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APPENDICES A-L

FINAL ENVIRONMENTAL IMPACT STATEMENT

DISPOSAL OF HANFORD DEFENSE HIGH-LEVEL, TRANSURANIC AND TANK WASTES

Hanford Site
Richland, Washington

DECEMBER 1987

U.S. DEPARTMENT OF ENERGY

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COVER SHEET

RESPONSIBLE AGENCY: U.S. Department of Energy

TITLE: Final Environmental Impact Statement, Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes, Hanford Site, Richland, Washington

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ABSTRACT: The purpose of this Environmental Impact Statement (EIS) is to provide environmental input into the selection and implementation of final disposal actions for high-level, transuranic and tank wastes located at the Hanford Site, Richland, Washington, and into the construction, operation and decommissioning of waste treatment facilities that may be required in implementing waste disposal alternatives. Specifically evaluated are a Hanford Waste Vitrification Plant, Transportable Grout Facility, and a Waste Receiving and Packaging Facility. Also an evaluation is presented to assist in determining whether any additional action should be taken in terms of long-term environmental protection for waste that was disposed of at Hanford prior to 1970 as low-level waste (before the transuranic waste category was established by the Atomic Energy Commission but which might fall into that category if generated today).

The following alternatives are considered in this EIS: 1) in-place stabilization and disposal, where waste is left in place but is isolated by protective and natural barriers; 2) geologic disposal, where most of the waste (by activity and to the extent practicable) is exhumed, treated, segregated, packaged and disposed of in a deep geologic repository; waste classified as high-level would be disposed of in a commercial repository developed pursuant to the Nuclear Waste Policy Act; transuranic waste would be disposed of in the Waste Isolation Pilot Plant near Carlsbad, New Mexico; 3) a reference alternative, where some classes of waste are disposed of in geologic repositories and other classes of waste are disposed of by in-place stabilization and disposal; 4) the preferred alternative, in which double-shell tank wastes, strontium and cesium capsules, and retrievably stored TRU wastes are disposed of according to the reference alternative, and in which decisions are deferred on disposal of single-shell tank wastes and on further remedial action for TRU-contaminated soil sites and pre-1970 buried suspect TRU-contaminated solid wastes (except the 618-11 site) until additional information is obtained on waste characterization, retrieval methods, and performance of near-surface disposal systems; and 5) a no disposal action alternative (continued storage).

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DOE/EIS-0113 (VOL. 2 of 5)
APPENDICES A-L

FINAL ENVIRONMENTAL IMPACT STATEMENT

DISPOSAL OF HANFORD DEFENSE HIGH-LEVEL, TRANSURANIC AND TANK WASTES

Hanford Site
Richland, Washington

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DECEMBER 1987

U.S. DEPARTMENT OF ENERGY
ASSISTANT SECRETARY FOR DEFENSE PROGRAMS
WASHINGTON, D.C. 20545

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FOREWORD

This environmental impact statement (EIS) provides analyses of environmental impacts for the selection and implementation of final disposal strategies for the high-level (HLW), transuranic (TRU) and tank wastes generated during national defense activities and stored at the Hanford Site near Richland, Washington. Also an evaluation is presented to assist in determining whether any additional action should be taken in terms of long-term environmental protection for waste that was disposed of at Hanford prior to 1970 as low-level waste (before the transuranic waste category was established by the Atomic Energy Commission (AEC) but which might fall into that category if generated today). This document also addresses environmental impacts associated with the construction, operation and decommissioning of waste treatment facilities that may be required to implement the waste disposal alternatives.

Several previous documents have addressed environmental aspects of the management of defense waste at the Hanford Site. The first comprehensive one, The Final Environmental Statement for Hanford Waste Management Operations (ERDA-1538), was issued in 1975. In that statement, waste management practices at Hanford were shown to protect the public health and safety and the environment on an interim basis. Those practices, however, were not and are not intended as final solutions for long-term isolation and disposal of high-level, TRU and tank wastes.

In 1977, the Energy Research and Development Administration (ERDA) issued the report Alternatives for Long-Term Management of Defense High-Level Radioactive Waste (ERDA-77-44), which included preliminary cost estimates and analyses of near-term risks associated with alternatives considered. That document examined 27 variations on four options for the processing and disposal of Hanford HLW, encompassing numerous final waste forms and storage and disposal modes.

In 1978, the National Research Council of the National Academies of Science and Engineering issued a report entitled Radioactive Wastes at the Hanford Reservation: A Technical Review, concluding that there has not been in the past, and is not at the present, any significant radiation hazard to public health and safety from waste management operations at Hanford. The Council recommended that long-term isolation and disposal of Hanford high-level waste become the main focus of waste management research and development.

The need to include retrievably stored TRU waste within the scope of wastes to be disposed of, and concerns about potential environmental impacts of wastes disposed of before 1970 as low-level wastes (before the Atomic Energy Commission established the TRU waste category but which might be classed as TRU if generated today), led to enlarging the earlier plan that was to issue an EIS covering high-level waste only. Accordingly, on April 1, 1983, the Department of Energy (DOE) published in the Federal Register (48 FR 14029) a Notice of Intent (NOI) to prepare an EIS on Disposal of Radioactive Defense High-Level and Transuranic Wastes at Hanford.

Eighteen comment letters were received in response to the Notice of Intent to prepare this EIS. Ten of the letters only requested copies of the draft EIS when issued; eight contained comments regarding its preparation. The draft EIS was published during March 1986, and its availability was published in the Federal Register on April 11 (51 FR 12547). During the 120-day agency and public comment period on the draft EIS, which began on April 11, 1986, 243 letters were received that provided about 2000 substantive comments on the draft EIS. In addition, oral testimony was heard on the draft EIS in public hearings held during July 1986, in Richland, Washington; Portland, Oregon; Seattle, Washington; and Spokane, Washington.

Excluded from consideration in this EIS are low-level radioactive wastes in liquid and solid disposal sites at Hanford (see ERDA 1538). These waste sites are presently being reviewed under hazardous-waste regulations. Also excluded are wastes generated by decontamination and decommissioning of surplus or retired facilities after the year 1983 (other than for those facilities directly associated with waste disposal). Those operations will be the subject of other National Environmental Policy Act (NEPA) reviews.

The Defense Waste Management Plan (DOE/DP 0015) states of the Hanford wastes: "Immobilization of new and readily retrievable high-level waste will begin about 1990 after sufficient experience is available from Savannah River's vitrification process. Other waste will be stabilized in place in the 1985-2015 time frame if, after the requisite environmental documentation, it is determined that the short-term risks and costs of retrieval and transportation outweigh the environmental benefits of disposal in a geologic mined repository."

It is necessary to understand the major differences between civilian and defense wastes and the programs to effect their disposal. Both types of waste include fission products and transuranic waste elements. On the other hand, the quantities of these elements, the physical and chemical forms of the wastes, and the technically sound alternatives for their disposal are markedly different. In all cases, for both civilian and defense, the final methods selected will have to meet the Environmental Protection Agency (EPA) standards (40 CFR 191) for the disposal of spent fuel and high-level and TRU wastes. The Nuclear Waste Policy Act of 1982 mandates a procedure to select the potential repository sites for detailed characterization.

A comparison of the Hanford waste inventory resulting from chemical processing of about 100,000 metric tons of nuclear reactor fuel with that of a commercial repository containing 70,000 metric tons of spent fuel elements is enlightening. In this comparison, the waste inventory from 100,000 metric tons of Hanford reactor fuel contains about 4% as much of the readily transportable (geohydrologically) isotopes $^{14}\mathrm{C}$, $^{99}\mathrm{Tc}$, and $^{129}\mathrm{I}$ as is contained in 70,000 metric tons of commercial spent fuel. It contains only 1% as much $^{90}\mathrm{Sr}$ and $^{137}\mathrm{Cs}$ and about 0.1% as much of the primary transuranics $^{239}\mathrm{Pu}$, $^{240}\mathrm{Pu}$, and $^{241}\mathrm{Am}$. The volume of the Hanford wastes is markedly larger than the civilian wastes cited above--410,000 m^3 of Hanford wastes as compared to 29,000 m^3 of commercial spent fuel.

The physical and chemical characteristics of existing and potential waste forms considered in this EIS are highly diverse: liquid waste in double-shell tanks, vitrified/canistered wastes (from processed double-shell tank wastes); sludge and salts in the single-shell tanks; strontium and cesium capsules that are further protected with a handling container; previously disposed of pre-1970 wastes in various forms and containers; and finally, low-level waste products, from the processing of double-shell-tank waste, in the form of grout.

In accordance with the requirements of NEPA, as amended, and implementing regulations of the Council on Environmental Quality (CEQ) published in the Code of Federal Regulations as 40 CFR 1500, this EIS was written early in the decision-making process to ensure that environmental values and alternatives are fully considered before any decisions are made that might lead to adverse environmental impacts or limit the choice of reasonable alternatives. This process will also help ensure that the public is fully informed and is involved in the decision-making process.

To comply with the NEPA's requirement for early preparation of environmental documentation, this EIS has been prepared early in the disposal decision process. As with any major action, it is expected that once a disposal decision is made, subsequent detailed engineering may enhance specific waste retrieval, treatment, handling, immobilization and/or disposal processes evaluated in the EIS. However, the processes evaluated in this document have been chosen such that, when finally implemented for any of the options, the processes would not be expected to result in environmental impacts that significantly exceed those described here. The DOE believes that bounding analyses performed in this EIS meet the requirements of CEQ regulations for analysis of all reasonably forseeable significant adverse impacts.

Implementation of defense waste disposal under the alternatives described in this EIS will be done in compliance with the letter and spirit of applicable federal and state environmental statutes, regulations and standards. To ensure that impacts of specific processes used during disposal implementation do not differ significantly from the results of the analyses set forth in this document, DOE will conduct environmental reviews of the specific processes as finally proposed. On the basis of these reviews, DOE will determine in accord with agency guidelines what additional NEPA documentation is required. The DOE anticipates that a supplemental EIS will be prepared prior to a decision on a disposal option for single-shell tank waste.

This document is not intended to provide the environmental input necessary for siting or constructing a geologic repository. For analysis of environmental impacts of alternatives involving geologic disposal, generic designs for either an offsite or onsite repository were used. Detailed environmental documentation required by the Nuclear Waste Policy Act of 1982 will be prepared before a geologic repository is sited, constructed and operated. A future EIS to address site selection is expected to include a discussion of cumulative impacts of the repository program at all candidate sites, including Hanford.

Other NEPA documentation relevant to this EIS includes the supplement to ERDA-1538, Double-Shell Tanks for Defense High-Level Radioactive Waste Storage at the Hanford Site (DOE/EIS-0063), and the Final Environmental Impact Statement--Operation of PUREX and Uranium Oxide Plant Facilities (DOE/EIS-0089). (The draft PUREX EIS with an addendum constituted the final PUREX EIS.)

Environmental considerations regarding disposal of Hanford's retrievably stored TRU waste at the Waste Isolation Pilot Plant (WIPP) (except for retrieval, processing, packaging, certification and transportation of waste from Hanford to WIPP, which are discussed in this EIS) are based on the Final Environmental Impact Statement--Waste Isolation Pilot Plant (DOE/EIS-0026). Environmental considerations associated with waste disposal in geologic repositories are based on information from the Final Environmental Impact Statement--Management of Commercially Generated Radioactive Waste (DOE/EIS-0046F). Alternatives to disposal of high-level waste in geologic repositories were described in that document.

Environmental considerations associated with borosilicate glass as a waste form for repository disposal of waste and with the construction and operation of a plant to provide vitrified waste are based in part on information developed in three previous DOE documents: Final Environmental Impact Statement--Defense Waste Processing Facility Savannah River Plant, Aiken, South Carolina (DOE/EIS-0082); Environmental Assessment--Waste Form Selection for SRP High-Level Waste (DOE/EA-0179); and Analyses of the Terminal Waste Form Selection for the West Valley Demonstration Project (WVDP-100 DOE).

The EIS has been structured to conform as closely as possible to the format described in CEQ Regulation 40 CFR Parts 1502.1 through 1502.18. To provide more information for the reader than can be reported within the text of Volume 1, more detailed information is included in 22 appendices (Volumes 2 and 3). Figure 1 in the Introduction to the Appendices (Volume 2, p. xxiv) shows the purpose of each appendix and how appendices relate to each other and to the text of Volume 1. Lines in the margins of Volumes 1, 2 and 3 indicate the areas where revisions were made. Volume 4 contains agency and public comments received and responses to them as well as the indication of location where revisions were made to the draft EIS. Volume 5 contains a reproduction of all of the comment letters received.

The final EIS is being transmitted to commenting agencies, made available to members of the public, and filed with the EPA. The EPA will publish a notice in the Federal Register indicating that the DOE has filed the final EIS. A DOE decision on proposed actions will not be made earlier than 30 days after the EPA has published the Federal Register notice for the final EIS. The DOE will record its decision in a publicly available Record of Decision (ROD) document published in the Federal Register.

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FIGURES

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TABLES

INTRODUCTION

Environmental impacts resulting from the implementation of disposal/no disposal alternatives are reported in Chapters 1, 3 and 5 of this EIS. To provide more information for the reader than can be reported within the text, the more-detailed information is included as appendices. There are 22 appendices that address, in general, such technical issues as 1) waste sources and disposal facilities, 2) assessment of operational impacts, and 3) post-disposal impact assessment.

More specifically, the appendices include detailed information on such topics as methods to calculate radiological doses and nonradiological consequences, assessment of transportation impacts, transport of radionuclides and chemicals through Hanford soils, long-term performance of disposal systems, and other technical issues. These and other data were needed to assess impacts resulting from disposal operations in the near term and waste disposal systems for the long term.

Figure 1 indicates the purpose of each appendix and how appendices relate to each other in the analysis. The rationale and bases for developing these analyses are described in the next section, entitled "Analytical Methodology."

EPA Regulations requiring predictions 10,000 years into the future are unique to the disposal of radioactive waste. A Record of Decision must be one that meets the requirements of the National Environmental Policy Act (NEPA). The analytical approach developed to meet both of these needs is also described in the Analytical Methodology section.

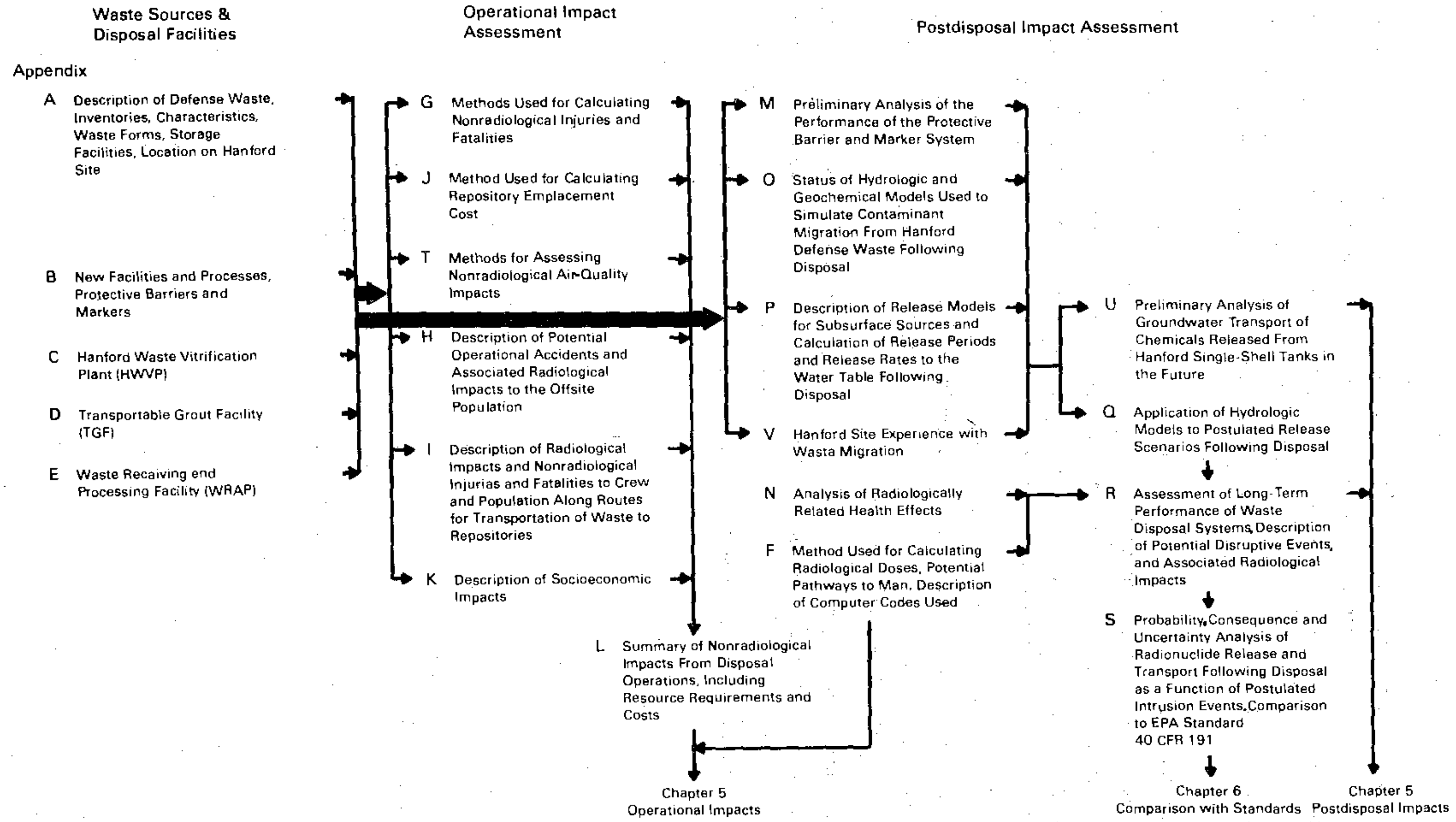

FIGURE 1. Interaction Among Appendices

ANALYTICAL METHODOLOGY

NEED FOR ANALYSIS

The purpose of the National Environmental Policy Act (NEPA) of 1969 (40 CFR 1500-1517) is to aid public officials in making decisions that are based on our understanding of the environment and in taking actions that protect and enhance the environment. NEPA's purpose is to foster excellent decisions, not to generate definitive scientific documents.

From the outset, a uniform body of law has developed that addresses the problem confronted by Federal agencies lacking definitive information to analyze the potential environmental effects of high-risk activities over long periods of time. The agencies are required to fully disclose the lack of important information and to provide a reasonable forecast of the likely consequences of proceeding without such information. Unequivocal proof of compliance is neither expected nor required because of the substantial uncertainties inherent in such long-term projections. This "rule of reason" encompasses analysis based on available data and, in some cases, engineering estimates. A "bounding" analysis is used throughout the EIS for estimating impacts. The DOE believes that bounding analyses performed in this EIS meet the requirements of analysis of all reasonably forseeable significant adverse impacts as noted in the U.S. Court of Appeals for the Eighth Circuit [Lamm vs Weinberger (86-1517, May 27, 1987)]. It is important that the nature of this analysis be understood and how it differs from other, perhaps more familiar, analyses.

The timing of NEPA statements necessitates an analysis prior to any major decision. After such a decision, compliance analysis may be necessary before obtaining a permit, license, start-up of a plant, etc. Such analysis may entail developing experimental or engineering data and validating codes that predict observed performance. In the case of Hanford defense waste (HDW), this work is under way or planned under the Hanford Waste Management Plan (DOE 1986).

Other types of analyses are sometimes used in environmental analysis. In the best-estimate analyses used in the repository Environmental Analyses, the major parameters are developed either as distributions in the Basalt Waste Isolation Project or as most likely values. Performance is assessed under unperturbed conditions using these values.

The approach used in this EIS bounds the impacts by assuming the worst set of circumstances that can be reasonably expected to occur at the same time. These sets are, in contrast to a probabilistic approach, assumed to occur with a probability of one. While there is no universally accepted definition of bounding analysis, the following is used in this EIS: A bounding analysis is deterministic and is performed with a set of data, modeling assumptions, and accidental release scenarios which, totaled, so <u>compounds conservatisms</u> that the calculated (predicted) environmental impacts should exceed those actually expected or experienced.

Individual parameters are not necessarily extreme values. They may be mean values in cases where ample data warrant such a choice or may be values well on the conservative[a] side of the expected mean for parameters with highly uncertain ranges of values. Because of uncertainty, the modeling assumptions are chosen to be conservative. Finally, accident scenarios are chosen to describe the most serious incidents that could reasonably occur.

KEY PARAMETERS

In discussing key parameters that most influence the impacts calculated in this analysis, it is important to note that the individual parameters themselves do not make the analysis bounding. Instead it is the systems analysis in which these values are used that bounds the reasonably expected impacts. "Bounding value" is thus not an appropriate term in the context of this EIS; to use it can only cause confusion.

Numerous modeling assumptions, scenario definitions, and data selections must be made in calculating all environmental impacts that could be generated by the Hanford Defense Waste Program. The resulting impacts, however, are highly sensitive to only a limited number of these parameters. The most important of these, listed in Table 1, are discussed. These values may come from a well-developed data set, from limited data where judgments are required in the selection of values, from pure engineering judgment where no data exist, or from assumptions that have no basis other than reason. In discussing these parameters an effort is made to point out the basis for each. Parameters are discussed in the order of their listing in Table 1.

Figure 2 depicts the major steps by which waste nuclides can migrate from the original waste form, through the environment, and eventually to humans. The figure illustrates how the results compound; long-term impacts are generally the product of factors shown. (Many other parameters of less sensitivity exist that are not shown.) Other impacts, such as operations and transportation, are additive in the total sense, but they have internal values that also compound conservatism. Thus, if reasonable values (whether based on data or engineering judgment) are chosen for these parameters, it can be seen that the radiological impacts are defensible as bounding on the basis of compounded conservatism.

Waste Inventories

Approach. Values of inventories used are those provided by Rockwell (1985), and unless specified otherwise, are best estimates. Average uncertainty on values provided was estimated by Rockwell to be within +50% -30%, but this value was not used in PNL's analysis. The distribution of radionuclides between single-shell and double-shell tanks may be more uncertain than that expressed by this range. To account for this uncertainty in existing tank waste, the tanked waste liquid inventory was assigned to both single-shell and double-shell tanks for calculating impacts.

(a) Conservative values as used in the HDW-EIS refer to those values or assumptions that tend to overestimate rather than underestimate impacts.

TABLE 1. Key Parameters in Calculated HDW Impacts

1. Waste Inventories
2. Waste Release Parameters
   a. Recharge Rate and Climatic Conditions
   b. Performance of Protective Barriers
   c. Waste Form Releases
   d. Tank Integrity
3. Geohydrologic Transport
   a. Transport in the Vadose Zone
   b. Transport in the Unconfined Aquifer
4. Pathways and Dosimetry
   a. Radiation Dosimetry
   b. Definition of Long-Term Exposure Scenarios
5. Short-Term Consequences
   a. Radiological Consequences of Operational Accidents
   b. Transportation Radiological Exposure Including Accidents
6. Radiologically Related Health Effects

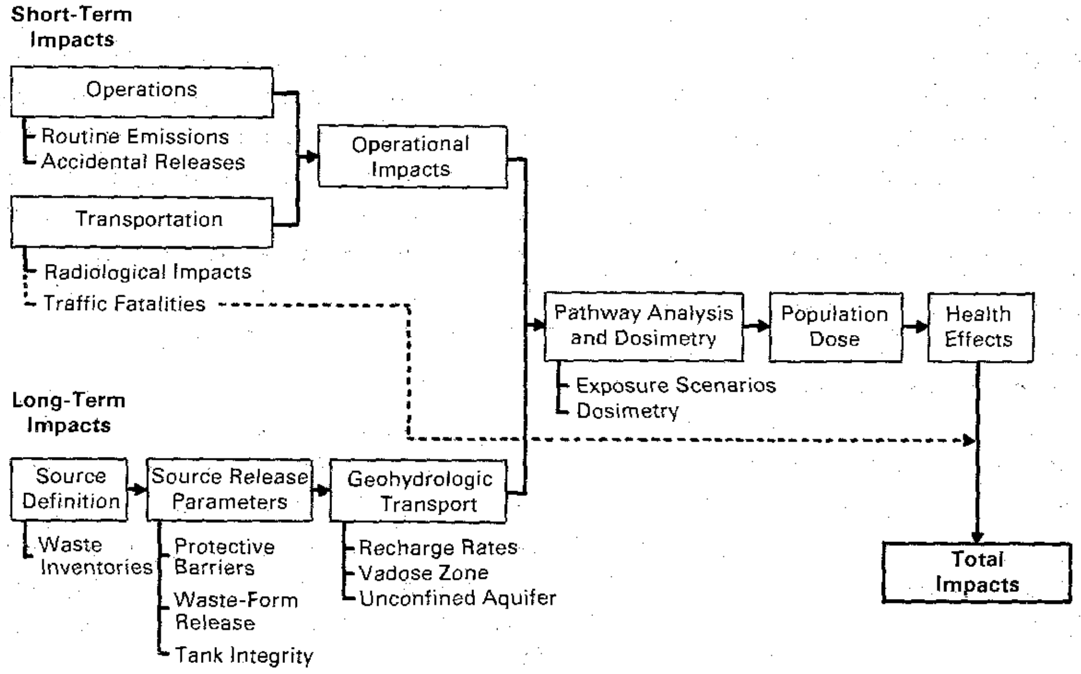

FIGURE 2. Key Impact Parameters

Features of This Approach. This so-called "double accounting" is a conservative assumption made to accommodate any uncertainties in the inventory distribution in tank wastes, specifically for the liquid distribution. For example, the total ^{90}Sr inventory in all tanks is estimated to be 60,000,000 Ci. Of this total, 40,000,000 Ci are assumed to be solid and are with some confidence located in the single-shell tanks. There is much greater uncertainty about the location of the balance, 20,000,000 Ci, assumed to be in solution. For impact analysis, this liquid inventory is assumed to be both in the single- and in the double-shell tanks. In effect, the impacts are calculated on the basis of an 80,000,000 Ci tank inventory, a result of 40,000,000 Ci (solid) plus 2 times 20,000,000 Ci (liquid) which equals 80,000,000 Ci.

For other waste classes the information provided is the best available and is expected to be reasonably conservative. Hence, actual impacts are not expected to exceed those presented.

Waste Release Parameters

a. Average Annual Recharge Rate and Climatic Conditions

Approach Used for Present Conditions. To be significant, the water pathway obviously requires some infiltration of moisture into the soils above the waste disposal site, desorption or dissolution of radionuclides and hazardous chemicals, and migration of the solute through the unsaturated zone to the water table. In transport calculations the operative factor is average annual recharge rate. Even though this is referred to simply as "recharge," the reader should remember that it is the long-term average that is pertinent to the issue. Recharge, in this context, is loosely defined as water that has drained below the zone where evapotranspiration can remove it from the system. Recharge depends on climatic variables, vegetation types and distributions, and the hydraulic conductivities of the various sediments through which the moisture may pass. Soil hydraulic properties, plant-water uptake, and climate variables were modeled for this EIS.

Under present climatic conditions, and with no barrier in place, the recharge rate on the 200 Areas plateau is assumed to be between zero and a few centimeters per year in undisturbed areas. At sites where surface soils are fine-textured and there is adequate plant cover, it is likely that an extremely low recharge rate exists under these conditions. Recharge is probably much higher where the soil has been removed and no plant cover exists. This conclusion is based primarily on several years of field lysimeter data, on the testing of both surface and deep-well observations of water contents in the 200 Area sediments as a function of time and depth, and on current understanding of the influence of soil texture on hydraulic properties. Unfortunately, there are no direct measurements of water balance and few, if any, in situ field measurements of hydraulic conductivities of 200 Area sediments that can be used to predict recharge. Thus, reliance has been almost entirely on assumed or estimated hydraulic conductivity values in calculations. This fact, combined with the limited resolution of the neutron probe measurements used to measure water content changes in the soil, dictates that quantitative predictions of water recharge rates are good only to

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within 2 or 3 cm/yr. Therefore, even if the present recharge on the 200 Area is zero, there are as yet no experimental data that confirm this.

Features of This Approach. For the EIS, a range of 0.5 to 5 cm/yr was selected on the basis of engineering judgment as a reasonable range of recharge rates to be expected on the 200 Areas plateau over the next 10,000 years. It seems likely that this range represents a current climate (dry) recharge rate of 0.5 cm/yr and a wetter climate at the 5-cm/yr value. Recharge can vary from zero to more than the annual precipitation rate (presently 16 cm/yr). Recharge depends on the surface conditions (topography, soil type, plant cover, etc.) as well as precipitation input; therefore it is site specific, particularly for an arid site like Hanford (see Appendix B of Fayer, Gee and Jones 1986). However, the single rate of 0.5 cm/yr is considered a reasonable representation of the site when considering unperturbed soil/plant conditions (e.g., climate, soil and plants) in the 200 Area, where the majority of the wastes addressed in this EIS are stored. This single value is in line with the range of recharge rates predicted by the U.S. Geological Survey (Bauer and Vaccaro 1986), who predicted that recharge on the 200 Areas plateau ranged from 0.1 to 1.2 cm/yr (based on climate records from 1957 to 1977). A great deal of uncertainty persists about recharge at Hanford, both present and future, and efforts are being made through the Hanford Site Performance Assessment Program to document the range of recharge that occurs on site under a variety of surface conditions. While significantly higher values could have been chosen and perhaps justified, the rate of 5 cm/yr was selected for the unperturbed soil/plant conditions. (While this number was developed and used for potential future climate conditions, its use in the calculational system permits examination of the impacts affected by this parameter.)

Approach Used for Potentially Wetter Conditions. Natural recharge rates, where no multimedia barrier is in place, may change in the thousands of years pertinent to this EIS performance assessment study. For this EIS, a future average annual recharge rate of 5 cm/yr was assumed because 1) experimental field evidence indicates that in wet years the present recharge rate in the 300 Area may be near 5 cm/yr, and 2) nationally recognized consultants have stated that for the Columbia Plateau, future average precipitation values are likely to be within the extremes of present values. Nationally recognized consultants support the assumption that rainfall for the present 100-year extreme is a reasonable value for the mean annual rainfall in a wetter climate scenario. Also selected and tested were both a fall (1948) and a spring (1947) bias in the temporal distribution of the annual rainfall used in the modeling. The rainfall in both years was normalized to an annual total of 30 cm. The 200 Areas have not yet been parametrically simulated using assumed future values for climatic variables and vegetation. This is primarily because the models have not yet been calibrated or demonstrated against experimental results. This type of activity is called for in the Hanford Waste Management Technology Plan (DOE 1986).

No future 200 Area onsite systematic irrigation effects such as sprinkling, leaky pipes, and canals are assumed. Although to assume no onsite artificial recharge is not conservative, it is consistent with the assumptions made regarding systematic intrusion within the boundary system with its warning markers.

Features of This Approach. The potential impacts of future offsite artificial recharge (e.g., irrigation, canal leakage, water losses beneath communities) are considered. Impacts primarily consist of changed elevations of the water table under the 200 Areas; these affect the direction and rate of water movement. These factors have been analyzed via predictive modeling of the shallow groundwater system in the Pasco Basin. The 5-cm/yr recharge is a reasonable estimate of potential recharge for conditions of grass cover on coarse soils. It would be consistent with values measured in the 300 Area in 1983, one of the wettest years on record. Based on consultants' advice, this is probably a reasonable value to evaluate potential impacts of a wetter climate.

b. Performance of the Conceptual Protective Barrier

Approach Used for Present Conditions. Computer modeling results to date reveal that a properly designed and built multi-layer barrier will store rainwater and snowmelt until it is removed naturally from the barrier system via evapotranspiration during the plants' growing season. The water is stored in the soil layer that overlies a gravel and basalt riprap zone. The fine soil can be thought of as acting like a sponge. There is little or no potential for the moisture to drain out of the overlying soil into the coarse gravel and riprap. As long as the soil does not saturate, the forces holding the water in the soil are greater than that of gravity. Thus the water remains suspended until plants can take it up and transpire the water away from the barrier system. The modeling parameters selected for the EIS were conservative; modeling efforts indicate that the 200 Area soils to be used are adequate. Field data are being collected as part of a barrier development effort (extending into the mid 1990s) to calibrate and demonstrate the model's adequacy at Hanford. This type of cover system has been used at solid waste (uranium tailings) sites in both humid and arid climates.

Approach Used for Potentially Wetter Conditions. The approach used here was discussed earlier under "Waste Release Parameters." Potentially wetter conditions over the next several thousand years might include a climate change, onsite artificial recharge, or a barrier disruption. Barrier disruptions that might be contemplated include erosion of the upper soil layer via wind and/or intense storm runoff, or subsidence of the barrier, or removal of soil by human activity. All of these potential disruptions are the subjects of engineering efforts. For instance, rock mulches are being designed for application to the surface of the barrier to preclude removal of the soil by wind should vegetative cover be removed. Subsidence control technologies are being tested in the field, and warning systems have been designed and tested to deter human intrusion and activity on the barrier.

The climate change scenario was analyzed by means of predictive modeling. The present 100-year extreme annual rainfall (30.1 cm/yr) was selected and applied to the barrier in successive years until an equilibrium water balance was attained. A water balance equilibrium is assumed when the barrier stops accumulating water (i.e., zero net storage). That is, evapotranspiration and drainage through the barrier equal the precipitation applied to the upper soil layer.

The preliminary results of modeling the future climate scenario reveal that, with fine-textured soil and shallow-rooted cheatgrass (a 152-day growing/transpiration cycle), a water balance would be reached before any water drained (recharge) below the top layer of the barrier. The 1.5-m, fine-textured soil would totally prevent natural water recharge for the mean precipitation (30 cm) assumed for the wetter climate scenario. A thicker soil layer was found to be undesirable unless deeper-rooted vegetation was used. As noted above, these results are based on preliminary modeling work. The computer codes are being calibrated and tested against actual barriers at Hanford that are subjected to present natural conditions and simulated future environmental conditions.

Future barrier failure scenarios were hypothesized: 1) a disruptive failure due to removal of soil by humans, and 2) a functional failure due to wind erosion, seismic events, subsidence, or use of construction materials that are out of specification. These potential disruptions were analyzed as follows. In the disruptive case it was assumed that the barrier failed in such a way that 10% of the waste inventory was subjected to direct leaching at 15-cm/yr recharge (the barrier failure is assumed to coincide with a wetter climate). In the functional barrier failure, it was assumed that 50% of the barrier was degraded to the point where it allowed 0.1 cm/yr to infiltrate the underlying wastes under wetter climate conditions (30-cm/yr annual average precipitation).

Features of This Approach. Experience (from non-Hanford projects) in barrier modeling provides confidence in the calculated performance results for the unperturbed barrier. For these calculations it was assumed that cheatgrass would grow on the barrier. It seems more likely that with increased average precipitation the barrier cover of fine silt/sand would support a natural grass with more extensive root and leaf systems as well as a longer growing season than exhibited by cheatgrass. Such vegetative cover would result in a significantly higher evapotranspiration potential than the one assumed.

The barrier performance was analyzed through an assumption that enough barrier topsoil was removed to result in direct leaching of 10% of the waste inventory. No available data support this assumption. Wind erosion may remove some of the topsoil from the barrier, particularly if no surface armoring is included in the design. While a flash flood in Cold Creek could inundate a corner of the plateau under extreme conditions, the topography is such that it would not erode the barriers, but could deposit sediments. Irrigation of large areas of the barrier surface is not deemed credible since it entails systematic intrusion of the messaged marker system. The most likely cause of barrier disruption is considered to be human intrusion. This means that the individual who disrupts the barrier must ignore records, the message-marked boundary, the waste marker obelisks, and the buried markers. This intrusion is considered unlikely, but not impossible.

c. Waste Form Releases

Approach. Releases take two forms: 1) wastes without a barrier that are leached by infiltrating water and 2) wastes with a barrier that are released via a diffusion-controlled transport pathway. Direct leaching is used also for inventories leached due to barrier failure. Release mechanisms for specific wastes are shown in Tables 2 and 3.

TABLE 2. Direct Leaching (without a barrier or for barrier failure)

Waste	Release Mechanism or Model
Grout	Release rate scaled from Savannah River Laboratory (SRL) data (DOE 1982)
Salt Cake and Sludge	Constant-solution concentration dissolution of nitrate, congruent release of radionuclides
Liquid	Distribution coefficient, K_d
TRU	Pu/Am: Constant-solution concentration dissolution of radionuclides Sr/Cs: Distribution coefficient, K_d
Sr/Cs Capsules	No release

TABLE 3. Wastes with an Operative Barrier

Waste	Release Mechanism or Model
Grout	Constant-solution concentration dissolution of nitrate and congruent release of radionuclides
Salt Cake and Sludge	Constant-solution concentration dissolution of nitrate and congruent release of radionuclides

Features of This Approach

- Grout release rate scaled from Savannah River Laboratory (SRL) data

  The release rate obtained from the SRL lysimeter study is scaled to Hanford grout geometry and applied to nitrate release. The rate, taken from some of the very early data, is projected to continue indefinitely in the model applied in this EIS. The actual release rate will, instead, decrease continuously with time. By applying a constant rate for a finite time and by using initial release data to define the rate, a conservative release is achieved. This implicitly assumes that the SRL grout and Hanford grout are similar entities. Because of their physical and chemical differences, it is necessary to confirm this assumption.

- Constant-solution concentration dissolution of nitrate and congruent release of radionuclides

Both the dissolution-controlled and the diffusion-controlled release models employ a concept of constant concentration of solution. For grout release from beneath a protective barrier and for all salt cake releases, the chemical component of interest is nitrate salt. A congruent release is assumed for all radionuclides. For TRU wastes, plutonium and americium releases are defined by solution concentrations of the radionuclide.

The calculation of salt-cake releases in single-shell tanks was based on solution concentrations reported by Schulz (1980) for interstitial liquors taken from single-shell tanks. These are the highest known published concentrations. Although it is unlikely that the chemical species in the salt cake and sludge will demonstrate solution concentrations as high as those measured in the tank liquors, these values are used for conservatism. The same solution concentrations are used for both direct-leaching and diffusion-controlled release scenarios.

The concentrations and oxidation state of plutonium and americium in TRU soil sites are defined as a result of research conducted on a decommissioned crib site. Thus, the release models for plutonium and americium in the TRU wastes are documented in the open literature. Although the applicable data are limited, the best available data have been used.

- Distribution coefficient

  Values of distribution coefficient used in the K_d release model applied to double-shell tank (liquid) and TRU wastes have been taken from Delegard and Barney (1983). As described in the discussion of transport in the vadose zone (where these same K_d values are used to model attenuation), the K_d represents the only data on synthetic or hypothetical Hanford single-shell tank wastes that are believed to be some of the more mobile wastes at Hanford. Therefore, desorption (i.e., release) and transport defined by these data represent the best that can be done with existing data. Use of single-value distribution coefficients (lumping complex chemistry into a single constant) has come under severe criticism recently. Despite the potential pitfalls of this approach, it has been chosen because of the limited data base. Double-shell tank liquids are characterized in the data as concentrated (salt content) and as complexed (organic complexant). It would strengthen the analysis considerably if data were available on the status of organic and inorganic complexation since this affects the effective K_d. Transuranic wastes are characterized as dilute and as non-complexed. The scarcity of data from which to develop averages and/or make comparisons requires that the data be considered realistic but not extreme values. However, because the data employed are taken from some of the more mobile wastes, there is confidence that releases are estimated conservatively.

- Availability of moisture

  Two issues exist regarding soil moisture and releases from single-shell tanks. First, is there a continuous water pathway over which diffusion can transport the release to the advection-controlled zone or the water table? Diffusion transport through vadose-zone sediments does occur even at the relatively low moisture contents that exist at Hanford. Therefore the transport pathway and mechanisms are plausible.

  Second, can soil moisture move to the waste from surrounding soils, dissolve the waste, and subsequently leach it to the soil? It is highly conservative to assume, as this EIS does, that an unlimited quantity of soil water is available for such a leach process. No studies have been conducted to determine viable mechanisms or approximate quantities of water involved. However, the high salt content of single-shell tank wastes will provide a hygroscopic force, i.e., a vapor potential gradient, that will draw soil moisture to the tank if 1) significant tank surface area has corroded or degraded (no credit is assumed for restriction of water movement afforded by the tanks past the year 2150) and 2) the heat of the tank contents does not completely counteract the hygroscopic force. Because the heat diminishes relatively soon, the operative assumption must be that soil moisture is drawn to the waste. While no study has examined the detailed situation, it is possible that soil moisture could supply water to a tank indefinitely if a moisture cycle is created. Therefore, without research contradicting the concept of the moisture cycle and quantifying the moisture available via the vapor potential gradient, the conservative assumption must be made that moisture is available indefinitely. Given the limited moisture content expected under the protective barriers, this assumption of water availability may prove overly conservative.

d. Tank Integrity

Approach. Hanford single-shell tanks currently contain salt cake and sludge; they also contain a limited quantity of liquid. The liquid is held within the salt cake and is slow to drain for removal by pumping. At issue is how salt cake and sludge are released from the structural encasement of the tanks. While the tank was designed to contain liquid wastes, it serves also as a barrier to the release of solid wastes. For example, even after welds fail, the tank walls isolate solid wastes from the soil. The approach taken assumes a time (year 2150) after which no credit is taken for tank structure as an inhibitor to the release mechanism.

No studies yet conducted identify specific mechanisms of release or quantify flux from single-shell tanks. Under current climatic conditions, however, corrosion and degradation of steel and concrete structures is slow. It will be hundreds, perhaps thousands, of years before existing single-shell tanks cannot be recognized as such. Some time before complete degradation but after containment (e.g., weld) failures, the tank surface will corrode and generally provide no barrier to soil moisture. It is hypothesized that hygroscopic forces will draw water vapor to the salt cake wastes, dissolve the solid wastes, and leach contamination to surrounding soil.

The approach taken assumes that existing and future localized structural failures do not represent degradation of a significant surface area. However, because corrosion and degradation studies are limited, there is no basis for taking credit for a significant period of isolation. Furthermore, there is currently no basis for a time-series model of tank failure. Consequently, an arbitrary assumption has been made that none of the tanks provides a barrier after the year 2150. This is equivalent to assuming the tanks provide a barrier to significant levels of vapor-phase transport of moisture for another 165 years.

Features of This Approach. The extreme value would assume no credit for the tank structure, that is, a zero for tank integrity. There are, however, no data to suggest that significant releases from the solid waste form are currently occurring.

Geohydrologic Transport

a. Transport in the Vadose Zone

Approach. Contaminants in the vadose zone are assumed to be transported by two mechanisms--advection and/or molecular diffusion. For analyses not involving the protective barrier, the source release is to the surrounding soil water and transport is dominated by the advective mechanism. When the protective barrier is in place, transport is calculated by a combination of diffusion and advection. Releases involving a protective barrier are accumulated at the bottom of the vadose zone to provide a fraction-remaining curve applicable at the water table.

Releases without a protective barrier are assumed to move in a uniformly vertical pathway to the water table. Travel time of water is approximated with a unit hydraulic gradient model. Solute transport is modeled by the advection-dispersion model and the retarded velocity concept, i.e., the linear-sorption isotherm. It is further assumed that these linear-sorption isotherms generated for saturated conditions apply also to unsaturated soils. Pore-water velocity is based on travel time and moisture content, and diffusion-type dispersion values used are appropriate to the vadose zone.

Releases with a protective barrier are modeled by dividing the vadose zone into advection-controlled and diffusion-controlled zones. Releases from the diffusion-controlled zone to the advection-controlled zone are simply translated in time to the water table by the retarded travel time of the radionuclide or chemical. Transport through the diffusion-controlled zone is more complex and is described below.

The protective barrier covers a multidimensional domain. It is modeled as a suite of one-dimensional transport pathways that transmit contamination from the source to either the advection-controlled vadose zone or the water table. Note that either the source or the pathway can act to control the overall release. Because the various sources are poorly characterized, it is assumed that the pathway completely controls the release. No credit is taken for the ability of a source to release at a slow or virtually negligible rate. Source control is neglected by assuming that the source maintains a constant concentration of solution at the source/soil interface. Key parameters are as follows:

1) Soil moisture characteristics and relative permeability curves
2) Average annual infiltration rates
3) Coefficient of molecular diffusion
4) Correction factors for diffusion in the vadose zone, i.e., tortuosity, viscosity and anion exclusion factors
5) Distribution coefficient
6) Constant concentration of solution at source/soil interface.

Features of This Approach. The modeling assumptions and many of the model parameters employed in analyzing transport through the vadose zone are realistic rather than extreme. For example, an ongoing detailed study of two-dimensional movement of moisture beneath the protective barrier is showing the existence of a diffusion-controlled region. Actual soils data are employed; however, soil analyses are too few to guarantee that the most transmissive soils have been used in the analysis. Infiltration rates employed do not represent rigorously defined average annual values because too few annual infiltration data are available. The values used, 0.5 and 5 cm/yr, do not represent extreme values at the high side of the known range. However, the coefficient of molecular diffusion in water is taken as 1 cm^2/day, or nearly twice the cited values and thus is conservative. A correction is applied to the diffusion coefficient to account for partial saturation of the pore space. This correction is based on the effects of tortuosity, viscosity, and anion exclusion observed in partially saturated media. A realistic correction factor is applied.

All distribution coefficients are taken from a study related to single-shell tank contents. The K_d values are reported as functions of salt concentration and of organic complexant content (i.e., dilute or concentrated salts, and complexed or non-complexed waste). These wastes are among the most mobile that could exist at Hanford. While what was introduced to the tanks is largely known, reactions within the tanks and transfers of contents cause some doubt as to their actual contents and their present chemical speciation.

As Hanford sediments are negatively charged under ambient conditions, it is very important to know whether the radionuclide or hazardous chemical is a cationic (positively charged), anionic (negatively charged) or neutral species. Most of the waste streams considered in this EIS are alkaline or neutral and are not expected to change the alkaline nature of the soils and sediments that exhibit a net negative charge. Anionic and neutral species tend to travel with water, whereas cationic species tend to be retarded by the negatively charged soils. A good example of the importance of this factor is a recent experience at Hanford in which uranium entered the groundwater. Carbonate complexing (uranium carbonate anionic complexes) appears to have played a significant role in the mobility of the uranium. This event has prompted review of the original assumptions on the transport of uranium, and K_d values of 0, 7, and 16-mL/g will be tested. Distribution coefficients categorized as complexed-concentrated or complexed-dilute are used for single-shell tank, double-shell tank, and grout waste forms. Only transuranic wastes (soil, buried, new) are modeled with non-complexed values of the distribution coefficients. Because of the mobility of the synthetic wastes tested to determine the distribution coefficients and because of the soil's ability to buffer and disperse the release, values of K_d applied to these wastes are believed to be on the extreme of known values.

Through use of constant concentration of solution at the source/soil interface, the waste form itself is not allowed to decrease or to control the level or rate of release.

The portrayal of transport and dose results contributes to overall conservatism in the HDW-EIS. A range of current climate is employed in the EIS, with annual average infiltration rate characterized as between 0.5 and 5 cm/yr. A low extreme of 0 cm/yr is not beyond reason. The low extreme implies no advective transport in the vadose zone, hence, no release to the water table in the 10,000-year period of interest. To maintain conservatism, however, these potential results (zero transport) are not highlighted in this EIS.

Finally, release calculations are made more conservative through the assumption of a 10-m protective barrier overhang in the simplified analyses. Construction will be preceded, however, by a design phase where an overhang of up to 30 m can be specified. This increased overhang distance will permit flexibility in the design phase for individual sources (e.g., tank farm). More detailed performance assessments based on source- and site-specific data are then assured of meeting or exceeding the performance calculated in this EIS.

b. Transport in the Unconfined Aquifer

Approach. Transport in the unconfined aquifer has been simulated through use of the Hanford groundwater model calibrated to existing conditions, combined with the effects of a constant infiltration rate assumed to apply over the Hanford aquifer watershed. For both annual infiltration rate cases, 0.5 cm/yr and 5 cm/yr, the steady-state version of the Variable Thickness Transient computer code was used to determine a groundwater potential distribution in the unconfined aquifer beneath the Hanford Site. The kinematic pathline method was used to determine a set of pathlines defining the streamtube associated with each given waste disposal site. Starting points for individual pathlines bounded the site perpendicular to the flow field. In the case of the 5-km well, the streamlines were intercepted at 5 km from the 200 Area fence line and the travel time was recorded. In the model of saturated transport, the variation in travel times was used to determine longitudinal dispersion along the flow tube.

Groundwater Modeling for Streamtube Generation. The calculated groundwater potential surfaces assume that after loss of institutional control the infiltration rate in the Pasco Basin remains constant, resulting in a steady-state groundwater system. This assumes no major disposal of imported water or major pumping from the unconfined aquifer.

The transmissivity distribution, representing the spatial distribution of the product of hydraulic conductivity and saturated formation thickness applied in the model of the unconfined aquifer, is determined from tests conducted on Hanford wells and adjusted through model calibration to reproduce the water table under transient conditions. Travel times generated using the modeled potential surface for the existing climate correlate well with those observed, in the Hanford Groundwater Monitoring Program, for low-level tritium and nitrate movement from the 200 Areas.

In the wetter-climate scenario, the net infiltration on hillsides and valleys upgradient from the edges (boundaries) of the model is accumulated and added as flow to the model boundaries. In the case of the lower water table (i.e., current climate), the bottom outcrops above the water table are determined from the approximate aquifer bottom topography and the final predicted water table.

In both climate scenarios, river elevations are held constant at levels corresponding to present average daily flow rates. This is reasonable for the 0.5-cm/yr rate because the base flows of the Columbia River are not determined by the local climate. The shift to a wetter climate, however, would be regional, and river flow and river elevations along the banks would be higher. As the lower elevations were used in this EIS, analysis is conservative since this assumption results in slightly greater gradients and shorter travel times to the Columbia River.

Streamtube Transport. The dimensions of the streamtube for a given waste site were determined by the site's vertical projection on the flow field and its dimension perpendicular to the flow field. The tube width at the 5-km well location is defined by the boundary streamlines. The concentration of a radionuclide at the 5-km well was calculated by using the width between bounding streamlines, the assumed 5-m screened depth of a well, and the average velocity of the groundwater past the well. Five meters was assumed as the nominal mixing depth, based on the fact that any pump capable of providing the 5-gal/min flow rate at a lift of about 70 m would require a submergence of at least 5 m. Any pumping would draw water from above, alongside, and below the pump location. The dimensions are believed to be conservative.

Features of This Approach. Lateral dispersion, or spreading outside of the defined streamtube, will further decrease the concentration of solute in the streamtube. Permitting no transverse or lateral dispersion is conservative because it maximizes solute concentration.

Pathways and Dosimetry

a. Radiation Dosimetry

Approach. Radiation dose is proportional to the quantity of energy deposited per unit mass. Definitions of length of time of exposure and length of time following exposure are what determine the format of the dose reported.

Five basic categories of public radiation doses are used in this EIS:

1) One-year dose from one year of exposure (external plus internal). This is the dose currently used for comparison with occupational exposure standards and the one originally used for comparison with public standards.

   The one-year dose is used in this EIS as a measure of potential short-term impact from accidental releases during waste management operations.

2) Committed dose from one-year external exposure plus extended internal dose accumulated as a result of a one-year intake (ingestion plus inhalation). A 70-year dose-commitment period is used in this EIS.

   The committed dose is used as a measure of the potential longer-term impact of accidents and as a measure of impact of routine releases.

3) Accumulated dose from a lifetime (70 years) of external exposure plus intake via ingestion and inhalation. This includes the effects of radionuclide accumulation or decay in the environment during the exposure period. This dose is most closely relatable to health effects from radiation exposure. This can also be construed as the lifetime committed dose from continuous exposure.

The accumulated dose is used as a measure of the total impact of any operation that results in chronic releases over periods of some duration, or from long-lasting, relatively constant groundwater contamination.

4) Maximum annual dose. This dose is calculated for each year of exposure accounting for each year's external exposure plus the internal dose from nuclides taken in during the year of interest and all previous years. The maximum annual dose is identified by inspection for each organ.

   The maximum annual dose is calculated for scenarios of human intrusion or long-term casual exposure to disposed-of wastes.

5) Integrated or collective doses from very long-term population exposure (up to 10,000 years). This dose is calculated as a sum of lifetime accumulated doses to populations over long time periods. It gives a measure of the total impact of a very long, time-dependent release of radionuclides to the environment.

   The integrated population dose is used for long-term groundwater and surface-water scenarios.

The dose model used is derived from that originally given by the International Commission on Radiological Protection (ICRP 1959) for body burden and maximum permissible concentration. Effective decay energies for radionuclides are calculated using the ICRP model. This model is based on the assumption that the entire quantity of a given radionuclide is located at the center of a spherical organ with an appropriate effective radius (Soldat 1976). Metabolic parameters for the Standard Man are used (ICRP 1975). Some of the parameters are updated from later ICRP publications and some use earlier values (1959).

The internal distribution of radionuclides following inhalation adds some complexity due to variable retention in the lungs. The model of the respiratory tract adopted by the Task Group on Lung Dynamics (ICRP 1966) forms the general basis for the mathematical models developed to calculate the dose from the inhalation of radionuclides.

Features of This Approach. The models used are based on the early ICRP reports. They are mathematically straightforward (using single-exponential retention functions to simulate human metabolic parameters), allowing the formulation of the five dose types described above. The newer dosimetry models, described in ICRP-26 and ICRP-30 and its supplements, are presently available only as a 50-year committed dose, and thus to date they preclude immediate conversion of the set of models used for the EIS analyses. Adaptation of the newer models to use the newer dosimetric data is planned, but it will entail several years' effort.

The actual "dose factors," the calculated values of rem per curie ingested or inhaled, vary both up and down between the ICRP-2 and ICRP-30 models. Most of the major changes are a result of revisions in the metabolic data, rather than in the actual calculational procedures. The ratios of 50-year dose commitments to critical organs for nuclides of importance calculated using ICRP-2 and ICRP-30 models are given in Table 4.

TABLE 4. **Ratios of Dose Factors Calculated Using ICRP-30 Dosimetry Methods for Selected Nuclides**

Radionuclide	Pathway	Critical Organ	Ratio (ICRP-2/ICRP-30) Critical Organ	Ratio (ICRP-2/ICRP-30) Body
Long-Term Groundwater Scenarios				
^{14}C	Ingestion	Bone	1.3	0.27
^{99}Tc	Ingestion	GI Tract	1.5	0.04
^{129}I	Ingestion	Thyroid	0.77	0.07
Operational Releases				
^{90}Sr	Ingestion	Bone	2.4	7.7
^{137}Cs	External	Body	1.0	1.0
^{239}Pu	Inhalation	Bone	0.17	0.15
Intrusion Scenarios				
^{90}Sr	Ingestion	Bone	2.4	7.7
^{137}Cs	External	Body	1.0	1.0
^{239}Pu	Inhalation	Bone	0.17	0.15

For long-term groundwater migration, the radionuclides identified by the transport assessment as being of major importance are ^{14}C, ^{99}Tc, and ^{129}I. The changes in critical-organ dose resulting from the new system shown in Table 4 are between 20% and 50% for these nuclides.

For the operational releases, most doses result from releases of the fission products ^{90}Sr and ^{137}Cs, with a small contribution from ^{239}Pu. The major pathway of exposure for ^{90}Sr is ingestion of crop plants from atmospheric deposition, for ^{137}Cs it is external exposure from fallout, and for ^{239}Pu it is inhalation. Since the doses are dominated by ^{90}Sr and ^{137}Cs, the values in Table 4 indicate that the result is conservatively modeled using the older dosimetry. Most of the difference in the ^{239}Pu dose factor is the result of a change in the definition of "bone" to "bone surface" in ICRP-30, resulting in an effective irradiated mass much less than that used in ICRP-2.

For the intrusion scenarios, the doses at times less than 400 years are controlled by external dose from ^{137}Cs and by ingestion of ^{90}Sr. At longer times, inhalation of resuspended ^{239}Pu becomes the controlling pathway. Thus the Table 4 data indicate that the ICRP-2 dosimetry conservatively models the doses at short times but may be low further in the future.

The ratios of the old "total-body dose" to the new effective "whole-body dose" show somewhat more variation in Table 4. This is because the definitions and methods of calculation have completely changed. However, if the total downriver population doses as reported (in Appendix R) were to be recalculated using the whole-body dose, the total number of health effects for the disposal alternatives would still be in the range 0 to 1 because the population doses are so small.

It can be seen that differences in the dosimetry affect the reported critical organ doses by factors of 2 or less, and total-body (or whole-body) doses by less than an order of magnitude. It must be remembered that the transport, bioaccumulation and uptake parameters of the models used have far more uncertainty and variability than this. The internal dosimetry method used is one of the least variable portions of the final estimate.

b. Definition of Long-Term Exposure Scenarios

Approach. Calculated projections of possible radiation dose have been used in this EIS as a measure of potential impact on public health and safety. The International Commission on Radiological Protection "believes that dose currently is the best available measure of the detriment to an exposed individual, and emphasizes that dose to individuals and collective doses to groups and populations provide the best bases for assessing the implications of radioactive materials in the environment" (ICRP 1979). The approach has been to select representative individuals and critical groups for a wide range of exposure scenarios involving all identified potential exposure pathways.

The potential pathways of exposure to buried wastes are diagrammed in Figure 3. Although no significant releases of radionuclides from disposed wastes are expected during planned operation of the HDW disposal systems, reasonably postulated events that might cause releases or exposures have been investigated. As illustrated in Figure 3, releases from undisturbed sites with functional barriers would be through slow migration of material to groundwater. The critical groups selected for analyses from this release mechanism are, first, users of downgradient domestic water wells in Hanford's unconfined aquifer, and ultimately, all people along the Columbia River between Hanford and the Pacific Ocean. For comparative purposes, sites without protective barriers, which therefore allow the other contaminant migration pathways of Figure 3, have also been analyzed. Erosion and biotic transport lead to localized surface contamination. The critical group for this set of pathways is onsite residents in the absence of institutional controls.

The natural pathways shown in Figure 3 can be abruptly short-circuited by human intrusion. It is impossible to accurately predict human behavior, and a determined individual can ignore, circumvent, or destroy any potential barrier. Therefore, an individual could potentially receive any exposure from incidental to total disruption of a site. For this reason, a set of scenarios has been used illustrating a range of exposure from negligible to total. For those intruding unintentionally who would remain ignorant of their exposure, the residential/house garden and drilling/post-drilling scenarios are used. For systematic intrusion, an excavation scenario is assumed. Because intrusive effects are localized, the critical individuals for these calculations are the intruders themselves.

Features of This Approach. While the scenarios chosen for analysis are representative of many types of potential exposure, the parameters chosen for each are selected to ensure

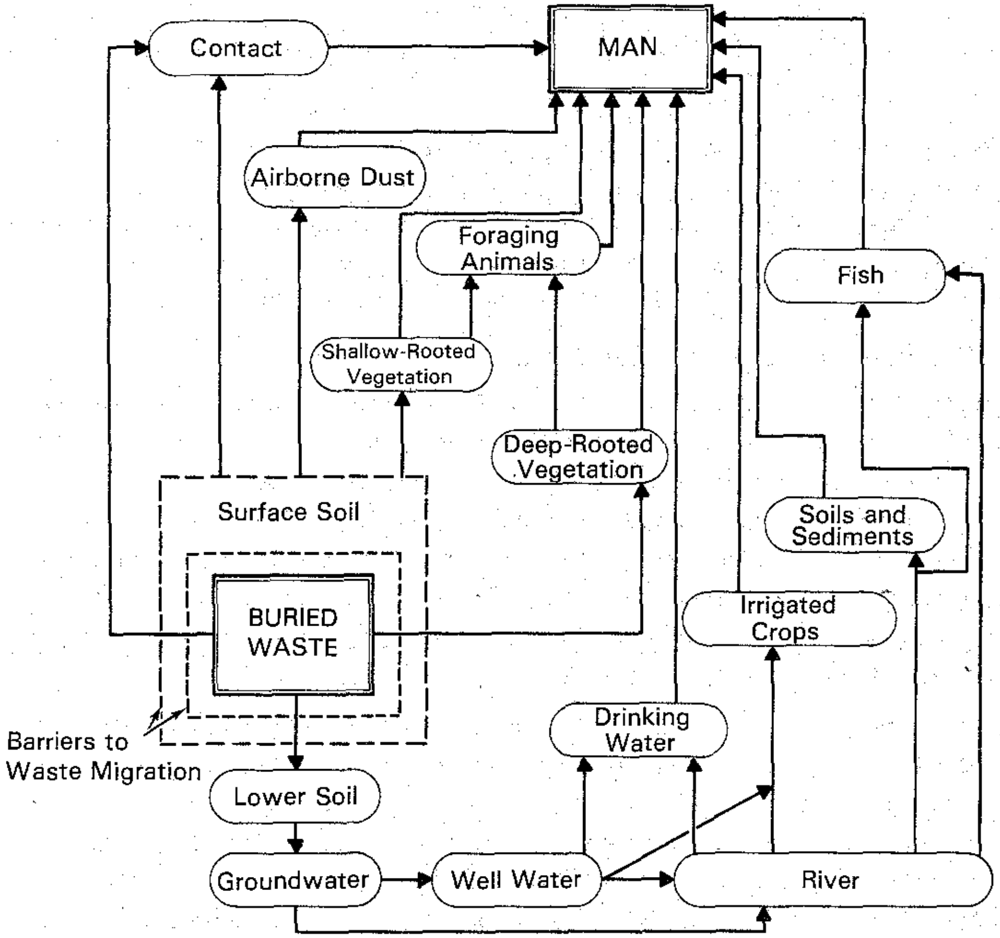

FIGURE 3. Potential Exposure Pathways

that the calculated results contribute toward representing the bounding analysis of consequences. That is, while the extreme scenarios like meteorites or nuclear war have not been analyzed, the more realistic scenarios that are used still give conservative results (i.e., high doses).

For the scenarios involving groundwater interception with a well, the well is assumed to be placed so that it draws water from the central, most concentrated, part of the contaminant plume. The potential uses of the contaminated well water are also maximized: for drinking-water calculations, 100% of the individual's annual liquid intake is assumed to come from the well, and for the extended scenario (i.e., gardening), 100% of the individual's annual diet is assumed to be grown with contaminated irrigation water. To calculate an upper bound on the population doses possible through this set of pathways, a community of small farms was postulated that withdraws 100% of the potentially contaminated water and uses it for irrigation and ingestion.

For the calculations involving the general population downriver of Hanford, essentially all of the radioactive material entering the groundwater eventually reaches the Columbia

River. For the time-variant calculation, a large increase in population over time was assumed. A future population about 10 times the size of the present one is assumed to develop, with essentially all of the people using river water. This large population increase required a large increase in the amount of irrigated land--essentially all productive land along the river between Hanford and the Cascade Mountains is assumed to be put under irrigation. No credit is taken for dilution of the contaminants in the Columbia from tributaries such as the Snake, Yakima, Deschutes, or Willamette Rivers. This additional dilution would potentially reduce the river concentrations by over half.

For the intrusion scenarios, the wastes in each disposal category are represented by the sites with the highest known inventories and/or concentrations of radionuclides. Those drilling or digging are assumed to hit the areas of highest activity. No reduction is taken for the probability, or relative unlikelihood, of a scenario's occurring; the intrusions are simply assumed to occur and the results are presented. For the biotic transport and farming portions of the scenarios, roots from all types of plants, including food crops, are assumed able to penetrate to a depth of up to 10 m; this is a conservative assumption.

Radiation doses to special subgroups, such as children, are not calculated, nor are specialized pathways such as direct ingestion of soil by children. All doses reported are for a Standard Man (ICRP 1977). However, since all doses are calculated on the basis of a lifetime of continuous exposure, these groups and pathways should contribute only a minor portion of the total dose.

Short-Term Consequences

a. Radiological Consequences of Operational Accidents

Approach. An estimate was made of the airborne releases resulting from the conditions imposed upon the materials involved during each unit operation in the proposed processing steps, and the accident with the greatest downwind radiological consequences was chosen to represent the processing step.

Features of This Approach. Each disposal alternative (geologic, in-place stabilization and disposal, reference and preferred) is made up of various processing steps to prepare each waste form for disposal. Processing steps can have more than one unit operation associated with them. The potential airborne release of radionuclides involved in each unit operation resulting from postulated significant accidents is assessed. The accident resulting in the greatest downwind radiological consequences for each unit operation was selected as the maximum credible accident. For processing steps with more than one unit operation, the accident selected as the maximum credible was the one resulting in the greatest downwind radiological consequences from all unit operations in that step. The maximum credible accident for all processing steps in a disposal alternative was selected as the maximum credible accident for that alternative. Wherever more than one accident resulted in similar downwind radiological consequences, both were considered.

The potential airborne releases from the postulated accidents were assessed by various means. For proposed operations that have been analyzed in the published literature, values given for the various scenarios were used. Additional scenarios not evaluated were subjectively considered to determine whether the values used could be exceeded by accidents not covered. For proposed operations not covered in the published literature, airborne releases were evaluated for accidents based upon the materials involved in the operations, the barriers and engineered safety systems assumed to be present, and the experience of the authors in assessing consequences of accidents. The accidents considered were those that result in conditions most conducive to the suspension of the radionuclide forms involved. The quantities of materials involved were limited only by values given for the operation (in some cases all the material). The releases were limited only by experimental data for the airborne release of the same or similar material under similar conditions or physical or chemical laws.

b. Transportation Radiological Exposure Including Accidents

Approach. Following is a list of specific items that are included in the transportation impact analysis. Items 1, 2, and 3 relate to routine (incident-free) exposures, and items 4 to 7 pertain to impacts of transportation accidents.

1) Point Source. Routine radiological exposures are calculated based on the assumption that the shipping cask is a point-source of external penetrating radiation.

   Features of This Approach. The point-source approximation tends to overestimate exposures at distances less than two times the cask length. Since the largest fraction of the routine dose is received by crew members and the population at stops, use of the point-source approximation overestimates the total dose.

2) Truck Stops. Doses to persons at truck locations and/or railyards are calculated assuming no shielding by intervening buildings and the surrounding terrain; it is further assumed that 50 persons are exposed at each truck stop.

   Features of This Approach. Most persons at truck stops are within buildings, such as restaurants, that would provide an additional level of shielding; Madsen and Wilmot (1983) estimate the dose to persons within buildings (assumed to be primarily of concrete) could be reduced by a factor of 7 for a more realistic estimate. This factor-of-7 reduction is not incorporated into the analysis. Madsen and Wilmot (1983) indicate also that an average exposure distance of 50 m is more appropriate than the value of 25 m used in this analysis. This results in a conservative factor of approximately 4.

   Madsen and Wilmot based these findings on the results of a study that followed actual shipments to determine realistic values for these parameters.

3) Crew Exposures. Routine exposures of crew members are based on an average exposure rate of 2 mrem/hr in the truck cab, which is the regulatory maximum value.

   Features of This Approach. Actual shipping experience indicates that dose rates in truck cabs are usually less than 0.2 mrem/hr because of the distance from the cask and shielding provided by intervening material (NRC 1977). Thus, use of the regulatory maximum value tends to overestimate the actual doses received by crew members. Also, according to Madsen and Wilmot (1983), having two truck drivers is common only for large-quantity shipments. Thus, this is the maximum number of drivers for shipments of wastes other than low-level.

The doses to truck or train crew and to persons at stops are found to dominate the total incident-free exposures. The other affected population groups, including persons in vehicles along routes and persons residing along routes, receive insignificant exposures when compared to the former two population groups. As a result, if the doses to crew members and the doses at stops can be considered conservative, the total dose can be considered conservative.

4) Rail Stops. A revised rail-stop exposure model was used to calculate exposures to rail employees that handle rail casks during rail stops (i.e., close-proximity exposures), to rail employees not handling the shipment, and to the general population that surrounds the railyard.

Features of This Approach. The rail-stop model in the RADTRAN computer code was recently revised to incorporate the results of a detailed analysis of rail operations that are important for transportation risk assessments. The report (Madsen et al. 1986) describes in detail the railroad operations that occur at classification yards. These details include such items as the number of times a railroad employee is close to the cask, the distance between the employee and the cask, exposure times, stop times, and population densities in and surrounding railyards. The effect of these revisions on the routine-exposure results is to reduce rail-stop exposures. The revised rail-stop model incorporates more realistic data than the previous model, which was extremely conservative because it lacked these detailed data. Thus, the results are believed to more accurately reflect real-world conditions.

The model is still believed to be conservative because of assumptions regarding the effectiveness of shielding between the cask and railroad employees and between the cask and the population surrounding the railyard. In addition, the population density used to calculate doses to rail employees not handling the cask and to the general population surrounding the railyard is conservatively assumed to be represented by a "suburban" population density (i.e., 719 persons/km^2). However, the more realistic yard population densities calculated from the results of the detailed rail operations analysis range from 30 to 110 persons/km^2. Moreover, areas nearest railyards are generally used for commercial and industrial development. These two items in combination indicate that suburban population density is a conservative assumption.

5) Groundshine, Resuspension, and Ingestion. RADTRAN III calculates radiological health effects (due to transportation accidents) that arise from several pathways. Three pathways considered are groundshine (external exposure to material deposited on the ground); resuspension (inhalation of material that is deposited and then resuspended); and ingestion (eating or drinking contaminated food or liquid).

Features of This Approach. These three pathways could potentially be avoided entirely, and in fact, in actual practice would largely be avoided. The three pathways model the health effects that would result from persons remaining in close proximity to the accident. In actual practice, the first responder to a potential accident involving a release (most likely local police or fire personnel) will close off the immediate area and evacuate nearby residents. Then the released material would be cleaned up and removed and the area would be decontaminated. If there were any serious threats to local inhabitants, they would not be allowed to return to their homes until the area was cleaned up. Any severely contaminated food supplies would not be allowed to be consumed. Therefore, inclusion of the health effects from these three pathways tends to overestimate actual risks of transportation accidents.

6) Accident Frequencies. Estimated accident frequencies used are based on analysis of extensive reports and surveys.

Features of This Approach. Accident frequencies for truck and rail shipments are based on extensive shipper surveys and accident/incident reports compiled by the Department of Transportation (DOT) and the NRC. These data were compiled and analyzed by McClure and Emerson (1980) during development of a "Radioactive Material Accident/Incident Data Analysis Program." Accident frequencies were chosen near the high end of the expected distribution, based on analysis of an extensive data base.

7) Respirable and Dispersible Fractions. Two fractions describe the dispersibility of radioactive materials after their release. Due to a lack of data, very conservative assumptions were made for strontium/cesium capsules.

Features of This Approach. For high-level waste (HLW), the fraction of glass that is assumed to be converted to respirable particles is assumed to be 5×10^{-8}. This value was chosen from the upper end of the distribution of particle size after a 30-km/hr impact test (Ross and Smith 1975). The value chosen for the EIS is a factor of 100 larger than that used for the Defense Waste Processing Facility (DWPF) EIS (DOE 1982). That is, an extreme value was chosen because of data limitations.

For strontium/cesium capsules, the respirable and dispersible fractions are each assumed to be 1.0. Thus, all of this material is assumed to consist of particles less than 10 microns in diameter. As no actual data exist related to the dispersibility and respirability of this waste form after a transportation accident, a highly conservative value was assumed.

It was assumed that TRU wastes in a cast-slag form would be represented by the dispersibility characteristics of a sintered ceramic such as UO_2. This was assumed because the dynamic response of cast slag to accident conditions is relatively unknown. It is believed that this material would behave similarly to a HLW glass waste form, but in the absence of adequate data, conservative values were chosen.

For retrievably stored and newly generated TRU wastes, respirable and dispersible fractions representative of loose, small powders were chosen to bound the impacts. These values are roughly equivalent to the airborne release fractions from combustion of contaminated, flammable (paper, plastic, rubber, etc.) wastes (Mishima and Schwendiman

1973) and represent average values near the high end of the distribution. An extensive testing program was performed to obtain these results.

8) Release to Dose Conversion. Factors that convert radioisotope release concentrations to a value of exposure were taken from several standard sources. The values of air submersion and inhalation dose factors are best estimates based on extensive testing and analysis of radionuclide uptake. The values of inhalation dose factors were based on the International Commission on Radiological Protection (ICRP 1977) and Hoenes and Soldat (1977). Air submersion dose factors were obtained from Kocher (1980). As these are best estimates rather than very conservative ones, they do not contribute to a bounding analysis.

Radiologically Related Health Effects

Approach. For the estimation of health effects resulting from exposure to radiation or radionuclides, a linear, nonthreshold relationship between total population dose and health effects is used. Health effects considered in the analysis are fatal cancers to the exposed individuals and specific genetic effects to all generations.

A range encompassing commonly used cancer risk factors is employed, compatible with BEIR III(a) and UNSCEAR,(b) but also including new data from revised studies of atomic bomb survivors that indicate the earlier estimates may be slightly low. The range of total fatal cancers used is an incidence of 50 to 500 per 10^6 man-rem (see Appendix N of this EIS for details).

A range of 50 to 500 specific genetic effects, caused by exposure of the germ cells (dominant and x-linked, chromosomal and multifactorial), is employed in the health-effect estimates. This is essentially the range recommended in the BEIR III report.

No special risks are considered to be associated with any specific radionuclide except as reflected in the calculation of their dose equivalent (in rem).

Features of This Approach. The full range of dose to health effects conversion factors is used, based on the most widely reported values. No correction is made for dose rate; only the total cumulative population dose is used in the calculations. Some data indicate that, at the very low dose rates generally reported in the EIS, there is a possibility of no effects.

In all cases where health effects are reported, the full range is used to give a low and a high estimate. Values in the lower range of the risk estimates are most appropriate for comparison with the estimated risks of other energy technologies. Values from the high end of the range are more appropriate for planning and as maxima for radiation-protection considerations. Combined with the conservatisms in the exposure and radiation-dosimetry calculations, these provide an upper estimate of the potential for long-term adverse effects.

(a) Report of the Advisory Committee on the Biological Effects of Ionizing Radiation, Division of Medical Sciences, National Academy of Sciences (NAS/NRC 1980).

(b) United Nations Scientific Committee on Atomic Radiation.

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APPENDIX A

WASTE SITE DESCRIPTIONS AND INVENTORIES

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APPENDIX A

WASTE SITE DESCRIPTIONS AND INVENTORIES

The waste sites addressed in Section 3.2 of this EIS are described below in more detail, together with the expected inventories of major radioactive materials at these sites. Inventories of selected nonradioactive materials are also given for wastes stored in tanks. The engineering data in this appendix were obtained primarily from the engineering support data provided by Rockwell Hanford Operations (Rockwell 1985); data without explicit references are generally from this source. The numerical information is the most accurate data available. The existing waste inventories are based on historical records and are believed to be adequate for the generic waste class descriptions. Future characterization of wastes will be necessary to provide more detail, and in some cases is already under way.

Most sites are described as of January 1984. Existing tank waste is described as of October 1983 (before resumption of operations at the PUREX Plant); wastes generated after that time are included in future tank waste. Projections of future tank waste and newly generated TRU solid waste are through 1996. Radioactive decay for all waste is calculated to December 31, 1995.

A.1 EXISTING TANK WASTE

Four major types of waste are contained in existing tank waste:

1. sludges produced from components of high-level waste that precipitate when the waste is neutralized
2. salt cake produced when waste supernatant liquids are concentrated beyond the solubility limit of a major component
3. double-shell slurry, which is the supernatant liquid after salt cake formation
4. complexed concentrate produced by concentration of wastes containing large amounts of organic complexing agents.

The first two types of waste are predominantly stored in older (single-shell) tanks, the latter two are in newer (double-shell) tanks. Some supernatant liquid is contained in older tanks and is being transferred as completely as possible to new tanks. This transfer is expected to be complete before disposal actions begin.

A.1.1 Waste Tank Descriptions

A schematic layout of the Hanford tank farm facilities is shown in Figure A.1. Detailed descriptions of the waste tanks are given in the final environmental statement for current waste management operations (ERDA 1975; DOE 1980). The single-shell and double-shell tanks are briefly described below.

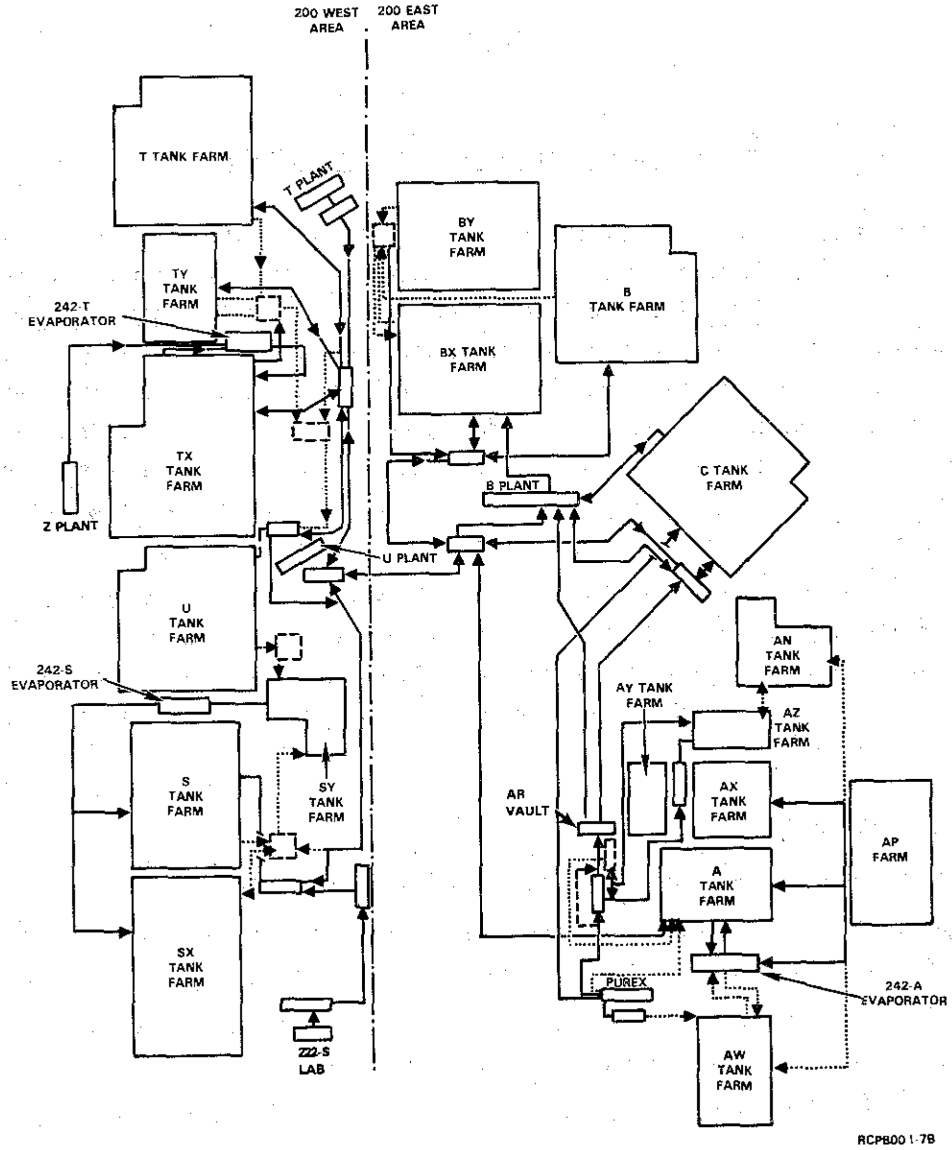
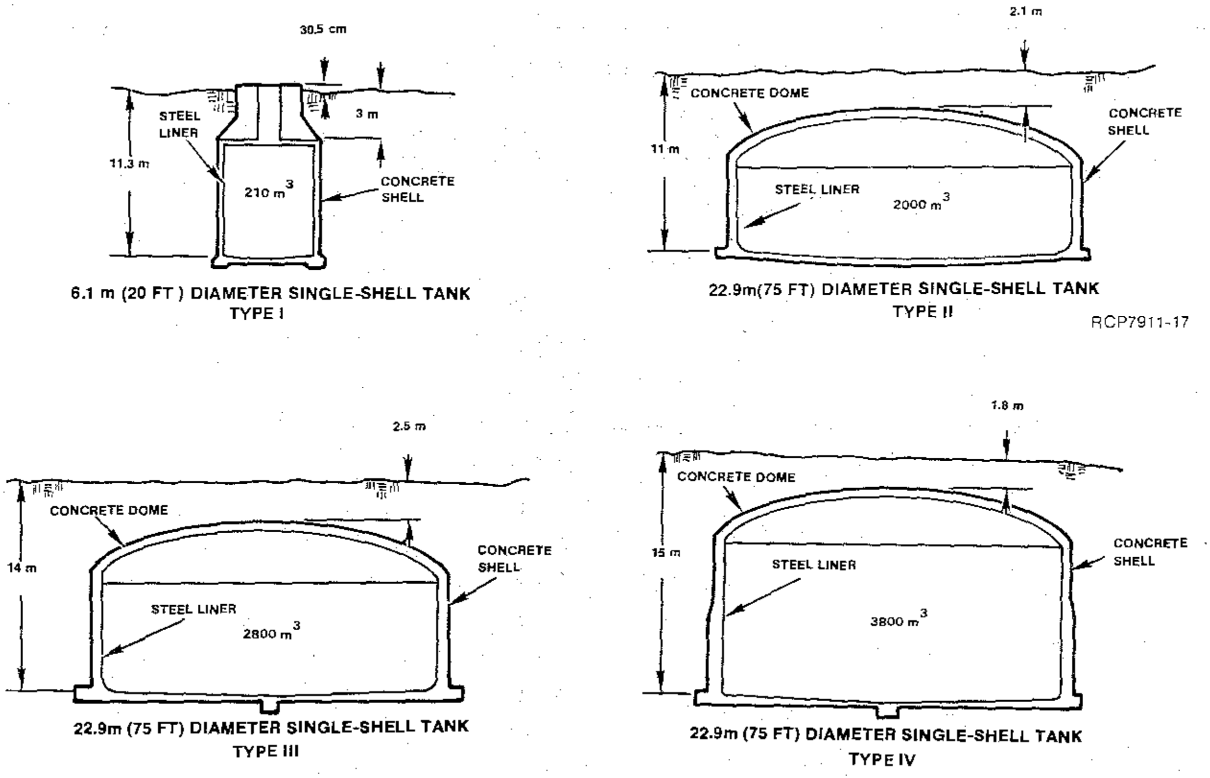


FIGURE A.1. **Schematic Representation of Hanford Tank Farm Facilities** (Not to scale. Farms in 200 East and 200 West Areas are about 10 km apart.)

A.1.1.1 Single-Shell Tanks

Hanford's underground single-shell tanks (Figure A.2) are reinforced concrete with carbon steel liners on the bottoms and sides. The 149 single-shell tanks, ranging in capacity from 210 to 3,800 m^3 (Table A.1), were constructed between 1944 and 1964. Distances from the tank bottoms to the water table are given in Table A.2.

Forced ventilation currently provides cooling for 16 tanks containing materials which, through radioactive decay, generate heat that could exceed established concrete temperature limits. Single-stage high-efficiency particulate air (HEPA) filters allow atmospheric



6.1 m (20 FT) DIAMETER SINGLE-SHELL TANK TYPE I

22.9m(75 FT) DIAMETER SINGLE-SHELL TANK TYPE II

22.9m (75 FT) DIAMETER SINGLE-SHELL TANK TYPE III

22.9m (75 FT) DIAMETER SINGLE-SHELL TANK TYPE IV

FIGURE A.2. Underground Single-Shell Tanks

breathing for tanks that do not require cooling. Gases generated by radiolytic decomposition disperse in this manner. Maintenance and repair of equipment, as well as testing of filters, is provided.

Concrete in the single-shell tanks has maintained its integrity, preventing tank collapse, during many years of service (ERDA 1977; DeFigh-Price 1982; Dahlke and DeFigh-Price 1983).

A.1.1.2 Double-Shell Tanks

A cross section of a typical double-shell tank is shown in Figure A.3. The tank-within-a-tank design provides double containment and ensures complete containment in the event of a leak in the primary (inner) shell. The steel inner tanks have been stress-relieved to prevent failure from stress-corrosion cracking.

Twenty double-shell tanks, each with a volume of 3,800 m^3 to 4,300 m^3, were constructed between 1970 and 1982 (Table A.1). Eight more have been constructed in AP Farm (see Figure A.1). Additional double-shell tanks will be constructed as necessary. Distances from tank bottoms to the water table are given in Table A.2.

Further details on the new double-shell tanks are given in the final environmental impact statement supplement for double-shell tanks (DOE 1980).

TABLE A.1. Tank Farm Summary

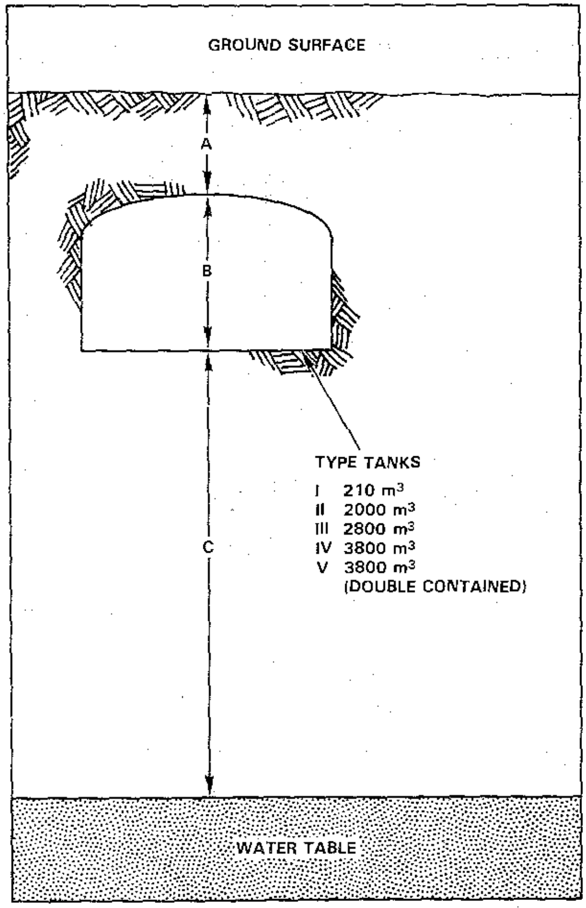
Tank Farm	Number of Tanks	Capacity, m^3 Per Tank	Capacity, m^3 Total
A	6	3,800	22,800
AN(a)	7	4,300	30,100
AP(b)	8	4,300	34,400
AW(a)	6	4,300	25,800
AX	4	3,800	15,200
AY(a)	2	3,800	7,600
AZ(a)	2	3,800	7,600
B	12	2,000	24,800
	4	210	
BX	12	2,000	24,000
BY	12	2,800	33,600
C	12	2,000	24,800
	4	210	
S	12	2,800	33,600
SX	15	3,800	57,000
SY(a)	3	4,300	12,900
T	12	2,000	24,800
	4	210	
TX	18	2,800	50,400
TY	6	2,800	16,800
U	12	2,000	24,800
	4	210	
Total	177(c)		471,000

(a) Double-shell tanks.
(b) Double-shell tanks under construction.
(c) Does not include four double-shell tanks, 4,300 m^3 each, in the proposed AQ tank farm.

A.1.2 Inventories in Existing Tank Waste

Volumes and compositions of waste in individual tanks or tank farms vary considerably, depending on the waste source and on past waste management practices at the respective tank farms. Alternatives for disposing of this waste are dependent on the storage mode, radioactive content, thermal release due to radioactive decay, and chemical and physical form of the waste and thus on the ease of retrieval and processing. Waste in single-shell tanks has generally been converted to solid forms (sludge or salt cake) to reduce the chance of content leakage to surrounding soil in the event of tank failure. Residual liquids (aqueous

TABLE A.2. Hanford Waste Storage Tanks



	NUMBER OF TANK TYPES					DIMENSIONS (m)		
TANK FARM	I	II	III	IV	V	A	B	C (APPROX)
A				6		2.3	14.3	70.1
AX				4		2.3	14.3	65.8
AY					2	2.1	14.9	65.8
AZ					2	2.1	14.9	59.4
B	4					1.8	6.1	64.3
B		12				1.8	8.5	61.9
BX		12				2.7	9.8	66.4
BY			12			2.4	11.9	60.0
C	4					1.8	6.1	69.5
C		12				2.7	8.5	66.1
S			12			2.4	11.9	43.9
SX				15		2.3	14.3	43.0
T	4					1.8	6.1	53.9
T		12				2.7	8.5	50.6
TX			18			2.4	11.9	51.2
TY			6			2.4	11.9	46.6
U	4					1.8	6.1	53.6
U		12				2.7	8.5	50.3
AW					6	2.1	14.9	69.5
AP					8	2.5	14.9	67.4
AN					7	2.1	14.9	63.4
SY					3	2.1	14.9	42.1

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solutions) in these tanks are jet pumped, leaving as small a liquid heel (less than 190 m^3) in the tanks as practicable. Waste inventories are presented as expected values after completion of this solution transfer.

The estimated mass of chemical components associated with existing in-tank waste is shown in Table A.3. The estimated radionuclide content in the tanks after completion of jet-well pumping is summarized in Table A.4. The maximum inventory of TRU and fission products in any one tank is given in Table A.5.

A.1.3 Single-Shell Tank Characterization

Single-shell tank waste characterization includes the development of a plan to assemble existing characterization data and to acquire additional data as required. The waste characterization data will be used to form a data base for the single-shell tank wastes.

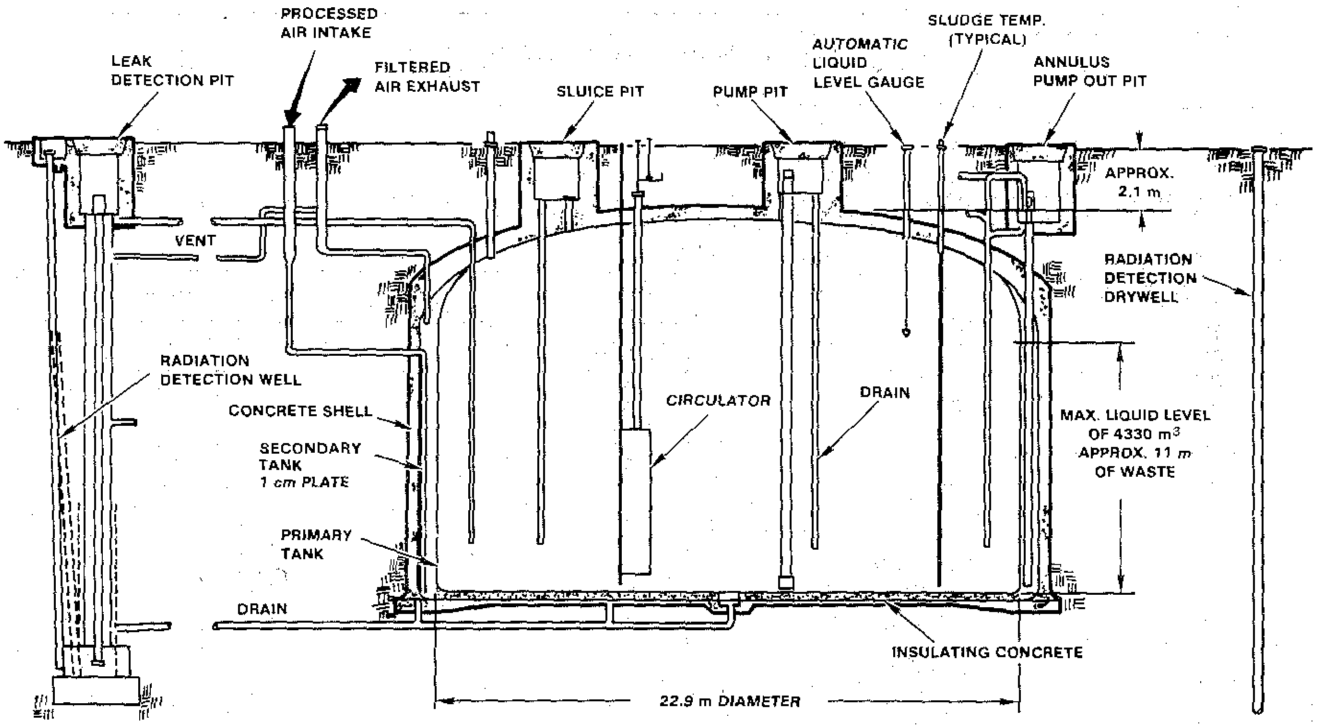

FIGURE A.3. Double-Shell (Type V) Storage Tank

Two methods are currently being pursued to characterize wastes in single-shell tanks: 1) simulation modeling and 2) sampling and analysis. Development work is required to calibrate the computer model used to estimate the total waste inventory and the distribution of waste components among tanks. Core-sampling equipment, which takes waste samples from the tanks while maintaining the waste layers, has been demonstrated. Wastes in nine tanks have now been sampled and analyzed for comparison with computer model predictions. The tanks chosen for sampling were predicted to contain significant amounts of key radionuclides (^{14}C, ^{129}I, ^{99}Tc, ^{239}Pu, ^{240}Pu, and ^{241}Am). Computer predictions for ^{14}C were about 1,000 times higher than the amount actually found, and may be highly conservative for this radionuclide. The actual amount of ^{239,240}Pu found is about three times the model prediction. Agreement was much better for the other radionuclides; the computer predictions were within the 95% confidence interval of analyses for at least half of the tanks (Rockwell 1987).

A.1.4 Transfer of Liquid from Single-Shell Tanks

The transfer of pumpable liquids from single-shell tanks to double-shell tanks is part of an ongoing program of waste stabilization. Waste disposal alternatives are based on completion of this program before initiation of disposal actions. The transfer process is briefly described below.

The typical single-shell tank contains a layer of salt cake on top of a layer of sludge (Figure A.4). Tanks may contain only salt cake or sludge. The void spaces of the salt cake and sludge contain liquids, which are removed by salt well jet pumps.

TABLE A.3. **Chemical Components Associated with Existing Tank Radioactive Wastes, t**[a]

Chemical[b]	Total Bulk Sludge	Total Bulk Salt Cake	Liquid	Double-Shell Slurry[c]	Complex Concentrates[c]	Total
Al	1,100	630	490	3,000	110	5,400
Bi	260					260
Ca	130					130
Cd	4					4
Ce	230					230
Cr	96					96
Fe	630				80	710
Hg	0.9					0.9
Mn	120					120
Na	15,000	34,000	2,300	15,000	2,300	69,000
Ni	180					180
Zr	250					250
CO_3	1,200	410	40	260	720	2,600
Cl	40					40
F	800		5	19	7	830
$Fe(CN)_6$	320					320
NO_3	15,000	80,000	1,800	12,000	3,100	110,000
NO_x	2,000	1,500	1,300	8,200	280	14,000
OH	5,200	2,400	1,600	9,700	580	19,000
PO_4	7,400	1,200	160	1,100	46	9,900
SO_4	500	1,100			120	1,700
Cancrinite[d]	2,700					2,700
Organic Carbon			200	430	1,100	1,700
H_2O	26,000	13,000	4,200	13,000	6,000	63,000
Totals	79,000	130,000	12,000	63,000	14,000	300,000

(a) After completion of jet pumping. The volume of single-shell tank waste is 140,000 m^3 and double-shell tank waste is 45,000 m^3, for a total of 190,000 m^3.
(b) Most minor components (<100 t total) are not listed.
(c) Based on limited sample analyses and predicted volumes of complex concentrate.
(d) Known silica additions are assumed to have reacted with aluminates and hydroxides to form cancrinite (assumed to be $2NaAlSiO_4 \cdot 0.52NaNO_3 \cdot 0.68H_2O$).

TABLE A.4. Estimated Major Radionuclide Inventories of Existing Tank Waste, Ci, Decayed to the End of 1995[(a)]

Radionuclide	In Single-Shell Tanks	In Double-Shell Tanks	Total Input to Tanks
^{241}Am	3×10^{4}	2×10^{4}	4×10^{4}
^{243}Am	2×10^{1}	2×10^{1}	3×10^{1}
^{14}C	3×10^{3}	2×10^{3}	5×10^{3}
^{244}Cm	7×10^{1}	7×10^{1}	1×10^{2}
^{135}Cs	7×10^{1}	7×10^{1}	1×10^{2}
^{137}Cs(b)	1×10^{7}	1×10^{7}	2×10^{7}
^{129}I	2×10^{1}	2×10^{1}	5×10^{1}
^{63}Ni	3×10^{5}	2×10^{4}	3×10^{5}
^{237}Np	3×10^{1}	3×10^{1}	6×10^{1}
^{238}Pu	4×10^{2}	1×10^{1}	5×10^{2}
^{239}Pu	2×10^{4}	8×10^{1}	2×10^{4}
^{240}Pu	5×10^{3}	2×10^{1}	5×10^{3}
^{241}Pu	5×10^{4}	4×10^{2}	5×10^{4}
^{226}Ra	2×10^{-5}	1×10^{-7}	3×10^{-7}
^{106}Ru(b)	7×10^{-1}	4×10^{-1}	1
^{79}Se	8×10^{2}	1×10^{2}	9×10^{2}
^{151}Sm	6×10^{5}	2×10^{5}	8×10^{5}
^{126}Sn	6×10^{2}	2×10^{2}	8×10^{2}
^{90}Sr(b)	4×10^{7}	8×10^{6}	5×10^{7}
^{99}Tc	2×10^{4}	1×10^{4}	3×10^{4}
^{235}U	2×10^{1}	9×10^{-2}	2×10^{1}
^{238}U	5×10^{2}	2	5×10^{2}
^{93}Zr	4×10^{3}	2×10^{2}	4×10^{3}

(a) Status after completion of jet pumping.
(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in modeling.

The jet pump is inserted into a salt well that reaches to the bottom of the tank (Figure A.4). A centrifugal pump circulates liquid through a submerged jet within the recirculation loop. A reduced pressure, caused by the jetted liquid, results in the removal of liquid that has drained to the bottom of the tank. The heels are expected to be less than 190 m^3 per tank, probably less than 120 m^3 (Murthy et al. 1983). Although no adverse radiological impacts have been postulated for tank leaks, including the ultraconservative assumption of a 40,000 m^3 release (Murthy et al. 1983), DOE policy is to reduce the potential for any liquid release whenever practicable.

TABLE A.5. Maximum Inventories in Any One Tank of Existing Waste, Ci, Decayed to the End of 1995

Radionuclide	TRU Tank(a)	Fission Product Tank(a)
^{241}Am	1×10^{4}	5×10^{2}
^{243}Am	1×10^{1}	2×10^{-1}
^{14}C	6×10^{2}	5×10^{-7}
^{244}Cm	5×10^{1}	--
^{135}Cs	8×10^{-1}	7
^{137}Cs(b)	2×10^{5}	6×10^{5}
^{129}I	2	--
^{63}Ni	4×10^{4}	3×10^{3}
^{237}Np	6×10^{-2}	3×10^{-3}
^{238}Pu	5×10^{1}	4×10^{1}
^{239}Pu	2×10^{3}	3×10^{3}
^{240}Pu	6×10^{2}	6×10^{2}
^{241}Pu	4×10^{3}	6×10^{3}
^{226}Ra	5×10^{-7}	1×10^{-7}
^{106}Ru(b)	2×10^{-1}	1×10^{-3}
^{151}Sm	1×10^{5}	2×10^{4}
^{126}Sn	9×10^{1}	1×10^{1}
^{90}Sr(b)	3×10^{6}	7×10^{6}
^{99}Tc	1×10^{3}	--
^{235}U	6×10^{-1}	1×10^{-1}
^{238}U	1×10^{1}	3
^{93}Zr	5×10^{2}	8×10^{1}

(a) Tank 105-C contains the maximum TRU inventory. Tank 106-C contains the maximum fission-product inventory.

(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in modeling of radiological impact.

A.1.5 Double-Shell Tank Waste Characterization

Ongoing efforts to characterize radioactive waste have included laboratory studies to identify organic constituents in the waste. Some double-shell tank waste contains a number of organic compounds that were used by the Hanford Site chemical processing facilities. Due to the thermal and radiolytic history of the waste, it is likely that a significant fraction of the organic materials has volatilized, decomposed, or polymerized.

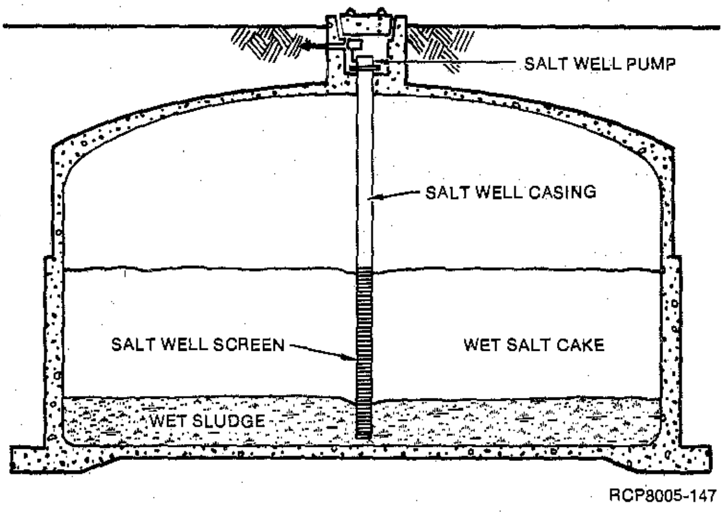

FIGURE A.4. **Jet Pumping of Solutions from Single-Shell Tanks**

Most of the organic compounds in the waste were introduced as chelating agents during strontium recovery processing at B Plant. The chelating agents used by B Plant were hydroxyacetic acid, citric acid, hydroxyethyl ethylene diamine triacetic acid (HEDTA) and ethylene diamine tetraacetic acid (EDTA). These compounds were used in approximately the following respective ratios: 25, 25, 40, and 10 wt%. Small amounts of these compounds may be found in multiple waste streams due to residuals after tank-to-tank transfers. Most of these organics and their degradation products are found in organic complexant waste.

Many of the chemical separation processes are based on extraction of the desired species from an aqueous solution by an organic solvent. Thus, trace quantities of organic solvents are present in the wastes. Organic solvents previously used or in use at the Hanford Site include bis-(2-ethylhexyl)-phosphoric acid, methyl isobutyl ketone, hexone, tributyl phosphate, normal paraffin hydrocarbon (kerosene) and carbon tetrachloride.

Sodium gluconate was introduced into the waste by the B Plant solvent cleanup process. Sugar (sucrose) is used in the PUREX plant to destroy nitric acid. Most of the sugar is decomposed by this process.

Recent analytical data have revealed that a large volume of organics has decomposed or polymerized. Preliminary data, presented in Table A.6, show that in a sample of neutralized cladding removal waste (NCRW), 95 wt% of the organic carbon present in the waste can be identified. The NCRW does not normally contain organics. The presence of organics in this sample was due to residual waste in the tank prior to the addition of the NCRW. However, analyses of double-shell slurry (DSS) can identify only 1 wt% of the organic carbon present. Future laboratory work will investigate postulated polymeric compounds present in DSS. Preliminary analytical data on organic complexant waste, as shown in Table A.7, allow 75 wt% of the organics present to be identified (Rockwell 1987). While it is probable that the decomposition or polymerization has reduced the complexing capability of the organics, further characterization and testing are needed to provide definitive information.

TABLE A.6. **Organics Identified in Neutralized Cladding Removal Waste and Double-Shell Slurry Wastes** (Rockwell 1987)[a]

Organics	Neutralized Cladding Removal Waste Carbon Concentration in Waste, (μg/g)	Double-Shell Slurry Wastes Carbon Concentration in Waste, (μg/g)
Solvent extractable		
Tri-n-butylphosphate	380	7
n-Undecane		<1
n-Dodecane	87	2
n-Tridecane	430	8
n-Tetradecane	390	4
n-Pentadecane		2
n-$C_{22}H_{46}$--n$C_{34}H_{70}$		
Butylbenzylphthalate		
Dicotylphthalate	24	5
Unknown phthalates		
Volatile		
Acetone		
Methylene chloride		
Chloroform		
Unknown (mol wt 75 or 76)		
Chelating complexing agents		
Citric acid	130	8
Ethylenediaminetetraacetic acid (EDTA)	170	22
Nitrilotriacetic acid (NTA)		2
Chelator fragments		
Methanetricarboxylic acid	64	1
Ethylenediaminetriacetic acid (ED3A)		1
Carboxylic acids		
Pentanedioic acid	130	
Hexanedioic acid	610	
Heptanedioic acid	330	
Octanedioic acid	1,000	
Nonanedioic acid	580	
Undecanoic acid		
Pentadecanoic acid	280	6
Heptadecanoic acid	310	2
Total organic carbon	5,200	5,900
Percent total organic carbon identified	95%	1%

(a) No entry indicates that compound is below detection level. Exact contributions of unknown organics to waste total organic carbon content cannot be determined unequivocally. Total organic carbon analysis performed by combustion with coulometrictitration.

TABLE A.7. Organics Identified in Organic Complexant Waste (Rockwell 1987)

Organics	Carbon Concentration in Waste, (µg/g)
Chelating/complexing agents	
Citric acid	3,800
N-(2-Hydroxyethyl)ethylenediaminetriacetic acid (HEDTA)	3,800
Ethylenediaminetetraacetic acid (EDTA)	3,100
Methane Tricarboxylic acid	1,200
Nitrilotriacetic acid (NTA)	440
Chelator fragments	
Ethylenediaminetriacetic acid (ED3A)	1,400
N-(2-Hydroxyethyl)ethylenediamine-N'N'-diacetic acid (HEDDA)	220
N-(ethylene)ethylenediaminetriacetic acid (E_2DTA)	190
N-(2-Hydroxyethyl)iminodiacetic acid (HEIDA)	150
N-(2-Hydroxyethyl)-N'-(methyl)ethylenediamine-'N,N'-diacetic acid (MeHEDDA'A)	170
N-(methyl)ethylenediamine-N,N'-diacetic acid (MeEDD'A)	70
Iminodiacetic acid (IDA)	9,100
Molecular weight (mol wt) species(a)	
A: mol wt 122	30
F: mol wt 173	20
J: mol wt 247	70
Carboxylic acids	
Docos-13en-oic acid	560
Hexanedioic acid	130
Hexadecanoic acid	330
Phthalic acid	80
Nonanedioic acid	60
Tetradecanoic acid	100
Pentanedioic acid	30
Octadecanoic acid	90
Hydroxybutanedioic acid	8
Butanedioic acid	8
Alkanes	
nC_{23}-nC_{35}	2,100
Phthalate esters	
Dibutylphthalate	190
Dioctylphthalate	8
Total organic carbon	37,000
Percent total organic carbon identified	75%

(a) Molecular weights assigned to unknown chelator fragments on the basis of electron impact (70 eV) gas chromatograph-mass spectrometer.

A.2 FUTURE TANK WASTES

Future tank wastes include PUREX Plant wastes generated during current operations, which began in November 1983, together with liquid wastes from other sources projected through 1995. All of these wastes are stored in double-shell tanks. The source, composition, and amount of future tank wastes are described below.

A.2.1 Future High-Level Tank Waste

High-level waste from PUREX Plant operations are neutralized and stored in tank farms.[a] The design specifications for the four tanks equipped with air lift recirculators limit the heat content to 1 x 10^7 Btu/hr/tank, or 2.9 MW (ERDA 1975). Approximately 11,000 m^3 of high-level waste (HLW) are generated from the processing of about 12,000 t of N Reactor fuel or its equivalent.

The following discussion addresses some of the characteristics of future tank waste to be considered in adopting any of the disposal alternatives.

A.2.1.1 Current Acid Waste

Although acid waste (in-process HLW) is not one of the wastes considered for disposal, it is described here because it is the immediate precursor of stored HLW. On restart of the PUREX facility in November 1983, the flowsheet used was essentially the same as that used when the plant was shut down in 1972. About 0.5% of the plutonium is not recovered in the PUREX process and will likely be in the acid waste or cladding removal waste. Plutonium is separated from uranium by reducing the plutonium to an oxidation state that is inextractable by the PUREX process solvent. Iron(II) sulfamate, used to effect this reduction, is converted to iron(III) and sulfate and becomes part of the acid waste.

The sulfate content in acid waste may be important to waste management, especially if the waste is to be vitrified. Only 0.1 to 0.2 wt% sulfate can be incorporated in borosilicate waste glass without introducing an undesirable crystalline phase in the glass that would decrease the product's stability and increase the rate of radionuclide leaching from the glass. Thus, although the waste volume may be kept small, the presence of sulfate in the waste could result in a large volume of glass. Therefore, the sulfate content of acid waste must be reduced or the waste will require sulfate removal before it can be converted to glass. The Department of Energy plans to make early plant tests to substitute other reductants in place of iron(II) sulfamate and thus to avoid introducing sulfate to the acid waste. This substitution, assumed in this EIS to begin in 1987, will reduce the amount of glass produced.

A.2.1.2 High-Level Waste (HLW)

The acid waste is neutralized and, until radiolytic heat declines, stored in double-shell tanks equipped with air-lift circulators to control boiling of the waste. Neutralization is necessary because the carbon steel tanks will corrode rapidly if acid solutions are

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(a) A proposed new tank farm (AQ) that would contain four double-shell tanks is not addressed in this EIS, but the waste that would be stored in these tanks is addressed.

stored in them. Circulators are necessary to prevent pressure surges and to minimize entrainment of radionuclides in the off gas caused by uneven boiling due to the radioactive decay heat. Circulators also serve to prevent overheating of tanks from sludge hot spots.

The HLW in double-shell tanks separates into two phases: 1) a solid phase (sludge of insoluble materials), consisting mostly of hydroxides or hydrated oxides that are insoluble in the highly alkaline aqueous solution and 2) a supernatant liquid consisting of an aqueous solution of sodium nitrate (resulting from the neutralization of nitric acid), sodium nitrite (resulting from radiolytic reduction of sodium nitrate), sodium sulfate, sodium hydroxide, and sodium aluminate (resulting from aluminum additions to complex fluoride ions).

The sludge contains most of the fission products (except cesium and technetium) and the TRU elements. The supernatant liquid contains essentially all of the cesium and technetium, iodine not removed in the head-end process, and some of the ruthenium. The supernatant liquid is expected to contain only small concentrations (i.e., <<100 nCi/g) of TRU elements.

The two-phase aqueous slurry made from high-sulfate-content acid waste cannot economically be processed directly to a borosilicate glass because the sulfate influences the quantity of waste that could be incorporated into glass. The sludge is separated from the supernatant liquid and washed free of sulfate. The washed sludge is then incorporated into glass with about 25 wt% dried sludge in the glass. The waste glass contains the insoluble fission products and actinides (residual uranium, plutonium, and other TRU elements in the waste). Thus, because of the sulfate-free sludge, the volume of glass product is not limited by sulfate content, and a smaller volume of waste glass is produced. The sulfate removal step also removes soluble aluminate from the sludge, further decreasing the volume of glass produced. The volume of glass produced from washed HLW sludge is therefore even less than that produced from HLW without sulfate. The supernatant liquid and sludge washes are incorporated into grout.

A.2.2. Future Non-High-Level Tank Waste

Cladding removal waste (CRW) is generated during operation of the PUREX Plant by dissolving Zircaloy cladding using the Zirflex process (DOE 1982). Organic wash waste (OWW) is generated during routine treatment of the PUREX Plant organic solvent to allow recycle and reuse. These two non-high-level waste streams and miscellaneous sump waste may be combined and stored in double-shell tanks (DOE 1980).

Three additional sources of waste--low-level waste from N Reactor fuel processing, waste from Plutonium Finishing Plant operations, and other customer waste--are dilute solutions that are not high-level when they are generated. However, after concentration at the tank farm, these wastes are likely to exceed 100 nCi TRU/g, the concentration above which waste becomes TRU waste.

All of these waste types are described below.

A.2.2.1 Cladding Removal Waste

The Zircaloy cladding on N Reactor spent fuel is dissolved chemically in the Hanford PUREX Plant by reaction with an aqueous ammonium fluoride solution containing ammonium nitrate to suppress hydrogen evolution (the Zirflex process). The dissolvent slowly attacks the uranium metal as it is exposed after cladding removal. The resulting uranium (IV) fluoride contains associated actinides and fission products. Centrifugation removes most of these insoluble fluorides from the waste, but 0.1 to 0.5% of the actinides may remain. Neutralization of the cladding waste causes precipitation of most of the zirconium as a hydrated oxide, removing essentially all of the actinides and fission products from the solution. Thus, the precipitate from untreated cladding waste may be a TRU waste (≥100 nCi/g) but the supernatant liquid is expected to be a non-TRU, low-level waste.

The Department of Energy intends to define by sample analyses the concentration of TRU elements in cladding waste and to explore technology for treating the waste to remove TRU elements. Since the TRU level in cladding waste is currently uncertain, a conservative method was selected for estimating plutonium and americium. This method predicts a conservatively high concentration of ^{241}Am in untreated cladding waste. Therefore, a TRU removal step (such as a rare-earth-fluoride scavenging precipitation with a TRU recovery efficiency of ~90%) is assumed to be implemented to treat cladding waste, unless future sample analysis demonstrates that it is non-TRU waste. Location of the processing step (PUREX Plant or B Plant) is not specified, and associated impacts (installation cost, incremental operating costs) are treated as the result of a near-term waste management operation decision, independently justified, and are thus outside the scope of this EIS. In other words, the implementation of additional TRU removal is a "given" that is common to all alternatives, not as a future waste processing step, but only as it controls the TRU content of the cladding waste.

A.2.2.2 Organic Wash Waste

As part of the PUREX process, the organic solvent, which consists of TBP dissolved in NPH, is washed to remove organic degradation products that would interfere with the process. The wash solution contains sodium carbonate and potassium permanganate. Following this wash, the organic solvent is reconditioned by treatment with dilute nitric acid. Trace concentrations of metal ions are also removed from the organic solvent by this treatment. Periodically, the aqueous wash solution becomes depleted in either carbonate or permanganate and is replaced. The depleted aqueous wash solution becomes the organic wash waste. It is combined with cladding waste for storage in double-shell tanks.

A.2.2.3 Miscellaneous Wastes

Additional wastes result from the PUREX process as ammonia scrubber wastes, miscellaneous sump waste, and low-level wastes from the later portions of the process. The scrubber and sump wastes are combined with the supernatant liquid of the neutralized cladding waste, and the low-level wastes are combined with other low-level wastes from operations elsewhere in the 200 Areas.

Plutonium Finishing Plant waste is generated during conversion of plutonium nitrate or oxide to metal, and includes associated laboratory wastes.

Customer waste consists of Hanford waste generated at other than the 200 Areas: principally, N Reactor in 100-N Area, Fast Flux Test Facility (FFTF) and laboratories in the 400 Area, and laboratories in the 300 Area. This waste is classified as low-level waste when generated.

The low-level waste from N Reactor fuel processing (wastes generated in the 200 Areas), Plutonium Finishing Plant waste, and customer waste all contain low concentrations of both chemical and radioactive components. Because of the high volume of these wastes, they are concentrated currently by about a factor of ten in the waste tanks. When concentrated, some of the wastes may be classified as TRU wastes since the TRU concentration may exceed 100 nCi/g.

A.2.3 Inventories in Future Waste Tanks

Quantities of chemicals and radioactive materials in future tank waste are listed in Tables A.8 and A.9, respectively. Neither volume nor water content is listed, since tank farm operations will appreciably concentrate the wastes. The volumes as generated will be 12,000 m^3 of HLW, 79,000 m^3 of cladding waste, 64,000 m^3 of miscellaneous 200-Area waste, 47,000 m^3 of Plutonium Finishing Plant waste, and 57,000 m^3 of customer waste for a total of 260,000 m^3. However, the available volume of double-shell tanks will be only 118,000 m^3 (or 114,000 m^3, assuming space equal to one empty tank in case of leakage). The volume of existing waste in double-shell tanks will be 45,000 m^3, so only 69,000 m^3 of space is available for future tank waste. Thus, 190,000 m^3 of water must be removed to place the projected waste into the available tanks. Additional concentration to 52,000 m^3 is planned to provide additional spare tank space. Less concentration may be needed if disposal actions make additional tank space available.

A.3 STRONTIUM AND CESIUM CAPSULES

Most of the high-heat-generating fission products, ^{90}Sr and ^{137}Cs, are contained in seal-welded, high-integrity double-shell metal capsules (Figure A.5) as strontium fluoride and cesium chloride, respectively. These are stored in shielded water basins (Figure A.6) for dissipation of decay heat and the reduction of exposure to operating personnel. Storage of the strontium and cesium capsules will be continued in the existing water basins until a disposal or other long-term storage alternative is selected. This method of storage provides multiple containment by double-shell metal capsules, water basins, a reinforced concrete building, and a ventilation system containing multiple HEPA filters. Some capsules have already been shipped off site for beneficial use as heat or radiation sources. Since the capsules are only leased, it is anticipated that they will be returned to Hanford for disposal after ^{137}Cs has decayed to levels below beneficial-use specifications.

The capsules are stored under 4 m of demineralized water in stainless steel-lined concrete basins with 0.76 m concrete shielding covers. The basin water is circulated through heat exchangers for cooling and is simultaneously monitored for detection of radioactivity in the event of a failed capsule. Should a capsule fail, it can be returned to a process cell, re-encapsulated, and replaced in the water basin. If contaminated, the basin water can be

TABLE A.8. Estimated Mass of Chemical Components Associated with Future Tank Radioactive Waste Through 1995, t

Chemical	High-Level Waste[a]	Cladding Removal Waste[b]	Plutonium Finishing Plant Waste	Other Waste	Total
Al	190		57		250
Ca			6	1	7
Cr	6				6
Fe	42		2		44
K		700			700
Mg			6	1	6
Mn	1	1		13	15
Na	1,200	1,700	280	750	3,900
U	12	12		110	140
Zr	6	820			830
CO_3		54		120	180
F	22	1,200	9		1,200
NO_3	800	73	480	560	1,900
NO_x	800	32		65	900
OH	740	1,100	190	110	2,100
PO_4				340	340
SO_4	78			120	200
Cancrinite[c]	17				17
Organic Carbon				5	5

(a) HLW from processing 12,000 t of N Reactor fuel, 7 t of Fast Flux Test Facility fuel, and 16.5 t of Shippingport fuel.
(b) Cladding waste plus organic wash waste, ammonia scrubber waste, and miscellaneous sump wastes from processing 12,000 t of N Reactor fuel and 16.5 t of Shippingport fuel.
(c) Known silica additions are assumed to have reacted with aluminates and hydroxides to form cancrinite (assumed to be $2NaAlSiO_4 \cdot 0.52NaNO_3 \cdot 0.68H_2O$).

processed to remove the radionuclides and returned to service. The capsule contents are soluble in water--cesium chloride more so than strontium fluoride. The basin atmosphere is routed to HEPA filters before venting to a stack, and all facility structures are designed to withstand maximum local seismic and natural events.

The estimated number of existing and projected strontium and cesium capsules, their curie content and heat release are summarized in Table A.10.

TABLE A.9. Estimated Major Radionuclide Inventories of Future Tank Waste Accumulated Through 1995, Ci

Radionuclide	High-Level Waste[(a)]	Cladding Removal Waste[(b)]	Plutonium Finishing Plant Waste	Other Waste	Total
^{241}Am	3.0×10^5	3.8×10^3	1.8×10^4	7.8×10^2	3.3×10^5
^{14}C	1.9×10^2	9.0×10^1			2.8×10^2
^{144}Ce(c)	1.0×10^8	1.0×10^5			1.0×10^8
^{137}Cs(c)	4.9×10^7	1.3×10^6			5.1×10^7
^{3}H	7.1×10^3				7.1×10^3
^{129}I	1.2×10^1				1.2×10^1
^{239,240}Pu	3.3×10^3	3.0×10^2	2.3×10^3	1.9×10^2	6.3×10^3
^{106}Ru(c)	1.3×10^7	3.2×10^5		7.6×10^4	1.3×10^7
^{151}Sm	3.9×10^5	3.9×10^2			3.9×10^5
^{90}Sr(c)	4.2×10^7	4.1×10^4			4.2×10^7
^{99}Tc	4.7×10^3	1.4×10^2			4.8×10^3
^{238}U	4.0	4.0		3.8×10^1	4.7×10^1

(a) HLW from processing 12,000 t of N Reactor fuel, 7 t of Fast Flux Test Facility fuel, and 16.5 t of Shippingport blankets.

(b) Cladding waste plus organic wash wastes, ammonia scrubber waste, and miscellaneous sump wastes from processing 12,000 t of N Reactor fuel and 16.5 t of Shippingport fuel.

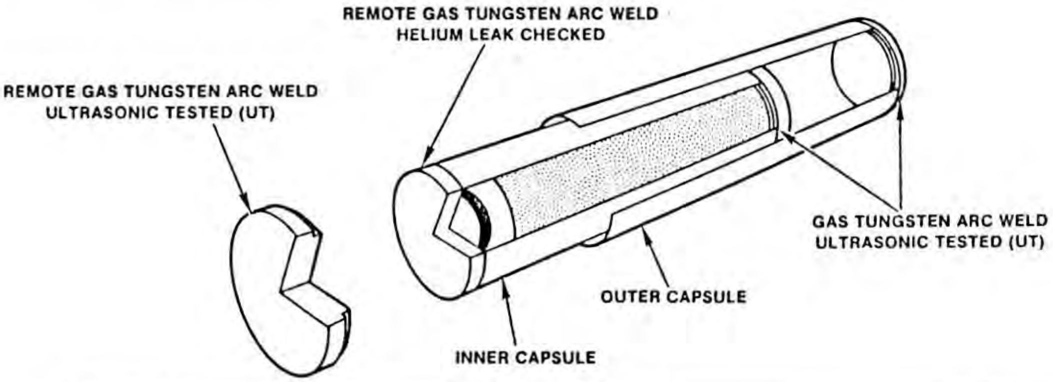
(c) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Short-lived daughters are accounted for in dose calculations.

A.4 TRU-CONTAMINATED SOIL SITES

Formerly used systems for discharging TRU-contaminated solutions to Hanford soils include:

- Cribs - buried structures (often wood or concrete) filled with aggregate that held or dispersed liquids and/or solutions for percolation into the ground.
- Ponds - surface depressions bordered by natural or manmade features used to contain and detain liquid waste.
- Trenches - open, usually long, narrow excavations used to deposit limited quantities of liquid waste.
- Ditches - open, unlined, long, narrow excavations used to transport and/or detain liquid waste.
- French Drains - large-diameter pipes buried vertically, normally less than 14 m deep, filled with rocks to allow the percolation of small, intermittent flows of liquid waste into the soil.

	FORM	LOADING	PER CENT OF THEORETICAL DENSITY BASED ON TOTAL VOID SPACE OF CAPSULE	TEMPERATURE			
				AIR		WATER	
				CENTER LINE	SURFACE	CENTER LINE	SURFACE
STRONTIUM FLUORIDE	COMPACTED POWDER	150 KCI (MAX)	68	860° C	430° C	660° C	71° C
CESIUM CHLORIDE	MELT-CAST	70 KCI	65	450° C	200° C	327° C	58° C



	CAPSULE									
	INNER					OUTER				
	MATERIAL	WALL THICKNESS	OUTSIDE DIAMETER	TOTAL LENGTH	TOTAL CAP THICKNESS	MATERIAL	WALL THICKNESS	OUTSIDE DIAMETER	TOTAL LENGTH	TOTAL CAP THICKNESS
STRONTIUM FLUORIDE	HASTELLOY C-276 (UT)	0.305 (UT)	5.72	48.39	1.02	STAINLESS STEEL 316-L (UT)	0.277 (UT)	6.67	51.05	1.02
CESIUM CHLORIDE	STAINLESS STEEL 316-L (UT)	0.241 (UT)	5.72	50.10	1.02	STAINLESS STEEL 316-L (UT)	0.277 (UT)	6.67	52.77	1.02
NOTE: ALL DIMENSIONS ARE IN CM										

FIGURE A.5. Strontium and Cesium Capsules

- Reverse Wells - well casings extending deep into the ground.
- Settling Tanks - single-shelled underground tanks or sumps made of concrete into which waste solutions were pumped. Solids settled in the tanks and liquids overflowed to a reverse well or other underground structure.
- Unplanned Releases - Releases of TRU-contaminated solutions to Hanford soils outside of engineered systems (caused by pipeline failure, overflows, etc.).

Movement of radionuclides into and through the soil varies considerably and depends on the soil chemistry of the elements involved (Murthy et al. 1983). Chemicals could be expected to behave similarly. Most anionic species, such as nitrates, move freely and change concentrations only by dilution as they move from the discharge point into the water table some distance below. The same is true of tritium, an isotope of hydrogen and a ternary fission product. Other elements, such as ruthenium, are retarded only slightly compared to the movement of the liquid phase. These components are useful in estimating the flow of the liquid through the soil column by measuring their concentrations in samples of water collected from the soil column and from the water table under the soil column. After the liquid reaches the water table, these component concentrations can be used to monitor the dispersion and movement of the water away from the soil column. Other elements move much more slowly through the column. In the absence of acids and complexing agents, the bulk of

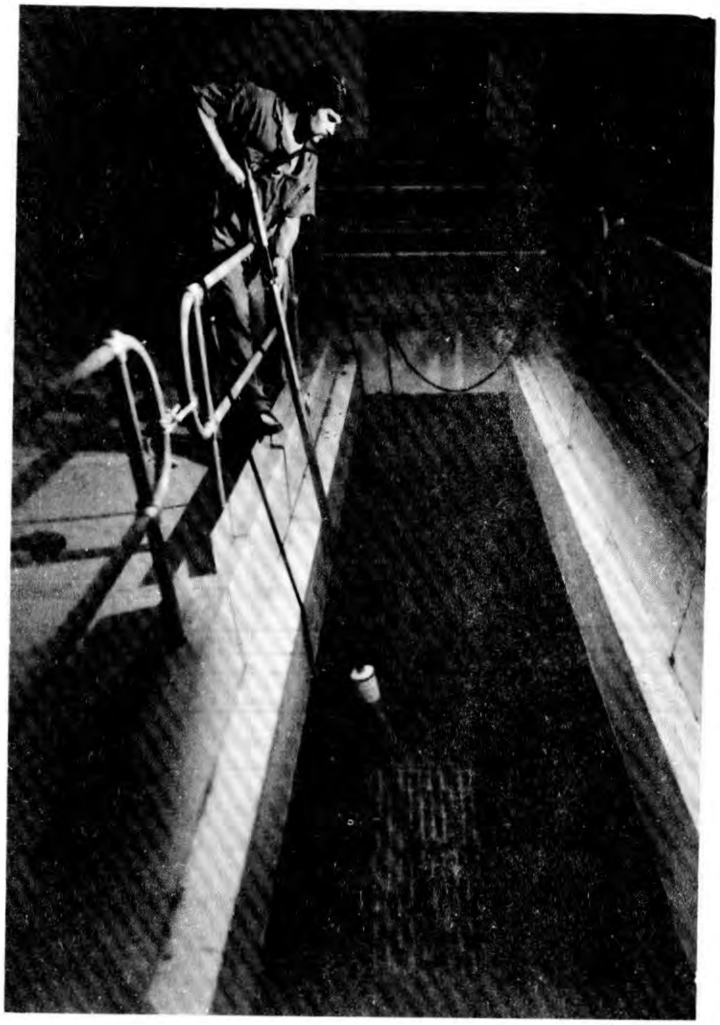


FIGURE A.6. Capsule Storage Basin

TABLE A.10. Characteristics of Strontium and Cesium Capsules(a)

Characteristics	Encapsulated before December 1983						Encapsulated After December 1983(b)		
	Strontium (447 capsules)			Cesium (1579 capsules)			Strontium (153 capsules)		
	1984	1995	2010	1984	1995	2010	1984	1995	2010
Cumulative MCi(a)	23	18	13	70	53	38	18	13	10
Cumulative kW	160	120	86	340	250	180	120	86	65
Average kCi/capsule(a)	52	39	28	44	33	25	110	86	63
Average W/capsule	350	270	190	210	160	120	780	560	430

(a) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. (Daughters are accounted for in modeling radiological impacts.)

(b) Stored as aqueous solution in B Plant as of December 1983.

TRU elements and compounds move through the soil only slightly and are held near (within a few meters) the point of discharge. Use of each TRU disposal site was discontinued before any radionuclide penetrated to the water table at a concentration exceeding the then-applicable concentration limits.

TRU-contaminated soil sites at Hanford that are estimated to contain TRU nuclides at concentrations greater than 100 nCi/g are listed in Table A.11. Relevant site characteristics are also included in the table. Twenty-four sites containing a total of approximately 190 kg of plutonium in 32,000 m^3 of contaminated soil have been identified as TRU-contaminated soil sites at Hanford.

TRU-contaminated soil sites with the maximum inventories are shown in Table A.12. Characterization data for the maximum TRU site show small volumes containing TRU concentrations up to 40,000 nCi/g. Such concentrations are projected to occur within the first 0.3-m depth from the distribution structure, dropping to <1000 nCi/g within 2 m and to <100 nCi/g within 15 m.

A.5 PRE-1970 BURIED SUSPECT TRU-CONTAMINATED SOLID WASTE(a)

Between 1944 and 1970, TRU-contaminated waste (soiled clothing, laboratory supplies, tools, etc., packed in cardboard, wood or metal containers), as distinguished from "industrial" waste (primarily items of large failed equipment, packaged in metal or concrete boxes), was buried in dry waste trenches (ERDA 1975). The concept of a burial trench is illustrated in Figure A.7.

A burial ground is defined as a pre-1970 buried suspect TRU-contaminated solid waste burial ground if the concentration of some containers there is estimated to exceed 100 nCi TRU/g (based on a soil density of 1.8 g/cm^3 and a peak-to-average concentration of 10:1). On this basis, 11 TRU burial sites have been identified on the Hanford Site, as listed in Table A.13. These sites involve 74,000 m^2 of surface area and 110,000 m^3 of TRU wastes and contaminated soil. Some of these sites contain TRU waste in caissons similar to the one shown in Figure A.8. Most of this waste is located in the 200 Areas. Two sites (618-1 and 618-2) are in or near the 300 Area, and the other (618-11) is located in an area near the WYE barricade (300-Y). Caissons were also used at the latter site. A recently completed study (DOE 1986), which examined records of inactive waste disposal locations on the Hanford Site, showed that the two 618 sites (618-1 and 618-2) each contained one gram of plutonium, rather than the previously listed 1,000 grams (Rockwell 1985). As a results of this lower quantity, both sites are now designated as low-level waste (Rockwell 1987).

Data for sites with the maximum inventories of TRU (218-W-2) and fission products (218-E-12B) are listed in Table A.14. Maximum TRU concentration in caissons (218-W-4B) is given in Table A.15.

(a) Also referred to as pre-1970 TRU solid waste.

TABLE A.11. TRU-Contaminated Soil Sites

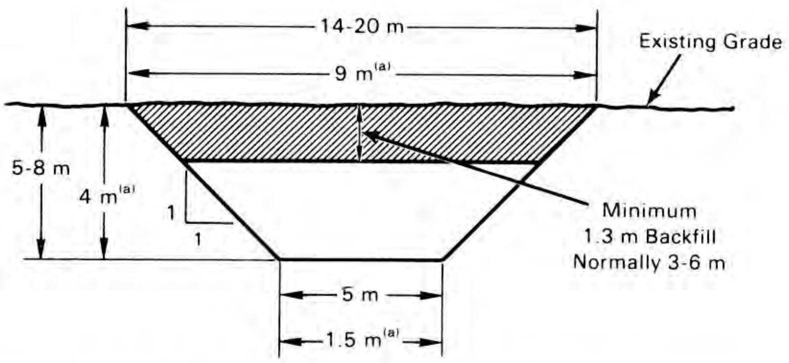
Site	Type	Total Plutonium, g	Area Over Waste Zone, m^2	Overburden Volume, m^3	Contaminated Soil Volume, m^3	Average TRU Concentration, nCi/g
216-T-3	Reverse well	1.3×10^3	1.1	1.8×10^3	2.9	2.5×10^4
216-Z-10	Reverse well	5.0×10^1	3.1×10^{-1}	1.3×10^3	1.7×10^{-1}	1.5×10^4
216-B-5	Reverse well	1.7×10^3	1.4	2.2×10^3	6.4×10^1	1.4×10^3
241-B-361	Settling tank	2.6×10^3	7.3	2.5×10^2	1.8×10^2	7.7×10^2
241-T-361	Settling tank	2.0×10^3	7.3	2.5×10^2	1.8×10^2	6.0×10^2
216-B-7A+B	Crib	4.3×10^3	3.1×10^2	1.3×10^3	4.3×10^2	5.3×10^2
216-Z-9	Trench	3.8×10^4	5.3×10^2	6.1×10^3	5.1×10^3	4.0×10^2
216-Z-8	French drain	4.8×10^1	2.0	2.5×10^2	5.8	4.5×10^2
216-Z-1+2TF	Crib	6.4×10^4	1.9×10^3	2.5×10^3	8.3×10^3	4.1×10^2
216-T-32	Crib	3.2×10^3	3.2×10^2	4.6×10^3	4.6×10^2	3.7×10^2
216-Z-12	Crib	2.5×10^4	1.9×10^3	1.1×10^4	5.4×10^3	2.5×10^2
216-E-15	Unplanned release	1.2×10^3	5.4×10^2	1.8×10^3	2.6×10^2	2.4×10^2
216-B-53A	Trench	1.0×10^2	7.0×10^1	4.3×10^2	2.4×10^1	2.2×10^2
216-Z-18	Crib	2.3×10^4	2.3×10^3	1.7×10^4	5.7×10^3	2.2×10^2
216-Z-3	Crib	5.7×10^3	5.5×10^2	1.7×10^3	1.5×10^3	2.0×10^2
216-Z-7	Crib	2.0×10^3	7.8×10^2	1.7×10^4	5.9×10^2	1.8×10^2
216-T-18	Crib	1.8×10^3	3.4×10^2	7.6×10^2	5.9×10^2	1.6×10^2
216-Z-11	Ditch	8.1×10^3	3.3×10^2	5.4×10^1	5.5×10^2	7.9×10^2
216-Z-5	Crib	3.4×10^2	2.1×10^2	2.0×10^3	2.1×10^2	8.6×10^1
216-T-6	Crib	3.9×10^2	2.8×10^2	2.6×10^3	2.9×10^2	7.2×10^1
216-S-1+2	Crib	1.2×10^3	7.0×10^2	8.7×10^3	1.7×10^3	3.8×10^1
216-Z-1	Ditch	1.4×10^2	1.5×10^2	4.8×10^1	3.8×10^1	2.0×10^2
216-Z-19	Ditch	1.4×10^2	3.0×10^2	0.0	7.3×10^1	1.0×10^2
216-U-10	Pond	2.2×10^1	5.9×10^2	0.0	1.9×10^2	6.1
Totals		1.9×10^5	1.2×10^4	8.3×10^4	3.2×10^4	

TABLE A.12. Radionuclide Inventory Data for TRU-Contaminated Soil Sites, Ci(a)

Radionuclide	Total Inventory	216-Z-1+2TF Maximum TRU Site	216-S-1+2 Maximum Fission Product Site
^{241}Am	3.8×10^{3}	1.3×10^{3}	2.5×10^{1}
^{60}Co	3.4×10^{-1}	2.0×10^{-2}	4.5×10^{-2}
^{137}Cs(b)	1.8×10^{3}	8.9×10^{-1}	9.6×10^{2}
^{3}H	1.0		
^{238}Pu	4.6×10^{2}	1.6×10^{2}	3.0
^{239}Pu	1.1×10^{4}	3.7×10^{3}	6.8×10^{1}
^{240}Pu	2.6×10^{3}	9.1×10^{2}	1.7×10^{1}
^{241}Pu	7.4×10^{3}	2.5×10^{3}	4.7×10^{1}
^{242}Pu	1.5×10^{-1}	5.3×10^{-2}	9.9×10^{-4}
^{106}Ru(b)		1.4×10^{-6}	1.7×10^{-9}
^{90}Sr(b)	3.4×10^{3}	8.8×10^{-1}	1.1×10^{3}
^{233}U	2.5	3.6×10^{-2}	1.0
^{234}U	2.6	3.7×10^{-2}	1.0
^{235}U	7.5×10^{-2}	1.1×10^{-3}	3.1×10^{-2}
^{238}U	1.9	2.7×10^{-2}	7.6×10^{-1}

(a) As of December 31, 1995.

(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Short-lived daughters are accounted for in dose calculations.



(a) Dimensions for Typical "Dry Waste" Trench; Cardboard Boxes, Barrels, etc. (Larger Dimensions are for Contaminated "Industrial" Solid Waste Trench; Failed Process Equipment in Large Metal or Concrete Boxes).

FIGURE A.7. Typical Solid Waste Burial Trenches

TABLE A.13. Pre-1970 Buried Suspect TRU-Contaminated Solid Waste Sites

Site	Area, m^2	Overburden Volume, m^3	Waste Volume, m^3	Pu, g	Transuranic Elements Ci	Transuranic Elements nCi/g
200 West Area						
218-W-1	6,500	7,400	9,000	94,000	9,000	560
218-W-2	15,000	24,000	23,000	130,000	12,000	290
218-W-3	16,000	55,000	25,000	68,000	6,500	140
218-W-4B	5,500	23,000	6,800	9,900	950	78
218-W-4A	20,000	80,000	25,000	35,000	3,400	76
200 East Area						
218-E-1	2,500	8,200	3,000	900	86	16
218-E-5A	960	1,300	2,200	1,400	130	34
218-E-12B	3,400	7,700	4,400	1,200	110	14
Outside the 200 Areas						
618-1(a)	390	620	470	1,000	96	110
618-2(a)	400	1,100	710	1,000	96	75
618-11	3,100	18,000	7,900	10,000	960	68
Totals	74,000	230,000	110,000	350,000	33,000	

(a) A recently completed study (DOE 1986), which examined records of inactive waste disposal locations on the Hanford Site, showed that two 618 sites (618-1 and 618-2) each contained 1.0 g of plutonium, rather than the previously listed 1,000 g (Rockwell 1985). As a result of this lower quantity, both sites are now designated as low-level waste. (Rockwell 1987).

A.6 RETRIEVABLY STORED AND NEWLY GENERATED TRU WASTE

TRU waste generated since 1970 has been retrievably stored. Most of this waste is in 55-gal drums, stored as shown in Figure A.9. The containers are covered with plywood, plastic-reinforced nylon sheeting, and a 1.3-m layer of uncontaminated soil to reduce surface radiation exposure rates. TRU waste unsuitable for asphalt pad or caisson storage because of size, chemical composition, security requirements, or surface radiation has been packaged in reinforced wood, concrete, or metal boxes, and stored in dry waste trenches. The trench is covered with plywood and a vinyl plastic and backfilled with dirt. Typical trench construction is similar to that for pre-1970 burial (see Figure A.7). If the surface dose rate exceeds 200 mrem/hr, the waste is classified as remote-handled (RH) and is either stored in caissons (Figure A.8) similar to those used for pre-1970 buried TRU solid waste (Section A.5) or packaged with sufficient shielding to meet requirements for contact handling.

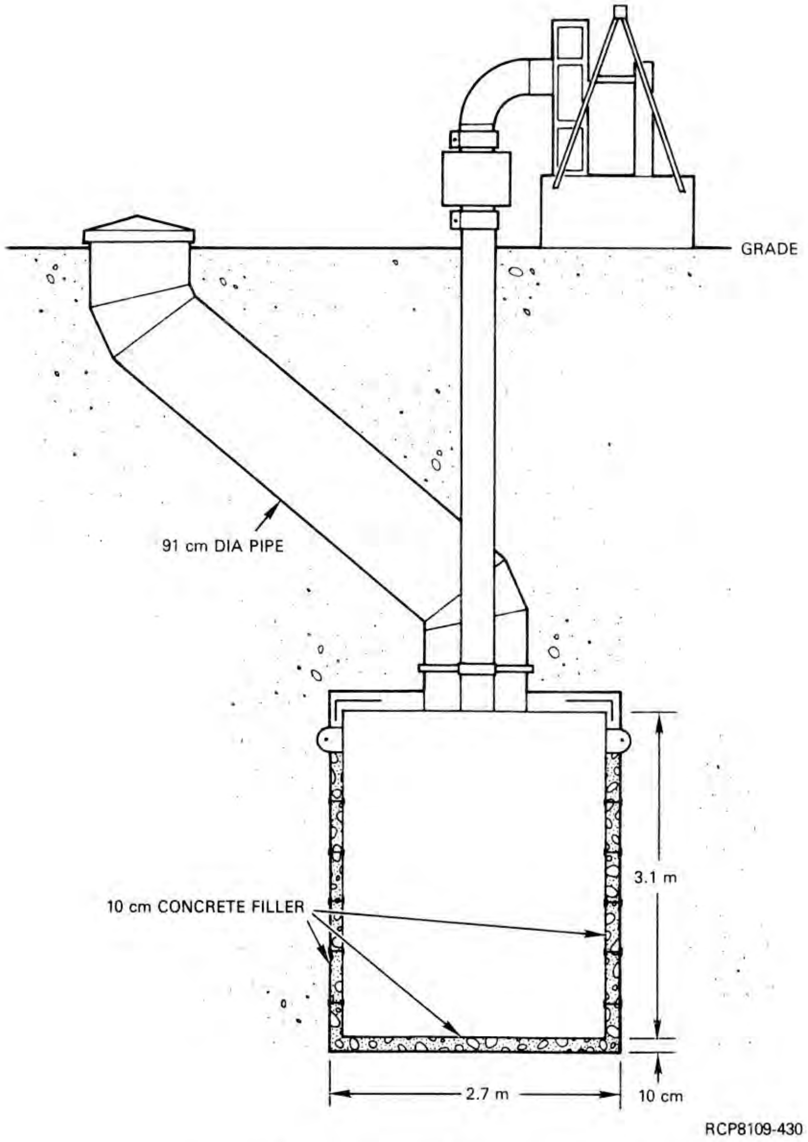


FIGURE A.8. Typical Caisson for TRU Storage

The current inventory of retrievably stored solid TRU waste is summarized in Table A.16. The estimated future inventory (through 1996) of newly generated TRU waste is given in Table A.17. The highest concentrations of radioactive materials are contained in the 218-W-4B alpha caissons.

TABLE A.14. Radionuclide Inventory for Pre-1970 TRU Burial Sites, Ci(a)

Radionuclide	Total Inventory	218-W-2 Maximum TRU	218-E-12B Maximum Fission Products
^{241}Am	7.1×10^{3}	2.5×10^{3}	2.4×10^{1}
^{60}Co	1.2×10^{4}		1.2×10^{4}
^{137}Cs(b)	2.1×10^{4}	4.2	1.9×10^{4}
^{3}H	5.3×10^{4}		
^{238}Pu	8.6×10^{2}	3.1×10^{2}	2.9
^{239}Pu	2.0×10^{4}	7.2×10^{3}	6.6×10^{1}
^{240}Pu	4.9×10^{3}	1.8×10^{3}	1.6×10^{1}
^{241}Pu	1.4×10^{4}	5.0×10^{3}	4.6×10^{1}
^{242}Pu	2.9×10^{-1}	1.0×10^{-1}	9.5×10^{-4}
^{106}Ru(b)	1.8×10^{-2}	1.8×10^{-11}	1.7×10^{-2}
^{90}Sr(b)	2.0×10^{4}	3.6	1.9×10^{4}
^{233}U	6.5×10^{1}	6.2×10^{-1}	1.3×10^{-2}
^{234}U	6.7×10^{1}	6.4×10^{-1}	1.3×10^{-2}
^{235}U	2.0	1.9×10^{-2}	3.9×10^{-4}
^{238}U	4.9×10^{1}	4.7×10^{-1}	9.5×10^{-3}

(a) As of December 31, 1995.

(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Short-lived daughters are accounted for in dose calculations.

TABLE A.15. Maximum TRU Concentration in a Pre-1970 Buried TRU Solid Waste Site (Caissons 218-W-4B)(a)

Radionuclide	Site Inventory, Ci	Average Concentration, nCi/g
^{241}Am	3.4×10^{1}	3.7×10^{2}
^{60}Co	9.6×10^{1}	1.1×10^{3}
^{137}Cs(b)	1.3×10^{3}	1.5×10^{4}
^{238}Pu	4.1	4.5×10^{1}
^{239}Pu	9.5×10^{1}	1.1×10^{3}
^{240}Pu	2.3×10^{1}	2.6×10^{2}
^{241}Pu	6.6×10^{1}	7.3×10^{2}
^{242}Pu	1.4×10^{-3}	1.5×10^{-2}
^{106}Ru(b)	2.2×10^{-3}	2.5×10^{-2}
^{90}Sr(b)	1.5×10^{3}	1.6×10^{4}
^{232}Th	7.4×10^{-3}	8.2×10^{-2}
^{233}U	1.3×10^{-1}	1.5
^{234}U	1.4×10^{-1}	1.5
^{235}U	4.2×10^{-3}	4.6×10^{-2}
^{238}U	1.0×10^{-1}	1.1
Total U, g	3.0×10^{5}	
Total Pu, g	1.7×10^{3}	
Volume, m^3	50	

(a) Site 218-W-4B also contains caissons with retrievably stored TRU waste (Section A.6).

(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Short-lived daughters are accounted for in dose calculations.

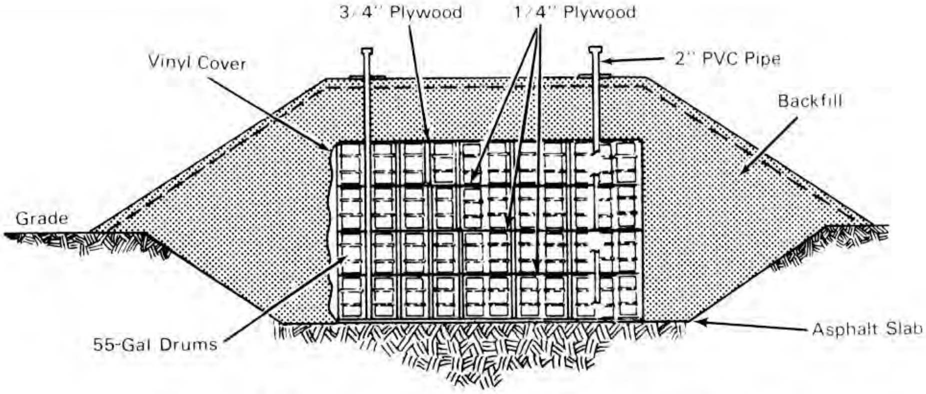


FIGURE A.9. TRU Asphalt Pad Storage (ERDA 1975)

TABLE A.16. **Inventory of Radionuclides in Retrievably Stored TRU Waste** (1970-1983), Ci

Radionuclide	Total Inventory(a)	Caissons 218-W-4B	Maximum TRU 218-W-4C	Maximum Fission Products 218-W-3A
^{241}Am	1×10^3	1×10^2	1×10^2	3×10^1
^{60}Co	9×10^2	3×10^2		4×10^2
^{137}Cs(b)	2×10^4	1×10^3	9×10^3	1×10^4
^{238}Pu	3×10^4	2×10^1	3×10^4	2×10^1
^{239}Pu	2×10^4	3×10^2	2×10^4	1×10^3
^{240}Pu	5×10^3	8×10^1	4×10^3	3×10^2
^{241}Pu	6×10^4	9×10^2	5×10^4	3×10^3
^{242}Pu	2	3×10^{-2}	1	1×10^{-1}
^{106}Ru(b)	7	7×10^{-1}	5	1
^{90}Sr(b)	2×10^4	1×10^3	8×10^3	1×10^4
^{233}U	6	2×10^{-1}	8×10^{-1}	4
^{234}U	6	3×10^{-2}	8×10^{-1}	4
^{235}U	2×10^{-1}	9×10^{-4}	2×10^{-2}	1×10^{-1}
^{238}U	4	2×10^{-2}	6×10^{-1}	3
Volume, m^3	1.3×10^4	2.2×10^1	4.9×10^3	3.9×10^3
Area, m^2	2.5×10^4		6.7×10^3	1.0×10^4

(a) Includes three sites not itemized by radionuclide in this table:

218-W-4B trenches	3,200 m^3 volume, 5,700 m^2 area
218-E-12B	640 m^3 volume, 2,100 m^2 area
212-N	220 m^3 volume.

(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Short-lived daughters are accounted for in dose calculations.

TABLE A.17. **Projected Radionuclide Inventory for Newly Generated TRU Waste, 1984 to 1996, Ci**

Radionuclide	Total Inventory(a)	Contact Handled	Remote Handled(b)	Fuel Hulls	"618" Sites(a)
^{241}Am	5×10^{3}	3×10^{3}	6×10^{1}	2×10^{3}	3×10^{2}
^{14}C	2			2	
^{60}Co	3×10^{3}		6×10^{2}	2×10^{3}	2×10^{2}
^{137}Cs(c)	5×10^{4}	3×10^{4}	5×10^{2}	2×10^{4}	2×10^{3}
^{3}H	7			7	
^{238}Pu	9×10^{2}	6×10^{2}	1×10^{1}	3×10^{2}	3×10^{1}
^{239}Pu	2×10^{4}	2×10^{4}	7×10^{1}	2×10^{3}	7×10^{2}
^{240}Pu	5×10^{3}	4×10^{3}	3×10^{1}	1×10^{3}	2×10^{2}
^{241}Pu	2×10^{5}	1×10^{5}	2×10^{3}	7×10^{4}	5×10^{2}
^{242}Pu	6×10^{-7}	3×10^{-7}	1×10^{-8}	3×10^{-7}	1×10^{-2}
^{90}Sr(c)	4×10^{4}	3×10^{4}	5×10^{2}	2×10^{4}	1×10^{3}
Volume, m^{3}	1.2×10^{4}	1.1×10^{4}	7.3	1.9×10^{2}	1.5×10^{3}

(a) Waste from the "618" sites (see Section A.5 and Table A.13) will be classed as newly generated TRU waste for the reference alternative only. This waste is not included in the total. There will be 300 m^{3} of remote-handled waste and 1,200 m^{3} of contact-handled waste from the "618" sites (after sorting to remove low-level waste).

(b) Includes 3.9 m^{3} generated through 1990 (caisson waste) and 3.4 m^{3} generated after 1990 (packaged for disposal).

(c) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Short-lived daughters are accounted for in dose calculations.

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APPENDIX B

DESCRIPTION OF FACILITIES AND PROCESSES

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APPENDIX B

DESCRIPTION OF FACILITIES AND PROCESSES

Some new construction would be required for all of the alternatives described in Section 3.3. Some large facilities or construction actions are described in this appendix. Processes for each alternative are also described, together with projected releases to the environment during processing, and characteristics of final waste forms. The engineering data in this appendix were obtained primarily from engineering support data provided by Rockwell Hanford Operations (1985 and 1987); data without explicit references are generally from this source. The numerical information is the best available data. The existing waste inventories are based on historical records and are believed to be adequate for the generic waste class descriptions. The projected future waste inventories, as well as the estimated release data for all waste types, are provided. Radioactive decay is calculated to December 31, 1995.

Three facilities for which appreciable detail is available are described in separate appendices: Appendix C, Hanford Waste Vitrification Plant; Appendix D, Transportable Grout Facility; and Appendix E, Waste Receiving and Processing Facility.

B.1 NEW FACILITIES AND CONSTRUCTION

New facilities would be required with all disposal alternatives for retrieval of wastes and for chemical or mechanical processing of wastes. Construction would be required for site stabilization (subsidence control) and isolation (barriers and markers).

B.1.1 Retrieval of Wastes

Wastes would be retrieved from all sites in the geologic disposal alternative. Special facilities would be required for retrieval from single-shell and double-shell tanks, from TRU-contaminated soil sites, and from pre-1970 solid TRU waste burial grounds. Retrieval from double-shell tanks would be required for the other alternatives as well. No special facilities are anticipated to be required for retrieval of strontium and cesium capsules or contact-handled retrievably stored and newly generated TRU solid waste.

B.1.1.1 Retrieval of Single-Shell Tank Wastes

Methods for removal of the contents of single-shell tanks must be carefully selected because the integrity of some of the tanks is suspect. Addition of liquids for removal of solid waste (sluicing) would increase the risk that some of the tank contents could leak to the surrounding soil. As of May 1982, 26 tanks among the 149 single-shell tanks at Hanford were designated as confirmed leakers (Murthy et al. 1983).

In response to comments received on the draft EIS, additional and previous methods for retrieval of single-shell tank waste were reviewed and three techniques for retrieval of single-shell tank wastes were briefly evaluated (Rockwell 1987). Other techniques, such as hydraulic cavitation, are planned to be evaluated in further studies. All methods considered are scheduled to be reviewed by the National Academy of Sciences (NAS). The three techniques evaluated are hydraulic sluicing, modified hydraulic sluicing and dry mechanical retrieval.

Hydraulic sluicing of tanks has been successfully performed both at the Hanford Site and at the Savannah River Plant. Hanford used a high-pressure sluicer and a multistage pump to transfer material through pipelines to vaults and on to B Plant for the recovery of strontium from high-level waste sludges. Savannah River Operations have used in-tank mixers and multistage turbine pumps to clean out their single-shell tanks and transfer material to their double-shell tanks. This is the most likely process Hanford would use if hydraulic sluicing were selected. Equipment includes a hydraulic sluice nozzle, multistage pumps, deep-well turbine pumps, and direct-buried and encased shielded piping to transfer the resulting slurry to nearby transfer vaults. From these vaults, the slurry would be transferred to a double-shell tank or directly to the processing facility for separation in the high-activity and low-activity fractions for further processing. Water sluicing can remove up to 99.95% of the residual wastes. However, less efficient radionuclide removal would be assumed due to past leaks and anticipated leaks during recovery processing. The risk is that dilute liquid could leak out of tanks during this process; however, analysis of a postulated 40,000-m^3 leak of liquid waste showed no significant environmental consequences (Murthy et al. 1983).

Modified hydraulic sluicing consists of a high-pressure sluicer coupled to a high-pressure vacuum head. A very similar technique has been developed to support decontamination and decommissioning efforts, and several commercial models are available. All would require some modification to work in a single-shell tank. Dust control (over dry retrieval) and the potential for leakage (over regular sluicing techniques) would be greatly reduced.

The third technique is dry mechanical retrieval. This technique reduces the potential for any additional leakage from the tanks due to retrieval operations. Figures B.1 through B.3 depict a concept of the equipment that would be used. The mechanical retrieval process is composed of three sequential operations: 1) in-tank recovery of waste; 2) removal of waste to a transfer point for emplacement in shipping containers; and 3) transfer of waste to an onsite processing facility. The system would be capable of retrieving all types of salt cake and sludge without the direct addition of liquids. Only minimal alterations to the tank farm structures, specifically, the tank dome, would be necessary. The system would be designed to avoid direct loads on the dome.

Waste tanks would be prepared by adding entry points (risers), if necessary, and removing above-ground obstructions to the retrieval platform. Where necessary, in-tank obstructions would be removed and waste would be prepared for excavation by breaking up large encrustations.

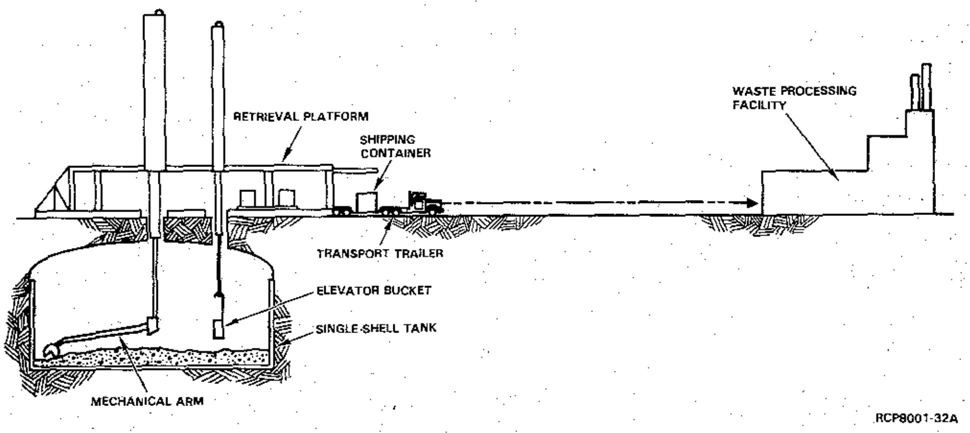

FIGURE B.1. Mechanical Retrieval of Wastes from Single-Shell Tanks

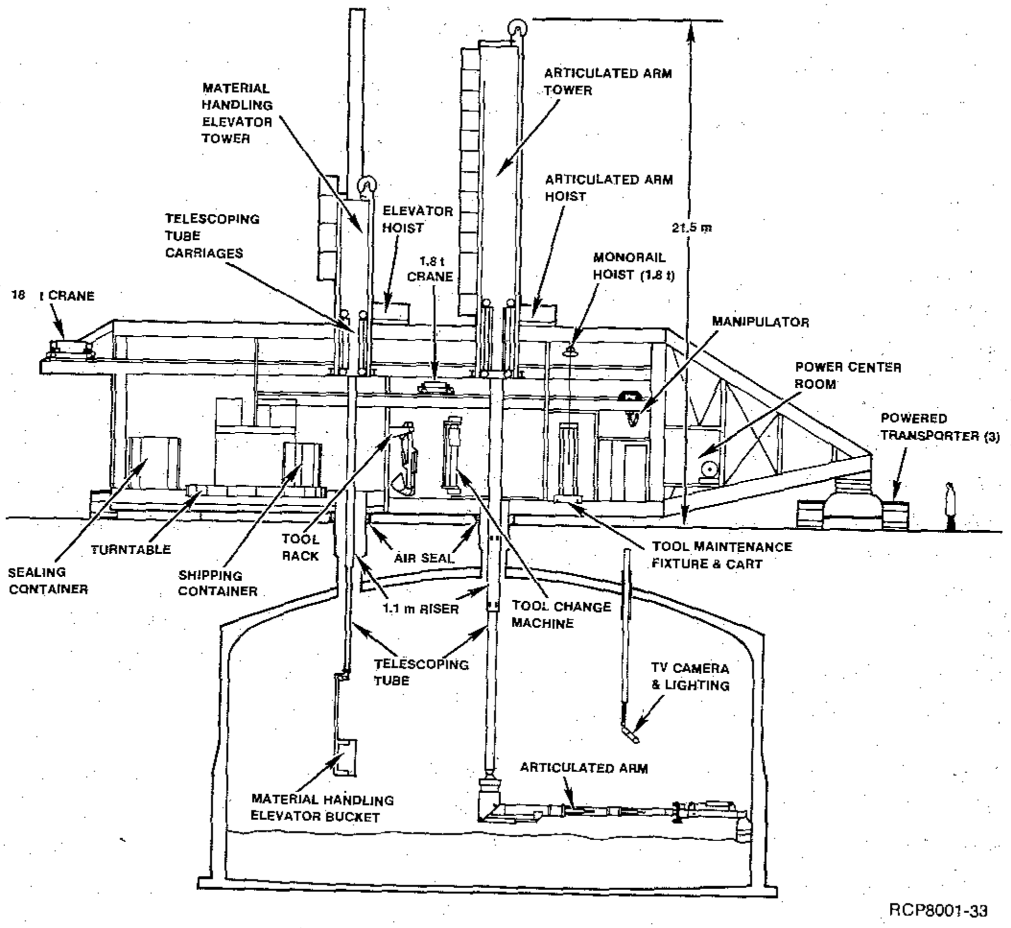

FIGURE B.2. Side View of Mechanical Waste Retrieval System

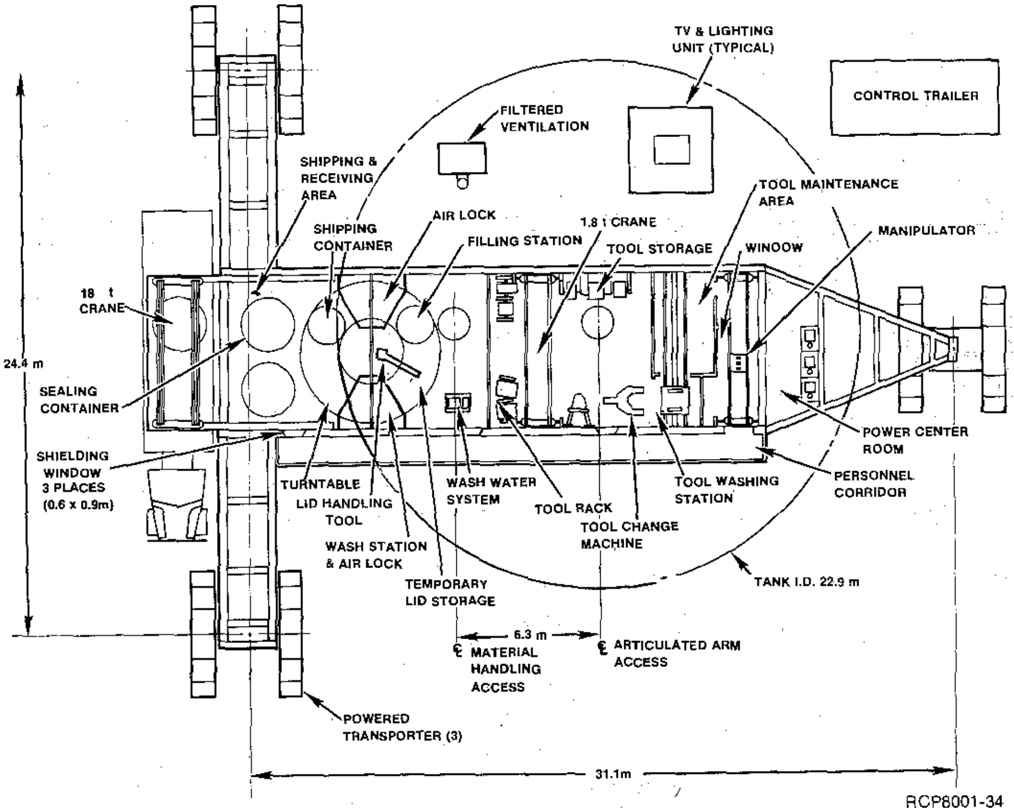

FIGURE B.3. Top View of Mechanical Waste Retrieval System

Support equipment and equipment necessary for waste recovery, removal, and containerization would be mounted on a platform that could be moved from tank to tank (Figures B.1 through B.3). The mobile platform, sized to the approximate 31-m tank spacing, would support the waste-handling apparatus and contain auxiliary systems necessary for safe retrieval of waste. A hydraulically actuated, articulated arm would be positioned by a tower-controlled telescoping tube and carriage mechanism.

The waste retrieval system would recover the waste mechanically using a clamshell bucket on the articulated arm and would deposit waste in an elevator bucket for transfer to the platform level. This recovered "as-is" waste would be unloaded from the elevator bucket to a shielded shipping container that would hold ~2.7 m^3 of waste. After being sealed and washed, the shipping container would be placed in a clean container that would also be sealed with a locking lid. A special tractor-trailer would transfer the containers over a dedicated roadway to an onsite waste processing facility where the waste would be combined with other waste materials and processed for conversion to a form suitable for geologic disposal. Additional safety in transit would be obtained by placing the outer container inside an enclosure of thick-walled shock-absorbing material secured to the trailer bed.

The waste tank would be maintained slightly below atmospheric pressure. Ventilation air would be discharged through two high-efficiency particulate air (HEPA) filters to maintain effluent radionuclide concentrations such that resultant doses would be less than those permitted by EPA standard 40 CFR 61: 0.025 rem/yr to the total body or 0.075 rem/yr to any organ of a member of the public, or other applicable standards.

The mechanical retrieval technique is estimated to remove at least 95% of the waste. The residual radioactivity could be removed by enhanced mechanical means (e.g., a revolving wire brush mounted on the articulated arm). Sluicing would also be possible if visual inspection and the presence of drainable liquid indicated that the tank was sound.

As it presents the upper range of radiological risks as a result of recovery operations, mechanical retrieval was selected as the bounding case evaluated in this EIS. The total cost of recovery of the most expensive technique is about 6% of the total processing cost; hence, the cost of the process chosen would not affect the overall decisions. These and additional options would be considered in more detail if the preferred alternative is chosen for implementation.

B.1.1.2 Hydraulic Sluicing from Double-Shell Tanks

Although hydraulic sluicing has not been assumed for the recovery of waste from single-shell tanks because of the suspect status of the tank liners (see Section B.1.1.1), the method would be readily acceptable for double-shell tanks. In these tanks, if a leak in the inner liner (primary tank) occurred, it would be detected in the annulus between the inner and outer liners, and the outer liner would prevent leakage into the soil under the tank.

The equipment and techniques for recovering and transferring radioactive liquids are well established and have been used at both Hanford and Savannah River. Existing methods involving multistage pumps, deep-well turbine pumps, and encased, shielded piping would be used to transfer liquid waste to nearby transfer vaults. From there the liquid would be transferred to other tanks or to processing facilities via encased, heat-traced, underground pipelines. A schematic of this process is shown in Figure B.4.

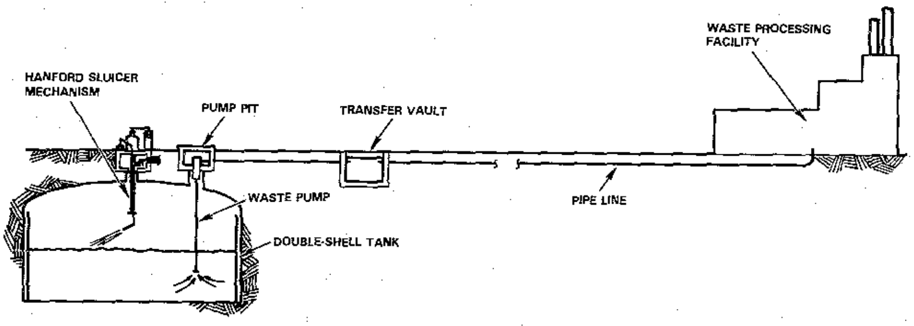

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FIGURE B.4. Hydraulic Retrieval of Waste from Double-Shell Tanks

Sluicers, which have been used in past years for cleanup of sludge heels from the bottoms of single-shell tanks, could be used to resuspend double-shell tank wastes. A sluicer is composed of two basic systems: 1) a high-pressure water supply system comprising a remote piping connector, vertical pipe, flex hose, rotary joint, and nozzle assembly and 2) a nozzle-aiming mechanism consisting of two concentric control rods in a guide tube, nozzle assembly turning arm, and gear rod for turning the rotary joint. Figure B.5 depicts one concept of hydraulic retrieval installation utilizing the sluicer assembly. The sluicers could use externally supplied liquid, recirculated in-tank liquid, or a combination of these. Floating pumps would be used to transfer the resuspended slurry. Transfer of the slurry to a waste processing facility or to another tank would be accomplished in the same manner as described above for liquid. Another hydraulic method utilizing rotating mixer pumps as demonstrated at Savannah River could be used to resuspend double-shell tank wastes. This system is composed of a low-pressure, recirculating pump which uses solution within the tank as the mixing medium. The intake is located at the bottom of the pump perpendicular to the waste and discharges through two opposing nozzles parallel to the waste surface while the entire pump rotates. Floating pumps would be used to transfer the resuspended slurry in the same manner as the sluicing operation.

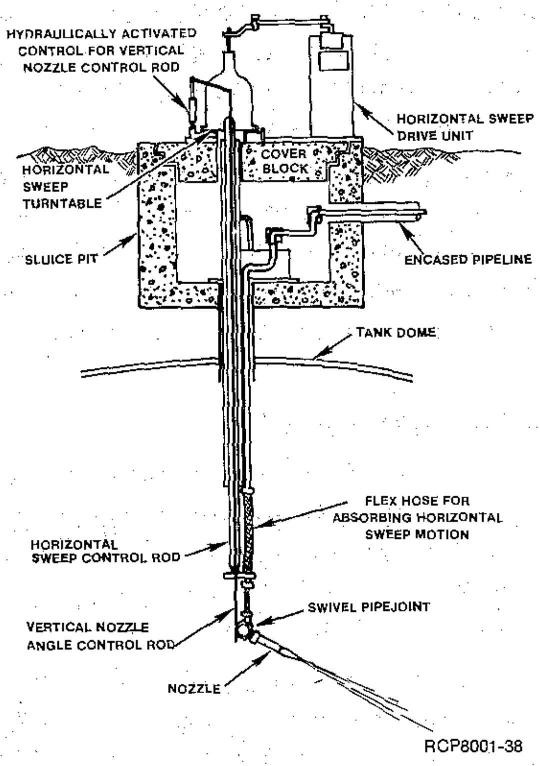

FIGURE B.5. **Hanford Sluicer Installation**

The tank would be maintained slightly below atmospheric pressure. Ventilation air would be discharged through two HEPA filters to maintain effluent concentrations less than maximum permissible concentrations for discharge to uncontrolled areas. A similar process has been implemented in past tank farm operations. This system should remove at least 99.95% of the waste in double-shell tanks.

B.1.1.3 Mechanical Retrieval of TRU-Contaminated Soil and Solid Waste Sites

The same facility would be used to retrieve waste from most contaminated soil sites and from most solid waste sites. Exceptions are described in Sections B.1.1.4 and B.1.1.5. To minimize the spread of contamination, the retrieval process would be carried out in a containment facility (Figure B.6), a double-shelled, prefabricated building designed to operate at negative air pressure. The approximate building dimensions would be: 46 m width, 92 m length, 12.5 m height. The building could be raised and moved about on wheels for ease of assembly at waste recovery sites. Auxiliary buildings would house airlocks, utilities equipment, standby generators, air compressors, power distributors, operations control centers, personnel support facilities, ventilation and filtering systems, and decontamination processes. These auxiliary facilities would be moveable when detached from the double-shelled recovery building.

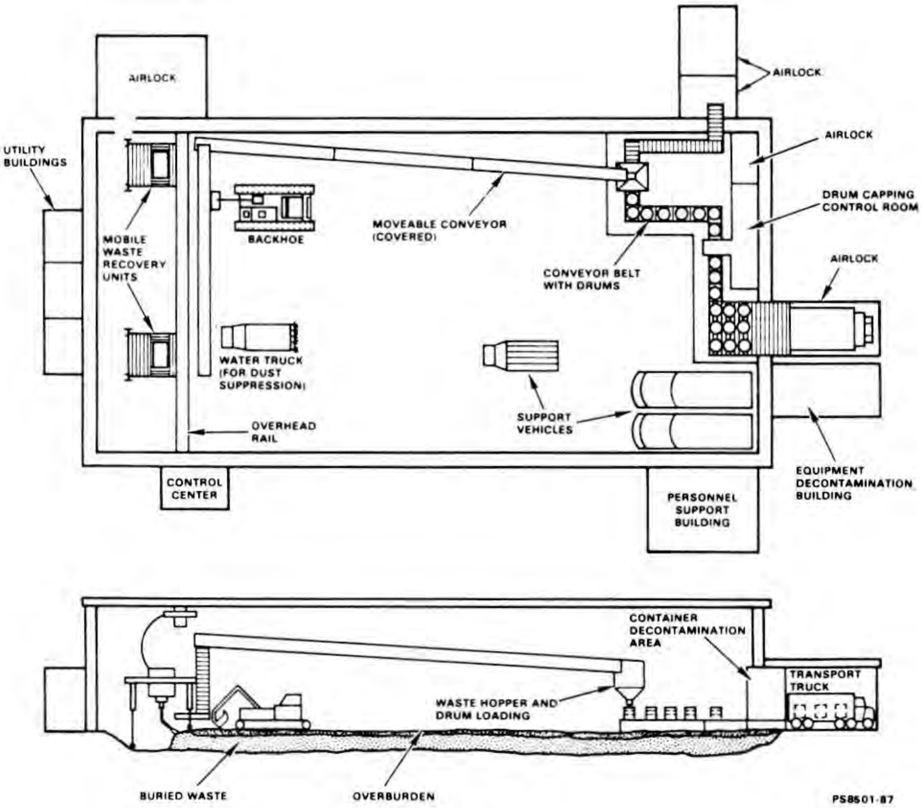

FIGURE B.6. Mechanical Retrieval of Wastes from Soil or Solid Waste Sites

Electrically operated equipment would be used to retrieve contaminated soil and solid wastes. Some equipment would be battery operated, while other equipment would be connected by cables to the building power supply. Recovery equipment would include waste retrievers, container haulers, waste sizers, dust precipitators, and heavy-duty backhoes. Recovery building vehicles would also be available for personnel transport, maintenance support, and firefighting.

The prefabricated recovery building and its auxiliaries would be erected to cover the first recovery area. The recovery building, all support equipment, and all recovery equipment would be thoroughly tested before excavation of an entry pit and radioactive waste. Dust within the pit would be controlled by spraying the working face of the pit with a dust suppressant and by operating dust precipitators. Mobile waste retrievers would excavate first the entry pit and then the contaminated area of the site within the facility. The retrievers would be equipped for both digging and lifting waste material, allowing them to excavate soil as well as to lift waste such as contaminated timbers or piping out of the excavation area. The retrievers would be operated from sealed, environmentally controlled cabs. The cabs would have built-in equipment for monitoring radiation and other health hazards. Uncontaminated overburden removed from the entry pit and the excavated area would be placed in a designated storage area for use in backfilling. The retrievers would place contaminated soil and waste into waste containers.

The filled containers would be moved to an area within the recovery building for analysis and capping. There the containers would be checked for radiation levels and TRU content, capped and then moved to a decontamination area. After decontamination the containers would be moved through an airlock into a special transportation trailer. Waste requiring repository disposal would be transported to a waste processing facility. Non-TRU waste would be disposed of by burial in near-surface low-level waste (LLW) disposal trenches. The trailer, which could hold four containers, would be towed by a highway tractor. The full containers would be exchanged for empty ones, and the trailer would then be returned to the recovery facility.

If a waste item were too large to be handled by the retriever, a sizing machine would reduce the item if possible. A track-mounted vehicle would hold the sizing machine, operated from an environmentally controlled cab. Built-in equipment in the cab would monitor radiation and other health hazards. Articulated manipulator arms equipped with attachments for sawing, shearing, hammering, and bending materials would be controlled from the cab. If the material could not be reduced to a size manageable by the retriever, a heavy-duty backhoe would be used to secure the oversized waste for special handling. Oversized items would be transported to a waste processing facility by a special tractor-trailer accompanied by escort vehicles.

The mechanical waste retrieval building would be maintained slightly below atmospheric pressure. Ventilation air would be discharged through two HEPA filters to maintain effluent radionuclide concentrations such that resultant doses would be less than those permitted by EPA standard 40 CFR 61: 0.025 rem/yr to the total body or 0.075 rem/yr to any organ of a member of the public.

As the recovery operation reached the far end of the recovery building, operations would cease and preparations to move the building would be made. The working face of the excavation would be covered with clean soil. A portion of the excavation near the working face would be backfilled to grade in order to establish a surface on which to seal the building. After the building had been moved, the excavation outside the building would be backfilled.

Utilities and auxiliary buildings would be disconnected to move the recovery building within a site. The recovery building would then be raised onto wheels and moved to its new position, where it would then again be lowered, assembled, and its systems tested. The cycle of building assembly, testing, waste recovery, and building movement would continue until waste recovery at the site was completed.

Movement of the recovery building to a new site would require thorough decontamination of the facility, disassembly, transport, and reassembly at the new site.

The recovery building would be decontaminated and decommissioned after completion of all waste recovery. Any TRU waste from decontamination would be packaged and sent to the waste processing center, and LLW would be buried in trenches. It is assumed that decommissioning would require 20% of the effort used for assembly of the recovery facility and equipment.

B.1.1.4 Mechanical Retrieval from Caissons

Retrieval operations from caissons would include site preparation, waste retrieval, waste packaging, cask decontamination, cask transport, and site stabilization. The caisson waste would be retrieved using the same large, moveable building used for retrieval of TRU-contaminated soil sites and pre-1970 TRU burial trenches (or a smaller but similar building if the operations described in Section B.1.1.3 were not performed).

Special equipment would be used to recover the waste in caissons. This equipment would not require an entry pit to gain access to the caissons. Movement of the building, however, would still require roadways. The recovery building would be positioned over the first caisson row and would contain a special remotely operated manipulator and associated equipment also assembled over the first caisson. A new entry cut would be made into the caisson. The retrieval operations would be controlled remotely from an auxiliary control room. A grappler housing equipped with a telescoping articulated boom would retrieve the caisson waste stored in 1-gal and 5-gal paint cans (Figures B.7 and B.8). An airlock and conveyor system would be used to transfer the remotely handled cask containing the retrieved caisson waste. This cask would be remotely sealed and decontaminated before placement on a truck. The cask would then be transported to a waste processing facility for conversion to a chemically inert, physically stable form for geologic disposal.

B.1.1.5 Mechanical Retrieval from Reverse Wells

Reverse wells differ from other contaminated soil sites in the depth of excavation necessary to reach the contaminated soil. A 6-m-dia access shaft would be excavated and retrieval would use remotely operated equipment as required. The recovery building would be similar to the one described in Section B.1.1.3. Special access shaft refrigeration equipment, used for freezing the surrounding water table during excavation, would be required at site 216-B-5 where contaminated soils extend to the groundwater.

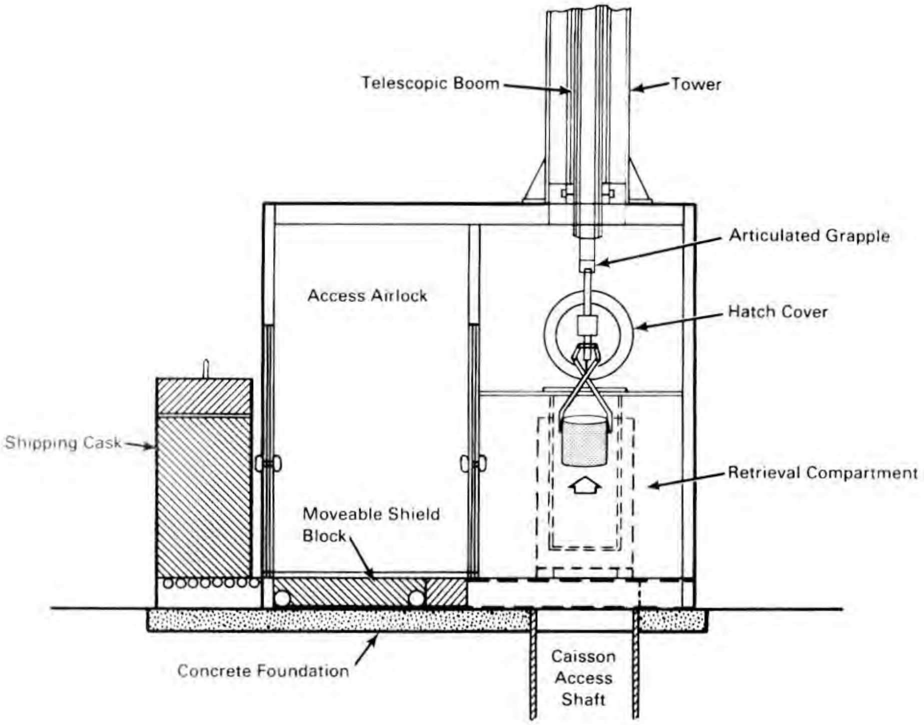

FIGURE B.7. Caisson Recovery Building, Side View

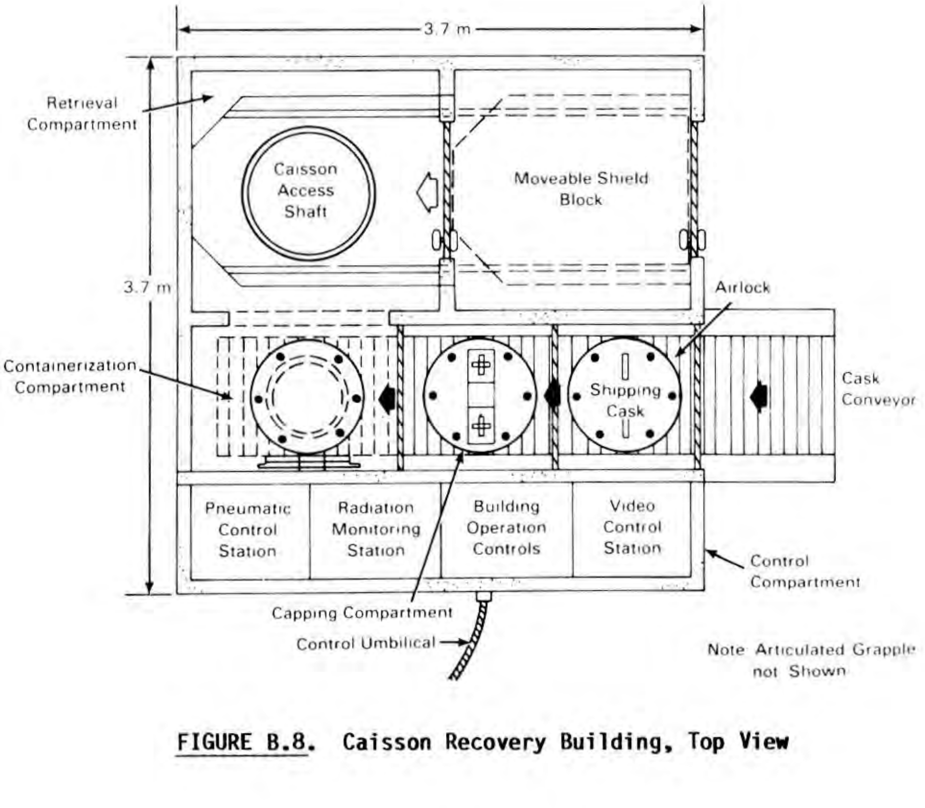

FIGURE B.8. Caisson Recovery Building, Top View

B.1.2 Chemical Processing

Retrieved wastes must be converted to a chemical form suitable for the proposed disposal mode. Some chemical separations would also be necessary to reduce the volume of high-level or TRU waste requiring permanent isolation from the environment.

B.1.2.1 Radionuclide Concentration for Geologic Disposal

Concentration of radionuclides for geologic disposal in the reference and preferred alternatives would involve relatively small volumes of waste and fundamental operations, and would be performed in existing facilities. In the geologic disposal alternative, a new facility would be required to process the large volume of existing tank waste currently in single-shell tanks.

The basic objective of radionuclide concentration would be to treat dissolved salt wastes for removal of most of the radionuclides with half-lives greater than 10 years (strontium, cesium, TRU elements, and technetium). The primary goal would be separation of these high-level wastes into two fractions: 1) a small volume of concentrated radioactive waste to be immobilized for long-term storage or disposal, and 2) a large volume of low-level waste that could be safely disposed of by relatively inexpensive means. If these feed processes were chosen as part of the Hanford defense waste disposal plan, the necessary facilities would have to be provided to process all existing and future tank waste. Feed preparation processes are summarized schematically in Figure B.9.

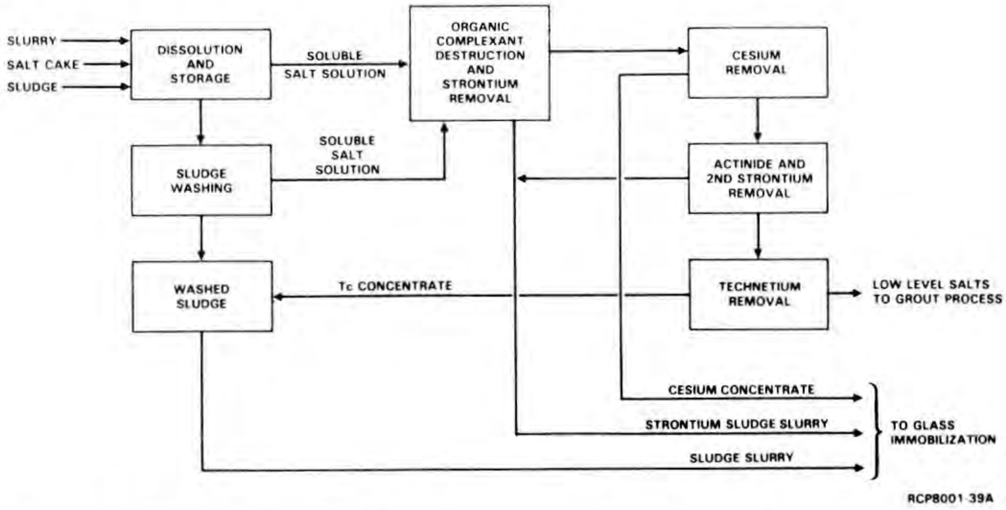

FIGURE B.9. Schematic Flowsheet for Radionuclide Concentration Processes

In the geologic disposal alternative, in which all waste from both single-shell and double-shell tanks would be processed, a new processing facility would be needed. Figure B.10 is a conceptual drawing for such a facility. The radionuclide concentration portion of the waste processing facility would consist of a canyon-type facility, a sludge-washing

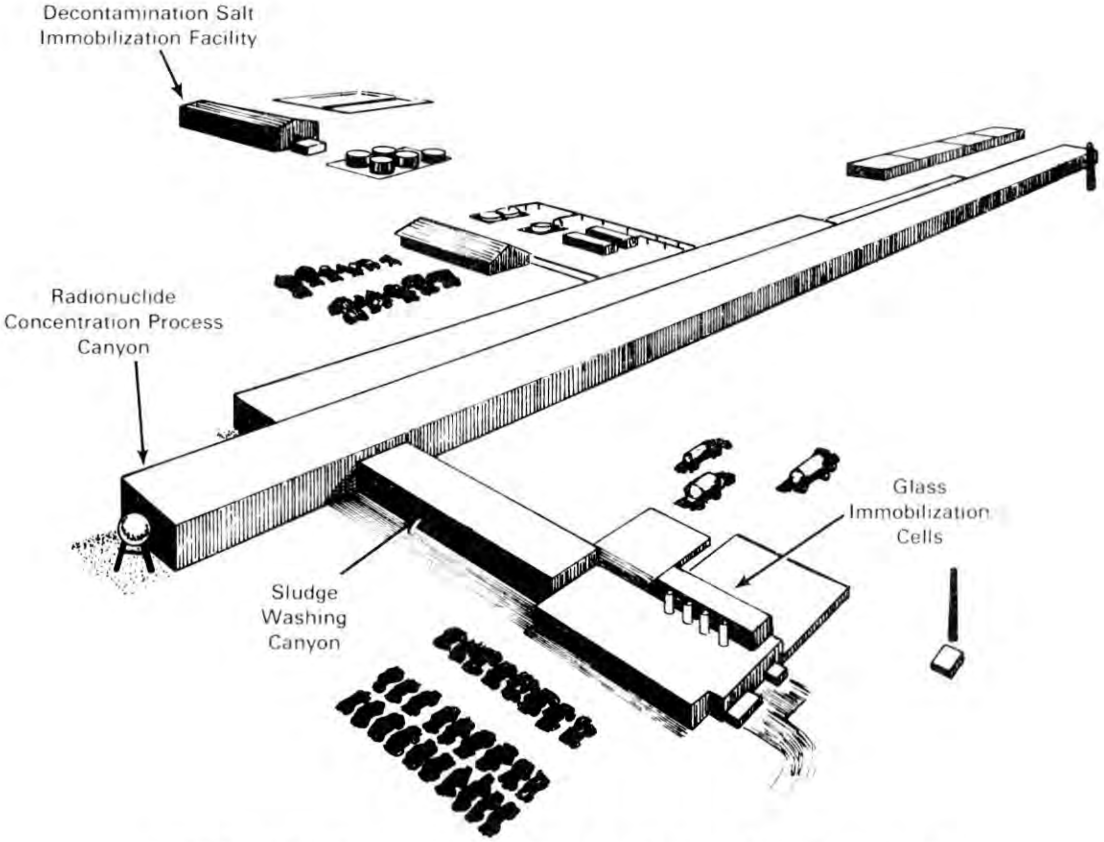

FIGURE B.10. Conceptual Facility for Radionuclide Concentration

canyon connecting the concentration process canyon to glass immobilization cells, and an attached support services facility. Figure B.11 is a cross section of the conceptual main canyon building.

Four distinct types of waste would be received at the radionuclide concentration section of the waste processing facility: bulk sludge, bulk salt cake, double-shell slurry, and complexed concentrate. The salt cake would be dissolved in water and stored for later blending with other waste. The bulk sludge would alternately be mixed with water to ensure dissolution of soluble salts and then centrifuged to separate the insoluble components. The centrifuged product would contain most of the strontium and actinides, and the salt solution would contain the cesium, technetium, iodine, and traces of strontium and actinides. The double-shell slurry and complexed concentrate would be treated as necessary for destruction of organic complexants, and stored for later blending with other waste.

The objective of solids-liquids separation and sludge washing operations would be to minimize the quantity of glass produced from high-level waste (HLW) by removing soluble salts that otherwise would remain entrained in the high-level solids. A second objective of these operations would be to maintain the natural partitioning of TRU nuclides between solids and liquid waste streams, resulting in low concentrations of TRU elements in supernatant liquids and spent wash streams.

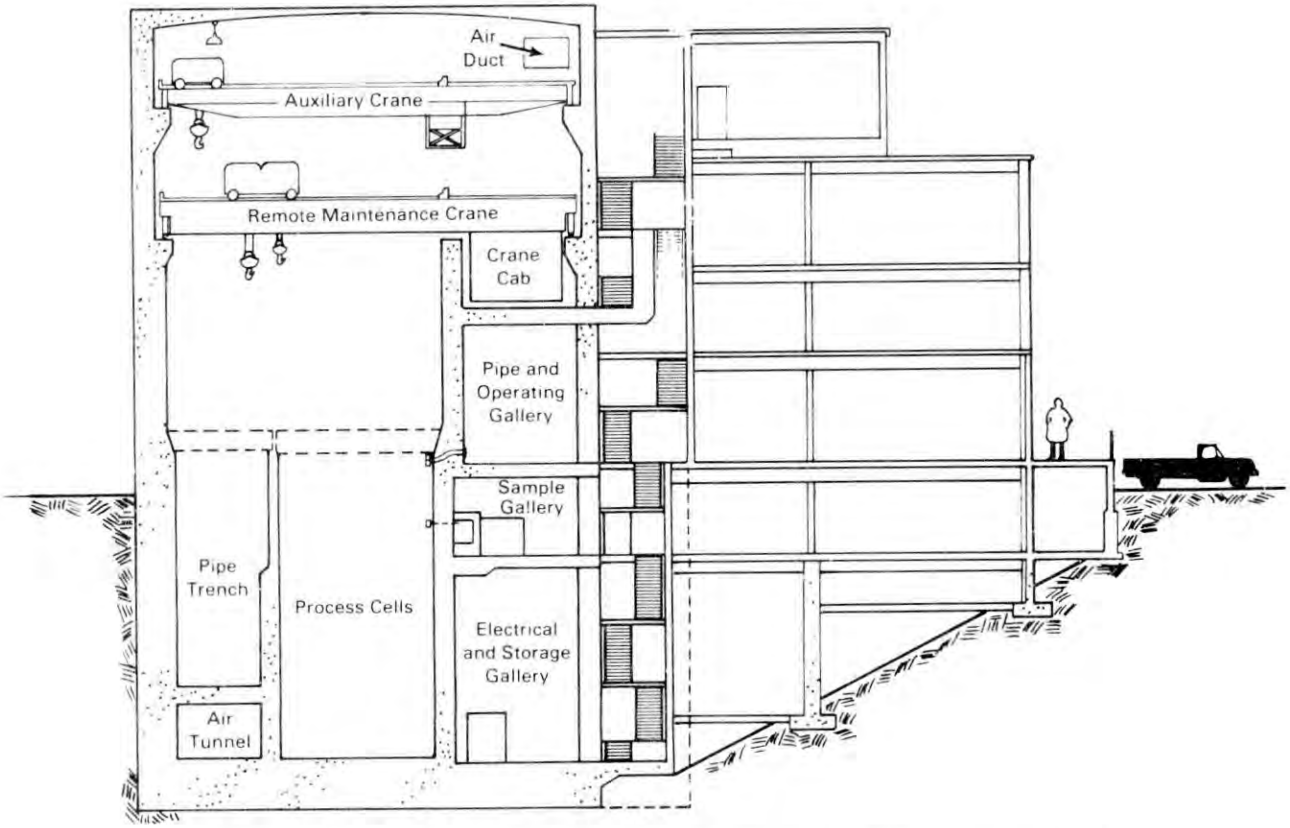

FIGURE B.11. Radionuclide Concentration Canyon Building, Cross Section

DOE continually seeks ways of improving the efficiency of its waste fractionization processes. As an example, recent technological developments have indicated that a promising solvent extraction technique is the transuranic extraction (TRUEX) process. The TRUEX process uses the bifunctional extractant octyl(phenyl)-N, N-diisobutylcarbamoylmethylphosphine oxide (CMPO). With this extractant, enough TRU elements may possibly be removed from the dissolved solids to make the residual waste suitable for grouting and shallow-land burial.

Applicability of alternative processes such as the TRUEX process will be based on results of technical feasibility and performance assessment analyses. Performance assessment is the analysis that evaluates the potential long-term isolation of a waste disposal system.

A process such as the TRUEX process could be used alone or with additional processes to provide a radionuclide disposition in the grout or glass that would cause minimal environmental impact. Use of the TRUEX process could allow large volumes of waste to be disposed of near the surface, because it substantially reduces the activity of the wastes. However, the predicted performance must be compared against accepted criteria and standards. Other processes, including sludge washing and radionuclide removal, will be considered. These processes could be used in conjunction with the TRUEX process if necessary, removing additional radionuclides so that environmental impacts can be minimized and applicable standards for near-surface disposal can be met.

Process off gases would be treated for removal of particulates, radionuclides, oxides of nitrogen (NO_x), and ammonia (NH_3) before release to the atmosphere. The nitric acid resulting from distillation of the technetium product would be recovered and reused. Ammonia and carbon dioxide would be recovered, recombined, and reused in elution of cesium from the ion exchange column. Off-gas treatment facilities would be designed to ensure that concentrations of NO_x and NH_3 in the gaseous effluent would meet limits for release to the atmosphere. Liquid effluents would be minimized by recycling. Cooling water would be reused after being passed through a suitable cooling process. Steam condensate would be returned to the process. Steam condensate and cooling water not recycled would be sampled and discharged to an evaporation pond if within release limits. Waste not meeting release limits would be concentrated and blended into the process.

The only routine effluent stream from solids-liquids separation and sludge washing operations would be an atmospheric release resulting from jet entrainment that passes through a vessel vent off-gas system.

The concentration of condensate occasionally released to surface ponds would be less than the maximum permissible concentration for release to uncontrolled areas for all radionuclides with the possible exception of tritiated water, which would be within the discharge limit for release to controlled areas.

Existing double-shell tank wastes would be treated to destroy organic complexants, as the presence of these complexants can retard or even prevent proper curing of the grout (Appendix D). Alternatives for complexant destruction include ozone treatment, peroxide oxidation, and pressurized aqueous combustion. Development of the technology to ensure an effective process is continuing. Wastes containing complexants would be pumped to a feed makeup tank where the pH and concentration would be adjusted. The feed solution would then be treated to destroy organic materials. The treated waste would be collected in a receiving tank where satisfactory destruction of organic materials would be confirmed. If organic destruction should be incomplete, the waste could be recycled. After satisfactory destruction of the organic complexants, the treated waste would be pumped to a double-shell tank where solids containing strontium and actinides would be allowed to settle. The non-TRU, complexant-free salt solution would be mixed with other LLW streams for disposal in grout.

The solids-liquids separation process would isolate a small volume of material containing most of the actinides, strontium, rare earths, and other fission products that are insoluble in alkaline solutions. This sludge fraction would be routed as feed to the immobilization process (Section B.1.2.2).

Strontium and actinides remaining in the salt solution would be removed by a combination of precipitation and adsorption onto a sodium titanate ($NaTi_2O_5H$) ion exchanger. Sand filters would remove trace solids from the supernatant liquid in the strontium precipitation processes. Spent sodium titanate would be added to the strontium precipitate and the slurry would be used as feed to the vitrification process. The cesium in the solution would then be removed using an ion exchange column loaded with Duolite ARC-359® (Diamond Shamrock Company) or a similar material. The cesium would be eluted with an ammonium carbonate solution which would then be steam-stripped to separate the eluent from the cesium product. The cesium solution would also be transferred to the immobilization process.

Technetium would be removed following cesium ion exchange by adsorption onto an anion resin. The technetium would be eluted with nitric acid and the product stream would be distilled to recover nitric acid. The concentrated product would be neutralized with sodium hydroxide and added to the sludge waste, which would be transferred to the immobilization process.

The waste stream leaving the technetium ion exchange column would be a sodium salt waste containing trace amounts of radionuclides. Ruthenium and iodine, like the sodium salts, are contaminants and would virtually be unaffected by the separation processes. Decay during long-term, underground tank storage before processing would reduce the concentration of radioruthenium by a factor of 1,000 for each decade of storage. The essentially decontaminated salt solution would be monitored and then transferred from collection tanks to a grout disposal process.

Four concentrated product streams would be produced in the waste preparation process: sludge slurry containing insoluble chemicals and radionuclides (strontium, actinides, and technetium); strontium sludge slurry containing strontium phosphate, sodium titanate, strontium, and actinides; a solution of cesium carbonate and sodium carbonate; and a solution of technetium. The concentrated products would be transferred to the vitrification facility.

B.1.2.2 Glass Immobilization for Geologic Disposal

Glass immobilization would be employed with the geologic disposal, reference, and preferred alternatives. For each alternative, glass immobilization would be in the Hanford Waste Vitrification Plant (Appendix C) or a similar facility.

The glass immobilization process discussed in this section would be used for geologic disposal in conjunction with the radionuclide concentration process. It is designed to be operated continuously at 72% operating efficiency for 18 years. The slurries from radionuclide concentration would be blended with a glass frit composed of silicon dioxide (SiO_2), boron oxide (B_2O_3), sodium oxide (Na_2O), and lithium oxide (Li_2O), and melted to a homogeneous glass in ceramic-lined melters that are heated internally by electrical conduction through the molten glass (joule heating). The molten glass stream poured from the melters would be cast directly into steel canisters 0.61 m in diameter by 3 m long. The product glass would contain about 25 wt% waste oxides.

The borosilicate glass product provides a waste form with properties of low dispersibility, low leachability, and relatively high thermal stability (Appendix C). The glass product could be expected to be moderately stressed and cracked.

Melter off gases, containing water, carbon dioxide (CO_2), nitrogen oxides (NO_x), sulfur oxides (SO_x), and traces of cesium, ruthenium, and other radionuclides, would be routed through a glass frit filter bed for removal and recycle of particulates and ^{137}Cs. Subsequent wet scrubber systems would remove the NO_x and SO_x, recycling nitrate and sulfate salts to the radionuclide concentration head-end process.

Radionuclide concentration process sludge and concentrate streams would be pumped as slurries from the main canyon facility (radionuclide concentration process) to the immobilization wing. The slurries would be blended with weighed quantities of glass-forming ingredients (SiO_2, B_2O_3, Na_2O, and Li_2O). A relatively small quantity of glass frit from the off-gas filter would be added to the feed batch as a recycle stream.

The melter concept under evaluation and development for Hanford waste immobilization is the slurry-fed, joule-heated, ceramic-lined continuous melter. This design offers the potential for long life, high processing rate, and high glass quality. To employ joule heating, the melter would be equipped with electrodes between which electrical energy would dissipate within the molten glass. Typical configuration of a continuous electric melter is shown in Figure B.12. The waste and glass additives would pass through three partially overlapping phases as they are incorporated into the glass pool: an evaporation phase in which the slurry is dried, a calcining phase in which dried wastes decompose to form oxides, and the molten glass phase. The relatively cool blanket of oxides and wet sludge (called a "cold cap") condenses most of the escaping volatile radionuclides and refluxes them to the molten pool. The resulting gaseous effluent would contain all of the water, NO_x, CO_2, and some of the SO_x in the melter feed, and (during infrequent periods of abnormal cold cap distribution) up to 5% of the cesium.

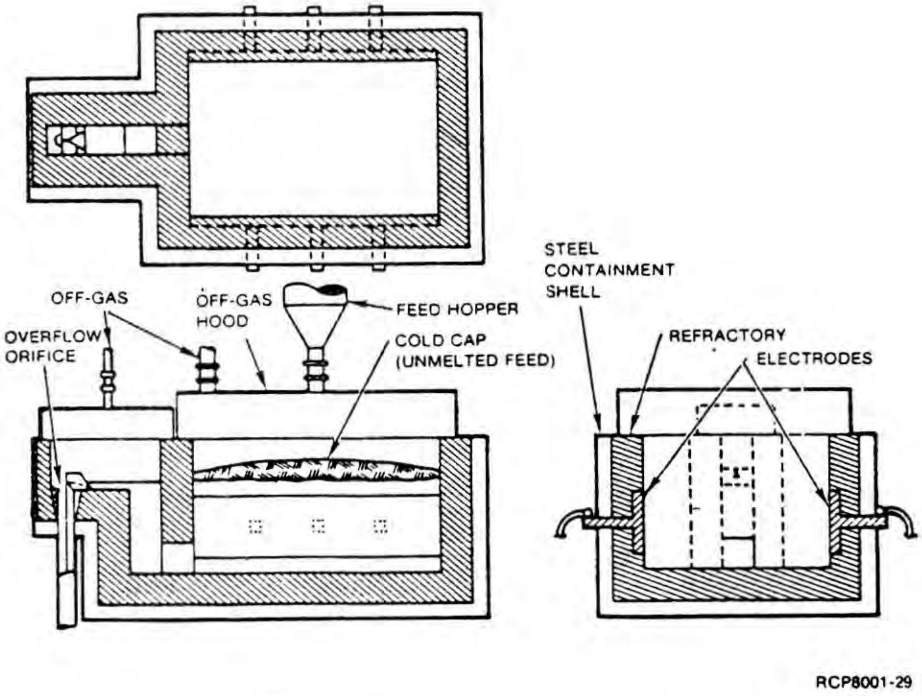

FIGURE B.12. Continuous Electric Glass Melter

Appendix C describes in detail the Hanford Waste Vitrification Plant. Present assumptions are for multiple melter units of this type, including spares as needed, to achieve an effective rate of 8 t per day. Glass would be allowed to pour from a melter to fill steel canisters in a casting operation. When a canister was full, the pour of molten glass would be stopped and the canister moved to another location for cooling. When cool, the canister would be moved to decontamination, welding, and nondestructive testing stations for final closure and inspection before transfer to the loadout facility.

The melter off-gas stream would be routed first through a rechargeable filter bed composed of a ground glass frit and maintained at an elevated temperature. Here dust particles would be filtered and some volatiles (cesium and ruthenium) condensed and trapped. When the bed was replaced, the trapped materials would be recycled to the feed blending system. Water vapor, NO_x, and SO_x would be finally removed via condenser and scrubber using a sodium carbonate (Na_2CO_3) solution. (The scrubber also would serve as a secondary decontamination step for volatilized cesium.) This contaminated solution of nitrate and sulfate salts would be recycled to the head-end of the radionuclide concentration process. The salts ultimately would leave the process in the decontaminated salt stream.

The glass immobilization facility would be constructed as a wing attached to the main (radionuclide concentration) canyon as shown in Figure B.10. Because of multiple mechanical operations and solids-handling steps in the process, a combination of in-cell cranes, manipulators, and viewing windows would be employed for remote maintenance and control. The glass conversion process would be conducted in three hot cells with shielding walls 1.1 m thick. The hot cells would provide a total cell floor area of 285 m^2.

A high bay or canyon would cover the entire cell complex and provide access by the 70-ton canyon crane to the cells below. An elevation view of the facility is shown in Figure B.11. Nonradioactive zone facilities would provide the normal auxiliary services of process control, chemical makeup, frit and canister storage, chemical storage, manipulator maintenance, service galleries, and utilities.

This glass immobilization technology has undergone considerable development at Hanford and Savannah River Plant and has been selected for use at both Savannah River (DOE 1982) and the West Valley Demonstration Project (DOE 1983).

B.1.2.3 Solid Waste Processing

In the geologic disposal alternative, retrieved solid TRU waste and contaminated soil would be sent to a waste processing facility to be combined and treated to form a chemically inert, physically stable, basalt-like slag. The waste processing facility would house the processing equipment and would be zoned for ventilation so that air would move from uncontaminated areas through successively more contaminated areas and finally would be discharged to the atmosphere through HEPA filters. The concentrations of radionuclides in the discharged air would be less than those that would lead to doses permitted by EPA standard 40 CFR 61: 0.025 rem/yr to the total body or 0.075 rem/yr to any organ of a member of the public.

The retrieved contaminated soil and waste containers would be transferred from transport trucks to a receiving airlock at the processing facility. All operations in the facility, from airlock waste entry to packaging of the output product, would be remotely controlled. After passing through the airlock, each container would be weighed, assayed, examined by x-ray, and stored for further processing.

Waste processing would begin with sorting and sizing. Any materials requiring special handling would be separated from the main waste stream and treated as needed. Large items would be sized by crushing, shredding, or flattening to a configuration suitable for incineration. After sizing, the waste would be blended to achieve some uniformity of feed to a process for converting the solid waste mixtures to a stable product.

For purposes of analysis, the slagging pyrolysis incineration (SPI) unit has been assumed to be the process used. A vertical furnace with two main components, a gasifier and a secondary combustion chamber, would constitute the incinerator unit as shown in Figure B.13. Feed consisting of radioactive solids mixed with solid fuel (coal, bark, etc.) would be subjected to drying, pyrolysis, and combustion in the gasifier. A secondary combustion chamber would complete combustion of the off gas, which would then be cooled and filtered before release to the environment. The slag would be poured into molds to form solid castings. After cooling, the molds and castings would be placed in 55-gal drums, assayed for TRU content and prepared for transport to the geologic repository.

For the reference or preferred alternatives, the Waste Receiving and Processing (WRAP) facility is proposed and is described in Appendix E. For calculation purposes, it has been assumed that the facility would not provide incineration for volume reduction, but would provide for sorting, shredding, compaction, cementation and certification of the waste for shipment to the Waste Isolation Pilot Plant (WIPP).

At the Idaho National Engineering Laboratory (INEL), solid waste processing facilities are currently under construction. At the Stored Waste Examination Pilot Plant, wastes will be certified for shipment to WIPP. Uncertifiable wastes will be processed into certifiable form at the Process Experimental Processing Plant and shipped to WIPP. The Idaho experience will be utilized in the Hanford Waste Management Program.

B.1.3 Mechanical Processing

Mechanical processing would be used to prepare strontium and cesium capsules for disposal, and also in the reference and preferred alternatives to prepare remote-handled TRU solid waste for shipment to a geologic repository.

Encapsulated strontium and cesium require some mechanical processing to provide additional outer containers for ease of handling and to manage the decay heat. The capsules containing solid strontium fluoride (SrF_2) or cesium chloride (CsCl) are currently stored in water basins in the Waste Encapsulation and Storage Facility (WESF). In this storage mode, cooling water, makeup water, ventilation, and maintenance of facility operating systems are

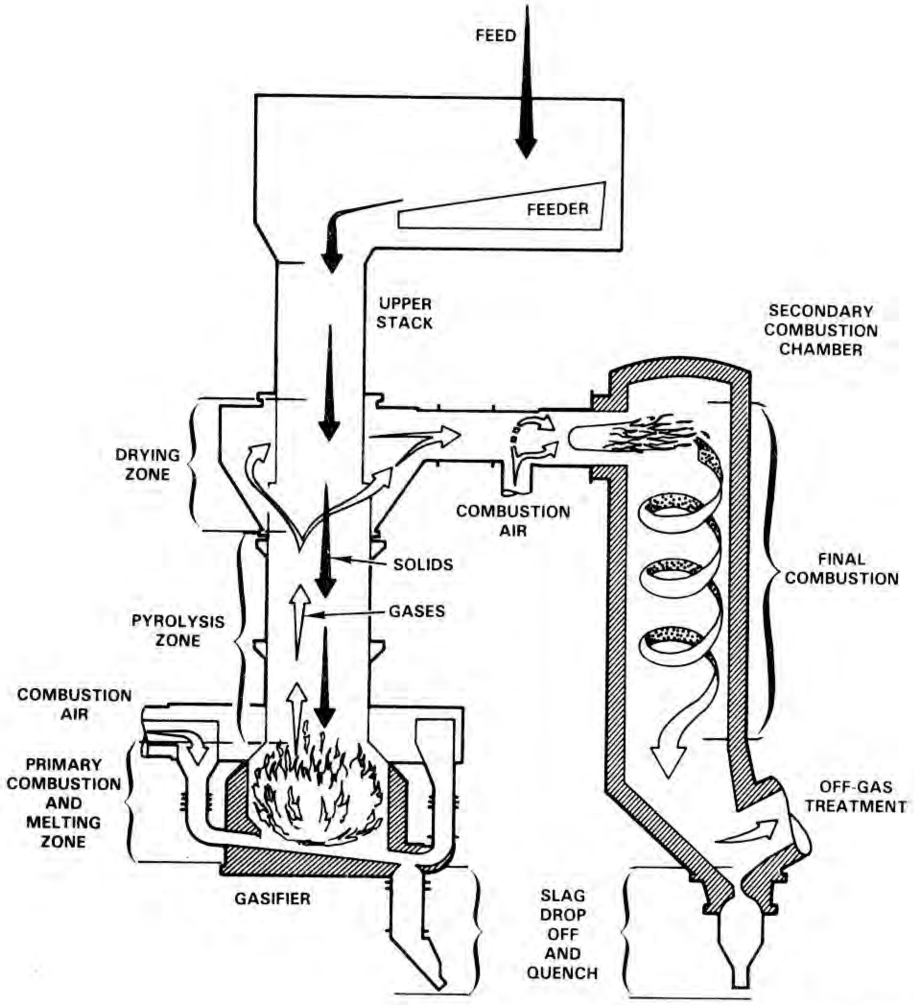

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FIGURE B.13. Slagging Pyrolysis Incineration Gasifier and Combustion Chamber

required to remove heat generated by radioactive decay. In contrast, each alternative considered in this EIS would use a passive mode of heat dissipation during storage or disposal and would eliminate the need for continued maintenance.

The capsules would be removed from the cooling basins and transported to a Capsule Packaging Facility (CPF) where the capsules would be placed in outer canisters and loaded into cask cars for transfer. In two alternatives, these canisters would then be transferred via bottom-loading transporter to the Drywell Storage Facility (DWSF). These facilities are described below.

B.1.3.1 Capsule Packaging

The Capsule Packaging Facility process is visualized as having a throughput rate of one canister per day. The strontium and cesium capsules would be placed in racks and inserted into canisters made from 0.3-m outside diameter carbon steel pipe about 2.7 m long with end plates. The canisters would be sealed, inspected, and surveyed for radioactive surface contamination. An air-cooled vault would provide lag storage space for the sealed canisters before their transfer to the drywells or a geologic repository. The following key equipment pieces would be operated in essentially the order listed to load one canister:

1. Seven-capsule holding vault
2. Three-station load/weld machine
   - At the "exit" station, a canister containing an empty capsule rack would be placed on the machine.
   - At the "load" station, an actuator, arm, and grapple would connect to the capsule rack and withdraw it vertically a distance of approximately 2 m from the canister. Capsules, handled by conventional hot cell manipulators, would be loaded onto the rack as it was lowered back into the canister.
   - At the "weld" station, a lid would be placed on the canister by manipulator or in-cell crane. A rotating weld head would make the weld closure.
3. Helium leak test unit
4. Ultrasonic weld penetration test unit
5. Electropolishing decontamination tank
6. Canister storage pods: an array of twelve 0.38-m inside diameter steel sleeves (with lids) that would penetrate the cell floor. A wind tunnel below would provide forced or natural convection air cooling.

The number of capsules loaded into a canister would vary according to heat dissipation capabilities of the storage medium (basalt, salt, or near-surface soil) and on thermal limitations of the capsule materials themselves.

The facility is envisioned as a series of three hot cells (each 4.9 m wide by 3 m, 14 m, and 7 m long) housed in an overall facility that would be 43 m long and 14 m high and occupy about 1,100 m^2. High-density concrete shielding walls of the hot cells would be ~0.9 m thick and have eight viewing windows and four manipulator pairs. Instead of building a new facility, the capsule packaging operations may be accomplished by modifying the existing Waste Encapsulation and Storage Facility.

B.1.3.2 Drywell Storage

The primary function of the Drywell Storage Facility would be passively cooled storage below ground of canisters containing encapsulated waste, with individual drywell containment for each canister. This concept was adapted from one proposed for interim storage of spent fuel assemblies from commercial light-water reactors.

The following guidelines were utilized in determining the number of capsules that could be placed in each canister and in determining the physical configuration of the drywells and storage area:

- Personnel exposure at ground level would not exceed 0.25 mrem/hr.
- Differential temperature (earth to air) at ground level would not exceed 38°C.
- Capsule wall temperature would not exceed 375°C.
- Cesium capsule centerline temperature would not exceed 420°C.
- Strontium capsule centerline temperature would not exceed 850°C.
- Canister voids would be filled with sand having thermal conductivity similar to that of the surrounding soil.
- Spacing of drywells would be 5 m from center to center.

Based on an initial analysis using these guidelines, indications are that about two strontium or four cesium capsules would be placed in each canister at the packaging facility and moved to the drywells by a shielded transporter as shown in Figure B.14. A total of 672 drywells would be required, arranged in a grid spaced 5 m from center to center, occupying ~18,000 m^2.

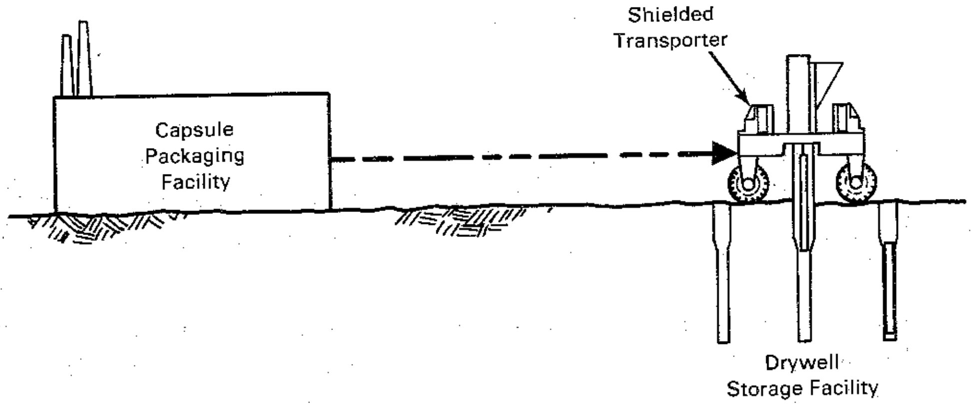


FIGURE B.14. Transfer of Strontium and Cesium Capsules from the Capsule Packaging Facility to the Drywell Storage Facility

Each drywell (Figure B.15) would consist of a cylindrical carbon steel canister encasement vessel that would extend ~0.15 m above and 4.6 m below the surface of the ground. The encasement vessels would be fabricated of 0.36- and 0.61-m diameter pipe joined by a standard pipe reducer. The encasement would be closed at the bottom by a pipe cap welded onto the 0.36-m diameter lower section of the encasement. Each drywell encasement vessel would be furnished with a carbon steel plate which would be field-welded to the top of the drywell after placement of a canister and shield sand. The closure plate would be furnished with lifting lugs and a sample valve assembly to obtain air samples and measure pressure inside the drywell. The sample valve would be protected by a detachable weather cover. A name plate would be provided on top of the closure plate for identification of the stored canister.

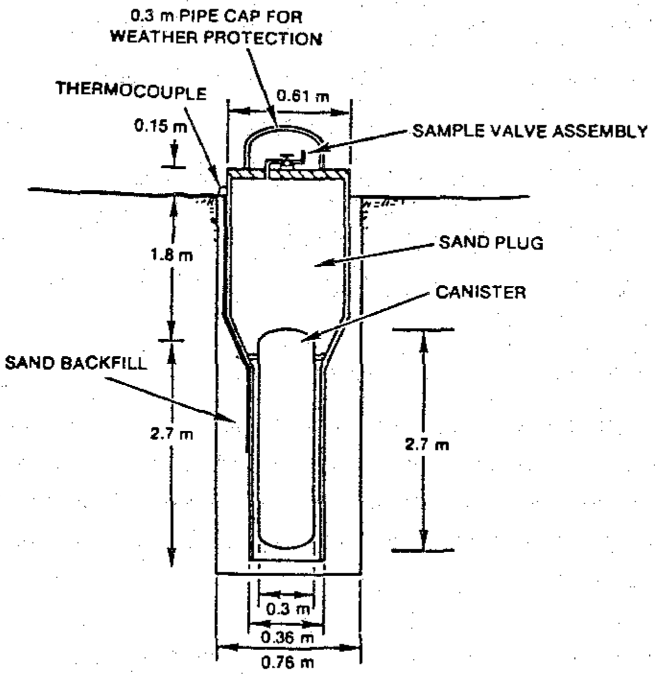


FIGURE B.15. Typical Drywell Assembly

A reusable metal cover would protect empty drywells from the weather before canister placement. A stainless steel thermowell attached to the exterior of the drywell encasement would protect a thermocouple used for periodic measurements of the drywell encasement exterior surface.

The canister would be transported to the storage field in a shielded cask transporter vehicle. The transporter would be positioned above an empty drywell, and the shielded cask would be lowered and pneumatically sealed to the top of the drywell encasement. The canister, attached to a grapple or hook device, would be lowered by a hoist mechanism built into the cask.

The canister would be suspended in the drywell by a dish-shaped steel support ring welded to the pipe reducer section of the encasement. A similar dish-shaped ring would be welded to the upper portion of the canister during fabrication of the canister. In addition to supporting the canister, the dish-shaped rings would provide a seal to retain the sand in the upper compartment of the encasement vessel. The ring configuration also would serve to center the canister in the drywell during placement.

After the canister was placed in the drywell, a loose sand fill would be placed in the compartment space above the canister to create a radiation shield plug. The sand materials would consist of a dry silica sand, washed to remove any organic materials, and dried to a minimum moisture content. To reduce the ground-surface-level radiation dose rate, approximately 0.5 m^3 of sand would be used for each sand plug to achieve a minimum sand fill of

1.8 m above the canister. A stainless steel tube through the sand shield plug would provide a connection between the spaces below and above the sand to permit sampling of the drywell interior for airborne activity.

A monitoring and surveillance program would be conducted. The interior of each drywell would be sampled periodically for pressure and airborne activity through the sample valve located on top of the drywell closure. Detection of abnormal thermal conditions would be provided by the thermocouple in the thermowell on the exterior of each drywell encasement.

B.1.3.3 Packaging of Remote-Handled TRU Solid Waste

In the reference and preferred alternatives, the retrievably stored and newly generated remotely handled TRU waste would be processed through a new facility, where it would be immobilized and packaged for shipment to a geologic repository (assumed for calculational purposes to be the WIPP). This differs from the geologic disposal alternative, in which the remotely handled TRU would be retrieved and sent to an incinerator facility (Section B.1.2.3) for final treatment. The new facility would be functionally similar to the Waste Receiving and Processing facility described in Appendix E, and would include specific processes required to immobilize and package the remotely handled solid waste. The unit operations envisioned include remote-handled and hot-cell operations to include size reduction, immobilization into a homogeneous waste form and packaging of the waste into approved waste canisters. These canisters would then be loaded into shipping casks and shipped to the geologic repository. Several alternatives for the facility are currently being investigated. The facility would be scheduled to begin operation in 1996. The new solid waste that does not require processing would be sent directly to a geologic repository.

B.1.4 Site Stabilization and Isolation

Void spaces in underground contaminated zones would be filled or compacted to stabilize the surface before covering the area with a protective barrier and marker system. This would prevent subsidence and isolate the sites from the biosphere. Research on subsidence control is being actively performed, and will continue under the Hanford Waste Management Technology Plan. The processes described below are feasible, but may be modified as a result of this research.

B.1.4.1 Subsidence Control for Waste Tanks

Drying of waste in single-shell tanks may be required so that the tank contents would support the material used for filling the void space in the tank, and to reduce the likelihood of migration of radioactive material from the tank. The use of radio-frequency energy to heat and thereby dry waste has been the most promising and economically feasible of the technologies evaluated. The radio-frequency technique could be used for drying in-tank waste by radiating radio-frequency energy into the interior of an underground tank, where the energy would be released as heat within the waste in a manner similar to microwave cooking. Due to its longer wavelength, radio-frequency energy would provide more even drying than that obtained from microwaves. To dry a tank, portable generators would transmit energy into the tank interior via coaxial cables. Dry, preheated air would be passed through the tank to remove water vapor and then discharged through two HEPA filters.

Dome fill would minimize the subsidence or collapse of the tank domes, thus minimizing the potential consequences of such a failure. It would also provide an isolating layer (barrier) between the surface environment and the residual waste materials.

Uniformly graded basalt gravel sized between 1 and 2 cm has been selected as dome fill material for use in single- and double-shell tanks.

Fill placement would be accomplished with a modified, commercially available centrifugal thrower. This equipment is used extensively for the transfer of granular and small lump materials at seaports and railroad terminals. The operating principle of the equipment is to change the direction of the falling gravel mass, using the kinetic energy of the mass to distribute the material laterally. The downward velocity of the gravel is redirected horizontally when the material is carried through an arc on a high-speed belt (Figure B.16).

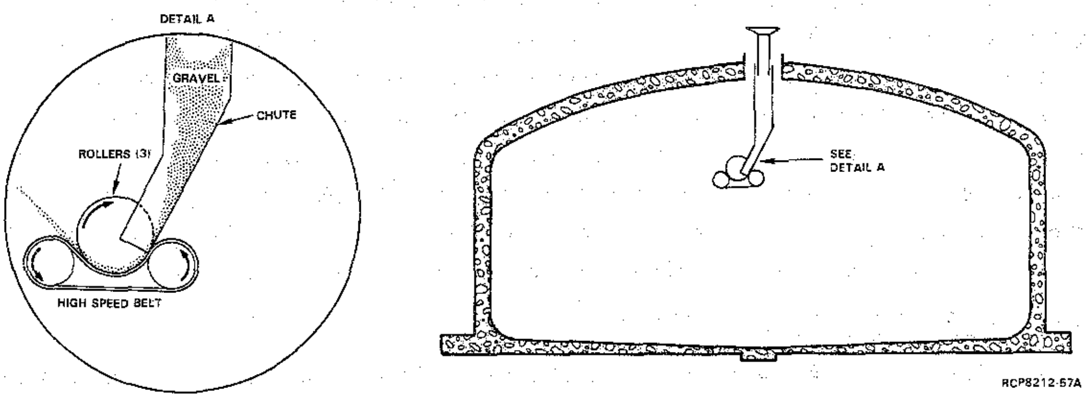


FIGURE B.16. Centrifugal Thrower for Filling Waste Tanks

The individual tanks would be maintained slightly below atmospheric pressure for the dome filling process. Ventilation air would be discharged through two HEPA filters to maintain effluent concentrations less than maximum permissible concentrations for discharge to uncontrolled areas. After completion of the fill, accessible risers and other penetrations to the tank dome would be sealed with nonradioactive grout.

In some alternatives, dome-fill material would be placed on top of waste in single-shell tanks. Porosity in the dome-fill material would allow the waste to migrate into the material and enhance gas and vapor release. Waste migration would not raise the waste level in existing tanks above the height of the steel liner in the waste tank.

B.1.4.2 Subsidence Control for Solid and Liquid Waste Sites

TRU burial grounds would be settled using a pile driver to inject rods (piles) into the waste zone. Figure B.17 illustrates the concept as applied to retrievably stored TRU waste. A diesel-powered vibratory hammer interfaced with a heavy-duty crane would inject the rods,

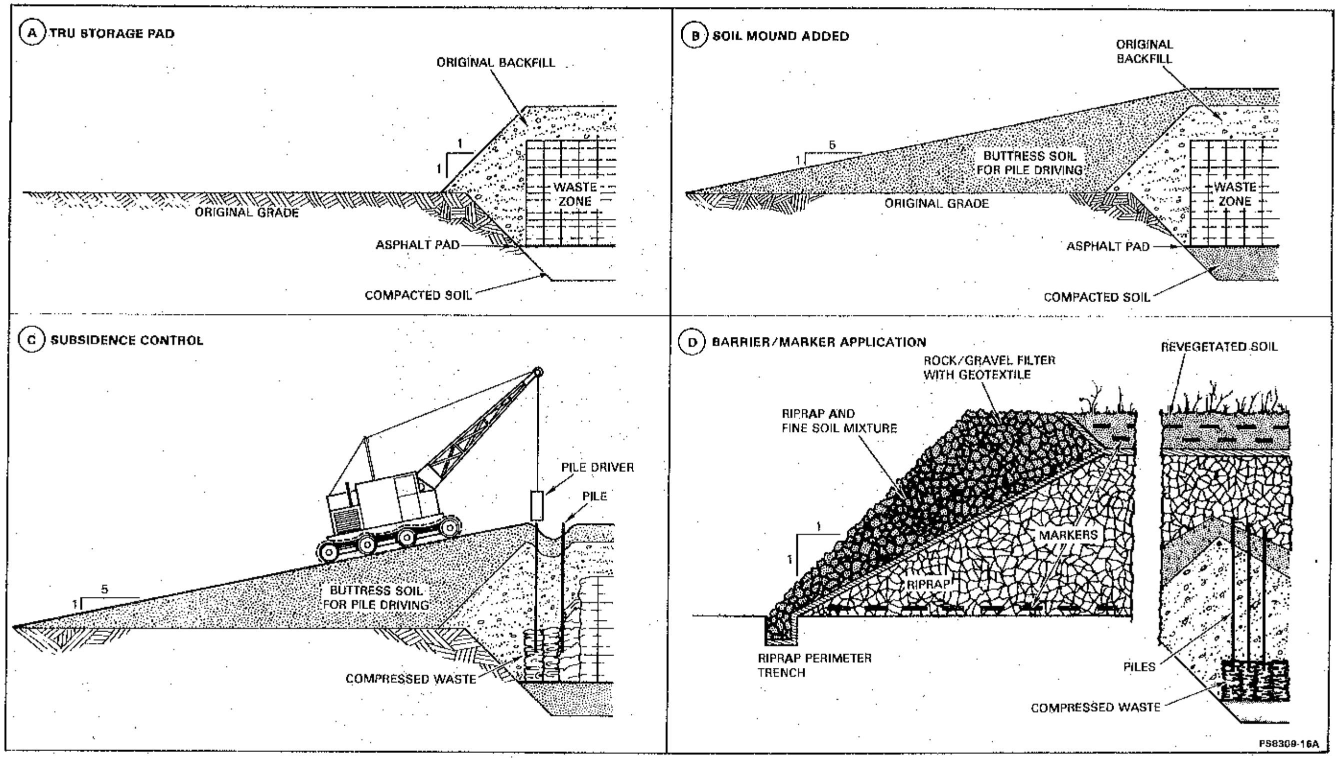
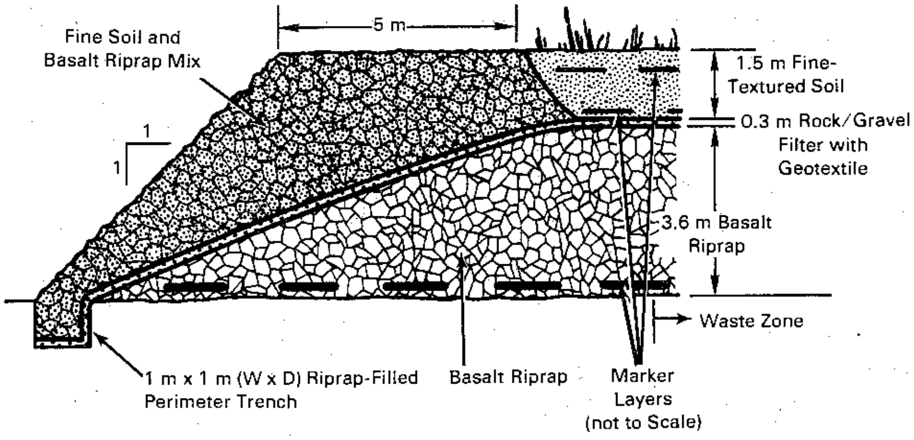


FIGURE B.17. Subsidence Control and Isolation of Solid Waste Sites causing compaction of buried drums and other containers. This approach would be used to dispose of retrievably stored TRU waste, as well as pre-1970 buried waste. The rods would be withdrawn unless contamination could be detected during rod withdrawal. If contaminated, the rods would be redriven for in-place disposal.

Significant subsidence is possible in liquid disposal sites with underground cavities that could collapse upon decay of the structure supporting the void space. Records describe the engineering features of void spaces in settling tanks and the concrete or timber-lined caverns of crib systems; other liquid disposal sites do not contain significant voids. Records would be supplemented with field examinations to select injection points for subsidence control. If no suitable openings exist into an underground cavity, a pipe would be installed. A cementitious grout would be injected into the cavity and allowed to harden. The quantity of grout injected would be monitored, and additional surveys would be conducted if the amount of grout were significantly less than the expected volume of the cavity.

B.1.4.3 Conceptual Protective Barrier and Marker System

In some alternatives, stabilized sites would be isolated by applying a protective barrier and marker system of the type shown in Figure B.18. Principal features of construction and application to various sites are described briefly below. Further details, including evaluation of performance, are described in Appendix M.



Detail of Barrier Perimeter

FIGURE B.18. Conceptual Protective Barrier and Marker System

The multilayer protective barrier system is composed of a 5.4-m-thick mound containing a 1.5-m-deep basin of revegetated soil in the upper surface over a 3.6-m layer of basalt riprap. The soil/basalt layer is laterally extended about 30 m beyond the surface projection of the waste zone or plume. Various aspects of biointrusion are to be analyzed under the Barrier Development Program. The basalt riprap consists of 12- to 25-cm-diameter rock material. A 0.3-m-thick rock/gravel layer separates the soil from the riprap and aids in minimizing the sifting of fine into riprap interstices. A 5-m-wide edge (or berm) of riprap is provided for slope protection. The sides of the barrier are constructed on a 1-to-1 slope, and a riprap-filled trench is provided at the toe of the barrier to prevent or further reduce the likelihood of animal intrusion. The berm consists of both riprap and fine soil mixed with riprap material separated by a gravel layer.

The materials (soil, gravel, basalt) for the multilayer cover are readily available on site. A thick geotextile mat, placed directly under the soil material to aid in layer construction, is being considered as an optional aid in preventing fine soil from sifting into the riprap. One analog found at Hanford, called cobble lenses, provides a practical basis for expecting long-term survival of the fines-rock interface necessary for the continued effective performance of the protective barrier.

Granite (or, possibly, quartzite) monoliths would provide surface markers for the completed barrier at the original grade. The marker base (Figure B.19) would be 1.5 m below grade while the apex would extend 3.8 m above grade. A series of repetitive messages would be engraved into each face of the monolith and possibly at the bottom base. The surface face near the message would be polished. The messages would be inscribed to a depth of at least 2.5 cm, based on extrapolations from data on weathering of tombstones. The actual message content has not been determined but would consist of simple wording such as "Caution: Buried Hazardous Waste Below" as well as a radiation symbol or simple pictograph. The actual distribution of these surface markers is still in question.

Barriers would be within a rectangular area about 6 km wide (north to south) by 13 km long (east to west). Markers may also be placed about every 200 m around the boundary of the rectangular area so that a person standing at one marker could see the markers on each side. The boundary markers would have wording to the effect "No Trespassing" or "Keep Out--Radioactive Waste Disposal Sites Ahead." This disposal site would include all the 200 Area waste disposal locations within the scope of this EIS. All markers would be built to the best of current technology, of materials having natural analogs that would suggest retention of integrity for millennia. Thus under this scenario the marker system would comprise protection in depth, a Hanford reservation boundary (performing during active institutional control), a waste disposal site with markers lasting during the period of passive institutional control, and the waste locations with monuments also lasting during the passive control period.

Three layers of subsurface markers would be distributed above the contaminated soil region of the barrier (Figure B.20). These layers would be approximately 0.6, 1.5, and 5 m from the top of the selected barrier. Markers in each layer would be spaced on 6-m centers. The two top layers would overlap to give an effective 3-m distribution so that any excavation would probably uncover at least one of the warning markers. The markers would consist of 12.7-cm-diameter porcelain or stoneware discs. These markers would be protected from surface erosion and should provide warning to potential intruders. The markers would warn the intruder about the potential hazard underlying the barrier, as well as protecting the barrier from further disruption.

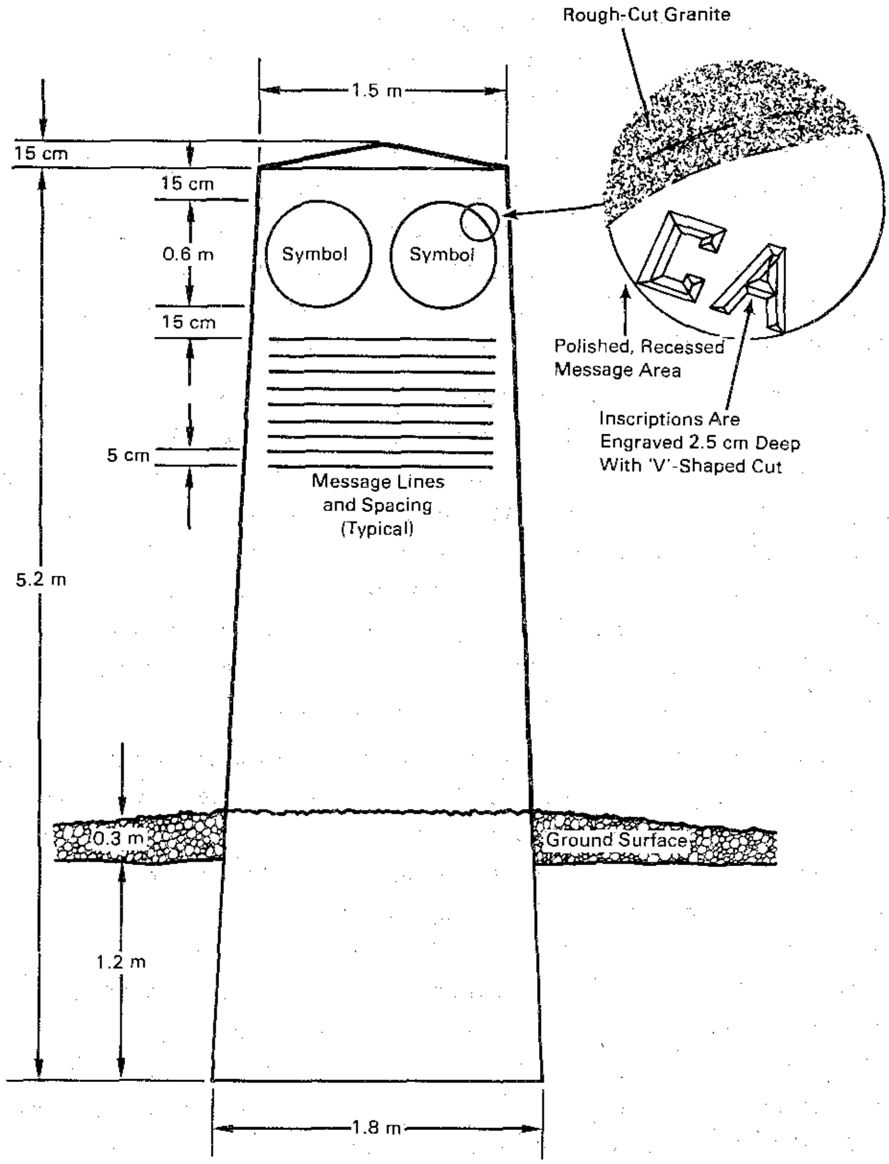


FIGURE B.19. Surface Marker Configuration and Placement

The barrier can be adapted as needed for application to any class of waste site as shown in Figure B.21 for waste tanks, Figure B.22 for TRU-contaminated soil sites, and Figure B.23 for all appropriate waste sites. Additional detail on the protective barrier and marker system is presented in Appendix M.

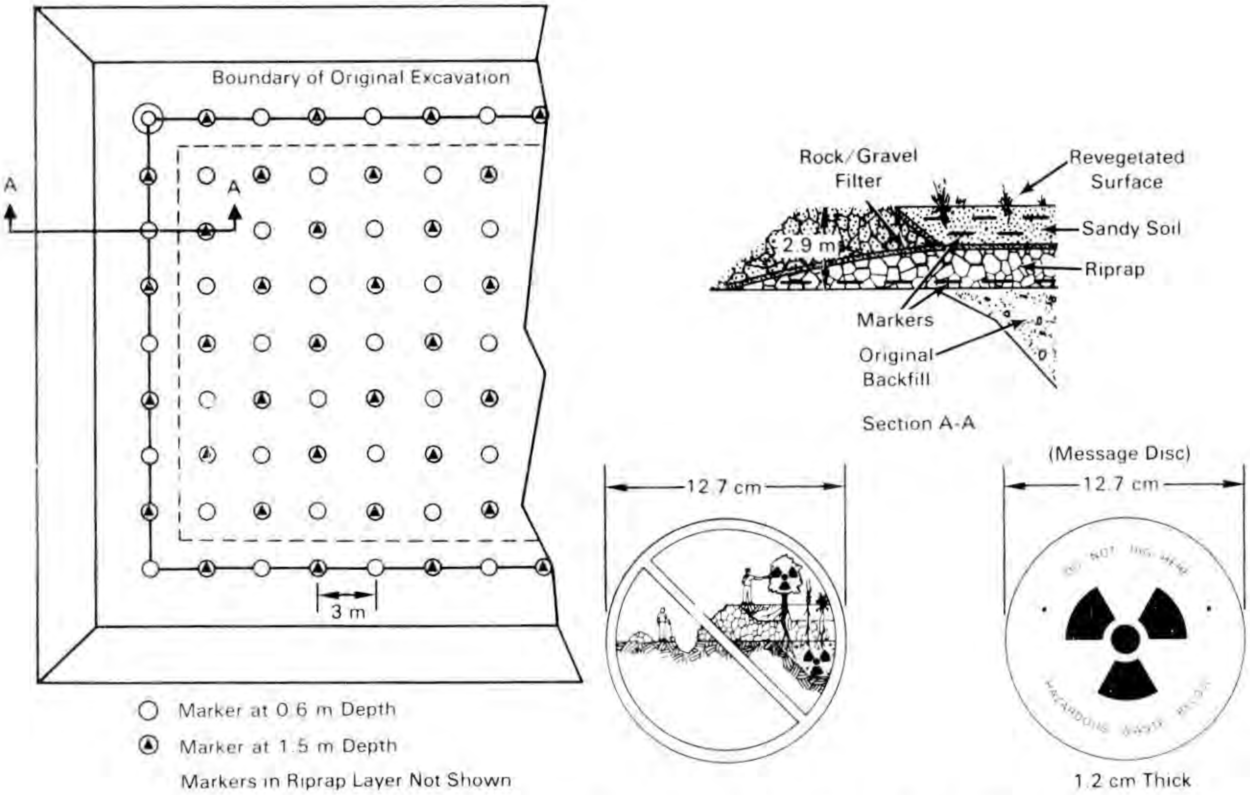


FIGURE B.20. Subsurface Marker Distribution and Marker Detail

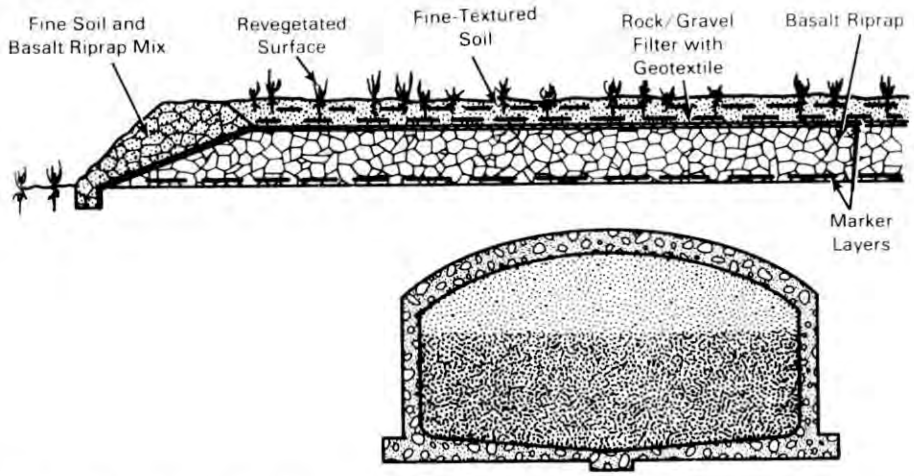


FIGURE B.21. Conceptual Protective Barrier and Marker System Applied to Waste Tanks

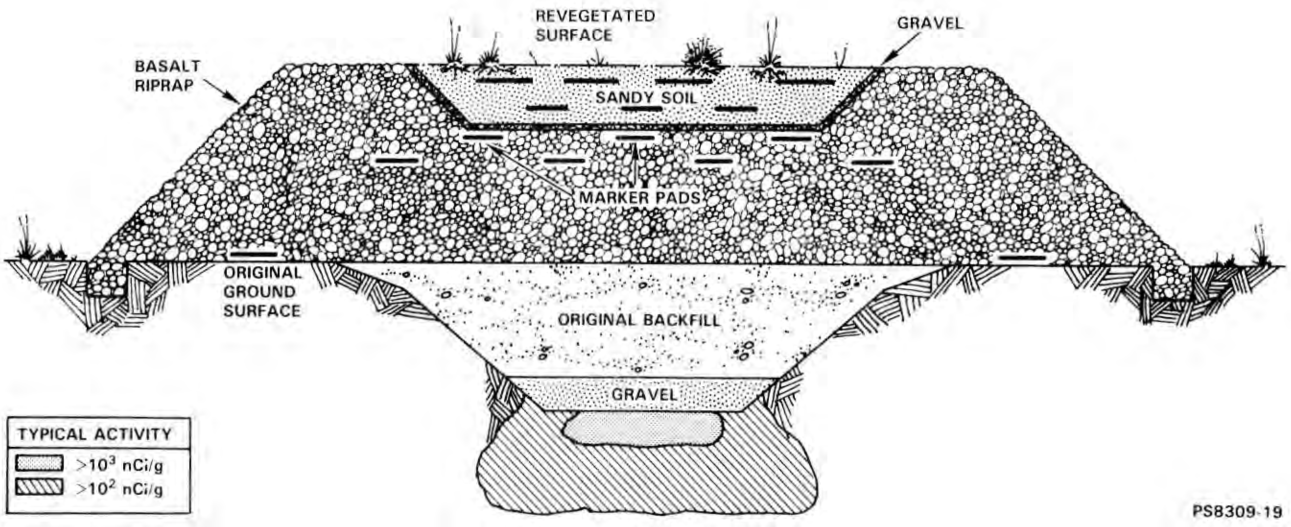


FIGURE B.22. Conceptual Protective Barrier and Marker System Applied to a TRU-Contaminated Soil Site

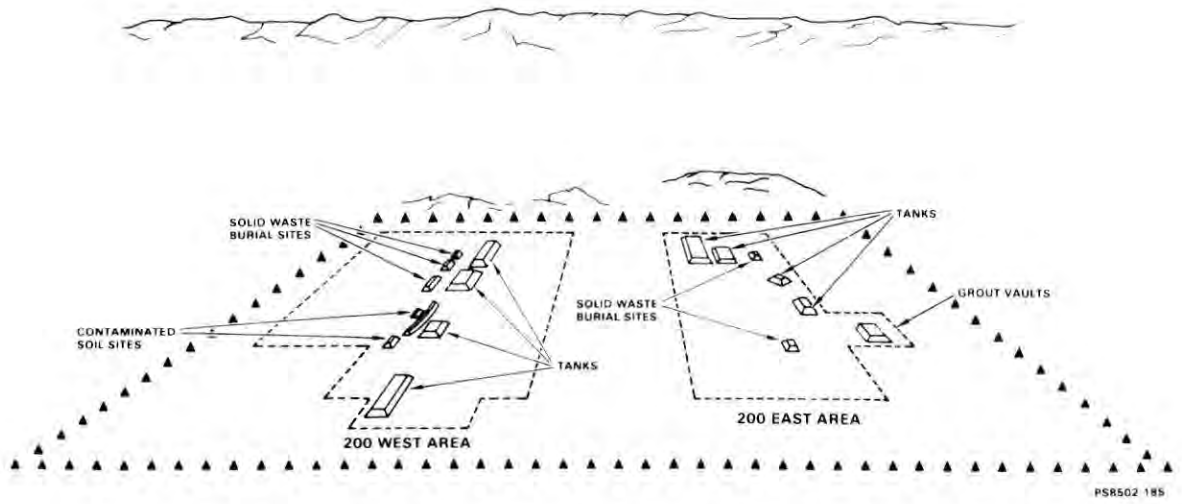


FIGURE B.23. Protective Barrier System in Place on 200 Areas Plateau

B.2 PROCESS DESCRIPTIONS AND OPERATIONAL RELEASES

Processes are described in this section for implementing each of the alternatives analyzed in this EIS, with references to new facilities and construction as appropriate. The methods described are attainable processes and are representative of those that would be used but are not necessarily now commercially available on the required scale. These descriptions are not intended to preclude more efficient methods that might be developed in the course of actual waste disposal operations.

B.2.1 Geologic Disposal

Material would be retrieved from each waste site and processed so that the bulk of the radioactive material would be disposed of in a geologic repository. Some emissions of radioactive materials would occur while retrieving the wastes and processing them to a stable chemical form. Most classes of waste would be divided into a high-activity low-volume and TRU component that goes to a repository, and a low-activity high-volume component that remains on site.

For calculational purposes in this EIS, it is assumed that waste in tanks would be retrieved using mechanical retrieval (Section B.1.1.1) for waste in single-shell tanks and hydraulic sluicing (Section B.1.1.2) for waste in double-shell tanks. Retrieved waste would be processed to remove soluble salts from the sludge and to destroy organic complexants as necessary. The supernatant liquid would be processed in a radionuclide concentration facility (Section B.1.2.1) to remove cesium, strontium, technetium, and TRU elements so that most long-lived radionuclides could be disposed of in a repository. Residual supernatant liquid after radionuclide removal would be LLW and would be disposed of in grout (Appendix D). The washed sludge would be combined with concentrates of cesium, strontium, technetium, and TRU elements for conversion to a stable chemical form (Section B.1.2.2). Emptied tanks would be stabilized by filling with gravel or grout, sealing all dome penetrations to the tanks (Section B.1.4.1), and covered with a barrier (Section B.1.4.3).

Capsules of strontium and cesium would be stored in the water basin until a geologic repository becomes available. At that time they would be placed in larger canisters for ease of handling (Section B.1.3.1), and transported to a geologic repository. Canisters would be packaged for uniform heat loading by either adjusting the number of capsules per canister or selecting capsules for each canister based on individual heat output. For geologic disposal, the maximum allowable heat per canister would be 1.17 kW of ^{90}Sr or 0.8 kW of ^{137}Cs, which corresponds to an average of about five cesium or three strontium capsules per canister. All materials, including the capsules described above, must meet applicable standards before they can be shipped to a repository.

Waste from TRU-contaminated soil sites and pre-1970 buried suspect TRU-contaminated solid waste would be retrieved mechanically (Section B.1.1.3), combined, and processed in a waste processing facility (Section B.1.2.3). The waste would be retrieved to a sufficient degree that residual contamination in each site would be classified as LLW (<100 nCi TRU/g). The retrieved waste would be sorted by TRU content to the extent practicable, and those portions that were shown to be LLW would be disposed of on site in accordance with normal procedures for LLW. At the waste processing facility the TRU wastes would be heated to produce a stable chemical form (a slag if the incineration process were selected for implementation). The product from this process would be packaged for disposal at a geologic repository (assumed to be WIPP for calculation purposes).

Retrieval of wastes from caissons and reverse wells would require modifications of the retrieval process described in Section B.1.1.3. Retrieval from caissons is described in

Section B.1.1.4, and retrieval from reverse wells is described in Section B.1.1.5. Material retrieved from caissons and reverse wells would be combined with other wastes and processed as described in the preceding paragraph.

Contact-handled TRU waste would be processed in the Waste Receiving and Processing Facility (Appendix E), with TRU waste being sent to the WIPP geologic repository, and with non-TRU waste being disposed of on site in the same manner as other LLW.

Caisson waste, which is remote handled, would be retrieved as described in Section B.1.1.4 and processed along with TRU-contaminated soil and pre-1970 buried TRU solid waste in the waste processing facility (Section B.1.2.3).

Airborne emissions of radioactive materials (Table B.1) would occur with all classes of waste, but the emissions from processing strontium and cesium capsules would be limited to those occurring during water basin storage before transferring capsules to the packaging facility. Emissions in the standby mode are estimated to be 10^{-6} Ci/yr of alpha and

TABLE B.1. **Annual Gaseous Emissions of Radionuclides, Geologic Disposal Alternative, Ci/yr**(a)

	Tank Waste		TRU Waste			
Nuclide	Existing	Future	Soil	Pre-1970	Retrievable	Total
^{241}Am	6×10^{-7}	2×10^{-6}	7×10^{-7}	1×10^{-6}	3×10^{-6}	8×10^{-6}
^{14}C	8×10^{-8}	2		5×10^{-9}	8×10^{-10}	2
^{137}Cs(b)	3×10^{-4}	3×10^{-4}	4×10^{-1}	5	1×10^{2}	1×10^{2}
^{3}H		8×10^{-2}	5×10^{-2}	5×10^{4}	7	5×10^{4}
^{129}I	6×10^{-10}	1×10^{-6}				1×10^{-6}
^{147}Pm		2×10^{-5}			2×10^{-7}	2×10^{-5}
^{238}Pu	4×10^{-9}	1×10^{-9}	9×10^{-8}	2×10^{-7}	3×10^{-6}	3×10^{-6}
^{239}Pu	2×10^{-7}	3×10^{-8}	2×10^{-6}	4×10^{-6}	1×10^{-5}	2×10^{-5}
^{240}Pu	6×10^{-8}	7×10^{-9}	5×10^{-7}	1×10^{-6}	3×10^{-6}	5×10^{-6}
^{241}Pu	5×10^{-7}	2×10^{-8}	1×10^{-6}	3×10^{-6}	9×10^{-5}	9×10^{-5}
^{106}Ru(b)	3×10^{-11}	3×10^{-4}		8×10^{-11}	2×10^{-9}	3×10^{-4}
^{151}Sm	9×10^{-6}	2×10^{-6}				1×10^{-5}
^{90}Sr(b)	8×10^{-4}	3×10^{-4}	9×10^{-4}	5×10^{-3}	9×10^{-2}	1×10^{-1}
^{99}Tc	4×10^{-7}	3×10^{-8}				4×10^{-7}
^{93}Zr	6×10^{-8}	1×10^{-8}				7×10^{-8}

(a) Based on Rockwell 1985.
(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in dose calculations.

10^{-5} Ci/yr of beta activity. Duration of emissions and operations leading to emissions vary with the waste class. It is assumed that tank waste would be processed over an 18-year period, with most emissions coming from either vitrification or tank ventilation. Carbon emissions would also be significant during hydraulic sluicing and ruthenium emissions during feed preparation. Emissions due to TRU-contaminated soil sites and pre-1970 buried TRU solid waste would largely arise over a 20-year period from the waste processing facility. However, tritium emissions from solid waste sites would occur primarily during waste retrieval. Emission during processing of retrievably stored and newly generated TRU waste would be roughly equal from the Waste Receiving and Processing Facility and from the waste processing facility used for caisson waste. Each facility would operate for about 12 years.

No contaminated liquid releases are anticipated for the geologic disposal alternative.

Although the majority of radioactive material would be sent to a geologic repository, some, from most of the waste classes, would remain on site. Tank waste would be divided between glass canisters being sent to geologic disposal and grout retained on site. Some waste would not be retrieved from its current location, but would remain as residue in tanks, TRU-contaminated soil sites, or pre-1970 TRU solid waste burial sites. Residual activity at current waste sites is expected to be greatest for single-shell tanks (5% of original) and least for double-shell tanks (0.05% of original). Essentially no residual onsite radioactivity is expected for strontium and cesium capsules or for retrievable and newly generated TRU solid waste. Estimated compositions of final waste forms (glass, grout, and slag) are listed in Table B.2.

The average concentration of alpha-emitting TRU radionuclides in the grout would be about 0.004 Ci/m^3 (equivalent to ~3 nCi/g, assuming a grout density of 1.4 g/cm^3). Since it would not be practical to make a uniform blend of all waste that goes into the grout, and since cladding removal waste or other waste may be TRU waste streams, individual grout batches may have a higher-than-average TRU concentration. If the TRU concentration in cladding waste could be reduced significantly, the chance for higher TRU concentrations in grout would reduce appreciably. Blending with the large volume of grout from existing tank waste would also reduce TRU concentrations to low levels.

B.2.2 In-Place Stabilization and Disposal

Material would be stabilized in its present location if practicable. Some processing and relocation of waste would be required to achieve the desired level of stabilization. Each site containing residual HLW or TRU waste would be covered with a protective barrier and marker system (Section B.1.4.3) to isolate the site from the environment.

Waste in single-shell tanks would be left in place, and dried as required to achieve adequate stability. Void space in the tank would be filled and surface-accessible dome penetrations would be sealed (Section B.1.4.1).

Immediate installation of barriers is a problem for approximately 12 tanks in A, C, and SX farms since these tanks may reach unacceptably high temperatures. Deferral of barrier

TABLE B.2. Average Composition of Final Waste Forms for Geologic Disposal Alternative, Ci/m^3(a)

Nuclide	Existing Tank Waste(b)		Future Tank Waste(c)		TRU Soil plus Pre-1970 TRU Solid Waste Sites(d)
	Grout	Glass	Grout	Glass	
^{241}Am	3×10^{-4}	4	3×10^{-2}	2×10^{2}	2×10^{-1}
^{14}C	7×10^{-3}		3×10^{-3}		2×10^{-5}
^{137}Cs(e)	3×10^{-1}	2×10^{3}	2×10^{1}	2×10^{4}	4×10^{-1}
^{3}H			7×10^{-2}		
^{129}I	6×10^{-5}		1×10^{-4}		
^{147}Pm			2×10^{-1}	2×10^{3}	
^{238}Pu	3×10^{-6}	4×10^{-2}	1×10^{-4}	1×10^{-1}	2×10^{-2}
^{239}Pu	1×10^{-4}	2	2×10^{-3}	2	6×10^{-1}
^{240}Pu	4×10^{-5}	4×10^{-1}	6×10^{-4}	6×10^{-1}	1×10^{-1}
^{241}Pu	3×10^{-4}	4	2×10^{-3}	2	4×10^{-1}
^{106}Ru(e)	8×10^{-7}	5×10^{-5}	7×10^{1}	3×10^{3}	3×10^{-7}
^{151}Sm	6×10^{-3}	7×10^{1}	2×10^{-2}	2×10^{2}	
^{90}Sr(e)	3×10^{-1}	4×10^{3}	2	2×10^{4}	4×10^{-1}
^{99}Tc	4×10^{-4}	2	2×10^{-3}	2	
^{93}Zr	3×10^{-5}	4×10^{-1}	6×10^{-4}	9×10^{-1}	

(a) Based on Rockwell 1985. To convert from Ci/m^3 to nCi/g, multiply by 714 for grout or by 357 for glass or slag.

(b) Volume of grout would be 736,000 m^3; volume of glass would be 12,300 m^3 in 19,800 canisters.

(c) Volume of grout would be 99,000 m^3; volume of glass would be 2,050 m^3 in 3,310 canisters.

(d) These two classes of waste would be combined during processing, and would result in 58,000 m^3 of slag.

(e) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in dose calculations.

construction over these tanks until the year 2030 would reduce the number of problem tanks to one. This tank would essentially be disposed of in 2030 by installing barriers and supplementing the normal heat dissipation with a system of heat pipes.

Heat pipes would be installed vertically around the outer perimeters of the tank. The lower ends of the heat pipes would be just above the tank base while the tops would extend above the barriers to dissipate heat into the air. A total of 102 pipes would be used. The heat pipe system would be maintained until the decay heat being generated was no longer sufficient to overheat the tank (about 2070). At that time, the pipes would be sealed or filled with clean grout.

Waste in double-shell tanks would be retrieved by hydraulic sluicing (Section B.1.1.2). HLW would be processed through B Plant and the Waste Encapsulation and Storage Facility for cesium removal and encapsulation. Wastes with high concentrations of organic complexants would be treated to reduce the complexant concentration to an acceptable level (Section B.1.2.1). The residues from HLW processing (except the cesium capsules) would be combined with residues from organic complexant destruction and all other waste from double-shell tanks. This combined waste would be processed through the Transportable Grout Facility (Appendix D) and disposed of in near-surface vaults or as fill for void space in tanks. Based on a preliminary cost and process feasibility analysis, it is assumed that vault disposal would be used. An estimated 29 ha of disposal area would be needed to isolate about 272,000 m^3 of grout. Emptied double-shell tanks would be filled (Section B.1.4.1) and sealed to isolate them from the environment. Tank sites and grout disposal sites would be covered with a protective barrier (Section B.1.4.3).

Strontium and cesium capsules would be retrieved from the water basin beginning about the year 2010, placed in larger canisters for ease of handling (Section B.1.3.1), and disposed of in drywells (Section B.1.3.2). For the drywells, the maximum allowable heat per canister is 500 W, which corresponds to an average of about two strontium or four cesium capsules per canister. The drywells would be covered with the protective barrier (Section B.1.4.3).

TRU waste would be left in its current near-surface locations. As additional actions, however, all sites would be stabilized either by filling void spaces with grout or by compaction of solid waste sites (Section B.1.4.2). Grout fill would be accomplished by injection of cementitious grout using mixing tanks, proportioning transfer pumps, hoses, and pneumatic drills. The stabilized site would be isolated by application of a protective barrier (Section B.1.4.3).

Airborne emissions of radioactive materials would occur primarily during processing of tank waste (Table B.3). Some emissions would occur during storage of strontium and cesium capsules in water basins, estimated to be 10^{-6} Ci/yr of alpha and 10^{-5} Ci/yr of beta activity. Airborne releases from TRU sites would be negligible in comparison to releases from other waste classes.

Discharges of small amounts of liquid radioactive tank wastes (Table B.3), resulting from waste concentration operations, would occur and would be handled as any other low-level liquid waste discharge.

The disposal form would be essentially identical to the present waste composition (Appendix A) for all classes of waste except those in double-shell tanks. Such wastes would be converted to grout of the composition shown in Table B.4. In addition, cesium removal would result in 811 capsules containing 4.7×10^7 Ci of ^{137}Cs. About 0.05% of the original waste would remain in double-shell tanks.

B.2.3 Reference Alternative

This alternative combines features of the geologic disposal alternative and the in-place stabilization and disposal alternative, with some process steps unique to this alternative. Strontium and cesium capsules and retrievably stored and newly generated contact-handled TRU

TABLE B.3. **Annual Releases of Radionuclides, In-Place Stabilization and Disposal Alternative, Ci/yr**[(a)]

	Airborne Emissions		Liquid Discharges	
Nuclide	Existing Tank Waste	Future Tank Waste	Existing Tank Waste	Future Tank Waste
^{241}Am	6×10^{-7}	2×10^{-6}	6×10^{-4}	5×10^{-4}
^{14}C	8×10^{-8}	2×10^{-9}	8×10^{-4}	1×10^{-4}
^{137}Cs(b)	3×10^{-4}	3×10^{-2}	4×10^{-1}	2×10^{-1}
^{3}H		3×10^{1}		3×10^{3}
^{129}I	6×10^{-10}	9×10^{-11}	8×10^{-6}	8×10^{-6}
^{147}Pm		3×10^{-5}		7×10^{-3}
^{238}Pu	4×10^{-9}	1×10^{-9}		4×10^{-7}
^{239}Pu	2×10^{-7}	3×10^{-8}	2×10^{-6}	8×10^{-6}
^{240}Pu	6×10^{-8}	8×10^{-9}	4×10^{-7}	2×10^{-6}
^{241}Pu	5×10^{-7}	2×10^{-8}		6×10^{-6}
^{106}Ru(b)	9×10^{-12}	8×10^{-5}	6×10^{-9}	4×10^{-1}
^{151}Sm	9×10^{-6}	2×10^{-6}	6×10^{-3}	8×10^{-4}
^{90}Sr(b)	7×10^{-4}	2×10^{-3}	4×10^{-1}	8×10^{-2}
^{99}Tc	4×10^{-7}	3×10^{-8}	6×10^{-3}	3×10^{-3}
^{93}Zr	5×10^{-8}	1×10^{-8}	4×10^{-6}	5×10^{-6}

(a) Based on Rockwell 1985.

(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in dose calculations.

waste would be handled as in the geologic disposal alternative (Section B.2.1). Retrievably stored remotely handled TRU waste in caissons would be processed in a shielded addition to the TRU waste processing facility or shipped off site. Waste in single-shell tanks, TRU-contaminated soil sites, and most pre-1970 suspect TRU solid waste burial grounds would be left in place, as for the in-place stabilization and disposal alternative, and a barrier and marker system would be applied. Solid wastes in a site outside the 200 Areas (site 618-11) would be retrieved as in the geologic disposal alternative and processed in the TRU waste processing facility. A recently completed study (DOE 1986), which examined records of inactive waste disposal locations on the Hanford Site, showed that two 618 sites (618-1 and 618-2) each contained one gram of plutonium rather than the previously listed 1,000 grams (Rockwell 1985). As a result of this lower quantity, both sites are now designated as low-level waste (Rockwell 1987).

Processing of wastes in double-shell tanks would be somewhat similar to the geologic disposal alternative, but with a much smaller amount of processing and new facility construction. Process steps would be similar to geologic disposal for waste retrieval (Section B.1.1.2), sludge washing, organic complexant destruction, and cesium removal

TABLE B.4. Average Radionuclide Content of Grout for In-Place Stabilization and Disposal Alternative, Ci/m^{3}[a]

Nuclide	Existing Tank Waste	Future Tank Waste
^{241}Am	9×10^{-2}	3
^{14}C	1×10^{-2}	3×10^{-3}
^{137}Cs[b]	6×10^{1}	4×10^{1}
^{3}H		2×10^{-2}
^{129}I	1×10^{-4}	1×10^{-4}
^{147}Pm		4×10^{1}
^{238}Pu	6×10^{-5}	2×10^{-3}
^{239}Pu	5×10^{-4}	5×10^{-2}
^{240}Pu	1×10^{-4}	1×10^{-2}
^{241}Pu	2×10^{-3}	3×10^{-2}
^{106}Ru[b]	3×10^{-6}	1×10^{2}
^{151}Sm	1	4
^{90}Sr[b]	5×10^{1}	4×10^{2}
^{99}Tc	8×10^{-2}	5×10^{-2}
^{93}Zr	1×10^{-3}	2×10^{-2}

[a] Based on 173,000 m^{3} of grout from existing tank waste and 99,000 m^{3} of grout from future tank waste. Derived from Rockwell 1985. To convert from Ci/m^{3} to nCi/g, multiply by 714.

[b] Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in dose calculations.

(Section B.1.2.1). However, a much smaller version of the radionuclide concentration facility would suffice, since wastes from single-shell tanks would not be processed. In fact, some or all of these waste processing operations may be done in existing facilities such as B Plant. There would be no recovery of technetium or strontium, thus minimizing new construction. The washed sludge would be combined with the recovered cesium and converted to a glass in a small vitrification facility (Appendix C). Accessible voids in emptied tanks would be filled with gravel (Section B.1.4.1) or other material, and dome penetrations into the tanks would be sealed.

Most releases of radioactive material would occur when processing wastes stored in tanks (Table B.5). Most of the gaseous releases from existing tank waste would occur over a 40-year period from tank ventilation systems, and releases from future tank wastes would occur over a 6-year period during all phases of waste retrieval and processing. Liquid discharges would occur as a result of evaporation to obtain a concentrated feed for grout preparation and disposal. No significant releases are anticipated from other classes of waste.

TABLE B.5. Annual Operational Releases of Radionuclides, Reference or Preferred Alternative, Ci/yr(a)

	Airborne Emissions		Liquid Discharges	
Nuclide	Existing Tank Waste	Future Tank Waste	Existing Tank Waste	Future Tank Waste
^{241}Am	6×10^{-7}	2×10^{-6}	6×10^{-4}	2×10^{-4}
^{14}C	8×10^{-8}	7×10^{-1}	8×10^{-4}	2×10^{-5}
^{137}Cs(b)	3×10^{-4}	3×10^{-2}	4×10^{-1}	5×10^{-2}
^{3}H		9		9×10^{2}
^{129}I	6×10^{-10}	5×10^{-6}	8×10^{-6}	2×10^{-6}
^{147}Pm		3×10^{-5}		2×10^{-3}
^{238}Pu	4×10^{-9}	2×10^{-9}	6×10^{-8}	7×10^{-8}
^{239}Pu	2×10^{-7}	3×10^{-8}	2×10^{-6}	1×10^{-6}
^{240}Pu	6×10^{-8}	8×10^{-9}	4×10^{-7}	4×10^{-7}
^{241}Pu	5×10^{-7}	2×10^{-8}	9×10^{-7}	1×10^{-6}
^{106}Ru(b)	3×10^{-11}	6×10^{-4}	6×10^{-9}	1×10^{-1}
^{151}Sm	1×10^{-5}	3×10^{-6}	6×10^{-3}	2×10^{-4}
^{90}Sr	8×10^{-4}	2×10^{-3}	4×10^{-1}	2×10^{-2}
^{99}Tc	4×10^{-7}	3×10^{-8}	6×10^{-3}	9×10^{-4}
^{93}Zr	5×10^{-8}	2×10^{-8}	4×10^{-6}	1×10^{-5}

(a) Based on Rockwell 1985.

(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in dose calculations.

Double-shell tank waste would be disposed of as grout containing low concentrations of radioactive materials and as a much smaller volume of glass containing the bulk of the radioactive materials (Table B.6). Assuming a grout density of 1.4 g/cm^3, the average TRU content of grout would be about 6 nCi/g for existing tank waste and 80 nCi/g for future tank waste. During processing, a uniform blend of the two waste types would be produced that would contain about 30 nCi TRU/g. This TRU concentration is much lower than the concentration in the average TRU-contaminated soil site. The remaining waste classes would be disposed of in a final form similar to their current form.

B.2.4 No Disposal Action (Continued Storage)

Wastes would be left in place to the extent practicable, and only those actions would be undertaken that were necessary to ensure safe interim storage. No actions would be taken that would make future waste retrieval or disposal more difficult.

Principal actions would relate to liquids associated with certain waste classes. Double-shell tanks contain large volumes of liquid, and leaks through both of the steel

TABLE B.6. Radionuclide Content of Final Disposal Forms for Reference or Preferred Alternative Applied to Double-Shell Tanks, Ci/m^3[(a)]

Nuclide	Existing Tank Waste(b)		Future Tank Waste(c)	
	Grout	Glass	Grout	Glass
^{241}Am	9×10^{-3}	5×10^{1}	1×10^{-1}	9×10^{2}
^{14}C	1×10^{-2}		3×10^{-3}	
^{137}Cs(d)	6×10^{1}		4×10^{1}	1×10^{5}
^{3}H			2×10^{-2}	
^{129}I	1×10^{-4}		1×10^{-4}	
^{147}Pm			1	1×10^{4}
^{238}Pu	6×10^{-6}	3×10^{-2}	2×10^{-4}	6×10^{-1}
^{239}Pu	5×10^{-5}	3×10^{-1}	5×10^{-3}	1×10^{1}
^{240}Pu	1×10^{-5}	8×10^{-2}	1×10^{-3}	3
^{241}Pu	2×10^{-4}	1	4×10^{-3}	8
^{106}Ru(d)	1×10^{-6}	8×10^{-4}	8×10^{1}	1×10^{4}
^{151}Sm	1×10^{-1}	5×10^{2}	1×10^{-1}	1×10^{3}
^{90}Sr(d)	5	3×10^{4}	1×10^{1}	1×10^{5}
^{99}Tc	8×10^{-2}		5×10^{-2}	
^{93}Zr	1×10^{-4}	5×10^{-1}	1×10^{-3}	5

(a) Based on Rockwell 1985. To convert from Ci/m^3 to nCi/g, multiply by 714 for grout or 357 for glass.
(b) Based on 173,000 m^3 of grout and 293 m^3 of glass in 473 canisters.
(c) Based on 99,000 m^3 of grout and 369 m^3 of glass in 595 canisters.
(d) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in dose calculations.

liners could result in release of the liquid wastes to the environment. Monitoring of the tanks would continue, and the wastes would be transferred to replacement tanks before the design life of the tanks was exceeded. Strontium and cesium capsules are stored in water basins, and loss of water from the basins could increase the temperature of capsules. The capsules would be moved to dry storage before the design life of the water basins was exceeded.

Other waste classes would be monitored and maintained as necessary to ensure continued safety for interim storage.

Most of the emissions would occur from waste tanks during storage (forced ventilation), with lesser amounts from hydraulic sluicing, and evaporation. Airborne emissions from existing waste tanks would occur primarily during the 100-year period of tank breathing ventilation. Emissions from future tank waste would be distributed among all phases of the operations. All liquid discharges from both existing and future tank waste would occur as a result of evaporation to remove the dilution water used in transferring wastes to replacement tanks. Table B.7 shows the expected annual releases.

TABLE B.7. Annual Releases of Radioactivity, Continued Storage Alternative, Ci/yr[(a)]

	Airborne Emissions		Liquid Discharges	
Nuclide	Existing Tank Waste	Future Tank Waste	Existing Tank Waste	Future Tank Waste
^{241}Am	6×10^{-7}	2×10^{-6}	6×10^{-4}	7×10^{-3}
^{14}C	8×10^{-8}	1×10^{-9}	8×10^{-4}	7×10^{-5}
^{137}Cs(b)	3×10^{-4}	3×10^{-4}	4×10^{-1}	1
^{3}H		1×10^{1}		1×10^{3}
^{129}I	6×10^{-10}	7×10^{-11}	8×10^{-6}	3×10^{-6}
^{147}Pm		2×10^{-5}		7×10^{-2}
^{238}Pu	4×10^{-9}	1×10^{-9}	2×10^{-7}	4×10^{-6}
^{239}Pu	2×10^{-7}	2×10^{-8}	2×10^{-6}	9×10^{-5}
^{240}Pu	6×10^{-8}	6×10^{-9}	4×10^{-7}	2×10^{-5}
^{241}Pu	5×10^{-7}	2×10^{-8}	8×10^{-6}	6×10^{-5}
^{106}Ru(b)	9×10^{-12}	6×10^{-5}	6×10^{-9}	2×10^{-1}
^{151}Sm	9×10^{-6}	2×10^{-6}	6×10^{-3}	7×10^{-3}
^{90}Sr(b)	7×10^{-4}	2×10^{-4}	4×10^{-1}	7×10^{-1}
^{99}Tc	4×10^{-7}	3×10^{-8}	6×10^{-3}	1×10^{-3}
^{93}Zr	5×10^{-8}	1×10^{-8}	4×10^{-6}	4×10^{-5}

(a) Based on Rockwell 1985.
(b) Does not include activity of short-lived daughters in equilibrium with parent radionuclide. Daughters are accounted for in dose calculations.

All of the waste classes would retain essentially the same chemical composition and radionuclide content described in Appendix A.

B.2.5 Preferred Alternative

This alternative combines features of the reference alternative and a deferred decision on disposal action. Existing tank waste in double-shell tanks, future tank waste, strontium and cesium capsules and retrievably stored and newly generated TRU solid waste would be handled as in the reference alternative (Section B.2.3). The low-activity fraction of double-shell tank waste would be solidified in a cement-based grout and disposed of near surface in double-lined vaults, specially designed to meet RCRA and long-term performance requirements. Waste in single-shell tanks, TRU-contaminated soil sites, and pre-1970 buried suspect TRU-contaminated solid wastes (except the 618-11 site) would be left in place pending decision between geologic disposal and in-place stabilization and disposal. Before a final decision is made, these latter wastes will continue to be stored and maintained under the no disposal action alternative in the short term.

Releases of radioactive material would be as described in Section B.2.3 and listed in Table B.5.

Waste forms for final disposal would be as described in Section B.2.3 and listed in Table B.6.

B.3. REFERENCES

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APPENDIX C

HANFORD WASTE VITRIFICATION PLANT

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APPENDIX C

HANFORD WASTE VITRIFICATION PLANT

This appendix describes the Hanford Waste Vitrification Plant at the "preconceptual" design stage and therefore the design details may differ when final design is complete.

The purpose of the Hanford Waste Vitrification Plant (HWVP) is to convert high-activity tank waste at the Hanford Site to a solid vitrified form (borosilicate glass), package it, and store it suitably for final disposal in a geologic repository. Construction of the HWVP is currently proposed to permit startup and operation in FY 1999.

Operation of HWVP will continue for a sufficient period of time to vitrify designated liquid defense HLW and TRU wastes at the Hanford Site. One scenario estimates this time period to be approximately 10 to 20 years (Rockwell 1987). The period of operation is dependent upon: 1) the type, timing, and availability of suitable liquid wastes; 2) the schedule for and the type of waste pretreatment required; 3) the throughput of HWVP; 4) the HLW and TRU content of the waste glass; and 5) the startup date of HWVP.

Current HWVP reference design recommendations estimate annual production of approximately 250 metric tons of glass containing a nominal 25 wt% waste oxide loading. This waste glass will be contained in canisters 0.6 m in diameter and 3 m in length.

The construction cost of HWVP is currently estimated to be approximately $920 million escalated to the midpoint of design and construction activities (about 1993). Operating costs are estimated to be approximately $40 million per year (1987 dollars). The HWVP is similar to the Defense Waste Processing Facility (DWPF) at the Savannah River Plant although throughput and waste feed compositions differ between the two facilities (DOE 1982a).

This appendix describes the HWVP as discussed under the reference disposal alternative. If it is determined that all or part of the single-shell tank wastes are to be retrieved, several alternative processes will be evaluated. The current HWVP design will accommodate some to all single-shell tank wastes, dependent on preprocessing constraints, final waste characteristics and final grout disposal criteria. However, alternatives will be evaluated to select the optimal final disposal process. National Environmental Policy Act (NEPA) documentation to evaluate options will be prepared before final single-shell tank waste disposal decisions are made. Also addressed here are the facility description (Section C.1), relationship to other facilities (C.2), process description (C.3), HWVP waste form (C.4), waste feedstreams (C.5), resource requirements (C.6), projected radiological impacts and emissions (C.7), nonradiological emissions (C.8), and cost estimates (C.9). The objective of this appendix is to provide conceptual information on the HWVP, its role in waste management operations, and general characteristics of the materials processed by the plant. Additional details are available in Appendix C of RHO-RE-ST-30 P (Rockwell 1985) and the addendum (Rockwell 1987).

C.1 FACILITY DESCRIPTION

The HWVP (see Figure C.1) is currently in the preconceptual design stage, with a definitive plant arrangement yet to be completed. The isometric, plan view, and process flow information in this appendix is based on a configuration developed for scoping purposes. The HWVP conceptual design layout will functionally be the same but may be rearranged to provide a more cost-effective design and more efficient operating configuration.

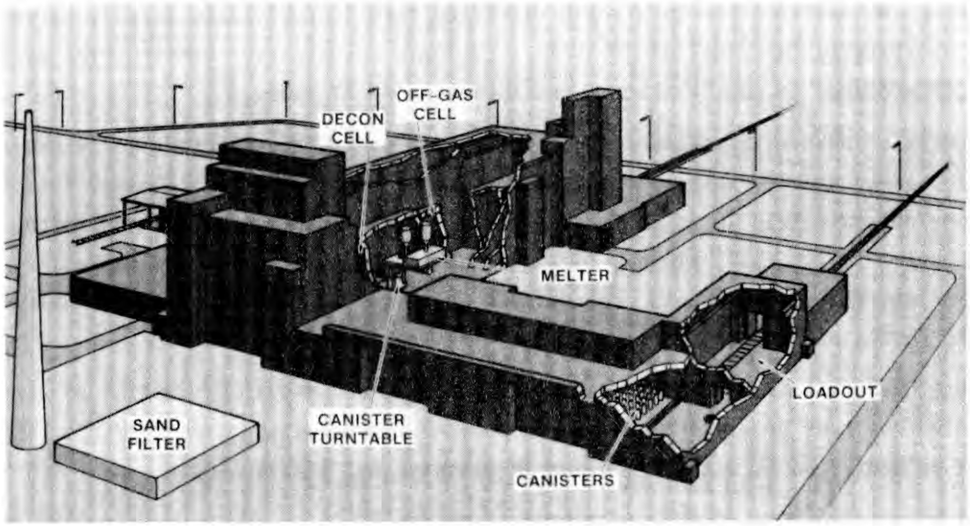

FIGURE C.1. Hanford Waste Vitrification Plant

The HWVP will use to the maximum extent the technology and design developed by other Department of Energy (DOE) programs. The Defense Waste Processing Facility (DWPF) and West Valley Demonstration Project (WVDP) programs each provide technology and design that will be used by the HWVP. The WVDP melter design is being considered for use by the HWVP while some other plant hardware systems such as canister decontamination, process off gas, and slurry feed system are based upon the DWPF technology.

The WVDP is a limited production facility designed for a 2-year campaign while the DWPF is a production facility designed for continued use. The HWVP production requirements are between these two production characteristics, and the Hanford waste chemical composition and heat load are different from those at West Valley and Savannah River. This means there will be a certain amount of "fine tuning" of the related technology and hardware systems to meet the Hanford needs.

As each of the WVDP and DWPF programs goes into operation, the HWVP program will also be in a position to take maximum advantage of this early operating experience.

The HWVP will vitrify blended waste feedstreams into a borosilicate glass matrix. These waste feedstreams are described in Section C.5.

The melter, which receives the slurried pretreated waste and glass-former mixture, melts this feed stock and yields a molten homogeneous product that upon cooling forms a glassy product. The molten glass product is transferred into canisters that will be temporarily stored at the HWVP site. The waste canisters will be transferred from the HWVP to a geologic repository when such a repository can receive these defense waste forms.

The building arrangement, as shown in Figure C.1, occupies a rectangular area approximately 87 m by 111 m on a site adjacent to B Plant in the 200 East Area. Height of this facility extends from approximately 22 m above grade to 5 m below ground level. This site has been selected for its economical advantages, convenient use of the existing B-Plant facilities, minimum interference with existing above-ground and underground structures, and because it is land already being used for industrial purposes.

The vitrification process for HWVP includes six major process systems:

- slurry feed
- melter/turntable
- canister closure
- process off gas
- canister decontamination
- canister storage.

The design of these six major process systems into process cells and ancillary support services for HWVP will be initially determined during conceptual design. The plan view of the process cells and support areas is shown in Figure C.2. The process area will include service galleries, operating galleries, a manipulator repair area, and support systems.

A canister storage area will provide storage for approximately 750 canisters that represent about 5 years of canister production. The canisters will be approximately 0.6 m in diameter and 3 m in length. The storage facility is of modular design to facilitate expansion, should a storage period longer than 5 years be necessary before shipment to the federal repository. Storage will be in sealed canisters stored in dry cells, with a negative pressure maintained through a monitored high-efficiency particulate air (HEPA) filter system. The vitrified waste form is a high-integrity waste form that can be stored safely in a monitored, filtered storage facility for a significant length of time if necessary.

C.2 RELATIONSHIP TO OTHER FACILITIES

Waste transfer lines connect the HWVP to the Hanford 200 East Area B Plant and tank farms where waste feed pretreatment operations will occur. Functions to be provided to the HWVP by B Plant include:

- pretreatment of the liquid high-level defense waste streams to provide an acceptable feed to the HWVP vitrification process
- treatment of chemical sewage
- treatment of nonradioactive condensate
- concentration of low-level waste

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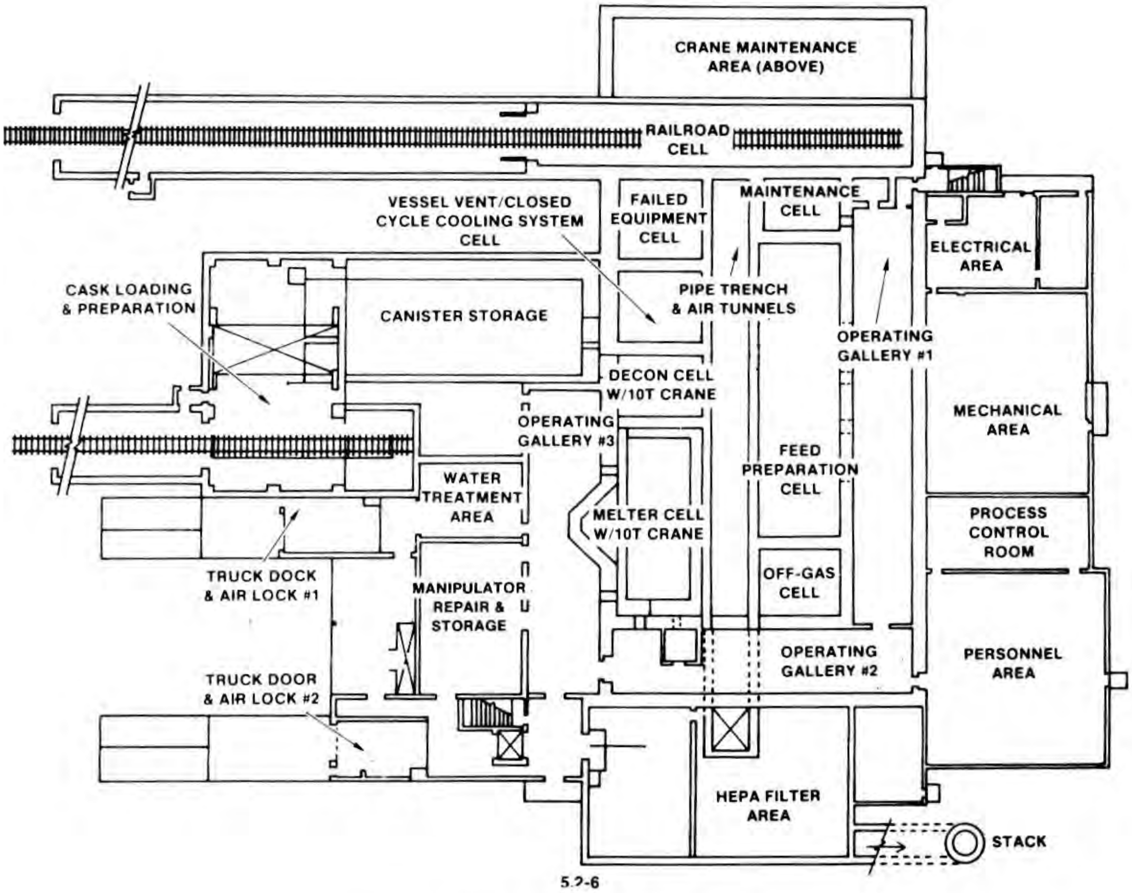


FIGURE C.2. HWVP Process Cell

- maintenance of facilities.

The relationship between the HWVP, B Plant, and the TGF (described in Appendix D) is shown in Figure C.3.

Wastes pretreated in B Plant are pumped into double-shell tanks in the 200 East Area for interim storage until they are pumped to the HWVP.

Additional interfaces with, and the functions provided by, other facilities include:

- analytical laboratories in the Hanford 200 West Area
- decontamination facilities at T Plant in the Hanford 200 West Area
- water, power, and steam from existing Hanford Site utilities[a]
- maintenance shops in the Hanford 200 East and West Areas for instrumentation support, replacement vessel fabrication, pump replacement or repair and other mechanical/electrical needs
- transfer pipelines from the tank farms to B Plant.

(a) These utilities may require expansion under separate projects to supply HWVP needs.

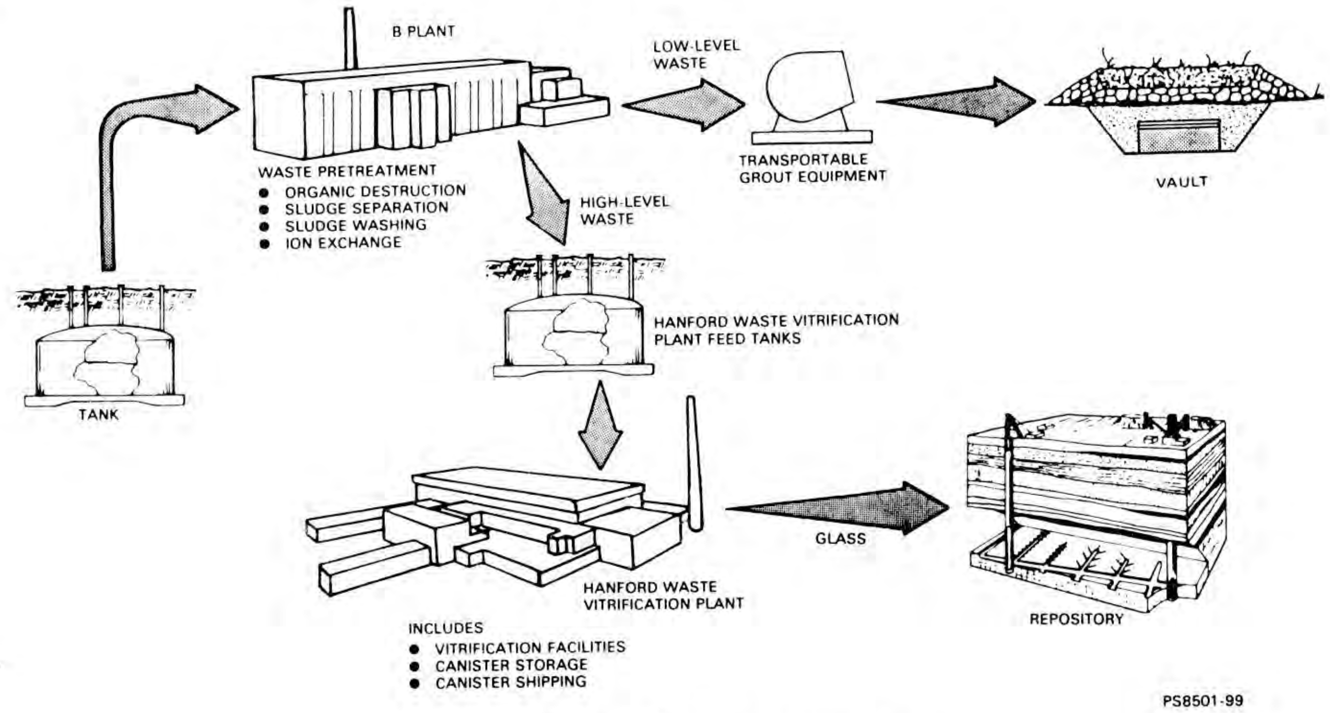


FIGURE C.3. B Plant Interconnections

C.3 PROCESS DESCRIPTION

The process steps required to vitrify pretreated waste feed in the HWVP are shown in Figures C.4 and C.5. The waste feed received at the HWVP will be concentrated by evaporation. The concentrated feed will be routed to the vitrification feed makeup system, and the evaporator overheads will be routed to the HWVP liquid-waste handling system for transfer to tank farms.

Following sampling and analysis, the concentrated feed will be batch-transferred to the feed makeup system. Glass formers will be added to the feed based on feed composition, concentration, and volume. The glass formers will be introduced as a slurry that has been prepared outside the process cell. The makeup feed slurry will be thoroughly blended prior to batch transfer to the melter feed system.

The melter feed system will transfer the made-up feed slurry to the melter at a nominal 100 L/hr-to-150 L/hr feed rate. The melter feed makeup system and melter feed system will be designed to allow continuous addition of feed to the melter while new feed slurry is prepared.

The waste vitrification melter system and the canister turntable are shown in Figure C.6. In the melter, the liquid waste is evaporated and the resultant waste and glass formers are converted to oxides that are melted into a vitreous material at about 1,150°C. The molten glass is joule-heated by an alternating current passed between electrodes that are in direct contact with the molten glass in the melter. Molten glass is transferred from the melter to a canister located in a turntable that is part of the canister handling system.

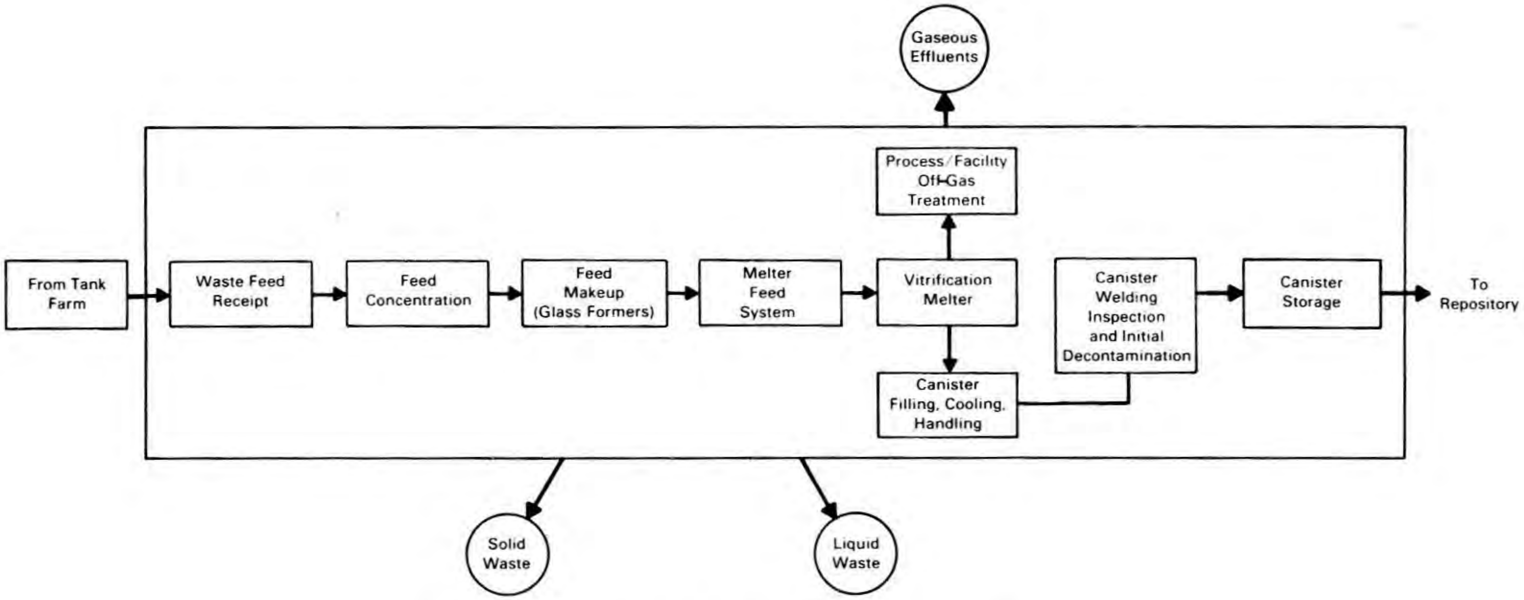


FIGURE C.4. Process Flow and Support Functions

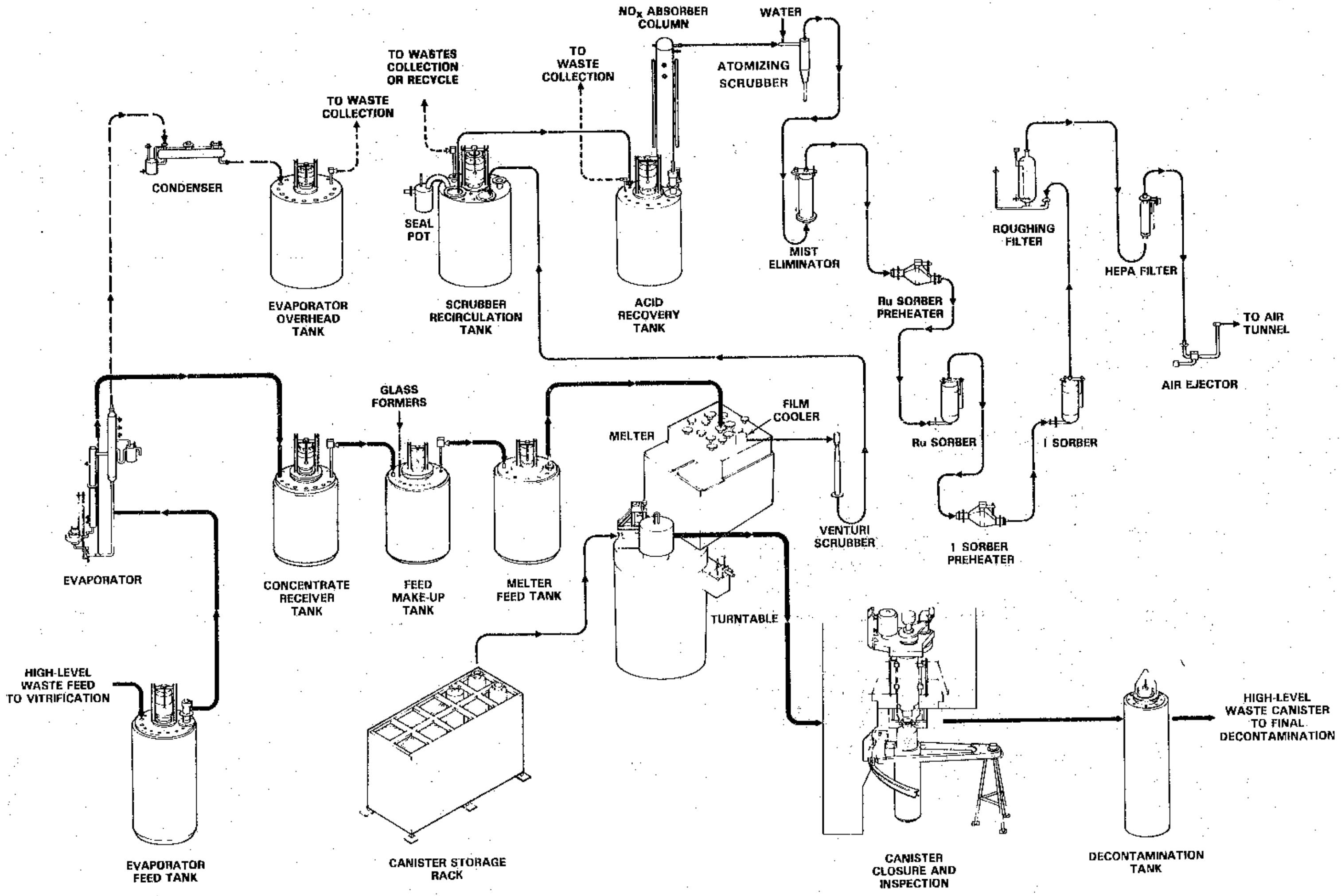


FIGURE C.5. Preconceptual HWVP Process

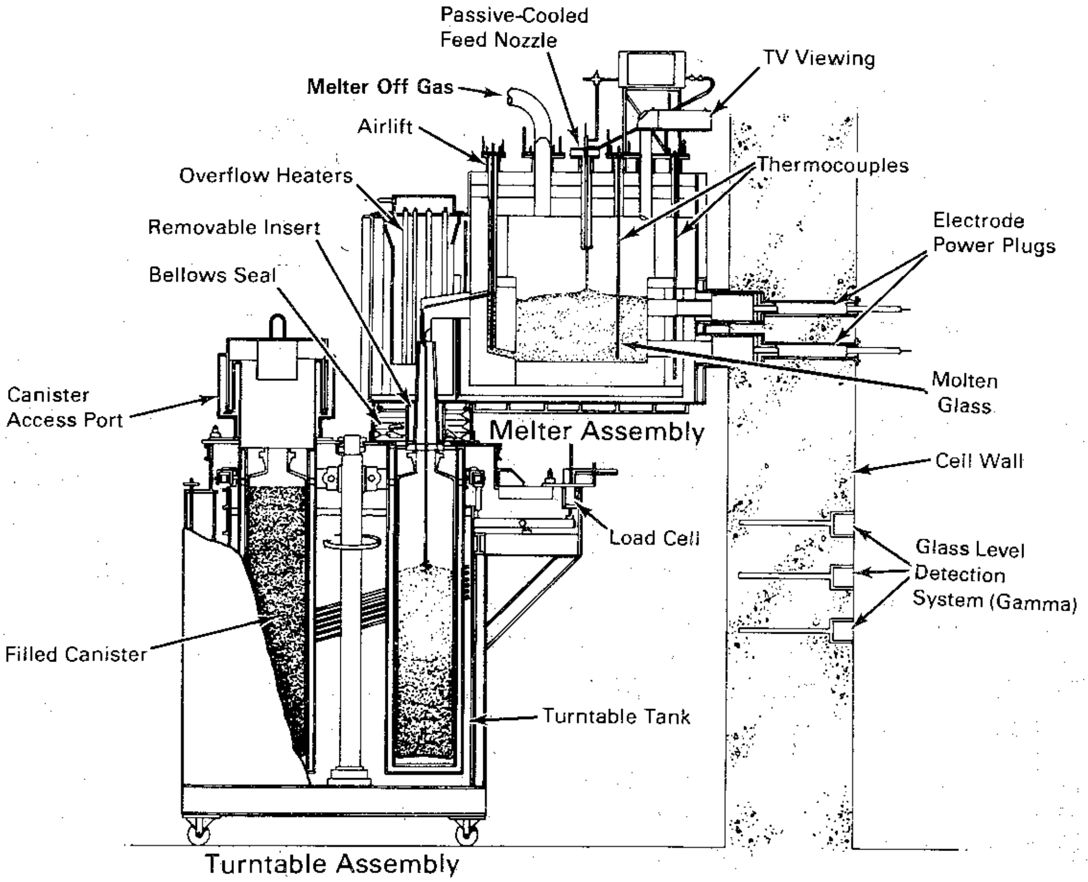


FIGURE C.6. **Waste Vitrification System**

The turntable system receives empty canisters, rotates canisters to the melter discharge port for filling, and holds the canisters during cooldown. Following cooldown, which will typically take between 36 and 72 hr, the filled canisters are removed from the turntable and transferred to a canister welding and inspection station where a lid is welded onto the waste canister and the weld is inspected. Following acceptance of the lid weld and before removal from the process cell, the canister is moved to a decontamination system for initial decontamination. After initial decontamination, the canister is transferred to a final decontamination station after which it will be transported to the facility's interim canister storage area. The HWVP design includes an in-cell crane to move canisters through the vitrification process within the melter and decontamination cell. The facility design also includes equipment to move canisters to various process cells and in and out of the interim canister storage area.

Off gas from the melter is routed to a process off-gas treatment system. The process off-gas system will remove condensibles, heat, particles, volatile radionuclides, and chemicals from the melter off gas prior to release to the facility ventilation exhaust system. The system will decontaminate the melter off gas so component concentrations at the stack, following filtration in the facility exhaust system, will meet release requirements such that resulting doses would be less than the standard in EPA's 40 CFR 61 subpart H of 0.025 rem/yr to the total body and 0.075 rem/yr to any organ of a member of the public. The process off-gas system also provides the vacuum operating conditions required by the melter and all components within the vitrification off-gas system. Condensate collected from the process off-gas system will be recycled to the waste feed concentration system and blended with incoming waste feed.

C.4 HWVP WASTE FORM

A variety of waste forms have been evaluated by DOE to select an acceptable waste form for high-level waste. Both borosilicate glass and crystalline ceramic waste forms were identified as having the highest potential for success for immobilization of high-level waste for disposal in a geologic repository (DOE 1982b). Evaluation showed that environmental effects resulting from disposal of the Salt Repository Project (SRP) HLW would not differ significantly between the two waste forms (DOE 1982b). Borosilicate glass was selected for the Defense Waste Processing Facility (DWPF) at SRP as the preferred waste form because process complexity, development requirements, and programmatic costs would be less for the borosilicate glass than for the crystalline ceramic. Furthermore, West Valley has selected borosilicate glass as the waste form for high-level waste stored at that facility (DOE 1983).

Although not yet explicitly defined, the expected properties of the HWVP borosilicate glass can be compared with the DWPF borosilicate glass. Favorable results from a comparison between the HWVP glass and the DWPF glass would then indicate that HWVP glass would be an acceptable waste form. Utilization of borosilicate glass at HWVP will minimize differences in waste forms and thereby simplify demonstration of compliance to geologic disposal criteria. Furthermore, HWVP will be able to rely upon the technology and information developed from the DWPF and West Valley experience.

Properties that help determine acceptability of the waste form are its leachability, mechanical strength, and thermal stability. These properties are compared to the SRP glass as follows:

- leachability - The leachability of the DWPF glass is described in the waste form selection Environmental Assessment (DOE 1982b) in terms of the normalized, steady-state leach rates of strontium, cesium, and plutonium. At the cited glass area-to-leachant volume ratio of 0.1 cm^{-1} and at steady state, solubility effects tend to constrain the leach rate of plutonium, which has a very low solubility, and of the alkaline-earth elements (e.g., strontium), whose concentrations in leachants tend to decrease with time (McVay 1980). Cesium, unlike other alkali metals, also has a decreasing concentration with time in static leach tests. Solubility effects will similarly constrain the leach rates of strontium, cesium, and plutonium from a HWVP glass (McVay 1980).

  Chick et al. (1984) found that the leachability of simulated nuclear waste glasses was insensitive to small variations in major glass components if the required processing properties such as melt viscosity and electrical conductivity were maintained. Because the major glass components of the HWVP glass are very similar to the major DWPF glass components, the durability of the HWVP waste form is expected to be comparable to the DWPF glass.

- mechanical strength - Wald et al. (1980) found that the tensile strength of a simulated HLW glass, designated PNL 76-68 (Mendel 1978), was significantly greater than for several other waste forms tested, including glass ceramic, cement, super calcine, and sintered materials. The standard deviation in the measurements was about 17%, due mainly to the distribution of defects in the waste form. Given the compositional similarity between HWVP glass and DWPF glass, the difference in mechanical strength between them will be small.
- thermal stability - Chick and Turcotte (1983) found that the effect of devitrification on leachability was not a strong function of glass composition, except for a high zinc glass. Specifically, a high iron glass, designated PNL 76-68 (Mendel 1978), and a rare earth glass, designated PNL 77-260 (Slate et al. 1981), had similar increases in leach rate after maximum devitrification was induced, and in each case the increase was about a factor of three.

The same factor-of-three increase in leach rates was also observed in DWPF glass with 40 wt% devitrification (Jantzen, Bickford and Karrater 1984). The time-temperature-transformation (TTT) curves associated with the devitrified DWPF glass are very similar to the TTT curve generated for the HWVP glass (Mitchell 1986). Therefore, given the similarity of the compositions and TTT curves of the DWPF and HWVP waste forms, the thermal stability and resulting durability of the two waste forms should be comparable.

The decay heat in the HWVP canister is a function of the following factors:

- glass content in the canister
- waste content in the glass
- waste composition
- "age" of the waste.

The calculated maximum canister heat generation rate, based upon the reference HWVP waste composition, is 1,400 watts per canister (Mitchell 1986). The 490°C glass transition temperature reduced by a safety margin of 100°C (i.e., 390°C) is the current selected upper waste form temperature limit at the time of shipment. For a reference thermal output of 1,400 watts per canister, the in-cell canister centerline temperature will be about 130°C after initial cooldown. Moreover, canister thermal outputs of up to 6,500 watts can be achieved before the canister centerline temperature exceeds 390°C. Draft criteria for commercial HLW canister acceptance at the Basalt Waste Isolation Project (BWIP) refer to a maximum decay heat of 2500 watts per canister (Rockwell 1983). The maximum anticipated HWVP canister decay heat is below this commercial limit.

C.5 WASTE FEEDSTREAMS

Waste streams will be routed from various storage areas at Hanford to B Plant for pretreatment before being sent to the HWVP for vitrification. In the B Plant, soluble salts will be removed and organic complexants will be destroyed. Waste stream pretreatment will also remove the low-level waste fraction separated in the sludge washing step. The remaining high-activity wastes will be blended and accumulated in HWVP feed tanks that are below-ground, double-shell storage tanks in the 200 East Area. The removed low-level wastes and soluble salts will be routed for disposal in grout. These pretreatment steps are required prior to processing in the HWVP to concentrate the feed. The pretreatment process will yield a waste feed that will provide maximum waste loading in the glass. Wastes to be pretreated include:

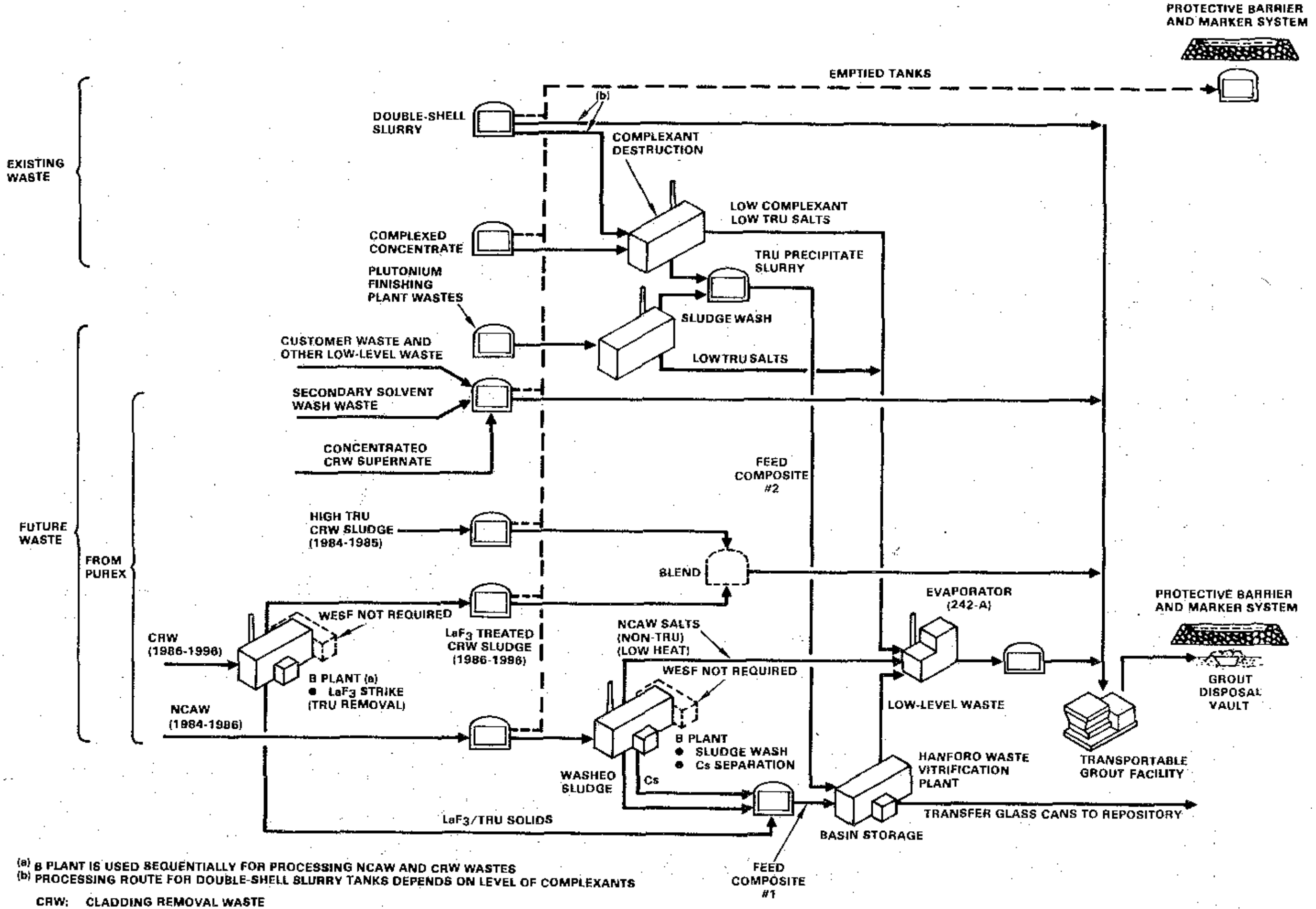
- neutralized current acid waste (NCAW)
- TRU solids from cladding removal waste (CRW)
- Plutonium Finishing Plant (PFP) wastes
- double-shell slurry (DSS)
- complexed concentrate (CC).

C.5.1 Waste Pretreatment Operations

Pretreatment operations, as well as glass and grout immobilization operations, are presented in Figure C.7 which indicates two segregated feedstreams (blended NCAW-CRW and complexant TRU solids from concentrates) entering the HWVP. Current planning assumes that these two streams will be vitrified in separate operating campaigns lasting several years each.

The first feed composite likely to be vitrified, the NCAW-CRW blend, is composed primarily of TRU-bearing sludge from neutralized current acid waste (NCAW) generated in the PUREX process. This sludge will have been washed in B Plant to keep soluble salts (Na^+, AlO_2^-, $SO_4^=$, NO_3^-) from entering the vitrification process and, hence, minimizing the volume of glass requiring disposal. As indicated in Figure C.7, two other relatively small waste streams will be blended with the washed NCAW sludge. These two waste streams consist of 1) LaF_3-TRU solids in a slurry resulting from the reaction of LaF_3 with PUREX cladding removal waste for the purpose of removing plutonium and americium contaminants, and 2) cesium crude concentrate in a concentrated solution of sodium and cesium carbonates resulting from the separation (via ion-exchange processing in B Plant) of radiocesium from the NCAW supernate and sludge wash solutions.

The second vitrification feed composite is expected to be a blend of complexed concentrate (CC) wastes and the TRU solids separated from future Plutonium Finishing Plant (PFP) wastes. Before blending, the CC wastes will be treated to destroy the organic complexants. Alternatives for complexant destruction include ozonization, peroxide oxidation, or pressurized aqueous combustion. A second option being investigated is to remove the TRU through solvent extraction. A 5-year to 10-year development period is expected for the solvent extraction or organic complexant destruction method selected. The resulting waste blend is, therefore, not projected to be generated until after 1995 and, thus, may be scheduled as the second major vitrification campaign.



[a] B PLANT IS USED SEQUENTIALLY FOR PROCESSING NCAW AND CRW WASTES
[b] PROCESSING ROUTE FOR DOUBLE-SHELL SLURRY TANKS DEPENDS ON LEVEL OF COMPLEXANTS

CRW:	CLADDING REMOVAL WASTE
PUREX:	PLUTONIUM-URANIUM REDUCTION/EXTRACTION
WESF:	WASTE ENCAPSULATION AND STORAGE FACILITY
NCAW:	NEUTRALIZED CURRENT ACID WASTE

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FIGURE C.7. Pretreatment Operations

C.5.2 Composite Waste Characteristics

Total oxide mass and composition of major chemical components for the first composite HWVP feedstream (NCAW-CRW sludge + Cesium concentrate) are given in Table C.1. Volume projections for the composite waste are similar to volumes expected for unwashed, aged NCAW (830 L/MTU).

TABLE C.1. Composite 1 Feedstream (NCAW-CRW Sludge + Cesium Concentrate)(a)

Component	Average Wt. Percent Waste Oxides Scenario A(b)	 Scenario B(c)
Al_2O_3	20	21
Na_2O	11	11
Fe_2O_3	42	30
SiO_2	2.6	3.0
$SO_4^{=}$	0.86	0.33
F^-	0.88	0.38
$FPO_{1.25}$	9.2	12
TOC(d)	2.8	--
Cr_2O_3	5.1	4.4
NiO	2.2	2.6
ZrO_2	2.4	3.7
Oxides (Ca, Be, Cu, Mg, Mn, Mo)		4.3
Nd_2O_3	0.22	~0
La_2O_3	0.63	~0
PuO_2	0.005	~0
Am_2O_3	0.015	~0
NpO_2	0.11	~0
U_3O_8	0.72	7.4
Total t waste oxides (includes F^- and TOC)	230(e)	210(e)

(a) Rockwell 1985.

(b) Scenario A: PUREX runs ferrous sulfamate flowsheet from late 1983 to early 1996.

(c) Scenario B: PUREX runs ferrous sulfamate flowsheet from late 1983 to early 1986. Hydroxylamine nitrate flowsheet from early 1986 to early 1996. Waste oxides contribution from CRW-TRU concentrate stream assumed negligible due to small fraction of CRW requiring treatment.

(d) Total organic carbon (TOC).

(e) Waste oxides in glass are less than these values due to volatilization of F^- and TOC.

Comparable data for the second HWVP composite feedstream (TRU solids from complexant concentrates plus PFP TRU concentrate) are given in Table C.2.

It should be noted that the waste compositions described in Tables C.1 and C.2 are not HWVP design base feed compositions, but rather they represent the average of several feedstreams. Two possible compositions are shown for composite 1 feedstream. This reflects

TABLE C.2. **Composite 2 Feedstream** (Complexant TRU Concentration and PFP TRU Concentrate)[a]

Component	Average wt% Waste Oxides
Fe_2O_3	63.0
Al_2O_3	18.0
MnO	4.3
CaF_2	4.3
MgO	3.4
Na_2O	3.1
NaF	<0.1
Na_2SO_4	<0.1
PuO_2	<0.1
Am_2O_3	<0.1
U_3O_8	3.8
Total Waste Oxides	280

uncertainties associated with future PUREX treatment chemistries and the potential need to modify the PUREX chemistry to reduce sulfate anion entering the glass melter.

C.6 RESOURCE REQUIREMENTS FOR HWVP

Major resource requirements expected for construction of HWVP are listed in Table C.3. This table gives total requirement estimates for personnel, water, energy, and materials. Annual personnel, water, energy, and materials requirements for the operation of HWVP are given in Table C.4.

No natural resources are significantly limited or stressed by the requirements of the HWVP facility.

C.7 RADIOLOGICAL IMPACTS AND EMISSIONS

Dose commitments to the general population and to the maximally exposed individual from routine operations of the HWVP are presented in Tables C.5 through C.8. Doses are much less than those permitted by EPA's 40 CFR 61, 0.025 rem/yr to the total body and 0.075 rem/yr to any organ of a member of the public. Impacts were calculated for existing and future tank waste over the operational period. Calculated doses are from estimated exposure during feed preparation and vitrification processes and are presented for 1-year and 70-year exposure periods. For comparison, the offsite population near Hanford receives about 25,000 man-rem per year from naturally occurring sources.

Annual radiological emissions during the operational period are shown in Table C.9. Projected radiological emissions would not exceed applicable guidelines. These emissions, averaged over the facilities' design lives for long-term dose calculations, are based on

TABLE C.3. Total Resource Requirements for Construction of HWVP[a]

Resource Requirements	
Personnel, man-yr	3,100
Land, ha	1
Water, m^3	15,000
Energy Consumed:[b]	
Electrical, GWh	9
Propane, m^3	16,000
Diesel, m^3	3,200
Gasoline, m^3	2,100
Materials:	
Concrete, m^3	33,000
Steel, t	6,300
Stainless Steel:	
Iron, t	900
Chrome, t	200
Nickel, t	100
Copper, t	300
Lumber, m^3	3,600
Argon, m^3	1,200

(a) Material requirements derived directly from RHO-SD-461-PCR-001 (Shah 1983).

(b) Energy related resource values and associated emissions, such as CO or particulates, are scaled from project capital costs ($281M assumed) using standard factors. These values may be refined as the HWVP design progresses.

preliminary flowsheets and engineering assumptions regarding radionuclide partitioning (Shah 1983; Rockwell 1985). Emissions of specific radionuclides shown in Table C.9 may be higher or lower, depending on final plant design and preprocessing efficiencies. Also, peaks and valleys around these average annual values will occur over the plant lifetime because the values assumed for calculations are based on an average waste stream, rather than on maximum values (as would be used for shielding design or permitting purposes). In no case would emissions exceed regulatory limits. That is, no member of the public would be exposed to greater than the EPA standard of 0.025 rem/yr total body (40 CFR 61 subpart H).

TABLE C.4. Annual Resource Requirements for Operation of HWVP

Resource Requirements	
Personnel, man-yr	86
Water Consumed, m^3	2.6×10^6
Energy Consumed:	
Electrical, GWh	13
Propane, m^3	520
Diesel, m^3	0(a)
Coal, t	2,100
Materials:	
Steel, t	60
NaOH, t	24
Glass Frit, t	180

(a) Canister transportation not included.

TABLE C.5. Population Total-Body Dose Commitments (man-rem) from Feed Preparation and Vitrification of Existing Tank Waste

Pathway	Exposure Period 1 yr	Exposure Period 70 yr
Air Submersion	4.0×10^{-10}	2.9×10^{-9}
Inhalation	1.0×10^{-6}	5.0×10^{-6}
Terrestrial (air paths)	2.0×10^{-6}	1.0×10^{-4}
Total Doses	3.0×10^{-6}	1.0×10^{-4}

TABLE C.6. Maximum Individual Total-Body Dose Commitment (rem) from Feed Preparation and Vitrification of Existing Tank Waste

Pathway	Exposure Period 1 yr	Exposure Period 70 yr
Air Submersion	3.2×10^{-15}	1.6×10^{-14}
Inhalation	8.3×10^{-12}	4.2×10^{-11}
Terrestrial (air paths)	3.5×10^{-11}	1.8×10^{-9}
Total Doses	4.3×10^{-11}	1.8×10^{-9}

TABLE C.7. Population Total-Body Dose Commitments (man-rem) **from Feed Preparation and Vitrification of Future Tank Waste**

Pathway	Exposure Period 1 yr	 70 yr
Air Submersion	1×10^{-5}	4×10^{-5}
Inhalation	3×10^{-3}	9×10^{-3}
Terrestrial (air paths)	1×10^{-2}	3×10^{-1}
Total Doses	2×10^{-2}	3×10^{-1}

TABLE C.8. Maximum Individual Total-Body Dose Commitment (rem) **from Feed Preparation and Vitrification of Future Tank Waste**

Pathway	Exposure Period 1 yr	 70 yr
Air Submersion	1×10^{-10}	3×10^{-10}
Inhalation	2×10^{-8}	7×10^{-8}
Terrestrial (air paths)	2×10^{-7}	4×10^{-6}
Total Doses	2×10^{-7}	4×10^{-6}

TABLE C.9. Annual Routine Radiological Emissions from Operation of HWVP

Radionuclide	Emissions, Ci/yr
^{3}H	0.4
^{14}C	1.0
^{90}Sr	2×10^{-4}
^{106}Ru	1×10^{-3}
^{129}I	7×10^{-6}
^{137}Cs	2×10^{-4}
^{144}Ce	6×10^{-4}
$^{239,240}Pu$	2×10^{-8}
^{241}Am	1×10^{-6}

C.8 NONRADIOLOGICAL EMISSIONS

Nonradiological emissions from construction and operation of HWVP are shown in Tables C.10 and C.11, respectively. These emissions reflect a case where the HWVP is processing NCAW sludge (plus cesium concentrate) at an annual glass production rate of 170 to 240 t glass per year (i.e., 30 kg to 45 kg glass/hr, 60% time operating efficiency with an

TABLE C.10. Total Nonradiological Emission Estimates for Construction of HWVP

Pollutant	Emissions[(a)]
Particulates, t	24
SO_x, t	18
CO, t	300
HC, t	36
NO_x, t	240
Fugitive Dust, t	130
Thermal, TJ	810

(a) Emissions are derived from energy resource consumption estimates. These values may be refined as the HWVP design progresses.

TABLE C.11. Annual Nonradiological Emission Estimates for Operation of HWVP

Pollutant	Emissions[(a)]
Particulates, kg	930
SO_x, kg	16,000
CO, kg	1,400
CO_2, kg	6,000
Hydrocarbons, kg	360
NO_x (NO_2), kg	6,200
F_2, kg	21
Thermal, TJ	190

(a) Calculation based on 240 t of glass/yr for HWVP. These values will be refined as the HWVP design progresses.

upward design envelope of 100 kg/hr) (Rockwell 1987). For perspective, this represents between 2,300 MTU and 3,450 MTU of N Reactor fuel processed through PUREX. Note that these emissions, averaged over the facilities' design lives, are based on preliminary flowsheets and engineering assumptions (Shah 1983; Rockwell 1985). Emissions of specific pollutants shown in this appendix may be either higher or lower, depending on final plant design, process flowsheets, preprocessing efficiencies, waste streams, and melter throughput. Also, peaks and valleys around these average annual values will occur over the plant lifetime as the emissions shown are based on continuous operations. In no case would emissions exceed applicable state and federal regulations or DOE guidelines. The HWVP will be designed and operated in full compliance with applicable hazardous waste regulations.

C.9 COST REQUIREMENTS FOR HWVP

The total design and construction cost for the HWVP facility is expected to be $920 million, escalated to the midpoint of design and construction activities (about 1993). This cost translates into $700 million in 1987 dollars. (The design and construction costs include a contingency factor that must be removed before further escalations or adjustments are made, and then the contingency factor must be reapplied.) The capital equipment (not related to construction) costs are estimated at $13 million in 1987 dollars. Prestartup expenses, including technology support and operations expenses, are estimated to be $225 million in 1987 dollars. The annual operating cost for facility operation is estimated to be $40 million in 1987 dollars. The cost to decontaminate and decommission the HWVP at the end of operation is estimated at 20% of construction cost.

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APPENDIX D

TRANSPORTABLE GROUT FACILITY

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APPENDIX D

TRANSPORTABLE GROUT FACILITY

The purpose of the Transportable Grout Facility (TGF) is to make a cementitious grout waste form of wastes designated for disposal in near-surface disposal sites located in the 200 East Area. A grouted waste slurry would be formed by blending Hanford defense liquid wastes with grout-forming solids. The grout slurry would be pumped into near-surface disposal vaults with greater than 5 m of cover, where it would solidify into large monoliths. Cement-based solidification materials have been used for radioactive wastes since the beginning of the nuclear industry in the United States (Kibbey and Godbee 1980) and are, in fact, used world-wide. Cement-based processes of solidification/immobilization are well documented (IAEA 1968; Spitsyn 1968, EPA 1978, 1980; Kibbey and Godbee 1980; EPRI 1983; Jolley et al. 1986).

This appendix describes the TGF currently planned to be used for disposal of designated wastes at Hanford in the reference disposal alternative described in Chapter 3 of the EIS. Topics addressed include the facilities (Section D.1), their relationship to other Hanford facilities (D.2), the grouting process (D.3), waste feedstreams (D.4), resource needs (D.5), nonradiological emissions (D.6), radiological impacts (D.7), and costs (D.8). Figure D.1 is a schematic of the grout process.

D.1 FACILITIES DESCRIPTION

The TGF would consist of two facilities: 1) the Dry Materials Receiving and Handling Facility (DMRHF), where the grout-forming solids would be blended, and 2) the transportable grout equipment (TGE) modules, where the blended solids would be mixed with liquid waste and from which the resulting slurry would be pumped to the disposal site (Figure D.2). A 3,800-m^3 underground waste storage tank would serve as the liquid feed tank for the grout process.

The DMRHF would include stationary equipment for storing and blending grout-forming solids such as Portland cement, blast-furnace slag, fly-ash (waste product from coal-burning power plants), and clays. Equipment associated with the DMRHF would include:

- rail car unloading station
- storage silos (for incoming and blended solids)
- solids conveyers
- solids blending system
- truck loading station (for transport of blended solids to the TGE).

All DMRHF equipment would be operated in a nonradioactive mode. Trucks would transport the blended grout-forming solids from the DMRHF to the TGE.

The TGE would consist of modules to mix blended solids with liquid wastes from current and future operations, including feedstream pretreatment wastes from B Plant, and wastes from

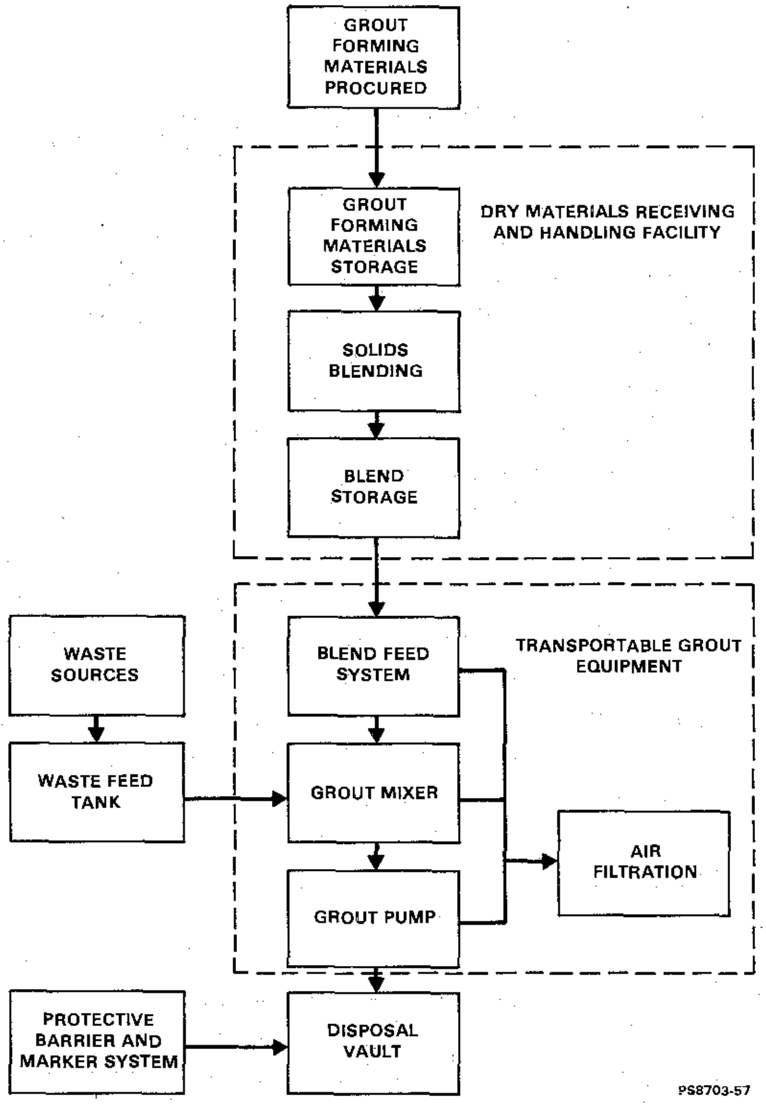

FIGURE D.1. Schematic of Grout Process

the Hanford Waste Vitrification Plant (HWVP, see Appendix C). HWVP wastes will consist of wastes from canister decontamination, drying of feed material, and off-gas treatment. The resulting slurry would then be pumped into the disposal sites. The system of modules would produce grout slurries safely and efficiently. The TGE would include:

- blended solids feed system (for providing solids to the grout mixer)
- grout mixing and pumping system
- off-gas exhausters and filters (for removing contaminants from process off gas)
- tanks for additives and decontamination solutions
- standby electric generator
- control room.

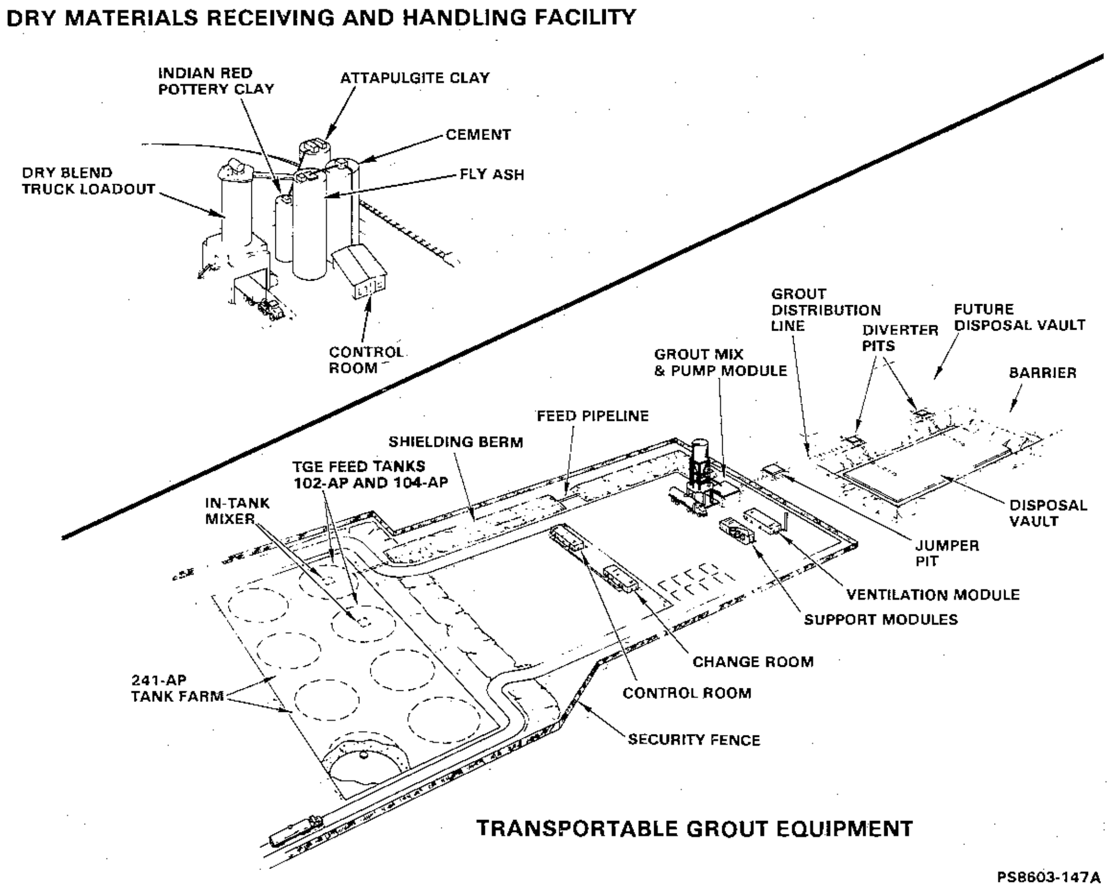

FIGURE D.2. Near-Surface Disposal of Grout

The grout slurry would be disposed of in prepared disposal vaults designed to accommodate the volume of grout (~5,300 m^3) expected from processing all liquid waste in the 3,800-m^3 feed tank. The conceptual vault design is shown in Figure D.3. The vault will be designed to conform to the technical requirements of the Resource Conservation and Recovery Act (RCRA) and Washington Administrative Code (WAC 173-303) for disposal of hazardous wastes. The grout process will meet the permit requirements for WAC 173-303.

D.2 RELATIONSHIP TO OTHER FACILITIES

Waste to be grouted originates from the operation of several existing and future facilities. These wastes will be processed through B Plant for pretreatment as required. Cladding removal waste (CRW) and neutralized current acid waste (NCAW) are generated by PUREX. Double shell slurry (DSS) and complexed concentrate (CC) would be obtained from underground storage tanks. Customer wastes are low-level wastes generated by the Plutonium Finishing Plant, B Plant, T Plant, S Plant, the PUREX Plant, N Reactor, and 300-400 Area facilities. In the future, customer wastes would include low-level liquid wastes from the HWVP. Customer wastes that have been mixed with other tank wastes are included within the scope of the EIS, while

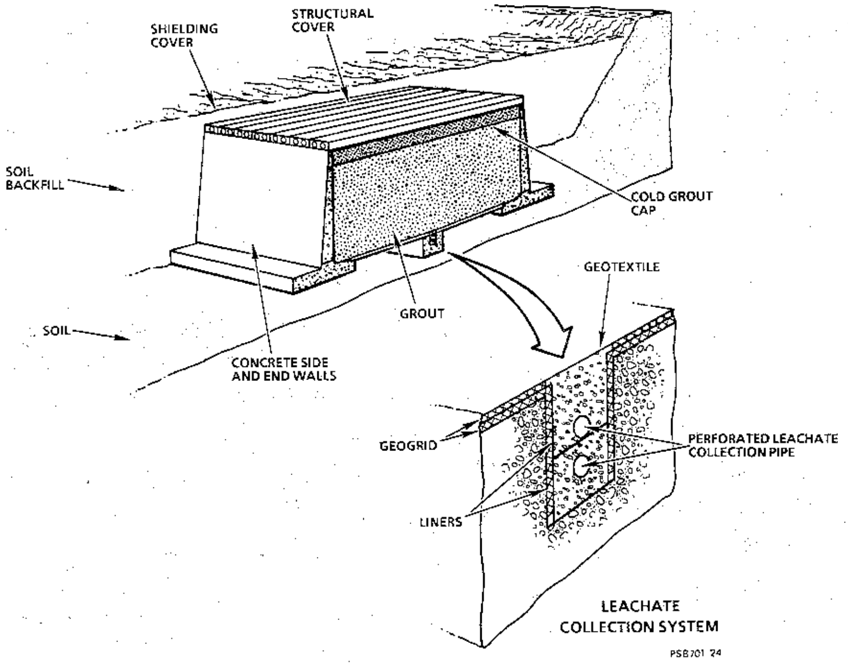

FIGURE D.3. **Grout Disposal Vault Concept**

those customer wastes stored in a separate tank are addressed in other evaluations (DOE 1986a). The NCAW supernate would be processed in B Plant to remove ^{137}Cs and TRU-bearing sludge. The ^{137}Cs and sludge would become feed for the HWVP while the remaining waste stream would be concentrated in the 242-A evaporator before it is grouted. The removal of DSS, CC, and CRW from double-shell tanks would provide space for storage of future waste or for disposal of solid wastes that require greater confinement. The grout feeds would be routed to the grout feed tank (initially Tank 102-AP) through an existing system of encased pipe, diversion boxes, and vaults. Most laboratory requirements for the grouting operation would be met in S Plant. Decontamination and maintenance of failed equipment would be conducted in T Plant or the 242-A evaporator building.

D.3. PROCESS DESCRIPTION

The process for making grout is depicted in Figure D.1. Figure D.2 shows the grout disposal facility. This process begins with development of a grout formulation and procurement of grout-forming solids. Next, the solids are blended in the dry state and then are mixed with the liquid waste to form a grout slurry. The grout is then pumped to a disposal vault, where it solidifies (Figure D.3).

D.3.1. Grout Formula Development

The formula developed must be compatible with safe and efficient operation of the grout process and must ensure effective immobilization of the waste for long-term public safety. A typical grout mixture might consist of Portland cement, a pozzolan (such as fly ash), clay, the liquid waste and probably one or more chemical admixtures to enhance specific characteristics of the grout (Rockwell 1985; Tallent et al. 1986). The proportions of each component can be adjusted to meet various processing and performance requirements.

Processing requirements include physical and rheological characteristics such as critical flow rate, gel strength, and frictional pressure drop. These requirements are affected by the amount and type of grout formers used, the presence of entrained air and admixtures, and the mixture's water content. These characteristics affect the ease of mixing, pumping and emplacing of the grout mixture.

Long-term grout performance depends on such physical and mechanical properties as density, porosity, compressive strength, thermal expansion, thermal conductivity, and leachability (Young 1982). In addition, the environment in which the waste-form material would be placed must also be considered (Roy et al. 1980). To formulate optimal material for a specific site, the probable effects of exposure to the surrounding conditions throughout the required life span must be evaluated. Changes in the grout after curing are expected to occur slowly and might affect performance. Long-term containment of wastes would be enhanced by the Hanford Site's arid climate, which limits the mobility of the hazardous chemical and radionuclide constituents in the wastes.

Grout formulas would be tailored to each type of waste to ensure that a durable, safe waste form is created. Tests will be conducted to provide data required to improve assessments of the operational and long-term performance characteristics of each type of grout (DOE 1986b). If it is not possible to develop a grout formula adequate for near-surface disposal of a particular waste, several options exist: 1) the waste stream may be treated to remove or neutralize the waste component(s) of concern, 2) the waste stream may be converted to borosilicate glass in the Hanford Waste Vitrification Plant, or 3) the waste stream may be converted to another solid form, such as drummed concrete, and disposed of at a federal waste repository.

D.3.2 Feed-Tank Filling

Grouting would be conducted in scheduled campaigns that are determined by the capacity of the 3,800 m^3 waste-feed tank and by the capacity of the grout facilities (nominally 0.2 m^3 of grout per min). After initial startup operations, there would be on the average about five grout campaigns per year, each lasting about 1 month. About 3,800 m^3 of waste feed would be mixed with the grout formers to produce a total grouted waste volume of about 5,300 m^3 per campaign. At a rate of five campaigns per year, it would take about 20 years of operations to grout the total volume of the candidate feed waste streams. The resulting grouted waste volume would be about 4.9×10^5 m^3.

A campaign would begin with the filling of the feed tank with liquid wastes that have been determined to be, through prior testing, acceptable for grouting. The contents of the tank would be mixed to ensure that the chemical composition falls within predetermined bounds. A sample of the waste would be tested before grouting to ensure that the waste and resultant grout properties fall within acceptable limits.

D.3.3 Solids Procurement and Blending

Grout formers such as Portland cement, blast furnace slag, fly ash, and clay would be procured in large quantities (3.6 million kg) at the beginning of each grouting campaign and would be transported to the DMRHF in covered hopper railcars or by truck. These dry, powdery solids would be pneumatically transferred from the railcars to assigned storage silos. The storage silos have the capacity to store the dry materials for only one-half of a campaign. Blending and unloading operations for the second half-campaign would be conducted concurrently with the grout mixing and pumping operation.

The grout formers would be weighed individually into a solids-blending system operating on a batch basis. The batch of blended solids would be pneumatically transferred to a storage silo, where it would await transport to the TGE. As needed, the blended grout formers would be transported by covered hopper trucks to the TGE. There the grout formers would be unloaded pneumatically into the blended solids feed system which is connected to the transportable grout mixing module. (The solids feed system holds about 4 hr of solids at a nominal grout production rate of 0.2 m^3/min.)

Samples of the individual grout formers and of the grout formers blend would be tested routinely for compliance with performance specifications. Material balances would be maintained to ensure that grout formers are blended in desired ratios. Samples may be obtained from hopper cars, during transfer operations, and from the silos.

D.3.4 Grout Mixing and Pumping

Liquid waste would be mixed with the dry, blended solids in the TGE and pumped to the disposal facility. Weighing devices with redundant instrumentation would provide assurance that the solids blend is added at the prescribed rate. Any significant variances in the feed rates would result in a shutdown of the process until the problem is resolved. The prescribed rate, based on the waste liquid flow rate, would be typically 1 t of solids per m^3 of waste. The volume of grout would be approximately 1.3 times the volume of waste feed.

Liquid waste would be continuously pumped from the feed tank through an encased pipe to the grout mixer. Redundant flow meters provide similar assurance that wastes are being delivered at the prescribed rate. Again, any significant variance in the flow rate would result in a shutdown of the process until the problem is resolved. Liquid waste and blended solids would be added at one end of the continuous mixer. The rotating action of the mixer paddles would mix the liquid and solids into a homogeneous slurry which would be discharged by gravity at the opposite end of the mixer. If required, chemicals might also be added to the mixer and/or the waste feed tank to control foaming, grout viscosity, and grout hardening rates.

The grout slurry would flow into the intake end of a progressive-cavity pump. The grout would then be pumped through an encased pipe to a preconstructed, lined concrete vault where the grout slurry would cure to form a solid monolith. Nominal expected pumping pressure would be about 250 psi. Before it was filled, the vault would be covered with structural and shielding covers to protect operating personnel and prevent the release of contaminants.

High-efficiency particulate air (HEPA) filters, connected to the mixing/pumping module, would be used to protect against the release of airborne contamination by pulling air from contaminated equipment. Continuous air monitoring would be conducted to detect filter failure. Activities within the mixing/pumping module will be monitored by television and liquid-level sensors to permit early detection of process problems. Small radioactive spills and leaks that may occur would be contained within the modules. Because of these small spills and leaks, equipment would be periodically flushed and decontaminated. Decontamination solutions used to clean up spills and leaks would be collected, mixed with grout formers and similarly disposed of as grout.

Components of the TGE would eventually wear out and require maintenance or replacement. All waste-processing equipment in contact with the grout would be designed for remote replacement, but not for extensive remote maintenance. Difficult maintenance activities, such as replacement of shaft seals, will require that failed equipment be replaced or transported to T Plant or the 242-A evaporator building for decontamination and maintenance.

In the event of electric power failure during operation, a standby electric generator would provide emergency power to all equipment required for a safe and orderly shutdown of the process.

D.3.5 Solidification

Initial grout solidification (curing) might take up to 4 weeks, depending on the grout formulation and the waste being grouted. After curing was completed, a nonradioactive grout slurry would be pumped into the vault to completely fill the vault and seal the radioactive grout surface. At a later time, a protective barrier and marker system would be placed over the disposal vault to protect against intrusion by plants and animals. The protective barrier and marker system is described in Appendix M. This system may be modified for grouted wastes that pose a very low health risk.

The grout waste form, cured and covered with a protective barrier, can be expected to isolate the waste in the arid Hanford environment for a long time. Evidence of the longevity of grout can be found in ancient structures in Europe (Roy and Langton 1983). Cement-based structures more than 3,000 years old are still standing. These ancient analogs provide additional confidence that grouts can be formulated to last for long time periods.

D.4 WASTE FEEDSTREAMS

Grout feedstreams include the low-activity fraction of wastes from current Hanford operations and future low-activity fraction of wastes from pretreatment of waste feedstreams destined for the HWVP. Wastes are immobilized in hydraulic cement-based grout in three ways: 1) chemical combination or adsorption with the cement constituents to form hydrated compounds, 2) containment in the pore structure of the grout matrix, and 3) mechanical binding of solid particles by the grout matrix. Some wastes may be difficult to solidify in a grout form due to chemicals in the waste that interfere with grout hardening. In that case, special additives may be required to ensure quality grout. Candidate waste feedstreams for grouting are shown in Table D.1. Customer waste (CW) is a blend of low-activity liquid wastes from several sources. Cladding removal waste, double-shell slurry, neutralized current acid waste and complexed concentrate would be treated to separate the soluble salts from the sludge fractions, destroy the organic complexants, and separate out the TRU, if such steps are required to produce acceptable grouts. The high-activity and transuranic (TRU) tank waste fractions would be feedstreams for the HWVP, while the low-activity fractions would be feedstreams for grouting.

TABLE D.1. Candidate Wastes for Grouting

Waste	Sources
Customer Waste (CW)	B Plant, Z Plant, T Plant, S Plant, PUREX, HWVP, N Reactor, 300-400 Areas
Cladding Removal Waste (CRW)	N Reactor fuel, Shippingport fuel processing
Double-Shell Slurry (DSS)	Existing waste in double-shell tanks
Neutralized Current Acid Waste (NCAW)	Newly generated waste from PUREX fuel processing
Complexed Concentrate (CC)	Existing waste in double-shell tanks

Grouting would be performed on a campaign basis with little blending of waste feeds. In this way, grout formulations could be tailored to each feedstream to optimize the grout's processability as well as its chemical and structural properties for inhibiting the release of nuclides, thereby improving long-term performance.

The compositions of some of the first wastes that would be grouted are shown in Table D.2. The estimated volumes are given in Table D.3. Two cases, Scenario A and Scenario B, are presented in Table D.3. Scenario A represents no volume reduction of waste feedstream and would result in the largest volume of grouted waste produced. Scenario B includes concentration of waste reducing the total volume to be grouted.

Initial grouting is currently scheduled to begin in 1988 with low-level wastes not within the scope of this EIS, resulting from decontamination operations at N Reactor (DOE 1986a). Impacts from operations for grouting this material were evaluated (DOE 1986a). Future grouting actions are expected to utilize the DMRHF and TGE constructed for this low-level waste. If an additional TGE is required, it would be expected to be similar in design with incorporation of appropriate technical advances. The second waste stream to be grouted will probably be double-shell slurry (DSS). This waste was generated when a number of different dilute waste streams were concentrated to form a saturated salt slurry. This waste stream may require further processing to reduce TRU concentration to <100 nCi TRU/g.

TABLE D.2. Typical Composition of Candidate Grout Feedstreams (g/L)

Stream Component	CW	DSS	CC	CRW	NCAW
Na^+	23	115	34		115
AlO_2^-		53	1.8		35
$NaOH$	0.4	24	4	16	36
$NO_3^- + NO_x^-$	0.54	150	22		150
SO_4^{-2}	0.96	2.9	0.96		19
PO_4^{-3}	28	85	0.38		
CO_3^{-2}		8.4	24		
CrO_4^{-2}		3.5			
F^-					1.5
NaF				3.8	
$NaNO_3$				1.7	
NH_3				6.8	
$ZrO_2 \cdot H_2O$				16	
$NaNO_x$	0.69			0.69	
TOC(a)		1.1	3.5		0.09

(a) TOC--Total organic carbon.

TABLE D.3. Estimated Total Volume of Candidate Grout Feedstreams (m^3)

Feedstream	Scenario A(a)	Scenario B(b)
CW	4.2×10^4	1.2×10^4
DSS	1.1×10^5	1.2×10^5
CC	1.4×10^5	1.5×10^4
CRW	8.0×10^4	5.0×10^4
NCAW	1.1×10^4	1.2×10^4
Total Volume	3.8×10^5	2.1×10^5

(a) Represents no concentration of feedstreams to reduce waste volumes to be grouted.

(b) Basis for cost and resource estimate presented in text. Assumes that feedstreams are concentrated to 5 M Na or 50 wt% solids.

Other candidate waste streams for grouting include complexed concentrate (CC), a waste concentrated after removing strontium and cesium from high-level waste supernatant; cladding removal waste (CRW), a waste generated when cladding is chemically removed from spent N Reactor fuels (this waste stream may require further processing to reduce TRU levels to <100 nCi TRU/g); and neutralized current acid waste (NCAW) supernatant, a waste stream that results from the separation of solids and cesium from the high-level waste stream generated by processing of defense reactor fuel in the PUREX Plant.

Certain chemicals, such as fluorides and organics in CRW, DSS, CC, and NCAW, may retard the hardening rate of grouts. Additional treatment may be required to remove or neutralize these chemicals if grout formulas cannot be developed that ensure acceptable grouts.

D.5 RESOURCE REQUIREMENTS

Resource requirements for grouting Hanford liquid wastes are shown in Tables D.4 and D.5. The resource requirements for construction and operation of the TGF are not considered significant in that they will not require large quantities of nonrenewable resources.

TABLE D.4. Resource Requirements--TGF Construction

Resource Requirements	
Personnel, man-yr	4.2×10^1
Land, ha	2.2×10^1
Water, m^3	3.7×10^2
Energy Consumed:	
Electrical, GWh	9.0×10^{-2}
Propane m^3	1.6×10^2
Diesel m^3	8.8×10^2
Gasoline m^3	2.2×10^1
Materials:	
Concrete, m^3	1.8×10^2
Steel, t	2.7×10^2
Lumber, t	8.0

D.6 NONRADIOLOGICAL EMISSIONS

Nonradiological emissions associated with construction of the TGF, as shown in Table D.6, include those from the grouting process only. The emissions listed do not include those from disposal vault preparation or final grout placement (see Appendix L). The only nonradiological emissions expected during operation of the TGF are small amounts of fugitive dust and thermal emissions. Dust reduction systems will be provided in the Dry Materials

TABLE D.5. Resource Requirements--TGF Operation

Resource Requirements	
Personnel, man-yr	2.3×10^2
Energy Consumed:	
Electrical, GWh	2.1
Materials:	
Cement, t	1.1×10^5
Fly Ash, t	1.1×10^5
Clay, t	5.4×10^4

TABLE D.6. Nonradiological Emissions--TGF Construction

Pollutant	Emissions
Aldehydes, kg	13
Particulates, kg	1.3×10^2
SO_x, kg	1.3×10^2
CO, kg	1.1×10^3
Hydrocarbons, kg	2.0×10^2
NO_x, kg	1.7×10^3
Thermal, J	2.0×10^{12}
Fugitive Dust, t	1.2

Receiving and Handling Facility (DMRHF) to keep fugitive dust down to insignificant levels. None of the nonradiological releases from either construction or operation of the TGF appear significant when compared to regulatory limits identified in Chapter 6.

D.7 RADIOLOGICAL IMPACTS

Dose commitments to the general population and to the maximally exposed individual for routine operations under the reference alternative are presented in Tables D.7 through D.10. Impacts were calculated for grouting operations associated with new and existing tank waste (note that those impacts were calculated using disposal trenches instead of disposal vaults. The impacts may be reduced by the subsurface disposal vaults with the waste form 5 m or more deep and covered with a protective barrier). The doses include those from the evaporation and grouting processes and are presented for exposure periods of one and 70 years. The projected doses are insignificant in comparison to the dose received from naturally occurring sources, which contribute 0.1 rem per year for each individual and 42,000 man-rem per year to the offsite population.

TABLE D.7. Population Total-Body Dose Commitments (man-rem) from Evaporation and Grouting of Existing Tank Waste

Pathway	Exposure Period 1 yr	Exposure Period 70 yr
Air Submersion	1.0×10^{-11}	2.0×10^{-10}
Inhalation	2.0×10^{-8}	5.0×10^{-7}
Terrestrial (air paths)	5.0×10^{-8}	1.0×10^{-5}
Totals	7.0×10^{-8}	1.0×10^{-5}

TABLE D.8. Maximum Individual Total-Body Dose Commitment (rem) from Evaporation and Grouting of Existing Tank Waste

Pathway	Exposure Period 1 yr	Exposure Period 70 yr
Air Submersion	9.2×10^{-17}	1.9×10^{-15}
Inhalation	1.9×10^{-13}	4.0×10^{-12}
Terrestrial (air paths)	8.1×10^{-13}	1.8×10^{-10}
Totals	1.0×10^{-12}	1.8×10^{-10}

TABLE D.9. Population Total-Body Dose Commitments (man-rem) from Evaporation and Grouting of New Tank Waste

Pathway	Exposure Period 1 yr	Exposure Period 70 yr
Air Submersion	6.0×10^{-12}	3.0×10^{-11}
Inhalation	3.0×10^{-4}	2.0×10^{-3}
Terrestrial (air paths)	2.0×10^{-3}	1.0×10^{-2}
Totals	2.0×10^{-3}	1.0×10^{-2}

TABLE D.10. **Maximum Individual Total-Body Dose Commitment (rem) from Evaporation and Grouting of New Tank Waste**

Pathway	Exposure Period 1 yr	 70 yr
Air Submersion	4.6×10^{-17}	2.7×10^{-16}
Inhalation	2.7×10^{-9}	1.6×10^{-8}
Terrestrial (air paths)	2.8×10^{-8}	1.6×10^{-7}
Totals	3.1×10^{-8}	1.8×10^{-7}

D.8 COSTS

Costs for grouting wastes according to the reference alternative include construction, operation, and decontamination and decommissioning (D&D), as shown in Table D.11. The costs associated with grouting are significantly greater than previously estimated in RHO-RE-ST-30 P (Rockwell 1985), and the reasons for the increase are also discussed by Rockwell (1987). The increase is primarily due to the costs of vault construction, compared to the earlier trench design.

TABLE D.11. **Cost for Grouting Under the Reference Alternative**

Phase	Cost, millions of $1987[(a)]
Construction	400
Operation	270
D&D[(b)]	14
Total	680

(a) Includes costs for research and development and construction of protective barriers. Data apply to the reference alternative (Rockwell 1987).

(b) D&D costs for a facility are assumed to be 20% of its construction cost.

Costs for grouting only SST wastes are shown in Table D.12. The data in Table D.12 are provided to permit a comparison between the reference and geologic alternatives. Again the costs include construction, operation, and decontamination and decommissioning. Changes in grouting requirements delineated for the reference alternative also apply when estimating costs for grouting SST waste.

TABLE D.12. Costs for Grouting Single-Shell Tank Wastes

Phase	Cost, millions of $1987[(a)]
Construction	860
Operation	430
D&D	40
Total	1,330

(a) Costs include research and development costs plus construction costs for protective barriers.

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WASTE RECEIVING AND PROCESSING FACILITY

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WASTE RECEIVING AND PROCESSING FACILITY

As discussed briefly in Chapter 3 of this EIS, the major functions of the Waste Receiving and Processing facility (WRAP) are: 1) to provide for examination, processing, packaging, and certification of retrievably stored contact-handled (CH) transuranic (TRU) wastes and 2) to provide for examination and certification of newly generated CH-TRU waste for repository disposal. This appendix describes the facility, the waste examination, processing, and packaging systems, the flow of materials through the WRAP facility, and the associated waste feedstreams. This appendix also summarizes the resource requirements, emissions, radiological impacts, and costs associated with construction, operation, and decontamination and decommissioning (D&D) of the facility.

E.1 INTRODUCTION

The WRAP facility is conceptually designed to support examination and certification of CH-TRU waste for repository disposal [assumed for calculation purposes to be the Waste Isolation Pilot Plant (WIPP)]. Processing and packaging capabilities for CH-TRU waste in 20-year retrievable storage will also be provided in the WRAP facility. On December 31, 1983, the total volume of retrievably stored CH-TRU waste at Hanford was about 13,000 m^3. This waste is stored below ground on asphalt pads as shown in Figures E.1 through E.5. The total volume of newly generated CH-TRU waste that will be generated at Hanford from 1984 through 1996 is about 13,100 m^3 (Rockwell 1985a). This waste must be inspected and certified to meet Waste Acceptance Criteria (WAC), for WIPP (Westinghouse 1983).

FIGURE E.1 **Variety of TRU Waste Containers**

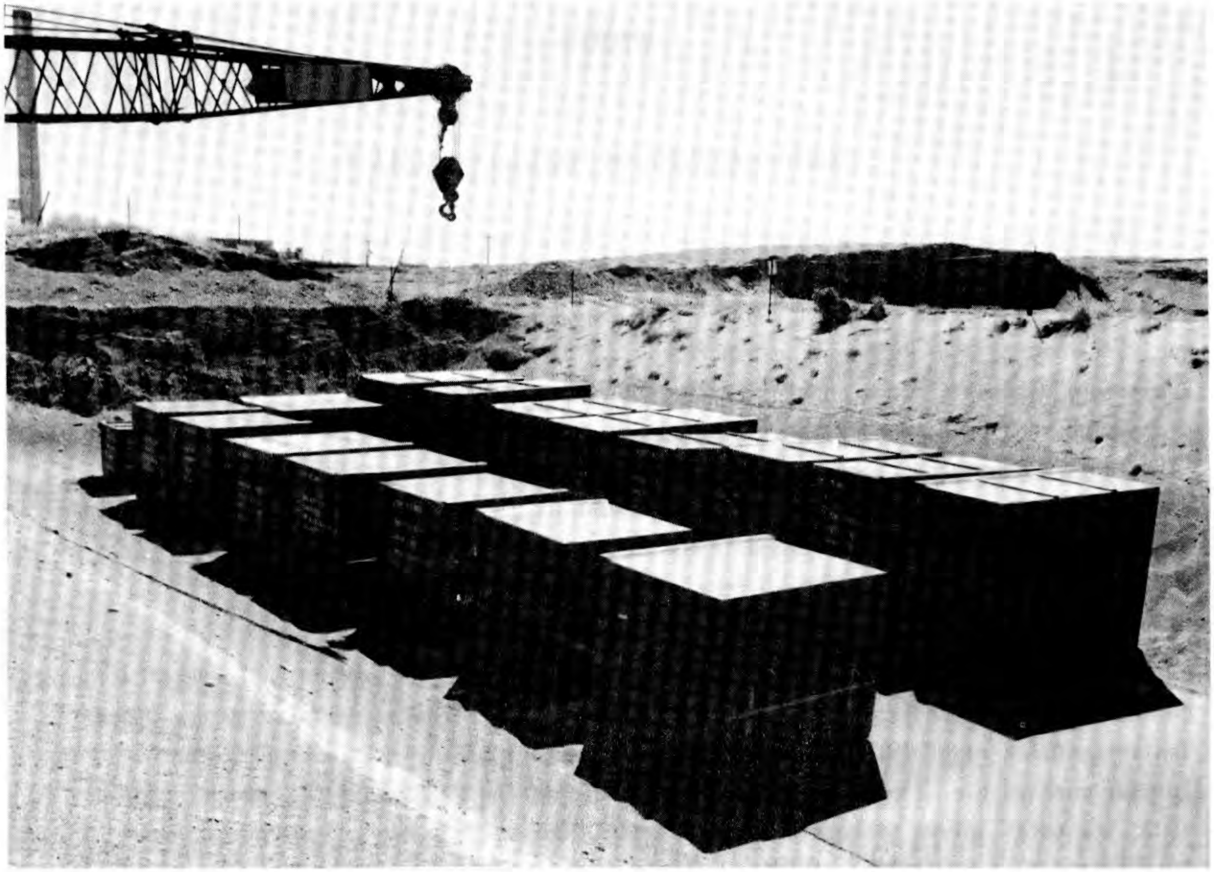

FIGURE E.2 **TRU Waste Fiberglass-Reinforced Plywood Boxes**

In estimating process costs, emissions, and volumes of waste, it is projected that 40% of all CH-TRU waste will be reclassified as low-level waste after the TRU waste content of each waste package is measured. The projected 40% of waste to be reclassified is based on engineering judgment and historical records (DOE 1984).

Current estimates are that about 4,000 m^3 of the retrievably stored and 5,200 m^3 of the newly generated TRU waste will be reclassified as low-level waste after being assayed. The low-level waste will be sent to the 200 Area burial grounds for disposal. Of the remainder, about 7,500 m^3 of the newly generated waste and none of the retrievably stored waste will be directly certifiable for shipment to WIPP. The balance (9,000 m^3 of retrievably stored and 400 m^3 of newly generated waste) will be processed within the WRAP facility to produce certifiable waste packages that will meet repository acceptance (WIPP-WAC).

The waste process systems being considered include waste package inspection, assaying, repackaging, size reduction, compaction, sorting, shredding, and waste immobilization in grout. Incineration will be implemented as an additional process step between shredding and grouting if deemed appropriate. A process flow diagram for WRAP, which uses the shredding process without incineration, is shown in Figure E.6.

FIGURE E.3. TRU Waste Storage Pad

The experience gained at the Idaho National Engineering Laboratory (INEL) will be utilized in the technology selection and development work for the WRAP facility. Solid waste processing facilities are currently being constructed there (Clements 1984). The Stored Waste Examination Pilot Plant at INEL will certify wastes for shipment to WIPP. The Process Experimental Pilot Plant, also at INEL, will treat and convert uncertified waste into certifiable packages for shipment to WIPP.

The WRAP facility is scheduled to be constructed in the 200 West Area during the 1990s. The operational period required for handling newly generated TRU is dependent on future activities at Hanford. For this EIS, it was assumed that all retrievably stored and newly generated TRU waste (generated through 1996) will be processed between 1994 and the year 2006 (Rockwell 1985a).

An estimated 22 m^3 of retrievably stored and 500 m^3 of newly generated remote-handled (RH) TRU waste will also require processing. This waste may be routed to a Special Handling and Packaging Facility designed to process RH-TRU waste (i.e., not in the WRAP facility).

FIGURE E.4. TRU Waste Storage Facilities

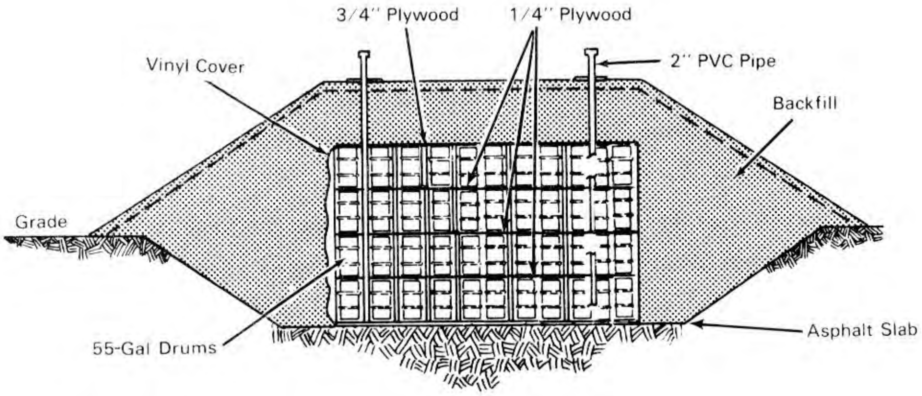

FIGURE E.5. Typical Newly Generated TRU Waste Interim Storage (ERDA 1975)

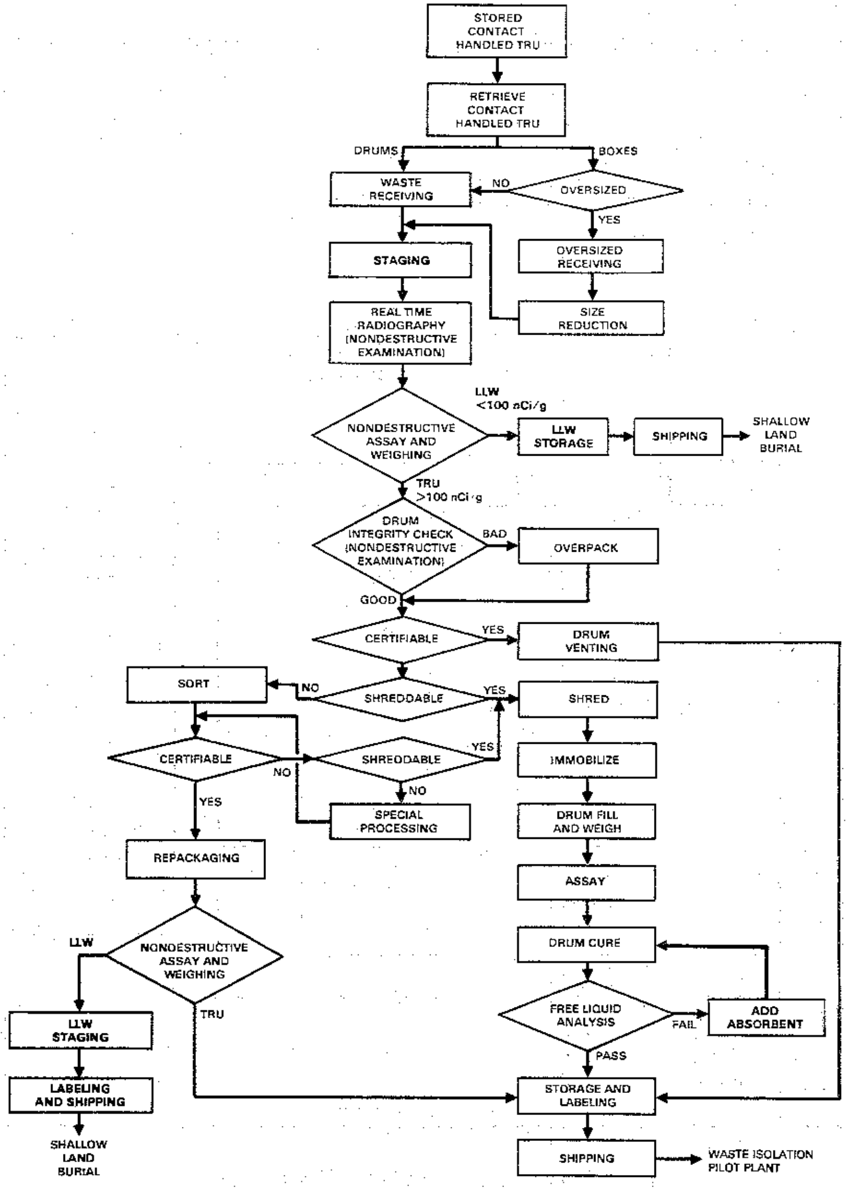

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FIGURE E.6. WRAP Process Flow Diagram

This RH-TRU waste is expected to be processed and stored with RH-TRU waste from the decontamination and decommissioning of Hanford facilities pending shipment to WIPP for disposal.

E.2 WRAP FACILITY DESCRIPTION

A conceptual floor plan for the WRAP facility is given in Figure E.7. The major areas are:

- receiving dock
- size-reduction room
- nondestructive assay and examination (NDA/NDE) room
- waste processing room
- shipping dock.

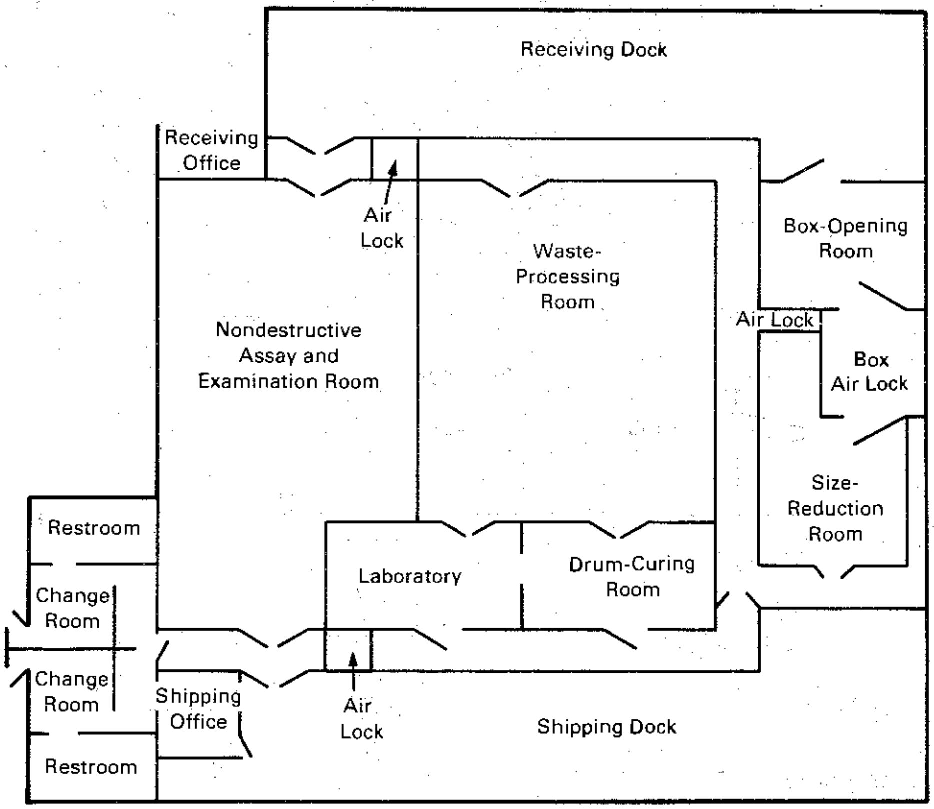

FIGURE E.7. **WRAP Facility Floor Plan**

The WRAP facility consists of about 1,700 m^2 of building floor space plus an additional 930 m^2 of dock space (Rockwell 1985b). The floor plan has been designed to group and centralize zones with high contamination potential and to separate the docks for waste receiving and shipping. Zones with high contamination potential are areas of the building where waste packages are opened and where glove-box, hood or cell operations are conducted. Operating galleries (access hallways) to such areas are also within these zones. Grouping such areas together will simplify ventilation system design and minimize air lock requirements. Physical separation of the waste-receiving and waste-shipping docks will help prevent inadvertent mixing of certified and yet-to-be-certified waste package shipments, as might occur on a common receiving/shipping dock. Location of the docks on opposite sides of the facility will also provide "straight-through" flow paths for waste packages. Figure E.8 depicts the waste package flow within the WRAP facility.

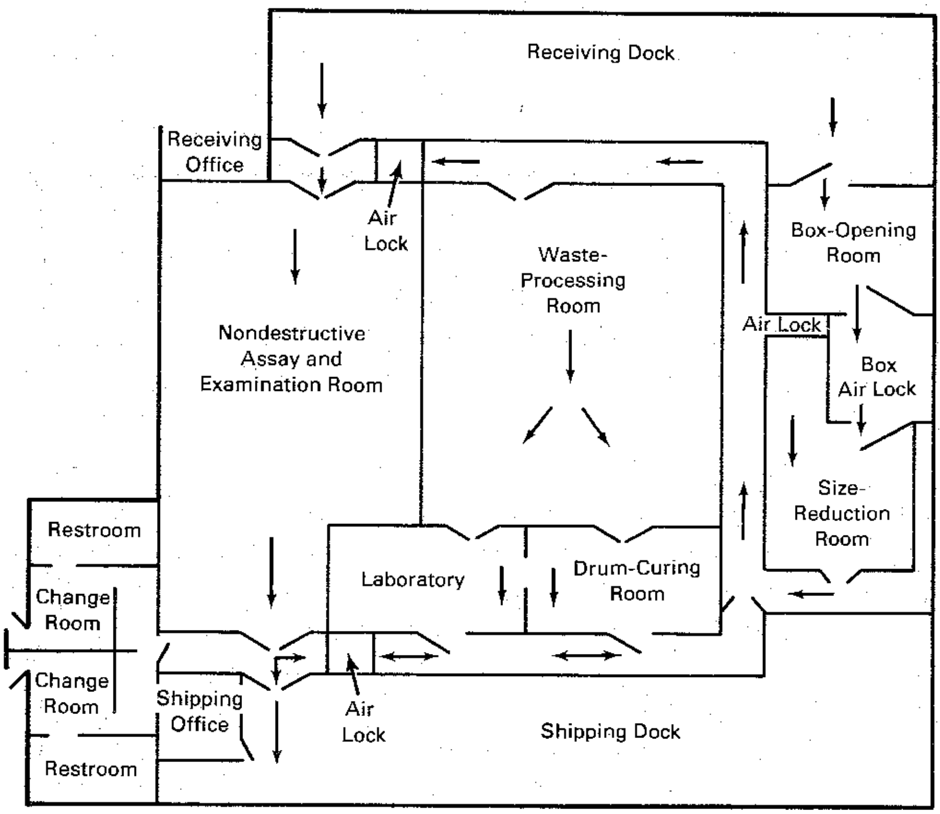

FIGURE E.8. **Waste Package Flow**

E.3 WASTE PROCESS DESCRIPTION

The WRAP facility will examine, process, and repackage, as necessary, 20-year retrievably stored TRU (Rockwell 1985b). Both 20-year retrievably stored TRU waste and newly generated TRU will be certified in the WRAP facility. The WRAP facility waste package flow is described below.

E.3.1 Receiving Dock

The first step in the waste package flow is off-loading the waste onto the receiving dock. The dock will be constructed to facilitate off-loading of trucks by forklift and possibly by crane. Once off-loaded, the waste packages will undergo initial inspection to determine whether incoming wastes meet WIPP criteria or whether further processing is required. For inspection, the receiving dock will be equipped with instruments that measure surface contamination, surface exposure rates, and physical dimensions. Packages with exposure rates greater than 200 mR/hr will be treated or placed in a canister overpack to reduce exposure rates. If it is not cost-effective to place waste packages in a canister overpack, thereby reducing exposure levels below contact handling limits, the waste will be treated as remote-handled TRU and transferred to remote-handled TRU waste storage.

E.3.2 Size-Reduction Room

Waste packages that exceed WIPP-WAC physical size requirements will be diverted to the size-reduction room. Here the waste will be repackaged into drums or steel boxes. The size-reduction area in the WRAP facility will consist of 1) a waste container opening chamber (box-opening room), 2) a waste-entry air lock, and 3) a size-reduction cell. The box-opening chamber will be equipped with commercially available equipment that will open boxes and sample for internal airborne contamination. The size-reduction cell will be a large stainless steel enclosure equipped with glove ports and viewing windows. Operations will be performed both remotely and manually. The room will be equipped with a positioning table that rotates horizontally and vertically, manipulators and cranes, lightweight dismantling tools, and metal sectioning equipment including nibblers, mechanical saws, abrasive saws, electric saws, and/or plasma torches.

E.3.3 Nondestructive Assay and Examination Room

Waste packages that meet size, contamination and exposure criteria will then be routed to the nondestructive assay and examination (NDA/NDE) room to determine 1) TRU waste content, 2) weight, and 3) the presence of noncompliance items such as free liquids, or cylinders of compressed gases. Equipment potentially required for NDA/NDE includes: scale systems (both in-floor, drive-on scales and smaller scales), neutron- and gamma-scan assayers, X-ray fluoroscopy equipment, ultrasonic and eddy current systems, and visual examination instruments. All directly certifiable waste (waste that requires no further processing to meet WIPP criteria) will be routed to the shipping dock for transport to WIPP. Waste that does not meet WIPP criteria will be diverted to the waste-processing room.

E.3.4 Waste-Processing Room

Noncertifiable drummed waste will be sent through the waste-processing room depicted in Figure E.9. It includes an opening and sorting glove box and a shredding and immobilizing processor. The opening and sorting glove box is shown in Figure E.10. The design provides for removal of drum lids and for lifting, tilting, and unloading of the drum to a sorting table. The sorting table will separate drum waste into certifiable categories and will be equipped with manipulator arms, glove ports, and tools. This glove box will also be able to crush empty drums and repackage waste.

The WIPP-WAC requires immobilization of all particulates and removal of free liquids. The shredder and immobilizer will process drum waste to meet these immobilization criteria. The shredding/immobilization process line is shown in Figure E.11. A slow-speed shredder with double rotors (Figure E.12) will be used to shred whole 55-gal and 83-gal drums and other similarly sized containers. The shredder will reduce the size of the waste package by ripping, tearing, and shearing it. To minimize contamination and the potential for fire or explosion, the shredding process will be designed to control dust and sparks.

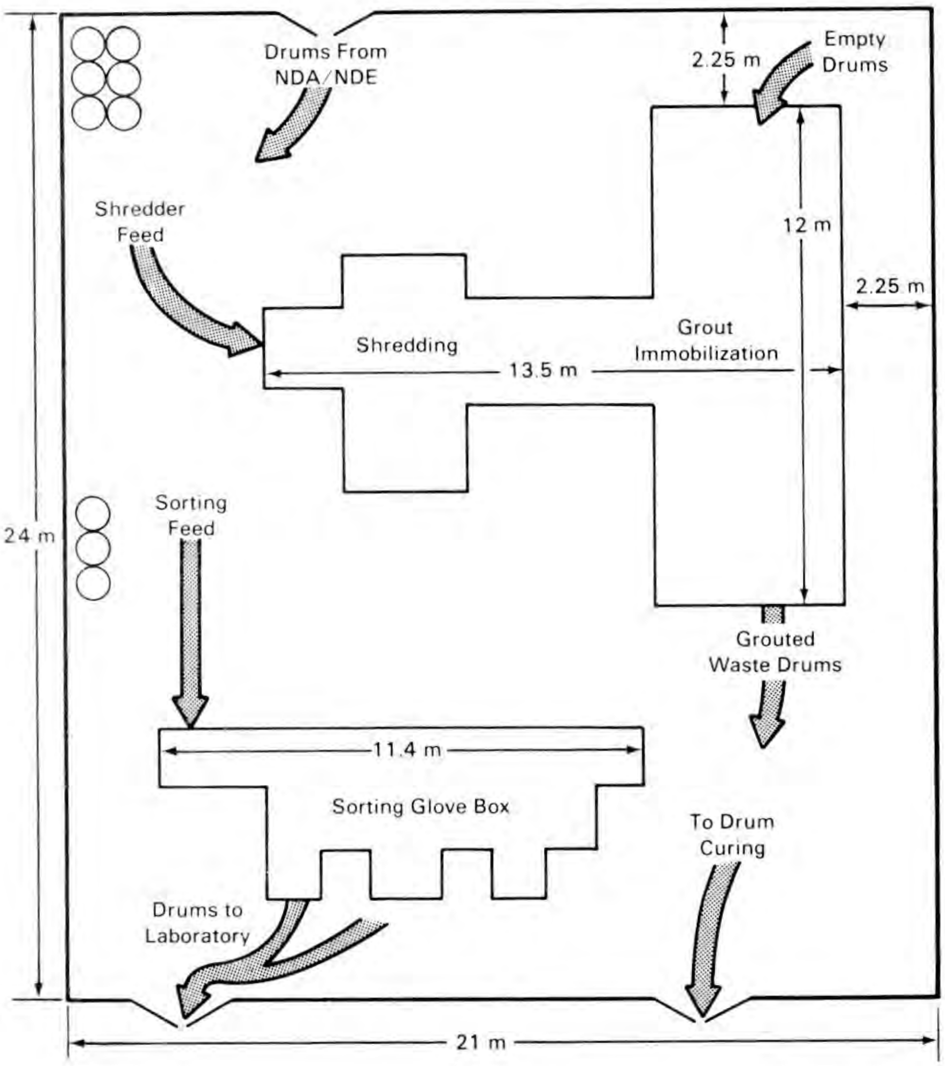

FIGURE E.9. Waste-Processing Room Floor Plan

Package opening and sorting will be used when direct shredding of unopened packages is not practical. Examples of nonshreddable waste include pressurized gas cylinders and drums with potentially flammable or explosive contents. Opened drums will be sorted to remove noncertifiable contents for further processing. Uncertifiable waste items will be processed via shredding and immobilization, direct immobilization, or other processes as required. Remote operation and maintenance will minimize any damage resulting from contact with unshreddable items.

When appropriate, processed waste will be transferred to a rotating grout-mixing chamber to be immobilized in grout. Grout formula(s) most suited to immobilize the shredded waste will be determined by experimental testing. To meet functional requirements, the grout must

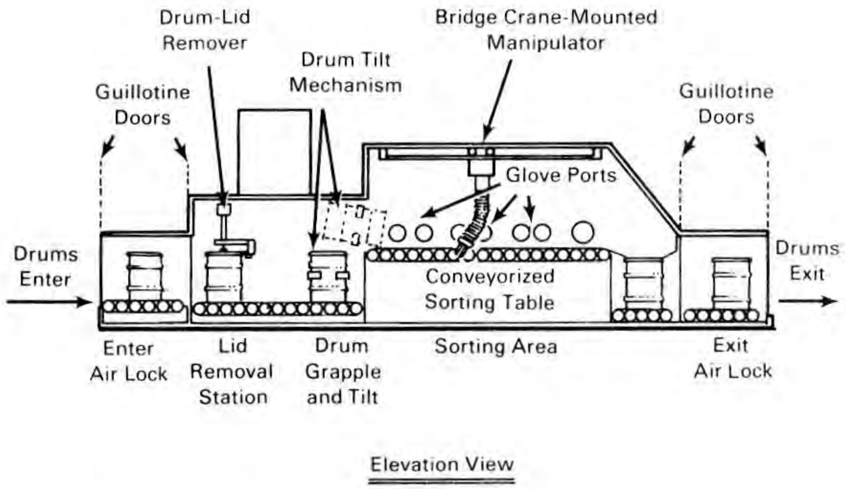

FIGURE E.10. **Drum Opening and Sorting Glove Box** (Elevation View)

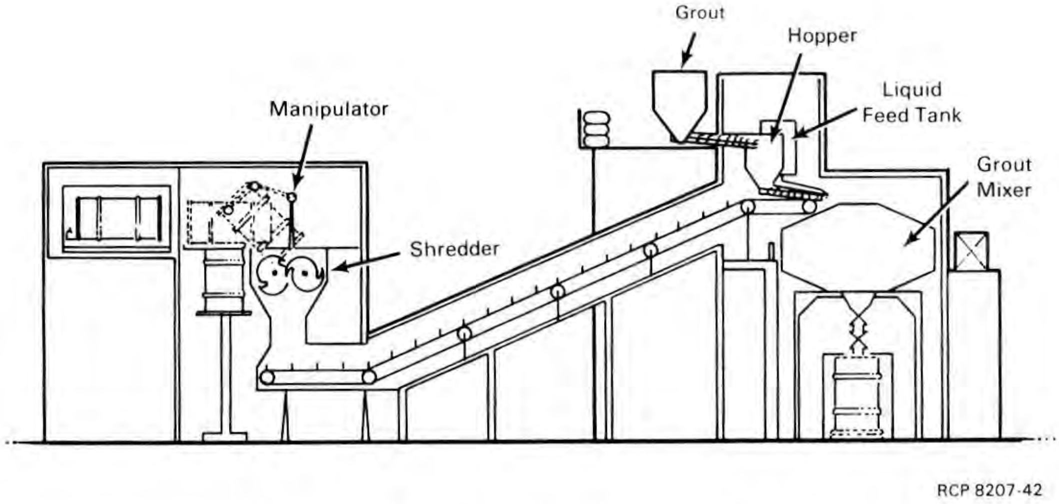

FIGURE E.11. **Shredding/Immobilization Process Line**

immobilize any particulates and free liquids generated as a result of the shredding process. Grouting must also eliminate any pyrophoric and/or corrosive characteristics of the waste. The grout/shredded waste mixture will be injected into drums and sent to the drum-curing room for solidification. The grouting process will also provide for direct immobilization of various liquid waste streams.

E.3.5 Shipping Dock

Three waste package streams will enter the shipping area of the WRAP facility: 1) low-level waste, 2) directly certifiable waste, and 3) waste made certifiable by processing. The low-level waste will be shipped to an onsite low-level burial ground, while all certifiable TRU waste packages will undergo final preparation for shipment to WIPP.

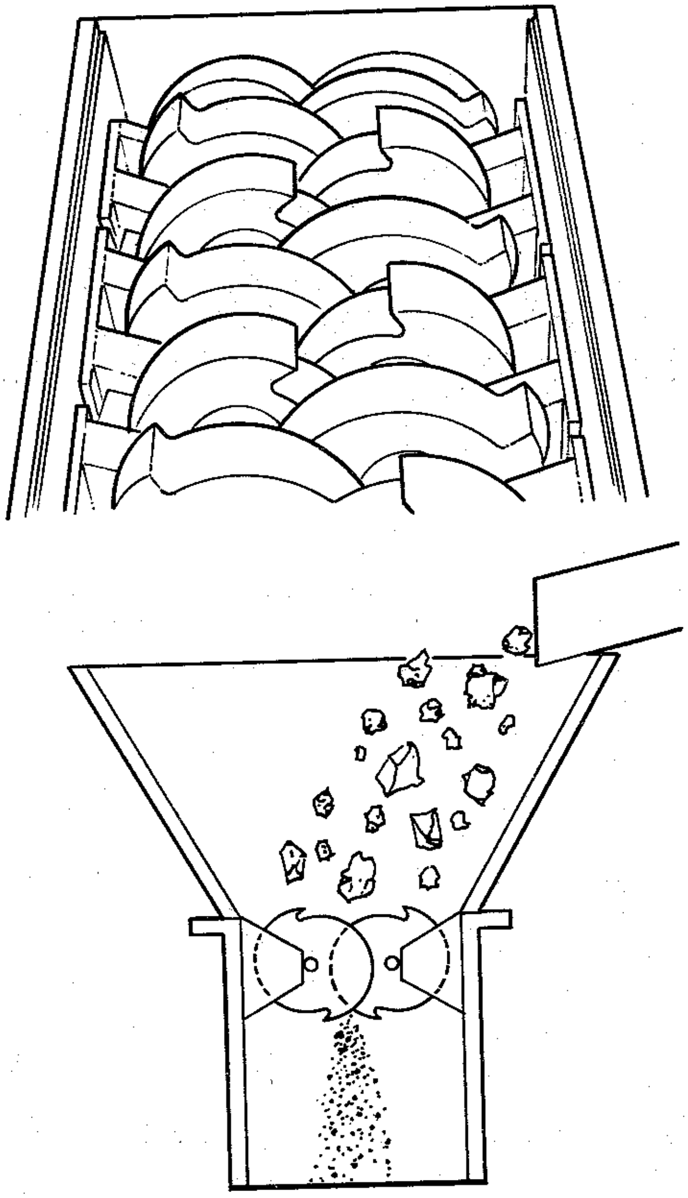

FIGURE E.12. **Schematic of a Typical Slow-Speed Shredder**

Identification letters and package weight labeling are required. The shipping dock will have a labeling station where this final certification of wastes will take place. Labeled TRU waste packages will be loaded by forklift into a TRUPACT transport container for shipment to WIPP by truck.

E.4 RESOURCE REQUIREMENTS

Major resource requirements expected for construction and operation of the WRAP facility are listed in Tables E.1 and E.2, respectively. The tables provide annual resource

TABLE E.1. **Total Resource Requirements for Construction of the WRAP Facility**

Resource Requirements	
Personnel, man-hr	1.8×10^5
Land, ha	6.0×10^{-1}
Energy Consumed:	
Electrical, GWh	1.9×10^3
Diesel, m^3	5.5×10^4
Gasoline, L	0
Materials:	
Concrete, m^3	1.2×10^3
Steel, t	3.2×10^1

TABLE E.2. **Annual Resource Requirements for Operation of the WRAP Facility**

Resource Requirements	
Personnel, man-yr	20
Water Consumed, L	2.5×10^5
Energy Consumed:	
Electrical, MWh	1.2×10^2
Diesel, m^3	2.2×10^2
Materials:	
Steel, t	7.3×10^1
Grout, t	5.3×10^2

requirements for operations and the total quantity of material needed for construction. No resource needs were identified for either construction or operation of the WRAP facility that would significantly deplete existing resources.

E.5 RADIOLOGICAL EMISSIONS

Beginning about 1994, about 3,550 m^3 of retrievably stored TRU waste and 9,560 m^3 of newly generated TRU waste were assumed to be processed in the WRAP facility. The retrievably stored TRU waste is assumed to be processed and repackaged during a 5-year period, and the newly generated TRU waste is assumed to be processed during a subsequent 8-year period. Due to uncertainties associated with the distribution of the radionuclide inventory, it is conservatively assumed that the entire radionuclide inventory is present in the fraction of waste drums and boxes that are shredded. The annual radiological releases to the atmosphere from routine operations at WRAP are presented in Tables E.3 and E.4. Projected annual releases from the WRAP facility are well below the limits established by DOE for release in uncontrolled areas (DOE 1986).

TABLE E.3. Routine Radiological Releases from Processing Retrievably Stored TRU Waste at WRAP

Isotope	Annual Release Ci/yr[(a)]
^{3}H	5.5×10^{-11}
^{14}C	1.3×10^{-11}
^{60}C	2.5×10^{-8}
^{90}Sr	2.3×10^{-7}
^{85}Kr	3.0×10^{-8}
^{106}Ru	2.3×10^{-7}
^{137}Cs	2.4×10^{-7}
^{144}Ce	4.3×10^{-7}
^{147}Pm	3.2×10^{-7}
^{155}Eu	2.9×10^{-10}
^{233}U	5.0×10^{-11}
^{234}U	4.7×10^{-11}
^{235}U	1.3×10^{-12}
^{238}Pu	2.6×10^{-7}
^{239}Pu	1.5×10^{-7}
^{240}Pu	3.6×10^{-8}
^{241}Pu	4.8×10^{-7}
^{242}Pu	1.4×10^{-11}
^{241}Am	1.0×10^{-8}
^{232}Th	5.2×10^{-13}

(a) Rockwell 1985a.

TABLE E.4. Routine Radiological Releases from Processing Future TRU Waste at WRAP

Isotope	Yr 2000 Isotope Inventory, Ci	Annual Release Ci/yr	Total Release Ci
^{238}Pu	5.7×10^{2}	9.0×10^{-9}	1.1×10^{-7}
^{239}Pu	1.9×10^{4}	3.1×10^{-7}	3.7×10^{-6}
^{240}Pu	4.3×10^{3}	6.9×10^{-8}	8.3×10^{-7}
^{241}Pu	1.2×10^{5}	2.8×10^{-6}	2.2×10^{-5}
^{241}Am	3.4×10^{3}	5.4×10^{-8}	6.5×10^{-7}

For calculation purposes, it was assumed that the decontamination factor of each high-efficiency particulate air (HEPA) filter is 2 x 10^3 and that two HEPA filters are in series. The airborne dust level within the process cell was assumed to be 0.01 g/m^3. The cell volume is about 4,200 m^3 and the ventilation rate is five air exchanges per hour. The average density of the waste is about 3 x 10^5 g/m^3 and the WRAP facility is assumed to operate one shift per day, 5 days per week, and 50 weeks per year (2,000 hr/yr).

E.6 NONRADIOLOGICAL EMISSIONS

Routine nonradiological releases for construction and operation of the conceptual WRAP facility are summarized in Table E.5. Estimates of total emissions are based on factors for grams of pollutants per liter of fuel consumed (EPA 1978, 1979). Not included in the totals are the emissions relating to the production of electric power, the refinement and transportation of petroleum, the manufacturing of cement, and the fabrication of equipment and other materials used in the construction and operation of the WRAP facility.

TABLE E.5. Nonradiological Emissions from the WRAP Facility

Emissions	Total Release
Particulates, kg	2.0 x 10^2
SO_x, kg	2.1 x 10^2
CO, kg	6.6 x 10^2
Hydrocarbons, kg	9.3 x 10^2
NO_x, kg	2.6 x 10^3
Aldehydes, kg	4.7 x 10^1
Thermal, J	7.2 x 10^{15}

Essentially all the nonradiological emissions occur during construction. Operational emissions are expected to be insignificant due to the absence of chemical treatment and the absence of gasoline- or diesel-fueled equipment. Annual emissions from the WRAP facility are projected to be well below regulatory limits (described in Chapter 6) established for atmospheric release.

E.7 RADIOLOGICAL IMPACTS

Dose commitments to the general population and to the maximally exposed individual are presented in Tables E.6 and E.7, respectively. The values presented include doses from the processing of retrievably stored and newly generated CH-TRU waste. Values are given for

TABLE E.6. Population Total-Body Dose Commitments (man-rem) **from the Processing of Retrievably Stored and Newly Generated CH-TRU at WRAP**

Pathway	Exposure Period 1 yr	 70 yr
Air submersion	5.0×10^{-11}	9.0×10^{-10}
Inhalation	1.2×10^{-5}	2.4×10^{-4}
Terrestrial (air paths)	2.0×10^{-7}	4.0×10^{-5}
Total Doses	1.2×10^{-5}	2.8×10^{-4}

TABLE E.7. Maximum Individual Total-Body Dose Commitment (rem) **from the Processing of Retrievably Stored and Newly Generated CH-TRU at WRAP**

Pathway	Exposure Period 1 yr	 70 yr
Air submersion	3.7×10^{-16}	7.3×10^{-15}
Inhalation	9.7×10^{-11}	2.1×10^{-9}
Terrestrial (air paths)	3.6×10^{-12}	7.4×10^{-10}
Total Doses	1.0×10^{-10}	2.9×10^{-9}

exposure periods of one year and 70 years. The projected population doses shown in Table E.6 are insignificant when compared to the 2.5×10^4 man-rem the offsite population will receive over the same time period from natural background radiation sources.

E.8 COSTS

The costs for the WRAP facility are presented in Table E.8. The construction costs include the capital costs of constructing the facility. The operational costs (through the year 2006) include shipping and disposal costs and all costs incurred during normal operation and maintenance while retrievably stored and newly generated TRU wastes are being processed.

TABLE E.8. Costs for the Waste Receiving and Processing Facility

Phase	Cost[(a)]
Construction	46
Operations	78
Decontamination and Decommissioning	9
Total	130

(a) Costs in millions of 1987 dollars (Rockwell 1987).

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