-06	1.9E-06	8.0E-07
Pu-239	1.7E-04	1.7E-05	1.0E-4	6.3E-06
Pu-240	8.5E-05	6.7E-06	0	1.1E-06
Am-241	5.5E-03	6.2E-04	0	3.6E-04
Cm-244	0	1.3E-05	0	1.6E-05
Total Dose (rem)	0.058	9.9E-03	2.2E-03	7.0E-03
LCF Probability	2.9E-05	4.9E-06	1.1E-06	3.5E-06
Dose to Gardener (rem)				
Sr-90	12	4.9	0.51	5.5
Y-90	1.2	0.49	0.051	0.55
Cs-137	59	10	2.4	7.1
Pu-238	0.29	0.097	0.052	0.023
Pu-239	5.2	0.52	3.1	0.19
Pu-240	2.6	0.20	0	0.033
Am-241	9.0	1.0	0	0.59
Cm-244	0	0.016	0	0.020
Total Dose (rem)	89	17	6.1	14
LCF Probability	0.045	8.7E-03	3.6E-03	7.0E-03

Hazardous Chemical Impacts		
Driller Impacts		
Hazardous Chemical	**PEL**	**Cancer Incidence**
Cadmium	9.8E-02	1.4E-09
Beryllium	17	1.3E-07
Lead	3,000	
Mercury	12	
Gardener Impacts		
Hazardous Chemical	**Hazard Quotient**	**Cancer Incidence**
Cadmium	0.01	2.0E-05
Beryllium	0.08	1.0E-04
Lead	3,900	
Mercury	77	

LCF = Latent Cancer Fatality

Carcinogenic health impacts from hazardous chemicals, including volatile organic compounds (VOC), were found to be negligible compared to radiological impacts (see Tables I-7 and I-8). The air concentration for the driller was 12, 17, and 27 times the permissible exposure limit (PEL) for mercury, beryllium, and lead, respectively, for CH-TRU waste; for RH-TRU waste, lead concentration in the air was 3,000 times the PEL. The 30-year gardener impacts ranged from 0.01 to 77 for CH-TRU waste and from 0.01 to 3,900 for RH-TRU waste. The hazard index for mercury was 77 for both CH-TRU and RH-TRU waste, while the hazard index for lead was 3,900 for RH-TRU waste and 36 for CH-TRU waste.

I.9.1.2 Intrusion into Surface-Stored Wastes

With the loss of institutional control, inadvertent intruders would be more likely to come into direct contact with waste in surface storage facilities than with buried waste. To estimate this impact, exposure calculations were performed for a hypothetical scavenger intruder in contact with surface-stored wastes during a 24-hour period at loss of institutional control. The scavenger was assumed to be exposed via inhalation of resuspended contamination and external and inadvertent ingestion of contaminated soil while at the site.

The estimated maximum dose to a hypothetical scavenger exposed to surface-stored CH-TRU wastes ranged from about 1.3 to 37.6 rem, corresponding to a maximum probability of an LCF occurring in the intruder of 6.4×10^{-4} to 0.02 (Table I-9). The estimated maximum dose to the scavenger from surface-stored RH-TRU wastes ranged from about 1.4 to 24.5 rem, corresponding to a maximum probability of an LCF occurring in the intruder of 6.9×10^{-4} to 1×10^{-2} (Table I-9). For CH-TRU wastes, the highest doses were estimated at RFETS. The highest doses for RH-TRU wastes were estimated for Hanford and INEEL.

Another potential intruder scenario involves a hypothetical family (2 adults and 2 children) that lives and farms on a plot of land immediately over the surface-stored waste, where the waste has degraded and become indistinguishable from the surrounding land. For these conditions, the maximally exposed intruder in the family could be exposed via ingestion of contaminated food crops grown in the contaminated soil, inhalation of resuspended contamination, external exposure to the soil, and inadvertent ingestion of contaminated soil. If this scenario occurred, the four-member family would receive high annual radiation dose equivalents over the 30-year exposure period (i.e., in excess of 400 rem per year). For the first year of farming the calculated maximum probability of an LCF occurring for this scenario ranges from 0.24 to greater than 1 (6.9) for CH-TRU waste and from 0.27 to greater than 1 (5.1) for RH-TRU waste. Table I-10 presents estimated radiation doses and resulting LCFs for all sites.

Health impacts due to hazardous chemicals would be significant following the loss of institutional control for surface-stored waste. For VOCs, the cancer incidence would be less than 1.0 for both the scavenger and 30-year farming scenarios. During the 24 hours the scavenger is on the site, the air concentration of heavy metals could be as much as 5 to 91 times the PEL for cadmium, mercury, and beryllium for CH-TRU waste and RH-TRU waste and 1,400 and 160,000 times the PEL for lead for CH-TRU waste and RH-TRU waste, respectively. Cancer incidence for the scavenger would be negligible.

Table I-9
Radiation Dose and Hazardous Chemical Impacts from Scavenger Intrusion into Surface-Stored Wastes

Radionuclide	Hanford	INEEL	LLNL	LANL	ORNL	RFETS	SRS
CH-TRU Waste Dose (rem)							
Sr-90	4.9E-05	1.0E-06	0	3.5E-06	8.9E-04	0	1.0E-06
Y-90	3.3E-04	6.8E-06	0	2.4E-05	6.0E-03	0	6.8E-06
Cs-137	0.03	2.2E-03	4.5E-09	2.7E-03	1.00	0	7.7E-04
Pu-238	1.37	0.51	0.07	2.16	0.73	0.22	19.01
Pu-239	1.05	0.71	0.35	3.52	1.03	12.04	0.76
Pu-240	0.24	0.17	0.14	0.00	0.34	4.83	0.19
Am-241	0.42	2.81	0.73	0.89	2.05	20.51	0.81
Cm-244	6.4E-05	2.1E-04	3.3E-03	1.6E-04	0.01	0	2.2E-03
Total Dose (rem)	3.11	4.22	1.29	6.58	5.19	37.60	21
LCF Probability	1.6E-03	2.1E-03	6.4E-04	3.3E-03	2.6E-03	0.02	0.01
RH-TRU Waste Dose (rem)							
Sr-90	0.03	0.01	N/A	1.1E-03	0.01	N/A	N/A
Y-90	0.19	0.07	N/A	7.7E-03	0.08	N/A	N/A
Cs-137	21.51	3.74	N/A	0.89	2.63	N/A	N/A
Pu-238	0.05	0.02	N/A	8.6E-03	3.7E-03	N/A	N/A
Pu-239	0.81	0.08	N/A	4.8E-01	0.03	N/A	N/A
Pu-240	0.40	0.03	N/A	0	0.01	N/A	N/A
Am-241	1.47	0.17	N/A	0	0.10	N/A	N/A
Cm-244	0	3.4E-03	N/A	0	4.3E-03	N/A	N/A
Total Dose (rem)	24.46	4.13	N/A	1.39	2.86	N/A	N/A
LCF Probability	0.01	2.1E-03	N/A	6.9E-04	1.4E-03	N/A	N/A

Hazardous Chemical Impacts

CH-TRU Hazardous Chemical	PEL	Cancer Incidence
Cadmium	5.2	2E-6
Beryllium	91	2E-4
Lead	CH: 1400/RH: 160,000	
Mercury	6.2	

LCF = Latent Cancer Fatality
N/A = Not Applicable

Table I-10
Radiation Dose and Hazardous Chemical Impacts
for First Year of Farmer Intrusion into Surface-Stored Wastes

Radionuclide	Hanford	INEEL	LLNL	LANL	ORNL	RFETS	SRS
CH-TRU Waste Impacts Dose (rem)							
Sr-90	7.5	0.16	N/A	0.56	140	0	0.18
Y-90	0.50	0.01	N/A	0.036	9.0	0	0.01
Cs-137	7.2	0.50	9.4E-7	0.57	210	0	0.17
Pu-238	500	190	26	800	270	80.92	7.0E+03
Pu-239	360	250	120	1.2E+03	360	4.2E+03	260
Pu-240	84	59	47	1.60	120	1.7E+03	64
Am-241	160	1.1E+03	280	340	780	7.9+03	310
Cm-244	0.02	0.08	1.23	0.06	3.2	0	0.84
U-233	0.54	2.6	7.6E-8	0.34	10	0.94	0.052
Total Dose (rem)	1.1E+03	1.6E+03	475	2.4E+03	1.9E+3	1.4e+04	7.7E+03
LCF Probability[a]	0.56	0.79	0.24	1 (1.2)	0.95	1 (6.9)	1 (3.8)
RH-TRU Waste Impacts Dose (rem)							
Sr-90	4.4	1.7E+03	N/A	180	1.9E+03	N/A	N/A
Y-90	280	110	N/A	12	120	N/A	N/A
Cs-137	4.5	780	N/A	190	550	N/A	N/A
Pu-238	18	5.91	N/A	3.18	1.37	N/A	N/A
Pu-239	281	28.00	N/A	167.01	10	N/A	N/A
Pu-240	139	11	N/A	0	1.80	N/A	N/A
Am-241	564	64	N/A	0	37	N/A	N/A
Cm-244	0	1.29	N/A	0	1.62	N/A	N/A
U-233	0.53	0.04	N/A	6.8E-4	6.2E-3	N/A	N/A
Total Dose (rem)	1.0E+4	2.7E+03	N/A	550	2.7E+03	N/A	N/A
LCF Probability[a]	1 (5.1)	1 (1.4)	N/A	0.27	1 (1.3)	N/A	N/A

Hazardous Chemical Impacts

Hazardous Index		Cancer Incidence
Cadmium	15	0.02
Beryllium	10	1.9
Lead	CH: 50,000/RH: 5.2E+6	
Mercury	100,000	

[a] Numbers in parentheses represent the actual calculated probability of an LCF.

LCF = Latent Cancer Fatality
N/A = Not Applicable

The hazard index to the MEI farmer would range from 10 to 5,200,000. The greatest indices would be for lead with 5,200,000 and 48,000 for RH-TRU waste and CH-TRU waste, respectively, and for mercury with 100,000 for both waste types.

Repeated farm family intrusion over a long period of time such as the 10,000-year impact period evaluated in Section I.9.2 could result in a large aggregate number of intrusion fatalities. As shown in Table I-10, the probability of a latent cancer fatality is very high at all sites, with some calculated values being 1.0 during the first lifetime after loss of institutional control. In the event of population pressure that resulted in individuals occupying the sites of former waste storage facilities, the number of LCFs from exposure to TRU waste could reach several thousand, assuming continuous occupation of the sites and four LCFs every 50 years, while the calculated probability of an LCF to an individual remained 1 or higher.

I.9.2 Impacts of Long-Term Environmental Release

Contaminants in TRU wastes stored in shallow burial trenches and surface storage facilities within site-specific environmental settings would eventually be released to the surrounding environments at the generator-storage sites. Contaminants within the buried or surface-stored wastes would be leached and released to underlying soils and aquifer systems at depth. Eventually, at most sites, contaminants would reach groundwater and migrate laterally to downgradient receptor locations. Contaminants may also eventually be discharged into nearby surface-water bodies, particularly at the Hanford, ORNL, and SRS sites. Once in these surface-water systems, dilute concentrations of the contaminants would become available to the public in nearby communities.

Wastes stored in surface facilities would also degrade and become available in the environment as a result of cyclic and ongoing processes, such as direct water and air erosion, deposition onto soils surrounding the site, and resuspension of contaminated soils in air. The general surrounding on-site and off-site populations would be exposed to contaminants redistributed into the environment by these processes.

For this analysis, the impacts to the hypothetical farm family living 300 meters (980 feet) downgradient of the waste storage area were estimated. It was assumed that the family would engage in farming activities such as growing and consuming their own crops and livestock. The family would use contaminated groundwater as a source of drinking water for themselves and the animals and for watering the crops. The MEI would be exposed via ingestion of food crops grown in the contaminated soil, inhalation of resuspended contamination, external exposure to the soil, and inadvertent ingestion of contaminated soil. This analysis also considered the off-site population that could potentially be exposed to environmental releases to the surface water and air. For analyses of buried waste releases, all CH-TRU and RH-TRU waste was combined into a single waste disposal unit, and only the groundwater pathway was considered. For analyses of surface-stored waste releases, all CH-TRU and RH-TRU wastes were combined into a single waste storage unit and were allowed to be released to all pathways.

The population impacts from exposure to radiation and hazardous chemicals were estimated based on current storage-site population distributions. At present, most sites are remote from large population centers. While it is unknown how populations at the sites will change over the next 10,000 years, populations may increase substantially over present day levels and encroach onto sites and locate near storage facilities. The latter point is certainly implied for the inadvertent human intrusion scenarios discussed above in Section I.9.1. Thus, the potential for additional impacts may increase and potential long-term radiological and hazardous chemical impacts could be considerably higher (i.e., an order of magnitude or more) than those estimated in this analysis.

I.9.2.1 Radiological Impacts

The estimated lifetime (70 years) radiological impacts over 10,000 years, as a result of environmental contaminant releases from buried and surface-stored wastes at the seven generator-storage sites, are presented in Table I-11. The MEI maximum lifetime radiation dose for all sites ranged from 5.2 x 10^{-3} to 7.8 rem per 70-year lifetime. Resultant maximum probability of an LCF ranged from 2.6 x 10^{-6} to 3.9 x 10^{-3}. The highest estimated probability of an LCF occurred at INEEL. The dose to the MEI at each site for each 70-year lifetime over 10,000 years is shown in Figure I-4.

Table I-11
Maximum Lifetime MEI and Population Impacts at Seven Major Generator-Storage Sites after Loss of Institutional Control for No Action Alternative 2

	Radiological Impacts			Chemical Carcinogenic Impacts	
Site	Lifetime Dose (rem/70 years)	Lifetime LCF [a]	Dominant Pathway	Lifetime Cancer Incidence	Dominant Pathway
MEI Impacts					
Hanford	0.3	1.2E-4	Inhalation	2.3E-6	Groundwater Ingestion
INEEL	7.8	3.9E-3	Groundwater Ingestion	5.4E-3	Groundwater Ingestion
LANL	0.09	4.5E-5	Inhalation	2.4E-4	Resuspended Soil Ingestion
LLNL	0.01	6.9E-6	Inhalation	1.1E-7	Groundwater Ingestion
ORNL	5.0E-3	2.6E-6	Groundwater Ingestion	5.9E-7	Groundwater Ingestion
RFETS	4.9	2.5E-3	Inhalation	3.1E-7	Groundwater Ingestion
SRS	1.4	6.5E-4	Groundwater Ingestion	3.7E-4	Groundwater Ingestion
Population Impacts					
Hanford	1.4	6.8E-4	Resuspended Soil Ingestion	1.5E-6	Surface Water Ingestion
INEEL	149	0.07	Inhalation	2.9E-6	Resuspended Soil Ingestion
LANL	162	8.1E-2	Inhalation	2.4E-4	Resuspended Soil Ingestion
LLNL	30	1.5E-2	Inhalation	1.1E-7	Resuspended Soil Ingestion
ORNL	0.07	3.6E-5	Inhalation	2.7E-7	Surface Water Ingestion
RFETS	14,200	7.1	Inhalation	2.8E-4	Resuspended Soil Ingestion
SRS	175	8.8E-2	Inhalation	5.6E-5	Surface Water Ingestion

[a] Lifetime LCF is the probability of an LCF for an MEI and the number of LCFs in a population.

The radiological impacts to populations over 70-year lifetimes were estimated from the air and surface water exposure pathways for all sites. The groundwater pathway was determined to not be a notable source of off-site drinking water, so it was not considered in the estimation of off-site population impacts. The estimated maximum lifetime population dose for all sites (see Table I-11) ranged from 0.07 to 14,200 person-rem. The resultant LCFs ranged from 3.6 x 10^{-5} to 7.1. The highest impact was estimated to occur in the population around RFETS, where the calculated number of LCFs was over an order of magnitude higher than any other site. The dose to the population around each site for each 70-year lifetime over 10,000 years is shown in Figure I-5.

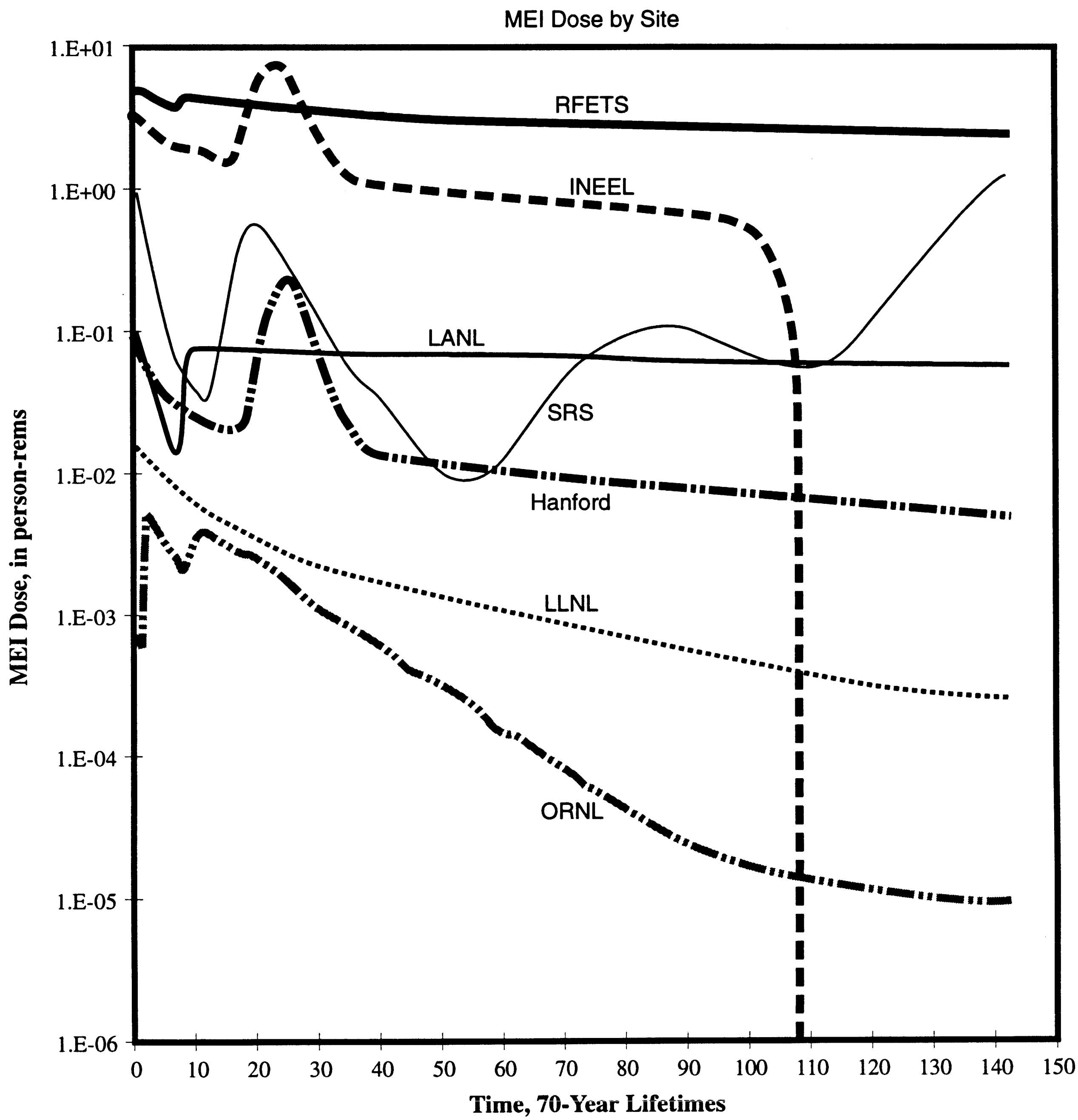

Figure I-4
Lifetime Doses to the MEIs at Seven Major Generator-Storage Sites over 10,000 Years Following Loss of Institutional Control Under No Action Alternative 2

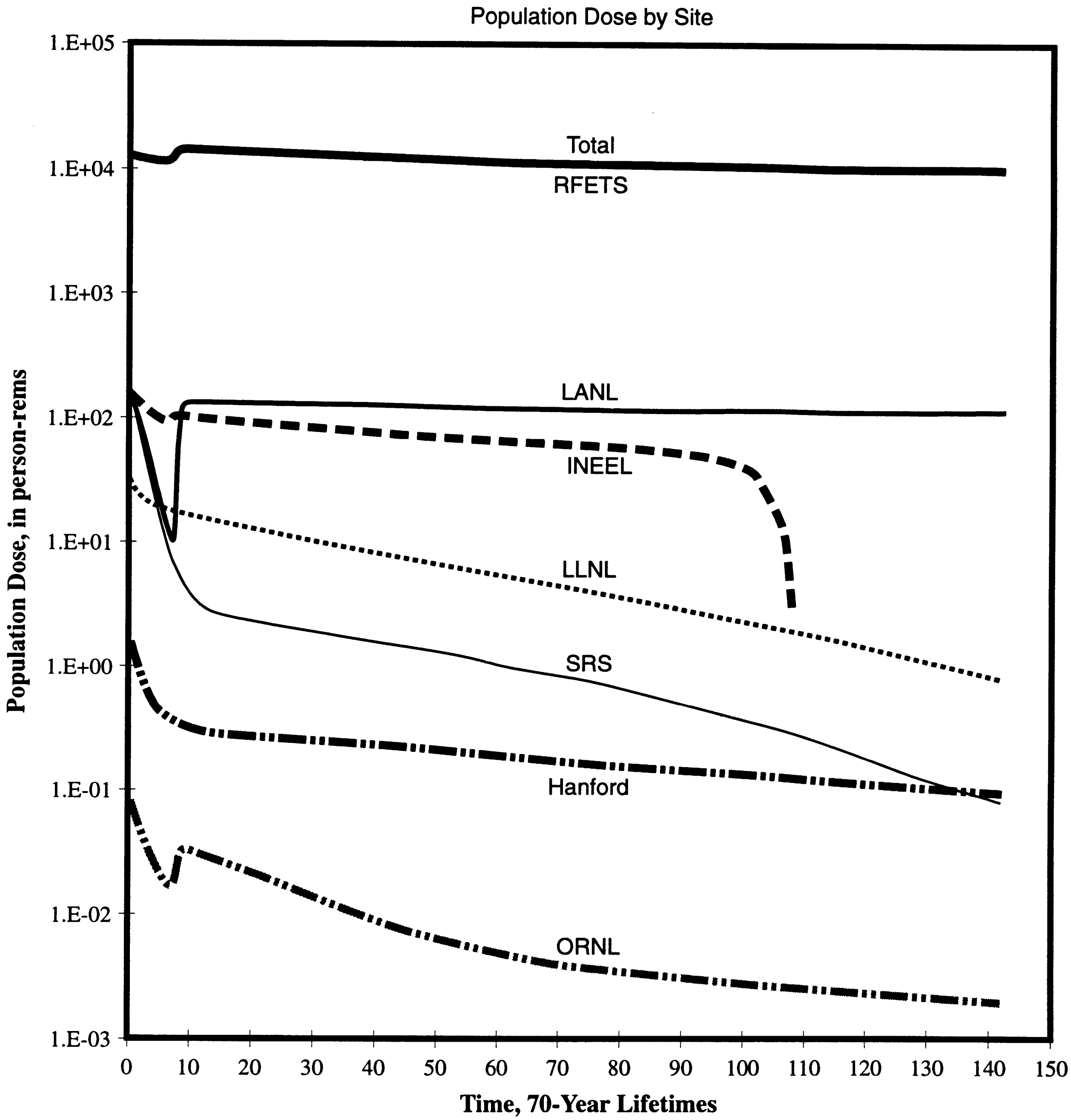

Figure I-5
Lifetime Population Doses for Seven Major Generator-Storage Sites over 10,000 Years Following Loss of Institutional Control Under No Action Alternative 2

The aggregate number of LCFs that could occur in off-site populations around the seven sites over 10,000 years (~ 142 70-year lifetimes) from release of the No Action 2 inventory was estimated by summing for each site the estimated LCFs for each lifetime, then summing the aggregate 10,000-year LCFs for the seven sites to provide the total estimated LCFs over 10,000 years. The aggregate number of LCFs was estimated to be 794 LCFs, with 98 percent of the estimated LCFs (781) occurring in the population around RFETS. In addition to the impact from release of the No Action Alternative 2 inventory, the number of aggregate LCFs at the seven sites was estimated for the Additional Inventory of Action Alternative 1 (see Tables 3-2 and 3-3) which would also remain in place at the sites under the No Action 2 Alternative. An additional 13 aggregate LCFs were estimated to occur from release of the Additional Inventory. Release of the combined inventories would result in about 800 LCFs. Estimates of site-specific and total aggregate LCFs are presented in Table I-12. As noted above these impacts were estimated based on current population distributions. These distributions may change substantially, creating the potential for significant increases over these estimates of aggregate LCFs.

Table I-12
Aggregate LCFs over 10,000 Years for Seven Major Generator-Storage Sites after Loss of Institutional Control for No Action Alternative 2

Site	Aggregate LCFs from Basic Inventory	Aggregate LCFs from Additional Inventory	Aggregate LCFs from Combined Inventory
Hanford	1.4E-02	1.0E-02	2.4E-02
INEEL	3.8E+00	7.7E+00	1.14E+01
LANL	8.5E+00	5.6E+00	1.41E+01
LLNL	4.5E-01	0	4.5E-01
ORNL	6.3E-04	3.0E-04	9.3E-04
RFETS	7.81E+02	0	7.81E+02
SRS	2.8E-01	1.1E-01	4.0E-01
Total	794	13	807

I.9.2.2 Chemical Carcinogen Impacts

The estimated maximum lifetime cancer incidence from chemical carcinogens to a MEI (see Table I-11) ranged from 1.3 x 10^{-7} to 5.4 x 10^{-3} for the seven sites. The highest cancer incidence was estimated for INEEL. The predominant impacts at nearly all sites, except LANL, resulted from ingestion of groundwater containing 1,1,2,2-tetrachloroethane.

The estimated maximum lifetime cancer incidence for exposed populations (see Table I-11) was estimated to range from 1.1 x 10^{-7} to 2.8 x 10^{-4} for the seven sites. The aggregate cancer incidence for the seven sites over 10,000 years was estimated to be 0.002.

I.9.2.3 Noncarcinogenic Impacts

The noncarcinogenic impact from chemicals was very low compared to the radiological impact. Maximum hazard indices for an MEI from exposure to noncarcinogenic chemicals for the seven sites ranged from 6.4 x 10^{-5} to 1.5 (see Table I-13). The highest hazard index was estimated for the SRS. The predominant impacts at all sites resulted from ingestion of either mercury or carbon tetrachloride except for RFETS where the impact was from inhalation of lead.

Table I-13
Noncarcinogenic Impacts to a Maximally Exposed Individual at Seven Major Generator-Storage Sites after Loss of Institutional Control for No Action Alternative 2 [a]

Site	Maximum Hazard Index	Key Chemical	Dominant Pathway
Hanford	0.2	Mercury	Groundwater Ingestion
INEEL	0.3	Carbon Tetrachloride	Groundwater Ingestion
LANL	1.7E-3	Mercury	Resuspended Soil Ingestion
LLNL	5.3E-4	Mercury	Resuspended Soil Ingestion
ORNL	2.8E-4	Mercury	Groundwater Ingestion
RFETS	6.4E-5	Lead	Inhalation
SRS	1.5	Mercury	Groundwater Ingestion

[a] The best available data was used to calculate concentrations of VOCs and mercury, but DOE recognizes that the data may not reflect actual usage of hazardous chemicals at all sites. Conservative assumptions, such as that all waste is mixed waste, were used to ensure the impacts stated in this document reflect a reasonable upper limit of likely impacts.

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APPENDIX J
UPDATED ESTIMATES OF THE DEPARTMENT OF ENERGY'S TRANSURANIC WASTE VOLUMES

The analyses presented in Chapter 5 of this second supplemental environmental impact statement (SEIS-II) are based on the transuranic (TRU) waste volume estimates published in 1996 in the *Transuranic Waste Baseline Inventory Report*, *Revision 3* (BIR-3) (DOE 1996a). The same waste volume estimates were used for the *Waste Isolation Pilot Plant Supplemental Draft Environmental Impact Statement* (Draft SEIS-II).

Since the completion of the Draft SEIS-II, though, the U.S. Department of Energy (DOE or the Department) has published *The National Transuranic Waste Management Plan* (DOE 1996b), which includes updated estimates of the stored and projected volumes of defense TRU waste at the generator and storage sites.[1] These more recent estimates include a 4 percent increase in the total volume of contact-handled (CH) TRU waste and an 86 percent decrease in the total volume of remote-handled (RH) TRU waste. The differences in the waste volumes presented in the BIR-3 and *The National Transuranic Waste Management Plan* are largely due to changes throughout the DOE Complex in the schedules for those remediation and decontamination and decommissioning (D&D) activities that may generate TRU waste during the next 35 years. Overall, the volumes in BIR-3 result in a more conservative estimate of the impacts of disposal at the Waste Isolation Pilot Plant (WIPP) and were chosen for the analyses in Chapter 5. This appendix has been prepared to allow a comparison of impacts had the volumes presented in *The National Transuranic Waste Management Plan* been the basis for the analyses presented in Chapter 5.

Table J-1 provides a comparison of the CH-TRU waste volumes in *The National Transuranic Waste Management Plan* and in the Basic Inventory (which is based on BIR-3). There are a number of differences in the site-specific volumes, as noted in the "percent difference" columns of Table J-1. Under the *National Transuranic Waste Management Plan*, there are major increases in waste volumes at Idaho National Engineering and Environmental Laboratory (INEEL) and Rocky Flats Environmental Technology Site (RFETS), while large decreases are reported for the Hanford Site (Hanford) and Los Alamos National Laboratory (LANL). Because the differences between TRU waste volumes at some sites differ significantly, a particular site's impacts could vary to a greater extent than indicated by the differences in the total volumes.

A comparison of RH-TRU waste volumes is provided in Table J-2. *The National Transuranic Waste Management Plan* reports more than 30,000 cubic meters (1,059,000 cubic feet) less of RH-TRU waste than the Basic Inventory. A major decrease in RH-TRU waste was reported at Hanford, where TRU waste from environmental restoration activities is now expected to be generated after the 35-year waste generation period that was evaluated.

A comparison of the total waste volumes that would be disposed of at WIPP under each of the action alternatives is shown in Table J-3. CH-TRU waste volumes using *The National Transuranic Waste Management Plan* increase slightly (2 to 5 percent), while RH-TRU waste volumes decrease dramatically (82 to 90 percent).

[1] *The National Transuranic Waste Management Plan* does not include estimates of radionuclide or hazardous chemical inventories.

Table J-1
Comparison of CH-TRU Waste Volumes in Basic Inventory and *The National Transuranic Waste Management Plan*

Site	NTRUWM Plan [a]			Basic Inventory [b]			NTRUWM Plan, Volume Difference and Percent Difference from Basic Inventory					
							Stored		To Be Generated		Total Inventory	
	Stored	Projected	(Total)	Stored	Projected	Total	Volume	Percent	Volume	Percent	Volume	Percent
Hanford Reservation (Hanford)	16,407	9,251	25,658	12,300	45,190	57,490	4,107	133%	-35,939	20%	-31,832	45%
Los Alamos National Laboratory (LANL)	7,770	9,259	17,029	11,050	9,980	21,030	-3,280	70%	-721	93%	-4,001	81%
Idaho National Engineering and Environmental Laboratory (INEEL)	65,102 [c]	81	65,183 [c]	28,150	0	28,150	36,945	231%	-924	8%	36,021	224%
Argonne National Laboratory - West (ANL-W)				7	1,005	1,012	in INEEL		in INEEL		in INEEL	
Savannah River Site (SRS)	9,165	3,773	12,938	2,880	9,180	12,060	6,285	318%	-5,407	41%	878	107%
Rocky Flats Environmental Technology Site (RFETS)	1,043	14,741	15,784	4,890	5,970	10,860	-3,847	21%	8,771	247%	4,924	145%
Oak Ridge National Laboratory (ORNL)	1,303	256	1,559	1,320	350	1,670	-17	99%	-94	73%	-111	93%
Lawrence Livermore National Laboratory (LLNL)	249	905	1,154	230	960	1,190	19	108%	-55	94%	-36	97%
Nevada Test Site (NTS) [d]	623	12	635	620	10	630	3	100%	2	120%	5	101%
Mound Plant (Mound)	239	12	251	300	0	300	-61	80%	12		-49	84%
Argonne National Laboratory - East (ANL-E)	83	12	95	25	180	205	58	332%	-168	7%	-110	46%
Bettis Atomic Power Laboratory (BAPL)	0	123	123	0	170	170	-		-47	72%	-47	72%
Sandia National Laboratories (SNL-NM)	7	6	13	7	10	17	-	100%	-4		-4	
Paducah Gaseous Diffusion Plant (PGDP)	2	0	2	0	8	8	2		-8		-6	
U.S. Army Material Command (USAMC)	3	0	3	3	0	3	-	100%	-		-	
Energy Technology Engineering Center (ETEC)	2	0	2	2	0	2	-	100%	-		-	
University of Missouri Research Reactor (U of Mo)	1	1	2	1	0	1	1	100%	1		-	
Pantex Plant (Pantex)	1	0	1	1	0	1	1	100%	-		-	
Ames Laboratory (Ames)	0	1	1	0	1	1	-		1		-	
Teledyne Brown Engineering (TBE)	1	0	1	1	0	1	1	100%	-		-	
Battelle Columbus Laboratories (Battelle)	0	0	0	0	0	0	-		-		-	
ARCO Medical Products Company (ARCO)	1	0	1				1		-		1	
Knolls Atomic Power Laboratory (Knolls)	0	0	0				-		-		-	
Lawrence Berkeley Laboratory (LBL)	1	1	2	.			1		1		2	
General Electric-Vallecitos Nuclear Center (GE-VNC) [e]	5	4	9				5		4		9	
Babcock & Wilcox (B&W) [e]	18	0	18				18		-		18	
West Valley Demonstration Project (WVDP) [f]												
Totals	102,025	38,437	140,462	61,787	73,014	134,801	40,242		-34,575		5,662	104%

[a] *National TRU Waste Management Plan* (DOE 1996b). Values < 1 are shown here as 1.
[b] Unrounded Basic Inventory values.
[c] Includes 27,000 cubic meters of alpha low-level waste. INEEL may also have 2,500 cubic-meters of commercial TRU waste.
[d] NTS may have an additional waste volume of approximately 120 cubic meters, composed of classified TRU waste and previously disposed of TRU waste not reported in BIR-3.
[e] GE-VNC and B&W are included only in the *National TRU Waste Management Plan*.
[f] WVDP is not included in either the *National TRU Waste Management Plan* or the Basic Inventory. WVDP has waste in the Additional Inventory.

Table J-2
Comparison of RH-TRU Waste Volumes in Basic Inventory and *The National Transuranic Waste Management Plan*

	NTRUWM Plan [a]			Basic Inventory [b]			NTRUWM Plan, Volume Difference and Percent Difference from Basic Inventory					
							Stored		To Be Generated		Total Inventory	
Site	Stored	Projected	Total	Stored	Newly Generated	Total	Volume	Percent	Volume	Percent	Volume	Percent
Hanford	200	2,420	2,620	200	29,200	29,400	-	0%	-26,780	-92%	-26,780	-91%
LANL	94	136	230	90	130	220	4	+4%	6	+5%	10	+5%
INEEL	86	53	139	-	-	-	66	+330%	-1,667	-97%	-1,601	-92%
ANL-W				20	1,720	1,740	in INEEL		in INEEL		in INEEL	
SRS	-	-	-									
RFETS	-	-	-									
ORNL	962	193	1,155	2,470	600	3,070	-1,508	-61%	-407	-68%	-1,915	-68%
LLNL	-	-	-									
NTS [c]	-	-	-									
Mound	-	-	-									
ANL-E	-	-	-									
Bettis	-	2	2	-	9	9	-		-7	-78%	-7	-78%
SNL-AL	1	2	3									
PGDP	-	-	-									
USAMC	-	-	-									
ETEC	6	1	7	6	1	7	-	0%	-		-	0%
U of M	-	-	-									
Pantex	-	-	-									
Ames	-	-	-									
TBE	-	-	-									
Battelle	581	-	581	580	-	580	1	0%	-		1	0%
ARCO	-	-	-									
Knolls	6	1	7									
LBL	-	-	-									
GE-VNC [d]	5	8	13									
B&W [d]	-	-	-									
WVDP [e]												
Totals	1,941	2,816	4,757	3,366	31,660	35,026	-1,437	-42%	-28,855	-91%	-30,292	-86%

[a] *The National TRU Waste Management Plan* (DOE 1996b).
[b] Unrounded Basic Inventory values.
[c] Includes approximately 1 cubic meter of RH-TRU waste.
[d] GE-VNC and B&W are included only in *The National TRU Waste Management Plan*.
[e] WVDP is not included in either *The National TRU Waste Management Plan* or the SEIS-II Basic Inventory. WVDP has waste in the SEIS-II Additional Inventory.

Table J-3
Comparison of Waste Volumes Among Alternatives Using
***The National Transuranic Waste Management Plan* and Basic Inventory Volumes**

	Proposed Action [a]		Action Alternative 1		Action Alternative 2		Action Alternative 3	
	CH-TRU	**RH-TRU**	**CH-TRU**	**RH-TRU**	**CH-TRU**	**RH-TRU**	**CH-TRU**	**RH-TRU**
Post-Treatment Disposal Volumes (cubic meters) for the Total Inventory								
NTRUWM Plan [b]	168,500	7,080	291,000	7,180	109,200	2,510	343,100	8,620
BIR-3 Based	168,500	7,080	281,000	55,000	107,000	19,000	334,000	66,000
Difference, cubic meters:	0	0	10,000	-47,820	2,200	-16,490	9,100	-57,380
NTRU percent difference	0%	0%	+4%	-87%	+2%	-87%	+3%	-87%
Pre-Treatment Consolidated Volumes (cubic meters) for the Total Inventory								
NTRUWM Plan [b]	168,500	7,080	279,700	7,180	280,400	7,180	279,700	7,180
BIR-3 Based	168,500	7,080	273,000	39,000	274,000	39,000	273,000	39,000
Difference, cubic meters:	0	0	6,700	-31,820	6,400	-31,820	6,700	-31,820
NTRU percent difference	0%	0%	+2%	-82%	+2%	-82%	+2%	-82%

[a] All volumes for the Proposed Action were adjusted to 168,500 cubic meters for CH-TRU waste and 7,080 cubic meters for RH-TRU waste.
[b] Alternative waste volumes in *The National TRU Waste Management Plan* waste inventory were substituted for the Basic Inventory.

Potential Changes in Estimated Impacts

The Proposed Action (Preferred Alternative) presents impacts adjusted to the treatment and disposal of 168,500 cubic meters (5,950,000 cubic feet) of CH-TRU waste (the maximum allowed under the Land Withdrawal Act [LWA]) and 7,080 cubic meters (250,000 cubic feet) of RH-TRU waste (the maximum allowed under the Agreement for Consultation and Cooperation [C & C] with the State of New Mexico). Using *The National Transuranic Waste Management Plan* in the same manner, no change would occur in the impacts for the Proposed Action except for the elimination of impacts related to storage of excess RH-TRU waste.

Under Action Alternatives 1, 2, and 3 and for the individual sites, the following changes would be expected:

- Negligible changes to the impacts would be anticipated in areas of land use and management, biological resources, cultural resources, noise, water resources and infrastructure, long-term performance, or consequences of lag storage accidents. No change would be expected in consequences from treatment accidents or WIPP disposal accidents.

- Changes in the estimated impacts for human health, life-cycle costs (except transportation costs), air quality, industrial safety, and economics would be directly related to changes in CH-TRU and RH-TRU waste volumes. Site impacts would change, as presented in the "percent difference" columns of Table J-1 for CH-TRU waste and Table J-2 for RH-TRU waste. Unlike most other impact areas, CH-TRU waste is a much higher contributor to impacts for involved workers than is RH-TRU waste; therefore, large decreases in RH-TRU waste volumes would have little impact for involved workers. However, at WIPP, large decreases in the amount of RH-TRU waste volumes would reduce the operations time needed for excavation and emplacement, reducing industrial safety and economics impacts.

- Changes in transportation impacts, including costs, are directly related to the number of shipments, which is dependent upon the type of waste treatment. Tables J-4 and J-5 present detailed CH-TRU and RH-TRU waste shipment information for all sites under all alternatives using *The National Transuranic Waste Management Plan* volumes. Table J-6 summarizes the differences in shipments and the percentages of the total inventories between *The National Transuranic Waste Management Plan* and the Basic Inventory.

Overall, impacts would be slightly lower using data from *The National Transuranic Waste Management Plan* because, although CH-TRU waste volumes are slightly higher, the RH-TRU waste volumes are markedly lower (except for the impacts to the involved workers, as noted above). The difference is quite marked for transportation impacts because of the reduction of nearly 61,000 shipments of RH-TRU waste under Action Alternative 3 (see Table J-6).

Changes in volume-dependent impacts would be expected at some sites. The changes in impacts would be greatest at INEEL, a major generator, treatment and potential consolidation site under all alternatives, where volume-dependent impacts could increase by about 124 percent. Impacts at RFETS would increase by about 50 percent and at Savannah River Site (SRS) by less than 10 percent. Impacts at Hanford would decrease dramatically: approximately 50 percent from CH-TRU waste and 90 percent from RH-TRU waste. Oak Ridge National Laboratory (ORNL) impacts, mainly from RH-TRU waste, would be about 60 percent lower. Impacts at LANL, the other key generator site, would also decrease by about 15 to 20 percent.

Table J-4
Number of CH-TRU Waste Shipments from Each Potential Consolidation Site to WIPP by Alternative Using *The National Transuranic Waste Management Plan* Volumes

	Proposed Action	Alternative 1			Alternative 2A			Alternative 2B			Alternative 2C			Alternative 3		
Sites	Basic	Basic	Additional	Total	Basic	Additional	Total	Basic	Additional	Total	Basic	Additional	Total	Basic	Additional	Total
ANL-E	12	11	0	11	0	0	0	0	0	0	11	0	11	0	0	0
Hanford	5,817	5,162	7,197	12,359	3,761	8,850	12,611	3,762	8,850	12,612	5,160	12,691	17,851	11,036	8,636	19,672
INEEL	12,466	11,046	6,758	17,804	9,230	12,469	21,699	13,834	14,408	28,242	11,061	9,738	20,799	22,947	8,049	30,996
LLNL	150	133	0	133	0	0	0	0	0	0	133	0	133	0	0	0
LANL	3,867	3,431	1,606	5,037	2,390	1,940	4,330	0	0	0	3,430	2,786	6,216	6,177	1,901	8,078
Mound	47	42	76	118	0	0	0	0	0	0	42	4	46	0	0	0
NTS	83	73	0	73	0	0	0	0	0	0	73	0	73	0	0	0
ORNL	221	196	13	209	0	0	0	0	0	0	179	13	192	0	0	0
RFETS	3,037	2,695	0	2,695	2,214	0	2,214	0	0	0	3,054	0	3,054	3,419	0	3,419
SRS	2,288	2,030	558	2,588	2,101	794	2,895	2,102	794	2,896	2,030	765	2,795	2,981	776	3,757
Ames	0	0	0	0	0	0	0	0	0	0	1	0	1	0	0	0
ARCO	0	0	0	0	0	0	0	0	0	0	1	1	2	0	0	0
B&W [a]	0	0	0	0	0	0	0	0	0	0	3	0	3	0	0	0
Battelle	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Bettis	0	0	0	0	0	0	0	0	0	0	15	0	15	0	0	0
ETEC	0	0	0	0	0	0	0	0	0	0	1	0	1	0	0	0
GE-VNC [a]	0	0	0	0	0	0	0	0	0	0	2	0	2	0	0	0
Knolls	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
LBL	0	0	0	0	0	0	0	0	0	0	1	1	2	0	0	0
PGDP	0	0	0	0	0	0	0	0	0	0	1	0	1	0	0	0
Pantex	0	0	0	0	0	0	0	0	0	0	1	0	1	0	0	0
SNL-AL	0	0	0	0	0	0	0	0	0	0	2	1	3	0	0	0
TBE	0	0	0	0	0	0	0	0	0	0	1	0	1	0	0	0
USAMC	0	0	0	0	0	0	0	0	0	0	1	0	1	0	0	0
U of M	0	0	0	0	0	0	0	0	0	0	1	0	1	0	0	0
WVDP	0	0	0	0	0	0	0	0	0	0	0	76	76	0	0	0
Totals	27,988 (29,766)[b]	24,819	16,208	41,027	19,696	24,053	43,749	19,698	24,052	43,750	25,204	26,076	51,280	46,560	19,362	65,922

[a] GE-VNC and B&W are included only in *The National TRU Waste Management Plan.*
[b] Total shipments when adjusted to the 168,500 cubic meters of CH-TRU waste allowed to be disposed of at WIPP under the LWA.

Table J-5
Number of RH-TRU Waste Shipments from Each Potential Consolidation Site to WIPP by Alternative Using *The National Transuranic Waste Management Plan* Volumes

Alternatives	Number of Shipments to WIPP for the Total Inventory				
	Hanford	**INEEL**	**LANL**	**ORNL**	**Totals**
Proposed Action	4,232	223	374	2,797	7,626 (7,957)[a]
Action Alternative 1					
Basic	4,210	223	395	2,797	7,625
Additional	1,229	121	269	2,265	3,884
Total	5,439	344	664	5,062	11,509
Action Alternative 2A					
Basic	1,690	0	0	979	2,669
Additional	566	0	0	793	1,359
Total	2,256	0	0	1,772	4,028
Action Alternative 2B					
Basic	1,690	0	0	979	2,669
Additional	566	0	0	793	1,359
Total	2,256	0	0	1,772	4,028
Action Alternative 2C					
Basic	4,828	0	0	2,797	7,625
Additional	1,617	0	0	2,265	3,882
Total	6,455	0	0	5,062	11,507
Action Alternative 3					
Basic	5,793	0	0	3,356	9,149
Additional	1,941	0	0	2,718	4,659
Total	7,737	0	0	6,074	13,808

[a] Total shipments when adjusted to the 7,080 cubic meters of RH-TRU waste allowed to be disposed of at WIPP under the C&C Agreement.

Table J-6
Comparison of Shipments Between Alternatives Using *The National Transuranic Waste Management Plan* and Basic Inventory

Alternatives	Number of Shipments to WIPP for the Total Inventory			
	NTRUWM Plan	**Basic Inventory**	**Difference (Cubic Meters)**	**NTRUWM Plan Percent Difference**
Proposed Action				
CH-TRU Waste	29,766[a]	29,766[a]	0	0%
RH-TRU Waste	7,957[b]	7,957[b]	0	0%
Action Alternative 1				
CH-TRU Waste	41,027	41,003	24	0%
RH-TRU Waste	11,509	62,162	-50,653	-81%
Action Alternative 2A				
CH-TRU Waste	43,749	42,775	974	+2%
RH-TRU Waste	4,028	21,895	-17,867	-82%
Action Alternative 2B				
CH-TRU Waste	43,750	42,774	976	+2%
RH-TRU Waste	4,028	21,895	-17,867	-82%
Action Alternative 2C				
CH-TRU Waste	43,431	41,206	2,225	+5%
RH-TRU Waste	11,507	62,160	-50,653	-81%
Action Alternative 3				
CH-TRU Waste	65,922	67,309	-1,387	-2%
RH-TRU Waste	13,808	74,606	-60,798	-81%

[a] Adjusted to the 168,500 cubic meters of CH-TRU waste allowed to be disposed of at WIPP under the LWA.
[b] Adjusted to the 7,080 cubic meters of RH-TRU waste allowed to be disposed of at WIPP under the C&C Agreement.

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APPENDIX K
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Affiliation: Battelle – Pacific Northwest National Laboratory

Education:
- B.S., Physical Metallurgy, Washington State University, 1981

Technical Experience: Fifteen years of experience as a risk and safety analyst and project manager in areas of radioactive and hazardous chemical material transportation risk analyses, transuranic solid waste management risks, and risk-based decision-making.

EIS Responsibility: Technical reviewer and contributor to the transportation impact analysis.

Name: JUDY DANKO

Affiliation: Battelle – Pacific Northwest National Laboratory

Education:
- B.A., Liberal Arts: History/English/Biology, 1981

Technical Experience: Sixteen years of experience as a technical editor, concentrating on technology management, energy, economics, and waste management.

EIS Responsibility: Assisted in the technical editing of the document.

Name: SALVATORE DIMARIA

Affiliation: Battelle - Albuquerque

Education:
- M.A., Geography, University of New Mexico, 1988
- B.S., Biology, University of New Mexico, 1972

Technical Experience: Eight years of experience with geographical, biological, and quantitative research methods.

EIS Responsibility: Technical lead for Chapter 4. Prepared the biotic resources, land use, and generator site sections of Chapter 4. Prepared the list of agencies and persons consulted. Contributed to the Comment Response Document.

Name: JAMES F. DONAGHUE

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- J.D., Law, Golden Gate University, 1990
- B.S., Civil Engineering, University of Arkansas, 1985

Technical Experience: Seven years of experience in environmental management and regulatory compliance.

EIS Responsibility: Contributed to Appendix A.

Name: FRANCIS C. DOUGLAS

Affiliation: Battelle - Albuquerque

Education:
- M.A., English, Kansas State University, 1994
- B.S., Journalism, Ohio University, 1982

Technical Experience: Eighteen years of experience as a writer and editor, including three years serving the environmental industry.

EIS Responsibility: Deputy Project Manager, Battelle Albuquerque activities. Contributor to Chapters 1, 2, 3 and 5 and comment response document. Conducted editorial reviews and provided technical document support. Lead for document production.

Name: AMY S. DUDA

Affiliation: Battelle - Albuquerque

Education:
- B.A., English, University of New Mexico, 1996

Technical Experience: Five years of experience in editing.

EIS Responsibility: Assisted in the technical editing of the document.

Name: LANCE W. ERRICKSON

Affiliation: Battelle – Albuquerque

Education:
- B.A., Geology, University of New Mexico, 1990

Technical Experience: Two years of experience working on WIPP technical data at Sandia National Laboratories.

EIS Responsibility: Assisted with document production.

Name: PAUL W. ESLINGER

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- Ph.D., Statistics, Southern Methodist University, 1983
- M.S., Mathematics, Washington State University, 1978
- B.S., Mathematics, George Fox College, 1976

Technical Experience: Seventeen years of experience in risk assessment, dose reconstruction, and performance assessment modeling.

EIS Responsibility: Lead preparer, Appendix A and the performance assessment sections of Chapter 5 and Appendix H. Contributed to the Comment Response Document.

Name: CHRISTIAN J. FOSMIRE

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Meteorology, Pennsylvania State University, 1993
- B.S., Meteorology, Pennsylvania State University, 1990

Technical Experience: Two years of experience in atmospheric diffusion modeling and risk assessment.

EIS Responsibility: Prepared the air quality sections of Chapter 5 and Appendix C. Assisted with the Comment Response Document.

Name: GARIANN GELSTON

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- B.S., Applied Mathematics, Mesa State College, 1991

Technical Experience: One year of experience in environmental assessment modeling.

EIS Responsibility: Modeled nonradiological impacts for the human health and facility accidents sections of Chapter 5 and Appendices F and G, and contributed to assessment of post-institutional control period for the No Action Alternative 2 in Chapter 5 and Appendix I.

Name: DONNA D. HAMMOND

Affiliation: Battelle – Albuquerque

Education:
- B.S., Education, New Mexico State University, 1995

Experience: Five years of experience in document production and database management.

EIS Responsibility: Contributed to text processing and the production of the document.

Name: GLEN T. HANSON

Affiliation: Battelle - Albuquerque

Education:
- M.A., Anthropology/Archaeology, Arizona State University, 1976
- B.S., Anthropology/Archaeology, Grand Valley State College, 1971

Technical Experience: Twenty-four years of experience in environmental and resource management, regulatory compliance, environmental assessment and impact analyses for NEPA documentation, facility siting, site characterization, cultural resource assessment and management, and environmental program management.

EIS Responsibility: Draft SEIS-II Project Manager, Battelle Albuquerque.

Name: PAUL L. HENDRICKSON

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- J.D., Law, University of Washington, 1971
- M.S., Industrial Management, Purdue University, 1972
- B.S., Chemical Engineering, University of Washington, 1968

Technical Experience: Twenty-three years of experience in energy and environmental studies with special emphasis on regulatory issues.

EIS Responsibility: Prepared the sections on land use impacts in Chapter 5.

Name: JAMES A. HILEMAN

Affiliation: Battelle - Albuquerque

Education:
- Ph.D., Seismology, California Institute of Technology, 1977
- M.S., Seismology, California Institute of Technology, 1971
- Geophysical Engineer, Colorado School of Mines, 1960

Technical Experience: Thirty years of experience in exploration, research, management, and review, particularly in siting critical facilities and assessing geologic hazards.

EIS Responsibility: Contributed to sections of Chapter 3.

Name: TRACY A. IKENBERRY

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Radiology & Radiation Biology, Colorado State University, 1982
- B.A., Biology, McPherson College, 1979

Technical Experience: Fifteen years of experience in radiological assessment, operational and environmental health physics. Diplomate, American Board of Health Physics, 1988.

EIS Responsibility: Project Manager, PNNL activities. Lead preparer, Appendices B and J. Technical and management reviewer. Contributed to Chapter 3 and the Comment Response Document.

Name: TERRY D. JAMES

Affiliation: Battelle – Columbus

Education:
- B.A., International Studies and English, Bradley University, 1969
- A.S., Life Sciences, Parkland College, 1986

Technical Experience: Twenty-seven years of experience in publications, including writing, editing, and proposal management.

EIS Responsibility: Contributed to the production of the draft final document.

Name: HAROLD JOHNSON

Affiliation: U.S. Department of Energy

Education:
- J.D., Law, Mercer University, 1976
- B.S., Math and Physics, Mercer University, 1971

Technical Experience: Nine years of experience in writing, reviewing, and approving NEPA documents.

EIS Responsibility: DOE Document Manager.

Name: CHRISTINE D. LADD

Affiliation: Battelle - Albuquerque

Education
- Certified in the beginning, intermediate, and advanced levels of Adobe Illustrator, University of New Mexico, 1995

Technical Experience: Two years of experience in document production and graphics design.

EIS Responsibility: Lead text processor and production coordinator. Contributed to the design of the graphics and production of the document.

Name: DAVID LECHEL

Affiliation: Lechel, Inc.

Education:
- M.S., Fisheries Biology, Michigan State University, 1974
- B.S., Fisheries Biology, Michigan State University, 1972

Technical Experience: Twenty-two years of experience in project management, regulatory compliance and permitting, NEPA planning and implementation for hazardous and radioactive waste disposal facilities.

EIS Responsibility: Technical reviewer. Lead preparer of Chapter 2, assisted in the development of alternatives, and contributed to the Comment Response Document.

Name: LEIGH LECHEL

Affiliation: Lechel, Inc.

Education:
- B.S., Speech Pathology, James Madison University, 1976

Technical Experience: Twenty years of experience in document production and administrative support.

EIS Responsibility: Contributed to the production of the document.

Name: ROBERT A. LECHEL

Affiliation: Battelle - Albuquerque

Education:
- B.S., Environmental Studies and Planning: Hazardous Materials Management, California State University - Sonoma, 1995

Technical Experience: One year of experience in performance assessment modeling.

EIS Responsibility: Contributed to the transportation sections of Chapter 5 and Appendix E. Assisted with the Comment Response Document.

Name: MARCUS K. LESTER

Affiliation: Battelle - Seattle Research Center

Education:
- M.A., Geography, University of Washington, 1991
- B.A., Geography, University of Washington, 1989

Technical Experience: Six years of experience in cartography, geographic information systems (GIS), and spatial analysis of environmental and socioeconomic data.

EIS Responsibility: Prepared the environmental justice section of Chapter 4.

Name: GEORGE A. MARINO

Affiliation: Battelle - Albuquerque

Education:
- M.E., Civil/Environmental Engineering, University of South Florida, 1994
- B.S., Mechanical Engineering, University of Pittsburgh, 1983

Technical Experience: Six years of experience in quality assurance, production engineering, and industrial hygiene and safety, and seven years of experience in environmental remediation.

EIS Responsibility: Technical Reviewer and assisted with the Comment Response Document.

Name: MACHELE McKINLEY

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- B.S., Environmental Engineering, Humboldt State University, 1992

Technical Experience: Three years of experience in computer modeling of saturated and unsaturated hydrology.

EIS Responsibility: Contributed to assessment of the post-institutional control period for No Action Alternative 2 in Chapter 5 and Appendix I.

Name: THOMAS McSWEENEY

Affiliation: Battelle - Columbus

Education:
- Ph.D., Chemical Engineering, University of Michigan, 1967
- M.A., Math, University of Michigan, 1964
- M.S., Chemical Engineering, University of Michigan, 1961
- B.S., Chemical Engineering, University of Notre Dame, 1960

Technical Experience: Twenty-eight years of experience in risk and safety analysis and ten years of experience in transportation risk analysis.

EIS Responsibility: Contributed to the transportation sections of Chapter 5 and Appendix E.

Name: TERRI B. MILEY

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Mathematics, University of South Carolina, 1986
- B.S., Mathematics, University of South Carolina, 1982

Technical Experience: Eight years of experience in performance assessment and groundwater risk assessments.

EIS Responsibility: Contributed to the performance assessment sections of Chapter 5 and Appendix H.

Name: TOM MONROE

Affiliation: Battelle – Columbus

Education:
- B.S., Journalism, Ohio University, 1995

Technical Experience: Two years of experience in technical writing/editing, proposal development, desktop publishing, and database development.

EIS Responsibility: Assisted in editing of the document.

Name: MARK T. MURPHY

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- Ph.D., Geology, Johns Hopkins University, 1989
- M.S., Geology, University of New Mexico, 1985
- B.S., Earth Science, University of California at Santa Cruz, 1977

Technical Experience: Eighteen years of experience in environmental geology and geological engineering.

EIS Responsibility: Contributed to the geology section of Chapter 4.

Name: ELIZABETH A. NAÑEZ

Affiliation: Battelle - Albuquerque

Education:
- B.S., Industrial Engineering, Texas Tech University, 1990

Technical Experience: Three years of experience in environmental engineering including one year of experience in NEPA document preparation.

EIS Responsibility: Comment Response Document Manager. Contributed to sections of Chapters 1, 3, 4, and 5, and compiled the list of preparers and contributors.

Name: IRAL C. NELSON

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.A., Physics, University of Oregon, 1955
- B.S., Mathematics, University of Oregon, 1951

Technical Experience: Forty years of experience in various aspects of health physics (radiation protection) and twenty-four years of experience in conducting NEPA reviews and preparing NEPA documentation. Diplomate, American Board of Health Physics, 1962.

EIS Responsibility: Co-preparer of Appendix B.

Name: WILLIAM E. NICHOLS

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Civil Engineering, Oregon State University, 1990
- B.S., Agricultural Engineering, Oregon State University, 1987

Technical Experience: Five years of experience in hydrologic and hydrothermal vadose zone modeling for performance assessment of waste isolation and disposal issues.

EIS Responsibility: Performed ground water transport modeling for the performance assessment sections of Chapter 5 and Appendix H.

Name: PAUL R. NICKENS

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- Ph.D., Anthropology, University of Colorado, 1977
- M.A., Anthropology, University of Colorado, 1974
- B.A., Anthropology, University of Colorado, 1969

Technical Experience: Twenty-one years of experience in southwestern archaeology and cultural resource management, site protection and preservation.

EIS Responsibility: Prepared the cultural resources sections of Chapters 4 and 5.

Name: TINA PAYNE

Affiliation: Battelle – Columbus

Education:
- B.A., Journalism, Otterbein College, 1994

Technical Experience: Three years of experience in technical editing and writing. Over fifteen years of experience in document production.

EIS Responsibility: Assisted in the technical editing of the document.

Name: TED M. POSTON

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Fisheries, Central Washington University, 1978
- B.S., Fisheries, Central Washington University, 1973

Technical Experience: Twenty years of experience in research, environmental assessment, and noise analysis.

EIS Responsibility: Performed the noise impact analyses for Chapter 5.

Name: MICHELLE R. POTTER

Affiliation: Battelle - Albuquerque

Education:
- M.S., Radiological Sciences and Control, University of Lowell, 1991
- B.H.S., Medical Technology, University of Missouri - Columbia, 1984

Technical Experience: Five years of health physics and technical writing experience.

EIS Responsibility: Quality Assurance Manager for Draft SEIS-II. Assisted in the technical reviews and editing of the draft document.

Name: RANDY F. REDDICK

Affiliation: Battelle - Albuquerque

Education:
- M.S., Environmental Health Engineering, University of Kansas, 1983
- B.S., Civil Engineering, University of Kansas, 1982

Technical Experience: Twelve years of experience with NEPA compliance, NEPA document preparation, and safety studies.

EIS Responsibility: Final SEIS-II Co-Project Manager, Battelle Albuquerque, technical support. Major contributor to Chapter 3. Conducted technical and management reviews of the document and contributed to the Comment Response Document.

Name: ROBERT A. ROOT

Affiliation: Battelle - Albuquerque

Education:
- Ph.D., Botany, Miami University, 1971
- M.S., Botany, University of Montana, 1968
- B.S., Botany, University of Maine, 1963

Technical Experience: Over twenty years of experience as an environmental scientist and project manager.

EIS Responsibility: Lead preparer of Chapter 3. Assisted with the Comment Response Document.

Name: STEVEN B. ROSS

Affiliation: Battelle - Albuquerque

Education:
- M.S., Nuclear Engineering, University of New Mexico, 1987
- B.S., Nuclear Engineering, University of New Mexico, 1985

Technical Experience: Ten years of experience in safety analysis, risk assessment, transportation, regulatory analysis, and fire risk assessment.

EIS Responsibility: Lead preparer of the transportation sections of Chapters 4, 5, and Appendix E. Contributed to the Comment Response Document.

Name: KEITH D. SHIELDS

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Environmental Science, Washington State University - Tri-Cities, 1995
- B.S., Physics, Whitman College, 1989

Technical Experience: Six years of experience in environmental assessments and modeling.

EIS Responsibility: Contributed to the assessment of post-institutional control period for No Action Alternative 2 in Chapter 5 and Appendix I.

Name: SANDRA F. SNYDER

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S.P.H., Radiological Hygiene, University of North Carolina-Chapel Hill, 1991
- B.S., Environmental Resource Management, Pennsylvania State University, 1986

Technical Experience: Seven years of experience in environmental health physics risk assessment.

EIS Responsibility: Lead preparer, human health and facility accidents sections of Chapter 5 and Appendices F and G. Contributed to the Comment Response Document.

Name: LISSA H. STAVEN

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Health Physics, Colorado State University, 1990
- B.S., Environmental Conservation, University of New Hampshire, 1984

Technical Experience: Six years of experience in environmental health physics.

EIS Responsibility: Contributed to the human health and facility accidents sections of Chapter 5, Appendices F and G, and assessment of post-institutional control period for No Action Alternative 2 in Chapter 5 and Appendix I.

Name: GARY STREILE

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- Ph.D., Soil Science (Soil Physics), University of California Riverside, 1984
- B.S., Physics, University of California Riverside, 1975

Technical Experience: Twenty years of experience in contaminant transport modeling and research into reactive contaminant transport processes.

EIS Responsibility: Contributed to the assessment of the post-institutional control period for No Action Alternative 2 in Chapter 5 and Appendix I.

Name: DENNIS L. STRENGE

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Chemical Engineering, University of Minnesota, 1968
- B.S., Chemical Engineering, University of Washington, 1966

Technical Experience: Twenty-seven years of experience in the fields of environmental health physics and risk assessment.

EIS Responsibility: Contributed to the assessment of post-institutional control period for No Action Alternative 2 in Chapter 5 and Appendix I.

Name: LUCINDA L. SWARTZ

Affiliation: Battelle – Albuquerque

Education:
- J.D. (Law), Washington College of Law, The American University, 1979
- B.A., Political Science and Administrative Studies, University of California, Riverside, 1976

Technical Experience: Seventeen years of experience in environmental law and regulation, most recently specializing in National Environmental Policy Act compliance.

EIS Responsibility: Final SEIS-II Co-Project Manager, Battelle Albuquerque. Technical reviewer of Comment Response Document and Final SEIS-II for conformity to NEPA and CEQ and DOE regulations and guidance.

Name: DESIREE THALLEY

Affiliation: Battelle – Albuquerque

Education:
- B.A., Journalism, University of New Mexico, 1983

Technical Experience: Thirteen years of experience in technical editing of Department of Energy and Department of Defense documentation.

EIS Responsibility: Assisted in the technical editing of the document.

Name: CARLOS A. ULIBARRÍ

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- Ph.D., Economics, University of New Mexico, 1992
- B.A., Economics and Spanish Literature, University of New Mexico, 1984

Technical Experience: Six years of experience in natural resource and environmental economics.

EIS Responsibility: Lead preparer of the socioeconomic sections of Chapters 4 and 5 and Appendix D.

Name: LANCE W. VAIL

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- M.S., Civil Engineering, Montana State University, 1981
- B.S., Environmental Resources Engineering, Humboldt State University, 1980

Technical Experience: Fifteen years of experience in surface and subsurface hydrology.

EIS Responsibility: Contributed to the hydrology section of Chapter 4.

Name: SANDRA K. WALTERS

Affiliation: Battelle - Albuquerque

Education:
- B.S., Environmental Health, Colorado State University, 1995

Technical Experience: Eight years of experience in data management, quality assurance and one year of experience in NEPA compliance.

EIS Responsibility: Prepared portions of Chapter 3, contributed to the preparation of the transportation sections of Chapter 5, and assisted with the Comment Response Document.

Name: TOMMYE S. WRIGHT

Affiliation: Battelle - Pacific Northwest National Laboratory

Education:
- B.S., Health Physics, Oklahoma State University, 1978

Technical Experience: Seventeen years of experience in occupational safety and health physics. Certified Safety Professional (CSP) and member of the National Registry of Radiation Protection Technologists (NRRPT).

EIS Responsibility: Prepared the industrial health and safety sections of Chapter 5.

APPENDIX L
DISTRIBUTION LIST

L.1 FULL SEIS-II

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U.S. Department of Energy
Albuquerque Operations Office Library
Albuquerque, NM

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National Research Council
National Academy of Sciences
Washington, DC

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Oak Ridge, TN

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Westminster, CO

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Laramie, WY

DOE Oak Ridge Public Reading Room
U.S. Department of Energy
Oakridge, TN

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Olympia, WA

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U.S. Department of Energy
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U.S. Department of Energy
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Planning
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Washington, DC

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Senator of California
United States Senate
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Donna Campbell
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Honorable William Clyburn
Representative
State of South Carolina
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Program Director
Concerned Citizens for Nuclear Safety
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United States Senate
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