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Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
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POTENTIAL RADIOLOGICAL IMPACTS OF UPPER-BOUND OPERATIONAL ACCIDENTS DURING PROPOSED WASTE DISPOSAL ALTERNATIVES FOR HANFORD DEFENSE WASTE

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FOREWORD

This document was prepared to present more detail with respect to operational accidents than could reasonably be contained in Appendix H, "Radiation Doses to the Public from Operational Accidents," contained in the environmental impact statement pertaining to disposal of Hanford high-level, transuranic and tank wastes, DOE/EIS-0113, to be issued in 1986.

SUMMARY

Three alternatives are being evaluated for disposal of Hanford defense high-level, transuranic, and tank wastes. The wastes have been identified - existing tank waste, future tank waste, cesium and strontium capsules, transuranic (TRU) contaminated soil, pre-1970 buried TRU solids, and retrievably stored and newly generated TRU solid waste. The three alternatives are the Geologic Disposal Alternative, the In-Place Stabilization and Disposal Alternative and the Reference Disposal Alternative.

Environmental impacts associated with disposal of these wastes according to the alternatives listed above include potential doses to the downwind population from operation during the application of the handling and processing techniques comprising each disposal alternative. Scenarios for operational accident and abnormal operational events are postulated, on the basis of the currently available information, for the application of the techniques employed for each waste class for each disposal alternative. From these scenarios, an upper-bound airborne release of radioactive material was postulated for each waste class and disposal alternative. Potential downwind radiologic impacts were calculated from these upper-bound events.

The dose to the maximally exposed individual and the total downwind population dose were obtained by using standard radionuclide transport and dose computer codes. The meteorological data used for the transport were selected from data collected at the Hanford Site over many years. Demographic data for the locale was taken from published information.

In all three alternatives, the single postulated event with the largest calculated radiologic impact for any waste class is an explosion of a mixture of ferri/ferro cyanide precipitates during the mechanical retrieval or microwave drying of the salt cake in single shell waste tanks. The anticipated downwind dose (70-year dose commitment) to the maximally exposed individual is 3 rem with a total population dose of 7000 man-rem. The same individual would receive 7 rem from natural background radiation during the same time period, and the same population would receive 3,000,000 man-rem. Radiological impacts to the public from all other postulated accidents would be less than that from

this accident; furthermore, the radiological impacts resulting from this accident would be less than one-half that from the natural background radiation dose. Since this is the postulated accident for each of the disposal alternatives, operational accidents in general should not constitute an important discrimination among alternatives.

CONTENTS

FIGURES

TABLES

1.0 INTRODUCTION

An important component of an environmental analysis of various waste disposal alternatives is the potential for radiological impacts on the downwind population from accidents that may occur during disposal operations. Accidents in which radiological material is released to the atmosphere usually result in exposure of the body by means of the inhalation pathway.

Accidents of concern are those with the highest releases of radioactive particles. Known as upper-bound accidents, these can provide an umbrella source term for all inadvertent releases for that technique. The most severe upper-bound accidents in terms of airborne release of material are those that include damage to the facility and/or its engineered safeguards. This damage could provide a direct path for radioactive particles to enter the ambient atmosphere.

Postulated accidents cover upper-bound releases. They are not anticipated; but on the basis of current knowledge of the systems and materials, they cannot be eliminated from consideration. However, some semblance of realism has been maintained in the development of worst-case accidents. This has been achieved by studying and evaluating the process, equipment, and confinement features of the operation.

Fractional airborne releases are calculated from published information currently available. This includes release data calculated for similar accidents postulated in the literature and experimental data for similar types and levels of stresses. The fractional airborne releases are combined with the anticipated inventories of radioactive materials at the accident location and mitigation during passage through the facility to estimate an atmospheric airborne release. A standard code is then used to calculate the downwind transport and resulting dose. Assumptions used for the current and future demography (population) are stated.

2.0 CONCLUSIONS

Accident scenarios for three alternatives for disposal of Hanford Defense Waste are evaluated. The three alternatives are designated as the Geologic Disposal Alternative, the In-Place Stabilization and Disposal Alternative, and the Reference Disposal Alternative (see Section 4.0). A "No Disposal Action" alternative is also discussed. Six types of Hanford Defense Waste are considered: existing tank waste, future tank waste, strontium and cesium capsules, transuranic (TRU) contaminated soil, pre-1970 TRU-contaminated solid buried waste, and retrievably stored and newly generated TRU waste (see Section 4.0).

The potential radiologic impacts of the various techniques (unit operations) required to convert the material in each waste class into its final form for each of the disposal alternatives is one of the factors in the value-impact evaluation used to select an alternative. In order to determine the potential radiologic impacts, the airborne release of radionuclides from potential accident or abnormal operation events during each unit operation for each waste type is required (covered in Sections 5.0, 6.0, 7.0, and 8.0). The information currently available on the proposed techniques ranges from extensive for techniques currently in use to preliminary for preconceptual descriptions. An upper-bound airborne release is used to provide an "umbrella" term bounding all releases anticipated for the application of a technique upon a waste class during an alternative. Estimates for the airborne releases are based upon scenarios for postulated accident and abnormal operation events. Although a recurrence rate cannot be specified, almost all the events chosen are severe (if warranted by the circumstances) and result in the airborne release of significant quantities of radionuclides although doses to the offsite population and maximally exposed individual are low.

To convert the postulated airborne releases to dose, the radionuclides released are quantified by using maximized inventories for the waste class involved (covered in Section 9.0), estimating the downwind transport of the released material (see Section 10.0), and evaluating the potential radiation imposed on individuals from all mechanisms (discussed in Section 10.0).

All the pertinent data for the upper-bound releases and doses are tabulated in Table 2.1 for the Geologic Disposal Alternative, Table 2.2 for the In-Place Stabilization and Disposal Alternative, Table 2.3 for the Reference Disposal Alternative, and Table 2.4 for the No Disposal Action Alternative.

The upper-bound events for the three disposal alternatives are the same: an explosion of ferri/ferro upgrade precipitates in the salt cake during mechanical retrieval or microwave drying of the salt cake in single shell waste tanks, and the pressurized release of liquid waste during the hydraulic retrieval of future tank waste. The total mass of material made airborne from an explosion is estimated to be 5×10^8 g of salt cake with 1.3×10^4 g of material in the respirable size fraction (assumed to be 10 μm or less Aerodynamic Equivalent Diameter for this analysis). The pressurized release of waste liquid results in the airborne release of 9×10^4 g of material with 4.5×10^3 g in the size range of concern.

The calculated downwind doses to the total population and maximally exposed individual are 7000 man-rem and 3 rem (70-year dose commitment) respectively for the explosion. The corresponding values for the pressurized release event are 4000 man-rem and 2 rem, respectively. In either case, the 1-year dose commitment to the maximally exposed individual does not exceed 0.2 rem.

TABLE 2.1. Summary of Upper-Bound Accidents and Radiation Doses Associated with the Geologic Disposal Alternative of Hanford Defense Waste

	Description of Upper Bound Accident	Maximum Individual Total Body Dose, rem		Population Total Body Dose, man-rem	
		1-yr Dose	Commitment	1-yr Dose	Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in single-shell tank waste during mechanical retrieval of the waste.	2×10^{-1}	3×10^{0}	4×10^{2}	7×10^{3}
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval of the waste.	9×10^{-2}	9×10^{1}	3×10^{2}	2×10^{3}
Strontium and Cesium Capsules	Rupture of a strontium capsule by improper handling during retrieval operations.	2×10^{-7}	3×10^{-6}	6×10^{-4}	1×10^{-2}
TRU-Contaminated Soil Sites	Deflagration of contaminated materials due to process malfunction in slagging pyrolysis incinerator	5×10^{-7}	2×10^{-5}	1×10^{-3}	4×10^{-2}
Pre-1970 TRU Solid Waste	Deflagration of contaminated material due to process malfunction of slagging pyrolysis incinerator.	5×10^{-6}	1×10^{-4}	1×10^{-2}	3×10^{-1}
Retrievably Stored and Newly Generated TRU	Pressurized release from waste drum rupture due to buildup of radiolytic gases in the wastes.	1×10^{-3}	5×10^{-2}	3×10^{0}	1×10^{2}

TABLE 2.2. Summary of Upper-Bound Accidents and Radiation Doses Associated with the In-Place Stabilization Disposal Alternative of Hanford Defense Waste

	Description of Upper Bound Accident	Maximum Individual Total Body Dose, rem		Population Total Body Dose, man-rem	
		1-yr Dose	Commitment	1-yr Dose	Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in the single-shell tank waste during drying of the wastes	2×10^{-1}	3×10^{0}	4×10^{2}	7×10^{3}
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval of liquid waste.	9×10^{-2}	9×10^{1}	3×10^{2}	2×10^{3}
Strontium and Cesium Capsules	Shearing of strontium capsule by improper handling during disposal operations	3×10^{-4}	4×10^{-3}	6×10^{-1}	1×10^{1}
TRU-Contaminated Soil Sites	Collapse of voids in soil site during grout injection/site stabilization activities	2×10^{-8}	9×10^{-7}	5×10^{-5}	2×10^{-3}
Pre-1970 TRU Solid Waste	Collapse of void space at waste site during subsidence control operations.	3×10^{-7}	7×10^{-6}	6×10^{-4}	2×10^{-2}
Retrievably Stored and Generated TRU	Breach of waste container during package disposal operations.	2×10^{-3}	4×10^{-2}	5×10^{0}	8×10^{1}

TABLE 2.3. Summary of Upper-Bound Accidents and Radiation Doses Associated with the Reference Disposal Alternative of Hanford Defense Waste

	Description of Upper Bound Accident	Maximum Individual Total Body Dose, rem		Population Total Body Dose, man-rem	
		1-yr Dose	Commitment	1-yr Dose	Commitment
Existing Tank Waste	Explosion of ferrocyanide precipitates in single-shell tank waste during drying of the waste	2×10^{-1}	3×10^{0}	4×10^{2}	7×10^{3}
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval of the waste	9×10^{-2}	9×10^{-1}	3×10^{2}	2×10^{3}
Strontium and Cesium Capsules	Rupture of strontium capsule by improper handling during retrieval operations	2×10^{-7}	3×10^{-6}	6×10^{-4}	1×10^{-2}
TRU-Contaminated Soil	Collapse of voids in the soil site during grout injection/site stabilization activities	2×10^{-8}	9×10^{-7}	5×10^{-5}	2×10^{-3}
Pre-1970 TRU Solid Waste	Collapse of voids in solid waste site during grout injection/site stabilization	3×10^{-7}	7×10^{-6}	6×10^{-4}	2×10^{-2}
Retrievably Stored and Newly Generated TRU	Pressurized release from waste drum due to container rupture; pressurization due to buildup of radiolytic gases in waste	2×10^{-3}	6×10^{-2}	4×10^{0}	1×10^{2}

TABLE 2.4. Summary of Upper-Bound Accidents and Radiation Doses Associated with the No Disposal Action Alternative of Hanford Defense Waste

	Description of Upper Bound Accident	Maximum Individual Total Body Dose, rem		Population Total Body Dose, man-rem	
		1-yr Dose	Commitment	1-yr Dose	Commitment
Existing Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval of the waste	6×10^{-2}	9×10^{-1}	1×10^{2}	2×10^{3}
Future Tank Waste	Pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval of the waste	9×10^{-2}	9×10^{-1}	3×10^{2}	2×10^{3}
Strontium and Cesium Capsules	Rupture of strontium capsule by improper handling during retrieval operations	2×10^{-7}	3×10^{-6}	6×10^{-4}	1×10^{-2}
TRU-Contaminated Soil	Collapse of voids in the soil site during site stabilization activities	2×10^{-8}	9×10^{-7}	5×10^{-5}	2×10^{-3}
Pre-1970 TRU Solid Waste	Collapse of voids in solid waste site during site stabilization	3×10^{-7}	7×10^{-6}	6×10^{-4}	2×10^{-2}
Retrievably Stored and Newly Generated TRU	Collapse of void space at waste site during site-stabilization activities	5×10^{-6}	6×10^{-5}	1×10^{-2}	2×10^{-1}

3.0 TECHNICAL APPROACH AND METHODS

3.1 TECHNICAL APPROACH

The potential radiological impacts of the proposed waste handling and processing techniques covered in this study are the doses to humans from the airborne release of radionuclides by postulated severe accidents and abnormal operations. This report assumes that the most significant hazard (inhalation, skin dose, immersion dose) results from the sudden airborne release of radionuclides. In order for the airborne release of material to be estimated, the level of force exerted by the event, the physical and chemical form of the radionuclides involved, the mechanism that deagglomerates or subdivides and disperses the radionuclide, the degree to which the released material is contained and deposited prior to release to the environs must all be considered.

Accidents are defined as credible situations which create demand upon the system beyond the capability of the process, equipment, or containment features. Because of the conservative nature of the study approach, mitigation by standby or engineered safety features is not considered. 'Credible,' in this case, signifies situations which cannot be eliminated by the design of existing systems and processes or laws of physics or chemistry under the stated conditions. Abnormal operations are defined as events resulting from malfunctions of the systems, improper operating conditions or operator errors. It is not necessary for this study to classify the events covered into these categories (clear separations may not always be possible); the important fact is to estimate the airborne releases from such events.

The estimate of the conditions generated and the quantity of radionuclides airborne are based upon scenarios (sequences of events and their consequences). Scenarios were chosen from accidents and abnormal operations that have occurred in waste handling/processing facilities used in the past or from reports of the consequences of potential accidents and abnormal events for proposed similar waste operations (Hayward and Jensen 1980, DOE 1982, Murphy and Holter 1980, Richardson 1980).

A conservative approach (i.e., one that is believed to overstate rather than understate the consequences) is used to estimate the potential airborne releases.

The quantity of radioactivity released with the airborne materials is estimated using the radionuclide inventories associated with the specific waste form involved in the operations covered. The reference radionuclide inventories for the waste classes were reported by Rockwell Hanford Operations (Rockwell 1985). Upper-bound reference radionuclide inventories used to assess the potential radiological impacts are based upon estimates of the most significant inventories (the largest quantity or highest concentration) which could be present or actual measurements available of these values.

Potential downwind transport and dose were evaluated using established Hanford transport and dose models (Section 9.0).

3.2 KEY ASSUMPTIONS

The following key assumptions were made for this analysis:

1. All facilities, processes, and operations are or will be designed, constructed, and used in a manner consistent with prudent and proven practices.
2. The processes, facilities, operations, radionuclide inventories, and waste forms are those described in the engineering support data (Rockwell 1985).
3. The upper-bound accident identified as having the greatest potential radiological consequences for a given operation is assumed to conservatively bound (upper limit) all other credible accidents that could occur during that particular operation.

3.3 METHODS

A great deal of information is required to make a precise estimate of the potential airborne release of radionuclides from an event. Some, but not necessarily all, types of information required are: description of the process and equipment involved, including flows, capacities, and auxiliary systems; the

characteristics of the materials involved (both the material being processed and the materials used in the process); the physical configuration and conditions under which the material is processed (cell and building characteristics and capabilities, building and cell systems such as off-gas and ventilation); the services and power used and the purpose for which they are used; control systems and the consequences of loss of control; the reliability of the equipment, process, and facility; the behavior of all materials involved to various levels of stress imposed (e.g., heat, pressure, and shock); and the types and levels of stresses that could make the material of concern airborne and compromise the integrity of all containment barriers.

The various techniques under consideration to process the six waste types into a final disposal form range from those currently in use to those in the preconceptual stage. The quantity of information available ranges accordingly. Thus, neither the amount of information nor the time available allows rigorous adherence to the outline given above for the assessment of the potential consequences from accidents and abnormal operations. The approach taken in this study was to search for published reports describing similar operations and to use the information of the consequences presented. When such information was available, the values given were used without further evaluation. In some instances, scenarios describing the postulated events were not given, and the applicability of the information to the operations covered in this report could not be evaluated.

Other events not covered in published reports were also included as they were suggested to the authors based on their experience in consequence assessment. If information on the consequences was not in the published literature available to the authors, assessment was made using their knowledge of the situations and experimental data on the airborne release of material under similar levels of stress. When analysis indicated that no significant quantity of radionuclides was released as a consequence of an event, that fact was so stated. Finally, when an estimate of the airborne release could not be made due to the lack of information, a method of evaluating the situation was outlined.

The scenarios selected for estimating the potential upper-bound radiological impact by a given operation tend to be those which result in large energy releases and involve radionuclides in a dispersible form. This is not surprising since energy is required to subdivide and disperse or to deagglomerate and disperse the radionuclides and also to compromise the integrity of the containment and barriers which normally prevent the entry of radionuclides into the environs. Events with potential for large energy releases, such as fires and explosions, are prime candidates. The level and type of stress over a given time period determines which materials will react in what way, what barriers/containment may survive, and which engineered safety features can function. Such mitigating features are considered where possible. The containment feature most susceptible to damage from energetic events is the high-efficiency particulate air filters, which usually are the last barrier to release to the atmosphere and are relatively fragile. (They are also often a great distance from the source of energy and protected by engineered safety features to prevent damage.) Operations involving uncontained, subdivided radioactive materials (i.e., powders) or performed outdoors with no containment were also prime candidates for upper-bound events.

4.0 WASTE FORMS AND DISPOSAL ALTERNATIVES

The study focused on the potential upper-bound radiological impacts of operational accidents during the operations proposed for three possible disposal alternatives performed on six waste forms. The six waste forms and three disposal alternatives are described in this section. A baseline case, continued storage, which consists of monitoring and maintaining the current waste storage sites without taking any disposal actions (sites are not stabilized, barrier and marker system is not installed, etc.) is also addressed.

4.1 WASTE FORMS

There are six waste forms: existing tank waste, future tank waste, strontium and cesium capsules, TRU-contaminated soil, pre-1970 solid waste, and retrievably stored and newly generated TRU solid waste. They are described in the Rockwell engineering packages (Rockwell 1985). Most of the forms are described as of January 1984 except those which do not currently exist, e.g., future tank waste, newly generated TRU solid waste.

4.1.1 Existing Tank Waste

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

1) sludge 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 (the supernatant liquid after salt cake formation)
4) complex concentrate produced by concentration of waste containing large amounts of organic complexing agents.

The first two classes of waste are stored primarily in the older (single-shell) tanks; the latter two in the newer (double-shell) tanks. Some supernatant liquid, contained in the older tanks, is being transferred as completely as possible to new tanks. The volumes and compositions of waste in individual tanks vary considerably, depending on the source of the waste and on past waste management practices at the respective tank farms. The processing schemes proposed to dispose of the type of waste depend upon the storage mode, radionuclide content, thermal release due to radioactive decay, and the chemical and physical form of the waste (ease of retrieval and processing).

4.1.2 Future Tank Waste

Future tank waste includes PUREX plant waste generated during the current operations, which began in November 1983, together with liquid wastes from other sources (including liquid waste projected for the operations of the Plutonium Finishing Plant) projected through 1995. All these wastes are stored in double-shell tanks. The sources and composition of future tank waste are described below.

4.1.2.1 Neutralized Current Acid Waste

High-level wastes from PUREX operations will be neutralized and stored. The neutralized current acid waste would be separated into two phases: 1) a solid phase (a suspended sludge of insoluble materials) primarily consisting of hydroxides or hydrated oxides 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 the nitric acid by sodium hydroxide), sodium nitrite (resulting from the radiolytic reduction of the sodium nitrate), sodium sulfate (resulting from the conversion of sulfamate to sulfate), sodium aluminate (resulting from the aluminum additions to complex fluoride ions), and sodium hydroxide.

The sludge would contain most of the fission products (except cesium and technetium) and unrecovered TRU. The supernatant liquid contains most of the cesium and technetium, the iodine not removed in the head-end process or decayed, and some of the ruthenium. The two-phase slurry cannot be economically vitrified since the sulfate limits the quantity of waste that can be

incorporated into the borosilicate glass. The sludge is separated from the supernate and washed free of sulfate (and also the soluble aluminate), minimizing the quantity of glass generated.

4.1.2.2 Cladding Removal Waste

The zircaloy cladding on N-Reactor fuels is dissolved chemically in the PUREX plant by reaction with an aqueous ammonium fluoride solution containing ammonium nitrate to suppress the evolution of hydrogen. The dissolvent slowly attacks the uranium metal as it becomes exposed after cladding removal. Solids and liquid are separated by centrifugation. A portion of the resulting uranium fluoride (with the actinide and fission product associated with it) not removed by centrifugation remains with the waste. Neutralization removes essentially all the TRU and fission products from solution. The supernatant liquid is thus non-TRU, low-level waste.

4.1.2.3 Organic Wash Waste

As part of the PUREX process, the organic solvent is washed to remove degraded organics that could otherwise interfere with the process. The wash solution contains sodium carbonate and potassium permanganate. Trace quantities of metal ions are removed from the organic solvent by this washing. As the aqueous wash becomes ineffective, the depleted aqueous wash solutions are combined with cladding removal waste for storage in double-shell tanks.

4.1.2.4 Plutonium Finishing Plant Waste Composite

Plutonium finishing plant waste composite is assumed to be a blend of waste from Z-plant (PUREX product metal reduction plus ash and oxide scrap recovery), S Area laboratory waste, and T-Plant waste. This excludes any special waste streams resulting from isotope separation efforts.

4.1.2.5 Miscellaneous Wastes

A variety of low-level wastes are included in this category. Some are additional wastes arising from PUREX operations, such as ammonia scrubber wastes, miscellaneous sump waste, and low-level waste from the latter portions of the process. Others are low-level wastes from operations elsewhere in the

200 Area (B-Plant, T-Plant, etc.). The scrubber and sump wastes are combined with the cladding removal waste, and the 200-Area wastes with the low-level wastes from the latter portions of the PUREX operations.

Other low-level wastes are generated by operations in other areas of Hanford. These and the low-level 200-Area wastes contain low concentrations of chemicals and radionuclides with large volumes. These two groups of low-level liquid waste will be concentrated by a factor of ten, and, when concentrated, may be classified as TRU waste.

4.1.3 Strontium and Cesium Capsules

Most of the high-heat generating fission products, ^{90}Sr and ^{137}Cs, have been extracted from the high-level waste and encapsulated in seal-welded, high-integrity double-wall metal capsules (see Figure 4.1) as strontium fluoride and cesium chloride. Capsule parameters are listed in Table 4.1. The capsules are currently stored in shielded water-filled basins for dissipation of decay heat and reduction of exposure to operating personnel. This method of storage provides multiple containment of the radionuclides by the double-wall capsules, water basin, reinforced concrete building, and a directional air system providing filtration by multiple stages of high-efficiency particulate air filters. In the event of a capsule failure, the damaged capsule can be returned to the process cell, re-encapsulated, and returned to the water basin. Contaminated water can be processed to remove the radionuclides and returned to service. Storage of the strontium and cesium capsules will continue in the existing water basins until a disposal or long-term storage alternative is selected.

4.1.4 TRU-Contaminated Soil

This waste form is primarily composed of systems formerly used to discharge TRU-contaminated liquids to Hanford soils and some incidental sites contaminated by other means. The types of systems used to discharge TRU-contaminated liquids include:

- cribs - buried structures (often wood or concrete) filled with aggregate that holds or disperses liquids and/or solutions for percolation into the ground

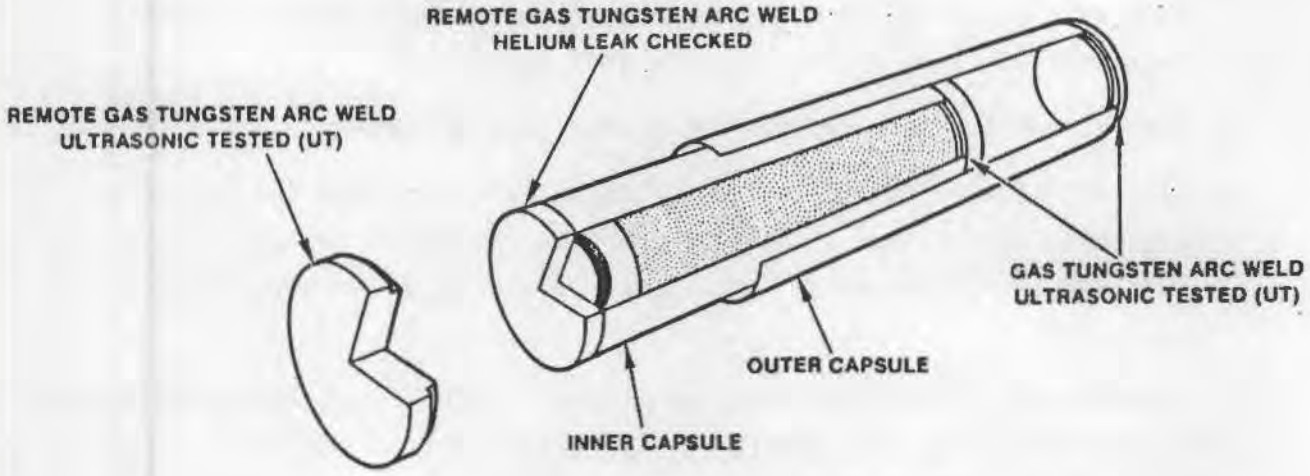


FIGURE 4.1. Strontium or Cesium Capsule

TABLE 4.1. Strontium and Cesium Capsule Parameters

Capsule	Containment Portion	Construction Material	Dimensions, cm Wall Thickness	Outside Diameter	Total Length	Total Cap Thickness
Strontium Fluoride	Inner	Hastelloy, C-276	0.305	6.72	43.39	1.02
	Outer	Stainless Steel, 316L	0.277	6.67	51.05	1.02
Cesium Chloride	Inner	Stainless Steel, 316-L	0.241	5.72	50.10	1.02
	Outer	Stainless Steel, 316-L	0.277	6.67	52.77	1.02

- ponds - surface depressions bordered by natural or manmade features used to contain and detain the liquid
- 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 wastes

- 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
- reverse wells - well casings going deep into the ground
- settling tanks - single-wall underground tanks or sumps made of concrete into which liquid waste is pumped. The solids settle in the tank and liquids overflow into a reverse well or other underground structure.

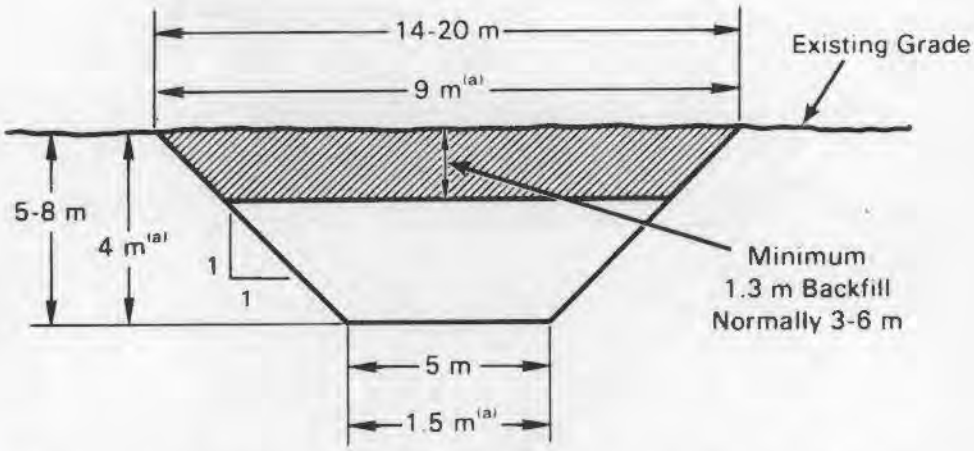
Movement of radionuclides into and through soil varies considerably and depends upon soil chemistry (Murthy et al. 1983). In the absence of complexing agents, the bulk of TRU elements and compounds move through the soil only slightly and are held (within a few meters) of the point of discharge.

4.1.5 Pre-1970 TRU Solid Waste Burial Ground

Between 1944 and 1970, TRU-contaminated waste (soiled clothing, laboratory supplies, tools, etc. packed in cardboard, wood or metal containers) was buried in "alpha" trenches. An "alpha" trench is an excavation in the ground 5 to 8 m deep with sloped sides and a minimum of 1.3 m of overburden (normally the depth of the overburden is 3 to 6 m deep). A schematic of such an arrangement is shown in Figure 4.2. A burial ground is defined as a TRU solid waste burial ground if the concentration of some containers at that location is estimated to exceed 100 nCi TRU/g (based upon a soil density of 1.8 g/cm^3 and a peak-to-average concentration of 10:1).

4.1.6 Retrievably Stored and Newly Generated TRU Solid Waste

TRU waste generated since 1970 has been retrievably stored. If the surface dose rate exceeded 200 mR/hr, the waste is classified as remote-handled and is either stored in caissons (see Figure 4.3) or packaged for direct shipment offsite. If the TRU waste was unsuitable for asphalt pad or caisson storage because of size, chemical composition, security requirements, or surface radiation, it was packaged in reinforced wooden boxes, concrete or metal boxes and stored in an alpha trench (see Section 4.1.5).



(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 4.2. Typical Solid Waste Burial Trench

TRU asphalt pad storage is shown schematically in Figure 4.4. Most of the waste is packaged in 55-gal metal drums. The containers are covered with plywood, plastic-reinforced nylon sheeting, and a 1.2-m layer of uncontaminated soil to reduce surface radiation exposures to less than 1 mR/hr. Ventilation is provided to reduce humidity and resultant rusting of the drums.

Newly generated TRU waste in approved packaging maybe be temporarily stored on pads with appropriate coverings.

4.2 DISPOSAL ALTERNATIVES

Three processing schemes or disposal alternatives to permanently dispose of the Hanford Defense Waste are under consideration. The three alternatives are labelled 'geologic disposal alternative', 'in-place stabilization and disposal alternative,' and 'reference alternative.' Basically, the geologic disposal alternative places most of the waste in a deep geologic repository (only the clearly low-level waste is grouted). The in-place stabilization and disposal alternative leaves most of the waste at its current sites, isolated from the ecosystems, but the high-level waste in the double-shell tanks is processed

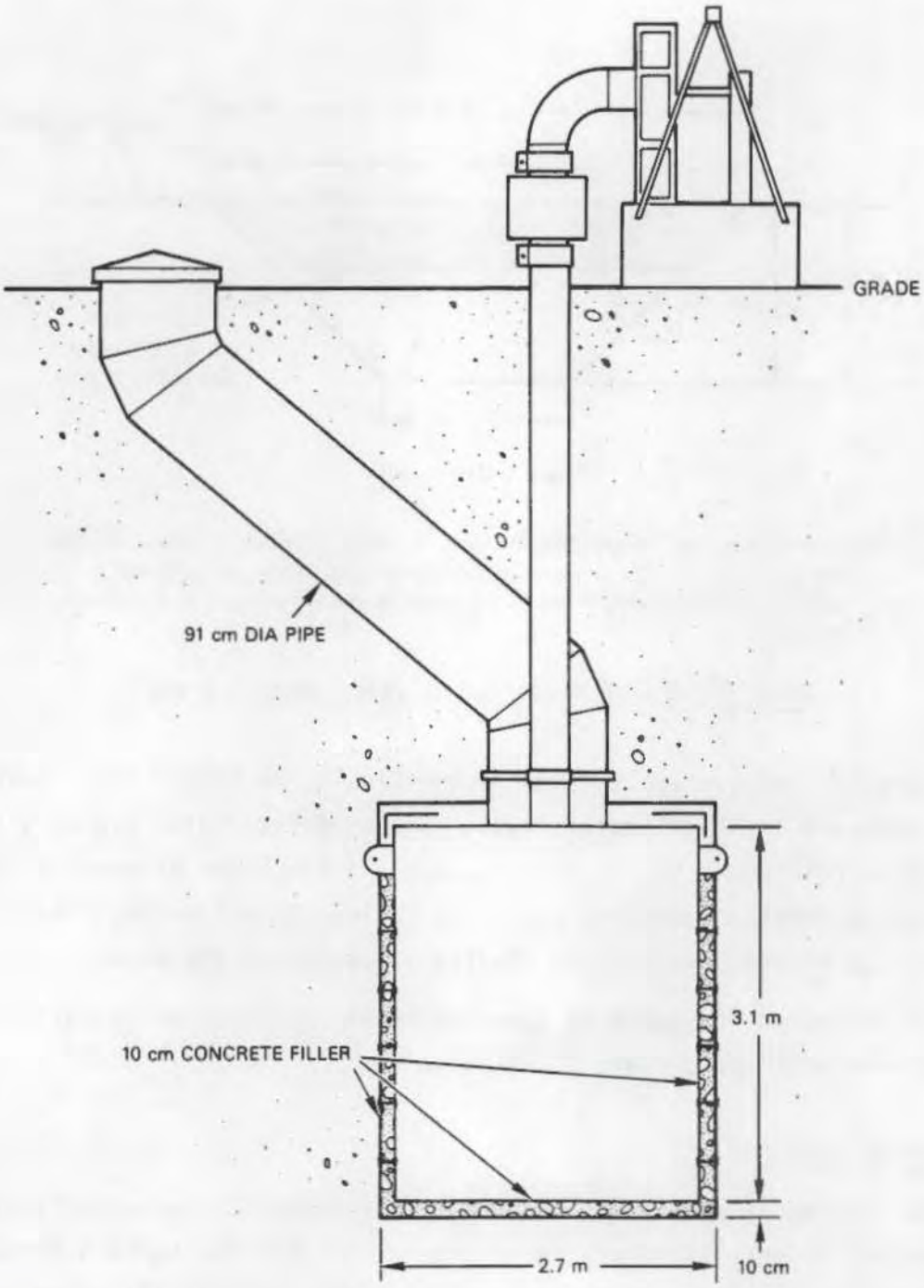


FIGURE 4.3. Caisson for TRU Storage

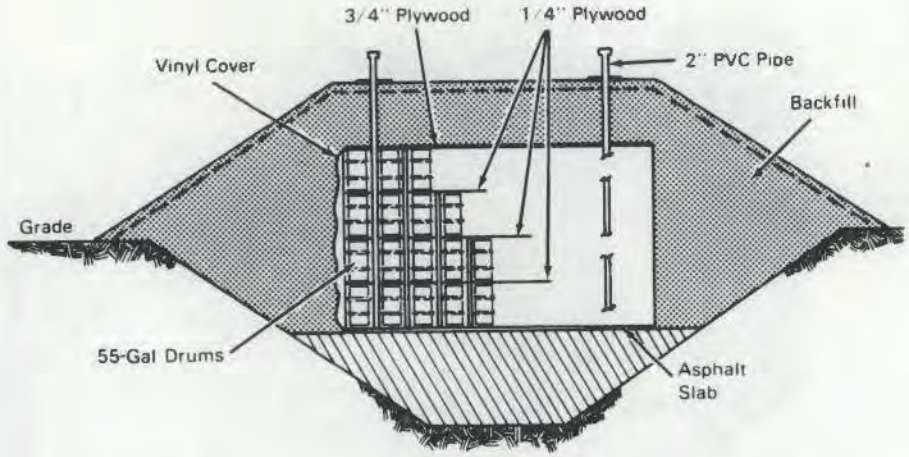


FIGURE 4.4. TRU Asphalt Pad Storage (ERDA 1975)

to remove the high-heat-generating radionuclides. In the reference alternative, the operations are chosen to be performed on the waste forms to balance the near- and long-term risks from processing or leaving the material in place. The three disposal alternatives are discussed in greater detail below.

4.2.1 Geologic Disposal Alternative

This alternative involves the retrieval, processing, segregation, packaging and placement of most waste in a deep geologic repository with some near-surface disposal. The location of the repository and its function, i.e. whether dedicated to defense waste or co-mingled with commercial wastes, are not known at this time but are unimportant for this analysis. A conceptual deep geologic repository is shown in Figure 4.5.

Under this alternative, some of the waste (e.g., existing tank waste described earlier in this section) is divided into high-level and low-level fractions. The high-activity, low-volume fraction is vitrified as borosilicate glass, packaged and deposited in a deep geologic repository. The larger low-activity fraction is made into a suitable cement (grout) and near-surface disposal is used. A schematic illustrating the various operations proposed for the six waste forms in this alternative is shown in Table 4.2.

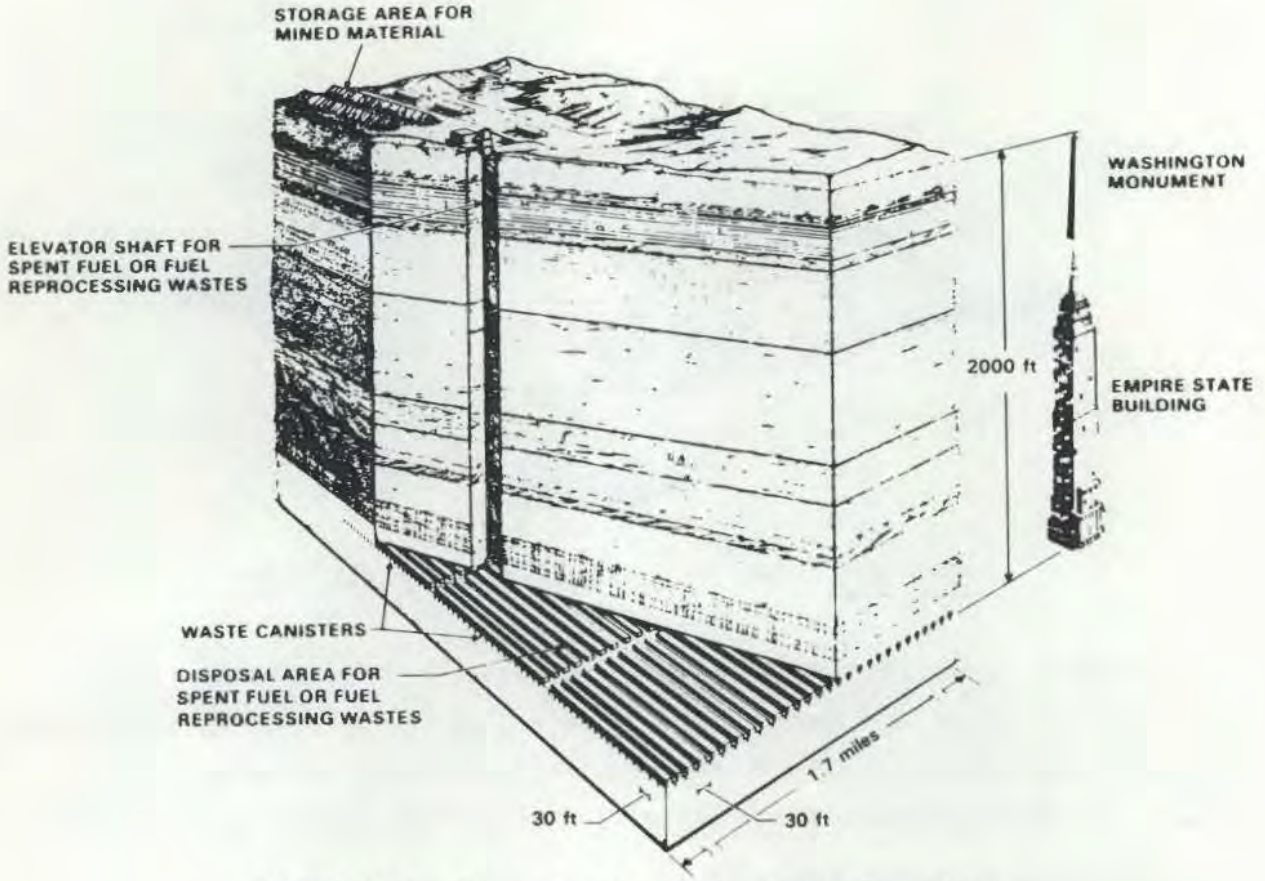


FIGURE 4.5. Conceptual Deep Geologic Repository

4.2.2 In-Place Stabilization and Disposal Alternative

The concept of in-place stabilization and disposal is to stabilize in-place all high-level and TRU waste at Hanford using a protective barrier and marker system to isolate the disposed materials from all ecosystems. An application of the barrier and marker system applied to one waste type, single-shell tank waste, is illustrated in Figure 4.6.

Very little processing is envisioned in this alternative except for the waste stored in double-shell tanks. Removal of the high-heat-generating radionuclides (i.e., cesium) to permit formulation of a suitable grout may be required. The cesium removed is encapsulated and handled in the same manner as the existing encapsulated materials (near-surface Drywell Storage Facility). Waste in the single-shell tanks would be dried, and interim heat removal systems would be provided as needed. Waste in all TRU sites (soil, pre-1970

TABLE 4.2. Waste Processing Steps for Geologic Disposal

Existing Tank Waste	Future Tank Waste	Strontium and Cesium Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU-Contaminated Solid Waste	Retrievably Stored and Newly Generated TRU Wastes
Retrieve at least 95% of single-shell tank contents and 99.95% of double-shell tank contents.	Retrieve 99.95% of wastes in double-shell tanks. Separate strontium, cesium, technetium, and TRU elements for vitrification and geologic disposal.	Package capsules in canisters for repository disposal.	Recover TRU waste to 100-nCi/g level.	Recover wastes defined as TRU.	Retrieve TRU waste and sort and package it to WIPP waste acceptance criteria.
↓	↓	↓	↓	↓	↓
Separate strontium, cesium, technetium, and TRU elements for vitrification and repository disposal. Send residual low-level wastes as grout to near-surface trenches or to existing tanks.	Immobilize the residual waste as grout and dispose of in near-surface trenches or in existing tanks.	Send canisters to geologic repository for disposal.	Treat, package, and send waste to Waste Isolation Pilot Plant (WIPP).	Treat, package, and send waste to WIPP.	Package and treat waste as required, and send it to WIPP.
↓	↓		↓	↓	↓
Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.	Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.		Backfill excavations and dispose of as low-level waste sites.	Backfill excavations and dispose of as low-level waste sites.	Dispose of residual waste as low-level waste.
↓	↓				
Leave soil contaminated from tank leaks in place.	Cover grout disposal waste sites and tanks with the protective barrier, emplace markers, and record.				
↓					
Cover emptied tank farms and grout disposal sites with the protective barrier, emplace markers, and record.					

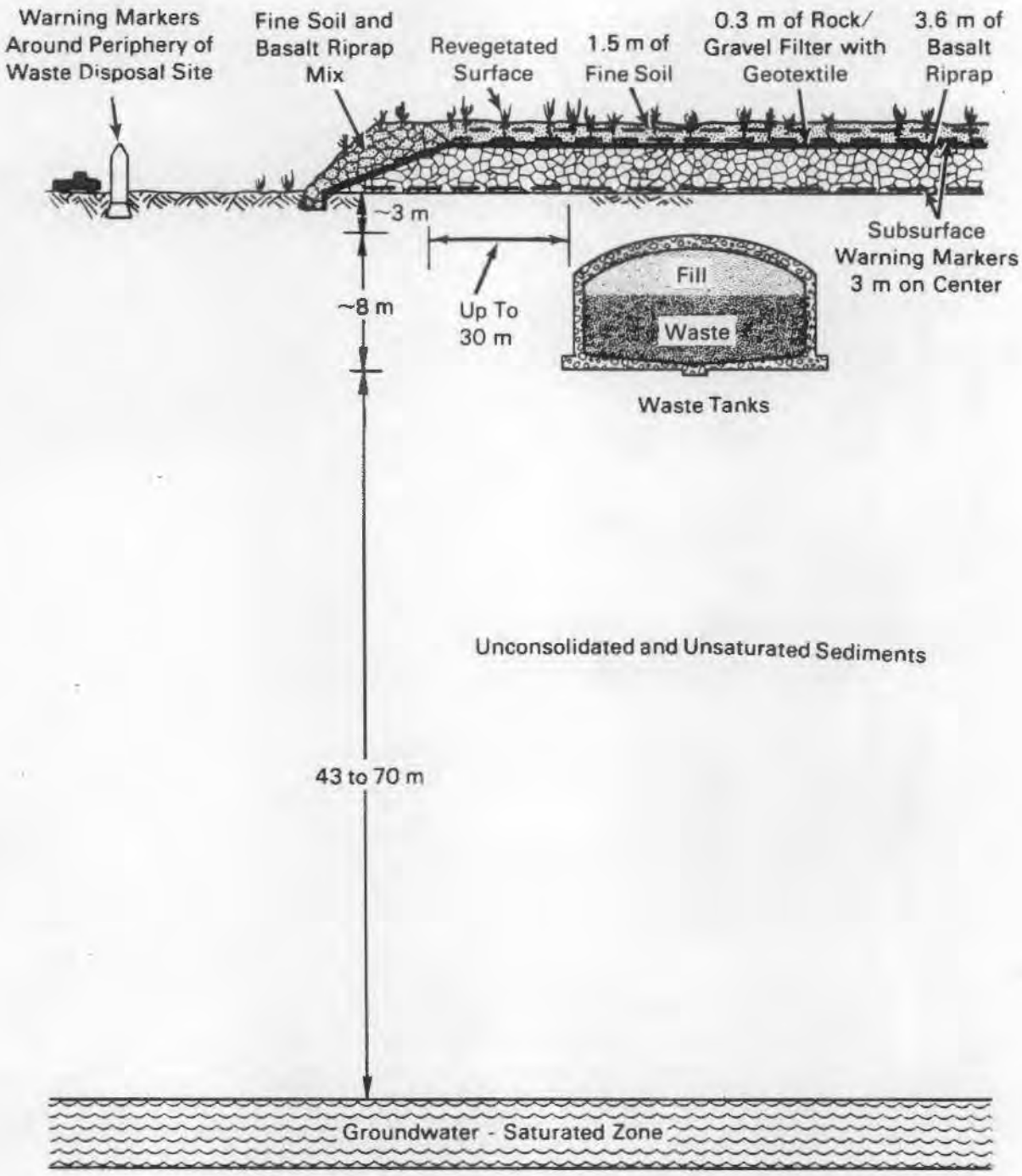


FIGURE 4.6. Schematic of Protective Barrier as Applied to Single-Shell Tank Waste

solid waste sites, and sites with retrievably stored or newly generated waste) would not be relocated but interred in place. All sites would be treated for subsidence control as needed and covered with a protective barrier and marker system. Although in-place stabilization would be considered permanent disposal, it does not preclude future generations from removing the waste. The operations proposed for the six waste forms for this alternative are shown schematically in Table 4.3.

4.2.3 Reference Alternative

The concept directing the reference alternative is to provide a balanced approach that would give a reasonable expectation that the long-term risks are limited without incurring unacceptable near-term risks (disturbing currently stable and/or difficult-to-retrieve waste sites). In a sense, this alternative is a combination of the two previously described alternatives, choosing the optimal processing scheme for each waste class. Geologic disposal would be used for the strontium and cesium capsules, for the high-activity portions of existing double-shell tank and newly generated tank wastes, and for most retrievably stored and newly generated TRU waste. Other waste that is not readily retrieved and for which the short-term environmental risks may outweigh the benefits of deep geologic disposal would be disposed of by in-place stabilization. The operations proposed for this alternative on the six waste forms are shown schematically in Table 4.4.

4.2.4 No Disposal Alternative

A no disposal action alternative, which amounts to continued storage of the wastes, was also considered in detail. In the short term (i.e., for periods less than 100 years), the no disposal action alternative can be considered as a "delay major action" alternative, after which time disposal alternatives could be considered. If DOE chooses the no disposal action alternative, waste would remain as disposed of or continue to be stored indefinitely using existing storage practices with planned improvements. The operations proposed for this alternative are shown in Table 4.5. Active

TABLE 4.3. Waste Processing Steps for the In-Place Stabilization Alternative

Existing Tank Waste	Future Tank Waste	Strontium and Cesium Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU-Contaminated Solid Waste	Retrievably Stored and Newly Generated TRU Wastes
Leave waste in single-shell tanks. Retrieve 99.95% of waste from double-shell tanks.	Retrieve 99.95% of wastes in double-shell tanks. Separate and encapsulate cesium, and dispose of as outlined for the strontium and cesium capsules, right.	Package capsules in canisters for near-surface disposal onsite.	Wastes remain dispposed of in place.	Wastes remain disposed of in place.	Dispose of stored TRU waste in place. (First remove building-stored waste to burial ground.)
↓	↓	↓	↓	↓	↓
Treat double-shell tank waste as necessary to immobilize in grout; place grout in near-surface trenches or in existing tanks.	Immobilize the residual waste as grout and dispose of in near-surface trenches or in existing tanks.	Place canisters in near-surface caissons after cooling.	Fill voids with grout to limit subsidence.	Compact in-place, disposed wastes as needed. Fill voids in old TRU caissons and other sites with grout to control subsidence.	Compact waste, fill voids with grout as needed for subsidence control, provide the protective barrier, emplace markers, and record.
↓	↓	↓	↓	↓	
Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.	Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.	Cover canister disposal area with the protective barrier, emplace markers, and record.	Cover all sites with the protective barrier, emplace markers, and record.	Cover all TRU buried solid waste sites with the protective barrier, emplace markers, and record.	
↓	↓				
Leave soil contaminated from tank leaks in place.	Cover grout disposal waste sites and tanks with the protective barrier, emplace markers, and record.				
↓					
Cover tank farms and grout disposal sites with the protective barrier, emplace markers, and record.					

TABLE 4.4. Waste Processing Steps for the Reference Disposal Alternative

Existing Tank Waste	Future Tank Waste	Strontium and Cesium Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU-Contaminated Solid Waste	Retrievably Stored and Newly Generated TRU Wastes
Leave waste in single-shell tanks. Retrieve 99.95% of waste from double-shell tanks.	Retrieve 99.95% of wastes in double-shell tanks. Separate high-level and TRU waste for vitrification and geologic disposal.	Package capsules in canisters for repository disposal.	Wastes remain disposed of in place.	Most wastes remain disposed of in place. Retrieve TRU sites not on 200 Area plateau.	Retrieve TRU waste and sort and package it to WIPP waste acceptance criteria.
Separate the TRU and high-level fraction of the double-shell tank wastes, immobilize the low-level as glass, and dispose of in a repository. Immobilize the non-TRU fraction of double-shell tank waste as grout, and dispose of in near-surface trenches.	Immobilize the residual waste as grout and dispose of in near-surface trenches or in existing tanks.	Send canisters to geologic repository for disposal.	Fill voids with grout to limit subsidence.	Compact in-place, disposed wastes as needed. Fill voids in old TRU caissons and other sites with grout to control subsidence.	Package and treat waste as required, and send it to WIPP.
Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.	Fill tanks sufficiently with appropriate material to limit subsidence, and seal all accessible penetrations.		Cover all sites with the protective barrier, emplace markers, and record.	Cover all TRU buried solid waste sites with the protective barrier, emplace markers, and record.	Dispose of residual waste as low-level waste.
Leave soil contaminated from tank leaks in place.	Cover grout disposal waste sites and tanks with the protective barrier, emplace markers, and record.			As required, package and treat the retrieved waste and send it to WIPP.	
Cover tank farms and grout disposal sites with the protective barrier, emplace markers, and record.					

TABLE 4.5. Implementation of the No Disposal Action Alternative

Existing Tank Waste	Future Tank Waste	Strontium and Cesium Capsules	TRU-Contaminated Soil Sites	Pre-1970 Buried TRU-Contaminated Solid Waste	Retrievably Stored and Newly Generated TRU Wastes
Leave waste in single-shell tanks; retank double-shell tank waste every 50 yr.	Retank double-shell waste every 50 yr.	Package capsules in canisters as necessary for onsite dry storage.	Leave sites as disposed of.	Leave sites as disposed of.	Leave waste as stored.
↓	↓	↓	↓	↓	↓
Monitor and maintain tanks, filling single-shell tank domes and unused old double-shell tank domes as required to prevent collapse and maintain surface.	Monitor and maintain tanks, filling unused, old double-shell tanks as required to prevent collapse.	Store canisters in near-surface caissons.	Monitor sites and continue site maintenance.	Monitor site and continue site maintenance.	Monitor waste and continue to maintain.
↓		↓			
Leave soil contaminated from tank leaks in place.		Monitor caissons and continue caisson maintenance.			

administrative control would be provided. Federal ownership and presence on the Hanford Site is planned in perpetuity, but for comparative analyses loss of active institutional control is assumed to occur in the year 2150. It must be emphasized that this scenario was defined simply for comparing alternatives.

5.0 POSTULATED RELEASES FROM POTENTIAL ACCIDENTS ASSOCIATED WITH THE GEOLOGIC DISPOSAL OF SIX WASTE FORMS

Several operations are required to process each of the six waste forms for geologic disposal. This section contains a brief description of each waste processing operation used and the facility involved. Upper-bound release is then estimated, and other potential accidents are discussed. The capability of the facility in which an operation is presumed to occur and the transport/mitigation before the release of the airborne material to the environment are also included. Upper-bound releases are listed in Tables 5.1, 6.1, 7.2, and 8.1 for the geologic, in-place stabilization and disposal, reference, and no disposal action alternatives, respectively.

5.1 EXISTING TANK WASTE

Tank wastes are retrieved, the strontium, cesium, technetium, and TRU are separated for vitrification, and residual wastes as grout are sent to low-level waste burial grounds or tanks. The tanks are filled with appropriate material and sealed, soil contaminated from tank leaks is left in place, and the tank farms are covered with soil.

5.1.1 Mechanical Retrieval

This operation retrieves salt cake and sludge from single-shell tanked waste without direct addition of water (Figure 5.1).

5.1.1.1 Description of Operation and Facility

The mechanical retrieval operation is designed to occur with minimum alterations to the tank dome structure and to the tank farms in general, and is structured to avoid direct loads to the dome. The conceptual recovery process is composed of three sequential operations:

- in-tank recovery of waste
- removal of waste from the tank to a transfer point for emplacement in a shipping container
- transfer of waste to an onsite processing facility.

TABLE 5.1. Geologic Disposal Alternative Potential Accidental Releases for Operations Involving Six Waste Forms

	Waste Form	Accident Source Quantity	Facility	Accident	Location	Fractional Release	Transmission Factor[a]	Atmospheric[b] Release of Concern, g	Release Point
Existing Tank Waste									
1. Mechanical retrieval	Salt cake	2000 m^3	Mobile platform	Explosion	Waste Tank	NA	1	1.3×10^4	Ground level
2. Hydraulic retrieval	Salt cake	9×10^4 g/min	NA[c]	Pressurized release	Diversion valve	5×10^{-2}	1	4.5×10^3	Ground level
3. Sr, Cs, Tc removal (radionuclide concentration)									
3.1 Sludge washing, solid/liquid separation	sludge	0.2 g/min	Sludge washing canyon	Filter failure	Filters	NA	1	0.2	Stack
3.2 Complexant destruction	No information available								
3.3 Radionuclide removal	Ion exchange resin		Radionuclide concentration	Ion exchange fire	Resin column	0.01	3×10^{-4}		Stack
4. Glass immobilization (vitrification)	Molten glass	0.2 g/min	Glass immobilization	Loss of filters	Filters	NA	1	0.2	Stack
5. Grout decontaminated salt solutions	Decontaminated salt solutions	10 m^3	Transportable grout	Liquid spray from line	Transfer line	1×10^{-4}	2.5×10^{-7}	3×10^{-4}	Stack
6. Fill and cover empty tank (dome fill)	Residual tank waste	5% of original fill	None	Dome collapse	Tank	5×10^{-2}	1	350	Ground level
Future Tank Waste									
1. Retrieval	NCAW[d]	9×10^4 g/min	NA	Pressurized release	Diversion valve	5×10^{-2}	1	4.3×10^3	Ground level
2. Solid/liquid separation	NCAW	0.2 g/min	Sludge washing canyon	Filter failure	Filters	NA	1	0.2	Stack
3. Sr, Cs, Tc, TRU removal	Ion exchange resin		Radionuclide concentration	Ion exchange fire	Resin column	0.01	3×10^{-4}		Stack
4. Grout decontaminated liquid	Decontaminated liquid	10^3	Transportable grout	Liquid spray from line	Transfer line	1×10^{-4}	2.5×10^{-7}	3×10^{-4}	Stack
5. Vitrification	Molten glass	0.2 g/min	Glass immobilization	Loss of filters	Filters	NA	1	0.2	Stack
6. Fill and cover empty tank (dome fill)	Residual tank waste	0.05% of original fill	None	Dome collapse	Tank	5×10^{-4}	1	3.5	Ground level
Strontium and Cesium Capsules									
1. Remove from WESF[e]	Encapsulated waste	2.2×10^3 g	WESF[e]	Capsule drop	Storage area	1×10^{-2}	2.5×10^{-7}	5.5×10^{-6}	Stack
2. Capsule packaging	Encapsulated waste	2.2×10^3 g	CPF[f]	Machinery impacts capsule	"Load" station	1.2×10^{-3}	2.5×10^{-7}	6.6×10^{-7}	Stack

TABLE 5.1. (contd)

	Waste Form	Accident Source Quantity	Facility	Accident	Location	Fractional Release	Transmission Factor [a]	Atmospheric[b] Release of Concern, g	Release Point
TRU-Contaminated Soil									
1. Retrieve	Contaminated soil		Retrieval facility	Explosion	Battery charging area		1	5	Ground level
2. Process	Slagging pyrolysis off gas		SPI[g]	Explosion	Gasifier		1	50	Ground level
Pre-1970 TRU Solid Waste									
1. Retrieve	Contaminated soil/waste		Retrieval facility	Explosion	Battery charging area		1	5	Ground level
2. Sorting and related operations	Packaged waste	2×10^5 g	SPI	Pressurized release	Drum	0.01	2.5×10^{-7}	5×10^{-4}	Stack
3. Process	Slagging pyrolysis off gas		SPI	Explosion	Gasifier		1	50	Ground level
Retrievably Stored and Newly Generated TRU									
1. Release									
1.1 RH TRU[h]	Packaged waste	3.7×10^3 g	Caisson retrieval	Pressurized release	Metal can	0.01	2.5×10^{-7}	9.3×10^{-6}	Stack
1.2. CH TRU[i]	Packaged waste	2×10^5 g	None	Pressurized release	Drum	0.01	1	2×10^3	Ground level
2. Sorting and Related Operations									
2.1 RH TRU	Packaged waste	2×10^5 g	SPI	Pressurized release	Drum	0.01	2.5×10^{-7}	5×10^{-4}	Stack
2.2 CH TRU	Packaged waste	2×10^5 g	CH WRAP	Pressurized release	Drum	0.01	2.5×10^{-7}	5×10^{-4}	Stack
3. Process									
3.1 RH TRU	Slagging pyrolysis off gas		SPI	Explosion	Gasifier		1	50	Ground level
3.2 CH WRAP	Packaged waste	1×10^5 g	CH WRAP	Fire	Drum	5×10^{-4}	2.5×10^{-7}	2.5×10^{-5}	Stack

(a) A transmission factor of 1 means the high-efficiency particulate air filters are breached or the operation is not enclosed.
(b) Default assumption - if release is in g/min, a 1 minute release is assumed.
(c) Not Applicable.
(d) Neutralized Current Acid Waste.
(e) Waste Encapsulation and Storage Facility.
(f) Capsule Packaging Process.
(g) Slagging Pyrolysis Incinerator.
(h) Remote-Handled.
(i) Contact-Handled Waste Retrieval and Packaging Facility.

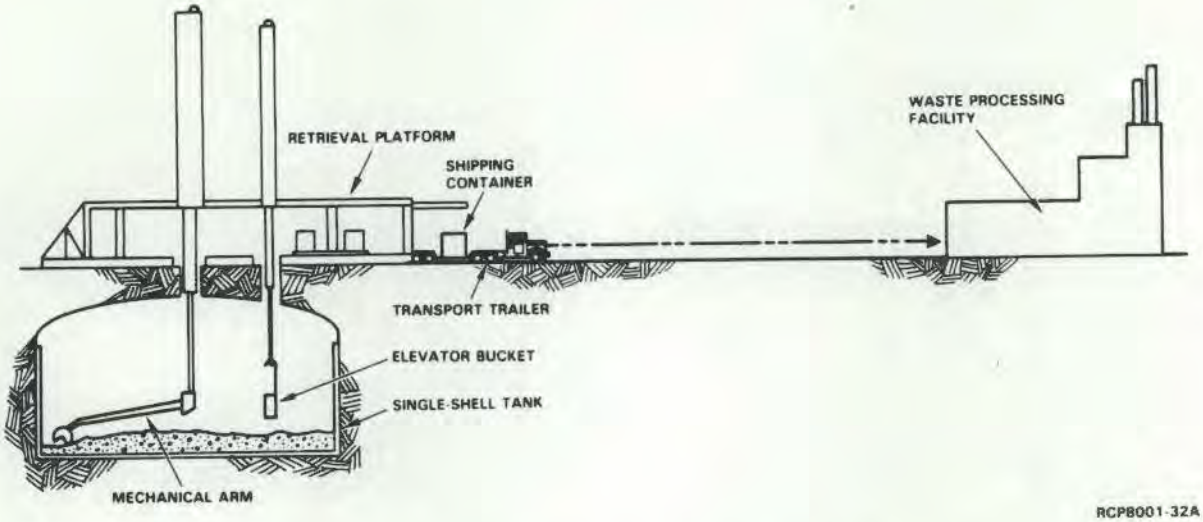


FIGURE 5.1. Mechanical Retrieval of Wastes from Single-Shell Tanks

Single-shell tanks are prepared by adding entry points (risers), if necessary, removing above-ground obstructions, and breaking up large encrustations of waste. The waste recovery, removal, containerization, and most of the supporting equipment are mounted on a movable platform. The mobile platform, sized to the approximate 31-m tank spacing, supports the waste-handling apparatus and contains most of the auxiliary systems necessary for safe retrieval of the waste. A hydraulically actuated articulating arm is positioned using a telescoping tube and carriage mechanism in a tower. The tank is maintained slightly below atmospheric pressure during these operations and ventilation air is discharged through two banks of high-efficiency particulate air filters.

The waste retrieval system recovers the waste mechanically with a clamshell bucket on the articulating arm and deposits the waste in the elevator bucket for transfer to the platform level. This recovered as-is waste is unloaded from the bucket elevator to a shielded shipping container that holds approximately 2.7 m^3 of waste. After being sealed and washed, the shipping container is placed in a clean, sealed container that is also sealed with a locking lid. A special tractor-trailer vehicle is used to transfer the container over a dedicated roadway to the onsite processing facility.

5.1.1.2 Postulated Upper-Bound Accident

Several sources for an explosive release of salt cake could be postulated during mechanical retrieval. The salt cake is composed of many salts; among these is sodium nitrate, a powerful oxidizer. Although experimental work (Beitel 1976) indicated that sodium nitrate can be considered stable below a temperature of 300°C, a potentially vigorous explosion might result if ferrocyanide precipitates were in the tank. Cyanide and nitrate ions could react violently during the heating and produce an explosion with the energy equivalent to 36 tons of TNT. This explosion would have an impact sufficient to breach the filters, thus releasing aerosol directly to the atmosphere.

5.1.1.3 Release Estimate for Upper-Bound Event 4.98E5 Kg = 498 t

Steindler and Seefeldt (1980) developed a method to predict aerosol production from a detonation, which is used to estimate this release. The source is 2000 m^3 of salt cake, which releases 4.98 x 10^8 g of aerosol directly to the atmosphere, including 1.3 x 10^4 g of respirable material 10 μm aerodynamic equivalent diameter and less. This latter value is the estimated release listed in Tables 5.1 and 5.2 and used for dose calculations.

$\frac{1.3E4}{4.98E8} = 2.6E-5$

5.1.1.4 Other Accidents Considered

Other accidents that appeared applicable to the mechanical retrieval were considered as the potential upper-bound release event. They are discussed briefly below and listed in Table 5.2.

Contaminated Soil Suspension During Sampling. It is believed that such an event would not result in any large, significant airborne release of radionuclides.

Retrieval Transporter Spill. During the mechanical retrieval, it is postulated that due to systems malfunction a shipping container fails to line up under the buck as the transfer operation proceeds. The entire elevator buck is filled with waste that can become airborne in the facility after a spill.

TABLE 5.2. Postulated Airborne Releases from Accidents During Mechanical Retrieval of Single-Shell Tank Waste

Event	Atmospheric Airborne Release
Explosion	1.3×10^4 respirable = 13 Kg
Contaminated soil suspension	Significantly below upper-bound
Retrieval transporter spill	Significantly below upper-bound
Waste spill	Significantly below upper-bound
Loss of filtration 1st stage Both stages	 5×10^{-6} g/m^3 1×10^{-2} g/m^3
Loss of services or power	No significant release

The fraction airborne under these circumstances can be estimated as 0.12%, based on experimental measurements of free-fall spill releases (Sutter, Johnston and Mishima 1981). It is assumed that the airborne material enters the transporter exhaust system and is filtered by two stages of high-efficiency particulate air filters before release to the atmosphere.

Waste Spill. If this event occurred on the platform level, the release would be that mentioned above. If the event occurred in the tank, then the release is postulated to be similar to that shown for the loss of filtration discussed below.

Waste Spill During Retrieval. It is postulated that material is dropped from the clamshell or elevator bucket as a result of equipment malfunction. The material is released in the waste tank and must therefore pass through two stages of high-efficiency particulate air filtration before release to the atmosphere. The exhaust rate through the tank is low; therefore it is not assumed that an inordinate amount of material would be made airborne. Based on aerosol behavior, a quasi-stable concentration of 100 mg/m^3 is assumed (ORNL 1970) with 10% (10 mg/m^3) as respirable particles less than 10 μm aerodynamic equivalent diameter.

Loss of Filtration. The quantity airborne during the mechanical retrieval operation of salt cake will be high due to the dusty nature of the operation. It is assumed that the mechanical subdivision of the salt cake will result in a coarse particle size distribution for the material airborne. A value of 100 mg/m^3 with 10% in the respirable size fraction is conservatively selected. Loss of one stage of filtration would increase the transmission factor of material challenging the filters to a 0.0005 fraction. Loss of both stages of filtration would increase the transmission factor to 1.

Loss of Services or Power. It is postulated that the only consequences of service or power loss would be the cessation of operations and that no significant airborne release of material would occur.

5.1.2 Hydraulic Retrieval

5.1.2.1 Description of Operation and Facility

This operation recovers and transfers radioactive liquids using multistage pumps, deep-well turbine pumps, and shielded piping as shown in Figure 5.2.

Slurries are removed from double-shell tanks with a sluicer such as the one shown in Figure 5.3. The sluicer is composed of two basic systems:

- the high-pressure water supply system, made up of a remote piping connector, vertical pipe, flex hose, rotary joint, and nozzle assembly

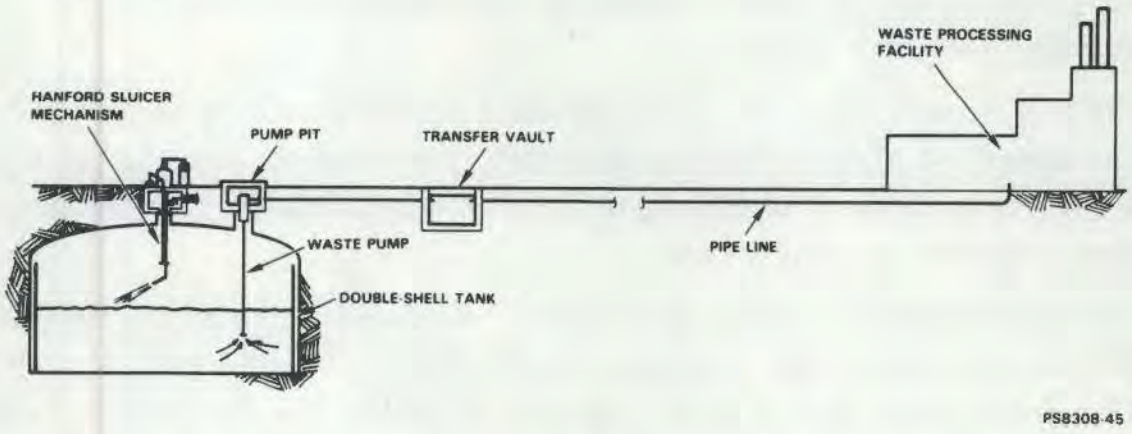


FIGURE 5.2. Hydraulic Retrieval of Waste from Double-Shell Tanks

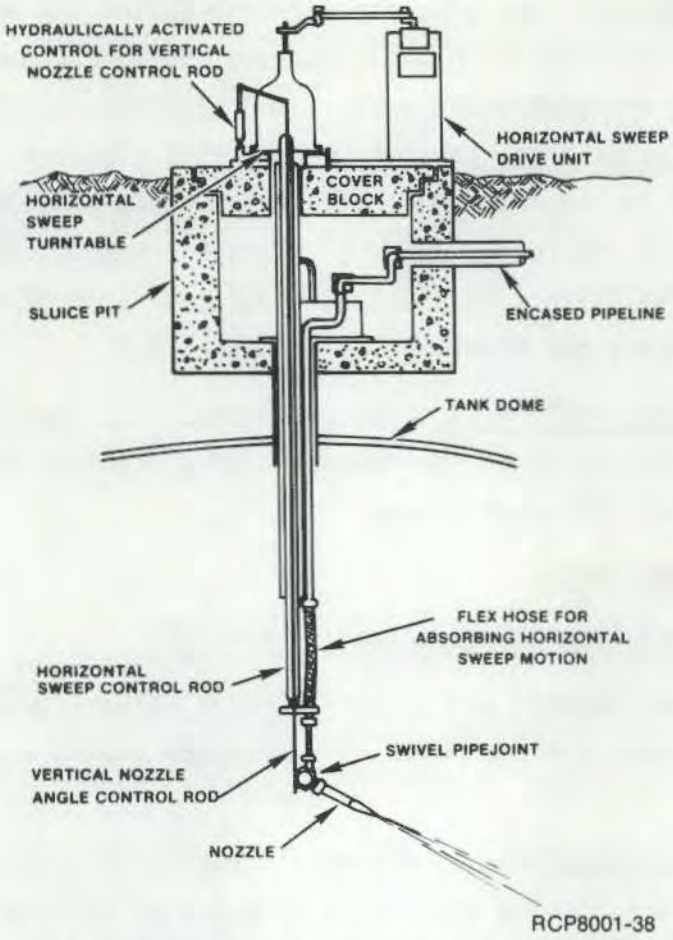


FIGURE 5.3 Hanford Sluicer Installation

- the 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.

The tank is maintained at a slightly negative pressure, and air is discharged through two high-efficiency particulate air filters. Transfer of the slurry and liquid to waste processing facilities or other tanks will be accomplished by methods currently in use.

The concentration of waste in the slurry is assumed to be in the form of insoluble particles and ~25% by volume, which is similar to slurries of coal or gravel that are pumped (Perry 1973). While some of the contamination is in the form of soluble nuclides, we are unable to identify this portion in the present

analysis. At 0.2 m^3/min pumped (ESG 1980), the 25% by volume (i.e., the radioactive material) pumped would be 0.05 m^3/min. Assuming a waste density of 1.8×10^6 g/m^3, this is 9×10^4 g/min of salt cake pumped.

5.1.2.2 Postulated Upper-Bound Accident

A pressurized release of the liquid waste is postulated as an upper-bound release event. Recycled liquid could be pumped to some manifold where that liquid or external liquid could be used for the sluicer. If the manifold is located in a facility which is not a nuclear-grade facility or the external tanks are not enclosed, failure of the diversion valve could result in the backflow of waste solution into the unenclosed area and the spray release of the liquid until pumping was stopped. The material released but not made airborne would be subjected to resuspension. The material could be jetted from the pipe and perhaps become airborne as a spray that is carried by the prevailing wind. Even at the nominal wind speed found at Hanford of 7.6 mph (Stone 1972), 5% of the sprayed material could be made airborne (Sutter 1980). If the wind velocities were substantially higher, even more could become airborne.

5.1.2.3 Release Estimate for Upper-Bound Accident

The parameters presented above can be used to calculate the release (listed in Table 5.3).

(pumped/min) (fractional release) = g/min

$$9 \times 10^4 \text{ g/min} \quad \times \quad 5 \times 10^{-2} = 4.5 \times 10^3 \text{ g/min} = 75 \text{ g/s}$$

This system would be monitored, and it could be assumed that a safety system would activate a response of shutting down the pump in a reasonable time. A one-minute release is assumed.

5.1.2.4 Other Accidents Considered

Accidents with lower releases were considered and are also listed in Table 5.3.

TABLE 5.3. Postulated Airborne Releases from Accidents During Hydraulic Retrieval of Double-Shell Tank Waste

Event	Atmospheric Airborne Release
Diversion valve failure	4.5×10^{3} g
Slurry spill	1×10^{-4} fraction, plus resuspension
Loss of service or power	No significant release
Loss of filters	
1st stage	4×10^{-5} g/min
2nd stage	7×10^{-2} g/min

Slurry Spill (Pipe Break). It is assumed that a pipe connection fails due to some external event which also damages the pipe shielding. The breach allows the entire flow to discharge to the soil, but remnants of the shielding prevent the interaction of the liquid and wind. Therefore, it is assumed that the immediate airborne release is similar to a liquid spill, and on the basis of experimental measurement of free-fall liquid spills, 0.01% of the slurry is estimated to become airborne (Sutter, Johnston and Mishima 1981). The total release would be the fraction airborne times the volume leaked plus the quantity resuspended by the wind until remedial measures could be effected. Sehmel (1979) suggested a resuspension rate of 10×10^{-8}/sec as a conservative value for yearly average Hanford conditions.

Loss of Services or Power. Loss of services or power is visualized as stopping any operations or systems. Loss of flow without loss of filtration would result in back diffusion after some period but is not viewed as a significant release potential. Cessation of the operation does not appear to have serious release potential for this operation.

Loss of Filtration. In the event of loss of one or more stages of high-efficiency particulate air filtration due to some indeterminant event (i.e.,

the accumulation of moisture upon the filters, pressurization within the waste tank, etc.), the material airborne within the tank could be released to the atmosphere.

During the operation, the tank ventilation system is assumed to operate at a rate of 20 m^3/min carrying 10 mg/m^3 meter of slurry. At an assumed slurry density of 1.2 g/cm^3, 0.07 g of slurry particles/min challenge the filters. With loss of a single stage of filtration without loss of flow (and neglecting the increase of flow due to the decreased pressure drop), the transmission factor through the filters increases to 0.0005 and the emission to the atmospheres rises to 3.5×10^{-5} g/min. Loss of both stages of filtration increases the transmission factor to 1, and the 0.07 g/min are released to the atmosphere.

5.1.3 Radionuclide Concentration

In the radionuclide concentration facility, strontium, cesium, technetium, and TRU elements would be removed from soluble salts to prepare material suitable for either glass immobilization or grouting. This is discussed as a single operation in the Hanford Defense Waste Environmental Impact Statement; however, it is really a series of processes as shown in Figure 5.4. These

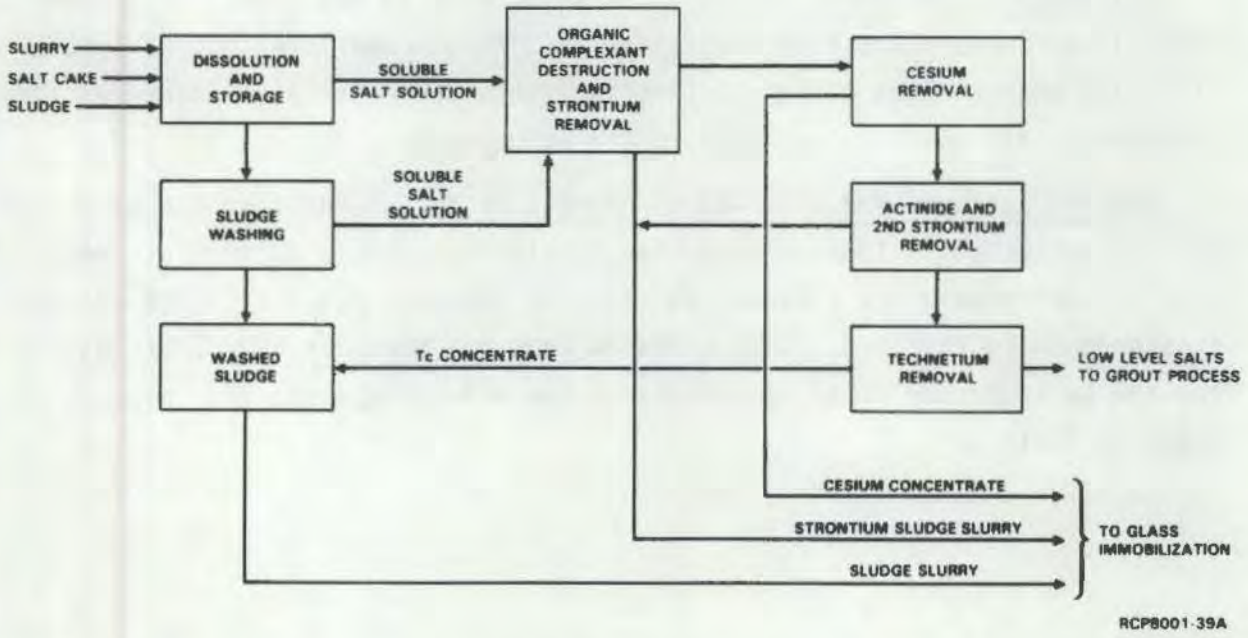


FIGURE 5.4. Schematic Flowsheet for Radionuclide Concentration Processes

processes are essentially three operations: sludge washing and solids/liquid separation, complexant destruction, and removal of various radionuclides. Releases from these operations are discussed in the following sections.

5.1.3.1 Sludge Washing and Solid/Liquid Separation

This operation dissolves salts that would otherwise remain entrained in the high-level solids. Removal of these salts minimizes the amount of glass being produced.

Description of Operation and Facility. Initial separation is performed by a solid-bowl centrifuge. Then separated sludge/solids are washed three times, again in the solid-bowl centrifuge. Centrifuge solids holdup capability, coupled with estimates of time cycles, provided an estimated nominal throughput capability of 10 metric tons of uranium per operating day. The effluent from jet entrainment in the solid/liquid separation and sludge washing operations would be discharged to the atmosphere through a vessel vent off-gas system.

Figure 5.5 is a conceptual drawing of a facility for radionuclide concentration with a sludge washing canyon identified. Additional information on the facility, equipment, or process is not available.

Postulated Upper-Bound Accident. Loss of flow is not viewed as a high-potential airborne release hazard; therefore, this scenario addresses loss of filtration without loss of flow. In this event, the material airborne within the process cell would be released to the atmosphere.

Release Estimate for Upper-Bound Event. It is assumed that the airborne mass concentration is 10 mg/m^3 and the ventilation rate is 20 m^3/min. Under these circumstances, 0.2 g/min of particulate material present in the process is assumed to be released. This scenario does not consider the particle loss from the duct work or other components of the off-gas system. The release is listed in Table 5.4.

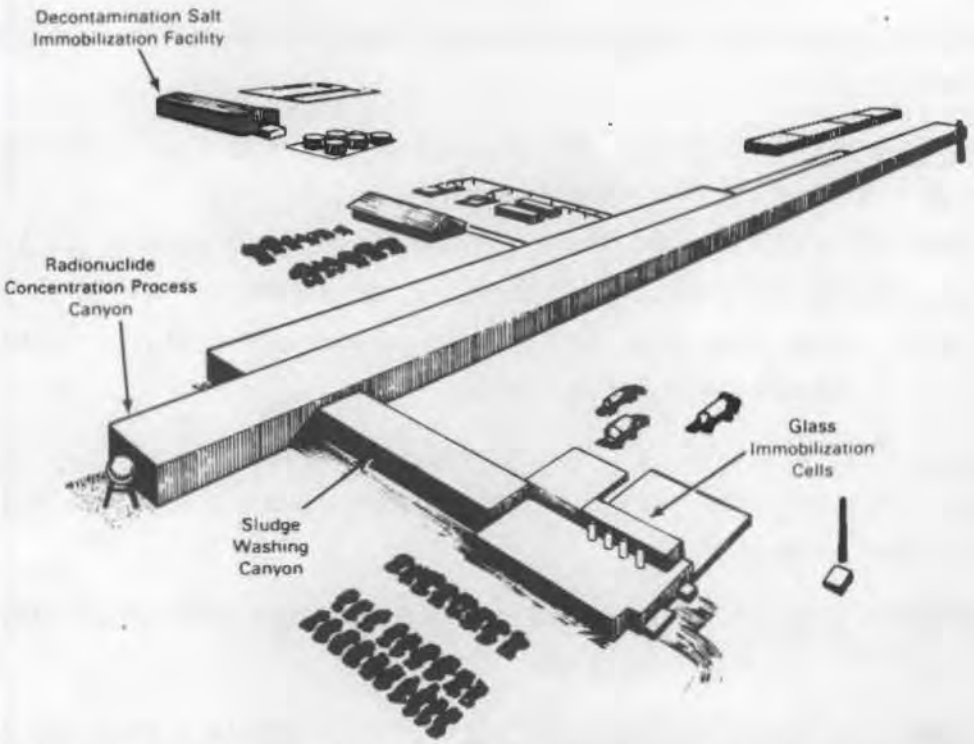


FIGURE 5.5. Conceptual Facility for Radionuclide Concentration

TABLE 5.4. Postulated Airborne Releases from Accidents During Sludge Washing and Solid/Liquid from Separation Tank Waste

Event	Atmospheric Airborne Release
Loss of exhaust system	0.2 g
Evaporation spill/ rupture	3×10^{-16} fraction[a]
Leaks	Significantly below upper-bound
Centrifuge rupture	7×10^{-14} fraction[b]
Hydrogen explosion	6×10^{-9} fraction[c]
Fire in cell	Significantly below upper-bound
Filter fire	Covered by upper-bound
Loss of services or power	No significant release

(a) Hayward and Jensen (1980).
(b) Richardson (1980).
(c) DOE 1982).

Other Accidents Considered. Releases from less than upper-bound accidents are included in Table 5.4.

- Evaporator Spill/Rupture. It is assumed that material is spilled or leaked from the evaporator and would be equal to the free-fall release of liquids. The energy or conditions generated by this event would not result in loss of filtration; therefore the airborne release consequences are limited. An estimated fractional release is 3×10^{-16} (Hayward and Jensen 1980).
- Leaks. Several tanks, sludge, slurry, or washed sludge, could leak. They could be assumed to have consequences similar to the evaporator leak.
- Centrifuge Rupture. This event had a postulated fractional release of 7×10^{-14} (Richardson 1980).
- Hydrogen Explosion in the Feed Tank. Four possible situations have been suggested (DOE 1982) that could result in an explosion: 1) red oil formation, 2) hydrogen accumulation, 3) mercury or silver compound inclusions, and 4) ammonium nitrate available. None of these events appears to compromise the filtration system and therefore the release consequences are less than loss of filtration. The fractional release was estimated at 6×10^{-9}.
- Fire in the Cell. Combustible materials are also prevalent in hot cells, generally in the waste, but also as lubricants, etc. Fires in the process cell would more likely plug than destroy the filters which are usually some distance from the cells. Therefore, the release from such an event is felt to be less than by loss of filtration.
- Filter Fire. Unless the radionuclides accumulated on the filter are volatile at elevated temperatures, subjecting the filter to heat is felt to result in the shrinking of the glass filter and trapping of the particles accumulated. The consequences would result in loss of filtration, which has already been covered.

- Loss of Services or Power. Loss of services or power would primarily lead to cessation of operations, which does not appear to result in serious airborne release hazards in this operation.

5.1.3.2 Complexant Destruction

This process will be performed, but the operation has not yet been defined. Therefore, no accidents are postulated.

5.1.3.3 Radionuclide Removal

This process removes all radionuclides with greater than 10-yr half-lives from soluble salt wastes. This separates high-level waste for immobilization, and a large volume of low-level chemical waste that can be disposed of in a relatively inexpensive way.

Description of Operation and Facility. Strontium and actinides remaining in solution after sludge removal are removed by a combination of precipitation and adsorption on sodium titanate ion exchanger (the conditions, materials and equipment used for the precipitation are not defined). Spent sodium titanate is added to the strontium precipitate solution, and the slurry is transferred to the immobilization portion of the facility. The supernatant liquid from the strontium precipitation is filtered through sand filters for the removal of trace solids. The cesium is then removed in an ion exchange column loaded with Duolite ARC-359® (Diamond Shamrock Company). The cesium is eluted with an ammonium carbonate solution, which is steam-stripped to separate the effluent from the cesium product. The cesium solution is also transferred to the immobilization portion of the facility.

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

The waste stream leaving the technetium ion exchange column is essentially a sodium salt waste containing trace amounts of radionuclides. Ruthenium and iodine, like the sodium salts, are contaminants and are virtually unaffected by the separation process. The ruthenium concentration is low because of the long

decay time. After being monitored in collection tanks, the salt solution is transferred to the grout mixing process.

Process off gases are treated for the removal of particulates, radionuclides, oxides of nitrogen, and ammonia before release to the atmosphere. The nitric acid resulting from the distillation of the technetium product is recovered and reused. Ammonia and carbon dioxide are recovered, recombined, and reused in elution of cesium from the ion exchange column. Liquid effluents are minimized by recycling. Cooling water is reused after passing through a suitable cooling process. Steam condensate is returned to the process. Steam condensate and cooling water not recycled are sampled and discharged to evaporation ponds. Wastes not meeting the limits for discharge are concentrated and blended into the process.

Postulated Upper-Bound Accident. It has been postulated in several references that an ion exchange column loaded for 2 days ignites and ruptures (DOE 1982, Hayward and Jensen 1980, ESG 1980). All of the activity on the resin is released and a portion becomes airborne. The building ventilation system is not impacted by the event and continues to function.

Release Estimate for Upper-Bound Event. In one scenario, all the activity on the resin is released, with 0.002% becoming airborne. The building ventilation system continues to function with a transmission factor of 10^{-5} (Rockwell 1980). A similar scenario is presented in DOE (1982) with a 0.01 fraction of the activity made airborne and with a transmission factor of 3×10^{-4}, resulting in an atmospheric fractional release of 3×10^{-6}. The larger values are listed in Table 5.5 and are suggested for dose estimates.

Other Accidents Considered. These less than upper-bound releases are included in Table 5.5.

- Ion Exchange Tank Leak. It is postulated that due to a loose connection, corrosion, etc., liquid leaks from a vessel containing concentrated solution from the process. The airborne release would be analogous to a free-fall spill, and the event does not compromise the filtration system. The airborne release is considered less than that given above for the ion exchange column fire.

TABLE 5.5. Postulated Airborne Releases from Accidents During Radionuclide Removal from Tank Waste

Event	Atmospheric Airborne Release
Ion exchange fire	3×10^{-6} fractional release
Ion exchange tank leak	Significantly below upper-bound
Equipment failure	No significant release

- Equipment Failure. Release values for equipment failure postulated by Hayward and Jensen (1980) indicate that the anticipated release is less than that for an ion exchange column fire.

5.1.4 Glass Immobilization (Vitrification)

5.1.4.1 Description of Operation and Facility

The glass immobilization process discussed in this section would be used in conjunction with the radionuclide concentration process in cells identified in Figure 5.5. 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 carbon steel canisters 0.61 m diameter by 3-m-long. The product glass would contain about 25 wt% waste oxides.

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 weighted 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. Estimated dimensions of a continuous electric melter for producing 4 t of glass per day are shown in Figure 5.6. 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 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 blanket distribution) up to 5% of the cesium.

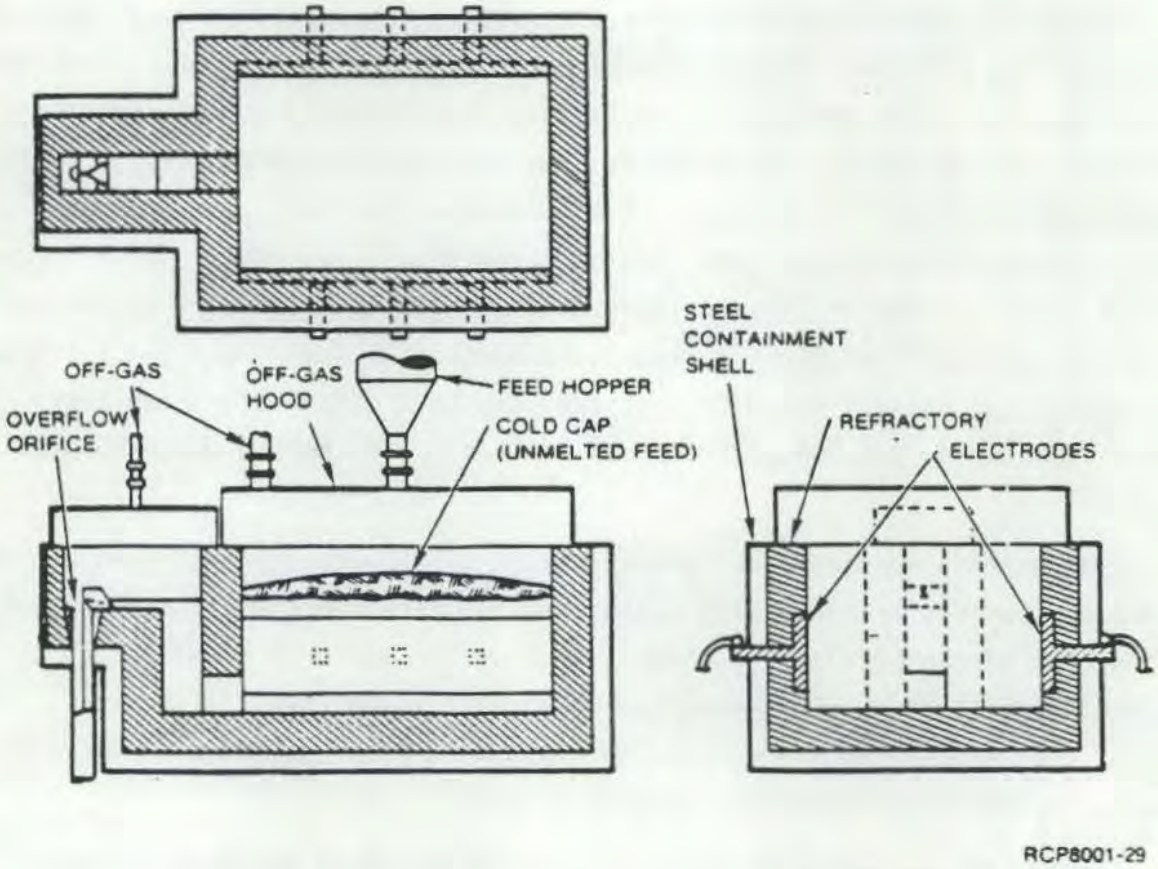


FIGURE 5.6. Continuous Electric Glass Melter

Present assumptions are for three melters--two operating and one spare--to achieve an effective rate of 8 t per day. Glass would be allowed to pour from a melter to fill carbon-steel canisters in a continuous 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 a temperature of about 157°C. 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.

Because of multiple mechanical operations and solids-handling steps in the process, a combination of in-cell cranes, manipulators, and viewing windows would be used 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.

5.1.4.2 Postulated Upper-Bound Accident

The simultaneous loss of both the melter and facility filtration appears to be an event with a low probability of occurrence. Even total loss of the facility filtration is highly unlikely. For the purposes of this analysis, however, it is postulated that such an event occurs and the material airborne in the canyon is released to the atmosphere without filtration.

5.1.4.3 Release Estimates for Upper-Bound Event

As in Section 5.1.2.4 of this report, the same mass concentration and flow are assumed, resulting in 0.2 g/min of glass being released. Since the melter has its own filtration system, the operation is not perceived to be a dusty one. A filter fire is once again assumed to result in the release of little of the accumulated material and is essentially the same airborne release as from the loss of filter alone. The release is listed in Table 5.6.

Cesium, if associated with the waste, could volatilize at elevated temperatures and release to the atmosphere if control systems fail. Cesium volatility is a function of temperature. Experimental cesium releases in percent per hour at 1000°C were: 1% (Albrethson and Schwendiman 1967), 1.5% (Gray 1976), and 4.2% (Walmsley et al. 1969). Volatility increases further at higher temperature levels.

5.1.4.4 Other Accidents Considered

Table 5.6 includes the releases from the other accidents considered.

Molten Glass Spill. "Should the molten glass spill from the melter onto the canyon floor, the glass would flow and solidify, releasing little

TABLE 5.6. Postulated Airborne Releases from Accidents During Waste Vitrification of Tank Waste

Event	Atmospheric Airborne Release
Failure of off-gas system	0.2 g
Molten glass spill	Significantly below upper-bound
Melter feed tank leak	Significantly below upper-bound
Explosion in glass melter/steam explosion	3×10^{-8} fractional release[a]
Waste canister failure	3×10^{-10} fractional release[a]
Cell fire	No significant release
Loss of services or power	No significant release

(a) DOE (1982).

radioactivity to the canyon" (DOE 1982). Thus, it is estimated that the airborne release to the atmosphere from such an event would not be significant, especially since any material released to the canyon must still pass through the filtration system.

Melter Feed Tank Leak. The consequences of this event are viewed by Hayward and Jensen (1980) as the leak of a liquid which has been covered several times in previous sections.

Explosion in Glass Melter/Steam Explosion in Glass Melter. DOE (1982) suggested the following event.

> "Although contact between the glass and water does not normally lead to a steam explosion, entrapment of water under molten glass, either in the melter or on the floor, in conjunction with the following factors would more likely cause such an event, 1) low water temperature, 2) high glass temperature, 3) shallow water depth, 4) rust on the surface beneath the water, 5) ionic content in water (e.g., salt), and 6) forced injection of glass into water. There is only a remote possibility that the water could get trapped beneath the molten glass in the melter. A failure of the cooling system and of the drain system followed by a failure of the glass melter could lead to entrapment of water beneath molten material, causing an explosion.
>
> The molten glass will fragment into a large number of small particles by the shock of the explosion and scatter throughout the canyon. Approximately 0.01 wt% of the fragmented glass is estimated to be carried into the ventilation system. At the time of the explosion, the melter is assumed to contain 1000 L of product. A filter factor of 3×10^{-4} is given resulting in a release of 3×10^{-8} fraction (or 0.03 cm^3 of glass)."

Waste Canister Failure. Waste canisters can be breached before encapsulation if they are dropped in handling operations (DOE 1982). For the purposes of this analysis, it is assumed that the rupture is equivalent to a cover block drop on an encapsulated canister.

Cell Fire. It is questionable that any combustibles could be tolerated in a cell with the potential high temperatures from the glass melting operation. Therefore, a fire is considered an unlikely scenario with no significant release.

Loss of Services or Power. Loss of service or power could result in a variety of consequences depending on which services failed and which continued

to operate. Loss of power would result in cessation of operations. However, some releases could continue until the melt cooled. They could be carried to the filters by diffusion since there would be no airflow, and filters would capture them. Loss of airflow without loss of power would mean continued heating and release carried to the filters. Emergency shutdown procedures would preclude this second event. No significant releases are postulated.

5.1.5 Grout Decontaminated Salt Solutions

The decontaminated salt solutions would be classified as low-level waste since they are neither high-level waste nor TRU waste. They will be disposed of by grouting and burial in near-surface trenches. These particular releases will be of lower radiological concern; however, the facility and releases are described here because later operations in them could involve higher levels of activity.

5.1.5.1 Description of Operation and Facility

A transportable grout facility will be used to make the grouted waste form by blending grout-forming solids with the liquid waste and pumping the slurry to the disposal site. A schematic of the grout process is shown in Figure 5.7, and the near-surface trench disposal of grout is shown in Figure 5.8.

The grout process would involve two new facilities: 1) the Dry Materials Receiving and Handling Facility, where the grout-forming solids would be blended, and 2) the Transportable Grout Equipment modules where the blended solids would be mixed with liquid waste and the resulting slurry pumped to the disposal site. The second facility is the only one with a radionuclide inventory. Thus, it is the one with potential radiological impact and the only one of interest for our analysis.

The transport grout equipment would consist of transportable modules that would mix blended solids with liquid wastes. The resulting slurry would be pumped into the disposal sites. The transportable grout equipment would include:

- blended solids feed system (for providing solids to the grout mixer)
- grout mixing and pumping system

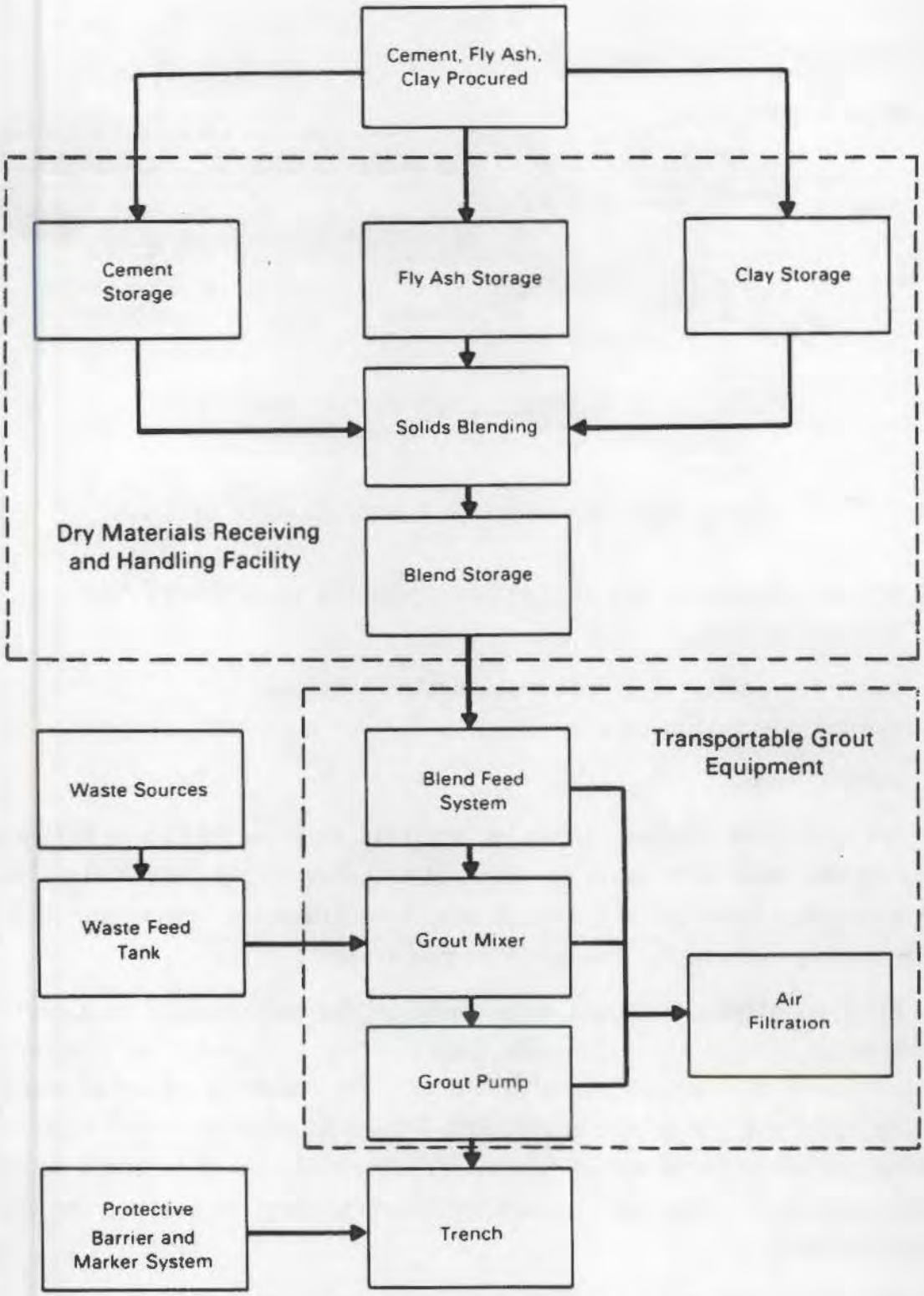


FIGURE 5.7. Schematic of Grout Process

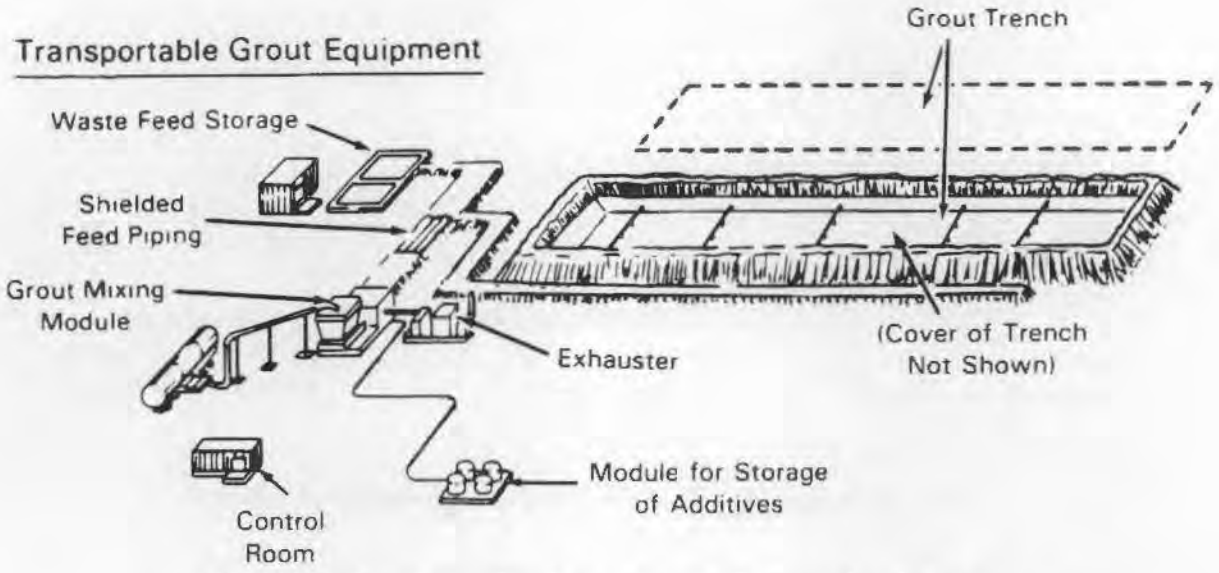


FIGURE 5.8. Near-Surface Trench Disposal of Grout

- off-gas exhausters and filters (for removing contaminants from process off gas)
- tanks for additives and decontamination solutions
- standby electric generator
- control room.

The suggested process for making grout is depicted in Figure 5.7; operations in the lower half would involve radionuclide-bearing materials. The operations would involve blending of the dry solids with liquid waste to form a grout slurry, and pumping the grout to the disposal site.

Blended solids and liquid waste would be fed continuously into one end of a continuous grout mixer. The rate, based on waste liquid flow rate, would typically be 1 t of solids per m^3 of waste. The rotating action of the mixer paddles would mix the solids and liquids into a homogenous slurry discharged by gravity at the opposite end of the mixer. Chemicals could be added to the mixer and/or waste feed tank to control foaming, grout viscosity, and grout-hardening rates.

The grouted slurry would flow by gravity into the intake end of a progressive cavity pump, and then pumped through a pipe to either a plastic-lined

trench (decontaminated salt), a culvert vault, or a retired underground storage tank. Maximum pumping distance would be 460 m at a pressure of 350 psi. During filling, the trench would be covered to retain moisture.

High-efficiency particulate air 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. Action within the mixing/pumping module would be monitored by television and liquid-level sensors to permit early detection of process problems. Small radioactive spills and leaks that might occur would be contained within the modules. Because of the small spills and leaks, equipment would have to be periodically flushed and decontaminated. Decontamination solutions used to clean up spills and leaks would be mixed with grout formers and similarly disposed of as grout. No details of the facility construction are available.

Wastes could be 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 blending of solid particles by the grout matrix. The wastes associated with each of these methods have not been identified.

5.1.5.2 Postulated Upper-Bound Accidents

The grout process is new and thus is not covered extensively in the literature. For the purposes of this analysis, it is postulated that the salt solution is sprayed from a penetration of the transfer line within the process. A liquid is chosen since less energy is required to subdivide the liquid than the grout, which is viscous. A second consideration in selecting a liquid is that the radionuclides are more concentrated in the liquid. There are typically 1 t of solids mixed with a m^3 of waste, so the grout dilutes the waste by about one-half.

5.1.5.3 Release Estimate for Upper-Bound Event

Even under conditions for formation of a spray, the quantity of particles less than 10 μm is only 0.01 wt% (Mishima and Ayer 1981). It is therefore assumed that 0.01 wt% of the liquid released is made airborne. Assuming a

density for the liquid of 1.3 g/cm^3 and a total flow of 10 m^3, a total of 1,300 g of salt solution is released to the facility. Assuming a filtration system with two stages of high-efficiency particulate air filtration and a transmission factor of 2.5×10^{-7}, 3×10^{-4} g of salt solution would be released to the atmosphere as listed in Table 5.7.

5.1.5.4 Other Accidents Considered

Releases from less than upper-bound accidents are discussed below and listed in Table 5.7.

Salt Solution Tank Leak. As mentioned in previous sections, the release would be analogous to the free-fall spill of a liquid.

Mixer Tank Leak. At worst, the airborne release from this event could approach that for the pressurized release of the salt solution due to the centrifugal force imparted to the liquid by the mixer. The amount of liquid available would be limited by its absorption by the dry components. Thus, it is not anticipated that the airborne release from this event would be as severe as that for the pressurized release.

Grout Spill. This event involves grout, which is more difficult to subdivide than a liquid under the same conditions. Thus, the potential airborne release from this event would be less than that for the liquid.

5.1.6 Fill Empty Tank

The residual tank waste (less than 5% of initial quantities in single-shell tanks and less than 0.05% in double-shell tanks) and the tanks themselves

TABLE 5.7. Postulated Airborne Releases from Accidents During Grouting Decontaminated Salt Solutions

Event	Atmospheric Airborne Release
Pressurized release of salt solution	3×10^{-4} g
Salt solution tank leak	Significantly below upper-bound
Mixer tank leak	Significantly below upper-bound
Grout spill	Significantly below upper-bound

would be disposed of in place. The single-shell and double-shell tanks and the annulus of double-shell tanks would be filled with grout, gravel, sand, soil, or other substances to control subsidence in the event of tank structural failure (dome collapse). This operation is called "dome fill" and is described in the Hanford Defense Waste Environmental Impact Statement.

5.1.6.1 Description of Operation (No Facility Indicated)

The operation is essentially the one described under subsidence control for waste tanks in the Hanford Defense Waste Environmental Impact Statement.

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 5.9).

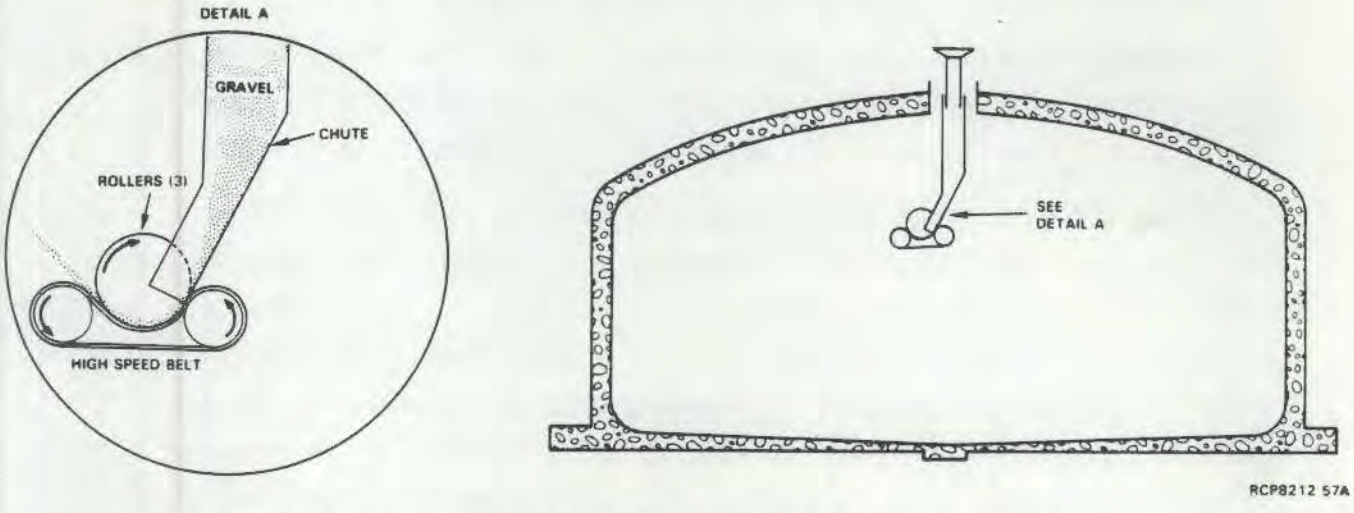


FIGURE 5.9. Centrifugal Thrower for Filling Waste Tanks

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

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 existing level above the height of the steel liner in the waste tank.

Individual tanks or whole tank farms would be covered with 1 m of soil. Contaminated soil, around and under tanks, resulting from past tank leaks is classed as low-level waste. That waste is relatively well fixed in the sediments and left in place.

5.1.6.2 Postulated Upper-Bound Accident

A dome collapse could be initiated if heavy equipment is inadvertently driven onto the tank. Since the operation is performed in the open, the release would be directly to the atmosphere.

5.1.6.3 Release Estimate for Upper-Bound Release Event

Hayward and Jensen (1980) suggested a 5×10^{-8} fractional release from a dome collapse. Rockwell (1980) estimated that 7 kg of dry saltcake as particles less than 10 μm in diameter could be released in a dome collapse.

Since the tank contains only residual material, up to 5% of the original quantity, the 7 kg release will be lowered correspondingly. Therefore, the release calculates to 350 g as listed in Table 5.8. This is a very conservative approach because much of the residual material could be hardened, fixed on the walls, and not susceptible to becoming airborne. In both cases (tank completely or partially full) the dome is being filled with inert material and the waste will conceivably be covered with this material; therefore, severity of the release will be mitigated. The upper-bound accident would occur as the operations begin before any fill is delivered to the tank.

TABLE 5.8. Postulated Airborne Releases from Accidents During Filling of Empty Tank

Event	Atmospheric Airborne Release
Dome collapse	350 g[a] 5×10^{-8} fraction[b,]
Loss of filtration 1st stage Both stages	 5×10^{-6} g/m^3 1×10^{-2} g/m^3
Loss of services or power	No significant release
Equipment failure	No significant release

(a) Based on Rockwell (1980).
(b) Hayward and Jensen (1980).

5.1.6.4 Other Accidents Considered

Additional accidents are developed in the following section, and releases are included in Table 5.8.

Loss of Filtration. This event could occur when equipment used in the dome-filling operation hits the filter and dislodges it. Any airflow (rate unknown) would therefore be unfiltered. It might be assumed, as in the dome collapse, that much of the airborne material would be inert rather than radioactive. If the accident happened before the dome filling began, the air in the dome would contain low levels of contamination. Any required safety systems could shut down the airflow. Therefore, these releases are postulated to be lower than in a dome collapse.

Loss of Services or Power. This would shut down the operation, and thus no releases would be envisioned for this event.

Equipment Failure. This event is not seen as leading to releases. Equipment is located outside the tank, radioactive materials inside. Operations would merely be shut down.

5.2 FUTURE TANK WASTE

Processing of future tank wastes would be integrated with that of existing waste. Insofar as practicable, all newly and future-generated high-level waste would be disposed of in a geologic repository. Many of the operations are the same as for existing tank wastes; therefore, the reader is directed to those sections when appropriate.

5.2.1 Retrieval

These wastes are stored in double-shell tanks, and the operation used will be hydraulic retrieval. Therefore the operation, facility, upper-bound accident, release estimates, and other accidents considered will be the same as those discussed in Section 5.1.2 and listed in Table 5.3.

5.2.2 Solid/Liquid Separation and Strontium, Cesium, Technetium and TRU Removal

These operations will share the same facility and operations as the existing tank waste. Therefore the operation, facility, upper-bound accident, release estimates, and other accidents considered will be the same as those described in Section 5.1.3 and listed in Tables 5.4. and 5.5.

5.2.3 Grout Decontaminated Liquid

The partially decontaminated liquid would be converted to grout, along with other wastes. Grout would be disposed of in shallow trenches. Thus these operations are the same as those for existing tank waste and the operation and facility, upper-bound accident, release estimate, and other accidents considered are the same as those for existing tank waste as described in Section 5.1.5 and listed in Table 5.7.

5.2.4 Vitrification

Future tank waste will be immobilized in a large vitrification facility operated in conjunction with the sludge washing and solid/liquid separation radionuclide removal facility. Therefore, the operations will be the same as those in Section 5.1.4, with the same upper-bound accident, release estimates, and other accidents considered. These are listed in Table 5.6.

5.2.5 Fill Empty Tanks

Future tank waste would be in double-shell tanks. The anticipated residual waste would be less than 0.05% of the initial quantity, thus reducing the magnitude of any potential release. The operations, upperbound accident, and other accidents considered would be the same as described in Section 5.1.6. In that section the postulated dome collapse accident had a 350-g estimated release, based on a 7-kg release from a full tank. Here the release is 0.05% of that from a full tank for a total of 3.5 g. This release is lower than that estimated in Section 5.1.6.3, since there is less residual waste in this tank. The other releases listed in Table 5.8 apply here.

5.3 STRONTIUM AND CESIUM CAPSULES

Strontium and cesium capsules (illustrated in Figure 4.1) would be stored in the Waste Encapsulation and Storage Facility until 1995, then removed for geologic disposal. The other operation discussed below is packaging the capsules.

5.3.1 Removal of Capsules from Water Basin

Strontium and cesium capsules would be stored in water-filled basins until the necessary modifications were made to the waste encapsulation and storage facility to allow overpacking. During the storage period, the capsules and basins would be periodically inspected and maintained as necessary. Storage of capsules at the facility is in an active mode that requires cooling water, makeup water, ventilation and maintenance of facility operating systems.

5.3.1.1 Description of Operation and Facility

Neither the waste encapsulation and storage facility, nor its equipment is described in the Hanford Defense Waste Environmental Impact Statement, nor are details of the capsule retrieval system. These are assumed to be those now in place and described by Braden et al. (1971) in the Safety Analysis Report for that facility. The facility contains a storage pool area, and a crane is available and used to remove cover blocks and capsules.

The operation and equipment design and usage are assumed in keeping with sound, prudent nuclear practices especially in the area of engineered safety systems such as the ventilation and exhaust system.

5.3.1.2 Postulated Upper-Bound Accident

The upper-bound airborne release event postulated is a rupture of the capsule. Waste canisters can be breached before they are encapsulated if they are dropped in handling operations (Hayward and Jensen 1980). Completely encapsulated canisters will not rupture even if they are dropped onto concrete from heights up to 6 m. Encapsulated canisters could not be ruptured when hit by a falling, heavy object such as a cell cover. However, the capsules may be stored for an extended period of time and could potentially deteriorate due to heated storage in water. Glass, which is the material contained in the Hayward and Jensen study, provides greater support for the casing than either the strontium fluoride or cesium chloride (salt forms) addressed in this study. Therefore, the capsule is assumed to rupture upon impact after a drop, and the material contained within the capsule is subdivided. The fraction subdivided and made airborne in the respirable size range is not available in the published literature. Thus, the internal pressure generated within the capsule required to rupture it is assumed to be high and the particulates produced are correspondingly fine.

Loss of basin cooling water resulting in the overheating of the cesium and strontium capsules was not considered as the upper-bound accident for this operation for the following reasons: 1) the basin is designed to withstand the drop of a capsule, 2) if the basin suffers a leak during normal storage, the leak would be slow and detected before significant effect since the facility is continually monitored, and 3) catastrophic loss of basin water is associated with severe natural phenomena (e.g., earthquakes) which are covered in another section.

5.3.1.3 Release Estimated for Upper-Bound Event

A value of approximately 1% was experimentally measured for less than 10 μm aerodynamic equivalent diameter airborne fraction of a fine depleted uranium dioxide powder released at 50 psig pressure (Sutter 1983) and this

experimental value is applied. Since it is postulated that this event occurs within a nuclear-grade facility with a filtration system that has two stages of high-efficiency particulate air filters, the respirable fraction released to the atmosphere is conservatively estimated as 2.5×10^{-9} of the source. Theoretic density of the salt forms are 3.988 and 4.24 g/cm^3 and the capsule contents are compacted to 75% of the theoretic density, so 3 g/cm^3 is considered a good value to use for the density of the radioactive material. The mass of the 7.42 cm^3 source is calculated to be 2.2×10^3 g, the respirable airborne fraction is 2.2×10^1 g and the atmospheric release is calculated to be 5.5×10^{-6} g. The release is listed in Table 5.9.

5.3.1.4 Other Accidents Considered

Other accidents could occur, but have lower releases as listed in Table 5.9.

Capsule Drop in Basin. The impact suffered from the capsules striking water will be significantly less than the impact of a cover block; therefore, it is estimated that the consequences will be much less.

TABLE 5.9. Postulated Airborne Releases from Accidents During Capsule Recovery

Event	Atmospheric Airborne Release
Capsule rupture	5.5×10^{-6} g
Capsule drop in basin	Significantly below upper-bound
Hydrogen accumulation and explosion	1×10^{-15} fraction[a]
Loss of filtration	
1st stage	5×10^{-6} g/m^3
Both stages	1×10^{-2} g/m^3
Fire	No significant release
Capsule failure in basin	No significant release
Loss of services or power	No significant release

(a) Richardson (1980).

Hydrogen Accumulation and Explosion. Hydrogen generated by the radiolysis of water in the basin could accumulate if ventilation flow through the area were lost for a long period of time. The rate of hydrogen production is dependent on the radiation level, which is dependent upon time for the concentrated cesium and strontium in the capsules. It is not felt that there is a great possibility of accumulating levels of hydrogen that could result in an explosion with significant damage to the building, much less to the capsules stored under many feet of water.

Loss of Filtration. Loss of filtration without loss of flow would result in the release of the material normally present in the facility. Since the areas around the water-filled basin are operating areas and would be at a radiation level safe for personnel, the consequences of such an occurrence are low.

Fire. Due to the presence of combustible materials (plastic - bags, sheeting and equipment; cellulosics - clothing, paper, etc.; and lubricating oils and greases) in the facility, the possibility of a fire cannot be ignored. The capsules are made of highly resistant metals, and high temperatures for long periods would be required to result in high enough internal pressures to cause rupturing. The quantity of combustibles would limit the extent of the fire, so no significant airborne release of the contained radionuclides is postulated.

Capsule Failure in Basin. Since storage is an active process in which the capsules and storage basin water are periodically checked, failure of the capsules in the water-filled basin is not considered to be a likely event and remedial measures (including decontamination of both the basin water and capsules) would be quickly implemented.

Loss of Services or Power. In this operation, loss of services or power would result in cessation of operations or loss of flow. Neither consequence is viewed as a significant release mechanism.

5.3.2 Capsule Packaging

The capsule packaging process is described in Appendix B of the Hanford Defense Waste Environmental Impact Statement.

5.3.2.1 Description of Operation and Facility

The capsule packaging 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 drywell storage facility 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 12 steel sleeves (0.38-m inside diameter, 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 capsule packaging 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, 14 m high, and that would occupy approximately 1100 m^2. High-density concrete shielding walls of the hot cells would be ~0.9 m thick, and would have eight viewing windows and four manipulator pairs.

5.3.2.2 Postulated Upper-Bound Accident

The literature reviewed (DOE 1982; Hayward and Jensen 1980, Richardson 1980) stated that encapsulated waste canisters cannot be failed unless impacted by a falling, heavy object such as a cell cover. Whether this applies to capsules that have been stored under water for long periods of time is not known. It is postulated that a capsule fails by impact with the machinery used in handling such as a crane head and releases a portion of its contents to the cell atmosphere.

5.3.2.3 Release Estimate for Upper-Bound Event

The maximum airborne release to the environs is a 3×10^{-10} fraction of the source based on values found in the literature (DOE 1982). The filter transmission is 2.5×10^{-7}; thus, the fractional release (back calculated) from the source is 1.2×10^{-3}. This is the failure of a single capsule; 2.64 g is airborne in the cell and 6.6×10^{-7} g is released to the atmosphere. This value is lower than that calculated in Section 5.3.1.3, but here only a portion of the capsule is involved in the release. The release is listed in Table 5.10.

5.3.2.4 Other Accidents Considered

Releases from the less than upper-bound accidents are developed below and listed in Table 5.10.

TABLE 5.10. Postulated Airborne Releases from Accidents During Capsule Packaging

Event	Atmospheric Airborne Release
Machinery impacts capsule	6.6×10^{-7} g
Fire	Significantly below upper-bound
Loss of off-gas exhaust system	
1st stage	5×10^{-6} g/m^3
Both stages	1×10^{-2} g/m^3

Fire. A fire in the facility is possible because of the combustibles normally found in such facilities. Its radiological consequences will be limited since the fire would have little effect upon the encapsulated radioactive materials.

Loss of Off-Gas/Exhaust System. Loss of filtration without loss of ventilation flow would release the material airborne in the facility to the atmosphere around the facility. The capsules handled in this operation have been checked for contamination and cleaned several times before this operation. The operation does not appear to be dusty; there would be minimal airborne material. Therefore, it is estimated that little radioactive material would be released for this occurrence.

5.4 TRU-CONTAMINATED SOIL

A TRU-contaminated soil site is one in which liquids (usually aqueous solutions) have been intentionally or inadvertently released to the soil. These sites consist of cribs, trenches, ponds, ditches, reverse wells, French drains, and unplanned releases.

5.4.1 Mechanical Retrieval of TRU-Contaminated Soil

The facility shown in Figure 5.10 and described in the Hanford Defense Waste Environmental Impact Statement will be used for recovery of both TRU-contaminated soil and TRU solid waste sites.

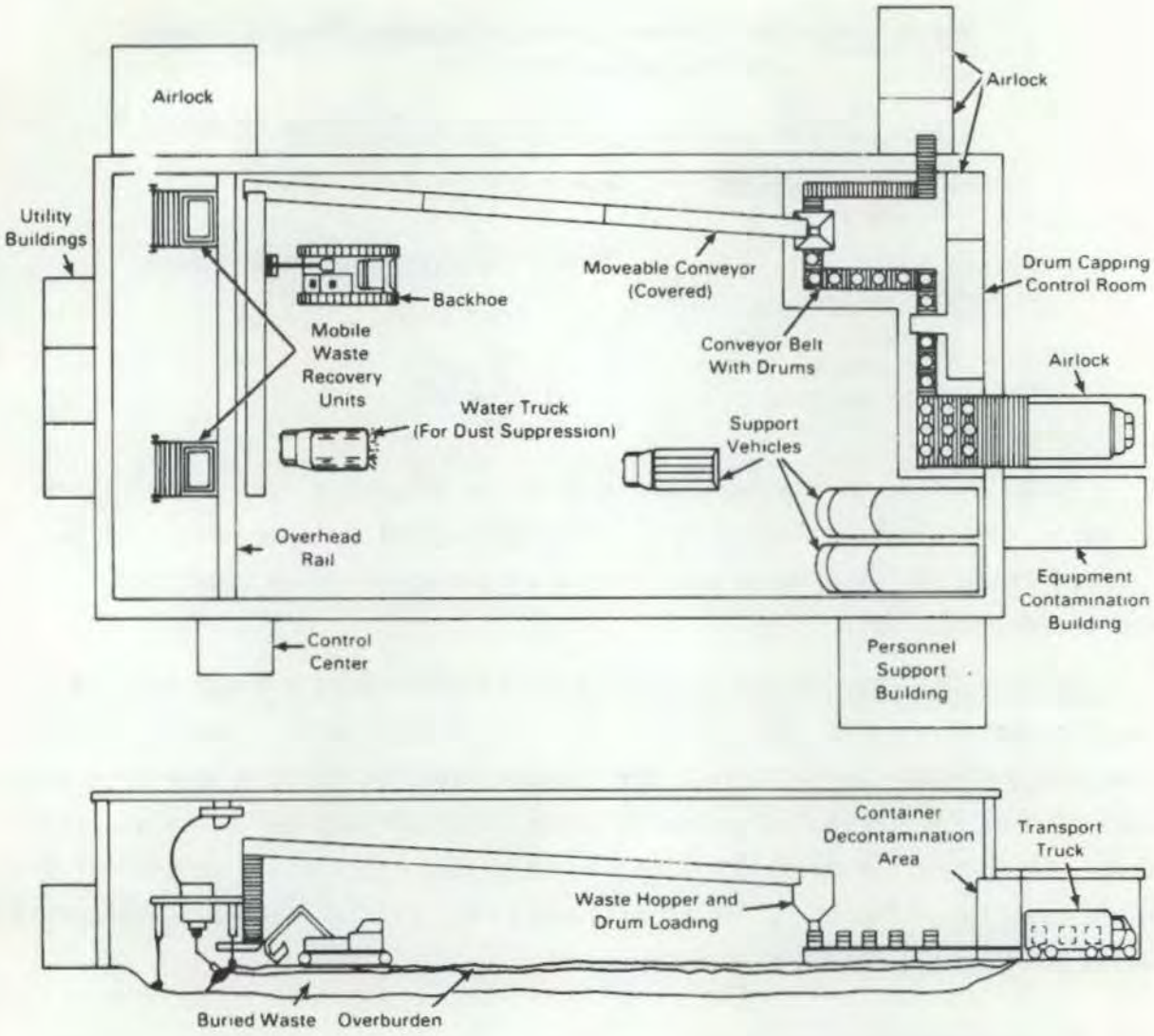


FIGURE 5.10. Mechanical Retrieval of Wastes from Soil or Solid Waste Sites

5.4.1.1 Description of Operation and Facility

Some of the contaminated soil sites are small, and the facility is large, so the facility is assumed to be used on soil sites as appropriate. The approximate building dimensions would be 46 m wide by 92 m long and 12.5 m high. This calculates to a 5.3×10^4 m^3 volume.

Electrically operated equipment would be used for retrieval of contaminated soil. Some equipment would be battery operated, while other equipment would be connected by cables to the building power supply. Equipment used

during soil retrieval would include waste retrievers, container haulers, dust precipitators, and heavy-duty backhoes.

The recovery building, support equipment, and recovery equipment would be thoroughly tested before the entry pit and radioactive waste were excavated. Dust within the pit would be controlled by spraying the working face of the pit with dust suppressants and operating dust precipitators. Mobile waste retrievers would excavate the soil and place it in containers. Filled containers would be checked for radiation levels and TRU content, decontaminated, and moved through an airlock into a special transportation trailer.

The retrieval building would be maintained slightly below atmospheric pressure, and air discharged through two stages of high-efficiency particulate air filters.

5.4.1.2 Postulated Upper-Bound Accident

The maximum release event postulated is an explosion that would breach a weak portion of the facility and provide a significant unfiltered pathway to the atmosphere for the particulate material generated by the event. An explosion could result from battery-generated hydrogen buildup in a confined volume, possibly occurring during off hours when the batteries were being recharged. It is assumed that all batteries would be brought to one portion of the building for this operation. Hydrogen produced by the recharging operation would accumulate with minimal ventilation. An explosion could be initiated by static electricity or by a match lighted in the vicinity of the hydrogen. No spray system would be working at the time of the event. The waste is assumed to be in piles.

5.4.1.3 Release Estimate for Upper-Bound Event

This event is postulated to breach the facility and generate a 1000-m^3 cloud with total mass airborne of 5 kg. It was assumed that the airborne mass might be as great as the maximum found in dust devils, 5 g/m^3 (Sinclair 1976). The waste airborne would be fragments of contaminated soil. Only a portion of the airborne material would be in the respirable size of concern less than 10 μm aerodynamic equivalent diameter. Sutter (1980) measured the size of some Hanford soil and determined that this respirable fraction was

0.088%. Using this value to estimate the respirable fraction reduces the release of concern to 5 g, as listed in Table 5.11.

5.4.1.4 Other Accidents Considered

A variety of accidents were considered before selecting the upper-bound event. These are developed below and listed in Table 5.11.

Spills. Spills of contaminated soil could occur. For one incident of this kind, a fractional release of 1×10^{-4} was assumed (DOE 1979). Sutter, Johnston and Mishima (1981) measured airborne releases from powders spilled in static air and found the maximum release fraction to be 0.12%. These values show the spill to have a low potential for release.

Facility Fire. A facility fire during retrieval would be unlikely since the equipment is battery-operated so there is no fuel source. There would be insufficient combustible material in the structure to support a fire. An external fire such as a range fire igniting the facility was considered an unlikely event because of the sparseness of vegetation in the work areas.

Filter Failure. High-efficiency particulate air filter failure or building leak are similar events. They could occur because of human error or equipment failure. Water sprays would mitigate the event. Murphy and Holter (1980) identified the air dust loading at 1.0 mg/m^3 and the calculated release for filter failure was a 1.2×10^{-6} fraction, a building leak three orders of magnitude less. They identified the frequency of filter failure at less than 1.0×10^{-5} per year, the building leak at greater than 1.0×10^{-9} per year.

TABLE 5.11. Postulated Airborne Releases from Accidents During During Mechanical Retrieval of TRU-Contaminated Soil

Event	Atmospheric Airborne Release
Explosion	5 g
Spills	1.2×10^{-3} fraction
Facility fire	No significant release
Filter failure	
1st stage	5×10^{-6} g/m^3
Both stages	1.2×10^{-2} g/m^3

Based on aerosol behavior (ORNL 1970) 10 mg/m^3 is assumed to be the maximum concentration of quasi-stable aerosol in the facility. Releases per cubic meter are listed in Table 5.11.

5.4.2 Processing

TRU-contaminated soil would be sent to a waste processing facility to be treated to form a chemically inert, physically stable, basalt-like slag that meets repository requirements for immobilized waste forms.

5.4.2.1 Description of Operation and Facility

The retrieved contaminated soil would be transferred from transport trucks to a receiving airlock at the processing facility. The waste processing facility would house the equipment; ventilation air would be discharged through high-efficiency particulate air filters. 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.

The processes for converting the waste to a stable product have not been selected. One possible concept used for analytical purposes, a vertical furnace with two main components, a gasifier and a secondary combustion chamber, would constitute a Slagging Pyrolysis Incineration unit as shown in Figure 5.11. The gasifier has three zones: drying, pyrolysis, and combustion. A secondary chamber completes combustion of the off gas, which is then cooled and filtered. Slag would be poured into molds, assayed, and prepared for transport to the geologic repository.

5.4.2.2 Postulated Upper-Bound Accident

A fire and deflagration involving contaminated materials in the fuel-rich gasifier is the postulated upper-bound airborne release event. This is suggested since explosions can result from the ignition of clouds of fine wood or coal dusts, diesel oil mists, or rich fuel mixtures (Orr 1966). Process malfunction allows carbon monoxide to reach the incinerator drying section where it reacts rapidly with the air introduced with the waste. The mixture deflagrates, resulting in failure of the upper portion of the gasifier due to overpressurization. The concrete structure is breached in the explosion.

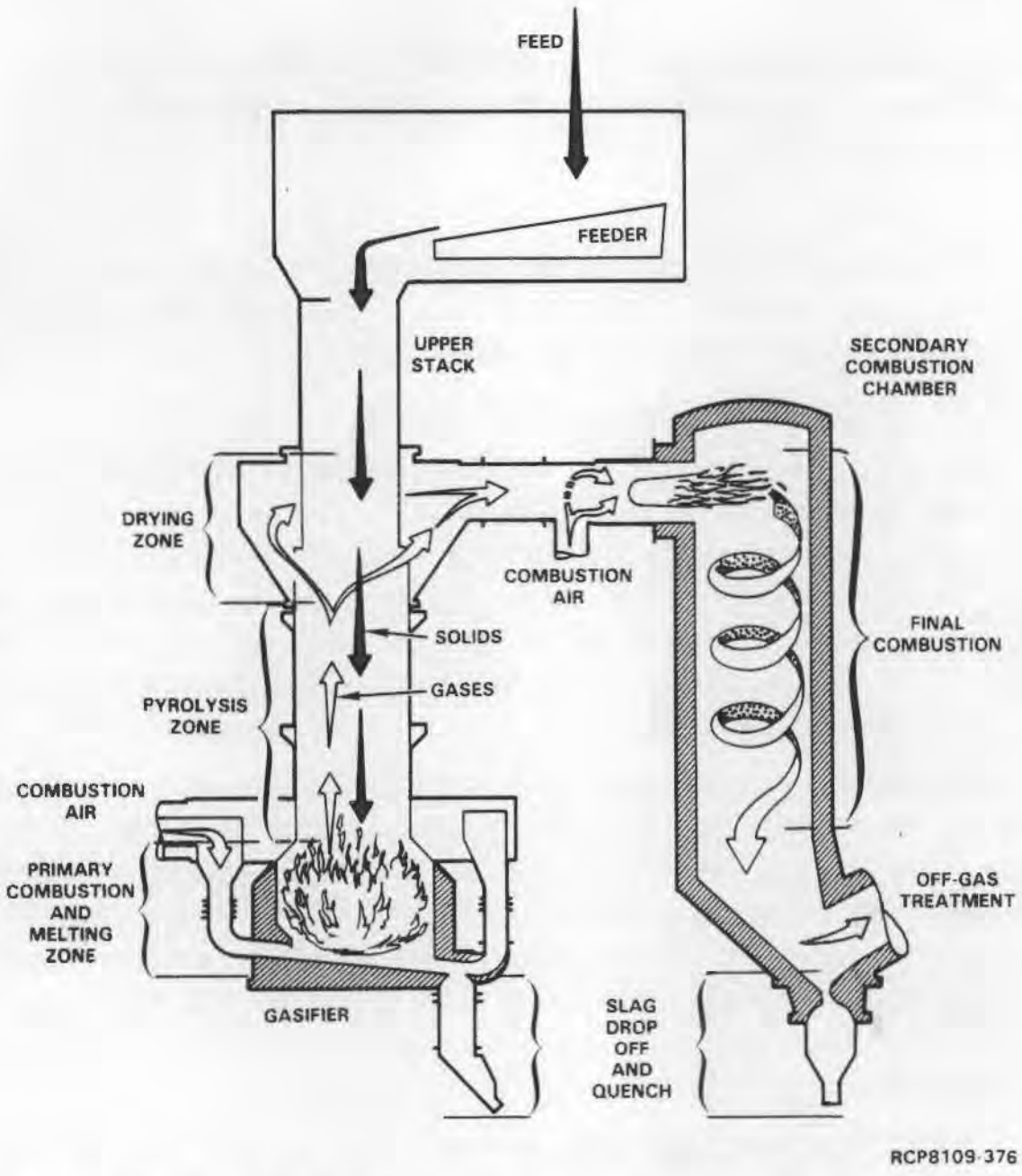


FIGURE 5.11. Slagging Pyrolysis Incineration Gasifier and Combustion Chamber

5.4.2.3 Release Estimate for Upper-Bound Event

A 1000 m^3 cloud filled with radioactive particles is assumed to be generated by the explosion. Mishima (1975) estimated that explosion generated clouds can attain a quasi-stable concentration of particles of 100 mg/m^3. The particles in the cloud are considered to have a size distribution typical of

particles in the secondary combustion chamber off gas with 50% in the fraction 10 μm aerodynamic equivalent diameter and less as suggested by the work of Christian et al. (1978) and Kirstein et al. (1979). The total release of respirable particles is 50 g. Only a portion of the release will be TRU-contaminated soil. However, for purposes of this analysis it is all assumed contaminated. Since the facility is breached, the release is atmospheric, as listed in Table 5.12.

5.4.2.4 Other Accidents Considered

Other accidents were considered before the upper-bound release was selected. They are developed in the following sections and listed in Table 5.12.

Criticality. A criticality would not be considered a credible accident because of administrative procedures. These could include screening waste for fissile content, introducing a small concentration of neutron absorbent to the feed, and using neutron detectors in the facility, particularly in the drying, pyrolysis, and slag regions (Close, Booth and Caldwell 1981). Criticality therefore would require multiple failure of administrative and system controls accompanied by a highly unusual fuel-fissile concentration or an accumulation of fissile material (Kirstein et al. 1979).

Spill. A slag spill could occur as the waste is cast. The release would be less than that anticipated for a powder spill, and is assumed to be similar to that from a liquid, 0.01% (Sutter, Johnston and Mishima 1981). If the amount spilled is equivalent to the amount in a drum, about 56 g could be airborne in the facility for an atmospheric release of 1.4×10^{-5} g. A release after the waste is cast could have lower releases, a 1×10^{-6} fraction.

TABLE 5.12. Postulated Airborne Releases from Accidents During Slagging Pyrolysis Incineration of TRU-Contaminated Soil

Event	Atmospheric Airborne Release
Explosion	50 g
Criticality	No significant release
Spill	1.4×10^{-5} g

5.5 PRE-1970 TRU SOLID WASTE

Buried TRU solid waste would be retrieved and processed using procedures similar to those proposed for use at TRU-contaminated soil sites.

5.5.1 Mechanical Retrieval of Pre-1970 TRU Solid Waste

Retrieval procedures would be similar to those described above (Section 5.4.1), with some additions because solid wastes are involved. Only additional operations are discussed.

5.5.1.1 Description of Operation and Facility

Additional equipment (beyond that used for TRU-contaminated soil) required for solid waste includes waste sizers that could perform sawing, shearing, hammering, and bending operations. If the size could not be reduced, a heavy-duty backhoe would be used to secure the oversize waste for special handling.

5.5.1.2 Postulated Upper-Bound Accident

This was described in Section 5.4.1.2.

5.5.1.3 Release Estimate for Upper-Bound Event

Since this is the same event as for TRU-contaminated soil, the estimate method is the same.

5.5.1.4 Other Accidents Considered

The other accidents considered are the same as those in Section 5.1.4, but since solid material can be included, there are other potential accidents as listed below. All releases are also listed in Table 5.13.

Leaks. A leak from a breach drum is an accident that could be considered. During waste retrieval operations at Idaho National Engineering Laboratory this type of accident occurred. During one period, 70% of the drums retrieved from one pit and all from another were breached. Free liquids leaked from 9% of the drum and 5% were externally contaminated (McKinley and McKinney 1978). In one case several liters of contaminated liquid leaked from a deteriorated container. Eventually 8000 cm^2 of soil surface was contaminated and exhibited mobility requiring an asphalt spray to fix the contamination to the soil. No activity passed through the high-efficiency particulate air filters

subsequent to this event (Harness and McKinney 1977). This experience indicates that little or no radioactive material would be released from the building as a result of a leak.

Pressurized Release. A pressurized release in the preparation area could occur if gases had built up in a drum. Various sources of gas are possible (e.g. radiolysis, decay of materials, spontaneous oxidation, etc.). The reactions which result in the loss of integrity of the vessel may also vary, but the forces that result in the airborne dispersion of particles are limited by the failure pressure of the container. The failure pressure of the container is a function of many characteristics of the container. The volume of gas increase that results in that pressure can also be small (a pressure of 50 psig is only a three-fold increase in the free volume of a container). A failure pressure of 50 psig was assumed to be conservative for 55-gallon drums buried for extended periods. At 50 psi, the release factor is 1% of the respirable particles, less than 10 μm aerodynamic equivalent diameter in diameter. The filters should not be damaged by this release.

TABLE 5.13. Postulated Airborne Releases from Accidents During Mechanical Retrieval of Pre-1970 TRU Solid Waste

Event	Atmospheric Airborne Release
Explosion	5 g
Filter failure	
1st stage	5×10^{-6} g/m^3
Both stages	1.2×10^{-2} g/m^3
Spills (powder)	1.2×10^{-3} fractional release
Facility fire	1.3×10^{-10} fractional release
Liquid leaks	Significantly below upper-bound
Pressurized release	1×10^{-2} fraction
Spread of surface contamination	Significantly below upper-bound
Container fire	1.3×10^{-13}

Spread of Surface Contamination. This release could occur when contaminated containers are recovered. It should be anticipated, and could possibly even be considered a routine release. The release would be at a low level and removed by the high-efficiency particulate air filters.

Fire. A fire in a waste container could occur if nitric acid leaked onto cellulosic material. This could lead to a possible detonation at slightly elevated temperatures due to self ignition (Mulkin 1975). The release fraction for combustibles contaminated with powder could be 5×10^{-4}. High-efficiency particulate air filtration would prevent a significant amount of radionuclides from entering the atmosphere, with the atmospheric fractional release 1.3×10^{-10}.

5.5.2 Sorting and Related Operations

The solid waste could require steps beyond those described for soil (5.4.2.1).

5.5.2.1 Description of Operation and Facility

Solid waste first would be sorted and sized. Any materials requiring special handling would be separated from the waste stream and treated as needed. Large items would be sized by crushing, shredding, or flattening suitable for incineration. After the waste was blended to achieve uniformity, it would be fed to a processing unit.

5.5.2.2 Postulated Upper-Bound Accident

A pressurized release from a ruptured container is postulated. The accidents could be the result of radiolytic gases building up within a drum.

5.5.2.3 Release Estimate for Upper-Bound Event

Based on experimental studies (Sutter 1983), a pressurized release of 1% of the contents is estimated becoming airborne as less than 10 μm aerodynamic equivalent diameter particles. This assumes the drum ruptures at 50 psi. Assuming a 210 L volume and waste density 0.96 g/cm^3, the drum could contain as much as 2×10^5 g of waste, thus 2×10^3 g could be airborne. The release will challenge two high-efficiency particulate air filters and 5×10^{-4} g is the atmospheric release listed in Table 5.14.

TABLE 5.14. Postulated Airborne Releases from Accidents During Sorting and Related Operations with Pre-1970 Solid Waste

Event	Atmospheric Airborne Release
Pressurized release	5×10^{-4} g
Leaking drum	1×10^{-4} fractional release
Spill	6×10^{-5} g
Fire	2.5×10^{-5} g
Contamination spread	Significantly below upper-bound
Facility fire	Not credible

5.5.2.4 Other Accidents Considered

These releases are also listed in Table 5.14.

Leak. A leaking drum with liquid spilling from it could have a release of 0.01% of the source, based on experimental measurements of liquid spills (Sutter, Johnston and Mishima 1981). Therefore, this accident would have a lower release than some of the other postulated events.

Spill. A drum breach/spill of powder could have a fractional release of 0.12% in a static air situation. This would be a release of 240 g in the facility, challenging the high-efficiency particulate air filters for an atmospheric release of 6×10^{-5} g.

Fire. A fire could occur in powder-contaminated combustibles in a drum. The experimental release factor (Mishima and Schwendiman 1973) developed for this type event would be 100 g/drum inside the facility, 2.5×10^{-5} g releasing to the atmosphere through 2 stages of high-efficiency particulate air filters. Since the facility and operation are still in the conceptual design phase, we cannot evaluate whether more than one drum would likely be involved in a fire. Since the drums are not opened, there would probably be no ignition source, so a fire is considered unlikely.

Contamination Spread. External contamination spread of smearable contamination from the exterior of waste packages could occur. This release would

probably involve only very small levels of contamination overlooked when the packages were emplaced or from spread of package leaks. They would be at a lower level than spill or fire releases.

Facility Fire. Facility fires could be started by an external event such as a range fire or fuel truck hitting the facility. However, sparseness of the desert vegetation would lower the probability of such an event. The facility design is still conceptual, but it would probably be built of materials that would not sustain a fire.

5.5.3 Processing

Retrieved solid waste would be sent to the same facility and treated by slagging pyrolysis incineration much the same as TRU-contaminated soil (Section 5.4.2).

5.5.3.1 Description of Operation and Facility

For purposes of the Hanford Defense Waste Environmental Impact Statement slagging pyrolysis is used, as described earlier.

5.5.3.2 Postulated Upper-Bound Accident

This will be the same as in Section 5.4.2.2.

5.5.3.3 Release Estimate for Upper-Bound Event

This is the same as for TRU-contaminated soil in Section 5.4.2.3, and is listed in Table 5.15.

5.5.3.4 Other Accidents Considered

These will be the same as for TRU-contaminated soil, with some additions considered because the waste is contained. Releases are included in Table 5.15.

Fire. Various fires could be suggested, but they would have a lower release than the upper-bound event, which postulated breaching the facility. One such fire suggested that 2000 ft^3 of waste in the shipping area had a fractional release of 1×10^{-9}, one in the waste preparation area with 1×10^{-6}

TABLE 5.15. Postulated Airborne Releases from Accidents During Processing of Pre-1970 TRU Solid Waste

Event	Atmospheric Airborne Release
Explosion	50 g
Criticality	Not postulated
Spill	1.4×10^{-5} g
Fire	1×10^{-6} fractional release[a]
Dropped container	2.5×10^{-11} fractional release

(a) DOE (1979).

fractional release (DOE 1979). No fire was identified that was severe enough to burn through the reinforced concrete walls or destroy the second high-efficiency particulate air filter.

Dropped Container. A dropped waste container was postulated to have a fractional release of 1×10^{-4} (DOE 1979), entering the filtering system.

5.6 RETRIEVABLY STORED AND NEWLY GENERATED TRU

Generated TRU solid waste in this category is remote handled or contact handled.

5.6.1 Waste Retrieval

Different retrieval methods are applied to remote-handled TRU and contact-handled TRU.

5.6.1.1 Remote-Handled TRU

Operation and Facility. Remote-handled TRU in caissons would be mechanically retrieved using an airtight, double-walled structure installed over the caissons. This building is shown in Figures 5.12 and 5.13, with dimensions included. The caisson access shaft area was estimated to occupy about a quarter of the building. A conveyor system is used to transfer remote handled casks containing retrieved waste.

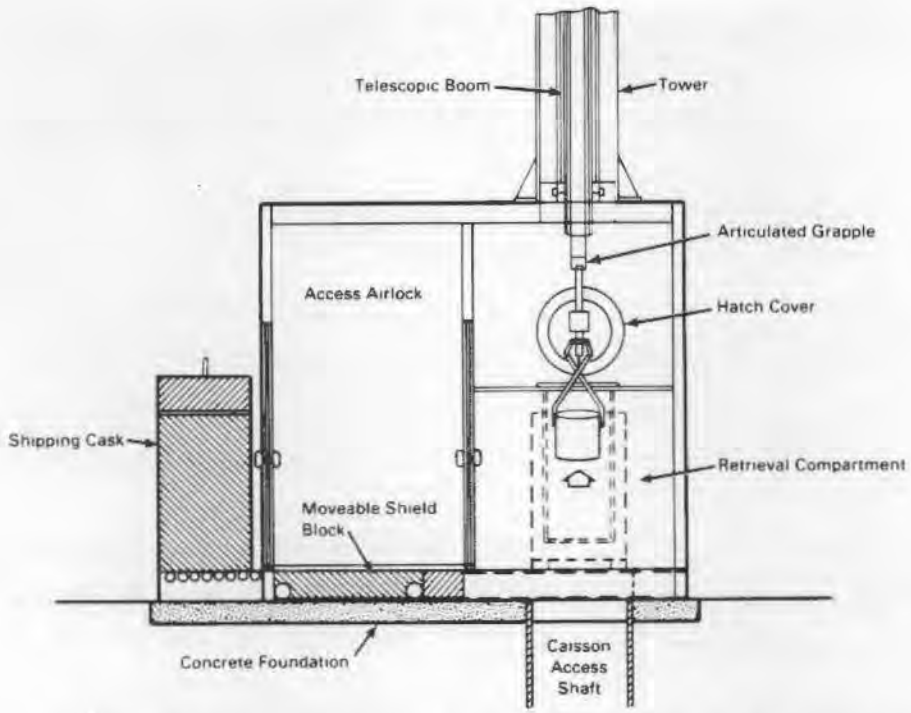


FIGURE 5.12. Caisson Recovery Building, Side View

The waste is retrieved using a grappler housing with a telescoping articulated boom. Operations are conducted remotely.

Postulated Upper-Bound Release Accident. The upper-bound accident is postulated as an explosion or pressurized release. The container is pressurized due to the buildup of radiolytic gases and subsequently ruptures.

Release Estimate for Upper-Bound Event. As noted in Section 5.5.2.3, 1% of the source would become airborne as respirable particles. Since the waste can be in 3.82-L metal cans, has a density of 0.96 g/cm^3, and the source is 3.7×10^3 g, 37 g is airborne in the facility. Damage to the filters was considered but not postulated since a missile must be shot in perfect alignment with the filter to penetrate it. The postulated atmospheric release in 9.3×10^{-6} g. The release is listed in Table 5.16.

Other Accidents Considered. These releases are also listed in Table 5.16.

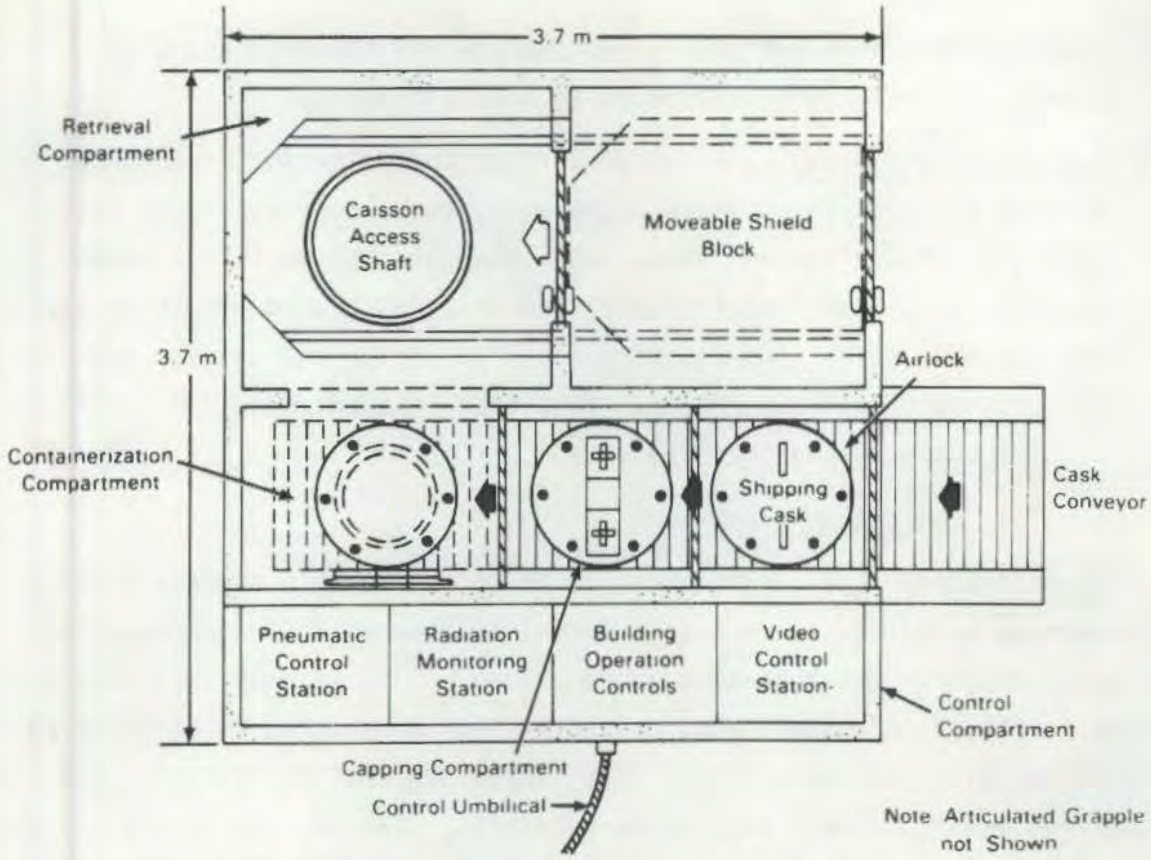


FIGURE 5.13. Caisson Recovery Building, Top View

TABLE 5.16. Postulated Airborne Releases from Accidents Retrieving Remote-Handled TRU in Caissons

Event	Atmospheric Airborne Release
Pressurized release	9.3×10^{-6} g
Spill	1×10^{-6} g
Contamination spread	Significantly below upper-bound

- Spill. A spill of material from a ruptured or breached package could release waste at a maximum level of about 0.12% as measured by Sutter, Johnston and Mishima (1981) in free-fall spills of powders in static air. The release would be within the facility or shaft with a minimal airflow, so a significantly higher release is not to be anticipated. Four grams would be airborne in the facility and

challenge the high-efficiency particulate air filters, and the atmospheric release is estimated as 1×10^{-6} g.

- Contamination Spread. Spread of surface contamination could occur because the outside of waste packages could be contaminated. Packages are allowed to fall freely when they are stored in the caissons (Geiger, Brown and Isaacson 1977) and this fall could result in package ruptures. The contamination could be inside the caisson and on the outside of waste packages. This type of event should not lead to significant atmospheric releases.

5.6.1.2 Contact-Handled TRU

Operation and Facility. Waste placed in retrievable storage trenches and above-ground buildings is free of external contamination and packaged to maintain integrity for a minimum of 20 years. It is assumed that this waste can be retrieved in an open environment without generating an airborne release of radioactivity (Rockwell 1985). The overburden will be removed using conventional equipment and hand digging as required. Once the overburden has been removed, the packaged waste will be removed by a forklift or crane.

Postulated Upper-Bound Accident. An explosion or pressurized release of the contents of a waste package are suggested as a maximum release event. The pressurized release is probably initiated by the buildup of radiolytic gases.

Release Estimate for Upper-Bound Event. A 210-L drum packaging would appear to be susceptible to pressure buildup. The fractional release for remote-handled TRU, 1% respirable, is applicable here. At a waste density of 0.96 g/cm^3, the source is 2×10^5 g and 2×10^3 g becomes airborne immediately. The release is listed in Table 5.17.

Other Accidents Considered. Releases from other postulated accidents are listed in Table 5.17.

- Spill. Waste containers can breach or rupture during various operations, and the contents spill. A portion of the contents could become airborne. An appropriate release fraction for this type of

TABLE 5.17. Postulated Airborne Releases from Accidents During Grouting Decontaminated Salt Solutions

Event	Atmospheric Airborne Release
Pressurized release	2×10^3 g
Spill	5×10^{-3} fractional release
Puncture	No significant release
Fire	5×10^{-4} fractional release
Range fire	No significant release
Contamination spread	No significant release
Equipment failure	No significant release

event might be 0.12%, as discussed in Section 5.6.1.1.4, if the event occurs in static air. However, if the operation is outside, it could be suspended by wind. If the wind is blowing at the average Hanford windspeed and 10% of the spilled material interfaces with the wind, the release fraction could be 5×10^{-3} (Sutter 1980).

- Puncture. A waste package could be punctured by a forklift during retrieval. If a single hole is punched into the package, only a small amount of the waste is pulled out during withdrawal of the probe. This type of event should not have significant releases.
- Fire. A fire in a package of combustible waste could occur if nitric acid, for example, spilled onto cellulosic waste. The fractional release could be 5×10^{-4} (Mishima and Schwendiman 1973). The largest box could contain 2×10^7 g of waste, 64 m^3 at a waste density of 0.32 g/cm^3. However, the package would not be anticipated to hold that quantity of combustible waste. These large packages would probably hold large equipment that could not be decontaminated, pumps, tools, etc. Therefore it is considered unlikely that this waste would sustain a fire.
- Range Fire. A range fire is not considered a threat to this operation. The gravel covering the waste would insure sparseness of vegetation so there would not be combustibles to fuel the fire.

- Contamination Spread. Spread of contamination could occur if packages of waste with exterior contamination are excavated. Administrative controls as the waste is emplaced should ensure that spread does not take place. Although a package could leak, it would not involve significant levels of contamination.
- Equipment Failure. An equipment-related event such as an overturning of a front-end loader could impact the waste. Such an event should lack sufficient force to cause significant releases since the vehicles will be operating at very low speeds.

5.6.2 Sorting and Related Operations

5.6.2.1 Remote-Handled TRU

Remote-handled TRU would be processed with the TRU-contaminated soil and pre-1970 solid waste. Therefore, the operation and facility, upper-bound accident, release estimate for the upper-bound event, and other accidents considered will be the same as those for sorting the solid waste. These were discussed in Section 5.5.2 and listed in Table 5.14.

5.6.2.2 Contact-Handled TRU

Contact-handled TRU waste will be assayed and segregated in the contact-handled Waste Receiving and Processing facility. Thus, wastes are identified as to whether they are low-level or whether they are handled in this facility. The operation and facility, upper-bound accident, release estimate for the upper-bound event, and other accidents considered will be the same as those for other sorting operations. These were discussed in Section 5.5.2 and listed in Table 5.14.

5.6.3 Processing

5.6.3.1 Remote-Handled TRU

Since the remote-handled TRU is processed by the slagging pyrolysis incinerator, the operation and facility, upper-bound accident, release estimate for upper-bound release event, and other accidents considered are the same as postulated in Section 5.4.2. The releases from the slagging pyrolysis incinerator are listed in Table 5.12.

5.6.3.2 Contact-Handled TRU

Operation and Facility. Processing will involve size reduction, shredding and immobilizing the waste in concrete. The size reduction operation will select materials for the shredding that will prepare the waste to a form readily adaptable to immobilization. The waste is immobilized using a concrete mixture.

Postulated Upper-Bound Accident. Fire is suggested as the bounding accident for this operation. It is postulated to occur during shredding combustible waste. Waste dumped into the feed hopper could inadvertently contain solvent that spills on cellulosic material. Sparks from the grinder could then ignite the waste.

Release Estimate for Upper-Bound Event. One of the larger packages used is an 80-gal (317 L) drum (Rockwell 1985). The material could be loosely packed cellulosics contaminated with radioactive powder. An overall density could be 0.32 g/cm^3, so the potential source is 1.5 x 10^5 g. The appropriate release factor is 5 x 10^{-4} of radioactive material (Mishima and Schwendiman 1973), and the calculated release in the facility is 50 g/drum of fine particles. It is conceivable that more than one drum load is in the hopper. Two drums would release 100 g in the facility to challenge the high-efficiency particulate air filters for the 2.5 x 10^{-5} g release listed in Table 5.18.

TABLE 5.18. Postulated Airborne Releases from Accidents During Sorting and Related Operations with Contact-Handled TRU

Event	Atmospheric Airborne Release
Fire	2.5 x 10^{-5} g
Explosion	No significant release
Handling Accident	2 x 10^{-11} fractional release
Spill	3 x 10^{-10} fractional release
Leak	2 x 10^{-11} fractional release
Pressurized release	Not postulates

Other Accidents Considered. Releases from the following less-than-upper-bound accidents are listed in Table 5.18.

- Explosion. An explosion was considered at the Idaho National Engineering Laboratory (DOE 1979), but no potential source was identified for an explosion strong enough to breach the walls of the facility or to fail both high-efficiency particulate air filters and their associated dampers.

- Handling Accident. A handling accident has been postulated to occur in the compaction area at the Idaho National Engineering Laboratory (DOE 1979) with a release fraction of 10^{-4} in the ventilation flow. Particles generated by this event would be larger than those produced in a fire, making it of lesser concern.

- Spill. A waste container breach and spill could have a maximum 0.12% fractional release (Sutter, Johnston and Mishima 1981) and produce fairly large particles. While the fractional release appears larger than for the fire, the size of the particles make it less of an inhalation hazard. A second mitigating consideration is that the spill could contain a larger portion of nonradioactive material.

- Leak. A liquid leak should result in a 0.01% fractional release. It would be assumed that very little liquid would be contained in the waste at this process step. Leaks would be identified in the earlier sorting and related operation.

- Pressurized Release. Pressurized releases from drums have been suggested in earlier process modules. The materials are not packaged during the sorting and shredding operations. Reactions by the materials would be most likely to occur, if any reactions are possible, during this time. Only after these operations is the material restrained in concrete. Pressurized release is not considered as a likely event for this process step.

6.0 POSTULATED RELEASES FROM POTENTIAL ACCIDENTS ASSOCIATED WITH IN-PLACE STABILIZATION AND DISPOSAL OF SIX WASTE FORMS

For the in-place stabilization and disposal alternative, there would be little processing or treatment of wastes except for those stored in double-shell tanks. Releases for each of the waste forms are developed in the following sections, and are listed in Table 6.1.

6.1 EXISTING TANK WASTE

6.1.1 Dry Single-Shell Tanks

Waste left in place in single-shell tanks would be dried as required to achieve adequate stability.

6.1.1.1 Description of Operation and Facility

One proposed drying method would use microwave energy to heat and thereby dry the waste. This technique could radiate microwave energy into the interior of an underground tank and heat the waste in a manner similar to microwave cooking. Portable generators would transmit energy into the tank interior via coaxial cables. Air would be discharged through two high-efficiency particulate air filters.

6.1.1.2 Postulated Upper-Bound Release Event

The postulated upper-bound release event would be the explosion described for single-shell tanks in Section 5.1.1.2.

6.1.1.3 Release Estimate for Upper-Bound Event

Section 5.1.1.3 develops an estimate of 1.3×10^4 g of respirable material airborne and is listed in Table 6.2.

6.1.1.4 Other Accidents Considered

Some other accidents considered are listed in Table 6.2.

TABLE 6.1. In-Place Stabilization and Disposal Alternative Potential Accidental Releases for Operations Involving Six Waste Forms

	Waste Form	Accident Source Quantity	Facility	Accident	Location	Fractional Release	Transmission Factor[a]	Atmospheric Release of Concern, g	Release Point
Existing Tank Waste									
1. Dry single-shell tanks	Salt cake	2000 m^3	Tank	Explosion	Tank	NA[b]	1	1.3×10^4 g	Ground level
2. Fill tank dome	Salt cake		Tank	Dome collapse	Tank	NA	1	7×10^3	Ground level
3. Hydraulic retrieval of residual liquid (double shell)	Liquid waste	9×10^4 g/min	Tank	Pressurized spray	Diversion valve	5×10^{-2}	1	4.5×10^3	Ground level
4. Complexant destruction	No information available								
5. Grout	Aqueous solutions	10 m^3	Transportable grout	Liquid spray from line	Transfer line	1×10^{-4}	2.5×10^{-7}	3×10^{-4}	Stack
6. Trench disposal	No release situation detected								
7. Fill empty tank dome	Residual tank waste	5% of original fill	None	Dome collapse	Tank	5×10^{-2}	1	350	Ground level
Future Tank Waste									
1. Hydraulic retrieval	Liquid waste	9×10^4 g/min	Tank	Pressurized spray	Diversion valve	5×10^{-2}	1	4.5×10^3	Ground level
2. Cesium removal	Liquid waste		B plant	Ion exchange fire	Resin column	0.01	3×10^{-4}		Stack
3. Cesium encapsulation	System off gas	2 g/min	B plant	Off-gas system failure	Filters			2	Stack
4. Grout	Liquid feed	10 m^3	Transportable grout	Spray from feed line	Transfer line	1×10^{-4}	2.5×10^{-7}	3×10^{-4}	Stack
5. Fill tank	Residual tank waste	0.05% of original fill	None	Dome collapse	Tank	5×10^{-4}	1	3.5	Ground level

TABLE 6.1. (contd)

	Waste Form	Accident Source Quantity	Facility	Accident	Location	Fractional Release	Transmission Factor[a]	Atmospheric Release of Concern, g	Release Point
Strontium and Cesium Capsules									
1. Remove from WESF	Encapsulated waste	2×10^3 g	WESF[c]	Capsule rupture	Storage area	1×10^{-2}	2.5×10^{-7}	5.5×10^{-6}	Stack
2. Capsule packaging	Encapsulated waste	2.2×10^3 g	CPF[d]	Machinery impacts capsules	"Load" Section	1.2×10^{-3}	2.5×10^{-7}	6.6×10^{-7}	Stack
3. Place in drywell storage	Encapsulated waste	2.2×10^3 g	DWSF[e]	Transporter shears capsule	DWSF	1×10^{-6}	1	2.2×10^{-3}	Ground level
TRU-Contaminated Soil									
1. Inject grout	TRU-contaminated soil	8.1×10^7 g	None	Void space collapse	CRIB	3×10^{-6}	1	2.6	Ground level
Pre-1970 TRU Solid Waste									
1. Grout caissons	Caisson	3.7×10^3 g	None	Equipment failure	Caisson	7×10^{-1}	1	3×10^{-4}	Ground level
2. Subsidence control	Solid waste site	8.1×10^7 g	None	Void space collapse	Burial site	3×10^{-6}	1	2.6	Ground level
Retrievably Stored and Newly Generated TRU									
1. Subsidence control	Retrievable stored TRU	8.1×10^7 g	None	Void space collapse	Burial site	3×10^{-6}	1	2.6	Ground level
2. Bury packaged waste	TRU waste	2×10^5 g	None	Package breach	Burial site	5×10^{-3}	1	1×10^3 g	Ground level

(a) A transmission factor of 1 means the high-efficiency particulate air filters are breached or not enclosed.
(b) Not Applicable.
(c) Waste Encapsulation and Storage Facility.
(d) Capsule Packaging Facility.
(e) Dry Well Storage Facility.

TABLE 6.2. Postulated Airborne Releases from Accidents During In-Tank Drying of Single-Shell Waste

Event	Atmospheric Airborne Release
Explosion	1.3×10^{4} g
Loss of filtration	
1st stage	5×10^{-6} g/m^3
Both stages	1×10^{-2} g/m^3
Loss of services or power	No significant release
Spattering	2.5×10^{-9} g/m^3

Spattering could occur if internal heating of the waste were uneven; pressure could build up, with subsequent ejecting of waste through the crust into the tank void space. It is estimated that 100 mg/m^3 (ORNL 1970) is in the airflow, with 10% less than 10 μm aerodynamic equivalent diameter in diameter. This event should not breach the filters, so the release would be at a lower level than the maximum.

6.1.2 Tank Dome Filling

Tank dome voids above the waste in both the single- and double-shell tanks will be filled.

6.1.2.1 Description of Operation and Facility

The same techniques used to fill empty tanks will be used to fill the dome voids. These were discussed in Section 5.1.6.1.

6.1.2.2 Postulated Upper-Bound Accident

A dome collapse would be the postulated upper-bound accident as discussed in Section 5.1.6.2.

6.1.2.3. Release Estimate for Upper-Bound Event

The source quantity available to be released will be larger from these full tanks than the emptied tanks. Therefore a larger release is anticipated from this dome collapse. Rockwell (1985) estimated a 7-kg release; Hayward and

Jensen (1980) suggested a 5×10^{-8} fractional release. The 7-kg value is the selected upper-bound release and is listed in Table 6.3.

6.1.2.4 Other Accidents Considered

These accidents will be the same as those discussed in Section 5.1.6.4 and are listed in Table 6.3. They include loss of filtration, loss of service or power, and equipment failure.

6.1.3 Hydraulic Retrieval of Residual Liquid

Residual liquor and other liquid waste from double-shell tanks would be retrieved hydraulically.

6.1.3.1 Description of Operation and Facility

The sluicing operation is described in Section 5.1.2.1.

6.1.3.2 Postulated Upper-Bound Accident

The sluicing operation will be the same as that described in Section 5.0; therefore, the accident hypothesized in Section 5.1.2.2, a pressurized release of sprayed liquid waste, is the upper-bound accident.

TABLE 6.3. Postulated Airborne Releases from Accidents During Dome Filling

Event	Atmospheric Airborne Release
Dome collapse	7 kg
Loss of filtration	
1st stage	5×10^{-6} g/m^3
Both stages	1×10^{-2} g/m^3
Loss of services or power	No significant release
Equipment failure	No significant release

6.1.3.3 Release Estimate for Upper-Bound Event

As described in Section 5.1.2.3, 9×10^4 g/min is pumped and has a 5×10^{-2} fractional release in the accident. Thus the release is 4.5×10^3 g/min, as listed in Table 6.4.

6.1.3.4 Other Accidents Considered

These are discussed in Section 5.1.2.4, and include a slurry spill, loss of services or power, and loss of filtration. Releases are listed in Table 6.4.

6.1.4 Complexant Destruction

Wastes with high concentrations of organic complexants would be treated to reduce the concentrations to an acceptable level. This process has not been developed, so no releases can be estimated.

6.1.5 Grout

Treated residual aqueous solutions would be treated to convert them to a cementitious grout. The waste would be processed through the transportable grout facility. The operation and facility, postulated upper-bound accidents, the release estimate for upper-bound event, and other accidents considered will be the same as those in Section 5.1.5. Releases are listed in Table 6.5.

TABLE 6.4. Postulated Airborne Releases from Accident During Hydraulic Retrieval of Residual Liquid

Event	Atmospheric Airborne Release
Pressurized liquid spray	4.5×10^3 g/min
Slurry spill	1×10^{-4} fraction, plus resuspension
Loss of services or power	No significant release
Loss of filtration	
1st stage	5×10^{-6} g/m^3
Both stages	1×10^{-2} g/m^3

TABLE 6.5. Postulated Airborne Releases from Accidents During Grouting Operation

Event	Atmospheric Airborne Release
Spray (solution)	3×10^{-4} g
Solution tank leak	2×10^{-8}(a)
Mixer tank leak	Significantly below upper-bound
Grout spill	Significantly below upper-bound

(a) Hayward and Jensen (1980).

6.1.6 Trench Disposal of Grout

Grouted waste will be disposed of in trenches.

6.1.6.1 Description of Operation and Facility

The grout burial trenches are of earthen construction with a membrane liner. They are 15 m wide, with sloping sides, and are 8 m deep. This trench, 100 m long and filled to a depth of 3 m, can hold 5400 m^3 of grouted waste. An air support bubble dome is used during filling for weather protection and to prevent airborne dispersal of the waste. Once the trench is filled and the grout is cured, the dome will be removed and the trench backfilled with 5 m of soil.

6.1.6.2 Postulated Upper-Bound Accident

No significant potential for airborne release was identified.

6.1.6.3 Accidents Considered

Since no releases were developed, a table was not developed.

Grout Leak. It is postulated that grouted liquid is inadvertently released. A variety of mechanisms could be responsible, such as a leaking connection or a break while making the grout. The quantity released would depend upon the circumstances. But the nature of the material makes a serious airborne release unlikely. The grout is viscous and difficult to subdivide

into fine particles (much of the material used to make the grout is coarse). The level of radioactivity in the grout is low. Therefore, it is estimated that this type of event would result in an insignificant (but unquantified) airborne release.

Grout Spray. In order for the grout to be sprayed, a considerable pressure would be required, much in excess of that needed simply to move the grout. No situation which could result in such a situation was postulated, and this event was not considered significant.

Slope or Trench Failure During Loading. The grout is laid in the bottom of a trench through multiple nozzles. If the sides of the trench were to collapse due to insufficient strength of the soil, the soil sliding into the trench could cause the grout to be splashed. As mentioned in the grout spill scenario, the airborne release would be insignificant and the most probable result of this event would be the inadvertent burial of the grout.

Failure of the Air Bubble Dome. Loss of the air bubble dome would expose the uncured grout to the atmosphere and, in inclement weather, might interfere with the proper curing and aging of the grout. These are operational concerns but do not appear to pose a serious airborne hazard.

Loss of Services or Power. Loss of services or power could result in cessation of operations or exposure of the grout. Neither consequence appears to pose a serious airborne release situation.

6.2 FUTURE TANK WASTE

6.2.1 Hydraulic Retrieval

Wastes in double-shell tanks (neutralized current acid wastes) would be retrieved by hydraulic sluicing.

6.2.1.1 Description of Operation and Facility

The sluicing operation is described in Section 5.1.2.1.

6.2.1.2 Postulated Upper-Bound Accident

The sluicing operations are the same, therefore the suggested accident in Section 5.1.2.2 applies, i.e., a release of pressurized liquid as a spray.

6.2.1.3 Release Estimate for Upper-Bound Event

As described in Section 5.1.2.3, 9×10^4 g/min is pumped and has a 5×10^{-2} fractional release in the accident. Thus the release if 4.5×10^3 g/min as listed in Table 6.4.

6.2.1.4 Other Accidents Considered

These are described in Section 5.1.2.4 and listed in Table 6.4.

6.2.2 Cesium Removal

6.2.2.1 Description of Operation and Facility

Due to the small volume of waste involved, the waste is processed through B Plant and the waste encapsulation and storage facility. The cesium is removed in an exchange column and converted to solid cesium chloride for encapsulation.

6.2.2.2 Postulated Upper-Bound Release Accident

The upper-bound is the ion exchange fire postulated in Section 5.1.3.3.

6.2.2.3 Release Estimate for Upper-Bound Release Event

These are discussed in Section 5.1.3.3 and listed in Table 6.6.

6.2.2.4 Other Accidents Considered

A tank leak and equipment failure were considered in Section 5.1.3.3. Some others follow, and are listed in Table 6.6.

Filter Fire. If the high-efficiency particulate air filters are subjected to high temperatures for extended periods of time, the glass fiber melts, resulting in loss of filtration. Whether the material accumulated can be released or is trapped by the shrinking glass fibers is yet to be defined. Loss of filtration without loss of flow would result in the release of the particles airborne in the cell. If no credit is taken for the removal of particles by other components of the off-gas system, all the airborne particles are assumed to be released to the atmosphere. Ion exchange removal is a wet process and is not particularly dusty. It is assumed that the mass airborne concentration is 10 mg/m^3 and the ventilation flow is 20 m^3/min resulting in the release of 0.2 g/min.

TABLE 6.6. Postulated Fractional Airborne Releases from Accidents During Cesium Recovery

Event	Atmospheric Fractional Release
Cesium ion exchange fire	3×10^{-6}(a) fractional release
Filter fire/filter failure	2×10^{-9}(b) 0.2 g/min fractional release
Loss of services or power	No significant release
Rupture of ion exchange column	Insufficient data

(a) DOE (1982).
(b) Rockwell (1980).

Loss of Services or Power. Loss of services or power could result in a variety of consequences depending upon the services and the equipment powered. Most of the consequences envisioned have been covered by the other events described, and a greater airborne release than estimated for these events is not anticipated.

Rupture of the Ion Exchange Column. Loss of containment of the resin and solution would result in the airborne release of some of the liquid by free-fall spill. If the resin is not recovered, it could dry and be available for combustion. Such an event is not likely since the operation would be actively monitored while high-level material is present. Thus it is not anticipated that this event could result in an airborne release greater than those already covered.

6.2.3 Cesium Encapsulation

Information on cesium encapsulation is taken from the Waste Encapsulation and Storage Facility safety analysis report (Braden et al. 1971). Although no description of the process was included in the Hanford Defense Waste Environmental Impact Statement, the process currently used at the facility is assumed to be similar to that described in the safety analysis report.

6.2.3.1 Description of Operation and Facility

The Waste Encapsulation and Storage Facility where the operation takes place is described in detail by Braden et al. (1971).

Cesium encapsulation begins when concentrated cesium carbonate is converted to cesium chloride by batch contact with 12 molar hydrochloric acid. Cesium chloride was selected as the optimal compound for encapsulation for various reasons, among which are high cesium density, compatibility with encapsulating materials, and negligible radiolytic decomposition for the anhydrous material. The cesium chloride solution is heated to boiling in the evaporator-melter, and after the drying step is slowly heated to its melting temperature (approximately 645°C). During the high-temperature drying and melting operation, the atmosphere inside the melter is purged with argon to reduce the corrosion potential. Some cesium chloride is volatilized during this operation, and the melter vent line is equipped with a de-entrainer for collecting this material. The collected material is recycled back to the melter and included in the next batch.

The molten cesium chloride is cast in cylindrical, type 316 L stainless steel, primary containers as shown in Figure 4.1. (Capsule parameters are listed in Table 4.1.) After the castings are cooled, caps are placed on the containers and welded to the containers using an inert atmosphere. Each capsule is leak tested, decontaminated by ultrasonic cleaning, weighed, weld-tested, and calorimetered. Interim storage is in a water-filled basin (described in Section 5.3.1). Equipment used in the process, including the off-gas system (caustic scrubbers, de-entrainer, heater, filters), are described by Braden et al. (1971).

6.2.3.2 Postulated Upper-Bound Accident

As previously discussed, loss of all components of the off-gas and ventilation systems (two separate systems) is not credible unless some events simultaneously occur within the systems with sufficient energy generated to breach the systems. A loss of the system can occur with loss of airflow or continued airflow. A loss with loss of flow results in venting the contained airborne materials slowly by diffusion. In the event of loss of function with continued

flow and melter operation, the release from this operation could be very high due to the volatilization of cesium chloride. Loss of filtration (loss of both stages of the high-efficiency particulate air filters - an unlikely event) without loss of ventilation flow would release the materials already airborne in the cell to the atmosphere, and is the postulated upper-bound accident.

6.2.3.3 Release Estimate for Upper-Bound Event

Cesium encapsulation may be a dusty operation, and a mass airborne concentration of 100 mg/m^3 is applied. Assuming the same ventilation rate as in previous discussions of this situation (20 m^3/min), the mass airborne release to the atmosphere would be 2.0 g/min, as listed in Table 6.7.

6.2.3.4 Other Accidents Considered

Other accidents considered are also listed in Table 6.7.

TABLE 6.7. Postulated Airborne Releases from Accidents During Cesium Encapsulation

Event	Atmospheric Airborne Release
Loss of off-gas exhaust system	2 g/min
Canister drop	8×10^{-6}(a) fractional release
Hydrogen accumulation and explosion	3×10^{-4}(b) fractional release
Spill of molten cesium chloride	Significantly below upper-bound
Fire	Significantly below upper-bound
Loss of services or power	No significant release

(a) Richardson (1980).
(b) DOE (1982).

Spill of Molten Cesium Chloride. It is postulated that due to a process malfunction, molten cesium chloride is released to the cell. As in the discussion of the spill of molten glass, it is believed that the molten material would cool rapidly and solidify. Some airborne release to the cell atmosphere would occur during the cooling but, if the exhaust system is operational, the release to the outside atmosphere would be minor.

Volatilization of Cesium Chloride. Some volatilization normally occurs during operational heating of the cesium chloride to melting. The volatilized material is controlled by the off-gas system. Loss of the off-gas system or the release of molten material to the cell has been discussed in previous sections.

Canister Drop. No scenario was postulated for this event by Richardson (1980); however, he postulated a release. It is assumed that some process malfunction causes the drop of an unencapsulated container full of cesium chloride. The airborne release to the atmosphere listed in the reference is shown in Table 6.7.

Fire. Fire in the in-cell filters was considered by Braden et al. (1971) to be the most serious radiologic hazard to the Waste Encapsulation and Storage Facility. It was postulated that all the accumulated material could be released to the facility filtration system, which is postulated to remain functional. In this operation, the material is volatile at higher temperatures and release is more probable. Since the facility high-efficiency particulate air filters are assumed to remain operational, the airborne release to the environs is estimated to be minor.

Hydrogen Accumulation and Explosion. A detonation of accumulated hydrogen from the radiolysis of water was postulated (Braden et al. 1971) in one of the tanks holding a concentrated radionuclide solution. The facility filters remain operational and the airborne release to the outside atmosphere is minor.

Loss of Services or Power. Various losses of services or power are discussed by Braden et al. (1971); vessel coil failure, loss of cooling water, loss of process or instrument air, loss of electrical power, etc. None of the release consequences from these events is as serious as those covered above.

6.2.4 Grouting Neutralized Current Acid Waste Supernatant Liquids

After removal of cesium, the neutralized current acid waste supernatant liquid would be concentrated by evaporation and converted to grout.

6.2.4.1 Description of Operation and Facility

The neutralized current acid waste residues would be combined with other wastes and processed through the transportable grout facility as described in Section 5.1.5.1.

6.2.4.2 Postulated Upper-Bound Accident

As in Section 5.1.5.2, the upper-bound release is a spray of the feed solution.

6.2.4.3 Release Estimate for Upper-Bound Event

The atmospheric release developed in Section 5.1.5.3 and listed in Table 6.8, is 3×10^{-4} g of salt solution.

TABLE 6.8. Postulated Airborne Releases from Accidents During Neutralized Current Acid Waste Grouting Operations

Event	Atmospheric Airborne Release
Spray of neutralized current acid waste solution	3.0×10^{-4}
Salt solution tank leak	2×10^{-8}(a) fractional release
Mixer tank leak	Significantly below upper-bound
Grout spill	Significantly below upper-bound

(a) Hayward and Jensen (1980).

6.2.4.4 Other Accidents Considered

Additional accidents beyond those listed in Section 5.1.5.4 are not considered.

6.2.5 Fill Tank

Filling the empty tanks will be the same operation used for future waste in Section 5.2.5. The operation and facility, postulated upper-bound accident, release for upper-bound event, and other accidents considered are described there. Releases are listed in Table 6.9.

6.3 STRONTIUM AND CESIUM CAPSULES

Storage of the strontium and cesium capsules in the Waste Encapsulation and Storage Facility continues to allow decay time. After 20 to 40 years the heat generated would be low enough to permit passive cooling of the encapsulated waste.

6.3.1 Capsule Retrieval

This is the same operation described in Section 5.3.1. It includes the description of the operation and facility, postulated upper-bound accident, release estimate for upper-bound event, and other accidents considered. Releases are listed in Table 6.10.

TABLE 6.9. Postulated Airborne Releases from Accidents During Filling of Empty Tanks

Event	Atmospheric Release
Dome collapse	3.5 g
Loss of filtration	
1st stage	5×10^{-6} g/m^3
Both stages	1×10^{-2} g/m^3
Loss of services or power	No significant release
Equipment failure	No significant release

TABLE 6.10. Postulated Airborne Releases from Accidents During Retrieving Capsules

Event	Atmospheric Airborne Release
Capsules rupture	5.5×10^{-6} g
Capsule drop in basin	
Cover drop	3×10^{-10}(a) fractional release
Hydrogen accumulation/ explosion	1×10^{-15}(b) fractional release
Loss of filtration	Significantly below upper-bound
Fire	No significant release
Loss of Services or power	No significant release
Capsule failure in basin	No significant release

(a) DOE (1982).
(b) Richardson (1980).

6.3.2 Capsule Packaging

This is the capsule packaging process described in Section 5.3.2 that includes a description of the operation and facility, postulated upper-bound accident release estimate for upper-bound event, and other accidents considered. Releases are listed in Table 6.11.

6.3.3 Place Capsules in Drywell Storage

Drywell disposal is the proposed canister storage for in-place stabilization.

6.3.3.1 Description of Operation and Facility

Canisters are transported to the drywell disposal area by a shielded-cask transporter (Figure 6.1). The canisters will be lowered into drywells (one per drywell is assumed) by the transporter vehicle that also discharges sand into

TABLE 6.11. Postulated Airborne Releases from Accidents During Capsule Packaging

Event	Atmospheric Airborne Release
Package element failure	6.6×10^{-7} g
Capsule damage during processing	3×10^{-10}(a)
Handling/canister drop	7×10^{-16}(b)
Fire	8×10^{-6}(c)
Loss of off-gas exhaust system	Significantly below upper-bound

(a) DOE (1982).
(b) Hayward and Jensen (1980).
(c) Richardson (1980).

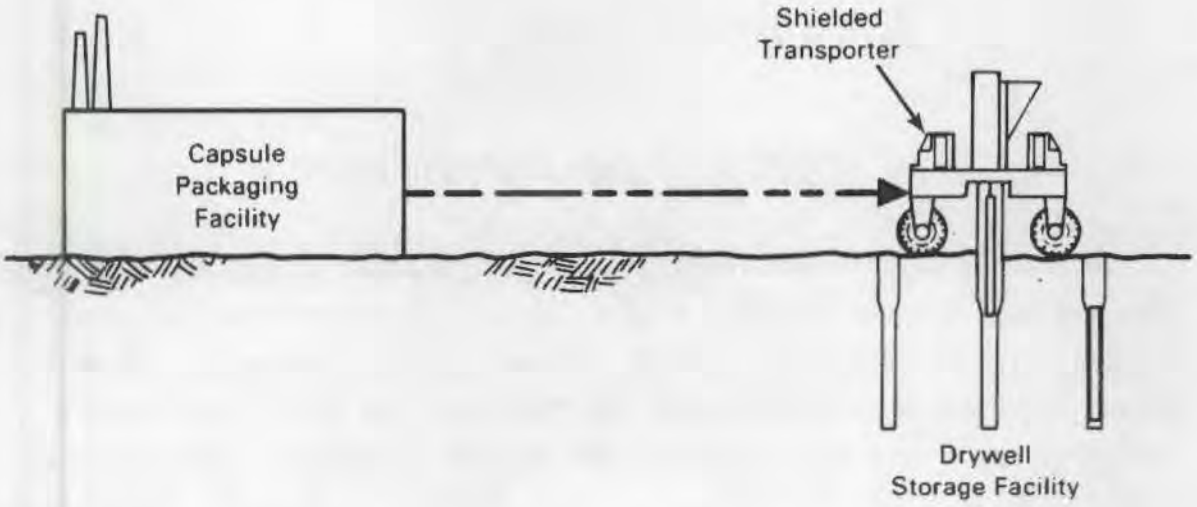


FIGURE 6.1. Transfer of Strontium and Cesium Capsules from the Capsule Packaging Facility to the Drywell Storage Facility

the space above the canister to fill the upper portion of the drywell. A typical drywell assembly is shown in Figure 6.2. Planned detailed engineering studies may show the feasibility of a dry, overpack disposal either in the open air or in a facility.

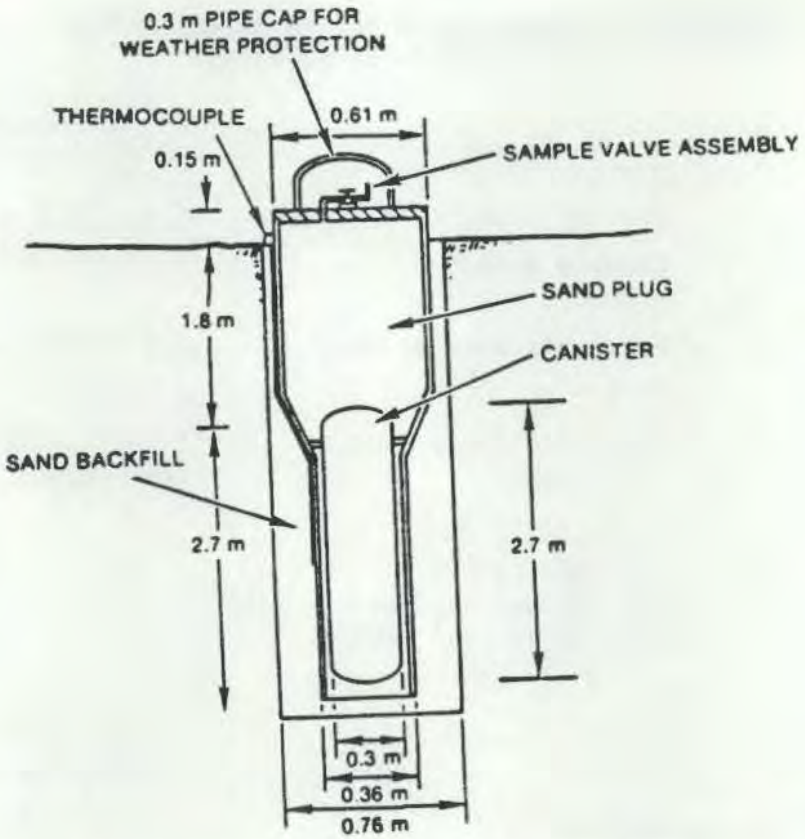


FIGURE 6.2. Typical Drywell Assembly

Given the drywell configuration assumed, a total of 274 drywells will be required. Each drywell consists of a cylindrical carbon steel encasement vessel that extends approximately 0.15 m above and 4.6 m below the ground surface. The encasement vessels are shop fabricated of 0.36- and 0.61-m-dia pipe joined by a standard pipe reducer. The vessel is closed at the bottom by a pipe cap welded onto the 0.36-m-dia lower section. Each drywell encasement vessel is furnished with a carbon steel plate that is field-welded to the top of the vessel after placement of the waste canister and sand. The closure plate is furnished with lifting lugs and a sample valve assembly to obtain air samples and measure pressure inside the drywell. The sample valve is protected by a detachable weather cover, and a nameplate is provided on top of the closure plate for identification. A reusable metal cover is used to protect the empty drywell from the weather before canister placement.

6.3.3.2 Postulated Upper-Bound Accident

It is postulated that the transporter does not correctly align the waste capsule in the drywell and moves with the capsule still partially in the transporter. The transporter shears the capsule and causes subdivision and dispersion of the particles generated. The operation is performed in the open without any enclosure over the drywell, and any material release would be directly to the atmosphere.

6.3.3.3 Release Estimate for Upper-Bound Event

The dispersion value given in DOE (1982) was 1×10^{-6} for a cover block drop, Richardson's (1980) value for a dropped shipping container was 1×10^{-5}. Since the area affected is less than either the container or cover block drop, the smaller value of 1×10^{-6} is applied.

As developed in Section 5.3.1.3, one capsule contains 2.2×10^{3} g, so the release is 2.2×10^{-3}g, as listed in Table 6.12. This release is directly to the atmosphere.

TABLE 6.12. Postulated Airborne Releases from Accidents During Drywell Storage of Waste Capsule

Event	Atmospheric Airborne Release
Improper transporter orientation during canister placement	2.2×10^{-3}g 10^{-6}(a) Fractional release
Package element failure/ canister drop	10^{-5}(b) Fractional release

(a) DOE (1982).
(b) Richardson (1980).

6.3.3.4 Other Accidents Considered

One other accident was considered and listed in Table 6.12.

Package Element Failure/Canister Drop. As stated in DOE (1982), it is not anticipated that a fully encapsulated waste container will be ruptured by routine handling mishaps. The capsules are further overpacked before disposal, even further reducing the potential for failure. It is not anticipated that the waste canisters will fail under these conditions.

6.4 TRU-CONTAMINATED SOIL

Sites would be surveyed to determine radiation and contamination status. Subsidence control is achieved by injecting grout into sites with high potential. Abandoned ponds, trenches, and ditches would be filled before covering if necessary. Trench drains, cribs, settling tanks, and reverse wells could be injected with grout.

6.4.1 Grout Injection

Grouting is the only disposal operation associated with the in-place stabilization and isolation alternative for these sites (Rockwell 1985).

Subsidence control is initiated by completing a geophysical survey of the high subsidence potential liquid waste sites (typically cribs) to identify their location and to identify grout injection points. To stabilize, these sites, a conceptual method is used involving a cementitious grout injected by a specially trained crew. The equipment needed includes mixing tanks, proportioning transfer pumps, hoses, and pneumatic drills. After grout injection, a construction crew is required to trim vent and feed piping with power saws under a tent-type containment. The equipment and personnel are transported by heavy-duty trucks. Following grout injection, the surface barrier can be applied (Rockwell 1985).

6.4.1.2 Postulated Upper-Bound Accident

The upper-bound airborne release event is suggested by Murphy and Holter (1980). This is a void space collapse initiated by equipment. They postulated

that an earthmover was engulfed in a 90 m^3 void space. This disturbed 45 m^3 of waste for an hour and produced airborne material.

6.4.1.3 Release Estimate for Upper-Bound Release Event

In the event postulated by Murphy and Holter (1980), the source was 130 Ci and 4.7×10^{-3} Ci is released; thus, the calculated atmospheric fractional release is 3.6×10^{-5}. At a soil density of 1.8 g/cm^3, the 45 m^3 is 8.1×10^7 g, and the calculated release is 2.9×10^3 g. Only a portion will be in the respirable size range. As discussed in Section 5.4.1.3, Hanford soil can have a 0.088% respirable component. This makes the release of concern 2.6 g, as listed in Table 6.13.

6.4.1.4 Other Accidents Considered

Other accidents considered are also listed in Table 6.13.

Grout Ejection. Grout injection at an excessive rate could possibly result in some minor amounts of waste being ejected through the air exhaust. If an air exit is not provided, pressure buildup could potentially eject some of the waste through the entry chute.

Excavating Contaminated Soil. Disturbing contaminated soil during excavation before marker placement should not be likely since clean material covers the waste to a depth of 5.4 m. Radiation surveys should preclude this accident.

TABLE 6.13. Postulated Airborne Release from Accidents During Grout Injection

Event	Atmospheric Airborne Release
Void space collapse	2.6 g
Grout ejection	Significantly below upper-bound
Digging into waste	No significant release
Fire	No significant release
Reaction with concrete	No significant release
Excavating contaminated soil	No significant release

Digging Into Waste. Digging into waste should not be likely, since little digging will be done. The method of preparing the grout injection has not been detailed, but it would probably be more of a core drilling operation than digging. The radiation surveys should preclude this accident.

Fire. Fire would be a limited event since there is a low level of combustible material in the cribs. The wood-constructed cribs could possibly burn, but no significant releases are postulated.

Reaction with Concrete. Thermal or reactive interaction of waste with the concrete does not seem likely. Small amounts of contained organics might have been inadvertently included when the waste was interred, but should not be in amounts large enough to be of concern.

6.5 PRE-1970 SOLID TRU WASTE

Pre-1970 TRU solid waste burial grounds would be stabilized as required.

6.5.1 Grout Caissons

Caissons containing TRU waste would be immobilized in place by being filled with grout or other stable fillers. The area would then be covered and marked.

6.5.1.1 Description of Operation

A conceptual method of injecting cementitious grout into caissons has been developed (Rockwell 1985). This is the same method Rockwell suggested for TRU-contaminated soil sites, described here in Section 6.4.1.

6.5.1.2 Postulated Upper-Bound Accident

For caissons, void space collapse does not seem to apply, since the waste is confined to the caisson. The postulated upper-bound event is equipment malfunction allowing grout injection at an excessive rate. This could result in an eruption of contaminated material through the air inlet. The waste could be from a package broken during placement. Surface contamination could contribute to the release. Some material is assumed to be ejected into the air, and the

potential for resuspension could also be considered. Since the caissons are located 4 m below grade, it seems unlikely that a large quantity of waste could be ejected.

6.5.1.3 Release Estimate for Upper-Bound Event

Waste should be packaged in the caisson. For this release estimate, it is postulated that the waste is stored in 3.82-L metal cans. It is assumed that the contents of one can are involved in the accident. The fractional release is assumed to be similar to that from a core drilling accident described by Murphy and Holter (1980), 7×10^{-8}. At a waste density of 0.96 g/cm^3, the source is 3.7×10^3 g and the release would be about 3×10^{-4} g as listed in Table 6.14. It would appear that only airborne material reaches the surface, so that there is no source for resuspension releases.

6.5.1.4 Other Accidents Considered

These accidents are also listed in Table 6.14.

Void Space Collapse. A void space collapse would not seem to apply to this situation, since the waste is confined to the caisson.

Spread of Surface Contamination. Surface contamination spread could only result in minor releases, which could occur if hoses are introduced into a contaminated chute and then withdrawn. Details of the operation not currently available could suggest whether this is a real possibility.

TABLE 6.14. Postulated Airborne Releases from Accidents During Grout Injection of Caissons

Event	Airborne Release
Injection at excessive rate	3×10^{-4} g
Void space collapse	No significant release
Spread of surface contamination	No significant release
Fires	

Fire. Fire such as a range fire is possible, but the depth of burial should preclude the possibility of releases from this event.

6.5.2 Subsidence Control

A pile-driving method is currently being developed for subsidence control in areas with high potential for subsidence (Rockwell 1985).

6.5.2.1 Description of Operation

Piles are injected into identified waste zones using a diesel-powered vibratory hammer/extractor attached to a vibratory crane. The 5-m-long piles are driven at 2.5-m centers within identified areas. Piles are dynamically driven through the waste zone and then withdrawn.

If the piles are found to have smearable surface contamination when withdrawn, they are redriven to grade. If no smearable contamination is detected, the piles are withdrawn and reused.

6.5.2.2 Postulated Upper-Bound Accident

The postulated upper-bound release event is a void space collapse described in Section 6.4.1.2 with the same estimated release, 2.6 g. The release is listed in Table 6.15.

6.5.2.3 Other Accidents Considered

No significant releases were postulated for the events listed in Table 6.15.

TABLE 6.15. Postulated Airborne Releases from Accidents During Subsidence Control

Event	Airborne Release
Void space collapse	2.6 g
Penetration of waste	No significant release
Fire	No significant release
Excavating contaminated soil	No significant release
Criticality	Not postulated

Penetration of Waste. Penetration of waste during rod emplacement could lead to releases when the rod is withdrawn. The diameter of the rods would be fairly small, so no large amount of waste would be available as a release source. When the waste is penetrated it is probably displaced and immobilized, and only small amounts clinging to the rod would reach the surface.

Fire. Fire could be considered, for example, if a range fire engulfed the diesel-operated vibratory hammer. The fire would be very limited because of low combustible loading, and should not result in releases of airborne contamination.

Excavating Contaminated Soil. The only soil excavation is done during marker placement. This is not a likely source of radioactive release since clean material is covering the waste to a depth of 5.4 m.

Criticality. Subsidence control is an operation to examine for criticality potential. While administrative controls should have prevented criticality during interment, this operation will change the fissile geometry within the soil matrix since the vibratory hammer will compress soil and waste. This type of system (soil, waste) does not tend to produce criticalities; they generally occur in solution systems with material accumulation in a tank and a surge to unsafe geometry. Nevertheless, Clayton (1974) suggests that criticality can occur in soil with concentrations as low as 2 g/L in relatively dry soil. He also suggests soil as having a 30% void volume; thus if soil with 2 g/1.4 L is compressed to 2 g/L it is a geometry that could conceivably go critical. Soils surrounding the packaged waste could be contaminated from leakage or ruptured packages. It is unlikely that there would be sufficient fissile material available to sustain a criticality. Radiation surveys should preclude criticality accidents.

6.6 RETRIEVABLY STORED AND NEWLY GENERATED TRU SOLID WASTE

Any TRU solid waste packages stored in above-grade facilities would be buried. All retrievably stored TRU solid waste would be treated the same as pre-1970 TRU solid waste. Thus the only releases not covered in Section 6.5

are from the burial operations. The subsidence control releases were included in Table 6.15, with the upper-bound release identified as the void space collapse.

6.6.1 Waste Burial

Burial operations are performed routinely at Hanford, thus are not detailed in either the Hanford Defense Waste Environmental Impact Statement or the support document for retrievably stored and newly generated transuranic solid waste (Rockwell 1985).

6.6.1.1 Description of Operation and Facility

Routine disposal of non-TRU waste by burial in trenches at Hanford is described by Geiger, Brown and Isaacson (1977). It is assumed that the process used here will be similar. Therefore, the waste would be placed in a wide-top, relatively narrow-bottom trench. All radioactive waste is covered at the end of the day. No enclosure is identified for the operation.

6.6.1.2 Postulated Upper-Bound Accident

The accident occurring when the waste container is in the open air rather than covered with dirt would have the largest release potential. A breach of a waste container, spilling the contents in the ambient air, is the postulated upper-bound accident.

6.6.1.3 Release Estimate for Upper-Bound Event

Based on the experimental work of Sutter (1980), 5% of the spilled waste could be entrained if the ambient winds are at the Hanford average 7.6 mph (Stone et al. 1972). The waste will be in 210-L drums, and at a waste density of 0.96 g/cm has a total mass of 2×10^5 g. The total is 1×10^3 g released directly to the atmosphere.

6.6.1.4 Other Accidents Considered

The other accidents considered before selecting the upper-bound release are discussed below and listed in Table 6.16.

Puncture. A puncture-type penetration of the waste container would have a lower release than would a complete breach of the container.

TABLE 6.16. Postulated Airborne Releases from Accidents During Subsidence Control

Event	Atmospheric Airborne Release
Container Breach	1×10^3 g
Puncture	Significantly below upper-bound
Equipment Impact	Significantly below upper-bound
Fire	No significant release

Equipment Impact. Equipment impacting the waste, if it generated sufficient energy to break a package, would not result in a release above the level estimated above for the upper-bound event.

Fire. Fire would be unlikely since the waste is packaged so that air is available. During this operation there is primarily dirt on the surface, and no combustibles are available, so a range fire is unlikely.

7.0 POSTULATED RELEASES FROM OPERATIONAL ACCIDENTS ASSOCIATED WITH REFERENCE DISPOSAL OF SIX WASTE FORMS

The reference alternative provides a balanced disposal approach to the waste disposal. This alternative should limit long-term (over 10,000 yr) population risk without incurring near-term risks by disturbing wastes that are now stable.

To some extent, the reference alternative would combine geologic disposal and in-place stabilization and disposal. It would be the same as the geologic disposal alternative for strontium and cesium capsules and portions of retrievably stored and newly generated TRU solid waste. It would be the same as the in-place stabilization and disposal alternative for waste in single-shell tanks, TRU-contaminated soil sites, and pre-1970 TRU buried solid waste sites. Table 7.1 lists the waste forms and indicates disposal alternative techniques that are applicable to each. Operational accidents will not be discussed again here, so the third column lists the report section in which they are developed. All upper-bound operational releases are listed in Table 7.2 for quick reference.

Thus there are three waste classes or portions thereof that could involve some variation of operational steps as listed below.

- Double-shell tank waste: grouted liquid waste could be returned to empty tanks
- Future tank waste: cesium removal only
- Retrievable remote-handled TRU: packaged for geologic disposal.

These wastes would be processed in facilities sized for the need of the reference alternative, rather than in the more extensive and much larger facilities required for the geologic disposal alternative.

7.1 DOUBLE-SHELL TANK WASTE

Operations for the double-shell tank wastes are essentially the same as those described in Section 5.1. Steps are hydraulic retrieval, sludge washing, high-level and TRU separation (without technetium or strontium recovery),

TABLE 7.1. Reference Disposal Techniques for Six Waste Forms

Waste Form	Applicable Alternative Disposal Method	Report Section Developing Accidents
Existing tank waste		
Single-shell	In-place stabilization and disposal	4.1
Double-shell	Geologic, part repository and part onsite	3.1
Future tank waste	New	
Strontium and cesium capsules	Geologic	3.3
TRU-contaminated soil	In-place stabilization and disposal	4.4
Pre-1970 solid waste	In-place stabilization and disposal	4.5
Retrievably stored and newly generated TRU		
Contact-Handled TRU	Geologic	3.6
Remote-Handled TRU	New	

vitrification, and repository disposal. Some operations will be conducted in a small vitrification facility, but will be the same as geologic disposal. For example, vitrification will use a joule-heated melter. Accidents and releases will be the same as Section 5.1, and are listed in Table 7.2.

7.2 FUTURE TANK WASTE

The reference alternative for future tank waste involves geologic disposal of high-level waste. Cesium only would be removed from neutralized current acid waste before grouting. Strontium and TRU elements from this waste would be contained primarily in the sludge. Cesium and the sludge would be vitrified. Accidental releases from all of these operations were developed in Section 5.1 and are listed in Table 7.2.

TABLE 7.2. Reference Alternative Potential Accidental Releases for Operations Involving Six Waste Forms

	Waste Form	Accident Source Quantity	Facility	Accident	Location	Fractional Release	Transmission Factor[a]	Atmospheric Release of Concern, g[b]	Release Point
Existing Tank Waste									
Single-shell									
1. Dry	Salt cake	2000 m^3	Tank	Explosion	Tank	NA[c]	1	1.3×10^4	Ground level
2. Fill Tank Dome	Salt cake		Tank	Dome collapse	Tank		1	7×10^3	Ground level
Double-shell									
1. Hydraulic Retrieval	Liquid waste	9×10^4 g/min	Tank	Pressurized spray	Diversion valve	5×10^{-2}	1	4.5×10^3	Ground level
2. Sludge washing	Sludge	0.2 g/min	Small vitrification	Filter failure	Filters	NA	1	0.2	Stack
3. Radionuclide removal	Ion exchange resin		Small vitrification	Ion exchange fire	Resin column	0.01	3×10^{-4}		Stack
4. Vitrification	Molten glass	0.2 g/min	Small vitrification	Loss of filters	Filters	NA	1	0.2	Stack
5. Grout solutions	Decontaminated solutions	10 m^3	Grout	Liquid spray from lines	Transfer line	1×10^{-4}	2.5×10^{-7}	3×10^{-4}	Stack
6. Fill empty tank	Residual tank	0.05% of original fill	None	Dome collapse	Tank	5×10^{-4}	1	3.5	Ground level
Future Tank Waste									
1. Hydraulic retrieval	NCAW[d]	9×10^4 g/min	NA	Pressurized release	Diversion valve	5×10^{-2}	1	4.3×10^3	Ground level
2. Cesium removal	Ion exchange resin		Small vitrification	Ion exchange fire	Resin column	0.01	3×10^{-4}		Stack
3. Grout solutions	NCAW supernatant	10 m^3	Transportable grout	Liquid spray from line	Transfer line	1×10^{-4}	2.6×10^{-7}	3×10^{-4}	Stack
4. Vitrification	Molten glass	0.2 g/min	Small vitrification	Loss of filters	Filters		1	0.2	Stack
5. Fill empty tank	Residual tank waste	0.05% of original fill	None	Dome collapse	Tank	5×10^{-4}	1	3.5	Ground level
Strontium and Cesium Capsule									
1. Remove from WESF[e]	Encapsulated waste	2.2×10^3 g	WESF	Capsule rupture	Storage area	1×10^{-2}	2.5×10^{-7}	5.5×10^{-6}	Stack
2. Capsule packaging	Encapsulated waste	2.2×10^3 g	CPF[f]	Machinery impacts capsule	"Load" station	1.2×10^{-3}	2.5×10^{-7}	6.6×10^{-7}	Stack

TABLE 7.2. (contd)

	Waste Form	Accident Source Quantity	Facility	Accident	Location	Fractional Release	Transmission Factor[a]	Atmospheric Release of Concern, g[b]	Release Point
TRU-Contaminated Soil									
1. Inject grout	TRU-contaminated soil	8.1×10^7 g	None	Void space collapse	CRIB	3×10^{-6}	1	2.6	Ground level
Pre-1970 Solid Waste									
1. Grout caissons	Caisson	3.7×10^3 g	None	Equipment	Caisson failure	7×10^{-1}	1	2×10^{-4}	Ground level
2. Subsidence control	Solid waste site	8.1×10^7	None	Void space collapse	Burial site	3×10^{-6}	1	2.6	Ground level
Retrievably Stored and Newly Generated TRU									
1. Retrieve									
1.1 RH[g] TRU	Packaged waste	3.7×10^3	Caisson retrieval	Pressurized release	Metal can	0.01	2.5×10^{-7}	9.3×10^{-6}	Stack
1.2 CH[h] TRU	Packaged waste	2×10^5	None	Pressurized release	Drum	0.01	1	2.5×10^3	Ground level
2. Sorting and related operations									
2.1 RH TRU	Packaged waste	2×10^5	RH WRAP[i]	Pressurized release	Drum	0.01	2.5×10^{-7}	5×10^{-4}	Stack
2.2 CH TRU	Packaged waste	2×10^5	CH WRAP	Pressurized release	Drum	0.01	2.5×10^{-7}	5×10^{-4}	Stack
3. Process									
3.1 RH TRU	No information available.								
3.2 CH WRAP	Packaged waste	1×10^5 g/drum	CH WRAP	Fire	Drum	5×10^{-4}	2.5×10^{-7}	2.5×10^{-5}[j]	Stack

(a) A transmission factor of 1 means the high-efficiency particulate air filters are breached or the operation is not enclosed.
(b) Default assumption - if release is in g/min, a 1 minute release is assumed.
(c) Not Applicable.
(d) Neutralized Current Acid Waste.
(e) Waste Encapsulation and Storage Facility.
(f) Capsule Packaging Facility.
(g) Remote-handled.
(h) Contact-handled.
(i) Waste Retrieval and Packaging Facility.
(j) Two drums involved in fire.

7.3 RETRIEVABLY STORED AND NEWLY GENERATED TRU

For the reference alternative, retrievably stored and newly generated TRU would be sent to a geologic repository. The waste would be processed in the same way as in the geologic disposal alternative, except for the remote-handled TRU waste. Since only remote-handled TRU waste would be processed, a smaller facility would be used. The waste processing facility proposed for the geologic disposal alternative was sized to accommodate TRU-contaminated soil sites and pre-1970 TRU solid waste burial grounds.

The remote-handled TRU waste would be processed in a facility that provides remote handling, and contains hot cells for size reduction, immobilization, and packaging. A remote-handled waste retrieval and packaging facility would include specific processes required to immobilize and package the waste (Rockwell 1985). However, the immobilization process is not identified, so no releases can be developed. Table 7.2 lists the postulated releases from the other operations.

8.0 POSTULATED RELEASES FROM OPERATIONAL ACCIDENTS ASSOCIATED WITH THE NO DISPOSAL ACTION ALTERNATIVE

Under the no disposal action, the wastes are placed in continued storage; this alternative does not implement a long-term solution for permanent disposal of the radioactive wastes. The wastes would continue to be stored essentially as they are now for the indefinite future. The waste-handling operations would include storage, necessary remedial actions, and waste surveillance. The accidents postulated for this alternative process are shown in Table 8.1. They involve the double-shell wastes (both existing and future tank wastes), the strontium and cesium capsules, and all TRU wastes. With the exception of the existing tank wastes and the retrievably stored and newly generated TRU, upper-bound accidents have been described in Sections 5.0, 6.0, and 7.0. For the existing tank wastes, the ferrocyanide explosion is no longer postulated to occur, as the single-shell tank wastes are left undisturbed. The dominant release for this waste class then becomes the pressurized release during hydraulic transfer of the existing double-shell tank wastes from tank to tank as required to assure tank integrity. For the retrievably stored and newly generated TRU, the collapse of a void space, similar to that described for the other TRU sites, is postulated to occur during subsidence control.

TABLE 8.1. No Disposal Action Alternative Accidental Releases for Operations Involving Six Waste Forms

	Waste Form	Accident Source Quantity	Facility	Accident	Location	Fractional Release	Transmission Factor[a]	Atmospheric Release of Concern, g	Release Point
Existing Tank Waste									
1. Hydraulic retrieval of residual liquid (double shell)	Liquid waste	9×10^4 g/min	Tank	Pressurized spray	Diversion valve	5×10^{-2}	1	4.5×10^3	Ground level
2. Fill empty tank dome	Residual tank waste	5% of original fill	None	Dome collapse	Tank	5×10^{-2}	1	350	Ground level
Future Tank Waste									
1. Hydraulic retrieval	Liquid waste	9×10^4 g/min	Tank	Pressurized spray	Diversion valve	5×10^{-2}	1	4.5×10^3	Ground level
2. Fill tank	Residual tank waste	0.05% of original fill	None	Dome collapse	Tank	5×10^{-4}	1	3.5	Ground level
Strontium and Cesium Capsules									
1. Remove from WSEF	Encapsulated waste	2.2×10^3 g	WSEF	Capsule rupture	Storage area	1×10^{-2}	2.5×10^{-7}	5.5×10^{-6}	Stack
2. Capsule packaging	Encapsulated waste	2.2×10^3 g	CPF	Machinery impacts capsules	"Load" section	1.2×10^{-3}	2.5×10^{-7}	6.6×10^{-7}	Stack
TRU-Contaminated Soil									
1. Subsidence	TRU-contaminated soil	8.1×10^7 g	None	Void space collapse	CRIB	3×10^{-6}	1	2.6	Ground level
Pre-1970 TRU Solid Waste									
1. Subsidence	Solid waste site	8.1×10^7 g	None	Void space collapse	Burial site	3×10^{-6}	1	2.6	Ground level
Retrievably Stored and Newly Generated TRU									
1. Subsidence	Retrievable stored TRU	8.1×10^7 g	None	Void space collapse	Burial site	3×10^{-6}	1	2.6	Ground level

(a) A transmission factor of 1 means the HEPA filters are breached or not enclosed.

9.0 REFERENCE RADIONUCLIDE INVENTORIES

Reference radionuclide inventories used to conservatively estimate potential radiation doses to the public from operational accidents during proposed waste disposal operations are covered in this section. These reference inventories represent the upper bound or maximum radionuclide inventory or concentration anticipated during these operations. The inventories are calculated from the data supplied by Rockwell either in their engineering data packages (Rockwell 1985) or by subsequent memoranda (Rockwell 1985b) or conversations with cognizant technical staff. Flowsheets are currently unavailable to adequately define the modification of the inventories during the various operations and, therefore, no credit is taken for the dilution or concentration of waste until it is processed into its final disposal form.

9.1 EXISTING TANK WASTE INVENTORIES

Existing tank waste is presently stored as salt cake, sludge, interstitial liquors (double-shell tank slurries) and complex concentrates. Existing tank waste in 149 older single-shell tanks and 20 new double-shell tanks contain the bulk of the radioactive waste (1.8×10^5 m^3) stored on the Hanford site (Rockwell 1985).

9.1.1 Tank or Process Waste Inventories

To generate a reference radionuclide inventory that would represent near upper-bound concentration levels of radionuclides for existing tank waste, the C-105 tank with the highest measured TRU concentration levels was selected as the reference waste. The C-105 tank radionuclide concentrations are shown in Table 9.1. This reference inventory is used to estimate the potential radiological impacts on the public from accidents postulated to occur during the handling, packaging and processing operations for existing tank waste.

9.1.2 Grout and Glass Inventories

Average concentrations of radionuclides in grout and glass for existing tank waste are shown in Table 9.2 for the three disposal alternatives. These data are taken from a Rockwell engineering package (Rockwell 1985) and

TABLE 9.1. Estimated Upper Bound Tank Inventory of Radionuclides for Existing Waste (Decayed to the end of 1990)

Radionuclide	Ci	Ci/kg[a]
^{241}Am	1×10^{4}	1×10^{-2}
^{243}Am	1×10^{1}	1×10^{-5}
^{14}C	6×10^{2}	7×10^{-4}
^{244}Cm	6×10^{1}	7×10^{-5}
^{135}Cs	8×10^{-1}	1×10^{-6}
^{137}Cs	2×10^{5}	2×10^{-1}
^{129}I	2	2×10^{-6}
^{63}Ni	4×10^{4}	5×10^{-2}
^{237}Np	6×10^{-2}	7×10^{-8}
^{238}Pu	5×10^{1}	6×10^{-5}
^{239}Pu	2×10^{3}	2×10^{-3}
^{240}Pu	6×10^{2}	7×10^{-4}
^{241}Pu	5×10^{3}	6×10^{-3}
^{226}Ra	9×10^{-9}	1×10^{-14}
^{106}Ru	5	6×10^{-6}
^{151}Sm	1×10^{5}	1×10^{-1}
^{126}Sn	9×10^{1}	1×10^{-4}
^{90}Sr	3×10^{6}	4
^{99}Tc	1×10^{3}	1×10^{-3}
^{230}Th	1×10^{-6}	1×10^{-12}
^{233}U	3×10^{-4}	4×10^{-10}
^{234}U	7×10^{-3}	8×10^{-9}
^{235}U	1	1×10^{-6}
^{238}U	2×10^{1}	2×10^{-5}
^{93}Zr	5×10^{2}	6×10^{-4}

(a) Based on C-105 tank inventory (Rockwell 1985).
150,000 gal (sp gr 1.47)
835,000 kg waste.

TABLE 9.2. Average Concentration of Radionuclides in Grout and Glass from Existing Tank Waste

Radionuclide	Onsite Stabilization and Isolation		Geologic Disposal				Reference Alternative			
	Grout		Grout		Glass		Grout		Glass	
	Ci/m^3[a]	Ci/kg[b]	Ci/m^3[a]	Ci/kg[b]	Ci/Canister[a]	Ci/kg[c]	Ci/m^3[a]	Ci/kg[b]	Ci/Canister[a]	Ci/kg[c]
^{241}Am	2×10^{-1}	1×10^{-4}	3×10^{-4}	2×10^{-7}	2	1×10^{-3}	2×10^{-2}	1×10^{-5}	6×10^{1}	3×10^{-2}
^{243}Am	2×10^{-4}	1×10^{-7}	2×10^{-7}	1×10^{-10}	2×10^{-3}	1×10^{-6}	2×10^{-5}	1×10^{-8}	6×10^{-2}	3×10^{-5}
^{14}C	2×10^{-2}	7×10^{-3}	3×10^{-2}	2×10^{-5}	---	---	2×10^{-2}	4×10^{-5}	---	---
^{244}Cm	1×10^{-3}	7×10^{-7}	1×10^{-6}	7×10^{-10}	1×10^{-2}	6×10^{-6}	1×10^{-4}	7×10^{-8}	4×10^{-1}	2×10^{-4}
^{135}Cs	6×10^{-4}	4×10^{-7}	1×10^{-6}	7×10^{-10}	5×10^{-3}	3×10^{-6}	6×10^{-4}	4×10^{-7}	---	---
^{137}Cs	1×10^{2}	7×10^{-2}	3×10^{-1}	2×10^{-4}	1×10^{3}	6×10^{-1}	1×10^{2}	7×10^{-2}	---	---
^{63}Ni	2×10^{-1}	1×10^{-4}	3×10^{-3}	2×10^{-6}	$2 \times 10^{+1}$	1×10^{-2}	2×10^{-2}	1×10^{-5}	$8 \times 10^{+1}$	5×10^{-2}
^{129}I	2×10^{-4}	1×10^{-7}	7×10^{-5}	5×10^{-8}	---	---	2×10^{-4}	1×10^{-7}	---	---
^{237}Np	3×10^{-4}	2×10^{-7}	4×10^{-7}	2×10^{-10}	3×10^{-3}	2×10^{-6}	3×10^{-5}	2×10^{-8}	1×10^{-1}	6×10^{-5}
^{238}Pu	6×10^{-5}	4×10^{-8}	3×10^{-6}	2×10^{-9}	3×10^{-2}	2×10^{-5}	6×10^{-6}	4×10^{-9}	2×10^{-2}	1×10^{-4}
^{239}Pu	5×10^{-4}	4×10^{-7}	1×10^{-4}	7×10^{-8}	1	6×10^{-4}	5×10^{-5}	4×10^{-8}	2×10^{-1}	1×10^{-4}
^{240}Pu	1×10^{-4}	7×10^{-8}	3×10^{-5}	2×10^{-8}	3×10^{-1}	2×10^{-4}	1×10^{-5}	7×10^{-9}	4×10^{-2}	2×10^{-5}
^{241}Pu	3×10^{-3}	2×10^{-6}	4×10^{-4}	3×10^{-7}	3	2×10^{-3}	3×10^{-4}	2×10^{-7}	1	6×10^{-4}
^{226}Ra	1×10^{-13}	7×10^{-17}	2×10^{-15}	1×10^{-18}	2×10^{-11}	1×10^{-14}	1×10^{-14}	7×10^{-18}	4×10^{-11}	2×10^{-14}
^{106}Ru	6×10^{-5}	4×10^{-8}	3×10^{-5}	2×10^{-8}	1×10^{-3}	6×10^{-7}	3×10^{-5}	2×10^{-8}	1×10^{-2}	6×10^{-6}
^{151}Sm	2	1×10^{-3}	6×10^{-3}	4×10^{-6}	5×10^{-1}	3×10^{-4}	2	1×10^{-3}	---	---
^{126}Sn	2×10^{-3}	1×10^{-6}	5×10^{-6}	4×10^{-9}	4×10^{-2}	2×10^{-5}	2×10^{-3}	1×10^{-6}	---	---
^{90}Sr	1×10^{2}	7×10^{-2}	4×10^{-1}	3×10^{-4}	3×10^{3}	2	1×10^{1}	7×10^{-3}	4×10^{4}	2×10^{1}
^{99}Tc	2×10^{-1}	1×10^{-4}	4×10^{-4}	3×10^{-7}	2	1×10^{-3}	2×10^{-1}	1×10^{-4}	---	---
^{230}Th	6×10^{-12}	4×10^{-15}	4×10^{-13}	3×10^{-16}	3×10^{-9}	2×10^{-12}	6×10^{-13}	4×10^{-16}	2×10^{-9}	1×10^{-9}
^{233}U	3×10^{-9}	2×10^{-12}	6×10^{-11}	4×10^{-14}	5×10^{-7}	3×10^{-10}	3×10^{-10}	2×10^{-13}	1×10^{-6}	6×10^{-10}
^{234}U	2×10^{-8}	1×10^{-11}	2×10^{-9}	1×10^{-12}	2×10^{-5}	6×10^{-8}	2×10^{-9}	1×10^{-12}	8×10^{-6}	5×10^{-9}
^{235}U	1×10^{-5}	7×10^{-9}	7×10^{-7}	5×10^{-10}	5×10^{-3}	3×10^{-6}	1×10^{-6}	7×10^{-10}	4×10^{-3}	2×10^{-6}
^{238}U	2×10^{-4}	1×10^{-7}	2×10^{-5}	1×10^{-8}	2×10^{-1}	1×10^{-4}	2×10^{-5}	1×10^{-8}	8×10^{-2}	5×10^{-5}
^{93}Zr	1×10^{-3}	7×10^{-7}	3×10^{-5}	2×10^{-8}	2×10^{-1}	1×10^{-4}	1×10^{-3}	7×10^{-7}	4×10^{-1}	---

(a) Ci/m^3 listed in the data package (Rockwell 1985).
(b) Calculated using a grout density of 1.4×10^3 kg/m^3.
(c) Calculated using a glass density of 2.8×10^3 kg/m^3, canister volume 0.62 m^3.

converted to Ci/kg of grout or glass. Grout densities can range in values from 1.27 to 1.43 g/cm^3 (McDaniel and Moore 1981, Table III). A value of 1.4 g/cm^3 (1.4 kg/m^3) was selected to calculate grout concentrations. Mendel et al. (1981, p. 2.2) compiled a list of glass waste densities. Values for borosilicate glass ranged from 2.6 to 3.4 g/cm^3. We calculated glass concentration using a 2.8 g/cm^3 (2.8 kg/m^3) value. Average concentration values are used for grout and glass since the waste is processed and blended prior to grouting or vitrification operations.

9.2 FUTURE TANK WASTE - REFERENCE RADIONUCLIDE INVENTORIES

This section describes the radionuclide composition of new waste from the processing of N-Reactor fuel in the PUREX facility and the planned operation of the Plutonium Finishing Plant. Estimates of the radionuclide concentrations in high-level waste from N-Reactor produced fuels from FY-1972 through FY-1990 for fission products, actinides, and activation products are shown in Table 9.3a. These inventories are accumulated to the end of FY-1990, then decayed annually through 1995. Table 9.3b lists cladding removal waste values. For dose calculations the larger inventory N-Reactor or cladding removal waste value is used. The concentrations reported represent the waste neutralized and accumulated before a vitrification facility is available in FY-1991 for the reference option. Plutonium finishing plant wastes, processed separately, have TRU concentrations lower by a factor of 2 to 3 than those shown for the PUREX wastes and use of the PUREX concentrations in the calculations of the radiological impacts will generate bounding values. Thus, plutonium finishing plant waste concentrations are not listed.

The average concentrations of radionuclides in grout and glass for new tanked waste is shown in Table 9.4 for the three disposal alternatives. These data are taken from Rockwell (1985) and converted to Ci/kg of grout or glass. Average concentrations values are used for grout or glass since the waste is processed and blended prior to grouting or vitrification.

TABLE 9.3a. Radioisotopes Accumulated in PUREX High Level Waste from N-Reactor Production FY-1972 to FY-1990 and Decayed through FY-1995

Fission Products			Actinides			Activation Products		
Radionuclide	Ci[a]	Ci/kg[b]	Radionuclide	Ci[a]	Ci/kg[b]	Radionuclide	Ci[a]	Ci/kg[b]
^{144}Ce	1.29×10^{8}	2×10^{-1}	^{241}Am	1.99×10^{5}	2×10^{-2}	^{14}C	1.9×10^{2}	2×10^{-5}
^{134}Cs	4.97×10^{5}	4×10^{-2}	^{239}Pu	2.07×10^{3}	2×10^{-4}	^{60}Co	2.58×10^{5}	2×10^{-2}
^{137}Cs	4.44×10^{7}	7	^{240}Pu	4.41×10^{2}	4×10^{-5}	^{55}Fe	2.42×10^{4}	2×10^{-3}
^{154}Eu	1.64×10^{5}	1×10^{-2}	^{241}Pu	1.92×10^{4}	2×10^{-3}	^{54}Mn	2.3×10^{3}	2×10^{-4}
^{155}Eu	8.94×10^{4}	7×10^{-3}	^{235}U	2.06×10^{-1}	2×10^{-8}	^{95m}Nb	3.6×10^{-3}	3×10^{-10}
^{3}H	7.2×10^{3}	6×10^{-4}	^{238}U	4.03	3×10^{-7}	^{59}Ni	4.3×10^{1}	4×10^{-6}
^{129}I	1.21×10^{1}	1×10^{-6}				^{63}Ni	3.9×10^{3}	3×10^{-4}
$^{106}RuRh$	2.5×10^{7}	2				^{125}Sb	4.6×10^{1}	4×10^{-6}
^{125}Sb	7.14×10^{5}	6×10^{-2}				^{113}Sn	4.75	4×10^{-7}
^{151}Sm	3.94×10^{5}	3×10^{-2}				^{119m}Sn	9.07×10^{1}	7×10^{-6}
^{126}Sn	2.2×10^{1}	2×10^{-6}				^{121m}Sn	6.76×10^{-2}	6×10^{-9}
^{90}Sr	4.08×10^{7}	3.5				^{93}Zr	7.65×10^{-7}	6×10^{-14}
^{99}Tc	4.7×10^{3}	4×10^{-4}				^{95}Zr	3.4×10^{-3}	3×10^{-10}
$^{93}ZrNb$	1.9×10^{3}	2×10^{-4}						

Quantity of Fuel Processed: 12,144 MTU
Note: Combined activities are listed for isotope pairs. No contingency included.

(a) Ci listed in Rockwell (1985).
(b) Volume 800 L/MTU; sp gr 1.25 at 20°C information obtained from Bob Watrous, Rockwell, February 6, 1984

TABLE 9.3b. Radioisotopes Accumulated by FY 1995 in Cladding Removal Waste from N-Reactor Production, FY 1972 through FY 1995

Fission Products			Actinides			Activation Products		
Radionuclide	Ci[a]	Ci/kg[b]	Radionuclide	Ci[a]	Ci/kg[b]	Radionuclide	Ci[a]	Ci/kg[b]
^{144}Ce	1.29×10^{5}	2×10^{-2}	^{241}Am	3.76×10^{3}	3×10^{-4}	^{14}C	9.0×10^{1}	6×10^{-6}
^{134}Cs	1.49×10^{4}	1×10^{-3}	^{239}Pu	2.69×10		^{60}Co	6.13×10^{1}	5×10^{-6}
^{137}Cs	1.33×10^{6}	2×10^{-1}	^{240}Pu	2.69×10^{2}	2×10^{-5}	^{55}Fe	4.90×10^{4}	4×10^{-3}
^{154}Eu	1.64×10^{2}	1×10^{-5}	^{241}Pu	2.69×10^{2}		^{54}Mn	6.89×10^{2}	6×10^{-5}
^{155}Eu	8.94×10^{1}	7×10^{-6}	^{235}U	2.06×10^{-1}	2×10^{-8}	^{95m}Nb	1.03×10^{5}	8×10^{-3}
^{106}Ru	7.50×10^{5}	6×10^{-2}	^{238}U	4.03	3×10^{-7}	^{59}Ni	2.08×10^{1}	2×10^{-6}
^{124}Sb	2.14×10^{4}	2×10^{-3}				^{63}Ni	1.83×10^{3}	2×10^{-4}
^{151}Sm	3.94×10^{2}	3×10^{-4}				^{125}Sb	6.00×10^{4}	5×10^{-3}
^{126}Sn	6.60×10^{-1}	5×10^{-8}				^{113}Sn	6.07×10^{3}	5×10^{-4}
^{90}Sr	4.08×10^{4}	7×10^{-3}				^{119m}Sn	1.16×10^{5}	1×10^{-2}
^{99}Tc	1.41×10^{2}	1×10^{-5}				^{121m}Sn	8.56×10^{1}	7×10^{-6}
$^{93}ZrNb$	6.96×10^{1}	6×10^{-6}				^{93}Zr	1.92×10^{1}	2×10^{-6}
						^{95}Zr	9.37×10^{4}	8×10^{-3}

Quantity of fuel processed: 12,144 MTU
Note: Combined activities are listed for isotope pairs, no contingency is included for unidentified production load; and LaF_3 precipitation step to remove TRU from cladding removal wastes is assumed to be implemented October 1, 1985.

(a) Ci listed in Rockwell (1985).
(b) Volume 800 L/MTU; sp gr 1.25 at 20°C.

TABLE 9.4. Activity in Terminal "Product" Forms from New Tanked Waste Decayed to 1995

Key Nuclides	Onsite Stabilization and Isolation Alternative			Geologic Disposal Alternative				Reference Alternative					
	Residual in Emptied Tanks[a] total Ci	Grout Product Ci/m³ (average)	Ci/kg[b]	Grout Product Ci/m³ (average)	Ci/kg[b]	Glass Product Ci/canister[c]	Ci/kg[b]	Grout Product[d] Ci/m³ (average)	Ci/kg[b]	Ci/canister	Ci/kg[b]	Glass Product PFP[e] (Ci/canister)[c]	Ci/kg[b]
^{241}Am	1.6×10^{2}	3.4	2×10^{-3}	3.0×10^{-3}	2×10^{-6}	1.6×10^{1}	9×10^{-3}	1.5×10^{-1}	1×10^{-4}	6.4×10^{2}	4×10^{-3}	1.4×10^{2}	8×10^{-2}
^{14}C	1.4×10^{-1}	2.9×10^{-3}	5×10^{-6}	6.3×10^{-3}	2×10^{-6}	0		2.9×10^{-3}	5×10^{-6}	0		---	
^{144}Ce	6.5×10^{4}	1.4×10^{3}[e]	1	8.9×10^{-1}	6×10^{-4}	5.7×10^{3}	3	4.3×10^{1}[e]	3×10^{-2}	2.7×10^{5}	2×10^{2}	---	
^{137}Cs	2.5×10^{4}	4.0×10^{1}	3×10^{-2}	2.0	1×10^{-3}	3.0×10^{3}	2	4.1×10^{1}	3×10^{-2}	1.0×10^{5}	6×10^{1}	---	
^{3}H	3.5	2.3×10^{-2}	2×10^{-5}	1.0×10^{-3}	7×10^{-7}	0		2.3×10^{-2}	2×10^{-5}	0		---	
^{129}I	6.0×10^{-3}	1.3×10^{-4}	9×10^{-8}	7.5×10^{-5}	5×10^{-8}	0		1.3×10^{-4}	9×10^{-8}	0		---	
^{147}Pm	2.2×10^{3}	4.6×10^{1}	3×10^{-2}	2.6×10^{-2}	2×10^{-5}	2.0×10^{2}	1×10^{-1}	1.4	1×10^{-3}	9.2×10^{3}	6	---	
$^{239,40}Pu$	3.0	6.0×10^{-2}	4×10^{-5}	2.6×10^{-4}	2×10^{-7}	1.3	7×10^{-4}	4.6×10^{-3}	3×10^{-6}	7.0	4×10^{-3}	1.8×10^{1}	1×10^{-2}
^{106}Ru	7.6×10^{3}	1.6×10^{2}[e]	1×10^{-1}	9.4[e]	7×10^{-3}	3.3×10^{2}	2×10^{-1}	1.0×10^{2}[e]	7×10^{-2}	1.2×10^{4}	7	---	
^{151}Sm	2.0×10^{2}	4.2	3×10^{-3}	1.8×10^{-2}	1×10^{-5}	5.7×10^{1}	4×10^{-2}	1.3×10^{-1}	9×10^{-5}	8.3×10^{2}	5×10^{-1}	---	
^{90}Sr	2.4×10^{4}	4.4×10^{2}	4×10^{-1}	6.1×10^{-1}	4×10^{-4}	4.2×10^{3}	3×10^{1}	1.4×10^{1}	1×10^{-2}	8.8×10^{4}	6×10^{1}	---	
^{99}Tc	2.4	5.1×10^{-2}	4×10^{-5}	5.5×10^{-4}	4×10^{-7}	1.5	9×10^{-4}	5.1×10^{-2}	4×10^{-6}	~0		---	
^{93}Zr	1.2	2.5×10^{-2}	2×10^{-5}	1.2×10^{-4}	9×10^{-8}	4.1×10^{-1}	8×10^{-1}	1.5×10^{-3}	1×10^{-6}	4.9	3×10^{-3}	---	
^{95}Zr	4.9×10^{1}	1.0	2×10^{-3}	9.0×10^{-2}	1×10^{-4}	1.0	1×10^{-3}	1.6	2×10^{-3}	1.1×10^{-1}	1.3×10^{-4}	---	
Quantity:	14 Tanks	99,000 m^3 grout		99,000 m^3 grout[f]		3,180 cans[e]		99,000 m^3 grout		463 cans		132	

(a) 0.05% of inventory remains in tank, Ci and Ci/m^3 values listed in Rockwell 1985.
(b) Ci/kg calculated using grout density of 1.4×10^{3} kg/m^3, or glass density 2.8×10^{3} kg/m^3, canister volume 0.62 m^3 (Rockwell 1985).
(c) Listed Ci/canister and number canisters are incremental values based on assumption that (for Reference Alternative only) a stand alone vitrification campaign is run for plutonium finishing plant waste.
(d) Assumes plutonium finishing plant supernate is blended with other grout feed streams (neutralized current acid waste, cladding removal wastes, Hanford Facility Waste) to make 98,000 m^3 of grout.
(e) High activity values arise from short cooled (1995 vintage) neutralized current acid waste and cladding removal wastes. In actual practice, cladding removal wastes waste could be aged in tanks several years before being converted to grout.
(f) Product quantities are incremental values arising from new waste being blended with existing waste. (Existing waste alone produces 736,000 m^3 grout and 19,540 canisters.)

9.3 CESIUM AND STRONTIUM CAPSULE INVENTORIES

Cesium and strontium wastes are concentrated and encapsulated in relatively pure form along with their respective stable isotopes counterparts. The characteristics of the existing capsules are shown in Table 9.5 (Rockwell 1985, p. 7). To conservatively estimate the potential radiological impacts to the public from operational accidents during the proposed operations, the capsules containing the highest curie content are assumed to be involved in any postulated accident. The curie concentration was calculated assuming a 2.2×10^3 g waste mass per capsule as developed in Section 5.3.1.3. Details on the number of capsules assumed, heat loading, and the number of canisters anticipated for each of the alternatives are shown in Table 9.6 (Rockwell 1985). The values in Table 9.6 assume that all the cesium and strontium aqueous waste currently in B-Plant will be processed and encapsulated.

9.4 TRU-CONTAMINATED SOIL SITES - REFERENCE RADIONUCLIDE INVENTORIES

Three upper-bound inventories are provided in order to cover the anticipated range of characteristics for TRU-contaminated soil sites. These inventories are an upper-bound TRU inventory site, an upper-bound fission product inventory site, and an upper-bound TRU concentration site. Data from characterized sites are presented to give a realistic estimate of the amount of radionuclides that could be encountered. Inventory data for more than one site are given because no single site exhibits the characteristics that makes it an upper-bound inventory site for all types of release scenarios.

9.4.1 Upper-Bound TRU Inventory Site

The site containing the highest inventory of TRU, 216-Z-1 + 2TF is chosen to represent the upper-bound TRU inventory site (Rockwell 1985). The inventory (Rockwell 1985) is shown in Table 9.7. The assumption and conversion factors shown in Rockwell (1985) apply to these data. Appropriate dimensions for the 216-Z-1+2TF site are 92-m long by 13-m wide for the area over TRU contaminated soil. The total estimated volume of TRU contaminated soil is 8,300 m^3 with a density of 1.81 g/cm^3 (15,000 MT of soil).

TABLE 9.5. Characteristics of Existing Strontium and Cesium Capsules (as of December 1983)

Characteristics (per capsule)	Strontium (447 Capsules)				Cesium (1,579 Capsules)			
	As Filled	Jan. 1, 1984	Jan. 1, 1995	Jan. 1, 2050	As Filled	Jan. 1, 1984	Jan. 1, 1995	Jan. 1, 2010
Cumulative megacuries	26.6	23.2	18.0	12.7	76.2	69.8	54.1	38.3
Cumulative kilowatts	178.7	157.8	122.4	86.5	366.4	335.0	259.9	183.8
Average kilocuries	59.4	51.9	40.3	28.5	48.3	44.2	34.3	24.6
Average watts	398.8	352.9	273.7	193.6	231.9	212.2	164.6	116.4
Highest curies loading (KCi)	137.1	115.9	89.9	63.6	68.5	64.0	49.6	35.1
Concentration, highest curies loading, kCi/g	0.062	0.053	0.041	0.029	0.031	0.029	0.023	0.016

TABLE 9.6. Assumed Strontium and Cesium Capsule and Capsule Canister Details

Waste Type	Number of Capsules	Head Load, kW	Average kW per capsule	Number of Canisters[a]	Average Number of Capsules Per Canister[b]
Geologic Disposal (1995)					
Strontium Capsules[c]	600	215	0.36	184	3.3
Cesium Capsules	1,580	260	0.17	325	4.9
Total	2,180	475	---	509	---
In-Place Stabilization and Disposal and Continued Storage (2010)					
Strontium Capsules[c]	600	152	0.25	304	2
Cesium Capsules	1,580	184	0.12	368	4.3
Total	2,180	336	---	672	---

(a) Based on thermal limit of 1.17 kW/canister (^{90}Sr) and 0.9 kW/canister (^{137}Cs) geologic repository heat load limits. A half-life of 30 years is assumed for cesium and strontium.
(b) Based on a Drywell Storage Facility heat load limit of 0.5 kW/canister.
(c) Includes a projected 153 capsules for existing B Plant strontium solutions.

TABLE 9.7. Upper-Bound TRU and Fission Product Site Inventory Data (Through December 1982) for TRU-Contaminated Soil Sites

Radionuclide	Maximum TRU Site, 216-Z-1+2TF Ci	Maximum[a,b] TRU Concentration, Ci/kg	Maximum Fission Product Site, 216-S-1+2 Ci	Maximum Fission Product Concentration, Ci/kg
^{241}Am	1.3×10^{3}	1×10^{-4}	2.5×10^{1}	9×10^{-6}
^{60}Co	1.1×10^{-1}	7×10^{-9}	2.5×10^{-1}	8×10^{-8}
^{137}Cs	1.2	8×10^{-8}	1.3×10^{3}	4×10^{-4}
^{238}Pu	1.8×10^{2}	1×10^{-5}	3.3	5×10^{-7}
^{239}Pu	3.7×10^{3}	3×10^{-4}	6.8×10^{1}	2×10^{-5}
^{240}Pu	9.1×10^{2}	6×10^{-9}	1.7×10^{1}	5×10^{-6}
^{241}Pu	2.5×10^{3}	9×10^{-4}	4.7×10^{1}	6×10^{-5}
^{242}Pu	5.3×10^{-2}	3×10^{-9}	9.9×10^{-4}	2×10^{-9}
^{106}Ru	1.1×10^{-2}	7×10^{-10}	1.3×10^{-5}	4×10^{-12}
^{90}Sr	1.2	8×10^{-8}	1.5×10^{3}	5×10^{-4}
^{233}U	3.6×10^{-2}	2×10^{-9}	1.0	3×10^{-7}
^{234}U	3.7×10^{-2}	2×10^{-9}	1.0	3×10^{-7}
^{235}U	1.1×10^{-3}	7×10^{-11}	3.1×10^{-2}	1×10^{-8}
^{238}U	2.7×10^{-2}	2×10^{-9}	7.6×10^{-1}	2×10^{-7}
Total Beta	2.5×10^{3}		5.7×10^{3}	
Total Uranium (grams)	8.1×10^{4}		2.3×10^{6}	
Total Plutonium (grams)	6.4×10^{4}		1.2×10^{3}	

(a) Based on 8,300 m^3 in site 216-Z-1 + 2TF and bulk density of Hanford soils = 1.83.
(b) Peak TRU concentrations 40,000 nCi/g ~40 times higher than average shown in column.

9.4.2 Upper-Bound Fission Product Site

A liquid-TRU-contaminated soil site containing the highest total quantity of fission product activity was selected as the representative site for this class of material. This allows consideration of the impact of release scenarios depending primarily on beta-gamma emissions. The inventory data for the site chosen, 216-S-1 + 2, are shown in Table 9.7. This site volume is 1700 m^3

(Rockwell 1985) or about 3100 MT of soil. The upper-bound site inventory (Rockwell 1985) is used to estimate the potential radiological impacts to the public from accidents postulated to occur during disposal operations for this class of sites.

9.4.3 Upper-Bound TRU Concentration

The upper-bound TRU concentration that might be encountered within a localized area is of concern for some release scenarios. Characterization data (Price et al. 1979) have reported TRU concentrations up to 40,000 nCi/g in small volumes. This value is roughly 200 times the average concentration for the upper-bound TRU site. This concentration is projected to occur within only the first 1/3-m depth from the source of contamination with the TRU concentration decreasing to less than 1000 nCi/g at a distance of 2 m to less than 100 nCi/g at 15 m.

9.5 PRE-1970 TRU SOLID WASTE BURIAL GROUND - REFERENCE RADIONUCLIDE INVENTORY

Three classes of material are described for this category of waste: an upper-bound TRU inventory site; an upper-bound fission product inventory site; and, an upper-bound TRU concentration site. Measured values from actual sites are used to give a realistic upper-bound estimate of the amounts of radionuclides that could be encountered. Inventory data for more than one site are given because no single site exhibits the characteristics that make it an upper-bound inventory site for all types of release scenarios.

9.5.1 Upper-Bound TRU Inventory Site

Inventory data for the 218-W-2 site, the site containing the largest total inventory of TRU, is shown in Table 9.8 and was taken from Rockwell (1985). Assumptions and conversion factors for determining isotopic ratios of uranium and plutonium in that document were applied to these data to estimate the appropriate radionuclide inventory. The approximate dimensions of the site are 124 m-long by 179 m-wide for the area over the TRU waste, with a total volume is listed in the Rockwell document and estimated at 23,000 m^3 (4×10^{10} g).

TABLE 9.8. Upper-Bound TRU Inventory Site (218-W-2) for Pre-1970 Solid Waste Burial Grounds

Radionuclide	Inventory, Ci
^{241}Am	2.6×10^{3}
^{137}Cs	5.6
^{238}Pu	3.4×10^{2}
^{239}Pu	7.2×10^{3}
^{240}Pu	1.8×10^{3}
^{241}Pu	5.0×10^{-3}
^{242}Pu	1.0×10^{-1}
^{90}Sr	5.0
^{233}U	6.2×10^{-1}
^{234}U	6.4×10^{-1}
^{235}U	1.9×10^{-2}
^{238}U	4.7×10^{-1}
Total Beta	5.0×10^{3}
Total U (grams)	1.4×10^{6}
Total Pu (grams)	1.3×10^{5}
Volume, m^3	2.3×10^{4}
Area, m^2	1.5×10^{4}
Hanford soil density, g/cm^3	1.8

9.5.2 Upper-Bound Fission Product Inventory Site

Although not defined as a TRU waste site, the 218-E-12B site containing the highest total fission product activity was selected to allow consideration of release scenarios depending primarily on beta-gamma emissions. The inventory data (Rockwell 1985) for the site are given in Table 9.9. Concentrations were calculated based on a 4400 m^3 (Rockwell 1985) waste volume.

9.5.3 Upper-Bound TRU Concentration Site Inventory

The upper-bound TRU concentration anticipated in a localized area during the proposed disposal operations for this class of materials is of concern for some release scenarios. Caissons have the highest concentrations of TRU found

TABLE 9.9. Upper-Bound Fission Product (not a TRU Site) Inventory Site (218-E-12B) for Pre-1970 TRU Solid Waste Burial Ground

Radionuclide	Inventory, Ci	Concentration, nCi/g
^{241}Am	2.4×10^{1}	3
^{60}Co	6.6×10^{4}	8×10^{3}
^{137}Cs	2.6×10^{4}	3×10^{3}
^{238}Pu	3.2	6×10^{-1}
^{239}Pu	6.6×10^{1}	8
^{240}Pu	1.6×10^{1}	2
^{241}Pu	4.6×10^{1}	6.9
^{242}Pu	9.5×10^{-4}	1×10^{-4}
^{106}Ru	1.3×10^{1}	1.6×10^{1}
^{90}Sr	2.6×10^{4}	3×10^{3}
^{233}U	1.3×10^{-2}	2×10^{-3}
^{234}U	1.3×10^{-2}	2×10^{-3}
^{235}U	3.9×10^{-4}	5×10^{-5}
^{238}U	9.5×10^{-3}	1×10^{-3}
Total Beta	1.4×10^{5}	
Total U (grams)	2.9×10^{4}	
Total Pu (grams)	1.2×10^{3}	
Volume, m^3	4.4×10^{3}	
Area, m^2	3.4×10^{3}	
Hanford soil density, g/cm^3	1.8	

in this class of waste. The inventory for the 218-W-4B caissons has an average concentration of 6,200 nCi/g. This concentration would be projected to occur within a caisson 3 m deep x 2 m in diameter under overburden that is 4.5 m deep. Inventories and concentrations to be used for this estimate of the potential radiological impacts from operational accident where upper-bound concentrations are used are shown in Table 9.10.

TABLE 9.10. Upper-Bound TRU Concentration Inventory Site (218-W-4B) for Pre-1970 TRU Solid Waste Burial Grounds

Radionuclide	Inventory, Ci	Concentration,[a] nCi/g
^{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
^{238}Pu	4.5	5.0×10^{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}
^{241}Am	3.5×10^{1}	3.8×10^{2}
^{237}Np	0.0	0.0
^{232}Th	7.4×10^{-3}	8.2×10^{-2}
^{106}Ru	1.7×10^{1}	1.9×10^{2}
^{90}Sr	2.0×10^{3}	2.2×10^{4}
^{137}Cs	1.8×110^{3}	2.0×10^{4}
^{60}Co	5.3×10^{2}	5.9×10^{3}
^{144}Ce	1.8	2.0×10^{1}
^{144}Pr	1.8	2.0×10^{1}
Total Beta	8.6×10^{3}	9.6×10^{4}
Total U (grams)	3.0×10^{5}	
Total Pu (grams)	1.7×10^{3}	
Volume, m^3	50	

(a) Note: Only parent radionuclide activities for ^{90}Sr-^{90}Y, ^{137}Cs-^{137m}Ba, and ^{106}Ru-^{106}Rh pairs are listed. Total beta, however includes daughters, and is the sum of specifically identified fission product isotopes, plus reported general activity (without specific isotopes given). In most cases, the general activity can be treated as approximately 50% each strontium and cesium activity.

Inventory values are as of December 31, 1982, except plutonium and americium alpha-emitting isotope values are decayed to maximum ^{241}Am buildup.

9.6 RETRIEVABLE STORED AND NEWLY GENERATED TRU SOLID WASTE - REFERENCE RADIONUCLIDE INVENTORIES

Three sites were selected as upper-bound inventory sites based on the descriptions and inventories of the retrievably stored TRU solid waste sites (Rockwell 1985). The sites are designated as the primary, secondary, and tertiary sites. The primary site was selected on the basis of the site that contained the highest TRU inventory without regard to fission product or TRU concentrations. The secondary site was selected on the basis of the highest total fission product inventory. The tertiary site was selected on the basis of the highest peak TRU concentrations radionuclide inventories and concentrations are listed in Table 9.11. Waste densities are included in the table, with contact handled TRU density 0.32 g/cm^3, caissons at 0.96 g/cm^3.

9.6.1 Primary Site

The primary site (the highest TRU inventory site) selected was the 218-W-4C burial ground; the inventories (Rockwell 1985) are listed in Table 9.11. The total TRU inventory is 5.20×10^4 Ci and contains retrievably stored TRU in trenches. If the isotopic values can be assumed to be average values and the peak to average concentration ratio is 10 to 1, then the peak concentration can be obtained by dividing by the volume, 4.90×10^3 m^3, and density D.32 g/cm^3 and multiplying by a factor of 10. This peak concentration should bound most of the retrievable TRU waste sites.

The following secondary and tertiary sites can be added to modify the characteristics of the primary site to develop additional upper-bound sites as required.

9.6.2 Secondary Site

The secondary site selected and listed in Table 9.11 was the 218-W-3A burial ground since it had the highest fission product inventory (Rockwell 1985). It is not defined as a TRU waste site. The primary radionuclides considered in this selection were ^{137}Cs and ^{90}Sr. The fission product

TABLE 9.11. Retrievably Stored TRU (1970 - FY-1983) Inventory[a]

Radionuclide	218-W-3A, Ci	218-W-3A, Ci/kg	218-W-5B Trenches, Ci	218-W-5B Trenches, Ci/kg	218-W-5B Caissons, Ci	218-W-5B Caissons, Ci/kg	218-W-4C Primary Site,[a] Ci	218-W-4C Primary Site,[a] Ci/kg
^{241}Am	2.6×10^{1}	2×10^{-5}	1.1×10^{3}	1×10^{-3}	1.21×10^{2}	9×10^{-3}	1.5×10^{2}	2×10^{-6}
^{243}Am		---		---		---	4.54×10^{-2}	3×10^{-8}
^{14}C	1.59	1×10^{-6}						
^{246}Cm							7.2	5×10^{-6}
^{60}Co	2.2×10^{3}	2×10^{-3}	8.7×10^{2}	9×10^{-4}	1.5×10^{3}	7×10^{-2}		
^{144}Ce	5.29×10^{4}	4×10^{-2}	1.6×10^{-1}	2×10^{-7}	5.22×10^{4}	3		
^{137}Cs	1.7×10^{4}	1×10^{-2}	8.4×10^{2}	8×10^{-4}	1.6×10^{3}	8×10^{-2}	1.2×10^{4}	8×10^{-3}
^{155}Eu	3.52×10^{1}	3×10^{-5}						
^{3}H	6.8	5×10^{-6}						
^{85}Kr	3.73×10^{3}	3×10^{-3}						
^{237}Np	7×10^{-9}	6×10^{-15}	6.5×10^{-2}	6×10^{-8}	1.5×10^{-2}	7×10^{-7}	3.5×10^{-4}	2×10^{-10}
^{147}Pm	4×10^{4}	3×10^{-2}						
^{144}Pr	5.3×10^{4}	4×10^{-2}	1.6×10^{-1}	2×10^{-7}	5.2×10^{4}	3×10^{0}		
^{238}Pu	$2.1 \times 10^{+1}$	2×10^{-5}	1.5×10^{2}	2×10^{-4}	2.2×10^{1}	7.5×10^{-4}	3.2×10^{4}	2×10^{-2}
^{239}Pu	1×10^{3}	8×10^{-4}	2.8×10^{3}	3×10^{-3}	3.0×10^{2}	1×10^{-2}	1.6×10^{4}	1×10^{-2}
^{240}Pu	2.5×10^{2}	2×10^{-4}	6.9×10^{2}	7×10^{-4}	7.5×10^{1}	2×10^{-2}	3.5×10^{3}	2×10^{-3}
^{241}Pu	3×10^{3}	2×10^{-3}	8.4×10^{3}	8×10^{-3}	9.1×10^{2}	1.9×10^{0}	4.8×10^{4}	3×10^{-2}
^{242}Pu	9.5×10^{-2}	8×10^{-8}	2.7×10^{-1}	3×10^{-7}	2.9×10^{-2}	1×10^{-6}	1.4	9×10^{-7}
^{106}Ru	5.8×10^{3}	5×10^{-3}	1.3×10^{2}	1×10^{-4}	3.3×10^{3}	2×10^{-1}	2.3×10^{4}	2×10^{-2}
^{90}Sr	1.67×10^{4}	1×10^{-2}	7.8×10^{2}	8×10^{-4}	1.5×10^{3}	7×10^{-2}	1.1×10^{4}	7×10^{-3}
^{232}Th	2.7×10^{-3}	2×10^{-9}	6×10^{-2}	6×10^{-8}	1.1×10^{-3}	5×10^{-8}	1.8×10^{-3}	1×10^{-9}
^{233}U	3.6	3×10^{-6}	1.3	1×10^{-6}	2.4×10^{-1}	1×10^{-5}	7.6×10^{-1}	5×10^{-7}
^{234}U	3.7	3×10^{-6}	1.3	1×10^{-6}	3.0×10^{-2}	1×10^{-6}	7.8×10^{-1}	5×10^{-7}
^{235}U	1.0×10^{-1}	8×10^{-8}	4.0×10^{-2}	4×10^{-8}	9.0×10^{-4}	4×10^{-8}	2.3×10^{-2}	1×10^{-8}
^{238}U	2.7	2×10^{-6}	1	1×10^{-6}	2.2×10^{-2}	1×10^{-6}	5.7×10^{-1}	4×10^{-7}
Volume, m^3		4×10^{3}		3×10^{3}		2×10^{1}		5×10^{3}
Area, m^2		1×10^{4}		6×10^{3}				7×10^{3}
Density, kg/m^3		320.4		320.4		961.2		320.4

(a) Peak concentrations are estimated to be a factor of 10 higher than average value given.

inventory is: ^{90}Sr, 1.67 x 10^4 Ci, and ^{137}Cs, 1.70 x 10^4 Ci. The peak anticipated concentration can be obtained by dividing the volume of 3.9 x 10^3 m^3 and the density and multiplying by a factor of 10.

9.6.3 Tertiary Site

The tertiary site selected on the basis of having the highest TRU concentration was 218-W-4B, and included caisson waste. A total of 21.7 m^3 (Rockwell 1985) of caisson waste was received from 1970 through FY 1983. Radionuclide inventories (Rockwell 1985) and concentrations for the 218-W-4B site are included in Table 9.11. Newly generated caisson waste can have slightly elevated levels of certain important radionuclides (Rockwell 1985). Because of the very small volume of waste involved, this site is not considered representative of the majority of waste within this category.

10.0 DOWNWIND TRANSPORT DOSE CALCULATIONAL METHODS AND RADIATION DOSES FROM POSTULATED OPERATIONAL ACCIDENTS

Population and maximum-individual dose estimates were calculated for each accident scenario postulated for a waste processing activity. Occupational doses for these accident situations were not addressed because of the unavailability of facility specific information (such as manpower requirements, shielding, distance from the source, etc.) essential to the analysis of occupational dose. The assumptions, models, and input parameters required for the calculation of maximum individual and population dose estimates for each of the waste disposal, along with the dose alternatives, are described below.

10.1 ACCIDENT SCENARIOS

Many different accident scenarios were developed as part of this project. Only those accidents which resulted in an airborne release of radioactive material to the offsite environment were considered in the dose analysis. Releases of radioactive material to the air and in liquid discharges were postulated, but only atmospheric accidental releases had the potential for migrating offsite during waste disposal activities.

The duration of a release during an accident can have a significant bearing on the radiological consequences of the event. In this study all releases were postulated to be of short duration (less than an hour). Even with a short-term, or acute, release there are many ways in which the radionuclides can continue to expose the population long after the release has been terminated. For example, in a typical accident scenario, a cloud (or plume) of contaminated material is postulated to be released. As this plume travels offsite, members of the public may be irradiated by the radionuclides contained in the cloud passing overhead. If they inhale some of the radioactive material from the cloud as it passes they can receive an additional exposure. If some of the radioactive material deposits on plants or on the ground it can result in long-term exposure to people residing in the area. While the possibilities for exposure seem large and complex, a set of computer programs has been

developed to calculate the dose consequences from all the major exposure pathways from each of the basic release scenarios. These programs, listed in Table 10.1, are discussed in the following section. All have been individually documented.

10.2 DOSE CODES

Several computer programs, or dose codes, were used to calculate the radiation dose to members of the public in the event of an accidental release of radioactive material to the environment. The standard method for evaluating the radiological impact of a release is to estimate the dose to the "maximally-exposed individual" (the single person receiving the highest dose from the release) and to the entire exposed population as a whole. The doses are reported in rem for the maximum individual and man-rem for the population.

The computer programs used to calculate dose to the maximally-exposed individual and to the regional population are shown in Figure 10.1. The programs SUBDOSA (Strenge, Watson and Houston 1975), DACRIN (Houston, Strenge and

TABLE 10.1. Computer Programs Used to Calculate Potential Radiation Doses from Releases During Waste Disposal Operations

Program	Type of Dose	Reference
SUBDOSA	One-year air submersion dose from acute (finite cloud) or chronic (semi-infinite cloud) releases,	Strenge, Watson and Houston 1975
DACRIN	Individual and collective inhalation doses from chronic or acute releases, one-year doses, dose commitments, and accumulated doses.	Houston, Strenge and Watson 1974; Strenge et al. 1975
PABLM	Individual and collective doses from contaminated farm products, from either air deposition or irrigation, one-year dose, dose commitment, and accumulated dose. Individual and collective doses from contaminated water and aquatic foods and aquatic recreation, one-year dose, dose commitment, and accumulated dose.	Napier, Kennedy and Soldat 1980
ALLDOS	Report generator using precalculated factors from SUBDOSA, DACRIN and PABLM. Simplifies repetitive calculations of individual and population doses.	Strenge et al. 1980

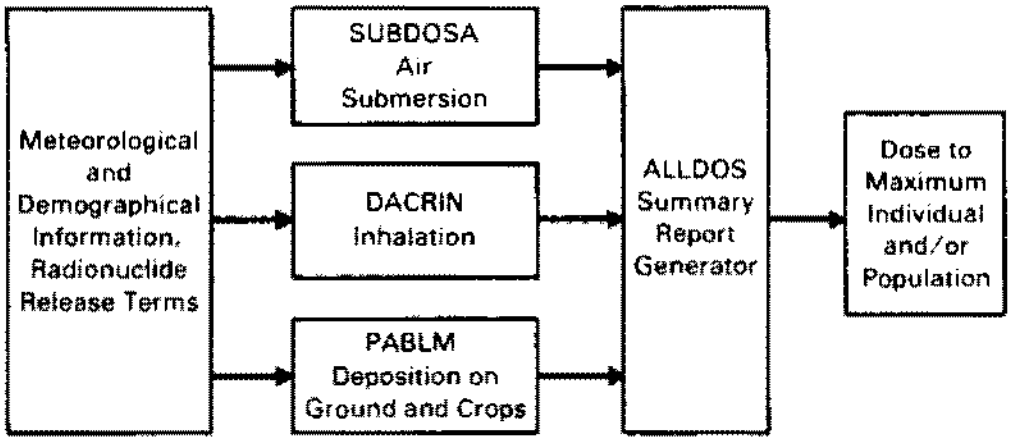


FIGURE 10.1. Computer Programs for Calculating Public Doses from Routine or Accidental Releases of Radionuclides During Operations

Watson 1974, Strenge et al. 1975), and PABLM (Napier, Kennedy and Soldat 1980) uses information about the radionuclides released, meteorology, and population distribution to calculate air submersion, inhalation, and ingestion doses, respectively. For cases in which repetitive calculations are necessary for the same environmental conditions (as for the various alternatives at Hanford), it was convenient to use a fourth program, ALLDOS (Strenge et al. 1980), to summarize the results of the calculations. This combination of computer programs was used for the acute accidental releases postulated for operations involving the Hanford wastes.

ALLDOS uses precalculated dose conversion factors to generate dose commitments to a maximum individual and the population in the region of the release site. The code was developed for calculation of radiation doses from postulated releases of aged radioactive wastes. These radionuclides are long-lived with decay half-lives of several weeks or longer. Therefore, radioactive decay in transit from the release point to the location of exposure in the environment is not considered.

10.3 STANDARD HANFORD METEOROLOGICAL PARAMETERS

The dose calculations rely on the use of meteorological data to provide an estimate of the manner in which radioactive material would most likely disperse

following an accidental release to the atmosphere. The longer the data have been collected, the more realistic the estimate. Meteorological data have been collected at the Hanford Meteorology Station, near 200 West Area, for the past 30 years (Stone et al. 1983). The results of these monitoring efforts are published in Hanford Annual Reports (Sula et al. 1982,1983).

For short-term accidental releases, the meteorological parameter used in the dose calculations is the value of air concentration of radionuclides per unit release that is not exceeded more than 5% of the time; it is referred to as E/Q, with units of sec/m^3. Typically the results of the meteorological efforts are tabulated and reported as X/Q', or Ci/m^3 per Ci/sec of release. The value of X/Q' can be converted to E/Q when the length of release is known or can be estimated. Values of X/Q' used in these calculations were based on data given in PNL-3777 Rev 1 (McCormack, Ramsdell, and Napier, 1984).

Demographical data also play an important role in the calculation of radiation dose. It is the combination of meteorological and demographical information that indicate which population group will receive the highest exposure from radioactive releases. In the case of accidental releases from the 200 Areas of the Hanford Site, the population projected to receive the greatest exposure lives 10 to 50 miles SE of the waste site. The population data used in this assessment came from Population Estimates for the Areas Within a 50-Mile Radius of Four Reference Points on the Hanford Site (Sommer, Rau, Robinson, 1981). Meteorological and population data used in the dose calculations are shown in Table 10.2.

For the maximally-exposed individual, the 95th Percentile center-line X/Q' values provided in Hanford Dose Overview Program: Standardized Methods and Data for Hanford Environmental Dose Calculations were used (McCormack, Ramsdell, and Napier 1984). The following assumptions were used to determine the location of the maximally exposed individual for accidental releases. For purposes of inhalation and submersion dose calculations, the maximally exposed individual was assumed to be positioned on Highway 240, 8.8 km south of the 200 Areas; but for ingestion dose calculations this person is presumed to live

TABLE 10.2. Population Values and Sector Averaged X/Q's Used in the Assessment of Radiation Dose

Distance, miles	Population Size	Ground Level X/Q'	Elevated X/Q'
0-10	0		
10-20	8,664	1.02×10^{-6}	4.21×10^{-7}
20-30	62,866	5.76×10^{-7}	2.04×10^{-7}
30-40	66,306	4.10×10^{-7}	1.33×10^{-7}
40-50	4,094	3.10×10^{-7}	1.03×10^{-7}
Population Weighted X/Q'		7.35×10^{-2}	2.57×10^{-2}

on a farm in Franklin County 24 km East of the 200 Areas. The values used in the calculations are shown in Table 10.3.

10.4 STANDARD HANFORD EXPOSURE PARAMETERS

Data required for the dose programs include dietary and recreational preferences and habits in the general population, as well as agricultural practices in the general region. The standard Hanford terrestrial pathway data are given in Sula and Blumer (1981). The growing period, yield, and irrigation rate reflect agricultural practices in the Columbia River Basin. The parameters for the average member of the population reflect the dietary habits of the Tri-Cities residents. Values used for the maximally exposed individual were selected to represent a worst-case individual supporting himself and family with a large garden and farm animals. Standardized input for Hanford environmental documentation is summarized in recent publications (Napier 1981, McCormack, Ramsdell and Napier 1984).

TABLE 10.3. Maximum Individual, 95th Percentile Centerline $\overline{X/Q'}$ Values

Pathway	Location	Elevation	
		Ground	60 Meters
Inhalation	5 mi-S	3.40×10^{-5}	1.05×10^{-5}
Ingestion	12 mi-E	1.50×10^{-5}	4.90×10^{-6}

10.5 RADIATION DOSE ESTIMATES FROM ACCIDENTAL ATMOSPHERIC RELEASES DURING WASTE DISPOSAL OPERATIONS

Doses to the maximally-exposed individual and to the population living downwind from the Hanford Site were estimated for each of the proposed waste disposal alternatives. The atmospheric releases used as the basis for the dose calculations are those given for each of the accident scenarios listed in Tables 5.1, 6.1, 7.2, and 8.1 and discussed in Sections 5, 6, 7, and 8.

10.5.1 Radiation Doses Associated with the Geologic Disposal Alternative

Twenty-six separate dose calculations were performed to analyze the potential radiological impact from the disposal of the Hanford Defense Wastes under the geologic disposal alternative. The dose estimates, listed in Tables 10.4a, b, and c, are divided according to the maximally exposed individual inhalation dose, the maximally exposed individual ingestion dose, and the population dose. For each of these three categories, the first year total body and critical organ dose[a] and the 70 year total-body and critical-organ dose commitments were calculated. The accident resulting in the greatest public dose was the ferro/ferricyanide explosion postulated for the handling of the single-shell tank wastes. It has been postulated that a layer containing ferro- or ferricyanide precipitates might be present in the single-shell tank wastes. Under the proper conditions, this material could react explosively with nitrates present in the waste. If ferrocyanide precipitates are present in the waste, the potential for an explosion, as discussed in Sections 5, 6, and 7, does exist. However, the presence of such material, in quantities sufficient to produce such an event is still a subject of some debate (RHO-LD-55, Rockwell 1980).

The accident with the second-highest radiological consequences is from the pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval of the tank wastes. This accident is described for both the existing and future tank wastes.

(a) The "critical organ" is the organ receiving the highest dose during the time period under consideration.

TABLE 10.4a. Geologic Disposal Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Max. Ind. Inhalation)

	Waste Form	Facility	Accident	Location	Reference[a] Inventory	Maximum Individual Dose,[b] rem — Maximum Inhalation — 1st Year Dose T. Body	1st Year Dose C. Organ	70 Year Dose Commitment T. Body	70 Year Dose Commitment C. Organ
Existing Tank Waste									
1. Mechanical retrieval	Salt cake	Mobile platform	Explosion	Waste Tank	T 8.1c2	9×10^{-2}	2×10^{0}L	1×10^{0}	2×10^{1}B
2. Hydraulic retrieval	Salt cake	NA[c]	Pressurized release	Diversion valve	T 8.1c2	3×10^{-6}	7×10^{-1}L	9×10^{-1}	6×10^{0}B
3. Sr, Cs, Tc removal (radionuclide concentration)									
3.1 Sludge washing, solid/liquid separation	Sludge	Sludge washing canyon	Filter failure	Filters	T 8.1c2	6×10^{-7}	1×10^{-5}L	6×10^{-6}	9×10^{-5}B
3.2 Complexant destruction	No information available								
3.3 Radionuclide removal	Ion exchange resin	Radionuclide concentration	Ion exchange fire	Resin column					
4. Glass immobilization (vitrification)	Molten glass	Glass immobilization	Loss of filters	Filters	T 8.2c6	3×10^{-7}	1×10^{-5}L	3×10^{-6}	4×10^{-5}B
5. Grout decontaminated salt solutions	Decontaminated salt solutions	Transportable grout	Liquid spray from line	Transfer line	T 8.2c4	6×10^{-14}	1×10^{-12}L	6×10^{-13}	9×10^{-12}B
6. Fill and cover empty tank (dome fall)	Residual tank waste	None	Dome collapse	Tank	T 8.1c2	2×10^{-3}	5×10^{-2}L	3×10^{-2}	4×10^{-1}B
Future Tank Waste									
1. Retrieval	NCAW[d]	NA	Pressurized release	Diversion valve	T 8.3A	4×10^{-7}	1×10^{0}L	4×10^{-1}L	7×10^{0}B
2. Solid/liquid separation	NCAW	Sludge washing canyon	Filter failure	Filters	T, 8.3A	6×10^{-7}	1×10^{-5}L	6×10^{-6}	1×10^{-4}B
3. Sr, Cs, Tc, TRU removal	Ion exchange resin	Radionuclide concentration	Ion exchange fire	Resin column	Insufficient Data				
4. Grout decontaminated liquid	Decontaminated liquid	Transportable grout	Liquid spray from line	Transfer line	T 8.4c5	2×10^{-13}	2×10^{-11}L	1×10^{-12}	3×10^{-11}L
5. Vitrification	Molten glass	Glass immobilization	Loss of filters	Filters	T 8.4c2	3×10^{-6}	7×10^{-5}L	4×10^{-6}	6×10^{-4}B
6. Fill and cover empty tank (dome fall)	Residual tank waste	None	Dome collapse	Tank	T 8.3A	3×10^{-5}	8×10^{-4}L	3×10^{-4}	5×10^{-3}B
Strontium and Cesium Capsules									
1. Remove from WESF[e]	Encapsulated waste	WESF	Capsule drop	Storage area	T 8.5C3 T 8.5C7	1×10^{-7} 2×10^{-8}	2×10^{-6}L 2×10^{-8}B	1×10^{-6} 2×10^{-8}	2×10^{-5}B 2×10^{-8}B
2. Capsule packaging	Encapsulated waste	CPF[f]	Machinery impacts capsule	"Load" station	T 8.5C1 T 8.5C7	1×10^{-6} 2×10^{-9}	3×10^{-4}L 2×10^{-7}B	2×10^{-7} 2×10^{-9}	3×10^{-6}B 3×10^{-9}B

TABLE 10.4a. (contd)

	Waste Form	Facility	Accident	Location	Reference[a] Inventory	Maximum Individual Dose,[b] Rem: Maximum Inhalation: 1st Year Dose: T. Body	1st Year Dose: C. Organ	70 Year Dose Commitment: T. Body	70 Year Dose Commitment: C. Organ
TRU-Contaminated Soil									
1. Retrieve	Contaminated soil	Retrieval Facility	Explosion	Battery charging area	T 8.7c2	4×10^{-8}	5×10^{-6}L	2×10^{-6}	4×10^{-5}L
2. Process	Slagging pyrolysis off gas	SPI[g]	Explosion	Gasifier	T 8.7c2	4×10^{-7}	5×10^{-5}L	2×10^{-5}	3×10^{-4}B
Pre-1970 TRU Solid Waste									
1. Retrieve	Contaminated soil/waste	Retrieval facility	Explosion	Battery charging area	T 8.9c2	4×10^{-8}	3×10^{-6}L	4×10^{-7}	6×10^{-4}B
2. Sorting and related operations	Packaged waste	SPI	Pressurized release	Drum	T 8.9c2	1×10^{-12}	4×10^{-11}L	1×10^{-11}	2×10^{-10}B
3. Process	Slagging pyrolysis off gas	SPI	Explosion	Gasifier	T 8.9c2	4×10^{-6}	2×10^{-4}L	1×10^{-6}	2×10^{-3}B
Retrievably Stored and Newly Generated TRU									
1. Release 1.1 RH[h] TRU	Packaged waste	Caisson retrieval	Pressurized release	Metal can	T 8.11c6	5×10^{-12}	6×10^{-10}L	1×10^{-10}	2×10^{-9}B
1.2. CH[i] TRU	Packaged waste	None	Pressurized release	Drum	T 8.11c6	1×10^{-3}	2×10^{-1}L	5×10^{-2}	1×10^{0}B
2. Sorting and Related Operations 2.1 RH TRU	Packaged waste	SPI	Pressurized release	Drum	T 8.11c6	3×10^{-10}	3×10^{-8}L	6×10^{-9}	1×10^{-7}B
2.2 CH TRU	Packaged waste	CH WRAP[j]	Pressurized release	Drum	T 8.11c6	1×10^{-10}	1×10^{-8}L	5×10^{-9}	1×10^{-7}B
3. Process 3.1 RH TRU	Slagging pyrolysis off gas	SPI	Explosion	Gasifier	T 8.11c6	8×10^{-5}	8×10^{-3}L	2×10^{-3}	4×10^{-2}B
3.2 CH WRAP	Packaged waste	CH WRAP	Fire	Drum	T 8.11c6	5×10^{-12}	6×10^{-10}L	2×10^{-10}	5×10^{-9}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Not applicable.
(d) Neutralized Current Acid Waste.
(e) Waste Encapsulation and Storage Facility.
(f) Capsule Packaging Facility.
(g) Slagging Pyrolysis Incinerator.
(h) Remote Handled.
(i) Contact Handled.
(j) Waste Retrieval and Packaging Facility.

TABLE 10.4b. Geologic Disposal Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Max. Ind. Ingestion)

	Waste Form	Facility	Accident	Location	Maximum Individual Dose,[b] rem — Maximum Ingestion — 1st Year Dose T. Body	1st Year Dose C. Organ	70 Year Dose Commitment T. Body	70 Year Dose Commitment C. Organ
Existing Tank Waste								
1. Mechanical retrieval	Salt cake	Mobile platform	Explosion	Waste Tank	6×10^{-2}	2×10^{-1}B	1×10^{0}	5×10^{0}B
2. Hydraulic retrieval	Salt cake	NA[c]	Pressurized release	Diversion valve	2×10^{-2}	8×10^{-2}B	5×10^{-1}	2×10^{0}B
3. Sr, Cs, Tc removal (radionuclide concentration)								
3.1 Sludge washing, solid/liquid separation	sludge	Sludge washing canyon	Filter failure	Filters	3×10^{-7}	1×10^{-6}B	7×10^{-6}	3×10^{-5}B
3.2 Complexant destruction	No information available							
3.3 Radionuclide removal	Ion exchange resin	Radionuclide concentration	Ion exchange fire	Resin column				
4. Glass immobilization (vitrification)	Molten glass	Glass immobilization	Loss of filters	Filters	2×10^{-7}	7×10^{-7}B	4×10^{-6}	1×10^{-5}B
5. Grout decontaminated salt solutions	Decontaminated salt solutions	Transportable grout	Liquid spray from line	Transfer line	4×10^{-13}	1×10^{-12}	1×10^{-12}	5×10^{-12}B
6. Fill and cover empty tank (dome fill)	Residual tank waste	None	Dome collapse	Tank	2×10^{-3}	5×10^{-3}B	3×10^{-2}	4×10^{-1}B
Future Tank Waste								
1. Retrieval	NCAW[d]	NA	Pressurized release	Diversion valve	6×10^{-2}	1×10^{-1}B	$5 \times 10^{1-}$	6×10^{0}B
2. Solid/liquid separation	NCAW	Sludge washing canyon	Filter failure	Filters	8×10^{-7}	2×10^{-6}B	7×10^{-6}	2×10^{-5}B
3. Sr, Cs, Tc, TRU removal	Ion exchange resin	Radionuclide concentration	Ion exchange fire	Resin column	Insufficient Data			
4. Grout decontaminated liquid	Decontaminated liquid	Transportable grout	Liquid spray from line	Transfer line	6×10^{-13}	2×10^{-12}B	2×10^{-12}	7×10^{-12}B
5. Vitrification	Molten glass	Glass immobilization	Loss of filters	Filters	3×10^{-6}	9×10^{-6}B	5×10^{-5}	2×10^{-4}B
6. Fill and cover empty tank (dome fill)	Residual tank waste	None	Dome collapse	Tank	4×10^{-5}	8×10^{-5}B	4×10^{-4}	1×10^{-3}B
Strontium and Cesium Capsules								
1. Remove from WESF[e]	Encapsulated waste	WESF	Capsule drop	Storage area	9×10^{-8} 8×10^{-8}	3×10^{-7}B 8×10^{-8}B	2×10^{-6} 1×10^{-7}	7×10^{-6}B 1×10^{-7}B
2. Capsule packaging	Encapsulated waste	CPF[f]	Machinery impacts capsule	"Load" station	1×10^{-8} 1×10^{-8}	4×10^{-8}B 1×10^{-8}B	2×10^{-7} 2×10^{-8}	9×10^{-7}B 2×10^{-8}B

TABLE 10.4b. (contd)

	Waste Form	Facility	Accident	Location	Maximum Individual Dose,[b] rem — Maximum Ingestion — 1st Year Dose T. Body	1st Year Dose C. Organ	70 Year Dose Commitment T. Body	70 Year Dose Commitment C. Organ
TRU-Contaminated Soil								
1. Retrieve	Contaminated soil	Retrieval facility	Explosion	Battery charging area	3×10^{-11}	4×10^{-10}B	2×10^{-9}	4×10^{-8}B
2. Process	Slagging pyrolysis off gas	SPI[g]	Explosion	Gasifier	3×10^{-10}	4×10^{-9}B	2×10^{-8}	4×10^{-7}B
Pre-1970 TRU Solid Waste								
1. Retrieve	Contaminated soil/waste	Retrieval facility	Explosion	Battery charging area	9×10^{-8}	1×10^{-7}B	5×10^{-7}	2×10^{-6}B
2. Sorting and related operations	Packaged waste	SPI	Pressurized release	Drum	3×10^{-12}	4×10^{-12}B	2×10^{-11}	5×10^{-11}B
3. Process	Slagging pyrolysis off gas	SPI	Explosion	Gasifier	1×10^{-6}	5×10^{-6}B	3×10^{-5}	1×10^{-4}B
Retrievably Stored and Newly Generated TRU								
1. Release								
1.1 RH[h] TRU	Packaged waste	Caisson retrieval	Pressurized release	Metal can	1×10^{-12}	2×10^{-12}B	7×10^{-12}	2×10^{-11}B
1.2. CH[i] TRU	Packaged waste	None	Pressurized release	Drum	5×10^{-5}	1×10^{-4}B	5×10^{-4}	2×10^{-3}B
2. Sorting and Related Operations								
2.1 RH TRU	Packaged waste	SPI	Pressurized release	Drum	6×10^{-11}	1×10^{-10}B	4×10^{-10}	1×10^{-9}B
2.2 CH TRU	Packaged waste	CH WRAP[j]	Pressurized release	Drum	4×10^{-12}	9×10^{-12}B	4×10^{-11}	2×10^{-10}B
3. Process								
3.1 RH TRU	Slagging pyrolysis off gas	SPI	Explosion	Gasifier	2×10^{-5}	3×10^{-5}B	1×10^{-4}	4×10^{-4}B
3.2 CH WRAP	Packaged waste	CH WRAP	Fire	Drum	2×10^{-13}	4×10^{-13}B	2×10^{-12}	9×10^{-12}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Not applicable.
(d) Neutralized Current Acid Waste.
(e) Waste Encapsulation and Storage Facility.
(f) Capsule Packaging Facility.
(g) Slagging Pyrolysis Incinerator.
(h) Remote Handled.
(i) Contact Handled.
(j) Waste Retrieval and Packaging Facility.

TABLE 10.4c. Geologic Disposal Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Total Population Dose)

	Waste Form	Facility	Accident	Location	Population Dose,[b] rem: 1st Year Dose T. Body	1st Year Dose C. Organ	70 Year Dose Commitment T. Body	70 Year Dose Commitment C. Organ
Existing Tank Waste								
1. Mechanical retrieval	Salt cake	Mobile platform	Explosion	Waste Tank	4×10^{2}	4×10^{3} L,B	7×10^{3}	6×10^{4}
2. Hydraulic retrieval	Salt cake	NA[c]	Pressurized release	Diversion valve	1×10^{2}	2×10^{3} L	2×10^{3}	2×10^{4} B
3. Sr, Cs, Tc removal (radionuclide concentration)								
3.1 Sludge washing, solid/liquid separation	sludge	Sludge washing canyon	Filter failure	Filters	2×10^{-3}	2×10^{-2} L,B	4×10^{-2}	3×10^{-1} B
3.2 Complexant destruction	No information available							
3.3 Radionuclide removal	Ion exchange resin	Radionuclide concentration	Ion exchange fire	Resin column				
4. Glass immobilization (vitrification)	Molten glass	Glass immobilization	Loss of filters	Filters	1×10^{-3}	3×10^{-2} L	2×10^{-2}	1×10^{-1} B
5. Grout decontaminated salt solutions	Decontaminated salt solutions	Transportable grout	Liquid spray from line	Transfer line	2×10^{-9}	7×10^{-9} B	5×10^{-9}	4×10^{-8} B
6. Fill and cover empty tank (dome fill)	Residual tank waste	None	Dome collapse	Tank	9×10^{0}	1×10^{2} L	2×10^{2}	1×10^{3} B
Future Tank Waste								
1. Retrieval	NCAW[d]	NA	Pressurized release	Diversion valve	3×10^{2}	2×10^{3} L	2×10^{3}	2×10^{4} B
2. Solid/liquid separation	NCAW	Sludge washing canyon	Filter failure	Filters	4×10^{-3}	3×10^{-2} L	4×10^{-2}	3×10^{-1} B
3. Sr, Cs, Tc, TRU removal	Ion exchange resin	Radionuclide concentration	Ion exchange fire	Resin column	Insufficient Data			
4. Grout decontaminated liquid	Decontaminated liquid	Transportable grout	Liquid spray from line	Transfer line	3×10^{-9}	5×10^{-8} L	1×10^{-8}	8×10^{-8} L
5. Vitrification	Molten glass	Glass immobilization	Loss of filters	Filters	2×10^{-2}	2×10^{-1} L	3×10^{-1}	2×10^{0} B
6. Fill and cover empty tank (dome fill)	Residual tank waste	None	Dome collapse	Tank	2×10^{-1}	2×10^{0} L	2×10^{0}	1×10^{1} B
Strontium and Cesium Capsules								
1. Remove from WESF[e]	Encapsulated waste	WESF	Capsule drop	Storage area	6×10^{-4} 3×10^{-4}	5×10^{-3} L,B 3×10^{-4} B	1×10^{-2} 5×10^{-4}	8×10^{-2} B 6×10^{-4} B
2. Capsule packaging	Encapsulated waste	CPF[f]	Machinery impacts capsule	"Load" station	7×10^{-5} 4×10^{-5}	7×10^{-4} L 4×10^{-5} B	1×10^{-3} 6×10^{-5}	9×10^{-3} B 7×10^{-5} B

TABLE 10.4c. (contd)

	Waste Form	Facility	Accident	Location	Population Dose,[b] rem: 1st Year Dose T. Body	1st Year Dose C. Organ	70 Year Dose Commitment T. Body	70 Year Dose Commitment C. Organ
TRU-Contaminated Soil								
1. Retrieve	Contaminated soil	Retrieval facility	Explosion	Battery charging area	9×10^{-5}	1×10^{-2}L	4×10^{-3}	9×10^{-2}B
2. Process	Slagging pyrolysis off gas	SPI[g]	Explosion	Gasifier	9×10^{-4}	1×10^{-1}L	4×10^{-2}	9×10^{-1}B
Pre-1970 TRU Solid Waste								
1. Retrieve	Contaminated soil/waste	Retrieval facility	Explosion	Battery charging area	4×10^{-4}	3×10^{-3}L	2×10^{-3}	2×10^{-2}B
2. Sorting and related operations	Packaged waste	SPI	Pressurized release	Drum	1×10^{-8}	1×10^{-7}L	8×10^{-8}	6×10^{-7}B
3. Process	Slagging pyrolysis off gas	SPI	Explosion	Gasifier	1×10^{-2}	6×10^{-1}L	3×10^{-1}	5×10^{0}B
Retrievably Stored and Newly Generated TRU								
1. Release								
1.1 RH[h] TRU	Packaged waste	Caisson retrieval	Pressurized release	Metal can	1×10^{-8}	1×10^{-6}L	3×10^{-7}	5×10^{-6}B
1.2. CH[i] TRU	Packaged waste	None	Pressurized release	Drum	3×10^{0}	3×10^{2}L	1×10^{2}	2×10^{3}B
2. Sorting and Related Operations								
2.1 RH TRU	Packaged waste	SPI	Pressurized release	Drum	9×10^{-7}	6×10^{-5}L	2×10^{-5}	3×10^{-4}B
2.2 CH TRU	Packaged waste	CH WRAP[j]	Pressurized release	Drum	3×10^{-7}	3×10^{-5}L	1×10^{-5}	2×10^{-4}B
3. Process								
3.1 RH TRU	Slagging pyrolysis off gas	SPI	Explosion	Gasifier	2×10^{-1}	2×10^{1}L	4×10^{0}	8×10^{1}B
3.2 CH WRAP	Packaged waste	CH WRAP	Fire	Drum	1×10^{-8}	1×10^{-6}L	5×10^{-7}	1×10^{-5}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Not applicable.
(d) Neutralized Current Acid Waste.
(e) Waste Encapsulation and Storage Facility.
(f) Capsule Packaging Facility.
(g) Slagging Pyrolysis Incinerator.
(h) Remote Handled.
(i) Contact Handled.
(j) Waste Retrieval and Packaging Facility.

10.5.2 Radiation Doses Associated with the In-Place Stabilization Alternative

Twenty separate dose calculations were performed to analyze the potential radiological impact from the disposal of the Hanford Defense Wastes under the in-place stabilization and disposal alternative. The dose estimates, listed in Tables 10.5a, b, and c, are divided according to the maximally exposed individual inhalation dose, the maximally exposed individual ingestion dose, and the population dose. For each of these three categories the first year total body and critical organ dose, and the 70 year total body and critical organ dose commitment were calculated. The accident resulting in the greatest public dose was the ferrocyanide/organic explosion postulated for the handling of the single-shell tank wastes. This upper bound accident is the same as described for the geologic disposal alternative.

The accident with the second-highest radiological consequences is from the pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval of the tank wastes. This accident is also the same as described in the geologic disposal alternative.

10.5.3 Radiation Doses Associated with the Reference Alternative Disposal Action

Twenty-seven separate dose calculations were performed to analyze the potential radiological impact from the disposal of the Hanford Defense Wastes under the Reference Alternative. The dose estimates, listed in Tables 10.6a, b, and c, are divided according to the maximally exposed individual inhalation dose, the maximally exposed individual ingestion dose, and the population dose. For each of these three categories the first year total body and critical organ dose, and the 70 year total body and critical organ dose commitment were calculated. The accident resulting in the greatest public dose was the ferrocyanide/organic explosion postulated for the handling of the single-shell tank wastes. This upper bound accident is the same as described for the geologic disposal alternative and for the in-place stabilization and disposal alternative.

The accident with the second-highest radiological consequences is from the pressurized release of liquid waste due to failure of a diversion valve during

TABLE 10.5a. In-Place Stabilization and Disposal Alternative Doses from Accidental Releases for Operations Involving Six Waste Forms (Max. Ind. Inhalation)

	Waste Form	Facility	Accident	Location	Reference[a] Inventory	Maximum Individual Dose,[b] rem — Maximum Inhalation — 1st Year Dose T. Body	1st Year Dose C. Organ	70-Year Dose Commitment T. Body	70-Year Dose Commitment C. Organ
Existing Tank Waste									
1. Dry single-shell tanks	Salt cake	Tank	Explosion	Tank	T 8.1c2	9×10^{-2}	2×10^{0}L	1×10^{0}	2×10^{1}B
2. Fill tank dome	Salt cake	Tank	Dome collapse	Tank	T 8.1c2	5×10^{-2}	1×10^{0}L	6×10^{-1}	1×10^{1}B
3. Hydraulic retrieval of residual liquid (double shell)	Liquid waste	Tank	Pressurized spray	Diversion valve	T 8.1c2	3×10^{-2}	7×10^{-1}L	4×10^{-1}	6×10^{0}B
4. Complexant destruction	No information available								
5. Grout	Aqueous solutions	Transportable grout	Liquid spray from line	Transfer line	T 8.2c2	1×10^{-11}	2×10^{-10}L	2×10^{-10}L	2×10^{-9}B
6. Trench disposal	No release situation detected								
7. Fill empty tank dome	Residual tank waste	None	Dome collapse	Tank	T 8.1c2	2×10^{-3}	6×10^{-2}L	3×10^{-2}	5×10^{-1}B
Future Tank Waste									
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	T 8.3a	4×10^{-2}	1.5×10^{0}L	4×10^{-1}	7×10^{0}B
2. Cesium removal	Liquid waste	B plant	Ion exchange fire	Resin column		Insufficient Data			
3. Cesium encapsulation	System off gas	B plant	Off-gas system failure	Filters	T 8.3a	6×10^{-6}	1×10^{-4}L	6×10^{-5}	9×10^{-4}B
4. Grout	Liquid feed	Transportable grout	Spray from feed line	Transfer line	T 8.4c3	1×10^{-10}	4×10^{-9}L	3×10^{-9}	2×10^{-8}B
5. Fill tank	Residual tank waste	None	Dome collapse	Tank	T 8.3a	3×10^{-5}	8×10^{-4}L	3×10^{-4}	5×10^{-3}B

TABLE 10.5a. (contd)

	Waste Form	Facility	Accident	Location	Reference[a] Inventory	Maximum Individual Dose,[b] rem — Maximum Inhalation — 1st Year Dose T. Body	1st Year Dose C. Organ	70-Year Dose Commitment T. Body	70-Year Dose Commitment C. Organ
Strontium and Cesium Capsules									
1. Remove from WESF[c]	Encapsulated waste	WESF	Capsule rupture	Storage area	T 8.5c3 T 8.5c7	1×10^{-7} 2×10^{-8}	2×10^{-6} L 2×10^{-8} B	1×10^{-6} 2×10^{-8}	2×10^{-5} B 2×10^{-8} B
2. Capsule packaging	Encapsulated waste	CPF[d]	Machinery impacts capsules	"Load" Section	T 8.5c3 T 8.5c7	1×10^{-8} 2×10^{-9}	3×10^{-7} L 2×10^{-9} B	2×10^{-7} 2×10^{-9}	3×10^{-6} B 3×10^{-9} B
3. Place in drywell storage	Encapsulated waste	DWSF[e]	Transporter sheare capsule	DWSF	T 8.5c3 T 8.5c7	2×10^{-4} 2×10^{-5}	3×10^{-4} L 2×10^{-5} B	2×10^{-3} 3×10^{-5}	6×10^{-3} L 3×10^{-5} B
TRU-Contaminated Soil									
1. Inject grout	TRU-contaminated soil	None	Void space collapse	CRIB	T 8.7c2	2×10^{-8}	3×10^{-6} L	9×10^{-7}	2×10^{-5} B
Pre-1970 TRU Solid Waste									
1. Grout caissons	Caisson	None	Equipment failure	Caisson	T 8.10c2	2×10^{-11}	2×10^{-9} L	6×10^{-10}	1×10^{-8} B
2. Subsidence control	Solid waste site	None	Void space collapse	Burial site	T 8.10c2	2×10^{-7}	1×10^{-5} L	5×10^{-6}	1×10^{-4} B
Retrievably Stored and Newly Generated TRU									
1. Subsidence control	Retrievable stored TRU	None	Void space collapse	Burial site	T 8.11c6	4×10^{-6}	4×10^{-4} L	9×10^{-5}	2×10^{-3} B
2. Bury packaged waste	TRU waste	None	Package breach	Burial site	T 8.11c6	2×10^{-3}	2×10^{-1} L	4×10^{-2}	8×10^{-1} B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Waste Encapsulation and Packaging Facility.
(d) Capsule Packaging Facility.
(e) Dry Well Storage Facility.

TABLE 10.5b. In-Place Stabilization and Disposal Alternative Doses from Accidental Releases for Operations Involving Six Waste Forms (Max. Ind. Ingestion)

	Waste Form	Facility	Accident	Location	Maximum Individual Dose,[b] rem — Maximum Ingestion — 1st Year Dose — T. Body	1st Year Dose — C. Organ	70-Year Dose Commitment — T. Body	70-Year Dose Commitment — C. Organ
Existing Tank Waste								
1. Dry single-shell tanks	Salt cake	Tank	Explosion	Tank	6×10^{-2}	2×10^{-1}B	1×10^{0}	5×10^{0}B
2. Fill tank dome	Salt cake	Tank	Dome collapse	Tank	4×10^{-2}	1×10^{-1}B	8×10^{-1}	3×10^{0}B
3. Hydraulic retrieval of residual liquid (double shell)	Liquid waste	Tank	Pressurized spray	Diversion valve	2×10^{-2}	6×10^{-2}B	5×10^{-1}	2×10^{0}B
4. Complexant destruction	No information available							
5. Grout	Aqueous solutions	Transportable grout	Liquid spray from line	Transfer line	2×10^{-11}	6×10^{-11}B	2×10^{-10}	7×10^{-10}B
6. Trench disposal	No release situation detected							
7. Fill empty tank dome	Residual tank waste	None	Dome collapse	Tank	2×10^{-3}	6×10^{-3}B	4×10^{-2}	1×10^{-1}B
Future Tank Waste								
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	6×10^{-2}	1×10^{-1}B	5×10^{-1}	6×10^{0}B
2. Cesium removal	Liquid waste	B plant	Ion exchange fire	Resin column	Insufficient Data			
3. Cesium encapsulation	System off gas	B plant	Off-gas system failure	Filters	8×10^{-6}	2×10^{-5}B	7×10^{-5}	2×10^{-4}B
4. Grout	Liquid feed	Transportable grout	Spray from feed line	Transfer line	5×10^{-11}	2×10^{-10}B	1×10^{-9}	4×10^{-9}B
5. Fill tank	Residual tank waste	None	Dome collapse	Tank	4×10^{-5}	8×10^{-5}B	4×10^{-4}	1×10^{-3}B

TABLE 10.5b. (contd)

	Waste Form	Facility	Accident	Location	Maximum Individual Dose,[b] rem — Maximum Ingestion — 1st Year Dose T. Body	1st Year Dose C. Organ	70-Year Dose Commitment T. Body	70-Year Dose Commitment C. Organ
Strontium and Cesium Capsules								
1. Remove from WESF[c]	Encapsulated waste	WESF	Capsule rupture	Storage area	9×10^{-8} 8×10^{-8}	3×10^{-7}B 8×10^{-8}B	2×10^{-6} 1×10^{-7}	7×10^{-6}B 1×10^{-7}B
2. Capsule packaging	Encapsulated waste	CPP[d]	Machinery impacts capsules	"Load" section	1×10^{-8} 1×10^{-8}	4×10^{-8}B 1×10^{-8}B	2×10^{-7} 2×10^{-8}	9×10^{-7} 2×10^{-8}B
3. Place in drywell storage	Encapsulated waste	DWSF[e]	Transporter sheare capsule	DWSF	1×10^{-4} 1×10^{-4}	4×10^{-4}B 1×10^{-4}B	2×10^{-3} 2×10^{-4}	9×10^{-3}B 2×10^{-4}B
TRU-Contaminated Soil								
1. Inject grout	TRU-contaminated soil	None	Void space collapse	CRIB	1×10^{-11}	2×10^{-10}B	8×10^{-10}	2×10^{-8}B
Pre-1970 TRU Solid Waste								
1. Grout caissons	Caisson	None	Equipment failure	Caisson	7×10^{-12}	3×10^{-11}B	2×10^{-10}	7×10^{-10}B
2. Subsidence control	Solid waste site	None	Void space collapse	Burial site	7×10^{-8}	3×10^{-7}B	2×10^{-6}	6×10^{-6}B
Retrievably Stored and Newly Generated TRU								
1. Subsidence control	Retrievable stored TRU	None	Void space collapse	Burial site	9×10^{-7}	1×10^{-6}B	6×10^{-6}	2×10^{-5}B
2. Bury packaged waste	TRU waste	None	Package breach	Burial site	3×10^{-4}	6×10^{-4}B	2×10^{-3}	8×10^{-3}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Waste Encapsulation and Packaging Facility.
(d) Capsule Packaging Facility.
(e) Dry Well Storage Facility.

TABLE 10.5c. In-Place Stabilization and Disposal Alternative Doses from Accidental Releases for Operations Involving Six Waste Forms (Total Population Dose)

	Waste Form	Facility	Accident	Location	Population Dose,[b] man-rem 1st Year Dose T. Body	1st Year Dose C. Organ	70-Year Dose Commitment T. Body	70-Year Dose Commitment C. Organ
Existing Tank Waste								
1. Dry single-shell tanks	Salt cake	Tank	Explosion	Tank	4×10^{2}	4×10^{3}L,B	7×10^{3}	6×10^{4}B
2. Fill tank dome	Salt cake	Tank	Dome collapse	Tank	2×10^{2}	2×10^{3}L,B	4×10^{3}	3×10^{4}B
3. Hydraulic retrieval of residual liquid (double shell)	Liquid waste	Tank	Pressurized spray	Diversion valve	1×10^{2}	2×10^{3}L	2×10^{3}	2×10^{4}B
4. Complexant destruction	No information available							
5. Grout	Aqueous solutions	Transportable grout	Liquid spray from line	Transfer line	1×10^{-7}	6×10^{-7}L,B	1×10^{-6}	8×10^{-6}B
6. Trench disposal	No release situation detected							
7. Fill empty tank dome	Residual tank waste	None	Dome collapse	Tank	9×10^{0}	1×10^{2}L	2×10^{2}	1×10^{3}B
Future Tank Waste								
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	3×10^{2}	2×10^{3}L,B	2×10^{3}	2×10^{4}B
2. Cesium removal	Liquid waste	B plant	Ion exchange fire	Resin column	Insufficient Data			
3. Cesium encapsulation	System off gas	B plant	Off-gas system failure	Filters	4×10^{-2}	3×10^{-1}L	4×10^{-1}	3×10^{0}B
4. Grout	Liquid feed	Transportable grout	Spray from feed line	Transfer line	4×10^{-7}	1×10^{-5}L	6×10^{-6}	5×10^{-5}B
5. Fill tank	Residual tank waste	None	Dome collapse	Tank	2×10^{-1}	2×10^{0}L	2×10^{0}	1×10^{1}B

TABLE 10.5c. (contd)

	Waste Form	Facility	Accident	Location	Population Dose,[b] man-rem: 1st Year Dose T. Body	1st Year Dose C. Organ	70-Year Dose Commitment T. Body	70-Year Dose Commitment C. Organ
Strontium and Cesium Capsules								
1. Remove from WESF[c]	Encapsulated waste	WESF	Capsule rupture	Storage area	6×10^{-4} 3×10^{-4}	5×10^{-3}L,B 3×10^{-4}B	1×10^{-2} 5×10^{-4}	8×10^{-2}B 6×10^{-4}B
2. Capsule packaging	Encapsulated waste	CPF[d]	Machinery impacts capsules	"Load" Section	7×10^{-5} 4×10^{-5}	7×10^{-4}L 4×10^{-5}B	1×10^{-3} 6×10^{-5}	9×10^{-3}B 7×10^{-5}B
3. Place in drywell storage	Encapsulated waste	DWSF[e]	Transporter shears capsule	DWSF	6×10^{-1} 4×10^{-1}	6×10^{0}L,B 4×10^{-1}B	1×10^{1} 6×10^{-1}	9×10^{1}B 7×10^{-1}B
TRU-Contaminated Soil								
1. Inject grout	TRU-contaminated soil	None	Void space collapse	CRIB	5×10^{-5}	5×10^{-3}L	2×10^{-3}	5×10^{-2}B
Pre-1970 TRU Solid Waste								
1. Grout caissons	Caisson	None	Equipment failure	Caisson	7×10^{-8}	3×10^{-6}L	2×10^{-6}	3×10^{-5}B
2. Subsidence control	Solid waste site	None	Void space collapse	Burial site	6×10^{-4}	3×10^{-2}L	2×10^{-2}	2×10^{-1}B
Retrievably Stored and Newly Generated TRU								
1. Subsidence control	Retrievable stored TRU	None	Void space collapse	Burial site	1×10^{-2}	9×10^{-1}L	2×10^{-1}	4×10^{0}B
2. Bury packaged waste	TRU waste	None	Package breach	Burial site	6×10^{0}	3×10^{2}L	8×10^{1}	2×10^{3}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Waste Encapsulation and Packaging Facility.
(d) Capsule Packaging Facility.
(e) Dry Well Storage Facility.

TABLE 10.6a. Reference Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Max. Ind. Inhalation)

	Waste Form	Facility	Accident	Location	Reference[a] Inventory	Maximum Individual Dose,[b] rem — Maximum Inhalation — 1st Year Dose T. Body	1st Year Dose C. Organ	Commitment T. Body	Commitment C. Organ
Existing Tank Waste									
Single-shell									
1. Dry	Salt cake	Tank	Explosion	Tank	T 8.1c2	9×10^{-2}	2×10^{0}L	1×10^{0}	2×10^{1}B
2. Fill Tank Dome	Salt cake	Tank	Dome collapse	Tank	T 8.1c2	5×10^{-2}	1×10^{0}L	6×10^{-1}	1×10^{1}B
Double-shell									
1. Hydraulic Retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	T 8.1c2	3×10^{-2}	7×10^{-1}L	4×10^{-1}	6×10^{0}B
2. Sludge washing	Sludge	Small vitrification	Filter failure	Filters	T 8.1c2	5×10^{-7}	1×10^{-5}L	6×10^{-6}	9×10^{-5}B
3. Radionuclide removal	Ion exchange	Small vitrification	Ion exchange fire	Resin column					
4. Vitrification	Molten glass	Small vitrification	Loss of filters	Filters	T 8.2c10	2×10^{-6}	5×10^{-5}L	3×10^{-5}	4×10^{-4}B
5. Grout solutions	Decontaminated	Grout	Liquid spray from lines	Transfer line	T 8.2c8	4×10^{-12}	2×10^{-11}L	2×10^{-11}L	2×10^{-10}B
6. Fill empty tank	Residual tank	None	Dome collapse	Tank	T 8.1c2	2×10^{-5}	6×10^{-4}L	3×10^{-4}	5×10^{-3}B
Double-shell suboption									
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	T 8.1c2	3×10^{-2}	7×10^{-1}L	4×10^{-1}	6×10^{0}B
2. Grout solution	Diluted waste	Grout	Liquid spray from lines	Transfer line	T 8.2c8	4×10^{-12}	2×10^{-11}L	2×10^{-11}	2×10^{-10}B
3. Fill empty tank	Residual tank waste	None	Dome collapse	Tank	T 8.1C2	2×10^{-3}	6×10^{-2}L	3×10^{-2}	5×10^{-1}B
Future Tank Waste									
1. Hydraulic retrieval	NCAW[c]	NA[d]	Pressurized release	Diversion valve	T 8.1c2	4×10^{-2}	1×10^{0}L	4×10^{-1}	7×10^{0}B
2. Cesium removal	Ion exchange	Small vitrification	Ion exchange fire	Resin column	Insufficient Data				
3. Grout solutions	NCAW supernatant	Transportable grout	Liquid spray from line	Transfer line	T 8.4c9	4×10^{-12}	3×10^{-10}L	3×10^{-11}	5×10^{-10}B
4. Vitrification	Molten glass	Small vitrification	Loss of filters	Filters	T 8.4c11	1×10^{-5}	4×10^{-4}L	1×10^{-4}	2×10^{-3}B
5. Fill empty tank	Residual tank	None	Dome collapse	Tank	T 8.3a	3×10^{-5}	8×10^{-4}L	3×10^{-4}	5×10^{-3}B

TABLE 10.6a. (contd)

	Waste Form	Facility	Accident	Location	Reference[a] Inventory	Maximum Individual Dose,[b] rem — Maximum Inhalation — 1st Year Dose — T. Body	1st Year Dose — C. Organ	50-Year Commitment — T. Body	50-Year Commitment — C. Organ
Strontium and Cesium Capsule									
1. Remove from WESF[e]	Encapsulated waste	WESF	Capsule rupture	Storage area	1 A,5C3 1 B,5C7	1×10^{-7} 2×10^{-8}	2×10^{-6}L 2×10^{-8}B	1×10^{-6} 2×10^{-8}	2×10^{-7}B 2×10^{-8}B
2. Capsule packaging	Encapsulated waste	CPF[f]	Machinery impacts capsule		1 A,5C3 1 B,5C7	1×10^{-8} 7×10^{-9}	3×10^{-7}L 2×10^{-9}B	2×10^{-7} 2×10^{-9}B	3×10^{-6}B 3×10^{-9}B
TRU-Contaminated Soil									
1. Inject grout	TRU-contaminated soil	None	Void space collapse	CRID	1 A,6C7	7×10^{-9}	3×10^{-6}L	9×10^{-7}	2×10^{-5}B
Pre-1970 Solid Waste									
1. Grout caissons	Caisson	None	Equipment failure	Caisson	1 A,1C7	7×10^{-11}	2×10^{-9}L	6×10^{-10}	1×10^{-8}B
2. Subsidence control	Solid waste	None	Void space collapse	Burial Site	1 A,1C7	7×10^{-7}	1×10^{-5}L	5×10^{-6}	1×10^{-4}B
Retrievably Stored and Newly Generated TRU									
1. Retrieve 1.1 RH[g] TRU			Pressurized release	Metal can	1 A,1B4	5×10^{-12}	5×10^{-10}L	1×10^{-10}	2×10^{-9}B
1.2 CH[h] TRU			Pressurized release	Metal can	1 A,1B1	7×10^{-9}	2×10^{-1}L	7×10^{-2}	1×10^{0}B
2. Sorting and related operations 2.1 RH TRU	Packaged waste	RH[i] WRAP	Pressurized release	Drum	1 A,1B6	1×10^{-10}	3×10^{-8}L	6×10^{-9}	1×10^{-7}B
2.2 CH TRU			Pressurized release	Drum	1 A,1B7	1×10^{-10}	1×10^{-8}L	5×10^{-9}	1×10^{-7}B
3. Process 3.1 RH TRU 3.2 CH WRAP	No information available		Fire	Drum	1 A,1B6 1 A,1B3	5×10^{-12}	6×10^{-10}L	2×10^{-10}	5×10^{-9}B

(a) C - Table, C - Column.
(b) Dose to B - Bone, B - Bone.
(c) Neutralized Current Acid Waste.
(d) Not Applicable.
(e) Waste Encapsulation and Storage Facility.
(f) Capsule Packaging Facility.
(g) Remote Handled.
(h) Contact Handled.
(i) Waste Retrieval and Packaging Facility.

TABLE 10.6b. Reference Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Max. Ind. Ingestion)

	Waste Form	Facility	Accident	Location	Maximum Individual Dose,[b] rem — Maximum Ingestion — 1st Year Dose — T. Body	1st Year Dose — C. Organ	70 Year Dose Commitment — T. Body	70 Year Dose Commitment — C. Organ
Existing Tank Waste								
Single-shell								
1. Dry	Salt cake	Tank	Explosion	Tank	6×10^{-2}	2×10^{-1}B	1×10^{0}	5×10^{0}B
2. Fill Tank Dome	Salt cake	Tank	Dome collapse	Tank	4×10^{-2}	1×10^{-1}B	8×10^{-1}	3×10^{0}B
Double-shell								
1. Hydraulic Retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	2×10^{-2}	8×10^{-2}B	5×10^{-1}	2×10^{0}B
2. Sludge washing	Sludge	Small vitrification	Filter failure	Filters	3×10^{-7}	1×10^{-6}B	7×10^{-6}	3×10^{-5}B
3. Radionuclide removal	Ion exchange	Small vitrification	Ion exchange fire	Resin column				
4. Vitrification	Molten glass	Small vitrification	Loss of filters	Filters	1×10^{-6}	6×10^{-6}B	4×10^{-5}	1×10^{-4}B
5. Grout solutions	Decontaminated	Grout	Liquid spray from lines	Transfer line	1×10^{-11}	2×10^{-11}B	4×10^{-11}	1×10^{-10}B
6. Fill empty tank	Residual tank	None	Dome collapse	Tank	2×10^{-5}	6×10^{-5}B	4×10^{-4}	1×10^{-3}B
Double-shell suboption								
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	2×10^{-2}	8×10^{-2}B	5×10^{-1}	2×10^{0}B
2. Grout solution	Diluted waste	Grout	Liquid spray from lines	Transfer line	1×10^{-11}	2×10^{-11}B	4×10^{-11}	1×10^{-10}B
3. Fill empty tank	Residual tank waste	None	Dome collapse	Tank	2×10^{-3}	6×10^{-3}B	4×10^{-2}	1×10^{-1}B
Future Tank Waste								
1. Hydraulic retrieval	NCAW[c]	NA[d]	Pressurized release	Diversion valve	6×10^{-2}	1×10^{-1}B	5×10^{-1}	6×10^{0}B
2. Cesium removal	Ion exchange	Small vitrification	Ion exchange fire	Resin column	Insufficient Data			
3. Grout solutions	NCAW supernatant	Transportable grout	Liquid spray from line	Transfer line	8×10^{-12}	1×10^{-11}B	4×10^{-11}	1×10^{-10}B
4. Vitrification	Molten glass	Small vitrification	Loss of filters	Filters	1×10^{-5}	2×10^{-5}B	1×10^{-4}	4×10^{-4}B
5. Fill empty tank	Residual tank waste	None	Dome collapse	Tank	4×10^{-5}	8×10^{-5}B	4×10^{-4}	1×10^{-3}B

TABLE 10.6b. (contd)

	Waste Form	Facility	Accident	Location	Maximum Individual Dose,[b] rem: Maximum Ingestion: 1st Year Dose: T. Body	1st Year Dose: C. Organ	70 Year Dose Commitment: T. Body	70 Year Dose Commitment: C. Organ
Strontium and Cesium Capsule								
1. Remove from WESF[e]	Encapsulated waste	WESF	Capsule rupture	Storage area	9×10^{-8} 8×10^{-8}	3×10^{-7}B 8×10^{-8}	2×10^{-6} 1×10^{-7}	7×10^{-6} 1×10^{-7}B
2. Capsule packaging	Encapsulated waste	CPF	Machinery impacts capsule		1×10^{-8} 1×10^{-8}	4×10^{-8}B 1×10^{-8}B	2×10^{-7} 2×10^{-8}	9×10^{-7}B 2×10^{-8}B
TRU-Contaminated Soil								
1. Inject grout	TRU-contaminated soil	None	Void space collapse	Okin	1×10^{-11}	2×10^{-10}B	8×10^{-10}	2×10^{-8}B
Pre-1970 Solid Waste								
1. Grout caissons	Caisson	None	Equipment failure	Caisson	7×10^{-12}	3×10^{-11}B	2×10^{-10}	7×10^{-10}B
2. Subsidence control	Solid waste	None	Void space collapse	Burial site	7×10^{-8}	3×10^{-7}B	2×10^{-6}	6×10^{-6}B
Retrievably Stored and Newly Generated TRU								
1. Retrieve								
1.1 RH[g] TRU	3.7×10^{3}		Pressurized release	Metal can	1×10^{-12}	2×10^{-12}B	7×10^{-12}	2×10^{-11}B
1.2 CH[h] TRU	2×10^{5}		Pressurized release	Metal can	7×10^{-5}	1×10^{-4}B	6×10^{-4}	3×10^{-3}B
2. Sorting and related operations								
2.1 RH-TRU	Packaged waste	RH-WRAP[i]	Pressurized release	Drum	5×10^{-11}	1×10^{-10}B	4×10^{-10}	1×10^{-9}B
2.2 CH-TRU	2×10^{5}		Pressurized release	Drum	4×10^{-12}	9×10^{-12}B	4×10^{-11}B	2×10^{-10}B
3. Process								
3.1 RH-TRU	No information available							
3.2 CH-WRAP	Package waste	CH-WRAP	Fire	Drum	2×10^{-13}	4×10^{-13}B	2×10^{-12}	9×10^{-12}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Neutralized Current Acid Waste.
(d) Not Applicable.
(e) Waste Encapsulation and Storage Facility.
(f) Capsule Packaging Facility.
(g) Remote Handled.
(h) Contact Handled.
(i) Waste Retrieval and Packaging Facility.

TABLE 10.6c. Reference Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Total Population Dose)

	Waste Form	Facility	Accident	Location	Population Dose, man-rem[b] 1st Year Dose T. Body	1st Year Dose C. Organ	70 Year Dose Commitment T. Body	70 Year Dose Commitment C. Organ
Existing Tank Waste								
Single-shell								
1. Dry	Salt cake	Tank	Explosion	Tank	4×10^{2}	4×10^{3}L,B	7×10^{3}	6×10^{4}B
2. Fill Tank Dome	Salt cake	Tank	Dome collapse	Tank	2×10^{2}	2×10^{3}L,B	4×10^{3}	3×10^{4}B
Double-shell								
1. Hydraulic Retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	1×10^{2}	2×10^{3}L	2×10^{3}	2×10^{4}
2. Sludge washing	Sludge	Small vitrification	Filter failure	Filters	2×10^{-3}	2×10^{-2}L,B	4×10^{-2}	3×10^{-1}B
3. Radionuclide removal	Ion exchange	Small vitrification	Ion exchange fire	Resin column				
4. Vitrification	Molten glass	Small vitrification	Loss of filters	Filters	1×10^{-2}	1×10^{-1}L,B	2×10^{-1}	1×10^{0}B
5. Grout solutions	Decontaminated	Grout	Liquid spray from lines	Transfer line	6×10^{-8}	1×10^{-7}B	2×10^{-7}	8×10^{-7}B
6. Fill empty tank	Residual tank	None	Dome collapse	Tank	1×10^{-1}	1×10^{0}L,B	2×10^{0}	2×10^{1}B
Double-shell suboption								
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	1×10^{2}	2×10^{3}L	2×10^{3}	2×10^{4}B
2. Grout solution	Diluted waste	Grout	Liquid spray from lines	Transfer line	6×10^{-8}	1×10^{-7}B	2×10^{-7}	8×10^{-7}B
3. Fill empty tank	Residual tank waste	None	Dome collapse	Tank	9×10^{0}	1×10^{2}L	2×10^{2}	1×10^{3}B
Future Tank Waste								
1. Hydraulic retrieval	NCAW[c]	NA[d]	Pressurized release	Diversion valve	3×10^{2}	2×10^{3}L	2×10^{3}	2×10^{4}B
2. Cesium removal	Ion exchange	Small vitrification	Ion exchange fire	Resin column	Insufficient Data			
3. Grout solutions	NCAW supernatant	Transportable grout	Liquid spray from line	Transfer line	4×10^{-8}	7×10^{-7}L	2×10^{-7}	2×10^{-6}B
4. Vitrification	Molten glass	Small vitrification	Loss of filters	Filters	7×10^{-2}	1×10^{0}L	6×10^{-1}	5×10^{0}B
5. Fill empty tank	Residual tank waste	None	Dome collapse	Tank	2×10^{-1}	2×10^{0}L	2×10^{0}	1×10^{1}B

TABLE 10.6c. (contd)

	Waste Form	Facility	Accident	Location	Population Dose, man-rem[b] 1st Year Dose T, Body	1st Year Dose C, Organ	70 Year Dose Commitment T, Body	70 Year Dose Commitment C, Organ
Strontium and Cesium Capsule								
1. Remove from WESF[e]	Encapsulated waste	WESF	Capsule rupture	Storage area	6×10^{-4} 3×10^{-4}	6×10^{-3}L,B 3×10^{-4}B	1×10^{-2} 5×10^{-4}	8×10^{-2}B 8×10^{-4}B
2. Capsule packaging	Encapsulated waste	CPF[f]	Machinery impacts capsule	Storage area	7×10^{-5} 4×10^{-5}	7×10^{-4}L 4×10^{-5}B	1×10^{-3} 6×10^{-4}	9×10^{-3}B 7×10^{-5}B
TRU-Contaminated Soil								
1. Inject grout	TRU-contaminated soil	None	Void space collapse	CRIB	6×10^{-5}	6×10^{-3}L	2×10^{-3}	6×10^{-2}B
Pre-1970 Solid Waste								
1. Grout caissons	Caisson	None	Equipment failure	Caisson	7×10^{-8}	3×10^{-6}L	2×10^{-6}	3×10^{-5}B
2. Subsidence control	Solid waste	None	Void space collapse	Burial site	6×10^{-4}	3×10^{-2}L	2×10^{-2}	2×10^{-1}B
Retrievably Stored and Newly Generated TRU								
1. Retrieve								
1.1 RH[g] TRU	3.7×10^{3}		Pressurized release	Metal can	1×10^{-8}	1×10^{-6}L	3×10^{-7}	5×10^{-6}B
1.2 CH[h] TRU	2×10^{5}		Pressurized release	Drum	4×10^{0}	4×10^{-2}L	1×10^{2}	3×10^{3}B
2. Sorting and related operations								
2.1 RH-TRU	Packaged waste	RH-WRAP[i]	Pressurized release	Drum	9×10^{-7}	6×10^{-5}L	2×10^{-5}	3×10^{-4}B
2.2 CH-TRU	2×10^{5}		Pressurized release	Drum	3×10^{-7}	3×10^{-5}L	1×10^{-5}	2×10^{-4}B
3. Process								
3.1 RH-TRU	No information available							
3.2 CH-WRAP	Package waste	CH-WRAP	Fire	Drum	1×10^{-8}	1×10^{-6}L	5×10^{-7}	1×10^{-5}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Neutralized Current Acid Waste.
(d) Not Applicable.
(e) Waste Encapsulation and Storage Facility.
(f) Capsule Packaging Facility.
(g) Remote Handled.
(h) Contact Handled.
(i) Waste Retrieval and Packaging Facility.

hydraulic retrieval of the tank wastes. This accident is also the same as described in the Geologic Disposal and In-Place Stabilization and Disposal Alternatives.

10.5.4 Radiation Doses Associated with the No Disposal Action Alternative

Thirteen separate dose calculations were performed to analyze the potential radiological impact associated with the no disposal action alternatiave. The dose estimates, listed in Tables 10.7a, b, and c, are divided according to the maximally exposed individual inhalation dose, the maximally exposed individual ingestion dose, and the population dose. For each of these three categories, the first year total body and critical organ doses, and the 70-year total body and critical organ dose commitments were calculated. The accident resulting in the greatest public dose was the pressurized release of liquid waste due to failure of a diversion valve during hydraulic retrieval operations. This upperbound accident is the same as described for the existing double-shell and future tank wastes in the geologic disposal alternative. The accident with the second-highest radiological consequences is the collapse of void space over the retrievably stored and newly generated TRU wastes.

10.5.5 Interpretation of the Radiological Impact from the Waste Disposal Alternatives

The highest total body dose to a maximally exposed individual from any of the waste disposal alternatives was calculated to be 0.2 rem in the first year and 3 rem over a 70-yr period. This dose is based on a summation of the ingestion and inhalation pathways, which were calculated separately in order to maximize the dose estimate (and to provide an upper-bound estimate of potential dose). The dose received by an actual individual in the event of such an accident would, in all likelihood, be much lower. The annual, or first-year, dose of 0.2 rem is below the DOE guideline of 0.5 rem/yr to a member of the population from occasional[a] releases at federal facilities. It is also equivalent

(a) Vaugh, W. A. 1985. "Radiation Standards for Protection of the Public in the Vicinity of DOE Facilities." Department of Energy memorandum, August 5, 1985.

to approximately twice the annual average background radiation dose received by a resident of the Tri-Cities from naturally occurring sources of radiation, such as cosmic rays (Price et al. 1984). The accident resulting in the greatest public dose was the explosion of the single-shell tank wastes during retrieval or handling operations. It has been postulated that a layer containing ferro- or ferricyanide precipitates might be present in the single-shell tank wastes (Rockwell 1980). Under certain conditions, this material could react explosively with nitrates present in the waste. If ferrocyanide precipitates are present, the potential for an explosion does exist. However, the presence of this material in quantities sufficient to produce a large explosion is still a subject of some debate (Rockwell 1980).

The Federal Government does not currently set limits for the maximum dose that can be received by a population as a whole. Rather, it specifies an individual limit based on an average dose to a suitable sample of the exposed population; this limit is one third the amount allowed for the individual at the point of maximum probable exposure. Consequently, one cannot compare the population dose to a specific DOE limit. However, it is possible to compare the estimated accidental dose to that which is routinely received by the same group of individuals from natural sources of radiation. Approximately 140,000 persons were presumed to be exposed from the postulated upper-bound accidental releases. Their first-year dose was estimated to be 500 man-rem. This same group of individuals receives approximately 100 mrem apiece each year from natural sources of radiation; this calculates to be 1.4 10^4 man-rem, or nearly thirty times the amount they might receive in the event of an accident during processing of the wastes for disposal.

TABLE 10.7a. No Disposal Action Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Max. Ind. Inhalation)

	Waste Form	Facility	Accident	Location	Reference[a] Inventory	Maximum Individual Dose,[b], rem Maximum Inhalation 1st Year Dose T. Body	1st Year Dose C. Organ	70-Year Dose Commitment T. Body	70-Year Dose Commitment C. Organ
Existing Tank Waste									
1. Hydraulic retrieval of residual liquid (double shell)	Liquid waste	Tank	Pressurized spray	Diversion valve	T 8.1c2	3×10^{-2}	7×10^{-1}L	4×10^{-1}	6×10^{0}B
2. Fill empty tank dome	Residual tank waste	None	Dome collapse	Tank	T 8.1c2	2×10^{-3}	6×10^{-2}L	3×10^{-2}	5×10^{-1}B
Future Tank Waste									
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	T 8.3a	4×10^{-2}	1.5×10^{0}L	4×10^{-1}	7×10^{0}B
2. Fill tank	Residual tank waste	None	Dome collapse	Tank	T 8.3a	3×10^{-5}	8×10^{-4}L	3×10^{-4}	5×10^{-3}B
Strontium and Cesium Capsules									
1. Remove from WESF[c]	Encapsulated waste	WESF	Capsule rupture	Storage area	T 8.5c3 T 8.5c7	1×10^{-7} 2×10^{-8}	2×10^{-6}L 2×10^{-8}B	1×10^{-6} 2×10^{-8}	2×10^{-5}B 2×10^{-8}B
2. Capsule packaging	Encapsulated waste	CPF[d]	Machinery impacts capsules	"Load" Section	T 8.5c3 T 8.5c7	1×10^{-8} 2×10^{-9}	3×10^{-7}L 2×10^{-9}B	2×10^{-7} 2×10^{-9}	3×10^{-6}B 3×10^{-9}B
3. Place in drywell storage	Encapsulated waste	DWSF[e]	Transporter shear capsule	DWSF	T 8.5c3 T 8.5c7	2×10^{-4} 2×10^{-5}	3×10^{-4}L 2×10^{-5}B	2×10^{-3} 3×10^{-5}	6×10^{-3}L 3×10^{-5}B
TRU-Contaminated Soil									
1. Subsidence	TRU-contaminated soil	None	Void space collapse	CRIB	T 8.7c2	2×10^{-8}	3×10^{-6}L	9×10^{-7}	2×10^{-5}B
Pre-1970 TRU Solid Waste									
1. Subsidence	Solid waste site	None	Void space collapse	Burial site	T 8.10c2	2×10^{-7}	1×10^{-5}L	5×10^{-6}	1×10^{-4}B
Retrievably Stored and Newly Generated TRU									
1. Subsidence	Retrievable stored TRU	None	Void space collapse	Burial site	T 8.11c6	4×10^{-6}	4×10^{-4}L	9×10^{-5}	2×10^{-3}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Waste Encapsulation and Packaging Facility.
(d) Capsule Packaging Facility.
(e) Dry Well Storage Facility.

TABLE 10.7b. No Disposal Action Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Max. Ind. Ingestion)

	Waste Form	Facility	Accident	Location	Maximum Individual Dose,[b] rem — Maximum Ingestion — 1st Year Dose T. Body	1st Year Dose C. Organ	70-Year Dose Commitment T. Body	70-Year Dose Commitment C. Organ
Existing Tank Waste								
1. Hydraulic retrieval of residual liquid (double shell)	Liquid waste	Tank	Pressurized spray	Diversion valve	2×10^{-2}	8×10^{-2}B	5×10^{-1}	2×10^{0}B
2. Fill empty tank dome	Residual tank waste	None	Dome collapse	Tank	2×10^{-3}	6×10^{-3}B	4×10^{-2}	1×10^{-1}B
Future Tank Waste								
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	6×10^{-2}	1×10^{-1}B	5×10^{-1}	6×10^{0}B
2. Fill tank	Residual tank waste	None	Dome collapse	Tank	4×10^{-6}	8×10^{-5}B	4×10^{-4}	1×10^{-3}B
Strontium and Cesium Capsules								
1. Remove from WESF[c]	Encapsulated waste	WESF	Capsule rupture	Storage area	9×10^{-8} 8×10^{-8}	3×10^{-7}B 8×10^{-8}B	2×10^{-6} 1×10^{-7}	7×10^{-6}B 1×10^{-7}B
2. Capsule packaging	Encapsulated waste	CPF[d]	Machinery impacts capsules	"Load" section	1×10^{-8} 1×10^{-8}	4×10^{-8}B 1×10^{-8}B	2×10^{-7} 2×10^{-8}	9×10^{-7} 2×10^{-8}B
3. Place in drywell storage	Encapsulated waste	DWSF[e]	Transporter sheare capsule	DWSF	1×10^{-4} 1×10^{-4}	4×10^{-4}B 1×10^{-4}B	2×10^{-3} 2×10^{-4}	9×10^{-3}B 2×10^{-4}B
TRU-Contaminated Soil								
1. Subsidence	TRU-contaminated soil	None	Void space collapse	CRIB	1×10^{-11}	2×10^{-10}B	8×10^{-10}	2×10^{-8}B
Pre-1970 TRU Solid Waste								
1. Subsidence	Solid waste site	None	Void space collapse	Burial site	7×10^{-8}	3×10^{-7}B	2×10^{-6}	6×10^{-6}B
Retrievably Stored and Newly Generated TRU								
1. Subsidence	TRU	None	Void space collapse	Burial site	9×10^{-7}	1×10^{-6}B	6×10^{-6}	2×10^{-5}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Waste Encapsulation and Packaging Facility.
(d) Capsule Packaging Facility.
(e) Dry Well Storage Facility.

TABLE 10.7c. No Disposal Action Alternative Potential Doses from Accidental Releases for Operations Involving Six Waste Forms (Total Population Dose)

	Waste Form	Facility	Accident	Location	Population Dose,[b] man-rem 1st Year Dose T. Body	1st Year Dose C. Organ	70-Year Dose Commitment T. Body	70-Year Dose Commitment C. Organ
Existing Tank Waste								
1. Hydraulic retrieval of residual liquid (double shell)	Liquid waste	Tank	Pressurized spray	Diversion valve	1×10^{2}	2×10^{3}L	2×10^{3}	2×10^{4}B
2. Fill empty tank dome	Residual tank waste	None	Dome collapse	Tank	9×10^{0}	1×10^{2}L	2×10^{2}	1×10^{3}B
Future Tank Waste								
1. Hydraulic retrieval	Liquid waste	Tank	Pressurized spray	Diversion valve	3×10^{2}	2×10^{3}L,B	2×10^{3}	2×10^{4}B
2. Fill tank	Residual tank waste	None	Dome collapse	Tank	2×10^{-1}	2×10^{0}L	2×10^{0}	1×10^{1}B
Strontium and Cesium Capsules								
1. Remove from WESF[c]	Encapsulated waste	WESF	Capsule rupture	Storage area	6×10^{-4} 3×10^{-4}	5×10^{-3}L,B 3×10^{-4}B	1×10^{-2} 5×10^{-4}	8×10^{-2}B 6×10^{-4}B
2. Capsule packaging	Encapsulated waste	CPF[d]	Machinery impacts capsules	"Load" Section	7×10^{-5} 4×10^{-5}	7×10^{-4}L 4×10^{-5}B	1×10^{-3} 6×10^{-5}	9×10^{-3}B 7×10^{-5}B
TRU-Contaminated Soil								
1. Subsidence	TRU-contaminated soil	None	Void space collapse	CRIB	5×10^{-5}	6×10^{-3}L	2×10^{-3}	5×10^{-2}B
Pre-1970 TRU Solid Waste								
1. Subsidence	Solid waste site	None	Void space collapse	Burial site	6×10^{-4}	3×10^{-2}L	2×10^{-2}	2×10^{-1}B
Retrievably Stored and Newly Generated TRU								
1. Subsidence	TRU	None	Void space collapse	Burial site	1×10^{-2}	9×10^{-1}L	2×10^{-1}	4×10^{0}B

(a) T = Table, C = Column.
(b) Dose to L = Lung, B = Bone.
(c) Waste Encapsulation and Packaging Facility.
(d) Capsule Packaging Facility.
(e) Dry Well Storage Facility.

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