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High-Level Waste Inventory, Characteristics, Generation, and Facility Assessment for Treatment, Storage, and Disposal Alternatives Considered in the U.S. Department of Energy Environmental Management Programmatic Environmental Impact Statement

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The following is a list of acronyms and abbreviations (including units of measure) used in this document.

ACRONYMS, INITIALISMS, AND ABBREVIATIONS

Ac	actinium
Ag	silver
Am	americium
ANL	Argonne National Laboratory
At	astatine
Ba	barium
BDAT	best demonstrated available technology
Bi	bismuth
BOY	beginning of year
BWR	boiling-water reactor
CC	complexant concentrate
CaF_2	calcium fluoride
Cd	cadmium
Ce	cerium
CFMUT	concentrator feed makeup tank
CH	contact-handled
CISV	canister interim storage vault
Cm	curium
Co	cobalt
CPC	chemical process cell
Cr	chromium
Cs	cesium
CSB	Canister Storage Building
CSS	Cement Solidification System
CSSF	Calcine Solids Storage Facility
CSTPB	cesium tetraphenyl borate
CTS	component test stand
CWC	Central Waste Complex
CY	calendar year
D&D	decontamination and decommissioning
DF	decontamination factor
DOE	U.S. Department of Energy
DST	double-shell tank
DWPF	Defense Waste Processing Facility

EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
Eu	europium
Fe	iron
Fr	francium
FSV	Fort St. Vrain
FTE	full-time equivalent
FY	fiscal year
GAO	U.S. General Accounting Office
Gd	gadolinium
GWSB	Glass Waste Storage Building
HEPA	high efficiency particulate air
HIP	hot isostatic pressing
HLW	high-level waste
HWVP	Hanford Waste Vitrification Plant
I	iodine
IAEA	International Atomic Energy Agency
ICPP	Idaho Chemical Processing Plant
IDB	Integrated Data Base
IDO	Idaho Operations Office
IFSF	Irradiated Fuel Storage Facility
In	indium
INC	Idaho Nuclear Corporation
INEL	Idaho National Engineering Laboratory
IRTS	Integrated Radwaste Treatment System
ISF	Interim Storage Facility
ISFSI	independent spent fuel storage installation
IWIF	Idaho Waste Immobilization Facility
K	potassium
KTPB	potassium tetraphenyl borate
La	lanthanum
LCC	life-cycle costs
LLW	low-level waste
LRWHF	Liquid Radioactive Waste Handling Facility
LWTS	Liquid Waste Treatment System
MCSB	modified canister storage building
MFHT	melter feed hold tank
MPSC	Multipurpose Processing and Storage Complex

MRS	monitored retrievable storage
MVDS	modular vault dry storage
$NaNO_2$	sodium nitrate
NaOH	sodium hydroxide
NaTi	sodium titinate
NaTPB	sodium tetraphenyl borate
Nb	niobium
NCAW	neutralized current acid waste
NCRW	neutralized cladding removal waste
Nd	neodymium
NEPA	National Environmental Protection Act
Ni	nickel
NO_x	nitrogen oxide
Np	neptunium
NRC	U.S. Nuclear Regulatory Commission
NWCF	New Waste Calcining Facility
O&M	operation and maintenance
ORNL	Oak Ridge National Laboratory
Pa	protactinium
Pd	palladium
PEIS	programmatic environmental impact statement
PFP	Plutonium Finishing Plant
Pm	promethium
PNL	Pacific Northwest Laboratory
Pr	praseodymium
PSCo	Public Service Company of Colorado
Pu	plutonium
PUREX	plutonium/uranium extraction
PWR	pressurized water reactor
Ra	radium
RCRA	Resource Conservation and Recovery Act
REDOX	reduction/oxidation
Rh	rhodium
RH	remote-handled
Rn	radon
ROD	Record of Decision
Ru	ruthenium
Sb	antinomy
SCT	shielded canister transporter
SFCM	slurry fed ceramic melter
SGCSB	second glass canister storage building

Sm	samarium
SMWS	Sludge Mobilization and Waste System
Sn	tin
SNF	spent nuclear fuel
SO_2	sulfur dioxide
SO_x	sulfur oxide
Sr	strontium
SRS	Savannah River site
SST	single-shell tank
STS	Supernatant Treatment System
Tb	terbium
Tc	technetium
TCLP	toxicity characteristic leaching procedure
Te	tellurium
Th	thorium
Ti	titanium
THOREX	thorium extraction
TPA	Tri-Party Agreement
TRUW	transuranic waste
TWRS	Tank Waste Remediation System
U	uranium
USDC	U.S. District Court
VF	Vitrification Facility
VOC	volatile organic compound
WAC	waste acceptance criteria
WCF	Waste Calcining Facility
WCSF	Waste Canister Storage Facility
WESF	Waste Encapsulation and Storage Facility
WM	Waste Management
WNYNSC	Western New York Nuclear Services Center
WSRC	Westinghouse Savannah River Company
WVDP	West Valley Demonstration Project
Y	yttrium
Zr	zirconium

UNITS OF MEASURE

Bq	becquerel(s)
°C	degree(s) Celsius
Ci	curie(s)
cm	centimeter(s)
cm^3	cubic centimeter(s)
d	day(s)
°F	degree(s) Fahrenheit
ft	foot(feet)
ft^3	cubic foot(feet)
g	gram(s)
gal	gallon(s)
GWh	gigawatt hour(s)
h	hour
ha	hectare(s)
in.	inch(es)
kg	kilogram(s)
km	kilometer(s)
kW	kilowatt(s)
L	liter(s)
lb	pound(s)
m	meter(s)
m^2	square meter(s)
m^3	cubic meter(s)
man-yr	man-year(s)
mm	millimeter(s)
mrem	millirem(s)
MT	metric ton(s)
MTHM	metric ton(s) of heavy metal
MTU	metric ton(s) of uranium
MW	megawatt(s)
rad	radiation absorbed dose
rem	roentgen equivalent man
s	second(s)
t	ton(s) (metric)
μm	micrometer(s)
W	watt(s)
worker-yr	worker-year(s)
wt%	weight percent
yr	year(s)

HIGH-LEVEL WASTE INVENTORY, CHARACTERISTICS, GENERATION, AND FACILITY ASSESSMENT FOR TREATMENT, STORAGE, AND DISPOSAL ALTERNATIVES CONSIDERED IN THE U.S. DEPARTMENT OF ENERGY WASTE MANAGEMENT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT

by

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ABSTRACT

This report provides data and information needed to support the risk and impact assessments of high-level waste (HLW) management alternatives in the U.S. Department of Energy Waste Management (WM) Programmatic Environmental Impact Statement (PEIS). Available data on the physical form, chemical and isotopic composition, storage locations, and other waste characteristics of interest are presented. High-level waste management follows six implementation phases: current storage, retrieval, pretreatment, treatment, interim canister storage, and geologic repository disposal; pretreatment, treatment, and repository disposal are outside the scope of the WM PEIS. Brief descriptions of current and planned HLW management facilities are provided, including information on the type of waste managed in the facility, costs, product form, resource requirements, emissions, and current and future status. Data sources and technical and regulatory assumptions are identified. The range of HLW management alternatives (including decentralized, regionalized, and centralized approaches) is described. The required waste management facilities include expanded interim storage facilities under the various alternatives. Resource requirements for construction (e.g., land and materials) and operation (e.g., energy and process chemicals), work force, costs, effluents, design capacities, and emissions are presented for each alternative.

1 INTRODUCTION

The two primary sources of high-level radioactive wastes (HLWs) in the United States are (1) defense wastes generated from the reprocessing of spent nuclear fuel (SNF) and weapons production targets and (2) commercial wastes generated from the power reactor fuel cycle. Spent

nuclear fuel was reprocessed for defense purposes at three sites: the Savannah River site (SRS), the Idaho National Engineering Laboratory (INEL), and the Hanford site. Spent nuclear fuel was commercially reprocessed at West Valley, New York.

Canisters of HLW immobilized in borosilicate glass or glass-ceramic mixtures will be produced at Hanford, SRS, West Valley, and INEL for ultimate disposal at a geologic repository. The existing and planned vitrification plants at SRS, Hanford, and West Valley are the Defense Waste Processing Facility (DWPF) (under testing), the Hanford Waste Vitrification Plant (HWVP) (planned), and the West Valley Demonstration Project (WVDP) (existing), respectively. At INEL, conversion of liquid HLW to a solid calcine form is currently underway at the New Waste Calcining Facility (NWCF). Section 1 of this report provides data on the potential immobilized waste form, interim storage requirements, and transportation and disposal issues.

1.1 BACKGROUND AND DEFINITION OF WASTE TYPE

As defined by the Nuclear Waste Policy Act, HLW is "(1) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including the liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations and (2) other highly radioactive material that the Nuclear Regulatory Commission (NRC), consistent with existing law, determines by rule to require permanent isolation" (U.S. Congress 1983).

U.S. Department of Energy (DOE) Order 5820.2A (DOE 1988b) requires proper handling and storage of HLW. It also requires each generator of HLW to develop a technology for permanent disposal of HLW in a geologic repository, when one becomes available. High-level waste is currently stored in an interim fashion in underground tanks at the Hanford Site, SRS, WVDP, and INEL. An evaluation of various HLW treatment technologies resulted in the selection of vitrification as the technology best suited for treating characteristics-mixed HLW (i.e., wastes exhibiting the characteristics of corrosivity and/or toxicity) for permanent disposal. The U.S. Environmental Protection Agency (EPA) considers this type of HLW treatment as a best demonstrated available technology (BDAT) under the Resource Conservation and Recovery Act (RCRA), as amended (40 CFR 268.42). The DOE approach to ending interim storage of HLW is to immobilize that part of the waste that is highly radioactive in a more stable glass form by using high-temperature vitrification to produce glass logs that are sealed in canisters. A glass made of boron and silicon (i.e., borosilicate glass) was chosen as the protective material for HLW immobilization because of (1) its long-term stability; (2) its resistance to the stresses of disposal in a repository; (3) its capability to withstand leaching under conditions that could potentially exist in a repository; and (4) its suitability for large-scale, remote operations with highly radioactive waste.

Two of the HLW sites (WVDP and SRS) plan to use cylindrical stainless steel canisters, 61 cm (24 in.) in diameter and 300 cm (118 in.) long, filled with borosilicate glass to about 85% of the canister volume. The canister design for the WVDP design has a smaller wall thickness and a wider fill neck than that proposed for SRS. On the basis of the current design, the canisters will be fabricated from 304L stainless steel.

Neither the canister dimensions nor the waste form has been decided for Hanford and INEL. The sizing of the HLW canisters to be produced at the Hanford site is in the conceptual stage and has not been finalized (DOE 1996a). For consistency, this analysis applies the dimensions of the Hanford reference canister for HLW disposal given in ORNL (1995): a diameter of 68 cm (26.8 in) and a length of 457 cm (180 in) with a nominal glass volume of 1.26 m^3 (44.5 ft^3) and an average waste oxide loading of 45%. It should be noted, however, that alternative canister designs with waste glass oxide volumes of 0.62 m^3 (22 ft^3) and 10 m^3 (360 ft^3) are being proposed in the Tank Waste Remediation System (TWRS) EIS (DOE 1996a). The mass of HLW waste oxide glass is estimated to be approximately 4,200 kg per canister, based on a canister volume of 1.26 m^3 and an assumed glass density of 3,300 kg/m^3.

In June 1995, DOE selected a preliminary waste form and a process technology to treat the sodium-bearing waste and immobilized calcine at INEL. The current reference waste form at INEL is a glass, with each canister assumed to nominally contain 0.92 m^3 (32.5 ft^3) of waste glass (ORNL 1995). The dimensions for the proposed INEL canister are not available but may be expected to be bounded by those at the other three HLW sites. The mass of HLW waste oxide glass is estimated in this study to be approximately 2,400 kg per canister, based on a canister volume of 0.92 m^3 and a glass density of 2,600 kg/m^3 (WINCO 1994).

Table 1.1 summarizes the canister dimensions, weights, and radioactivity for the four HLW sites. The values for radioactivity and decay heat are estimates of the maximum values at the time of filling. The values in Table 1.1 may change, depending on future changes in HLW management plans (potential changes for the various sites are noted in the section for each site).

1.2 STORAGE TECHNOLOGIES

Under the DOE approach to long-term HLW management, liquid HLW in current storage will be vitrified, placed in canisters, and stored in an interim on-site storage facility awaiting transport to a geologic repository. Canisters produced at WVDP would be placed in storage racks that hold four canisters each (Peters and Leap 1987). The canisters would then be transported in the racks to the on-site Waste Canister Storage Facility (WCSF). The immobilized HLW would be temporarily stored in a previously decontaminated and refurnished process cell known as the chemical process cell (CPC), which has been modified for HLW interim storage. The racks containing the HLW canisters would be stored on two levels because this configuration provides a

TABLE 1.1 Dimensions, Weights, and Radioactivity of HLW Canisters

	WVDP	SRS	Hanford	INEL
Outer diameter (cm)	61	61	68	NA[a]
Overall height (cm)	300	300	457	NA
Construction material	304L stainless steel	304L stainless steel	304L stainless steel	304L stainless steel
Nominal wall thickness (cm)	0.34	0.95	NA	NA
Weight (kg)				
Canister	252	500	NA	NA
Glass or ceramic	1,900	1,682	4,200	2,400
Total	2,152	2,182	NA	NA
Radioactivity per canister (Ci)	104,300	234,400	12,700	9,000
Decay heat per canister (W)	311	709	38	26

[a] NA = not available.

storage area for failed equipment. The canister storage area would be equipped with two coolers to remove the heat generated by the vitrified HLW. Storage capacity is available for 344 HLW canisters.

The interim canister storage facility at SRS for the vitrified HLW is designed to hold canisters in vertically sealed cavities within a concrete structure that forms the storage vault (see the Appendix for further information on this storage technology). The Glass Waste Storage Building (GWSB) at SRS will be an air-cooled, dry storage vault for vitrified HLW. Exhaust fans will be used to create a slight negative pressure for cooling, thereby removing heat generated by the stored HLW. A dedicated shielded canister transporter will be used to move filled canisters from the vitrification building to the GWSB. The transporter removes the concrete plug from the top of the canister storage vault, inserts the canister, and replaces the plug. The transporter maintains the canister in a vertical position during transport and emplacement, and is lowered by gravity into the plug hole. No inspection is performed. Activities at the GWSB includes the receipt and unloading of transportation casks containing canisters of vitrified waste, and storage of the waste until transfer to a geologic

repository. The GWSB will consist of rows of tubes placed below grade into which the canisters will be lowered. The canisters will not be stacked within the storage tubes, and concrete plugs will provide a cover for the tubes. The canisters will be stored in sealed sleeves so that the cooling air will not directly come into contact with the potentially contaminated surfaces of the HLW canisters. Storage capacity is currently provided for 2,286 canisters; this capacity accommodates approximately five years of vitrification operations at the DWPF. The storage capacity of the existing facility was predicated on the assumption that a geologic repository would be available by 1992, the time at which fresh waste would have been processed. Additional storage capacity for 2,286 HLW canisters (to be made available in the year 2007) will be required if the opening of the geologic repository is delayed to FY 2015.

Canister estimates for Hanford and INEL are less certain because pretreatment and waste minimization processes have not yet been finalized. The construction of the Hanford Waste Vitrification Plant was delayed because of insufficient capacity to vitrify the HLW fraction of all Hanford double-shell tank (DST) and single-shell tank (SST) waste in the planned time frame. It is estimated that approximately 2,000 standard-sized (0.62 m^3) canisters containing glass logs will be produced from the HLW currently stored within the Hanford DSTs. The number of canisters from SST wastes is currently unknown because the pretreatment process has not been selected; a maximum of 70,000 canisters has been projected for minimal pretreatment (DOE 1996a). The estimated number of HLW canisters to be produced from vitrification of the Hanford Site tank waste depends on site-specific decisions regarding pretreatment and disposal reached under the TWRS Environmental Impact Statement (EIS) currently under development (DOE 1996a). For the purposes of this study, it is assumed that an estimated 15,000 canisters will be produced from treating existing Hanford HLW. The vitrified HLW canisters at Hanford will be placed in an on-site interim storage facility, where they would be stored awaiting shipment and disposal at the national geologic repository. The interim storage method at Hanford is assumed in this analysis to be similar to that employed at SRS.

The only other HLW forms produced at Hanford that may require treatment and storage are strontium (Sr) and cesium (Cs) capsules. These capsules were produced over the years when the high-heat-emitting isotopes (Sr-90 and Cs-137, plus their progeny) were removed from the old waste, converted to solids (strontium fluoride and cesium chloride), placed in double-walled capsules, and stored in a water basin. These isotopes are currently contained in seal-welded, high-integrity, double-encapsulated stainless steel or Hastelloy capsules (DOE 1987). Each capsule has an external diameter of approximately 6.7 cm (2.6 in.) and an overall length of about 53 cm (21 in.). Six hundred five Sr-90 capsules and 1,338 Cs-137 capsules are available for disposal (ORNL 1995). The cesium and strontium capsules currently are classified as waste by-product and have potential commercial value as irradiation or heat sources. Additional measures will be required to remediate the capsules if and when they are reclassified as waste. The cesium and strontium capsules may not meet the current waste acceptance criteria of the national geologic repository because the waste is in a corrosive form (cesium and strontium salts). It is possible that these capsules may be opened at

Hanford and their contents combined with other feeds to vitrification. DOE policy is to dispose of readily retrievable HLW in a national geologic repository, and therefore it would appear that the strontium and cesium will go to the repository; whether these wastes will go as capsules enclosed in overpacks or in vitrified form has not been decided (DOE 1996a). The reference case of the Hanford TWRS is to overpack the capsules (Wodrich 1992), which would comply with the Record of Decision (ROD) for the disposal of Hanford HLW (DOE 1988a). In this report, it is assumed that the cesium and strontium capsules will be packaged in 0.3-m (1-ft)-diameter, 2.7-m (8.5-ft)-tall overpack canisters in accordance with repository waste acceptance criteria and will not require vitrification; a total of approximately 320 overpack canisters will require interim storage.

It is assumed that interim storage of the overpack canisters will be at the Multipurpose Processing and Storage Complex (MPSC) (Wodrich 1992; Nyman et al. 1993). The MPSC is designed to provide interim dry storage of special-case wastes and will be located in the vicinity of the Central Waste Complex (CWC) in the 200 West Area at Hanford. The overpack canisters could conceivably be stored on an interim basis by using one or more of the following dry storage systems: metal casks, concrete casks, horizontal concrete modules, dual-purpose casks, and modular concrete vaults. Because the cesium/strontium canisters will presumably be shipped to a geologic repository for final disposal if and when they are reclassified as waste, the dry storage technology will most likely be similar to that utilized for the glass canisters (i.e., modular concrete vaults).

High-level waste generated during reprocessing at INEL initially took the form of an acidic liquid. In 1963, INEL began to calcine this waste into a dry, granular solid that was then stored in 4-m (13-ft)-diameter, 14-m (46-ft)-tall stainless steel bins housed in reinforced concrete vaults with an expected lifetime of at least 500 years. Of the four DOE sites containing HLW, INEL HLW will be processed last because it is in a fairly stable solid form. Full-scale immobilization is projected to start by FY 2015 (DOE 1995c). It is assumed that on-site storage will be required for the entire inventory of HLW glass canisters produced at INEL, pending transport off site and disposal in a geologic repository.

The method of interim storage at INEL has not been determined, but it is indicated in WINCO (1994) to be similar to the enclosed vault storage concept outlined in Feizollahi and Shropshire (1993). The technical characteristics of the enclosed vault storage concept appear to be analogous to the modular vault storage concept applied at SRS, in that waste packages are stored below-grade in storage vaults, and are loaded and retrieved remotely by an overhead bridge crane. In this analysis, it is assumed that the HLW glass canisters at INEL are stored in modular vaults.

Table 1.2 gives a comparison of the interim storage facilities.

TABLE 1.2 Interim Storage Facilities for HLW Canisters

			Hanford		
	WVDP	SRS	DST and SST Wastes	Strontium/Cesium Capsules	INEL
Facility	CPC	GWSB	TBD[a]	MPSC	TBD[a]
Storage capacity (HLW canisters)	344	12,286[b]	15,000	320	1,700
Storage method	Process cell	Modular concrete vault	Modular concrete vault	Modular concrete vault[c]	Modular concrete vault[c]
Footprint (m^2)	190	4,343	12,200	≈1,500[d]	3,200[d]
Vault volume (m^3)	2,490	63,404	141,000	≈18,000	47,000
Cooling method	Air cooler	Exhaust fans	Natural convection	Natural convection[c]	Natural convection[c]

[a] To be determined. Facility not yet constructed; conceptual design being evaluated.

[b] Additional storage capacity of over 2,286 canisters will also be required if the repository opening is delayed until FY 2015.

[c] Assumed; actual dry storage technology is to be determined (Nyman et al. 1993).

[d] Determined by correlation, assuming single canister stacking within a storage tube (due to high heat generation rate).

2 INVENTORY AND GENERATION VOLUMES

The following sections describe the current inventory of HLW at the four sites that reprocessed SNF—Hanford, INEL, SRS, and WVDP. Projections of the quantities of HLW to be produced at each site in upcoming years are provided, as are projections of the number of canisters of treated HLW to be produced at each site. The principal assumptions used in making these projections are noted.

2.1 ASSUMPTIONS

DOE decided in 1992 to phase out operations involving the reprocessing of SNF for the recovery of enriched uranium or plutonium in support of national defense activities. Thus, the quantity of HLW to be treated, stored, and disposed of within DOE Waste Management (WM) programs will consist primarily of the material already in storage at each of the four sites plus waste generated during decontamination and decommissioning (D&D) activities with radioactivity levels high enough to warrant management similar to HLW. Very little additional HLW is expected to be generated as a result of environmental restoration activities (DOE 1993b) and, therefore, such waste is not considered in this analysis.

It is assumed that only SRS will add to the current HLW inventory (about 15% of the total SRS HLW inventory in FY 2015). The F- and H-Canyon stabilization and decommissioning projects may be expected to average about 1,000 m^3/yr of material assumed to be managed as HLW.

The SRS WM EIS (DOE 1995a) provides estimates of the minimum and maximum amounts of wastes that may be generated in the future, to account for wastes that may be generated during interim management of nuclear materials. This analysis applies the inventory from the expected waste forecast.

Radioactive, sodium-containing liquid waste produced at INEL from decontamination and solvent recovery operations is stored in the Idaho Chemical Processing Plant (ICPP) tank farm and is managed as HLW. Currently, the most effective method of processing the sodium-containing waste consists of calcining it as a blend with fluorine waste generated from fuel reprocessing operations. The ICPP is required to continue to calcine liquid wastes in accordance with the Notice of Noncompliance Consent Order, which stipulates that the removal of wastes from existing tank farms should begin by the late 1990s.

HLW is not projected to be generated at either WVDP or Hanford. Projections for future HLW inventories at Hanford are based on the assumptions that spent fuel reprocessing at the PUREX Plant will not resume and that irradiated N-Reactor fuel at Hanford will remain in storage

without further processing that may result in the generation of HLW. Newly generated waste at Hanford would typically consist of liquid low-activity waste from several facilities at Hanford, a fraction (about one-third) of which was produced by facility transition activities. Additionally, approximately 1,930 cesium and strontium capsules are currently classified as waste by-product. One alternative in the TWRS EIS (DOE 1996a) considers vitrifying the cesium and strontium capsules with the Hanford tank waste after reclassifying them as HLW. This analysis assumes the cesium and strontium capsules to be waste by-product, but does examine the consequences of their potential reclassification as HLW.

The waste in underground tank storage at the Hanford site consists of mixed hazardous HLW, mixed hazardous TRUW, and mixed hazardous LLW, but is managed as HLW at Hanford. It is assumed that tank wastes that could be classified as TRUW will be blended with other HLW streams and vitrified for disposal at the national geological repository.

2.2 VOLUMES BY SITE

The radioactive waste inventory indicates that most of the wastes stored have already undergone one or more treatment processes, such as neutralization or precipitation, and that such wastes are not in the same form as when they were initially generated or characterized.

The total volume of HLW at all sites at the end of CY 1994 was about 379,000 m^3 (100 million gal) with a total radioactivity of about 1×10^9 Ci. The DOE began vitrification of some of the current inventory in 1996; therefore, the net projected total cumulative volume of stored HLW at the end of 2000 is projected to be 341,000 m^3 (90 million gal) at a total radioactivity of approximately 7×10^8 Ci.

2.2.1 Hanford

The HLW currently stored at the Hanford Site is a result of fuel reprocessing operations carried out since 1943. The waste, abundant in fission products and transuranics, is currently stored in 177 underground storage tanks built between 1943 and 1986. The 177 tanks are comprised of 149 older SSTs and 28 newer DSTs. The SSTs, built between 1943 and 1964, are approaching their operating lifetime of 50 years, and as a result of safety concerns, HLW has been transferred from SSTs to DSTs since the 1960s.

At the end of 1991, approximately 238,900 m^3 (63.1 million gal) of HLW was stored at Hanford. This volume represents the largest DOE inventory at a single site and is about 63% of the total DOE HLW. The radioactivity level of this waste totaled about 348 million Ci at the end of 1991, which is approximately 36% of the total HLW radioactivity. Some of this waste is stored in

149 SSTs. No waste has been added to the SSTs since 1980. This waste consists of about 83,200 m^3 (22 million gal) of liquid waste consisting of free tank supernatant in single- and double-shell tanks and drainable interstitial liquid in single-shell tanks, and 155,800 m^3 (41 million gal) of solid tank waste consisting of sludge and salt cake in single- and double-shell tanks. Waste was first put in the SSTs in the 1940s and considerable uncertainty exists regarding its composition because of poor record keeping of waste inventories and the mixing of waste streams that has occurred over the years. The waste in the SSTs consists of HLW, TRUW, and several LLWs but is managed as HLW at Hanford. The HLW inventory in 2015 (neglecting volume reduction due to treatment) is projected to be on the order of 213,211 m^3 (56 million gal).

Projections for current and future HLW inventories at Hanford are taken from the *Final Environmental Impact Statement Safe Interim Storage of Hanford Tank Wastes* (DOE 1995d) and the *Draft Environmental Impact Statement for the Tank Waste Remediation System* (DOE 1996a), and are based on the assumptions that spent fuel reprocessing will not resume and that irradiated spent fuel at Hanford will remain in wet storage without further processing that may result in the generation of HLW.

Most of the Sr-90 and the Cs-137 (plus their radiological progeny) has been removed from the original waste during 1974 and 1984 and converted to strontium fluoride and cesium chloride, which are solids. These solids have been placed in double-walled capsules and stored in a water basin. DOE leased about half of the 1,577 cesium capsules to commercial enterprises, primarily for medical sterilization purposes. Of the 1,577 cesium capsules and 640 strontium capsules produced, a total of 249 cesium capsules and 35 strontium capsules have been dismantled. In 1988, DOE recalled the capsules after one leaked a very small amount of radioactive material. The final shipment of cesium capsules back to the Hanford site occurred during September 1996. A total of 1,328 cesium capsules (totaling 2.5 m^3 [88 ft^3]) and 605 strontium capsules (totaling 1.1 m^3 [39 ft^3]) are currently in storage at Hanford.

The TWRS Final EIS, issued in August 1996, addresses actions to manage and dispose of approximately 212,000 m^3(56 million gal) of radioactive, hazardous, and mixed waste within the TWRS program at Hanford (61 FR 45949). The EIS also addresses actions to manage and dispose of the cesium and strontium contained in approximately 1,930 metal, double-walled capsules. The EIS identifies Phased Implementation as the preferred alternative for remediating Hanford's high-level tank waste. Under the Phased Implementation alternative, the high-level tank waste would be remediated in a two-phase process. Phase 1 would involve design, construction, and operation of demonstration-scale treatment facilities. Phase 2 would be implemented following Phase 1 and would involve the design, construction, and operation of full-scale treatment facilities to remediate the remainder of the tank waste. Under both phases of the preferred alternative, the HLW would be vitrified and placed into canisters for interim storage pending off-site disposal at a geologic repository. Vitrification of all Hanford HLW is expected to be completed by 2028.

The estimated number of HLW canisters to be produced from vitrification of the Hanford Site tank waste depends on the performance of separations and treatment processes implemented to treat the HLW, and the canister size. For purposes of this analysis, an estimated 15,000 standard canisters (each containing 0.62 m^3 [22 ft^3] of borosilicate glass) were assumed to be produced from treating existing HLW (Walters 1995). This value is based on the preliminary analysis performed for the TWRS. The *Draft Environmental Impact Statement for the Tank Waste Remediation System* (DOE 1996a) indicates a total of 33,400 HLW canisters, with a range between 13,600 to 70,000 canisters. Recent communication between S. Folga (Argonne National Laboratory) and R. Lober (DOE/Richland) on July 25, 1996, indicated that the canister totals provided in the draft TWRS EIS should be considered to be fairly conservative, and that a value of 23,000 canisters may be more appropriate. More recent information received from P. LaMont (DOE/Richland) on July 29, 1996, indicates a new canister total of 12,200 (each containing 1.15 m^3 [40.6 ft^3] of borosilicate glass), based on Hanford's submittal to the Integrated Data Base 1996 data call and the current revision of the draft TWRS EIS. Given the current uncertainty concerning the total number of canisters estimated to be generated at Hanford, it was conservatively assumed to be 15,000 canisters, in agreement with the canister inventory applied in the Draft Waste Management Programmatic EIS (DOE 1995b) (WM PEIS).

Preliminary analysis indicates that the impacts associated with the canister total of 12,200 indicated in DOE (1996a) are similar to those for the canister inventory applied in the Final WM PEIS (i.e., 15,000) (Table 2.1). The Final WM PEIS forecasts conservative results because the number of canisters applied in the Final WM PEIS analyses is greater than in the TWRS EIS (DOE 1996a) and an individual canister has a greater radiological activity (by approximately 50%) compared with the TWRS EIS (as indicated by Table 2.2). Thus, an accident associated with this canister would have greater impacts (as an example) and the occupational dose received by the work force during interim storage of HLW canisters pending disposal at the national geologic repository appears to be greater using the WM PEIS canister inventory, again due to the higher radiological activity of a single canister. The overall risk of transportation would decrease on the average by 10% and at most by 12% for the No Action Alternative using the lower Hanford canister number (Table 2.1). In general, the final number of canisters at Hanford would depend on the performance of separations and treatment processes implemented to treat the HLW. It would appear that the uncertainty in the total number of canisters at Hanford would not significantly affect comparisons among alternatives within the WM PEIS concerning HLW management.

Recently, DOE and state and federal environmental regulators agreed on a modification to the Hanford Tri-Party Agreement (TPA 1994). Under this new agreement, tank waste will undergo limited pretreatment to maximize the routing of radioactivity to the HLW vitrification facility while directing the bulk of the tank waste material to the LLW vitrification facility and all tank wastes will be vitrified. The proposed changes include delaying the HLW vitrification project so that immobilization of low-activity radioactive waste can be factored in. Construction of the HLW treatment facility is to begin in 2002 and facility operations are to begin in December 2009; vitrification is to be completed by December 2028.

TABLE 2.1 Variance in Estimated Transportation Risk When 12,200 versus 15,000 Canisters Are Assumed to Be Produced from Hanford's HLW

WM PEIS Alternative	Truck (%)		Rail (%)	
	Crew	Public	Crew	Public
No Action	-11	-11	-12	-10
Decentralization	-10	-11	-11	-10
Regionalization, Case 1	-10	-11	-11	-9
Regionalization, Case 2	-10	-10	-11	-9
Centralization, Case 1	-11	-11	-11	-10
Centralization, Case 2	-8	-8	-9	-8

TABLE 2.2 Comparison of HLW Glass Canister Contents between the Final WM PEIS and the Final TWRS EIS for Hanford

Major Radionuclides in Hanford HLW	Final TWRS EIS[a]			Final WM PEIS[b]	Variance (%)
	Total Activity for HLW Glass Canisters (Ci)	Total Number of HLW Canisters	Activity per HLW Canister (Ci/canister)	Activity per HLW Canister (Ci/canister)	
SR-90	2.713E+07	12,200	2.22E+03	2.98E+04	-93%
Y-90	2.713E+07	12,200	2.22E+03	2.98E+04	-93%
Tc-99	3.209E+04	12,200	2.63E+00	7.50E+00	-65%
Cs-137	1.760E+07	12,200	1.44E+03	3.61E+04	-96%
Ba-137m	1.667E+07	12,200	1.37E+03	3.40E+04	-96%
Sm-151	8.205E+05	12,200	6.73E+01	6.98E+02	-90%
Pu-238	1.090E+03	12,200	8.93E-02	4.40E-01	-80%
Pu-239	2.632E+04	12,200	2.16E+00	1.20E+00	80%
Pu-240	6.667E+03	12,200	5.46E-01	3.90E-01	40%
Pu-242	2.801E-01	12,200	2.30E-05	7.60E-05	-70%
Am-241	1.007E+05	12,200	8.25E+00	2.84E+02	-97%
Am-242	5.373E+01	12,200	4.40E-03	2.20E-01	-98%
Total	8.955E+07	12,200	7.34E+03	1.37E+05	-95%

[a] DOE (1996a).

[b] DOE (1996b).

This analysis assumes the DWPF canister design for the vitrified Hanford tank waste, with a canister volume of 0.62 m^3 (22 ft^3) of borosilicate glass at a waste loading of 25 wt%. The radionuclide composition of the HLW canisters assumed in this analysis is shown in Table 2.2 along with the mass, radioactivity, and thermal power for each radionuclide. These values represent the activity level of the HLW at the beginning of 1990 and are based on nominal values of the neutralized cladding acidic waste (NCAW) reference feed to the HWVP. It was previously intended by DOE to remove the NCAW prior to other wastes (GAO 1993). The NCAW constitutes about 6 percent of the waste in DSTs, has a relatively well defined composition, and is currently stored in two DSTs.

The current Hanford approach is to blend the HLW retrieved from all 177 tanks prior to pretreatment. It would be expected that canisters produced from the vitrification of the blended DST and SST wastes will have significantly lower values for radioactivity content and thermal power than for the canisters produced from only the DST waste at Hanford. One approach to estimate the "average" canister radioactivity for the blended DST and SST wastes is to divide the total radioactivity of the two streams (2.12×10^8 Ci) by the estimated number of canisters (15,000) to arrive at an average value of approximately 14,000 Ci per canister, which is an order-of-magnitude lower than that estimated based on the DST waste alone (137,000 Ci/canister). This result indicates that the nominal NCAW radionuclide concentration may be too conservative in comparison with the blended SST and DST waste compositions. It should be noted that a more accurate approach which accounts for the amount of radioactivity associated with the low-level waste (LLW) fraction after waste pretreatment would result in a canister radioactivity lower than the value of 14,000 Ci per canister calculated above. However, because of the uncertainty concerning tank retrieval and pretreatment, the data used in this report for the canister contents at Hanford are from (Mitchell and Nelson 1988), which is based on the HLW in the 10 DSTs at Hanford.

As previously mentioned, the specific disposal method for the existing strontium and cesium capsules has not been selected. One possibility is to place the capsules in overpacks for repository emplacement. An issue is that this overpack concept may not meet the waste acceptance criteria (WAC) for disposal in a national repository. Another possibility is to open the capsules and combine the strontium and cesium with one of the low-heat feeds to the HWVP for vitrification. If this is done, the maximum radioactivity and thermal output per canister could conceivably exceed the quantities shown in Table 2.3.

The HLW canister assumed in this analysis is made of 304L stainless steel pipe with an outside diameter of 61 cm (24 in.), a length of 300 cm (118 in.), and a thickness of 0.95 cm (0.37 in.). The canister is identical to that planned for use at the Savannah River DWPF. Table 2.4 gives additional information on the canister and the HLW glass. The fill level of the canister is approximately 85% of the available internal canister fill volume, which results in a canister glass volume of 0.63 m^3 (22 ft^3). A 15% void volume minimizes the potential of canister overfill. Table 2.5 gives the decay of radioactivity and thermal power for a single vitrified HLW canister.

TABLE 2.3 Radionuclide Content per Hanford Canister for NCAW Glass: Nominal Case[a]

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Fe-55	7.2E-03	1.8E+01	6.1E-04
Ni-59	1.4	1.1E-01	4.3E-06
Co-60	1.3E-03	1.5	2.3E-02
Ni-63	2.0E-01	1.2E+01	4.9E-03
Se-79	4.5E-02	3.1E-03	7.8E-07
Sr-89	1.8E-17	5.4E-13	1.9E-15
Sr-90	2.2E+02	2.9E+04	3.5E-01
Y-90	5.5E-02	2.9E+04	1.7E+02
Y-91	5.6E-15	1.4E-10	5.0E-13
Nb-93m	2.2E-03	6.2E+01	1.1E-04
Zr-93	4.2E+02	1.1	1.2E-04
Zr-95	1.4E-13	2.9E-09	1.5E-11
Nb-95	1.7E-13	6.7E-09	3.2E-11
Tc-99	4.4E+02	7.5	3.8E-03
Ru-103	1.0E-22	3.4E-18	1.1E-20
Rh-103m	9.3E-26	3.0E-18	7.0E-22
Ru-106	1.3E-02	4.2E+01	2.5E-03
Rh-106	1.2E-08	4.2E+01	4.0E-01
Pd-107	5.9E+01	3.0E-02	1.8E-06
Ag-110m	4.7E-07	2.2E-03	3.7E-05
Cd-113m	3.9E-02	8.5	1.4E-02
In-113m	6.0E-15	1.0E-07	2.4E-10
Sn-113	1.0E-11	1.0E-07	1.7E-11
Cd-115m	1.3E-22	3.2E-18	1.2E-20
Sn-119m	1.5E-06	6.8E-03	3.5E-06
Sn-121m	1.3E-03	7.8E-02	1.6E-04
Sn-123	4.4E-09	3.7E-05	1.1E-07
Sn-126	1.3E+01	3.7E-01	4.6E-04
Sb-124	6.6E-19	1.2E-14	1.5E-16
Sb-126	6.1E-07	5.1E-02	9.4E-04
Sb-126m	4.7E-09	3.7E-01	4.6E-03
Sb-125	2.5E-01	2.5E+02	7.9E-01
Te-125m	3.4E-03	6.2E+01	5.2E-02
Te-127	2.5E-12	6.6E-06	8.8E-09

TABLE 2.3 (Cont.)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Te-127m	7.1E-10	6.7E-06	3.6E-09
Te-129	1.5E-30	3.1E-23	1.1E-25
Te-129m	1.6E-27	4.8E-23	8.4E-26
I-129	7.3E-02	1.3E-05	6.0E-09
Cs-134	7.2E-02	9.3E+01	9.5E-01
Cs-135	1.8E+02	2.0E-01	6.7E-05
Cs-137	4.2E+02	3.6E+04	4.0E+01
Ba-137m	6.3E-05	3.4E+04	1.3E+02
Ce-141	1.0E-26	2.9E-22	4.3E-25
Ce-144	2.5E-02	8.0E+01	5.3E-02
Pr-144	1.1E-06	8.0E+01	5.9E-01
Pr-144m	5.3E-09	9.6E-01	3.3E-04
Pm-147	5.6	5.2E+03	1.9
Pm-148m	2.9E-23	6.2E-19	7.9E-21
Sm-151	2.7E+01	6.7E+02	8.2E-02
Eu-152	8.1E-03	1.4	1.1E-02
Gd-153	3.8E-09	1.4E-05	1.2E-08
Eu-154	5.4E-01	1.5E+02	1.3
Eu-155	2.9E-01	1.4E+02	1.0E-01
Tb-160	8.4E-17	9.5E-13	7.7E-15
U-234	7.3E-01	4.6E-03	1.3E-04
U-235	8.8E+01	1.9E-04	5.0E-06
U-236	6.5	4.2E-04	1.1E-05
U-238	1.0E+04	3.5E-03	8.9E-05
Np-237	2.2E+02	1.6E-01	4.8E-03
Pu-238	2.6	4.4E-01	1.5E-02
Pu-239	1.9E+01	1.2	3.6E-02
Pu-240	1.7	3.9E-01	1.2E-02
Pu-241	1.2E-01	1.3E-01	3.9E-04
Pu-242	2.0E-02	7.6E-05	2.2E-06
Am-241	8.3E+01	2.8E+02	9.4
Am-242	2.7E-07	2.2E-01	2.5E-04
Am-243	1.9E-01	3.8E-02	1.2E-03

TABLE 2.3 (Cont.)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Cm-242	5.5E-05	1.8E-01	6.7E-03
Cm-244	6.2E-02	5.0	1.8E-01
Total	≈ 1.3E+04	≈ 1.4E+05	≈ 3.9E+02

[a] This table identifies the nominal expected activity of HWVP canisters at the time of vitrification. Canister contains 1,650 kg of HLW glass (85% fill).

Source: Mitchell and Nelson (1988).

A package consisting of immobilized radioactive wastes must have adequate shielding in order to be handled safely. Gamma rays are emitted from the decay of the fission products, neutron particles from the spontaneous fission of any actinides present, and alpha particles from alpha-emitting radionuclides. Table 2.6 gives the calculated neutron production rates from spontaneous fission and (alpha, n) reactions for a single vitrified HLW canister on the basis of NCAW glass. The neutron production rates shown are uniform source terms for the glass rather than dose rates at the exterior; consequently, the shielding effects of the borosilicate glass and canister wall have not been included. The results show that (alpha, n) reactions account for more than 93% of the total neutron production.

The strontium and cesium capsules in storage at Hanford are currently classified as waste by-product, which means that they are available for productive use if uses can be found. DOE is attempting to find uses for the strontium and cesium capsules in storage at Hanford; for example, the strontium could be used as a source of heat and the cesium could be used to sterilize medical equipment or to irradiate food to extend its shelf life. If the decision is made to reclassify the strontium and cesium capsules as HLW requiring disposal in a national geologic repository, the strontium and cesium capsules are assumed in this analysis to be placed in racks and inserted into canisters. Two proposed designs (thin wall and cast steel) for the overpack canisters have been proposed (ORNL 1992a). The design currently proposed in the Final TWRS EIS (DOE 1996a) is the Hanford Multi-Purpose Canister, which would be approximately 4.6 m (15 ft) long and 1.4 m (4.5 ft) in diameter. Although the proposed canister design has not yet been decided, in this report, it is assumed that the Hanford Multi-Purpose Canister will be used to contain the strontium and cesium capsules. The annular region between the capsules and the canister may be filled with material that would immobilize the capsule contents upon breakdown of the capsule integrity; examples are natural ion-exchange materials such as chabazite and clinoptilolite.

TABLE 2.4 Projected Characteristics of the Immobilized HLW Form and Canister at Hanford

High-Level Waste Form and Canister Characteristics	
Waste form	Borosilicate glass in sealed canister
Canister material	Stainless steel type 304L
Waste glass density (g/cm^3)	2.64
Weight per canister (kg)	
Empty canister	500
Borosilicate glass	1,650
Total loaded weight	2,150
Waste loading in glass (wt%)	25
Glass volume per canister (m^3)	0.626[a]
Canister volume filled with glass (%)	85
Canister dimensions	
Outside diameter (cm)	61
Height overall (cm)	300
Wall thickness (cm)	0.95
Inside volume (m^3)	0.736
Radionuclide content (Ci/canister)[b]	
Nominal	137,000
Maximum	298,000
Thermal power (W/canister)[a]	
Nominal	389
Maximum	869
Radioactivity 50 yr after filling	
Ci/canister	58,000
W/canister	180
Neutron production (neutrons/s/canister)	
Spontaneous fission	1,700,000
Alpha, n	24,800,000
Total	26,500,000

[a] Average fill temperature of 825°C (1,517°F).

[b] All values shown are based on NCAW reference feed with 25 wt% waste oxide in glass. Activities and thermal power are at time of filling canister. The range of values shown is from Mitchell and Nelson (1988), in which estimated activities and radionuclide compositions are given for two NCAW feeds referred to as nominal and maximum.

Sources: Salmon and Notz (1991); ORNL (1992a,b).

TABLE 2.5 Radioactivity and Thermal Power per Hanford HLW Canister over Time

Decay Time after Immobilization (yr)	Total Radioactivity per Canister (Ci)	Total Thermal Power per Canister (W)
0	136,900	389
1	132,600	380
2	128,500	370
5	118,200	344
10	104,200	306
15	92,500	273
20	82,300	243
30	65,200	194
50	41,000	125
100	13,100	44
200	1,570	10
300	375	6.2
350	260	5.6
500	157	4.3
1,000	70	2.0
2,000	24	0.44
5,000	12	0.06
10,000	12	0.05
20,000	11	0.04
50,000	10	0.03
100,000	9.2	0.03
500,000	5.3	0.05
1,000,000	3.6	0.04

Source: Adapted from ORNL (1992a). Based on NCAW reference feed with 25 wt% waste oxide in glass.

The number of strontium or cesium capsules per canister is limited by heat load. An internal rack would support either three or four strontium or cesium capsules along the axis of the canister. This arrangement would allow the decay heat to be distributed along the entire canister area, thus avoiding excessive local heat rates. The number of capsules per canister would depend on the thermal limitations of the capsules themselves and the thermal limit based on the heat load limit of the HLW geologic repository. The total thermal power of the capsules at the end of 2010 would be 111 kW for the strontium capsules and 181 kW for the cesium capsules, on the basis of the capsule overpacking completed at the end of 2010 (ORNL 1992a). The values would decrease to 85.1 kW and 143.9 kW for the strontium and cesium capsules, respectively, based on December 31, 2019. If it is assumed that the strontium and cesium capsules will not be placed together in a canister and that

the thermal limits for strontium and cesium overpacks are 1.17 kW and 0.8 kW, respectively, then each overpack canister will contain eight strontium or cesium capsules. A total of 76 strontium canisters and 166 cesium canisters would result (242 total canisters). Table 2.7 gives the characteristics of the strontium and cesium capsules. The thermal power for the two canister contents was estimated on the basis of the radionuclide content, using the "Q" values reported in ORNL (1992a).

2.2.2 Savannah River

SRS was established in 1950 by the U.S. Atomic Energy Commission to produce nuclear materials for the nation's defense (DOE 1995a). The production of these nuclear materials resulted in radioactive waste by-products and hazardous waste that have been stored at SRS. In the early 1980s, the DOE initiated efforts to end the interim storage of its HLW at SRS by developing plans to vitrify the waste in the DWPF.

TABLE 2.6 Detailed Rates of Neutron Production per Hanford Canister (neutrons/s)[a]

Actinide	(Alpha, n)	Spontaneous Fission	Total
U-238	5.1E+01	1.4E+02	1.9E+02
Pu-238	3.2E+04	1.2E+02	3.2E+04
Pu-239	4.7E+04	6.2E-01	4.7E+04
Pu-240	1.8E+04	2.2E+03	2.0E+04
Pu-242	3.5	5.8E+01	6.1E+01
Am-241	2.4E+07	2.1E+02	2.4E+07
Cm-242	3.0E+04	3.3E+03	3.4E+04
Cm-244	6.4E+05	1.7E+06	2.4E+06
Total	2.5E+07	1.7E+06	2.7E+07

[a] Neutron production rates are source terms in the glass; shielding effects of the glass and the canister wall have not been calculated.

Source: ORNL (1992a) based on NCAW reference feed with 25 wt% waste oxide in glass.

TABLE 2.7 Projected Characteristics of the Strontium and Cesium Canisters at Hanford

Canister Characteristics	Strontium Fluoride	Cesium Chloride
Estimated number of repository canisters	76	166
Material	Hanford Multi-Purpose Canister	Hanford Multi-Purpose Canister
Dimensions		
Outer diameter (cm)	140	140
Overall height (cm)	460	460
Radionuclide content as of 1990 (Ci/canister)	Sr-90: 2.8E+06 Y-90: 2.8E+05	Cs-137: 2.6E+05 Ba-137m: 2.5E+05
Thermal power as of 1990 (W/canister)	3,460	1,240
Radionuclide content as of 2019 (Ci/canister)	Sr-90: 1.7E+05 Y-90: 1.7E+05	Cs-137: 1.8E+05 Ba-137m: 1.7E+05
Thermal power as of 2019 (W/canister)	1,120	867

Sources: Adapted from ORNL (1992a,b).

The SRS has a large inventory of HLW. At the end of 1994, approximately 126,300 m^3 (34 million gal) of HLW with a radioactivity level of about 535 million Ci was in storage in underground, double-walled tanks. These figures represent about 33 and 56% of the total volume and radioactivity, respectively, of DOE HLW (ORNL 1995). At the end of 1994, the forms of HLW at SRS included about 58,100 m^3 (15 million gal) of liquid tank waste and 68,200 m^3 (18 million gal) of solids consisting of sludge, salt cake, and precipitate. Radioactivity levels at the end of 1994 were about 231 million Ci in the sludge and about 302 million Ci in the solids (ORNL 1995). As generated, most of the waste was in the form of an acidic liquid. Alkaline liquid and sludge are produced when the acidic liquid is treated with caustic (sodium hydroxide) and as the mixture settles with time, respectively. Salt cake is produced when the supernatant liquor is concentrated in evaporators and the precipitate results from treatment of the salt cake with an in-tank precipitation process.

Table 2.8 gives the projections of the HLW inventory at SRS based on the *Savannah River Site Waste Management Final Environmental Impact Statement* (DOE 1995a). This document provides estimates of the minimum and maximum amounts of wastes that may be generated in the future, to account for wastes that may be generated during interim management of nuclear materials. This analysis applies the inventory from the expected waste forecast. Additions to the current HLW inventory at SRS (about 15% of the total SRS HLW inventory in FY 2015) are primarily due to facility stabilization activities such as the F- and H- Canyon stabilization and decommissioning

TABLE 2.8 Projected HLW Inventory at SRS

	Liquid HLW Volume (m^3)		
Year	Expected	Minimum	Maximum
1995	2,598	705	2,598
1996	4,317	1,317	4,358
1997	3,752	1,158	4,358
1998	2,432	1,240	4,321
1999	1,788	326	2,611
2000	2,175	387	2,174
2001	2,175	387	2,174
2002	857	387	850
2003	228	387	227
2004	126	387	227
2005	126	387	227
2006	126	387	227
2007	126	387	227
2008	126	387	227
2009	126	387	227
2010	126	387	227
2011	126	387	227
2012	126	387	227
2013	126	387	227
2014	126	387	227
2015	126	387	227
Total (1995-2015)	21,834	10,938	26,395
Current (1994)	130,581		
Total SRS HLW Inventory at 2015	152,415		

projects, which may be expected to average about 1,000 m^3/yr of material assumed to be managed as HLW.

For purposes of this study, the estimated total number of canisters is assumed to equal the total planned interim storage capacity of HLW canisters, or 4,572 canisters. This value is derived from the documents, *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility* (DOE 1994b) and the *Savannah River Site Waste Management Final Environmental Impact Statement* (DOE 1995a). The value of 4,572 canisters estimated to be generated at SRS is referenced in the recent document *Storage and Disposition of Weapons-Usable Fissile Materials Draft Programmatic Environmental Impact Statement* (DOE 1996c), based on the two previously-cited documents. The current storage capacity is stated to be 2,286 canisters, with capacity for additional 2,286 canisters to be available in year 2007, for a total of 4,572 canisters.

The numbers of canisters applied in the WM PEIS analyses are based on the current operating plans and projected funding assumed within the Defense Waste Processing Facility (DWPF) SEIS and the SRS WM EIS, which indicated DOE expected by 2018 that the HLW at SRS would have been processed into borosilicate glass and the HLW tanks would be empty. Operation of the DWPF would occur from FY 1996 to 2018. Decisions made pursuant to other National Environmental Protection Act (NEPA) analyses pursued after the DWPF SEIS and the SRS WM EIS could extend the period of HLW generation and increase the number of expected canisters.

The latest version of the HLW System Plan (DOE 1995h) presents a total of approximately 6,000 canisters projected to be produced at SRS. Preliminary analysis indicates that the impacts associated with the higher total number of canisters (i.e., 6,000) are similar to those for the canister inventory applied in the WM PEIS (i.e., 4,572). The WM PEIS forecasts conservative results because although the numbers of canisters applied in the WM PEIS analyses are lower than the HLW System Plan (DOE 1995b), an individual canister has a greater radiological activity compared with the HLW System Plan, and thus an accident associated with this canister would have greater impacts (as an example). Similarly, the occupational dose received by the work force during interim storage of HLW canisters pending disposal at the national geologic repository appears to be greater using the WM PEIS canister inventory, again due to the higher radiological activity of a single canister. The overall risk of transportation would increase on the average by 11% and at most by 14% for the Centralized Alternative, Case 2 using the higher SRS canister number (see Table 2.9). It would appear that the uncertainty in the total number of canisters at SRS would not significantly affect comparisons among alternatives in the WM PEIS concerning HLW management.

The DWPF was built to treat HLW at SRS. Construction of this facility was initiated in 1983 and completed in 1990. An 18-month testing program for the DWPF began in March 1993. The program started with cold chemical runs to test the DWPF's ability to receive and process waste streams and will be followed with hot operations using surrogate radioactive waste stimulants. The HLW will initially be pretreated with in-tank processing. The salt portion will be decontaminated

TABLE 2.9 Variance in Estimated Transportation Risk When 4,572 Versus 6,000 Canisters Are Assumed to Be Produced from the Savannah River Site's HLW

	Truck (%)		Rail (%)	
WM PEIS Alternative	Crew	Public	Crew	Public
No Action	12	12	11	13
Decentralization	12	11	10	13
Regionalization, Case 1	11	11	10	12
Regionalization, Case 2	11	11	10	13
Centralization, Case 1	3	3	3	3
Centralization, Case 2	14	14	13	14

by precipitation and sorption for disposal as LLW. The sludge will be washed with a water-caustic mixture to remove those nonradioactive components that are soluble. The salt precipitates, sorption products, and washed sludge will then be mixed with glass frit and other additives and sent to the melter for vitrification.

SRS completed the DWPF SEIS in November 1994 (DOE 1994b). The Record of Decision (March 28, 1995) describes DOE's decision to complete construction and begin operation of the DWPF. The DWPF became operational on March 12, 1996, and 27 HLW canisters were produced as of July 15, 1996. Vitrified HLW canister production is expected to be completed in 2020.

Each canister is assumed to contain 0.637 m^3 (22.5 ft^3) of borosilicate glass containing HLW. The glass incorporates 36 wt% oxides from waste (28 wt% from spent fuel and 8 wt% from processing chemicals) and 64 wt% oxides from the nonradioactive glass frit. At the end of this first year of operation, the average radioactivity content of each canister will be 110,000 Ci and the average heat content will be about 263 W. It is projected that approximately 300 to 400 canisters will be produced in each subsequent year; so that after 24 years of operations, 4,572 canisters will have been produced. At this time, the average radioactivity level of the 4,572 canisters will be about 80,000 Ci and the average heat content about 254 W (ORNL 1992a,b). By the end of 2020, these averages will have decreased to 70,000 Ci and 201 W. Table 2.10 gives the radionuclide composition of the HLW canisters produced at the DWPF. These values are based on the radiological properties of the waste as of the beginning of 1990 and also the most highly radioactive glass likely to be made from in-tank processing of the sludge-supernatant (i.e., sludge aged an average of 5 years and a cesium-containing precipitate derived from processing the supernatant aged an average of 15 years). Given a total canister activity of 234,400 Ci and a thermal power of 709 W, it is

TABLE 2.10 Radionuclide Content per SRS Canister[a]

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Cr-51	1.0E-21	9.3E-17	2.0E-20
Co-60	1.5E-01	1.7E+02	2.6
Ni-59	3.2E-01	2.4E-02	9.5E-07
Ni-63	4.8E-02	3.0	3.0E-04
Se-79	2.4	1.7E-01	4.2E-05
Rb-87	1.0E+01	8.7E-07	7.3E-10
Sr-89	1.5E-09	4.3E-05	1.5E-07
Sr-90	3.4E+02	4.7E+04	5.4E+01
Y-90	8.8E-02	4.8E+04	2.7E+02
Y-91	3.1E-08	7.6E-04	2.7E-06
Zr-93	4.4E+02	1.1	1.3E-04
Zr-95	4.7E-07	1.0E-02	5.1E-05
Nb-94	5.1E-04	9.6E-05	9.8E-07
Nb-95	5.4E-07	2.1E-02	1.0E-04
Nb-95m	3.3E-10	1.2E-04	1.7E-02
Tc-99	1.8E+02	3.1	1.5E-03
Ru-103	5.2E-13	1.7E-08	5.8E-11
Ru-106	6.7E-01	2.3E+03	1.3E-01
Rh-103m	5.0E-16	1.6E-08	3.8E-12
Rh-106	6.3E-07	2.3E+03	2.2E+01
Pd-107	2.9E+01	1.5E-01	8.7E-07
Ag-110m	2.6E-05	1.3E-01	2.1E-03
Cd-113	1.5E-01	5.0E-14	8.4E-17
Cd-115m	4.8E-14	1.2E-09	4.5E-12
Sn-121m	1.3E-03	7.9E-02	1.6E-04
Sn-123	3.1E-05	2.5E-01	8.0E-04
Sn-126	1.6E+01	4.4E-01	5.5E-04
Sb-124	4.1E-12	7.1E-08	9.4E-10
Sb-125	8.2E-01	8.5E+02	2.7
Sb-126	7.4E-07	6.2E-02	1.1E-03
Sb-126m	5.6E-09	4.4E-01	5.6E-03
Te-126m	1.5E-02	2.8E+02	2.3E-01
Te-127	4.6E-08	1.2E-01	1.6E-04
Te-128m	1.3E-05	1.2E-01	6.6E-05
Te-129	1.5E-19	3.1E-12	1.1E-14
Te-129m	1.6E-16	4.7E-12	8.3E-15
Cs-134	2.6E-01	3.4E+04	3.4
Cs-135	8.6E+01	9.9E-02	3.3E-05

TABLE 2.10 (Cont.)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Cs-136	1.1E-44	7.8E-40	1.1E-42
Cs-137	4.9E+02	4.3E+04	4.8E+01
Ba-136m	3.2E-50	8.6E-39	1.0E-41
Ba-137m	7.7E-05	4.2E+04	1.6E+02
Ba-140	1.4E-41	1.0E-36	2.9E-39
La-140	7.7E-43	4.3E-37	7.2E-38
Ce-141	1.3E-15	3.6E-11	5.3E-14
Ce-142	4.0E+02	9.6E-06	0
Ce-144	3.1E-02	9.8E+03	6.5E-02
Pr-143	1.8E-34	1.2E-34	2.3E-38
Pr-144	1.3E-04	9.8E+03	7.3E+01
Pr-144m	6.5E-07	1.2E+02	4.1E-02
And-144	4.1E+02	4.9E-10	0
And-147	1.6E-49	1.3E-43	3.0E-47
Pm-147	2.6E+01	2.4E+04	8.7
Pm-148	4.2E-16	7.0E-11	5.4E-13
Pm-148m	4.7E-14	1.1E-09	1.3E-11
Sm-147	8.8E+01	2.0E-06	2.7E-08
Sm-148	1.9E+01	5.8E-12	6.9E-14
Sm-149	7.4	1.8E-12	0
Sm-151	9.4	2.5E+02	2.9E-02
Eu-152	2.1E-02	3.7	2.8E-02
Eu-154	2.3	6.2E+02	5.5
Eu-155	1.0	4.8E+02	3.5E-01
Eu-156	9.5E-37	5.2E-32	5.4E-34
Tb-160	9.9E-11	1.1E-06	9.1E-09
Ti-208	3.8E-12	1.1E-03	2.6E-05
U-232	6.3E-04	1.3E-02	4.3E-04
U-233	1.6E-04	1.6E-06	4.6E-08
U-234	5.5	3.4E-02	9.9E-04
U-235	7.3E+01	1.6E-04	4.1E-06
U-236	1.7E+01	1.1E-03	3.1E-05
U-238	31,220	1.1E-02	2.7E-04
Np-236	1.3E-06	1.7E-08	3.5E-11
Np-237	1.3E+01	8.9E-03	2.7E-04
Pu-236	2.3E-04	1.2E-01	4.2E-03
Pu-237	7.4E-16	8.9E-12	3.3E-15

TABLE 2.10 (Cont.)

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Pu-238	8.7E+01	1.4E+03	4.9E+01
Pu-239	2.1E+02	1.3E+01	4.0E-01
Pu-240	3.8E+01	8.7	2.7E-01
Pu-241	1.6E+01	1.7E+03	5.2E-01
Pu-242	3.2	1.2E-02	3.6E-04
Am-241	3.2	1.1E+01	3.7E-01
Am-242	1.8E-08	1.4E-02	1.6E-05
Am-242m	1.5E-02	1.4E-02	5.7E-06
Am-243	2.9E-02	5.8E-03	1.9E-04
Cm-242	1.1E-05	3.5E-02	1.3E-03
Cm-243	1.1E-04	5.6E-03	2.0E-04
Cm-244	1.3	1.1E+01	3.8
Cm-245	3.9E-05	6.7E-06	2.2E-07
Cm-246	1.7E-06	5.3E-07	1.7E-08
Cm-247	7.1E-09	6.6E-13	2.1E-14
Cm-248	1.6E-10	6.9E-13	8.5E-14
Total	≈3.4E+04	≈2.3E+05	≈7.1E+02

[a] Quantities shown are for sludge-precipitate glass and are based on Baxter (1988), assuming sludge aged an average of five years and supernatant aged an average of five years, with a canister load of 1,682 kg of glass. Radionuclide contents are at time of filling canister.

Source: Adapted from Baxter (1988).

expected that the maximum values of radioactivity and thermal power of any HLW glass produced at SRS will not exceed the values shown in Table 2.10.

Table 2.11 provides information on the radiological, chemical, and physical characteristics of the HLW canisters produced at SRS. The density of the HLW glass is 2.85 g/cm^3 (178 lb/ft^3) at 25°C (77°F); consequently, on average, each canister will contain a total of 1,682 kg (3,700 lb) of HLW glass on the basis of a reference-case fill level of 85%. The decay of radioactivity and thermal power over time is presented in Table 2.12; the maximum total activity and thermal power at the time of filling are 234,400 Ci and 709 W per canister. The canister radioactivity and thermal power are estimated to have decreased to less than 0.00170% and 0.003%, respectively, of their initial values after one million years. Table 2.13 shows the calculated rates of neutron production per canister of HLW glass at the time of filling on the basis of radionuclide composition in Table 2.10;

TABLE 2.11 Projected Characteristics of the Immobilized HLW Form and Canister at SRS

Immobilized HLW Form and Canister Characteristics at SRS	
Waste form	Borosilicate glass in closed canister
Canister material	Stainless-steel type 304L
Canister fill level (%)	85
Glass density (g/cm^3)	
At average fill temperature of 825°C	2.69
At 25°C	2.85
Glass volume (m^3)	
At average fill temperature of 825°C	0.637
At 25°C	0.601
Canister inside volume (m^3)	0.736
Canister dimensions (cm)	
Outside diameter	61.0
Overall height	300.0
Wall thickness	0.95
Weights (kg)	
Canister	500
Glass	1,682
Total	2,182
Waste loading (wt% waste oxides in the glass)	36.0
Radioactivity at time of filling	
Ci/canister	234,400
W/canister	709
Radioactivity 50 years after filling	
Ci/canister	57,000
W/canister	200
Neutron production	
Spontaneous fission	15,050,000
(Alpha, n)	64,110,000
Total	79,160,000
Canister surface temperature at air temperature of 38°C	58°C
Canister centerline temperature at air temperature of 38°C	89°C
Radiation level at canister surface (rad/h)	5,600

Sources: Baxter (1988); DOE (1987); ORNL (1992a,b); Salmon and Notz (1991).

TABLE 2.12 Radioactivity and Thermal Power per SRS Canister over Time

Years after Vitrification	Radioactivity per Canister (Ci)[a]	Thermal Power per Canister (W)[a]
0[b]	234,400	709
1	208,500	627
2	193,800	586
5	169,300	527
10	145,800	467
15	128,400	418
20	113,900	374
30	90,000	301
50	56,500	198
100	17,900	75
200	2,100	17
300	390	7.2
350	227	5.2
500	95	2.7
1,000	42	1.1
1,050	41	1.1
2,000	29	0.72
5,000	24	0.54
10,000	20	0.43
20,000	16	0.30
50,000	11	0.16
100,000	9.2	0.11
500,000	4.8	0.05
1,000,000	2.4	0.02

[a] Includes contribution of actinides and activation products, as well as fission products.

[b] Based on 5-year cooled sludge and 15-year cooled supernate.

Source: ORNL (1992a).

TABLE 2.13 Detailed Rates of Neutron Production per SRS Canister (neutrons/s)[a]

Actinide	(Alpha, n)	Spontaneous Fission	Total
U-238	1.3E+02	4.0E+02	5.3E+02
Pu-238	5.8E+07	2.3E+05	5.8E+07
Pu-239	4.0E+05	5.7	4.0E+05
Pu-240	2.7E+05	3.5E+04	3.0E+05
Pu-242	3.1E+02	5.4E+03	5.7E+03
Am-241	4.2E+05	4.0	4.2E+05
Cm-242	2.0E+03	2.3E+02	2.2E+03
Cm-244	5.1E+06	1.5E+07	2.0E+07
Total	6.4E+07	1.5E+07	7.9E+07

[a] Neutron production rates are source terms in the glass; shielding effects of the glass and the canister wall have not been calculated.

Source: ORNL (1992a).

(alpha, n) reactions account for more than 80% of the total neutron production. The neutron production rates are the source terms within the glass itself and do not account for the shielding effects of the glass and the canister wall.

2.2.3 Idaho National Engineering Laboratory

Reprocessing activities have generated liquid HLW at the ICPP since 1952. High-level waste created in the reprocessing of SNF at the ICPP initially takes the form of an acidic liquid. This waste has been collected and stored in underground stainless steel tanks housed in concrete vaults. Most of the liquid HLW previously generated has been calcined. Roughly 28,000 m^3 (7.4 million gal) has been calcined during nine campaigns of the Waste Calcining facility (WCF) and three campaigns of the NWCF. Approximately 1,900 m^3 (500,000 gal) of liquid HLW is currently stored in four 1,100-m^3 (300,000-gal) tanks and three 70-m^3 (18,500-gal) tanks. Sodium-containing waste has been generated primarily from decontamination efforts but also from processes related to the recovery of uranium. It is anticipated that this waste will continue to be generated, although research is underway to minimize future waste production. At the end of 1994, 7,200 m^3 (1.9 million gal) of sodium-containing waste is stored in seven 1,100-m^3 (300,000-gal) tanks. The liquid acidic waste

represents the majority of the initial volume but a minority of the initial radioactivity (approximately 2.4 million Ci).

Liquid HLW has been treated in the NWCF to result in solid HLW calcine at the ICPP since 1962 and is contained in five CSSFs that have been constructed and filled sequentially. Two additional CSSFs have been constructed and will be filled during future calcination operations. The bins containing the calcine are expected to retain their integrity for at least 500 years (DOE 1983).

By 1994, about 3,800 m^3 (124,000 ft^3) of calcine has been produced with a total radioactivity of 57 million Ci. The production of calcine is anticipated to continue from the solidification of existing liquid wastes and potential future liquid wastes generated from decontamination and decommissioning (D&D) activities. Without calcination, the ICPP tank farm would have been filled to capacity, precluding generation of waste from reprocessing or future decontamination and decommissioning programs. Calcination of liquid HLW at INEL results in a solid that is safer to store than liquid waste (due to its physical form and lower dispersibility) but does not meet NRC requirements for repository disposal.

Current and projected inventories of liquid HLW at INEL are presented in Table 2.14. The values in Table 2.14 represent generation of liquid HLW only and do not take into account potential volume reduction due to evaporation and calcination operations. INEL uses its HLW tank system to manage radioactive wastewaters (technically not HLW) from the following sources: sodium-bearing waste from decontamination and decommissioning (D&D) activities, acid waste, solutions generated during calciner operations, and filter leach processing. The projections shown in Table 2.14 were made on the assumption that the New Waste Calcining Facility would resume operations to pretreat HLW liquids and that radionuclide partitioning would be performed on both liquid and calcined waste streams. It was also assumed that no new HLW would be produced in the future from reprocessing activities; the reprocessing facilities at the INEL were placed into cold standby pending D&D during FY 1992.

In general, the radionuclide content in all of the various calcine types is less than 1 wt% with a radionuclide distribution typical for U-235 fissioning and neutron capture. Calcine has a relatively low activity and heat generation rate (approximately 24,000 Ci/m^3 and 70 W/m^3 [Ernold et al. 1993], respectively, for aged calcine) because of the low concentration of radionuclides. However, these values vary significantly with age, fuel irradiation exposure, and type of fuel reprocessed, among other factors.

For planning purposes, the following assumptions have been made: (1) disposal of calcine will not be allowed, (2) no pretreatment of calcine or liquid HLW will occur such that inert materials will be removed prior to immobilization, and (3) HLW will be converted into a glass

TABLE 2.14 Projected HLW Inventory at INEL

Year	Unprocessed Sodium Waste Volume (gal)	Sodium Waste Reduction Factor	Processed Sodium Waste Volume (m^3)	Unprocessed Acid Waste Volume (m^3)	Acid Waste Reduction Factor	Processed Acid Waste Volume (m^3)	Number of Calciner Operating Months	Gallons of Solution Generated by Calciner	Solution Generated by Calciner (m^3)	Number of Existing Filters Processed	Number of Other Filters Processed	Filter Leach Volume (m^3)	Annual Volume Added to Existing Waste Inventory (m^3)
1995	50,725	1.80	107	11,500	1.21	36	0	5,778	0	36	14	6	148
1996	46,850	1.97	90	15,500	1.14	51	4	5,778	87	37	38	8	237
1997	49,583	1.97	95	118,500	1.60	280	12	5,778	262	0	75	8	646
1998	47,216	1.97	91	18,500	1.28	55	2	5,778	44	0	75	8	197
1999	46,400	1.97	89	4,000	1.00	15	10	5,778	219	1	74	8	331
2000	46,400	1.97	89	4,000	1.00	15	8	5,778	175	13	62	8	287
2001	58,400	1.97	112	4,000	1.00	15	4	5,778	87	37	38	8	223
2002	58,400	1.97	112	4,000	1.00	15	12	5,778	262	0	75	8	397
2003	57,767	1.97	111	4,000	1.00	15	2	5,778	44	49	26	9	178
2004	106,567	3.11	130	11,500	1.00	44	0	5,778	0	61	14	9	182
2005	102,567	3.18	122	19,000	1.00	72	0	5,778	0	61	14	9	203
2006	97,567	3.28	113	19,000	1.00	72	10	5,778	219	1	74	8	411
2007	97,567	3.28	113	19,000	1.00	72	8	5,778	175	5	62	7	366
2008	97,567	3.28	113	19,000	1.00	72	4	5,778	87	0	38	4	276
2009	97,567	3.28	113	19,000	1.00	72	12	5,778	262	0	75	8	455
2010	97,400	3.28	112	19,000	1.00	72	2	5,778	44	11	38	5	233
2011	97,400	3.28	112	19,000	1.00	72	10	5,778	219	0	74	7	411
2012	97,400	3.28	112	4,000	1.00	15	8	5,778	175	0	62	6	309
2013	97,400	3.28	112	4,000	1.00	15	0	5,778	0	0	14	1	129
2014	97,400	3.28	112	4,000	1.00	15	0	5,778	0	0	14	1	129
2015	45,600	1.97	88	4,000	1.00	15	4	5,778	87	0	38	4	194
Total (1995–2015)			2,248			1,106			2,450			138	5,942
Average (1995–2015)			107			53			117			7	283
Current (1994)													10,400
Total HLW Inventory at 2015													16,342

and placed in stainless steel canisters. Under U.S. Nuclear Regulatory Commission (NRC) regulations, calcine is not considered to be an adequate final waste form because of the occurrence of significant leaching of radionuclides on contact with water (i.e., calcine would not pass a toxicity characteristic leaching procedure (TCLP) test using EPA requirements). This regulation may prohibit disposal of the calcine. The calcination process may be classified as an interim best demonstrated available technology by EPA under RCRA, pending development of a process to produce a stable glass, ceramic, or glass/ceramic that will meet both Atomic Energy Act and RCRA requirements. The final treatment process and waste form for the HLW at INEL is under investigation. Evaluations of various waste immobilization processes are underway. Characteristics of the INEL waste suggest the final waste form for disposal will be a borosilicate glass material, which is assumed in this analysis.

The estimated number of HLW canisters to be produced from immobilization of the HLW at the INEL depends on the performance of separation and treatment processes implemented to treat the HLW, and the canister size. For purposes of this analysis, an estimated 1,700 canisters was assumed to be produced from treating existing HLW. This value is derived from a number of references, as presented below.

The most recent NEPA documentation concerning HLW activities at the INEL is the *Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995c). The Record of Decision for this EIS (May 30, 1995) states in the first paragraph on page 15 that "The technology selecting (for HLW treatment) is radionuclide partitioning for radioactive liquid and calcine waste treatment . . . and glass (vitrification) for immobilizing the resulting high-activity waste stream."

The number of HLW canisters that would result from the various processing options is not presented in this document. Information concerning HLW immobilization that allows calculation of the number of HLW canisters at the INEL is, however, presented in Section C-4.3.2 ("Waste Immobilization Facility") of Volume 2, Part B of the above-cited document. Table C-4.3.2-1 on page C-4.3.2-7 presents the volume of high-activity waste that would result depending upon the degree of pretreatment and immobilization technology chosen. The option listed in Table C-4.3.2-1 consistent with the ROD is Option 3a (i.e., the high-activity waste form would be glass); the final waste volume for the high-activity waste form (which would be disposed in the national repository as HLW) for Option 3a is 870 m^3. Based on an 85% fill level and a standard canister size of 0.61 m^3 (from the source of the values in Table C-4.3.2-1 [WINCO 1994]), the number of HLW canisters at the INEL is estimated to be:

$$(870\ m^3) / 0.85 / (0.61\ m^3 \text{ per canister}) \cong 1{,}680 \text{ canisters.}$$

The *Integrated Data Base Report-1994: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections, and Characteristics* (DOE 1995e) states on page 48 that HLW canister production is assumed to start in the year 2019 and continue through 2045. Table 2.7 on page 64 of the same reference presents the canister production schedule for the INEL, consistent with this projected schedule. Summation of the annual canister production rates (27 canisters produced the first year, 53 canisters the second year, 33 canisters the third year, and 66 canisters annually for the remaining years) results in the following total number of HLW canisters produced at the INEL:

$$27 + 53 + 33 + \{ 66 \times (2045 - 2022 + 1) \} = 1{,}697 \text{ canisters},$$

which is consistent with the totals calculated from the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995c). A conservative value of 1,700 canisters is applied in this analysis.

The immobilization facility at the ICPP is assumed to begin operation in 2015 with an average annual production rate of about 81 canisters per year. All HLW would be treated so that it is ready to be moved out of Idaho by a target date of 2035 (*Public Service Company of Colorado v. Philip E. Batt*, U.S. District Court for the District of Idaho, Oct. 17, 1995).

The dimensions of the proposed canister proposed in this analysis for disposal of the glass waste form are 0.7-m (27.5-in.) outer diameter and 3.28-m (129-in.) high, with a projected useable volume of 1.25 m^3 (44 ft^3). The canister dimensions are similar to the proposed Yucca Mountain "hybrid" canister (E.R. Johnson Associates 1989) that is 0.7 m (28 in.) by 4.76 m (187.5 in.) and designed to hold four boiling water reactor (BWR) and three pressurized water reactor (PWR) assemblies. Based on the glass formulations provided in Table 2.17 of ORNL (1995), the glass density varies from 2.36 to 2.69 g/cm^3. Assuming an average glass density of 2.55 g/cm^3, a single INEL canister would contain approximately 2,900 kg (6,400 lb) of glass. Table 2.15 gives the estimated radionuclide content of an INEL canister; Table 2.16 gives the characteristics of the immobilized HLW form and canister. Although the final chemical composition is currently unknown, the glass product is expected to be composed of 50-60 wt% silica, 8-13 wt% boron oxides, 2-20 wt% alkali oxides, 20-45 wt% waste oxides, and 0-7 wt% other additives (ORNL 1995).

Table 2.17 presents the estimated radioactivity and thermal power per canister as functions of decay time ranging from zero to one million years. The values given represent the average radioactivity of the fission and activation products per canister expected to be generated in the future. The heat generation rate of a single canister is less than the restriction of 800 W per canister at the time of shipment to the repository.

TABLE 2.15 Radionuclide Content per INEL Canister

Radionuclide	Mass (g/canister)	Radioactivity (Ci/canister)	Thermal Power (W/canister)
Sr-90	1.5E+01	2.1E+03	2.4E+00
Y-90	3.8E–03	2.1E+03	1.2E+01
Ru-106	7.4E–19	2.5E–15	1.5E–18
Rh-106	6.9E–25	2.5E–15	4.7E–17
Cs-134	3.1E–09	4.0E–06	4.0E–08
Cs-137	2.9E+01	2.5E+03	2.5E+00
Ba-137m	4.4E–06	2.4E+03	9.3E+00
Ce-144	4.6E–22	1.5E–18	9.6E–22
Pr-144	1.9E–26	1.5E–18	1.1E–20
Pm-147	1.9E–07	1.8E–04	6.5E–08
Eu-154	2.4E–03	6.6E–01	6.0E–03
Total	4.4E+01	9.0E+03	2.6E+01

Source: Adapted from ORNL (1995).

2.2.4 West Valley Demonstration Project

The 1994 inventory of HLW at WVDP included approximately 2,180 m^3 (580,000 gal) (24.7 million Ci of mixed fission products and transuranic radionuclides) stored in two underground tanks (Brown and Gordon 1988; ORNL 1995). One tank contained 2,040 m^3 (about 540,000 gal) of liquid alkaline waste while the other tank contained 140 m^3 (37,000 gal) of solid waste composed of both alkaline sludge and inorganic zeolite ion-exchange material contaminated with radioactive cesium.

Table 2.18 shows the current HLW inventory and projected levels at WVDP. No new HLW has been generated at WVDP for several years, and no additional HLW will be produced. Thus, the volume projections represent the effects of pretreating the existing inventory to reduce the quantity of material requiring vitrification and the effects of the vitrification procedure itself.

HLW pretreatment at WVDP has been completed. Supernatant in the alkaline tank was treated by ion exchange to remove cesium. The sludge has been washed to remove nonradioactive, soluble components. The sludge-washing process was completed in 1993. The final THOREX wash has recently been performed. The decontaminated supernatant and the sludge wash water will

TABLE 2.16 Projected Characteristics of Immobilized HLW Form and Canister at INEL

High-Level Waste Form and Canister Characteristics	
Waste form	Borosilicate glass in closed container
Canister material	Stainless-steel type 304L
Glass-ceramic density (g/cm^3)	2.55
Weight per canister (kg)	
Empty canister	985[a]
Glass	2,932[a]
Total loaded weight	3,917
Glass volume per canister (m^3)	1.15[a]
Canister dimensions (cm)	
Outside diameter	64.1[b]
Height overall	475[b]
Fraction of fill void (%)	77.5[b]
Average radioactivity	
Ci/canister	9,000[c,d]
W/canister	26[c,d]

[a] Source: adapted from ORNL (1995).

[b] Repository waste container based on commercial waste canister (Bolon et al. 1991); adapted from ORNL (1995).

[c] At time of immobilization; adapted from ORNL (1995).

Source: adapted from Salmon and Notz (1991).

eventually be disposed of as LLW. The washed sludge, spent ion-exchange media, and the acidic liquid will be combined, mixed with glass-forming chemicals, and sent to the melter for vitrification.

The plant began vitrification operations in 1996 with an expected annual production of about 100 canisters per year. Three HLW canisters have been produced as of July 8, 1996. Each canister contains 0.7 m^3 (25 ft^3) of borosilicate glass incorporating the HLW solids.

The total number of HLW canisters at the WVDP applied in this analysis is 340. This value is derived from the document *Draft Environmental Impact Statement for Completion of the West Valley Demonstration Project and Closure or Long-Term Management of Facilities at the Western New York Nuclear Service Center* (DOE and NYSERDA 1996). The total of 340 HLW canisters of vitrified HLW at the WVDP includes canisters containing spent fuel fines, Nuclear Regulatory Commission-Licensed Disposal Area (NDA) fuel assemblies, and HLW tank sludge. The fuel fines

TABLE 2.17 Radioactivity and Thermal Power per INEL HLW Canister over Time

Decay Time after Immobilization (yr)	Total Radioactivity per Canister (Ci)	Total Thermal Power per Canister (W)
0	9,030	26
1	7,400	20
10	4,400	12
100	530	1.4
1,000	0.32	1.7E–04
10,000	0.30	1.6E–04
100,000	0.24	1.1E–04
1,000,000	0.056	1.1E–05

Source: Scaled from ORNL (1992a) on the basis of a load of 2,932 kg borosilicate glass/canister.

TABLE 2.18 Current and Projected HLW Inventory at WVDP

Year	Volume (1,000 m^3)	Radioactivity (10^6 Ci)	Thermal Power (10^3 W)	Number of Canisters (cumulative)
1990	1.23	27.3	81	0
1994	2.18	24.7	78	0
2000	0.21	21.8	64	340
2005	0.21	19.5	57	340
2010	0.21	17.4	51	340
2015	0.21	15.5	46	340
2020	0.21	13.8	41	340

Source: ORNL (1992b; 1995).

(a total of 14 canisters) were assumed in this analysis to be HLW (for conservatism), although its classification is not yet known.

Table 2.19 gives the radionuclide contact for a canister containing 1,900 kg of HLW glass, on the basis of radioactivity levels at the beginning of 1990. The average radioactivity level in each canister at the end of 1998 (when vitrification is expected to be completed) is estimated to be 110,000 Ci and the average heat content of 330 W. The average activity of a WVDP HLW canister may be lower than 110,000 Ci due to the presence of the spent fuel fines and HLW tank sludge. The activity level of the spent fuel fines (approximately 80 Ci per canister) is three orders of magnitude lower than that of the vitrified HLW, and as such, will result in lower occupational doses during the loading/unloading and interim storage phases than those provided in the WM PEIS. For conservatism, an average canister activity of 110,000 Ci is applied in this analysis towards the total WVDP inventory of 340 canisters.

During the approximate 30 months of anticipated operation of this facility, a total of about 340 canisters will have been filled. By the end of 2020, the average canister values will have decayed to approximately 54,000 Ci and 160 W.

The canisters holding the HLW-loaded borosilicate glass will be made of stainless steel (type 304L) with an outside diameter of 61 cm, a height of 300 cm, and a wall thickness of 0.34 cm. The borosilicate glass will contain about 24% plutonium/uranium extraction (PUREX) and thorium extraction (THOREX) wastes and about 17% cesium-loaded zeolite. Table 2.20 provides further summary information on the canister and its contents.

Table 2.21 gives the calculated radioactivity and thermal power per canister. Table 2.22 gives estimates of neutron production rates in a canister from the (alpha, n) reaction and from spontaneous fission.

TABLE 2.19 Radionuclide Content per WVDP Canister

Radionuclide	Mass (g/canister)	Radioactivity (C/canister)	Thermal Power (W/canister)
Fe-55	1.1E-03	2.8	9.3E-05
Co-60	2.7E-03	3.0	4.7E-02
Ni-59	5.5	4.2E-01	1.7E-05
Ni-63	4.9E-01	3.0E+01	3.0E-03
Se-79	2.0E-01	1.4E-02	3.4E-06
Sr-90	1.9E+02	2.6E+04	3.1E+01
Y-90	4.8E-02	2.6E+04	1.5E+02
Zr-93	4.3E+02	1.1	1.2E-04
Nb-93m	2.5E-03	7.2E-01	1.3E-04
Tc-99	2.5E+01	4.3E-01	2.1E-04
Ru-106	1.7E-05	5.5E-02	3.3E-06
Rh-106	1.6E-11	5.5E-02	5.3E-04
Pb-107	8.4E+01	4.3E-02	2.6E-06
Cd-113m	3.8E-02	8.3	1.4E-02
Sn-121m	1.2E-03	6.9E-02	1.4E-04
Sn-126	1.4E+01	4.1E-01	5.1E-04
Sb-125	2.8E-02	2.9E+01	8.9E-02
Sb-126	6.9E-07	5.7E-02	1.1E-03
Sb-126m	5.2E-09	4.1E-01	5.2E-03
Te-125m	3.9E-04	7.0	5.9E-03
Cs-134	1.6E-02	2.0E+01	2.1E-01
Cs-135	5.5E+02	6.3E-01	2.1E-04
Cs-137	3.3E+02	2.8E+01	3.1E+01
Ba-137m	5.0E-05	2.7E+04	1.1E+02
Ce-144	8.0E-07	2.6E-03	1.7E-06
Pr-144	3.4E-11	2.6E-03	1.9E-05
Pm-146	9.6E-05	4.3E-02	2.1E-04
Pm-147	3.7E-01	3.5E+02	1.2E-01
Sm-151	1.3E+01	3.3E+02	3.9E-02
Eu-152	8.3E-03	1.4	1.1E-02
Eu-154	1.4	3.8E+02	3.4
Eu-155	2.0E-01	9.4E+01	6.8E-02
Ti-207	1.7E-10	3.2E-02	9.4E-05
Ti-208	4.3E-11	1.3E-02	3.0E-06

TABLE 2.19 (Cont.)

Radionuclide	Mass (g/canister)	Radioactivity (C/canister)	Thermal Power (W/canister)
Pb-209	1.8E-10	8.3E-40	9.5E-07
Pb-211	1.3E-09	3.2E-02	9.7E-05
Pb-212	2.5E-08	3.5E-02	6.7E-05
Bi-211	7.7E-11	3.2E-20	1.3E-03
Bi-212	2.4E-09	3.5E-02	6.0E-04
Bi-213	4.3E-11	8.3E-04	3.5E-06
Po-212	1.3E-19	2.3E-02	1.2E-03
Po-213	6.2E-20	7.9E-04	4.0E-05
Po-215	1.1E-15	3.2E-02	1.4E-03
Po-216	1.0E-13	3.5E-02	1.4E-03
At-217	5.1E-16	8.3E-04	3.5E-05
Rn-219	2.5E-12	3.2E-02	1.3E-03
Rn-220	3.8E-11	3.5E-02	1.3E-03
Fr-221	4.7E-12	8.3E-04	3.2E-05
Fr-223	1.2E-11	4.3E-04	1.1E-06
Ra-223	6.3E-07	3.2E-02	1.1E-03
Ra-224	2.2E-07	3.5E-02	1.2E-03
Ra-225	2.1E-07	8.3E-04	5.8E-07
Ra-228	2.6E-05	6.0E-03	4.6E-07
Ac-225	1.4E-08	8.3E-04	2.9E-05
Ac-227	4.5E-06	3.2E-04	1.6E-07
Ac-228	2.7E-09	6.0E-03	5.2E-05
Th-227	1.0E-06	3.2E-02	1.2E-03
Th-228	4.3E-05	3.5E-02	1.2E-03
Th-229	3.9E-03	9.3E-04	2.5E-05
Th-230	1.2E-02	2.4E-04	6.7E-06
Th-231	6.7E-10	3.5E-04	2.0E-07
Th-232	5.9E+04	6.5E-03	1.6E-04
Th-234	1.4E-07	3.1E-03	1.3E-06
Pa-231	1.3	6.0E-02	1.8E-03
Pa-233	4.4E-06	9.2E-02	2.1E-04
Pa-234m	4.6E-12	3.1E-03	1.6E-05
U-232	1.3E-03	2.7E-02	8.7E-04
U-233	3.7	3.6E-02	1.0E-03
U-234	2.6E-02	1.7E-02	4.7E-04

TABLE 2.19 (Cont.)

Radionuclide	Mass (g/canister)	Radioactivity (C/canister)	Thermal Power (W/canister)
U-235	1.6E+02	3.5E-04	9.3E-06
U-236	1.7E+01	1.1E-03	3.0E-05
U-238	9.3E+03	3.1E-03	8.0E-05
Np-236	2.8	3.7E-02	7.5E-05
Np-237	1.3E+02	9.2E-02	2.8E-03
Np-239	2.1E+01	1.4	3.3E-03
Pu-236	6.2E-06	3.3E-03	1.1E-04
Pu-238	1.9	32.6	1.1
Pu-239	1.0E+02	6.4	2.0E-01
Pu-240	2.1E+01	4.7	1.5E-01
Pu-241	3.1	3.2E+02	9.8E-03
Pu-242	1.7	6.4E-03	1.9E-04
Am-241	6.1E+01	2.1E+02	7.0E-02
Am-242	1.4E-06	1.2	1.3E-03
Am-242m	1.2E-01	1.2	4.6E-04
Am-243	6.8	1.4	4.4E-02
Cm-242	2.9E-04	9.6E-01	3.5E-02
Cm-243	1.0E-02	5.3E-01	1.9E-02
Cm-244	3.7E-02	3.0E+02	1.0
Cm-245	2.0E-02	3.5E-03	1.1E-04
Cm-246	1.3E-03	3.9E-04	1.3E-05
Total	≈7.0E+04	≈1.1E+05	≈3.3E+02

Source: ORNL (1992a).

TABLE 2.20 Projected Characteristics of the Immobilized HLW Form and Canister at WVDP

High-Level Waste Form and Canister Characteristics	
Waste form	Borosilicate glass in sealed canister
Canister material	Stainless steel type 304L
Borosilicate glass density (g/cm^3)	2.7
Weight per canister (kg)	
Empty canister	234
Cover	18
Borosilicate glass	1,900
Total	2,152
Reference fill	85%
Waste loading in glass (wt%)	40
Glass volume per canister (m^3)	0.7
Canister dimensions (cm)	
Outside diameter	61
Overall height	300
Wall thickness	0.34
Radioactivity (as of 12/91)	
Ci/canister (average)	104,300
W/canister	311
Radioactivity 50 years after filing	
Ci/canister	33,890
W/canister	105
Neutron production (neutron/s/canister)	
Spontaneous fission	4,155,000
(Alpha, n)	11,070,000
Total	15,230,000

Source: ORNL (1992a).

TABLE 2.21 Radioactivity and Thermal Power per WVDP Canister over Time

Decay Time after End of 1989 (yr)	Calculated Radioactivity per Canister (Ci)	Calculated Thermal Power per Canister (W)
0	109,600	326
1	106,900	319
2	104,300	311
5	97,080	290
10	86,230	258
15	76,660	230
20	68,180	205
30	53,970	164
50	33,890	105
100	10,730	37
200	1,260	8.9
300	291	5.4
350	202	4.8
500	128	3.7
1,000	63	1.9
1,050	60	1.8
2,000	26.7	0.70
5,000	15.8	0.33
10,000	13.2	0.26
20,000	10.3	0.18
50,000	6.8	0.08
100,000	5.2	0.04
500,000	3.8	0.04
1,000,000	3.1	0.03

Source: ORNL (1992a).

TABLE 2.22 Detailed Rates of Neutron Production per WVDP Canister (neutrons/s)[a]

Actinide	(Alpha, n)	Spontaneous Fission	Total
U-238	4.0E+01	1.2E+02	1.6E+02
Pu-238	3.0E+04	5.0E+03	3.5E+04
Pu-239	4.7E+03	2.8	4.7E+03
Pu-240	3.6E+03	1.9E+04	2.3E+04
Pu-242	4.0	2.8E+03	2.8E+03
Am-241	2.1E+05	7.6E+01	2.1E+05
Cm-242	1.3E+03	6.3E+03	7.9E+03
Cm-244	2.8E+04	3.4E+06	3.4E+06
Total	3.0E+05	3.4E+06	3.7E+06

[a] Neutron production rates are uniform source terms in the glass rather than dose rates at the exterior and are calculated for the end of 1989.

Source: ORNL (1992a).

3 FACILITY ASSESSMENT

3.1 HANFORD SITE: EXISTING FACILITIES

3.1.1 History

The Hanford site began storing HLW in underground tanks in 1944. The waste came from a variety of sources, including plutonium and uranium recovery from irradiated SNF. Hanford's HLW is contained primarily in 177 underground storage tanks built between 1943 and 1986. These tanks range in capacity from 208 m^3 (55,000 gal) to more than 3,800 m^3 (1 million gal). The waste at Hanford is found in four main forms: sludge, salt cake, slurry (i.e., a combination of liquid and suspended solid waste), and liquid.

3.1.2 Facility Descriptions and Environmental Releases

The chemical processing plants (T-Plant, U-Plant, Z-Plant, and Reduction/Oxidation [REDOX] Plant) and the associated tank farms (T, TY, TX, U, S, SY, and SX) are located in the 200 West Area at Hanford (DOE 1992b). The B-Plant and PUREX Plant, along with their associated tank farms (A, AN, AP, AW, AX, AY, AZ, B, BX, BY, and C) are located in the 200 East Area (Figures 3.1 and 3.2). The chemical processing plants were constructed between 1943 and 1955; the tank farms were constructed between 1943 and 1964.

The basic tank design of the first tank farms (T, TY, TX, U, S, SX, A, AX, B, BX, BY, and C) was a reinforced-concrete, domed shell with an open top carbon-steel liner known as a SST. Between 1943 and 1964, 149 SSTs were built in the 200 East and 200 West areas for storing radioactive wastes. The SSTs are located in 12 tank farms of 4 to 18 tanks each. Most SSTs are 22.9 m (75 ft) in diameter and have a capacity ranging from 2,000 to 3,800 m^3 (530,000 to 1 million gal), depending on the depth of the tank (with the exception of 16 smaller 208-m^3 [55,000-gal] tanks in the first tank farms—B, C, T, and U). The SSTs can be categorized into one of four types on the basis of size and construction. Three of the four SSTs are 22.9 m (75 ft) in diameter, have a domed top made of reinforced concrete, and can hold volumes of approximately 2,000 m^3 (530,000 gal), 2,900 m^3 (766,000 gal), and 3,800 m^3 (1 million gal), respectively. The fourth tank type has a 6.1-m (20-ft) diameter, a flat, reinforced concrete top, and a volume of about 210 m^3 (55,000 gal).

Unit characteristics are given in Table 3.1. Forced ventilation currently provides cooling for 16 tanks that contain materials that generate heat from radioactive decay. Single-stage high efficiency particulate air (HEPA) filters allow atmospheric breathing for tanks that do not require cooling. The radioactive waste stored in these underground tanks has come from various sources,

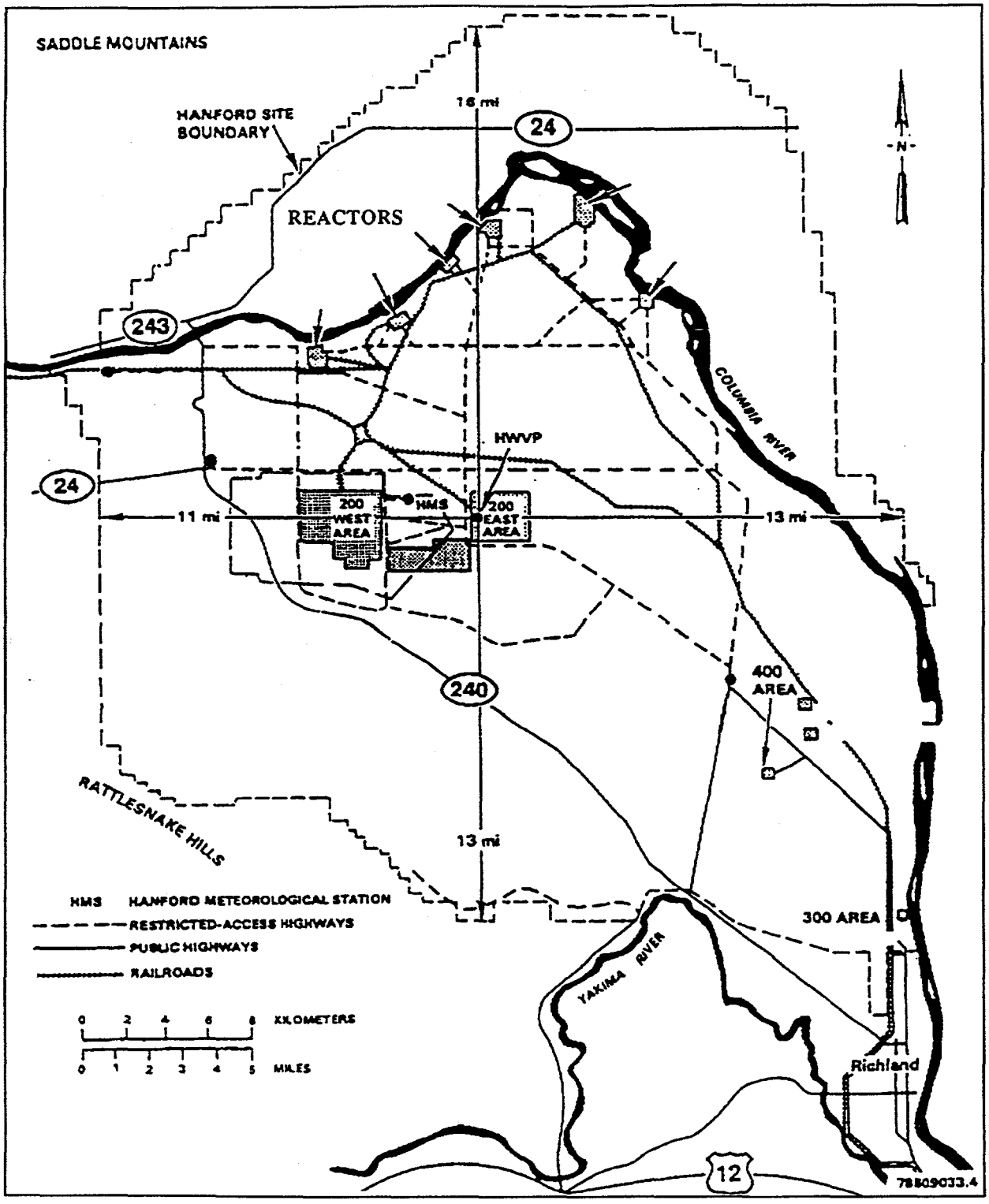


FIGURE 3.1 Map of the Hanford Site (Source: DOE 1992b)

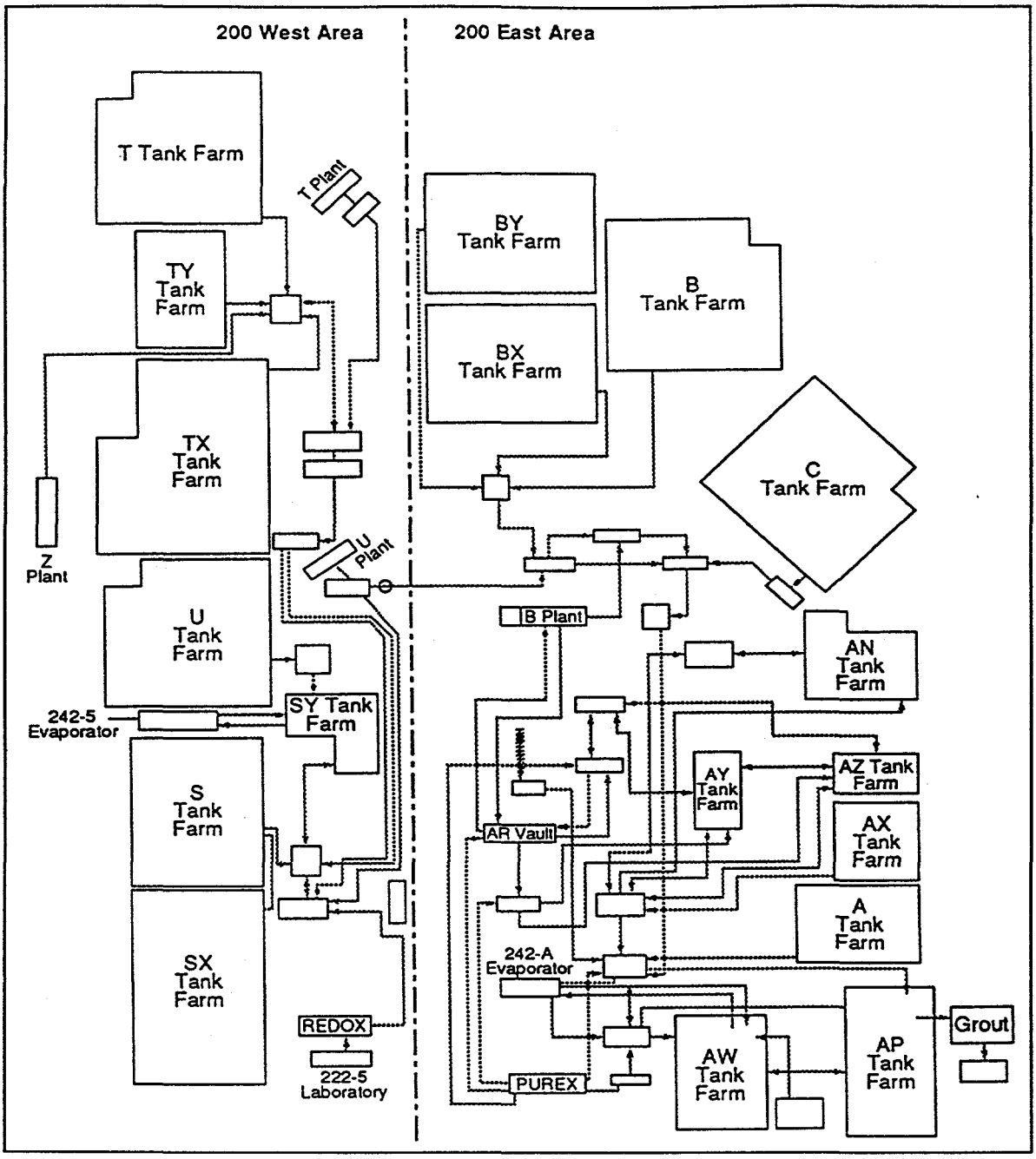


FIGURE 3.2 Schematic of the 200 West and 200 East Chemical Process Areas (Source: DOE 1992b)

TABLE 3.1 Current Single-Shell Tank (SST) Storage at Hanford[a]

Hanford Tank Type	IV	III	II	I
Storage method	Underground reinforced concrete shell with carbon-steel liner on bottom and sides; top made of reinforced concrete			
Tank top type	Dome	Dome	Dome	Flat
Number of tanks	25	48	60	16
Tank capacity (m^3)	3,785	2,870	2,018	208
Surface area of unit	Diameter: 22.9 m Height: 13.2 m Surface: ≈1,770 m^2	Diameter: 22.9 m Height: 11.5 m Surface: ≈1,650 m^2	Diameter: 22.9 m Height: 8.9 m Surface: ≈1,460 m^2	Diameter: 22.9 m Height: 8.3 m Surface: ≈220 m^2
Number of tanks at 200 East Farm	10	12	36	8
Number of tanks at 200 West Farm	15	36	24	8
Distance below ground	Top of tank is 2.1 to 2.5 m below grade.	Top of tank is 2.4 m below grade.	Top of tank is 1.8 to 2.7 m below grade.	Top of tank is 1.8 m below grade.
Approximate distance to water table	Varies from 42.1 to 70.1 m above water table	Varies from 43.9 to 60 m above water table	Varies from 50.3 to 66.1 m above water table	Varies from 53.6 to 69.5 m above water table
Curie content of waste tanks	47 million Ci	910,000 Ci	590,000 Ci	18 million Ci

[a] Assumes tanks are cylinders for calculating surface area.

Sources: DOE (1987); Opitz et al. (1990)

including three different plutonium and uranium recovery processes, three different radionuclide recovery processes from waste, and various associated sitewide operations. All SST wastes are from various site activities prior to 1980. The different processes that discharged waste into the SSTs include the bismuth phosphate ($BiPO_4$), REDOX, and PUREX processes.

Some of the SSTs at Hanford will soon exceed their original operating life of 50 years. The integrity of the SST concrete structure was evaluated in the late 1970s and early 1980s. It was concluded that the SST concrete structure is adequate to support proposed disposal plans, although the tanks could not be expected to contain any liquid materials. A total of 67 SSTs are known or suspected to have leaked. No waste has been added to the SSTs since 1980, and the pumpable liquids have been removed from 115 SSTs (as of July 1996) and transferred to DSTs through a jet-pumping program. Current plans call for removal of pumpable liquids from all SSTs. The remaining waste will be primarily sludge and saltcake. Saltcake can be characterized as a hard, dry crystalline solid, while sludge is a wet solid. Saltcake is made of various salts, primarily sodium salts of hydroxide, nitrate, nitrite, carbonate, aluminumate, and phosphate. About 83% of the saltcake consists of sodium nitrate and nitrite salts. The non-saltcake portions of the SST sludge are primarily metal oxides and metal hydroxides formed by precipitation from the caustic SST solution. The average bulk density ranges from 1.4 to 1.7 g/cm^3. The radioactive components consist of fission products, a small amount of activation products, and actinide elements such as uranium, thorium, plutonium, americium, and neptunium. About 99% (by activity) of the radionuclide content is Sr-90, Cs-137, and their radioactive progeny (see Table 3.2).

The DSTs containing the wastes that will be processed by the HWVP are in both the 200 East and 200 West areas at Hanford. The tanks in tank farms AN, AP, AW, AY, AZ, and SY are DSTs. The DSTs are constructed with both primary and secondary steel liners within the concrete shell. The annulus between the liners permits leak detection and leak confinement should the primary liner ever develop a leak. As of December 1992, however, no leaks have been detected (GAO 1993). The DST design was adopted in 1968 at Hanford, and all tanks built later than 1968 utilize this design. Twenty DSTs, each with a volume of 3,800 m^3 (1 million gal) to 4,300 m^3 (1,140,000 gal), were constructed between 1970 and 1982; eight additional tanks were constructed in the AP tank farm. Most tanks are 22.9 m (75 ft) in diameter with a height of 14.9 m (49 ft) (DOE 1987). The high-level radioactive wastes contained within the DSTs that will be vitrified are (1) NCAW, (2) neutralized cladding removal waste (NCRW), (3) Plutonium Finishing Plant (PFP) waste, and (4) complexant concentrate (CC) waste. The NCAW is a two-phase (solid-liquid) waste produced by neutralizing high-level first-cycle raffinate from the reprocessing of N-Reactor fuels at the PUREX Plant; it will be the primary feed to the vitrification plant. The NCRW is an alkaline solid-liquid waste resulting from chemical dissolution of the zirconium-alloy N-Reactor fuel cladding material; following dissolution, sodium hydroxide is added to the decladding waste. The CC waste contains organic compounds and their degradation products, which resulted from the use

TABLE 3.2 Radionuclide Inventory of Hanford Single-Shell Tank Waste

Radionuclide	Curies	Radionuclide	Curies
C-14	3.0E+03	Pa-233	6.0E+01
Ni-63	2.7E+05	Pa-234m	4.7E+02
Se-79	9.1E+02	U-233	5.7E-03
Sr-90	4.4E+07	U-234	1.2E-01
Y-90	4.4E+07	U-235	2.0E+01
Zr-93	3.9E+03	U-238	4.6E+02
Nb-93m	3.2E+03	Np-237	3.2E+01
Tc-99	1.1E+04	Np-239	3.1E+01
Ru-106	3.8E–02	Pu-238	4.6E+02
Rh-106	3.8E–02	Pu-239	2.2E+04
Sb-126	8.8E+01	Pu-240	5.4E+03
Sb-126m	6.3E+02	Pu-241	5.7E+04
Sn-126	5.7E+02	Am-241	2.7E+04
I-129	2.4E+01	Am-242	6.8E+01
Cs-135	7.3E+01	Am-242m	6.8E+01
Cs-137	1.2E+07	Am-243	1.9E+01
Ba-137m	1.1E+07	Cm-242	5.6E+01
Sm-151	6.8E+05	Cm-244	8.4E+01
Th-231	2.0E+01	Cm-245	1.0E-02
Th-234	4.7E+02	Total	1.2E+08

Sources: Adapted from TWRS EIS (DOE 1996a).

of chelating agents in strontium recovery processing. The PFP waste is a solid-liquid mixture generated during solvent extraction, ion exchange, plutonium nitrate-to-metal conversion, scrap stabilization, and laboratory operations at the PFP (Z-Plant).

Of the 177 underground storage tanks at Hanford, 149 are SSTs and 28 are DSTs. Vitrification will be performed on the waste in 10 DSTs. Current plans call for the waste in the remaining 18 DSTs to be treated as low-level mixed waste.

The individual tank farms are interconnected to diversion boxes and valve pits with buried transfer lines. Jumpers are used in the diversion boxes to connect the transfer lines necessary to accomplish the desired waste-transfer operations. Six cross-site transfer lines connect the 200 West and 200 East Area tank farms (DOE 1992b).

The tanks themselves contain ancillary equipment and instrumentation for waste monitoring, pumping, agitation, condensate recovery, and ventilation control. Table 3.3 gives the design characteristics of the various DST types; the radionuclide content of the DSTs is shown in Table 3.4. Table 3.5 shows the staff required to maintain the tanks as well as the doses that workers receive. The low worker dose rate results from the tanks being located below grade with at least 1.8 m of soil cover, which provides a measure of shielding to minimize radiation exposure to personnel. Personnel receive routine exposures through surveying and monitoring of the tanks.

Finally, Table 3.6 shows the release fraction to be used to calculate emissions from current storage. Airborne emissions occur during current storage primarily due to forced ventilation, tank breathing, and evaporation. Emissions can be calculated by multiplying the inventories given in Tables 3.2 and 3.4 by the release fraction given in Table 3.6:

$$[\textit{Emission rate}\ (\text{Ci/yr})]\textit{storage} = \{\ [\textit{Species in DST waste}\ (\text{Ci})] + [\textit{Species in SST waste}\ (\text{Ci})]\ \} \times [\textit{Release fraction}\ (\text{yr}^{-1})]. \quad (3.1)$$

Non-radiological emissions were determined to be insignificant (e.g., on the order of 1×10^{-12} tons per year of organics). Table 3.7 gives representative stack characteristics for atmospheric dispersion calculations.

The strontium and cesium capsules are currently stored in stainless-steel-lined concrete basins at the Waste Encapsulation and Storage Facility (WESF) under 4 m (≈13 ft) of demineralized water. The WESF is located in Building 225-B in the 200 East Area at Hanford. The water basins dissipate the decay heat and reduce the radiation exposure to the workers. Table 3.8 gives the worker dose rates and labor requirements for current storage of these capsules. The number of exposed workers for surveillance and current storage was determined from the total manpower requirement for 100 years of continued storage (1,300 person-yr) for the No Disposal Action alternative in DOE (1987). The worker dose rate was determined by dividing the occupational dose for continued capsule storage at Hanford [420 person-rem/100 yr (DOE 1987)] by the total manpower requirement (1,300 person-yr). The operation duration of about 13 years is based on canisterization starting in FY 2006 (ORNL 1992b).

Various processes are located in the WESF. Emission data for 1989 reports a total of less than 1×10^{-6} Ci/yr gross alpha and 1.04×10^{-5} Ci/yr gross beta (Brown et al. 1990). Both Sr-90 and Cs-137 (and their radioactive progeny) are beta emitters; alpha emissions will not be measured if the reported emissions are solely from the storage basins. In this report, it is assumed that insignificant emissions will occur during current storage because of the integrity and double encapsulation of the capsules.

TABLE 3.3 Current HLW Storage at Hanford[a]

Waste Type	NCAW	NCRW	PFP	CC
Storage method	DST: carbon-steel tank within a steel-lined concrete tank	DST: carbon-steel tank within a steel-lined concrete tank	DST: carbon-steel tank within a steel-lined concrete tank	DST: carbon-steel tank within a steel-lined concrete tank
Tank capacity	3,800 m^3	4,300 m^3	4,300 m^3	3,800 - 4,300 m^3
Number of tanks	2	2	1	5
Percent capacity filled	95% (average)	68% (average)	59%	90% (average)
Surface area of unit	Diameter: 22.9 m Height: 14.9 m Surface: ≈ 1,895 m^2	Diameter: 22.9 m Height: 14.9 m Surface: ≈ 1,895 m^2	Diameter: 22.9 m Height: 14.9 m Surface: ≈1,895 m^2	Diameter: 22.9 m Height: 14.9 m Surface: ≈ 1895 m^2
On-site location	200 East Area	200 East Area	200 West Area	200 West and 200 East areas
Distance below ground	Top of tank is 2.1 m below grade.	Top of tank is 2.1 m below grade.	Top of tank is 2.1 m below grade.	Top of tank is 2.1 m below grade.
Approximate distance to water table	Bottom of tank is 59.4 m above water table.	Bottom of tank is 69.5 m above water table.	Bottom of tank is 42.1 m above water table.	Varies from 42.1 to 65.8 m above water table (depends on specific tank).
Curie content of waste tanks	47 million Ci	910,000 Ci	590,000 Ci	18 million Ci

[a] Assumes tanks are cylinders for calculating surface area.

Sources: DOE (1987); Hanlon (1992); Johnson (1992).

TABLE 3.4 Radionuclide Inventory of Hanford Double-Shell Tank Waste[a]

Radionuclide	Curies	Radionuclide	Curies
H-3[b]	6.3E+03	Ba-137m	3.4E+07
C-14	4.4E+02	Ce-144	1.6E+05
Co-60	1.2E+03	Eu-152	3.0E+02
Ni-63	9.1E+03	Eu-154	3.6E+03
Sr-90	2.6E+06	Eu-155	3.7E+03
Y-90	2.6E+06	Np-237	2.1
Nb-95	2.6E+05	Pu-238	1.8E+03
Zr-95	1.4E+05	Pu-239	2.0E+04
Tc-99	6.1E+03	Pu-240	7.4E+03
Ru-106	1.8E+05	Pu-241	1.2E+05
Sb-125	4.2E+04	Am-241	5.4E+05
I-129	1.6	Pu-242	1.2E-01
Cs-134	4.8E+03	Cm-244	1.2E+02
Cs-137	3.6E+07	Total	6.3E+07

[a] NCAW, NCRW, PFP, and CC waste only — 10 DSTs.

[b] H-3 derived from C-14 inventory.

Source: Adapted from Lowe (1991).

TABLE 3.5 Estimated Labor Requirements for Current Storage of HLW in DSTs and SSTs at Hanford

Category	Number of Personnel (FTE[a])	Duration (yr)[b]	Dose Rate[c] (rem/yr)
Exposed workers	108	≈33	0.11

[a] Full-time equivalent.

[b] Duration estimated assuming completion of vitrification operations by FY 2028.

[c] Worker dose rate determined by dividing occupational dose for continued waste storage at Hanford by annual requirement: (1,200 man-rem/100 yr) ÷ (108 man-yr/yr).

Source: Adapted from DOE (1987).

TABLE 3.6 Atmospheric Releases of Radionuclides from Current HLW Storage at Hanford

Type of Release	Release Fraction
Tritium (H-3)	1.0E-03
All other radionuclides	5.0E-12

Source: Adapted from DOE (1987).

TABLE 3.7 Release Point Characteristics for Atmospheric Emissions for Current Storage and Transfer of HLW at Hanford

Stack Parameter	Value
Release point	296-A-40
Stack height (m)	4.1
Height above surrounding structures (m)	4.1
Stack diameter (m)	0.25
Stack exit velocity (m/s)	8.7
Exit temperature	Ambient

Source: DOE (1990a).

TABLE 3.8 Estimated Personnel Requirements and Exposure for Current Storage of Strontium and Cesium Capsules at Hanford

Category	Number of Personnel (FTE[a])	Duration (yr)	Dose Rate (rem/yr)
Exposed (all)	13	13	0.32

[a] Full-time equivalent.

Source: Adapted from DOE (1987).

Prior to vitrification in the HWVP, the waste in the DSTs and SSTs will be retrieved for pretreatment. The retrieval technology for the SST contents is the basis of a future study to develop a system that will clean the tanks to the desired level and at the desired rate, yet function in tanks that may leak. It is assumed for this study that transfer of the SST contents will be by a hydraulic retrieval system using slurry transfer (pumping) to convey the tank waste out of the tank. This retrieval approach is similar to that proposed for DST waste retrieval and has been successfully applied on SSTs in the past. There is some uncertainty in its future application due to the addition/control of water to facilitate waste removal during retrieval may not be allowed (Krieg et al. 1990). Table 3.9 gives the staff required for the transfer of the waste from the DSTs and SSTs to the pretreatment facility (and its associated exposure). The total number of personnel for SST retrieval has been reported to be 1,462 (Boomer 1992), of which 80 percent are exposed to radiation; the value of 0.44 rem/year for the worker dose rate is based on the maximum allowable dose rate for retrieval equipment operations personnel of 0.2 mrem/hour (Krieg et al. 1990) and an assumed 2,200 hours per person-year. Table 3.10 gives the atmospheric release fractions during the transfer process. Emissions during the transfer process can be calculated by using the following equation:

$$[\textit{Emission rate}\ (\text{Ci/yr})]\textit{retrieval} = \{\ [\textit{Species in DST waste}\ (\text{Ci})] + [\textit{Species in SST waste}\ (\text{Ci})]\ \} \times [\textit{Release fraction}]\ /\ 20\ \text{yr}, \qquad (3.2)$$

on the basis of a 20-year retrieval period (based on the planned high-level vitrification start-up and completion dates in the Hanford Federal Facility Agreement and Consent Order and the assumption that the pretreatment facility would be sized to accommodate the vitrification rate) and by using the inventories in Tables 3.2 and 3.4.

TABLE 3.9 Estimated Personnel Requirements and Exposure for Transfer of HLW at Hanford

Category	Number of Personnel (FTE[a])	Duration (yr)	Dose Rate (rem/yr)
Operations	1,194	20	0.44

[a] Full-time equivalent.

Source: Adapted from Boomer (1992); Krieg et al. (1990).

TABLE 3.10 Atmospheric Releases of Radionuclides from Transfer of HLW at Hanford[a]

Release Description	Release Fraction
Assume 10% of the liquid becomes airborne annually via evaporation and is released from the tank farm ventilation system with an overall decontamination factor of 5.0E+08.	2.0E-10 for gas 1.0E-01 for tritium

[a] The release characteristics at Hanford are taken to be the same as at WVDP. The release fraction for gases is calculated by dividing 10% by 5.0E+08. For tritium, the release fraction is 10% because the filter system is assumed to be ineffective in removing tritium.

Source: Adapted from DOE (1982b).

3.1.3 Current Status

Numerous tank waste activities are ongoing to provide for the continued safe storage of the tank waste until remediation measures are implemented. These activities consist of a number of routine activities as well as a number of additional activities required for safe storage.

Routine operations include management oversight, regulatory compliance and reporting activities, and operations and maintenance of facilities and equipment. Tank monitoring activities support waste management by gathering information on waste temperature, liquid levels, solid levels, and tank status. Leak detection activities involve in-tank liquid level monitoring, leak detection monitoring of the annulus for the DSTs, dry-well monitoring around tanks for increase in radioactivity levels, and groundwater monitoring.

Safety management activities include the following:

- Calculating operational waste volume projections that involve comparing projected waste volumes against tank capacity. The projections also provide for identification and management of risk that could negatively impact available tank storage space;

- Combining compatible waste types. Transferring tank waste between tanks and tank farms through the existing cross-site transfer system to provide the required tank space and to address safety issues;

- Implementing a waste minimization program to reduce the generation of new waste requiring storage in the tanks;

- Screening and characterizing the waste on a tank-by-tank basis to gather data in support of safety and remedial action design activities;

- Isolating and removing pumpable liquid from SSTs to reduce the potential of future leakage (interim stabilization by saltwell pumping); and

- Operating the 242-A Evaporator to concentrate waste and treat residual liquid to remove the contaminants.

On December 1, 1995, DOE and Ecology published their Record of Decision for the Safe Interim Storage EIS in the Federal Register (60 FR 61687). The decision was to implement most of the actions of the preferred alternative, including:

- Construct and operate a replacement cross-site transfer pipeline system;

- Continue operating the existing cross-site transfer pipeline system until the replacement system is operational; and

- Continue operating the mixer pump in tank 101-SY to mitigate the unacceptable accumulation of hydrogen and other flammable gases.

The existing cross-site transfer system has been used to transfer waste from the 200 West Area for 40 years. This underground pipeline system is at the end of its original design life. Currently, four of the six lines are out of service and unavailable to perform transfers because of plugging. The two useable lines do not meet current engineering standards, such as double containment and leak detection, which are required for waste management facilities. Construction of the replacement cross-site transfer system has begun.

The mixer pump in tank 101-SY was proven to be effective in mitigating the flammable gas as a safety issue in that tank during more than one year of operations. It was determined that the construction of new tanks to resolve safety concerns was not necessary.

Based on new information available to DOE regarding nuclear criticality safety concerns during retrieval, transfer, and storage actions, DOE has decided to defer a decision on the construction and operation of a retrieval system in tank 102-SY. Based on the quantities of plutonium in tank 102-SY sludge, retrieval of the solids falls within the scope of the criticality safety issues that will be evaluated over the next few months.

Currently, DOE is considering contracting with private companies for waste remediation services for the tank waste. DOE is interested in encouraging industry to use innovative approaches, and in using competition within the private marketplace to bring new ideas and concepts to tank waste remediation. The goal of the privatization effort is to streamline the TWRS mission, transfer a share of the responsibility, accountability, and liability to industry, improve performance, and reduce cost without sacrificing worker and public safety or environmental protection.

DOE plans on issuing contracts to perform the first phase of the work in late summer 1996. As currently envisioned, DOE would select contractors to construct and operate commercial demonstration facilities for two tank waste separations and low-activity waste (LAW) immobilization facilities, one of which may include a HLW vitrification facility. If these commercial demonstrations are successful, DOE may use the lessons learned from those demonstration facilities and proceed with contracting for full-scale facilities to remediate the remainder of the tank waste.

3.2 HANFORD SITE: PLANNED FACILITIES

3.2.1 Hanford High-Level Waste Vitrification Facility

The DOE is currently considering building two vitrification plants at Hanford, one for LLW and the other for HLW. The 1988 TPA among DOE, Washington State, and the U.S. Environmental Protection Agency (EPA) establishing cleanup milestones has been altered to now mandate that the contents of the SSTs be processed as well. The HLW vitrification facility will be designed to be similar to the DWPF at SRS; originally, it was designed to process four types of waste from 10 of the DSTs: (1) NCAW, (2) NCRW, (3) PFP waste, and (4) CC waste. The NCAW was produced from the reprocessing of N-Reactor fuels at the PUREX Plant. This waste contains approximately 80% of the radioactive inventory and would therefore be the primary high-heat source of feed to vitrification. It was expected to be the initial feed to the vitrification plant and is also the reference feed for the design of the HWVP. The remaining feeds are expected to follow in the order of NCRW, PFP, and CC.

The ability of the HLW vitrification facility to process both the DST and SST waste, will depend on the pretreatment process selected and the resulting quantities of HLW that will be fed to the facility.

Figure 3.3 shows the flowsheet for the borosilicate glass immobilization process at Hanford. The process consists of (1) transferring the HLW from current storage to a pretreatment process, (2) transferring the pretreated HLW to the vitrification facility for processing into canisters, (3) storage of the canisters in an interim storage facility, and (4) final disposal in a geologic repository.

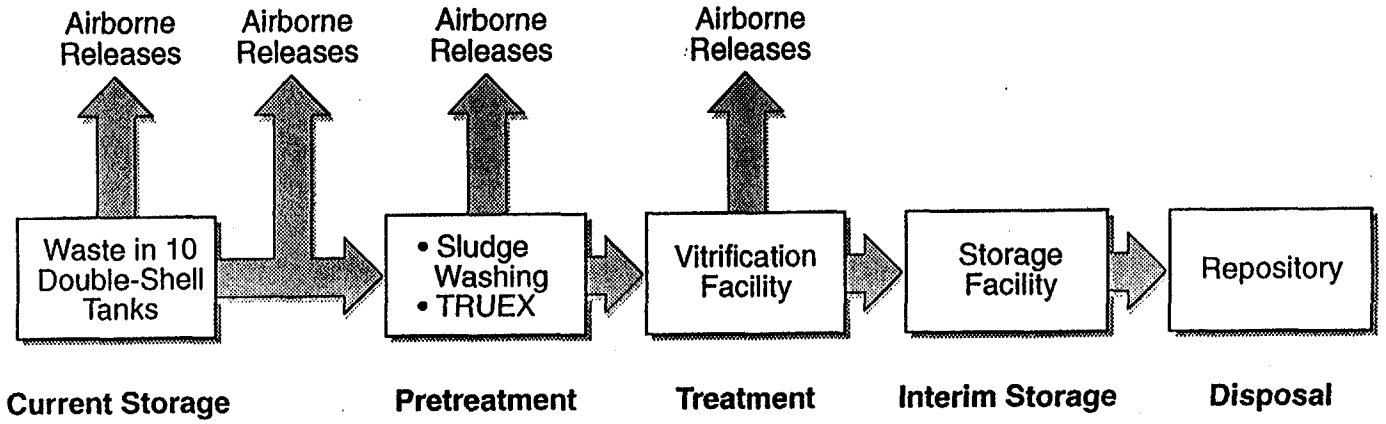


FIGURE 3.3 Flow Diagram of the HWVP Process

The HLW vitrification facility is intended to be a freestanding, single train plant designed to produce 20 metric tons of HLW glass per day. The HLW vitrification facility would have six operational areas that would include feed receipt and sampling, process evaporation, melter operations, maintenance areas, canister loading, cold chemical makeup, and off-gas processing. The facility would have an overall height of 45 m (150 ft), of which 13 m (45 ft) would extend below grade. The facility's dimensions would be 55 by 165 m (175 by 545 ft) with an area of 8,800 m^2 (94,700 ft^2).

The facility's process levels would contain feed receipt and sampling equipment, centrifuges, process evaporation equipment, melter operations equipment, and the maintenance area. The feed tanks would be located in an adjacent structure. Three other areas would provide the remainder of the support facilities. The 13 m (45 ft) level would house the canister loading and handling equipment. The +20 m (+65 ft) level would provide crane maintenance and cold chemical storage makeup.

The final HLW glass form would be a glass canister measuring 0.61 by 4.57 m (2.0 ft by 15.0 ft). The HLW interim on-site storage facility would allow enough interim storage space for all of the HLW glass produced.

The HLW vitrification facility would receive HLW slurry from the sludge washing operation and cesium solution from the LAW separation facility. After sampling, water would be removed first by centrifuging and then evaporating the centrate. The solids and the slurry from the evaporator would be recombined to feed the HLW melter feed system. The feed would be sampled and analyzed. Based on the resulting analyses, fluxes would be added to provide the desired vitreous product, a borosilicate glass that would contain 20 percent waste oxides.

A cold cap melter would use joule heating where current is passed through the molten charge that serves as the resistance element for the furnace. This type of furnace would have a crust over the surface of the melt, which receives the slurry feed, hence the term cold cap. The water in the slurry would be evaporated from the cold cap, and the dried waste would sink as the bottom of the cap enters the melt.

The hot glass would be semi-continuously poured into cylindrical stainless-steel canisters, which would be 0.61 m (2 ft) in diameter and 4.57 m (15 ft) high.

A canister would be moved from storage into position under a filling tube that would be lowered to mate tightly with the canister. The fill tube would contain a passage for molten glass to flow into the canister and a separate passage for air to vent out of the canister. The canister would be filled with molten glass. After canister filling was completed, the filled canister would then be transferred to the canister weld cell where it would be welded shut.

A transfer cart would move the canister into the decontamination cell from the weld cell. The crane would lift the canister from the cart and move it to a decontamination area. Decontamination solution would be sprayed onto the canister followed by a water rinse. After the canister is dried, the crane would transport it to the smear test cell, where the canisters would be smear-tested for surface contamination. If the canisters fail the test they would be returned to the decontamination cell. If the canisters pass the test they would be forwarded to the load-out cell and then transferred to interim storage pending shipment to a potential geologic repository for disposal.

It should be noted that the design information is at an early planning stage. The details are likely to change as the planning and design process matures.

Table 3.11 summarizes information on safety classifications of the high-level waste vitrification buildings and structures, the dimensions of the vitrification building, electric energy requirements, and approximate construction material requirements.

3.2.2 Schedules and Uncertainties

The processing rate of the HLW vitrification facility will be affected by the pretreatment option selected. Pretreatment reduces the amount of waste feed to the immobilization plant; the number of HLW canisters is therefore dependent on the extent of the pretreatment performed. The previous DOE approach was to remove strontium, cesium, technetium, and transuranic waste (TRUW) elements from the soluble salts, which would be then combined with sludges and other streams containing high concentrations of fission products and TRUW elements. Pretreatment may potentially include an ion-exchange process for extracting cesium and strontium from the liquid portion of the waste and sludge washing for dealing with the precipitate at the bottom of the tanks. The exact pretreatment option is to be assessed between 1994 and 1997; a design for the HLW pretreatment facility is to be completed by March 1998. Based on an estimated total of 15,000 Hanford HLW canisters and the planned high-level vitrification start-up and completion dates (December 2009 and December 2028, respectively) in the Hanford Federal Facility Agreement and Consent Order, the average annual processing rate is estimated to be approximately 790 canisters per year.

3.3 SAVANNAH RIVER SITE: EXISTING FACILITIES

3.3.1 History

The SRS was established in 1950 by the U.S. Atomic Energy Commission to produce nuclear materials for the nation's defense. The production of these nuclear materials has resulted in

TABLE 3.11 Resource Requirements for Construction of the HLW Vitrification Facility and Support Structures at Hanford

Hanford High-Level Waste Vitrification Facility	
Safety classifications	
Vitrification Building	1[a]
Auxiliary structures	3[b]
Number of levels in Vitrification Building	4[c]
Exterior walls of Vitrification Building	Reinforced-concrete, shielded construction
Vitrification Building dimensions	
Length (m)	165
Width (m)	55
Height (m)	45
Building footprint (m^2)	8,800
Annual electric energy requirement (GWh)	620
Approximate Construction Material Requirements	
Concrete (m^3)	715,000
Steel (t)	142,000
Stainless steel	24,400
Hastelloy/Inconel (t)	3,130
Water (m^3)	678,000
Diesel (m^3)	28,400

[a] Exterior concrete walls and roof slab are Safety Class 1. Some parts interior to the building are Safety Classes 2 and 3. A discussion of Hanford's facility usage categories is given in Wodrich et al. (1991).

[b] The sand filter building is Safety Class 1. Inter-area transfer lines are Safety Class 2.

[c] First level (13 m) is below grade.

Sources: DOE (1996a).

radioactive waste that has been stored at SRS. In the early 1980s, DOE initiated efforts to end the storage of the liquid high-level radioactive waste at SRS by developing plans to treat the waste and convert it into a more stable glass form that would then be shipped to a geologic repository for permanent disposal. Figure 3.4 is a schematic of the HLW management approach to be implemented at SRS.

The high-level radioactive waste that was generated during reprocessing of SNF and target materials was transferred from the separations facilities to large underground tanks (McIntosh and Papouchado 1990). Prior to storage in 51 underground carbon-steel tanks, the waste stream was alkalinized to pH 13 for corrosion prevention. The fresh waste consisted of two phases: liquid and solid. The liquid phase is a solution of salts in water under alkaline conditions. The solid phase is composed of chemicals that are insoluble in alkaline solution. The bulk of these solids settles in the storage tanks and is referred to as "sludge." The sludge phase consists primarily of oxides and hydroxides of aluminum, iron, and manganese. Except for Cs-137, virtually all radionuclides can be found in the sludge. The sludge contains more than 60 percent of the radionuclides. To minimize the volume of waste stored, the salt solution is concentrated by evaporation, which reduces the waste volume by about 60%. The chemical salts in the waste crystallize out and a salt cake forms.

3.3.2 Facility Descriptions and Environmental Releases

Existing HLW facilities at SRS include the Liquid Radioactive Waste Handling Facilities (LRWHFs) and the DWPF (Figure 3.5). The LRWHFs, commonly referred to as the tank farms, are located in the F- and H-areas at SRS; the tank farms occupy about 20 ha (50 acres) of the 80,000-ha (198,000-acre) SRS. The tank farm facilities consist of underground waste tanks with a nominal capacity of 3,800 m^3 (1 million gal) each. The HLW tank farms also maintains four evaporators for waste concentration (two are currently operational), transfer pipelines, 14 diversion boxes, 13 pump pits, and associated tanks, pumps, and piping for transferring the waste (DOE 1995a).

The HLW tanks are located in two tank farms in the 241-F and 241-H areas. The F-Area farm accommodates 22 tanks, and the H-Area farm houses 29 tanks. The tanks can be one of four different types; each type has unique design characteristics. The HLW tanks are built of carbon steel and reinforced concrete. Types I through III are constructed of double-walled carbon-steel with forced cooling (outer diameter ranging from 23 to 26 m [75 to 85 ft]) and between 7.5 and 10 m [24.5 and 33 ft] high); Type IV tanks have a single carbon-steel wall without forced cooling (26-m [85-ft] outer diameter by 10-m [33-ft] high). Tank types I, II, and III are used primarily for high-heat waste; Type IV tanks are reserved for low-heat waste. The Type III design is the newest design and is most commonly used for receipt of fresh waste from chemical processing operations (Hobbs et al. 1992). Table 3.12 gives the number of tanks, capacities, and construction times for tanks of each design.

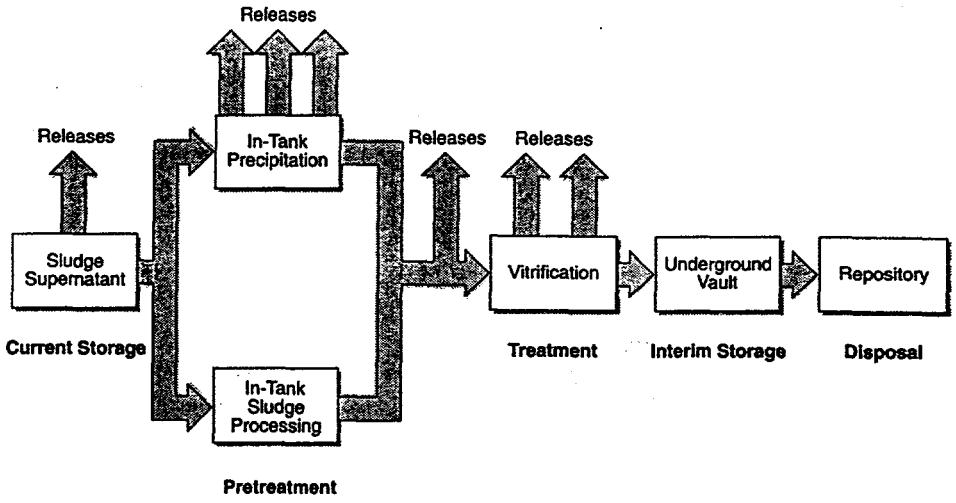
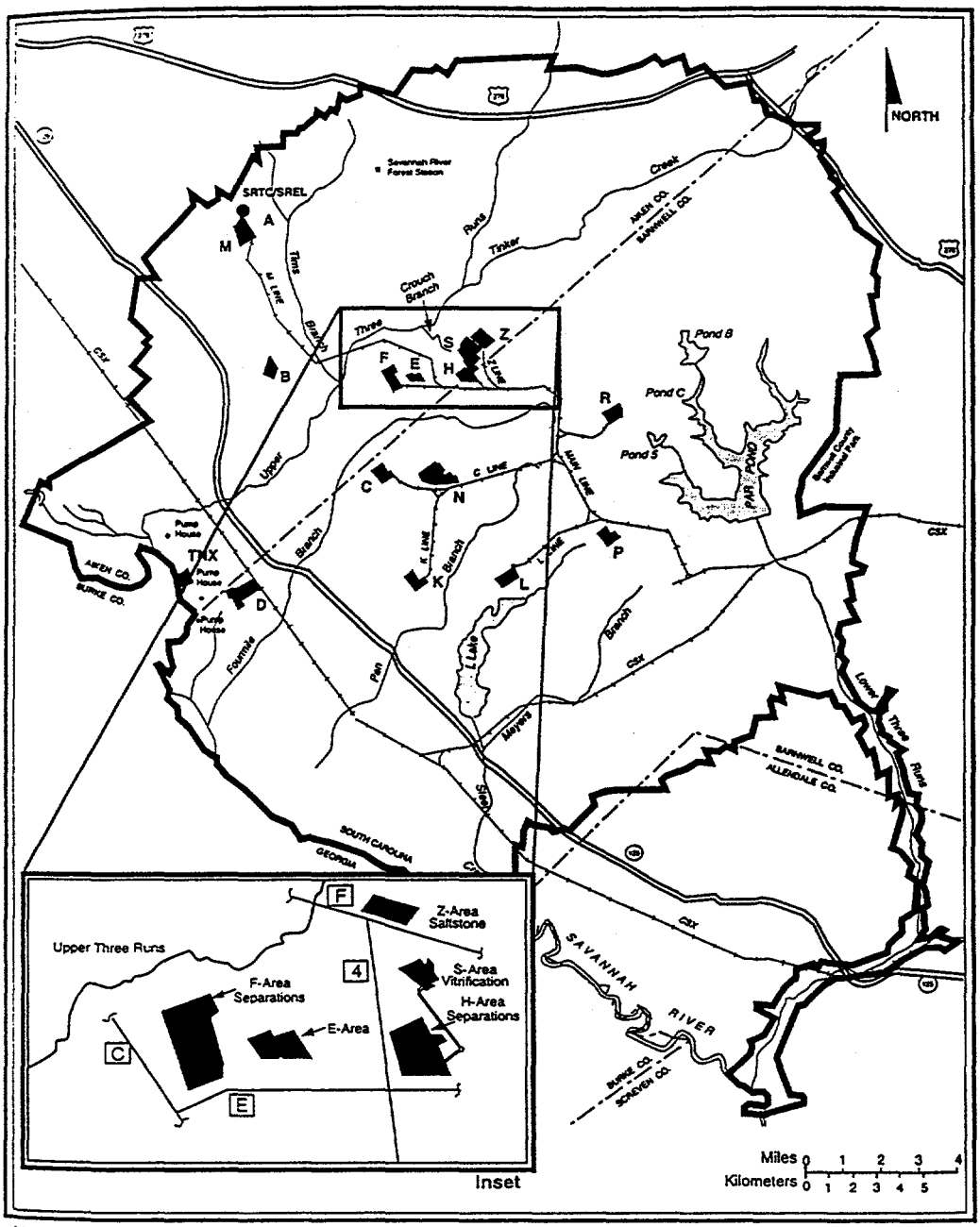


FIGURE 3.4 Flow Diagram for Defense Waste Processing Facility

The size of the tanks ranges from 2,840 m^3 (750,000 gal) to 4,921 m^3 (1.3 million gal) with a total capacity of 188,000 m^3 (49.7 million gal). Table 3.13 gives the design characteristics of each tank type. At present, the tanks are at greater than 90 percent of usable capacity (DOE 1995a). Of the 50 tanks currently in use, 23 (Types I, II, and IV designs) do not meet criteria for leak detection and secondary containment, and will be retired by 1997. This will leave 27 tanks for HLW storage with a total capacity of about 133,000 m^3 (35 million gal). The radionuclide distribution of the waste currently stored in the tanks is given in Table 3.14.

The ventilation system for the waste storage tanks vary; some do not have active ventilation while others maintain negative pressure to ensure that unfiltered air would flow into the potentially contaminated tank during loss of containment. For most tanks, air enters a tank through a HEPA filter on one side, passes to the other side of the tank where it exits through a demister, condenser, heater, and HEPA filter. The air then exits through an exhaust blower to the atmosphere. All of the Type III and some of the Type I, II, and IV tanks have permanently installed instruments that continuously monitor the vapor space for flammable vapor mixtures. The other tanks are monitored with portable instruments on a weekly basis.

Atmospheric releases from the HLW storage tanks are the result of the continuous venting of the tanks. The venting is required to remove the hydrogen that is generated by the radiolytic decomposition of organic material in the tanks. On the basis of experience at WVDP, 1% of the liquid inside the tanks is expected to evaporate per year and become airborne (Table 3.15). The radionuclides entrained in the ventilation air stream are assumed to be passed through a HEPA filter with an overall decontamination factor (DF) of 5×10^8 (except for tritium, H-3). With this information, an annual release fraction of 2×10^{-11} for all the gaseous entrainments can be



Note: DWPF facilities are located in S-, H-, and Z-Areas.

FIGURE 3.5 Map of the Savannah River Site (Source: Davis et al. 1989)

TABLE 3.12 Savannah River Waste Tank Designs

Tank Type	Number of Tanks	Capacity (m^3)	Cooling	Constructed
I	12	2,840	Yes	1951-1953
II	4	4,000	Yes	1955-1956
III	27	4,920	Yes	1967-1981
IV	8	4,920	No	1958-1961

Source: Hobbs et al. (1992).

calculated. Because HEPA filters are ineffective for tritium, the release fraction for hydrogen-3 is 0.01/yr. The release fractions have to be multiplied with the inventory for each radioisotope provided in Table 3.14 to obtain annual emissions in curies per year. The emissions are released to the atmosphere through a 7.62-m (25-ft) stack with an inside diameter of 0.254 m (10 in.) (Table 3.16).

The HLW storage tanks must be monitored and maintained on a regular basis. For the tank farm surveillance and maintenance, labor requirements and dose rates can be found in DOE (1979). After scaling the original number to the current storage volume, it was determined that a total of 33 workers are needed until 1997 when all Type I, II, and IV tanks will be retired (Table 3.17). After 1997, 21 workers will be required for this task. The individual worker dose is estimated to be 0.36 rem/yr.

Waste from the separation canyons flows to a receiving tank in the tank farm through a diversion box in which appropriate jumper connections are made to establish the desired flow path. The individual tank farms are interconnected to diversion boxes and valve pits with buried transfer lines. The original waste transfer lines consists of a stainless steel pipe surrounded by a bitumen-coated carbon-steel pipe for secondary containment which is itself surrounded by a waterproof concrete encasement. Hydraulic pumps and steam-jet eductors are used for waste transfer (WSRC 1992).

The F- and H-Area tank farms, located approximately 3.2 km (2 mi) apart, are connected by inter-area transfer lines. The inter-area transfer lines consist of a pair of parallel 7.62-cm (3-in.) stainless steel lines enclosed in 10.16-cm (4-in.) carbon-steel pipe jackets for secondary containment and are located below grade.

TABLE 3.13 Current HLW Storage at SRS

Tank Characteristics	Tank Type I	Tank Type II	Tank Type III	Tank Type IV
Storage method	Underground, carbon-steel, double-walled tank with forced cooling; sits within a pan that is 1.5 m deep and 1.5 m larger in diameter than tank; encapsulated in a concrete vault.	Underground, carbon-steel, double-walled tank with forced cooling; sits within a pan that is 1.5 m deep and 1.5 m larger in diameter than tank; encapsulated in a concrete vault.	Underground, stress-relieved carbon-steel, double-walled tank with forced cooling; sits within a tank of same height; encapsulated in a concrete vault.	Underground, carbon-steel, single-walled tank without forced cooling; encapsulated in a concrete vault without water stops.
Radiation protection	2.7 m of soil cover over 0.56 m concrete roof	Not available	Elevated above ground, surrounded with mounded earth; no soil cover over 1.2 m of reinforced concrete roof.	F-Area, 0.8 m of soil cover over 0.18-m concrete roof; H-Area, 1.1 m of soil cover over 0.18-m concrete roof.
Unit capacity	2,838,750 L	3,898,550 L	4,920,500 L	4,920,500 L
Number of units	12	4[a]	27	8
Percent of total waste stored in this tank type	12	4	77	7
Percent of total radioactive content	27	8	64	<1
Date of retirement[b]	1997	1997	Not available	1997

TABLE 3.13 (Cont.)

Tank Characteristics	Tank Type I	Tank Type II	Tank Type III	Tank Type IV
Surface area of unit[c]	Diameter: 22.9 m Height: 7.5 m Surface area: 1,363 m^2	Diameter: 25.9 m Height: 8.2 m Surface area: 1,721 m^2	Diameter: 25.9 m Height: 10.1 m Surface area: 1,876 m^2	Diameter: 25.9 m Height: 10.1 m Surface area: 1,876 m^2
Location	8 tanks in 241-F Area; 4 tanks in 241-H Area	4 tanks in 241-H Area	10 tanks in 241-F Area; 17 tanks in 241-H Area	4 tanks in 241-F Area; 4 tanks in 241-H Area

[a] One of the four Type II tanks (i.e., tank 16H) has been taken out of service and is currently empty.

[b] DOE (1989a, 1991a, 1995a).

[c] Surface area calculated by assuming that the tanks are cylindrical in shape.

Sources: DOE (1979, 1982a, 1989a); McIntosh and Papouchado (1990).

TABLE 3.14 Radionuclide Content of Current HLW Inventory at SRS

Radionuclide	Curies	Radionuclide	Curies	Radionuclide	Curies
H-3	3.2E+03	Sb-126	1.4E+02	U-235	3.7E-01
C-14	9.2E-02	Sb-126m	1.0E+03	U-236	2.6
Co-60	4.0E+05	Te-125m	6.8E+05	U-238	2.4E+01
Ni-59	5.5E+01	I-129	3.6E-02	Np-237	5.0
Ni-63	7.3E+03	Cs-134	8.7E+05	Pu-236	2.9E+02
Se-79	4.2E+02	Cs-135	2.6E+02	Pu-238	3.6E+06
Rb-87	2.1E-03	Cs-137	1.1E+08	Pu-239	3.1E+04
Sr-90	1.1E+08	Ba-137m	1.0E+08	Pu-240	2.1E+04
Y-90	1.1E+08	Ce-144	2.3E+07	Pu-241	4.0E+06
Zr-93	2.6E+03	Pr-144	2.3E+07	Pu-242	2.7E+01
Zr-95	2.4E+01	Pr-144m	2.7E+05	Am-241	2.6E+04
Nb-95	5.0E+01	Pm-147	5.7E+07	Am-242	3.4E+01
Tc-99	7.6E+03	Sm-151	5.7E+05	Am-242m	3.4E+01
Ru-106	5.3E+06	Eu-152	8.6E+03	Am-243	1.4E+01
Rh-106	5.3E+06	Eu-154	1.5E+06	Cm-242	8.2E+01
Pd-107	3.4E+01	Eu-155	1.1E+06	Cm-243	1.3E+01
Sn-121m	2.1E+02	U-232	3.1E+01	Cm-244	2.6E+05
Sn-126	1.0E+03	U-233	3.7E-03	Total	5.7E+08
Sb-125	1.9E+06	U-234	8.2E+01		

Source: Adapted from ORNL (1992b).

TABLE 3.15 Atmospheric Releases of Radionuclides during Current Storage of HLW at SRS

Release Description	Release Fraction[a]
Assume 1% of liquid becomes airborne by evaporation into ventilating system air stream with overall decontamination factor of 5.0E+08 before release to the atmosphere.	2.0E-11/yr for gases 1.0E-02/yr for tritium

[a] The release characteristics at SRS are taken to be equal to those of WVDP. The release fraction for gases is calculated by dividing 1% by 5.0E+08. For tritium, the release fraction is 1% because the filter system is assumed to be ineffective in its removal.

Source: DOE (1982b).

TABLE 3.16 Release Point Characteristics for Atmospheric Emissions for Current Storage of HLW at SRS

Stack Parameter	Value
Release point	Tank Type III[a]
Stack height (m)	7.6
Height above surrounding structures (m)	7.6
Stack diameter (m)	0.25
Stack exit velocity (m/s)	4.9
Exit temperature (°C)	40

[a] Because tank types I, II, and IV are in the process of being retired, the release point characteristics of tank type III are used for storage.

Source: Hunter (1990).

TABLE 3.17 Estimated Personnel Requirements and Exposure for Current Storage of HLW at SRS

Category	Number of Personnel (FTE)[a]	Duration (date)	Dose Rate (rem/yr)[b]
Exposed (all)	33	1994-1997	0.36
Exposed (all)	21	1998-2007	0.36

[a] The quoted number of 21 full-time-equivalent (FTE) workers for tank-farm maintenance and surveillance is based on 25 Type III tanks. This number is used for the period after 1997 when all waste tanks other than Type III will be retired. For the period until 1997, the 21 workers are scaled up to account for the 24 tanks of types I, II, and IV. The scaleup is not done linearly to account for the gradual phaseout of these waste tanks. A factor of $(51/25)^{0.6}$ is applied (DOE 1979).

[b] 7.6 rem/yr for all workers divided by 21 workers (DOE 1979).

The HLW will be transferred from the F- and H-Area tank farms, where the waste is stored and the pretreatment facilities (in-tank precipitation and in-tank sludge processing) are located, to the DWPF for treatment. For the transfer, a system of above-grade and underground transfer lines will be used, together with diversion boxes and pump pits. The diversion boxes and pump pits are below-grade concrete cells with concrete covers that will allow SRS to transfer waste from tank to tank, from one tank farm to the other, or from the tank farms to the DWPF. Pump pits are located at a low point in the system and contain tanks to receive the waste and pumps to transfer the waste through the connecting lines.

According to experience at WVDP, air entering the transfer system via the pits causes about 10% of the liquid to evaporate and become entrained in the system air stream. Before being released to the atmosphere through a 25-ft stack, the air is passed through a HEPA filter with an overall DF of 5×10^8 (except for tritium). The release fractions can be calculated as 2×10^{-10}/yr for gases and 0.1/yr for tritium, as given in Table 3.18. The release point characteristics are assumed to be the same as for current storage presented in Table 3.16.

Table 3.19 gives the labor requirements for transferring the HLW at Savannah River from the tank farms to the DWPF. The estimates in DOE (1979) were used and scaled accordingly to reflect the lower throughput of the transfer system. To transfer the current HLW backlog of about 132,000 m^3 (34.8 million gal), it is estimated that four workers will be needed during the 15 years of transfer operations. The worker dose rate is estimated to be about 0.42 rem/yr.

Pretreatment will take place in seven Type III storage tanks (four for in-tank precipitation and three for sludge processing); a pump tank at the New Waste Transfer Facility (capacity of 29.3 m^3 [7,750 gal]) will be used for the sludge feed to the DWPF. Four Type III tanks will be allocated to the in-tank salt processing, which consists of three steps. The feed for in-tank salt processing will be the salt cake that has been redissolved with corrosion-inhibited water plus recycled wash water and agitated by slurry pumps. When the solution reaches saturation, it will be removed in batches and sent to a salt processing tank. Salt concentrate from the in-tank sludge processing plus any liquid waste recycled from the DWPF are blended into the feed. The first step removes cesium and potassium from the 35 wt% salt solution. The addition of an aqueous sodium tetraphenyl borate (NaTPB) solution causes Cs-137 and potassium (K) to precipitate out as cesium tetraphenyl borate (CsTPB) and potassium tetraphenyl borate (KTPB). For complete precipitation, NaTPB is added 50% in excess of the stoichiometric amount. Portions of mercury in the stream are precipitated in this step as diphenylmercury. Next, Sr-90 and plutonium are removed by adsorption on sodium titanate (NaTi) particles after mixing the salt solution with a 17 wt% NaTi slurry in alcohol. Finally, the precipitate is separated from the salt solution by cross-flow filtration with about 28.4 million L (7.5 million gal) per year of decontaminated filtrate going to the Saltstone Facility. The slurried precipitate is sent back to the processing tanks and mixed with the next batch that undergoes in-tank salt processing.

TABLE 3.18 Atmospheric Releases of Radionuclides during Transfer of HLW at SRS[a]

Release Description	Release Fraction
Assume 10% of liquid becomes airborne annually by evaporation and is released from the tank-farm ventilation system with an overall decontamination factor of 5.0E+08 before release to the atmosphere.	2.0E-10 for gases 1.0E-01 for tritium

[a] The release characteristics at SRS are taken to be equal to those at WVDP. The release fraction for gases is calculated by dividing 10% by 5.0E+08. For tritium, the release fraction is 10% because the filter system is assumed to be ineffective in its removal.

Source: DOE (1982b).

TABLE 3.19 Estimated Personnel Requirements and Exposure for HLW Transfer at SRS

Category	Number of Personnel (FTE)[a]	Duration	Dose Rate (rem/yr)[b]
Exposed (all)	5	15	0.42

[a] DOE (1979) states that it will take 10 workers over 10 years to transfer 60×10^6 gal (227×10^6 L) of waste. As planned, 132×10^6 L will be transferred to the DWPF over 15 years. By scaling to the lower throughput, the number of workers is determined as follows:

$$[100 \text{ worker-yr} \times (132/227)^{0.6}] / 15 \text{ yr.}$$

[b] The total personnel exposure of 4.2 rem/yr is divided by the number of workers to obtain the individual dose rate of 0.42 rem/worker-yr (DOE 1979).

One full cycle of in-tank salt processing involves three consecutive batches. Sodium tetraphenyl borate and NaTi are added only to the first batch since their increased use provides virtually no additional gains in decontamination. At the end of the cycle, the accumulated precipitates are washed and corrosion inhibitors (caustic [NaOH] and sodium nitrite [$NaNO_2$]) added. The precipitates are then sent to the salt processing cell in the DWPF in 30-m^3 (8,000-gal) batches.

The slurried sludges will be transferred from the storage tanks to three Type III tanks dedicated to in-tank extended sludge processing. Sludges low in aluminum will be sent there directly. High-aluminum sludges, however, will first be reacted with 50 wt% hot caustic to dissolve an expected 75% of the hydrated alumina (Al_2O_3). The excess caustic and the resulting soluble sodium aluminate plus other soluble salt species will be washed out with water, which reduces the waste volume by approximately 50% and improves the waste characteristics significantly, particularly viscosity. The sludge will be allowed to settle over a period of about three weeks. Next, the supernatant will be decanted, concentrated by evaporation, and fed to the in-tank salt processing. The complete wash-settle-decant cycle will take approximately one month. It is estimated that the first batch will provide feed to the DWPF for about three years of operation (2.5 million L [660,000 gal] of 13 wt% sludge). A pump tank at the New Waste Transfer Facility will be used to feed sludge to the DWPF; pump tank capacity will be 7,750 gallon. The sludge will be delivered to the DWPF in microbatches of about 30 m^3 (8,000 gal).

The vitrification process at SRS is similar to that at Hanford. After processing to remove mercury from the pretreated HLW, glass-forming frit is added and the mixture concentrated by evaporation. The resulting mixture is fed to a melter where it is heated, and the molten glass is poured into stainless steel canisters. The outer surfaces of the canisters are decontaminated and the top is welded closed. Each canister will contain about 0.62 m^3 (165 gal) of vitrified waste. An annual average production rate of 190 canisters per year is assumed in this analysis. The "High-Level Waste System Plan" (WSRC 1995) indicates a maximum production rate of about 300 canisters per year.

The facility design adopted for the DWPF is based on the separations canyons operated successfully at Hanford and SRS for processing SNF. Design of the DWPF is based on totally remote operation and maintenance (O&M) of all chemical and mechanical processes that handle any radioactive materials or are conducted in a radioactive environment (McKibben et al. 1990). The principal operating facility is the Vitrification Building, which is designed as a reinforced-concrete shielded facility 110 m (360 ft) long, 35 m (115 ft) wide, and 27 m (90 ft) in height (DOE 1994b). This structure contains the glass melter and associated facilities for solidification of the high-level radioactive waste supplied from the tank farms. It is designed with two parallel canyons. The east canyon contains the maintenance repair cell, backup off-gas cell, melter cell, and decontamination cell. The west canyon contains most of the conventional wet chemical processing equipment. The principal cell areas of the DWPF are the process cell (sludge acidification, mercury removal, and glass frit addition), the melter cell (glass melter and cooldown station for filled canisters), the

canister decontamination cell (canister decontamination), and the weld test cell (to seal weld the canisters).

The DWPF vitrification building is designed as a Category 1 facility (DOE 1989b; Kennedy et al. 1990). Table 3.20 summarizes information on the dimensions of the Vitrification Building, electric energy requirements, and other process requirements. Further information concerning the DWPF is provided in the Defense Waste Processing Facility Final Supplemental EIS (DOE 1994b).

A second major facility is the GWSB, which is designed as a natural convection air-cooled vault for storage of 2,286 canisters with a maximum canister heat load of 470 W.

3.3.3 Current Status

SRS completed the DWPF EIS in November 1994 (DOE 1994b). The DWPF includes the HLW pretreatment process, the Vitrification Facility, saltstone manufacturing and disposal (LLW resulting from the pretreatment of HLW), radioactive glass waste storage facilities, and associated support facilities. The Record of Decision (March 28, 1995) describes the Department's decision to complete construction and begin operation of the DWPF. The DWPF became operational on March 12, 1996. Vitrified HLW canister production began in 1996, with 27 canisters produced as of July 15, 1996.

3.4 SAVANNAH RIVER SITE: PLANNED FACILITIES

Vitrification of the HLW at SRS is expected to take 24 years. The glass canisters are stored on-site in an interim storage facility until the year 2015, when it is assumed that a geologic repository will be available.

Storage capacity is currently provided for 2,286 vitrified HLW canisters. In accordance with the Final DWPF Supplemental EIS (DOE 1994), a second glass canister storage building will be constructed in 2007 for an additional estimated 2,286 HLW canisters. The second facility, like the first storage building, will hold canisters in vertically sealed cavities within a concrete structure forming the storage vault. The second glass canister storage building is projected to accommodate canisters with heat loads up to 670 W, in comparison with the 470 W limit of the first building (WSRC 1991). Table 3.21 summarizes estimated facility characteristics for the second glass canister storage building on the basis of a storage capacity of 2,286 canisters and a maximum loading rate of 410 canisters per year; further details on the derivation of the various parameters in Table 3.21 are given in the Appendix. (Facility characteristics for the first glass storage facility at SRS are given in Table 1.2.)

TABLE 3.20 Defense Waste Processing Facility at SRS

Building features	Seismic- and tornado-resistant concrete structure; designed with two parallel canyons; east canyons for maintenance repair cell, backup off-gas cell, melter cell, and decontamination cell; west canyon for most of the conventional wet chemical processing equipment.
Number of stories in DWPF building	3[a]
Shielding for operating personnel	1.2-m thick[a]
Building dimensions[b]	
Building length (m)	110
Building width (m)	36
Building height (m)	28
Approximate footprint (m^2)	3,960
Total land requirements (m^2)	161,880[b]

Resource Commitments during Operations[c]

Electricity (GWh/yr)	32
Water (m^3/yr)	7,600
Glass frit (t/yr)	680
Sodium hydroxide (t/yr)	1,490
Sodium tetraphenylborate (t/yr)	245

[a] Carlson (1989).

[b] State of South Carolina (1984).

[c] Defense Waste Processing Facility Final Supplemental EIS (DOE 1994b).

TABLE 3.21 Facility Characteristics for the Second Glass Canister Storage Building at SRS

Construction		Operations	
Parameter	Quantity	Parameter	Quantity
Resource Requirements			
Water (m^3)	9,300	Water (m^3/yr)	
Concrete (m^3)	250,000	Loading	5,700
Steel (t)	41,000	Storage	2.5
		Electricity (GWh/yr)	
		Loading	0.6
		Storage	Negligible
Land Requirements			
Occupied land (ha)	0.44	Number of floors	1
Disturbed land (ha)	0.80	Facility size (m^2)	4,800
Employment			
Peak (persons)	390	Storage (FTE[a])	9
Total effort (worker-yr)	51	Loading (FTE)	41
Construction duration (months)	52	Support (FTE)	18
Costs			
Capital cost ($ million)	≈43	Loading O&M ($ million/yr)	12
		Storage O&M ($ million/yr)	0.66
Secondary Waste Streams			
Hazardous	Negligible	Contact-handled LLW: loading (m^3/yr)	940
Sewage	Negligible	Contact-handled LLW: storage (m^3/yr)	200
Air Emissions			
Fugitive Dust (t)	33	Radiological	Negligible
NO_x (t)	5.7	Hazardous	Negligible
SO_x (t)	0.5	Potential pollutant	Negligible

[a] Full-time equivalent.

3.5 IDAHO NATIONAL ENGINEERING LABORATORY: EXISTING FACILITIES

3.5.1 History

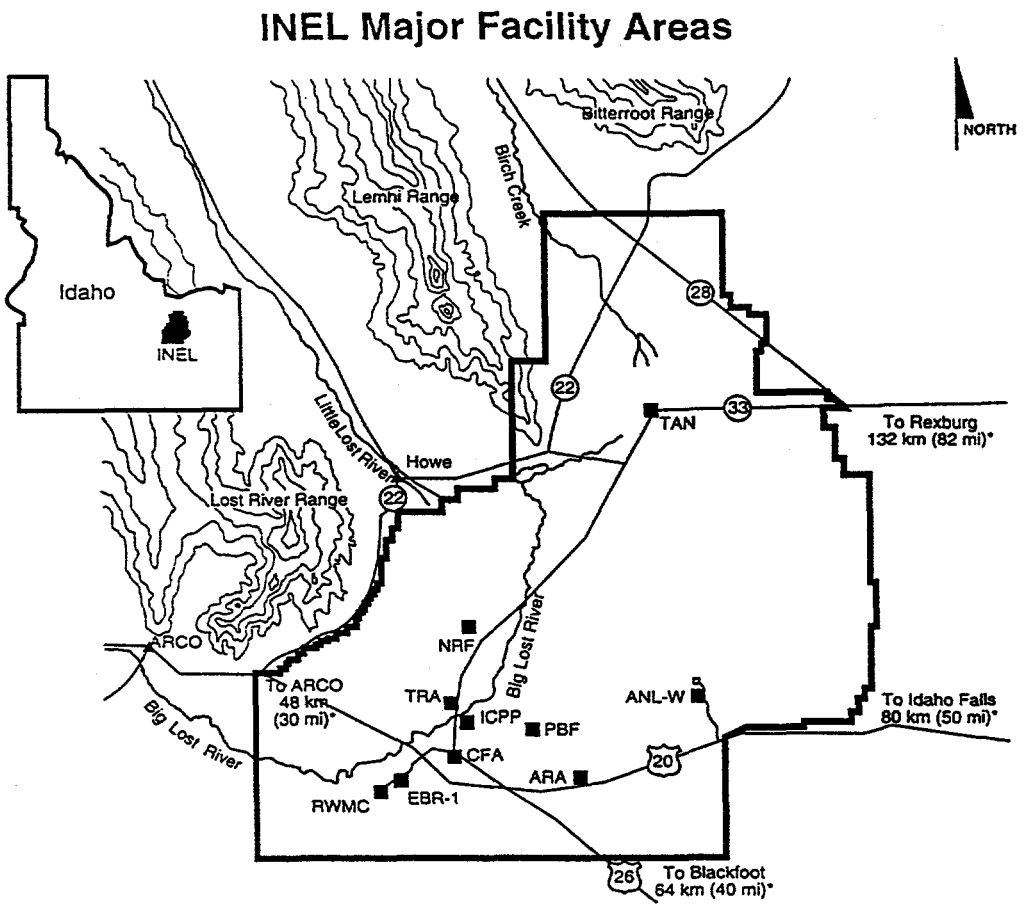
The INEL, formerly known as the National Reactor Testing Station, was established in 1949 as a site for constructing, testing, and operating nuclear facilities. Eventually, 52 nuclear reactors were constructed and operated; as of 1992, 13 were currently operational (DOE 1992a). In addition, various support facilities were built and tested to demonstrate the applications of nuclear reactor technology, conduct safety research, and support defense programs. Currently, INEL activities include treating and storing HLW.

Recovery of nuclear materials at INEL is centered at the ICPP. This plant was built in 1951 and is located on an 82-ha (203-acre) site in the southwest portion of INEL. The ICPP processes high-level liquid radioactive waste generated at the facility and that generated from other INEL activities. These wastes, which are generated primarily by the processing of SNF and decontamination operations, are concentrated by evaporation and converted to a solid in the NWCF. The waste at INEL has two main forms: acidic liquid and solid calcine.

3.5.2 Descriptions and Environmental Releases

Existing HLW facilities include the HLW tank farm (Tanks WL-101, WM-100 through WM-102 and WM-180 through WM-190), CSSFs (CPP 729, 742, 746, 760, 765, 791, and 795), and the NWCF (CPP-659), which are all located at the ICPP site (Figure 3.6).

Two types of liquid HLW generated at INEL are fluorinel and sodium-containing waste. Fluorinel waste is generated during SNF reprocessing; sodium-containing waste is primarily produced during solvent cleanup and from decontamination waste streams. The sodium-containing waste, although not generated by reprocessing of SNF, is managed the same way as is HLW because of site practice. Decontamination wastes and process condensates are evaporated prior to tank farm storage. In general, about 70-90% of the liquid HLW at INEL is fluorinel waste (Bolon and Nelson 1992). The acidic, liquid HLW contains aluminum nitrate, zirconium fluoride, and sodium-containing wastes. Current practice is to blend sodium-containing waste with fluorinel waste prior to calcination. As of 1994, the inventory of calcine at INEL is approximately 3,800 m^3 (134,000 ft^3); 7,200 m^3 (1.9 million gal) of liquid HLW remain. The calcine (at the end of 1994) accounted for about 50 million Ci with the remaining acidic, liquid portion of the waste representing the majority of the initial volume but a minority of the initial radioactivity (about 2 million Ci). The fluorinel-sodium calcine is expected to be the most prevalent composition at the ICPP. Table 3.22 gives the compositions of the fluorinel-sodium high-level liquid waste and calcine (Berreth and Knecht 1987).



*Miles from Central Facilities Area

Legend:

ARA	Auxiliary Reactor Area
ANL-W	Argonne National Laboratory-West
CFA	Central Facilities Area
EBR-1	Experimental Breeder Reactor - I
ICPP	Idaho Chemical Processing Plant
NRF	Naval Reactors Facility
PBF	Power Burst Facility
RWMC	Radioactive Waste Management Complex
TAN	Test Area North
TRA	Test Reactor Area

Miles 0 2 4 6 8
Kilometers 0 4 8 12

FIGURE 3.6 Map of the Idaho National Engineering Laboratory (Source: DOE 1993d)

TABLE 3.22 Major Components of ICPP High-Level Wastes

Sodium-Bearing Liquid Waste		Calcined High-Level Waste	
Component	Moles per m^3	Aluminum (kg per m^3)	Zirconium Mix (kg per m^3)
Fluoride (F^-)	64	–	360
Nitrate (NO_3^-)	4,700	45	68
Zirconium (Zr^{+4})	6.4	–	230
Aluminum (Al^{+3})	540	530	140
Cadmium (Cd^{+2})	1.9	–	8.1
Boron (B^{+3})	16	2	14
Sodium (Na^+)	1,300	25	34
Potassium (K^+)	150	–	7.4
Chloride (Cl^-)	23	–	–
Sulfate (SO_4)	33	13	4.7
Density (kg/m^3)	1,240	1,100	1,600

Note: – = not present in significant quantity.

Source: WINCO (1994).

The ICPP tank farm has been storing liquid HLW since 1953; the newest tank was placed in service in 1964. The ICPP tank farm consists of eleven 1,140-m^3 (300,000-gal) stainless steel tanks housed in underground concrete vaults (WM-180 through WM-190); one tank (WM-190) is kept empty for use as a spare should a leak occur in one of the other 10 tanks. The 1,140-m^3 (300,000-gal) tanks are 18.9 m (62 ft) in diameter and 4.1 m (13.5 ft) in height; the top of the vault is a minimum of 3 m (10 ft) below grade. The vault structure is founded 16 m (52.5 ft) below ground level on bedrock, with a 4.3 m (14 ft) separation between vaults (Uldrich and Malik 1991). Additional storage capacity is available from four 69.6-m^3 (18,400-gal) tanks resting on concrete pads with curbing (WL-101 and WM-100 through WM-102). These tanks are generally kept empty and may be used for waste storage only with prior approval by DOE-IDO. Tank farm activities include continuous level, density, temperature, and pressure monitoring; transfer operations; tank off-gas system operation; tank cooling; and periodic corrosion monitoring tests. Each tank and vault is separately monitored to detect any possible leak; none of the storage tanks has ever leaked. Table 3.23 gives the design characteristics of the underground HLW tanks.

HLW calcine is stored in stainless steel underground bins called calcine solids storage facilities (CSSFs) also located at the ICPP site. The CSSFs are designed to provide interim storage

TABLE 3.23 Current HLW Storage at INEL

Waste Storage Type	Liquid HLW Storage Tanks[a]	Calcine Solids Storage Facilities
Storage method	Underground stainless steel tanks in concrete vault	Reinforced-concrete vaults containing 3 to 7 stainless steel bins
Unit capacity	≈1,136 m^3	≈ 1,785 m^3
On-site unit number	WM-180 through WM-190	CPP 729, 742, 746, 760, 765, 791, and 795
Number of units	10 filled with 1 empty, used as spare	7
Percent capacity filled	80%	CSSF 1 to 5 full, CSSF 6 and 7 currently being filled
Dimensions/surface area of unit	Tank diameter: 18.9 m Tank height: 4.1 m Surface area: ≈805 m^2	Vault diameter: 18 m Vault height: 32 m Surface area: ≈2,320 m^2
Distance below/above ground	Top of vault is minimum of 3 m below grade.	Variable; nominal design is between 6.1 to 9.1 m above ground.

[a] Tanks WL-101 and WM-100 to WM-102 are not included because of low percentage of utilization.

Sources: IDO (1982); Berreth (1988); Wheeler et al. (1986).

(up to 500 years) for the calcined solids and consist of vertical stainless steel bins that are housed in reinforced concrete vaults anchored on bedrock. Each vault contains from three to seven stainless steel storage bins bolted to the vault floor, with the top of the vault projecting above grade. The vault walls and roof provide required radiation shielding as well as structural support. The bins and the vaults provide the primary and secondary containment, respectively, for the calcined solids. The heat generation from the radionuclides in the solid calcine is removed from the calcine bins by natural convection; the maximum storage temperature is 600°C (1,122°F). A representative vault is 18 m (59 ft) in diameter and 32 m (105 ft) tall. Five calcined solids storage facilities are currently filled at the ICPP, with a sixth still receiving calcine and a seventh ready to receive calcine. Table 3.23 gives the design characteristics of the calcine storage facilities.

Public information on the radionuclide content of INEL's high-level liquid waste is generally unavailable. However, radionuclide content data for a repository canister containing immobilized three-year-old calcine are available (Table 2.11 [ORNL 1992a]). Although, in general, the feed calcine to the Idaho Waste Immobilization Facility (IWIF) may be older than three years,

this appears to be the only publicly available information. Calcines older than three years would have a lower activity so that the radionuclide composition given in Table 2.11 (ORNL 1992a) for a repository canister containing 2,725 kg of calcine can be considered to represent the maximum activity. Assuming that the amount lost (from atmospheric emissions, etc.) of a given radionuclide is negligible, so that most of the radionuclide in the liquid HLW will be immobilized, the "representative" liquid HLW composition (shown in Table 3.24) can be calculated from the volume of the current liquid HLW inventory (2.3×10^6 gal) through the use of the following equation for nonvolatile radionuclides (i.e., other than H-3, C-14, and I-129):

$$[\textit{Liquid HLW composition}\ (\text{Ci})] = [\textit{Canister radioactivity}\ (\text{Ci/canister})]/ (2.725 \times 10^6\ \text{g/canister}\ \textit{calcine}) \times (1.4\ \text{g/cm}^3\ \textit{calcine})/(6.3985\ \text{cm}^3\ \textit{liquid HLW/cm}^3\ \textit{calcine}) \times (3{,}785.4\ \text{cm}^3\text{/gal}) \times (2.3 \times 10^6\ \text{gal}\ \textit{liquid HLW}). \quad (3.3)$$

Elements such as H-3, C-14, and I-129 volatilize completely before the immobilization process and thus will not appear in the canister contents. Their composition was determined by assuming that these elements are completely emitted during primary pretreatment, and that their emission rate from the NWCF is proportional to the product of their liquid HLW concentration and its calculated release fraction during primary pretreatment:

$$[\textit{Liquid HLW composition}\ (\text{Ci})] = ([\textit{Emission from primary pretreatment}\ (\text{Ci/yr})]/ [\textit{Component release fraction}])/([\text{Sr-90}\ \textit{Emission from primary pretreatment}\ (\text{Ci/yr})]/ [\text{Sr-90}\ \textit{Release fraction}]) \times [\text{Sr-90}\ \textit{in liquid HLW}\ (\text{Ci})], \quad (3.4)$$

where

Component release fraction = 1.0 (for H-3, C-14, I-129),
Sr-90 release fraction = 1.805×10^{-11},
Component emission from primary pretreatment (Ci/yr) = 1.1×10^{-10} for nonvolatiles,
Component emission from primary pretreatment (Ci/yr) = 3.9×10^{-9} for semivolatiles,
Sr-90 emission from primary pretreatment (Ci/yr) = 2.09×10^{-4}, and
Sr-90 in liquid HLW (Ci) = 2.4754×10^7.

Strontium-90 is used as the reference radionuclide; other radionuclides used from Table 3.24 gave similar results. Comparison of the values in Table 3.24 with a limited set of radionuclides in Schindler (1991) showed order-of-magnitude agreement.

Off-gases are generated during venting of the liquid HLW storage tanks. The annual release of radionuclides during current liquid HLW storage at INEL is calculated on the basis of the current liquid HLW inventory (2.3×10^6 gal), the assumption that 1% per year of the liquid becomes airborne into the ventilating system air stream (DOE 1982b), and a DF of 5×10^8. The DF is based

TABLE 3.24 Radionuclide Inventory of Liquid HLW at INEL

Radionuclides in INEL Liquid HLW	Canister Radioactivity (Ci/canister)	Calculated Liquid HLW Activity (Ci)	Calculated Liquid HLW Composition (g/gal)
H-3	Not given	6.48E+02	4.63E-06
C-14	Not given	6.37E-01	2.03E-09
I-129	Not given	1.45E-01	4.63E-10
Se-79	1.74E-01	1.22E+02	7.60E-04
Rb-87	9.81E-06	6.86E-03	3.40E-02
Sr-90	3.54E+04	2.48E+07	7.89E-02
Y-90	3.54E+04	2.48E+07	1.98E-05
Zr-93	8.85E-01	6.18E+02	1.02E-01
Nb-93m	2.04E-01	1.43E+02	2.20E-07
Tc-99	5.72E+00	4.00E+03	1.03E-01
Ru-106	2.64E+03	1.85E+06	2.40E-04
Rh-106	2.64E+03	1.85E+06	2.26E-10
Pd-107	5.45E-03	3.81	3.22E-03
Sn-126	8.72E-02	6.09E+01	9.34E-04
Sb-126m	8.72E-02	6.09E+01	3.37E-13
Sb-126	8.72E-02	6.09E+01	3.17E-10
Cs-134	8.99E+03	6.28E+06	2.11E-03
Cs-135	2.04E-01	1.43E+02	5.39E-02
Cs-137	3.54E+04	2.48E+07	1.24E-01
Ba-137m	3.27E+04	2.28E+07	1.85E-08
Ce-144	2.23E+04	1.56E+07	2.13E-03
Pr-144	2.23E+04	1.56E+07	8.99E-08
Pm-147	3.27E+04	2.28E+07	1.07E-02
Sm-151	4.63E+02	3.24E+05	5.35E-03
Eu-154	4.90E+02	3.43E+05	5.52E-04
U-233	3.27E-09	2.28E-06	1.03E-10
U-234	1.17E-07	8.19E-05	5.70E-08
U-235	4.90E-06	3.43E-03	6.89E-04
U-236	2.72E-05	1.90E-02	1.28E-04
U-237	1.31E-08	9.14E-06	4.87E-17
U-238	2.72E-11	1.90E-08	2.46E-08
Np-237	1.31E-04	9.14E-02	5.64E-05
Pu-238	1.91E+02	1.33E+05	3.39E-03
Pu-239	1.91	1.33E+03	9.32E-03

TABLE 3.24 (Cont.)

Radionuclides in INEL Liquid HLW	Canister Radioactivity (Ci/canister)	Calculated Liquid HLW Activity (Ci)	Calculated Liquid HLW Composition (g/gal)
Pu-240	1.77	1.24E+03	2.36E-03
Pu-241	4.36E+02	3.05E+05	1.29E-03
Pu-242	4.90E-03	3.43	3.90E-04
Am-241	2.48	1.73E+03	2.19E-04
Am-243	2.26E-02	1.58E+01	3.45E-05
Cm-242	1.77	1.24E+03	1.63E-07
Cm-244	1.42	9.90E+02	5.32E-06

on the current practice of passing the liquid HLW tank-farm emissions at the ICPP through a mist eliminator, a heater, one bank of HEPA filters, a second heater, and a second bank of HEPA filters. The overall release fraction is thus 2×10^{-11}. The emission rate during current storage is calculated by using the following formula:

$$[\textit{Emission rate} \text{ (Ci/yr)}] = [\textit{Species "i" in liquid HLW} \text{ (Ci)}] \times [\textit{Release fraction}]. \quad (3.5)$$

It is assumed that negligible emissions would occur during calcine storage because of the calcine's stable (i.e., nonvolatile at ambient conditions), solid form and the absence of venting within the CSSF.

Nonradiological atmospheric emissions during current storage can be calculated on the basis of similar assumptions for radiological emissions; approximately 5.9×10^{-8} tons of NO_x and 6×10^{-10} tons of sulfur dioxide (SO_2) are emitted annually.

The CPP-708 Main Stack is the common release point for off-gases from the liquid HLW tank farm vessels, the NWCF, and other ICPP operations. Stack characteristics are given in Table 3.25 (State of Idaho 1989). The projected height of the INEL Waste Immobilization Facility (60 ft) was used to determine the value of the height above surrounding structures. The exit flow rate was calculated on the basis of the stack's other parameters.

Worker dose rates and labor requirements for current storage were estimated as follows. The number of exposed workers was determined on the basis of SRS experience (i.e., 21 workers for surveillance and monitoring of 25 liquid HLW tanks [DOE 1979]) and scaled by the number of HLW storage tanks at INEL; this resulted in a total of nine workers for liquid HLW storage and six workers for calcine storage. The worker dose rate for current storage of liquid HLW is based on SRS

TABLE 3.25 Release Point Characteristics for Atmospheric Emissions for the Various HLW Processes at INEL

Stack Parameter	Value
Release Point	CPP-708 Main Stack
Stack height (m)	76
Height above surrounding structures (m)	58
Stack diameter (m)	≈2
Stack exit velocity (m/s)	≈20
Exit temperature (°C)	38

Source: State of Idaho (1989).

whole body occupational exposure experience (7.6 rem/yr for 21 workers [DOE 1979]) or 0.36 rem/yr; the worker dose rate during calcine storage is not known and is assumed to be equal to that for liquid HLW storage.

Liquid HLW from INEL underground storage tanks is pumped to pretreatment. Buried waste lines are fabricated of stainless steel and are enclosed in pipe encasements. Three types of encasements are used: split tile, stainless-steel-lined concrete, and stainless steel pipe. Encasements drain to sumps instrumented to detect radioactivity. Routing of liquid waste within the tank farm area and from process areas to the tank farm is controlled by manually operated valves.

Off-gases are generated during liquid HLW transfer and calcine retrieval. The annual release of radionuclides during liquid HLW retrieval is calculated on the basis of WVDP's reported release fraction of 2.0×10^{-10} (DOE 1982b) for liquid HLW retrieval and a liquid HLW retrieval rate of approximately 7.5×10^5 gal/yr (operating at 3,000 gal/d for 250 d/yr):

$$[\textit{Emission rate}\ (\text{Ci/yr})] = [\textit{Species "i" in liquid HLW}\ (\text{Ci})] \times [\textit{Release fraction}] \times [\textit{Liquid HLW retrieval rate}\ (\text{gal/yr})] \,/\, (2.3 \times 10^6\ \text{gal}\ \textit{liquid HLW}). \quad (3.6)$$

The annual release of radionuclides during calcine retrieval is calculated on the basis of a release fraction of 5.0×10^{-14} (IDO 1982) and a calcine retrieval rate of 8.91×10^5 kg/yr:

$$[\textit{Emission rate}\ (\text{Ci/yr})] = [\textit{Species "i" in canister}\ (\text{Ci/canister})]/ (2.725 \times 10^6\ \text{g/canister}\ \textit{calcine}) \times [\textit{Release fraction}] \times [\textit{Calcine retrieval rate}\ (\text{kg/yr})] \times (1{,}000\ \text{g/kg}). \quad (3.7)$$

Labor requirements for liquid HLW transfer were estimated on the basis of WVDP experience (five workers to transfer 2×10^6 L over three years) and scaled by the operating capacity of the NWCF (3,000 gal/d), assuming 250 operating d/yr; the number of exposed personnel is 21. The number of personnel for calcine retrieval was estimated by assuming four worker-shifts per week; one worker per shift is required for each of the following activities: (1) controlling the location of the retrieval nozzle within the calcine bin; (2) monitoring the discharge of calcine from the cyclone separator into the receiver, (3) monitoring the operation of the suction pump(s) to ensure proper calcine transport (correct pressure drop to prevent particle saltation) to the vitrification facility, and (4) monitoring the discharge from the off-gas treatment system. The number of exposed personnel during calcine retrieval is assumed to be 16. Two worker dose rate values have been found in the literature for liquid HLW transfer: the higher worker dose rate of 0.65 rem/yr is taken from DOE (1982a). This rate was developed on the basis of experience at the Hanford tank farms; the lower value of 0.42 rem/yr is from historic SRS data (DOE 1979). The higher value is used here for conservatism. The worker dose rate for calcine retrieval is assumed to equal that for liquid HLW (0.65 rem/yr).

Liquid HLW in acidic solution is stored in stainless steel tanks that may not meet all seismic regulations and do not have a secondary containment system that is acid resistant. The Idaho Operations Office entered into a consent order in April 1992 to resolve secondary containment issues. This consent order requires continued calcination, thus reducing waste volume and resulting in a material that is much easier to handle and store. The acidic, liquid HLW is calcined to form a dry granular solid. Calcination is desirable because the solid calcine is less corrosive than the original liquid waste, consumes approximately one-seventh to one-eighth of the original liquid waste volume, and is chemically and physically more stable than the radioactive liquids.

Without calcination, the ICPP tank farm would have filled to capacity, precluding generation of waste from reprocessing or future decontamination and decommissioning programs. Since startup of the original WCF, 26,000 m^3 (6.9 million gal) of liquid HLW has been converted to 3,600 m^3 (127,000 ft^3) of solid calcine. The ICPP was the first plant in the world to use calcination for converting highly radioactive liquid waste into a radioactive solid waste. The NWCF started operation in September 1982 and has a design throughput of 11.4 m^3 (3,000 gal) per day. The building is a concrete and steel structure approximately 76 m (250 ft) long by 38 m (125 ft) wide by 23 m (75 ft) high; half of the structure is located below grade (Waite 1989). The processing equipment is located below grade and is divided by subsystem into isolated cells separated by concrete shielding walls. The principal cell areas of the NWCF are the blend and hold cell (calciner feed preparation), the calciner cell (calciner and associated equipment), the off-gas cell (cooling of off-gas and removal of solids), the filter cell (containing HEPA filtration units), and the adsorber cell (containing absorber cells for ruthenium removal).

Additional cells hold valves, sampling equipment, and other equipment. An attached decontamination facility is located in the west side of the NWCF building. The NWCF is designed

as a Category 1 facility (DOE 1989b; Kennedy et al. 1990). Table 3.26 summarizes the relevant technical information.

The NWCF uses a fluidized bed calcining process that is operated at 500°C (932°F) by internal combustion of kerosene. The initial fluidized bed material is a dolomite rock (calcium magnesium carbonate) with a particle size between 0.3 and 0.6 mm (0.012 to 0.024 in.). Calcium is added in a 10-to-40-mol% excess to react with the fluoride in the liquid HLW to form an inert calcium fluoride (CaF_2). Compressed air is used to atomize the acidic HLW, which is then sprayed into the fluidized bed. The moisture in the HLW evaporates and leaves the residual radioactive solids to deposit on the fluidized bed material. The calcined solids are pneumatically transferred from the NWCF to the nearby CSSFs.

The resulting HLW calcine is stored in stainless steel underground bins (CSSFs) also located at the ICPP site. The CSSFs are designed to provide interim storage (up to 500 years) for the calcined solids and consist of vertical stainless steel bins that are housed in reinforced concrete vaults anchored on bedrock. Each vault contains from three to seven stainless steel storage bins bolted to the vault floor. The bins and the vaults provide the primary and secondary containment, respectively, for the calcined solids. The heat generation from the radionuclides in the solid calcine is removed from the calcine bins by natural convection; the maximum storage temperature is 600°C (1,112°F). A representative vault is 18 m (59 ft) in diameter and 32 m (105 ft) tall. The ICPP currently has seven CSSFs; six have been filled. Table 3.23 gives the design characteristics.

3.5.3 Current Status

The latest NWCF campaign went from December 1990 through 1993. Approximately 570 m^3 (150,000 gal) of waste was calcined during the campaign. By court order (U.S. District Court [USDC] 1993), DOE is required to calcine all high-level liquid radioactive waste that does not contain sodium on or before January 1, 1998. In addition, this court order requires that DOE calcine or otherwise process as much sodium-containing high-level liquid radioactive waste (sodium liquid waste) as DOE and the State of Idaho mutually agree is practicable by January 1, 1998. Therefore, it is not possible to determine the exact schedule for NWCF operations at this time.

An amended Notice of Noncompliance Consent Order was issued to DOE in March 1994 by the Idaho Department of Health and Welfare. The Order included provisions for proposing waste treatment technologies and deadlines for removing liquid waste from specific tanks.

The Record of Decision (May 30, 1995) for the *Final Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995c) indicated DOE's decision to construct a facility to treat the calcined HLW and remaining liquid waste, and selected the

TABLE 3.26 New Waste Calcining Facility at INEL

Building features	Seismic- and tornado-resistant concrete structure; decontamination facility attached to west side; process equipment located below grade with the above-grade portion primarily being a clean area.
Number of stories in NWCF building	4[a]
Shielding for operating personnel	1.2-m thick[a]
Building dimensions[a]	
Length (m)	76
Width (m)	38
Height (m)	23
Approximate footprint (m^2)	≈2,800
Total land requirement (m^2)	8,000-12,000[b]
Resource Commitments during Operations	
Electricity (kW)	500-2,000
Calcium magnesium carbonate (t/yr)	≈450
Fuel oil (m^3/yr)	570-1,900

[a] Includes two subgrade floors and two above-grade floors (Waite 1989).

[b] Includes NWCF and CSSFs (Idaho Nuclear Corporation [INC] 1974).

Source: INC (1974).

technology for potential use in a treatment facility. The technology selected is radionuclide partitioning for radioactive liquid and calcine waste treatment, grout for immobilizing the resulting low-activity waste stream, and vitrification for immobilizing the resulting high-activity waste stream. The Record of Decision also stated that two INEL projects related to HLW management would be implemented in the future: the Tank Farm Heel Removal Project, and the Calcine Transfer Project.

3.6 IDAHO NATIONAL ENGINEERING LABORATORY: PLANNED FACILITIES

Five of the liquid HLW tanks must cease operations by March 2009, with the remaining six tanks by June 2015. The Tank Farm Heel Removal Project would provide for the design,

construction, and operation of equipment to perform tank internal rinsing and removal of the 5,000-to-20,000 gallon heel (liquid and solids remaining when the tanks have been emptied using existing transfer equipment) from the eleven 300,000 gallon storage tanks at the ICPP. Further information concerning the proposed Tank Farm Heel Removal Project is described in DOE (1995f).

A mobile calcine retrieval system has not yet been designed to remove the calcine from the storage bins and transfer it for treatment. A demonstration project has been scheduled to prove retrieval methods for extracting calcine from bin sets (DOE 1995f). Calcine retrieval is assumed to be accomplished by vacuuming the calcine particles from the bins, followed by transport by a pneumatic conveyor to the vitrification facility. Further information concerning the proposed Calcine Transfer Project for bin set #1 is described in DOE (1995f).

Of the four DOE sites containing HLW, INEL HLW will be processed last because it is in a fairly stable solid form. The schedule for construction of a facility to treat the calcined HLW (and any remaining liquid waste) at the INEL is to be negotiated between DOE and the State of Idaho under the Federal Facility Compliance Act. However, in this analysis, it is assumed that physical construction of the Idaho WIF is to begin in FY 2005, with hot operation commencing in FY 2015 (WINCO 1994).

The facility design adopted for the Idaho WIF is based on the vitrification design provided in DOE (1995f). The IWIF would incorporate totally remote O&M of all chemical and mechanical processes that handle any radioactive materials or are conducted in a radioactive environment. The principal operating facility is the immobilization building, which is designed as a reinforced-concrete shielded facility approximately 110 m (360 ft) long, 36 m (120 ft) wide, and 45 m (150 ft) in height. This structure contains the separations process used to partition the incoming HLW into high- and low-activity fractions (and, if necessary, to remove heavy metals from the low-activity stream), and the vitrification unit and associated operations for solidifying the high-level radioactive waste supplied from either the CSSFs or the NWCF. The principal cell areas are the vitrification cell (containing vitrification melter and feed mix tanks), the maintenance cell, the separation cell, the canister decontamination cell, and the off-gas processing cell.

The Idaho WIF is assumed to be designed as a Category 1 facility (DOE 1989b; Kennedy et al. 1990). Table 3.27 summarizes information on the dimensions of the HLW immobilization building and construction, electric energy, and other process requirements.

Treatment of the calcine to a form suitable for permanent disposal in a geologic repository is assumed in this analysis to begin in FY 2015 (DOE 1995c). Treatment by vitrification is assumed in this analysis. The calcine would first be dissolved to allow separation of the radioactive liquid and calcine waste into low-activity waste and high-activity streams (Figure 3.7). The resulting high-activity waste stream would then be immobilized using vitrification. In the vitrification process, the

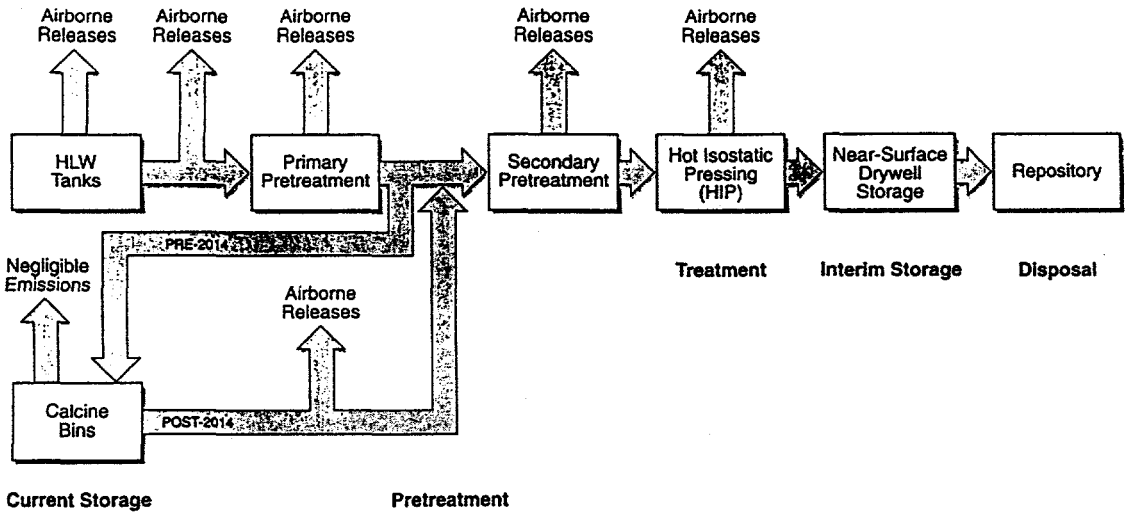


FIGURE 3.7 Flow Diagram for the Idaho Waste Immobilization Facility (Source: Adapted from DOE 1995f)

dissolved calcine would be slurried with glass-forming sands of varying composition (frit) and introduced to the vitrification melter. The calcine would first have to be thoroughly mixed with the frit to obtain a homogeneous melter feed and might have to be stabilized and ground to improve the melter operation efficiency. The molten glass mixture from the vitrification melter would be cast directly into canisters, and then cooled.

The method of interim storage at INEL has not been determined, but dry storage in modular vaults is assumed in this analysis. The facility would hold canisters in vertically sealed cavities within a concrete structure forming the storage vault. Table 3.28 summarizes estimated facility characteristics for the INEL glass canister storage building on the basis of a storage capacity of 1,700 canisters and a maximum loading rate of 81 canisters per year.

Annual releases of radionuclides, under normal operations, would be limited to those from canister surface contamination. Surface contamination levels may be expected to be less than 370 Bq/m^2 for alpha contaminants and less than 3,700 Bq/m^2 for beta-gamma contaminants (Gurley and Minor 1985), assuming that INEL's HLW canisters will undergo the same level of decontamination as at HWVP. Given the canister dimensions in Table 1.1, the amount of alpha contaminants adhering to a single canister is approximately 1×10^{-7} Ci, with about 1×10^{-6} Ci of beta-gamma contaminants. It is therefore expected that radiological atmospheric emissions will be negligible.

TABLE 3.27 Resource Requirements for HLW Immobilization at INEL

Conceptual Waste Immobilization Facility	
Assumed number of stories in building	3[a]
Exterior walls	Reinforced-concrete, masonry unit construction
Shielding for operating personnel	1.2 m thick[b]
Approximate building dimensions	
Length (m)	110[c]
Width (m)	36[c]
Height (m)	45[d]
Approximate footprint (m^2)	4,000[e]
Electrical energy consumed annually (GWh/yr)	40[e]
Annual water usage (liter/yr)	150 million[e]
Approximate Construction Requirements	
Peak construction manpower (person)	300[e]
Concrete (m^3)	32,000
Reinforcing steel (t)[f]	3,800
Structural steel (t)[g]	1,700
Water (liter)	11.5 million[e]

[a] IDO (1982) states that the waste immobilization facility will be a two- to three-story reinforced-concrete-mat-foundation structure; the three-story structure was chosen here because of the inclusion of the calcine stabilization process.

[b] Adapted from Grantham et al. (1983).

[c] A three-to-one aspect ratio assumed.

[d] Building height taken to equal that for the conceptual Hanford High-Level Waste Vitrification Facility (DOE 1996a).

[e] Adapted from DOE (1995c).

[f] Amount of reinforcing steel taken to be the average value for a BWR power plant (120 kg/m^3 of concrete) (United Engineers & Constructors, Inc. 1977; Ball et al. 1984).

[g] Amount of structural steel taken to be the average value for a BWR power plant (54 kg/m^3 of concrete) (United Engineers & Constructors, Inc. 1977; Ball et al. 1984).

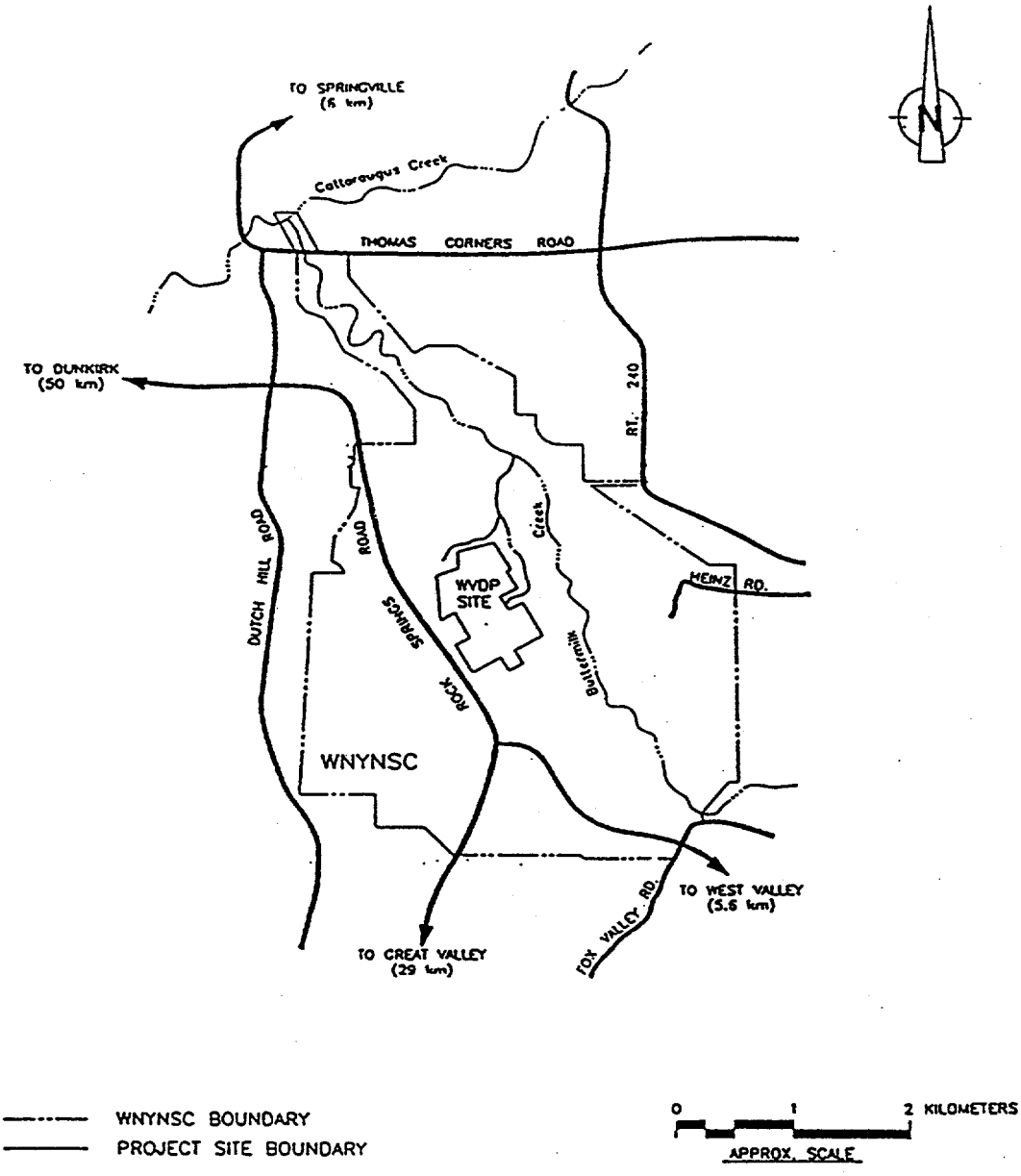


FIGURE 3.8 Map of the West Valley Demonstration Project (Source: WVDP 1991)

TABLE 3.28 Facility Characteristics for the Glass Canister Storage Building at INEL

Construction		Operations	
Parameter	Quantity	Parameter	Quantity
Resource Requirements			
Water (m^3)	8,000	Water (m^3/yr)	
Concrete (m^3)	19,000	Loading	1,100
Steel (t)	31,000	Storage	1.9
		Electricity (GWh/yr)	
		Loading	0.3
		Storage	Negligible
Land Requirements			
Occupied land (ha)	0.38	Number of floors	1
Disturbed land (ha)	0.68	Facility size (m^2)	3,800
Employment			
Peak (persons)	310	Storage (FTE[a])	8
Total effort (worker-yr)	39	Loading (FTE)	21
Construction duration (months)	45	Support (FTE)	16
Costs			
Capital cost ($ million)	≈36	Loading O&M ($ million/yr)	5.6
		Storage O&M ($ million/yr)	0.5
Secondary Waste Streams			
Hazardous	Negligible	Contact-handled LLW: loading (m^3/yr)	190
Sewage	Negligible	Contact-handled LLW: storage (m^3/yr)	150
Air Emissions			
Fugitive Dust (t)	24	Radiological	Negligible
NO_x (t)	4.3	Hazardous	Negligible
SO_x (t)	0.3	Potential pollutant	Negligible

[a] Full-time equivalent.

3.7 WEST VALLEY DEMONSTRATION PROJECT: EXISTING FACILITIES

3.7.1 History

The WVDP is being undertaken at the Western New York Nuclear Services Center (WNYNSC) near West Valley, New York (Figure 3.8), in order to conduct a HLW management operation of immobilizing the 2,500 m^3 (660,000 gal) of HLW which was stored on-site. The only commercial nuclear fuel reprocessing facility in the United States operated at West Valley from 1966 to 1972. The facility was shut down in 1972 to expand its capacity. However, increasing state and federal regulations made the required investment more costly than planned, and the operator decided to close the facility in 1976 (Borisch et al. 1988). In 1980, the West Valley Demonstration Project Act (Public Law 96-368) was enacted to demonstrate solidification that can be used to prepare HLW for disposal. This act authorized DOE to conduct a nuclear waste management project at West Valley that includes waste minimization, reduction, treatment, and storage. The DOE assumed control of the West Valley project premises in February 1982.

3.7.2 Facility Descriptions and Environmental Releases

Existing HLW facilities include the Vitrification Facility Building, adjacent glass canister storage area in the CPC, Integrated Radwaste Treatment System (IRTS), Sludge Mobilization and Wash System (SMWS), and a tank farm consisting of four tanks.

Most of the HLW stored at West Valley is alkaline waste that was produced in the PUREX reprocessing of low-burnup commercial and Hanford N-Reactor spent fuel. Prior to the initiation of pretreatment, the HLW inventory in storage at WVDP was 2,270 m^3 (600,000 gal) of alkaline PUREX waste and 45 m^3 (12,000 gal) of acidic THOREX waste. Ninety percent by volume of the alkaline waste was in the supernatant liquid form; the remaining 10% was in a sludge form at the bottom of the tank (McIntosh et al. 1992). The acidic waste can be considered to be a single-phase solution (ORNL 1992b).

The current HLW inventory of 2,180 m^3 consists of 2,040 m^3 of liquid alkaline sludge and 140 m^3 of solid waste composed of alkaline sludge and inorganic zeolite ion-exchange material contaminated with cesium. Alkaline waste was stored in an underground carbon-steel tank (8D-2) in the form of a supernatant alkaline liquid containing Cs-137 and an alkaline sludge containing Sr-90. Tank 8D-2 resides in a tank farm consisting of four tanks enclosed in three underground concrete vaults (Tanks 8D-3 and 8D-4 are contained in the same vault). Tank 8D-1 is identical to 8D-2 and contains loaded zeolite; Tank 8D-1 has a capacity of 2,800 m^3 (740,000 gal), an outer diameter of 21 m (70 ft), a height of 8.2 m (27 ft), and is located about 2.4-2.7 m (8-9 ft) below grade. Tank 8D-4 is constructed of stainless steel and contains the acidic THOREX waste; it has a

capacity of 57 m^3 (15,000 gal), an outer diameter of 3.7 m (12 ft), a height of 4.8 m (15.75 ft), and is located about 1.8 m (6 ft) below grade. Tank 8D-3 is identical to 8D-4 and serves as its backup. Table 3.29 gives the design characteristics of the various HLW tanks.

To estimate atmospheric releases from current storage, it is assumed that 1% of liquid THOREX waste becomes airborne annually by evaporation into the ventilation system air stream. The ventilation system is assumed to have an overall DF of 5×10^8 for all species except tritium; it is further assumed that the off-gas system is ineffective in removing tritium and gaseous organic chemicals from the tank off-gas stream (DOE 1982b). This assumption results in a release fraction of 2×10^{-11}/yr for gases and 1×10^{-2}/yr for tritium. Table 3.30 gives the mass amounts of various compounds in the THOREX liquid waste, and Table 3.31 gives the waste's radionuclide composition as of the end of 1989 (ORNL 1992a). The emission rate for species *i* during current storage is given by the following equation:

$$[\textit{Emission rate}\ (\text{Ci/yr})]_{cs} = [\textit{Species "i" activity}\ (\text{Ci})] \times [\textit{Release fraction}\ (\text{yr}^{-1})]. \quad (3.8)$$

TABLE 3.29 Current HLW Storage at WVDP

Waste Type	Alkaline Sludge	Acidic Liquid	Cesium-Load Zeolites
Storage method	Underground carbon-steel tank in concrete vault	Underground stainless steel tank in concrete vault	Underground carbon-steel tank in concrete vault
Unit capacity	2,800 m^3	57 m^3	2,800 m^3
Number of units	1	1	1
Percent capacity filled[a]	≈3%	≈82%	≈3%
Surface area of unit	Diameter: 21.3 m Height: 8.2 m Surface area: ≈1,260 m^2	Diameter: 3.7 m Height: 4.8 m Surface area: ≈77 m^2	Diameter: 21.3 m Height: 8.2 m Surface area: ≈1,260 m^2
Location on installation	North end of installation	North end of installation	North end of installation
Distance below grade	Top is minimum of 2.4 m below grade.	Top is minimum of 2.4 m below grade.	Top is minimum of 2.4 m below grade.

[a] It is assumed that all the alkaline sludge has been washed, that washing reduced its volume by 61% to 89 m^3, and that the washed sludge has been returned to Tank 8D-2.

Sources: Ploetz and Leonard (1988); DOE (1982b).

TABLE 3.30 Mass Amounts of Various Compounds in the THOREX Liquid Waste at WVDP

Compound	Mass (kg)	Compound	Mass (kg)
$AgNO_3$	8.0E-02	Na_2SiO_3	1.3E+02
Al_2O_3	1.0E+03	Na_3PO_4	1.2E+01
B_2O_3	2.7E+02	NaCl	5.0E+01
BaO	1.6E+01	NaF	1.0
CaO	1.0E+01	$NaNO_3$	2.3E+02
CoO	1.2	$NaTcO_4$	1.1E+01
Cr_2O_3	6.1E+02	NiO	2.4E+01
Cs_2O	2.0E+01	$Pd(NO_3)_2$	8.0
CuO	3.0E-01	PuO_2	4.0E-01
FeO	2.5E+03	RhO_2	5.1
HNO_3	2.8E+03	RuO_2	1.6E+01
$KMnO_4$	9.8E+02	SrO	7.8
KNO_3	1.9E+02	ThO_2	1.7E+04
La_2O_3	1.1E+01	$UO_2(COOH)_2$	4.6
MgO	1.6E+01	Y_2O_3	5.7
Na_2MoO_4	5.4E+01	ZnO	4.3
Na_2SO_4	1.8E+02	ZrO_2	4.4

Source: ORNL (1992b).

Table 3.32 gives information on the stack release from the current storage facility (and transfer operations) on the basis of WVDP's permit application (WVDP 1992).

To estimate atmospheric releases during transfer, it was assumed that 10% of the liquid THOREX waste becomes airborne via evaporation and is released from the current storage facility's ventilation system with a DF of 5×10^4. It was also assumed that the overall DF for the current storage facility and its stack filtration system is 1×10^4 (DOE 1982b), which yields a release fraction of 2×10^{-10} for gases and 1×10^{-1} for tritium. Assuming three years of HLW retrieval and the above release fractions, the annual release of radionuclides during HLW transfer is given by the following equation:

$$[\textit{Emission rate}\ (\text{Ci/yr})]_{rt} = [\textit{Species "i" activity}\ (\text{Ci})] \times [\textit{Release fraction}] / 3. \qquad (3.9)$$

TABLE 3.31 Radionuclide Composition of THOREX Liquid HLW at WVDP as of the End of 1989

Radionuclide	Ci	Radionuclide	Ci	Radionuclide	Ci
H-3	1.5	Sm-151	4.7E+03	Th-230	4.4E-02
C-14	1.3E-01	Eu-152	4.1E+01	Th-231	5.2E-03
Fe-55	2.5E+02	Eu-154	2.0E+03	Th-2342	1.6
Co-60	7.7E+02	Eu-155	5.6E+02	Th-234	7.1E-05
Ni-59	2.0E+01	Tl-207	8.2	Pa-231	1.5E+01
Ni-63	2.4E+03	Tl-208	2.2	Pa-233	3.0E-01
Se-79	3.4	Pb-209	2.1E-01	Pa-234m	7.1E-05
Sr-90	4.2E+05	Pb-211	8.2	U-232	2.7
Y-90	4.2E+05	Pb-212	6.1	U-233	2.1
Zr-93	1.6E+01	Bi-211	8.2	U-234	2.2E-01
Nb-93m	1.1E+01	Bi-212	6.1	U-235	5.2E-03
Tc-99	1.0E+02	Bi-213	2.1E-01	U-236	9.8E-03
Ru-106	8.0E-02	Po-212	3.9	U-238	7.1E-05
Rh-106	8.0E-02	Po-213	2.0E-01	Np-236	1.2E-01
Pd-107	1.1E-01	Po-215	8.2	Np-237	3.0E-01
Cd-113m	3.2E+01	Po-216	6.1	Np-239	7.8
Sn-121m	5.8E-01	At-217	2.1E-01	Pu-236	1.1E-02
Sn-126	3.1	Rn-219	8.2	Pu-238	4.7E+02
Sb-125	1.4E+02	Rn-220	6.1	Pu-239	1.5E+01
Sb-126	4.4E-01	Fr-221	2.1E-01	Pu-240	8.1
Sb-126m	3.1	Fr-223	1.1E-01	Pu-241	7.4E+02
Te-125m	3.3E+01	Ra-223	8.2	Pu-242	1.2E-02
I-129	1.8E-01	Ra-224	6.1	Am-241	2.4E+02
Cs-134	1.1E+02	Ra-225	2.1E-01	Am-242	6.7
Cs-135	5.5	Ra-228	1.5	Am-242m	6.7
Cs-137	4.4E+05	Ac-225	2.1E-01	Am-243	7.8
Ba-137m	4.2E+05	Ac-227	8.2	Cm-242	5.5
Ce-144	9.6E-03	Ac-228	1.5	Cm-243	2.2E-01
Pr-144	9.6E-03	Th-227	8.1	Cm-244	1.2E+01
Pm-146	3.5E-01	Th-228	6.1	Cm-245	2.0E-02
Pm-147	4.1E+03	Th-229	2.1E-01	Cm-246	2.3E-03

Source: ORNL (1992a).

TABLE 3.32 Release Point Characteristics for Atmospheric Emissions during Current Storage and Transfer of HLW at WVDP

Stack Parameter	Value
Release point	Main Stack for HLW Storage
Stack height (m)	3.7
Height above surrounding structure (m)	-4.3
Stack exit velocity (m/s)	2.6
Stack diameter (m)	0.076
Exit temperature (°C)	25

Source: WVDP (1992).

To estimate the occupational exposure from current HLW storage, it is assumed that two full-time workers are needed — a technician and a health physicist. It is also assumed that average radiological exposure information for WVDP operations for 1987 is applicable to current storage operations. In that year, a collective dose of 70 rem was received by 425 workers who received a measurable dose (PNL 1989). The average dose per worker is therefore taken to be 165 mrem annually, or 0.08 mrem/h.

Radioactive process operations for pretreatment are planned to be conducted totally within the existing HLW storage tanks 8D-1, 8D-2, and 8D-3. To accomplish this, a new transfer line was constructed from Tank 8D-2 to a Supernatant Treatment System (STS) using interconnected double-contained piping housed in a containment conduit. The waste transfer facility includes piping, a containment trench, pump pits, and mobilization pumps.

Occupational exposure is also associated with transfer of the HLW from the current storage facilities to the vitrification facility. The estimate given here is based on information contained in the Phase I EIS (DOE 1982a). A dose rate of 0.65 rem/worker-yr was assumed on the basis of experience with exposure to workers at the tank farm at Hanford. The Phase I EIS estimates that five persons are needed for transfer operations and that the operations continue for four years (i.e., three years to retrieve the waste and one year to flush and decontaminate the tanks used in current storage). Table 3.33 summarizes information on the number of personnel required and worker exposure.

The IRTS consists of the STS, Liquid Waste Treatment System (LWTS), Cement Solidification System (CSS), and the Drum Cell. The LWTS, CSS, and Drum Cell are concerned primarily with LLW processing; the zeolite product from the STS is managed as HLW.

TABLE 3.33 Estimated Personnel Requirements and Exposure for Transfer of HLW at WVDP

Category	Number of Personnel (FTE[a])	Duration (yr)	Dose Rate (rem/yr)[b]
Exposed (all)	5	3[c]	0.65
Exposed (all)	5	1[d]	0.65

[a] Full-time equivalent.

[b] Dose rates taken from experience at Hanford tank farms (DOE 1982a).

[c] Duration of operations phase.

[d] Duration of tank-flushing phase.

The STS removes Cs-137 from the supernatant wash solutions in Tank 8D-2 by running it through three zeolite ion-exchange columns located in Tank 8D-1; the decontaminated supernatant is then transferred to the LWTS for further processing. The major process components include a prefilter, feed tank, supernatant cooler, the four ion-exchange columns, and a filter for decontaminated supernatant. The solution is cooled to less than 13°C (55°F) to improve the cesium removal efficiency. Once the solution is cooled, it is pumped, in series, through the ion-exchange columns. Four columns are required, and each column contains 1.5 m^3 (53 ft^3) of zeolite. The spent ion-exchange medium is stored underwater in Tank 8D-1.

The SMWS is designed to resuspend the insoluble solids in the sludge in Tank 8D-2, wash the salts from the sludge, and decontaminate the wash solutions to ensure that the cement product does not exceed NRC radioactive waste Class C limits (Schiffhauer and Thompson 1993). The system consists of five low-pressure, high-flow, long-shafted centrifugal pumps installed in Tank 8D-2. The mobilization pumps mix the settled sludge solids and added caustic solution, resulting in dissolution of the salt crystals. The mixing is stopped when the desired salt concentration is reached and the solids are allowed to settle. The wash solution is then drawn from Tank 8D-2 with a floating suction removal pump and transferred to the IRTS for processing.

Both sludge-wash processing cycles were completed in 1994. The washed sludge, the acidic waste, and the cesium-loaded zeolite are to be combined.

The vitrification process at WVDP is similar to that at Hanford and SRS (Figure 3.9). High-level waste enters the vitrification process through the concentrator feed makeup tank (CFMUT). In the CFMUT, excess water is removed from the HLW, which is then mixed with glass-forming chemicals to form a slurry. The slurry is then fed to a holding tank, that is, the melter feed hold tank

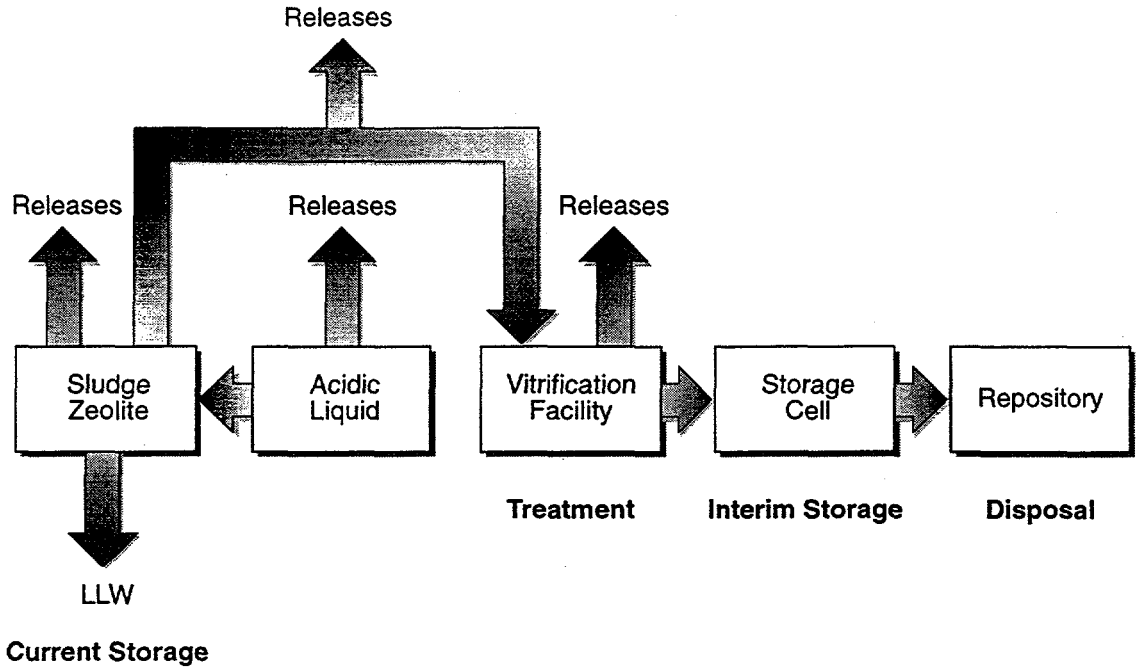


FIGURE 3.9 Flow Diagram for West Valley Demonstration Project

(MFHT). The slurry fed ceramic melter (SFCM) is where the vitrification into borosilicate glass actually takes place. The slurry is fed to the SFCM from the MFHT by an air displacement pump that meters the amount of slurry fed into the melter. The output from the melter overflows into canisters held on a turntable (Barnes et al. 1988).

During the initial part of the project, full-scale developmental testing was conducted in the Component Test Stand (CTS) by using nonradioactive materials to simulate the sludge expected in the storage tanks. The existing CTS is now being converted for radioactive operations into the Vitrification Facility (VF). The major components of the VF, including parts of the off-gas purification system, are housed in a new below-grade, shielded pit and Butler-type building constructed in the mid-1980s. The remainder of the off-gas system is located in the former reprocessing facility (McIntosh et al. 1988). A list of the various process equipment characteristics is given in Barnes et al. (1988).

3.7.3 Current Status

Construction of the VF began on August 12, 1985, and was completed during FY 1995. On June 24, 1996, liquid HLW was transferred to the Vitrification Facility for immobilization. The first canister of glass was completed on July 5, 1996, with three canisters produced as of July 8, 1996.

Vitrification operations using the HLW at WVDP are expected to proceed from 1996 to 1998, during which an estimated total of 340 HLW canisters will be produced.

4 DEFINITION OF ALTERNATIVES

4.1 METHODOLOGY AND ASSUMPTIONS

The DOE Waste Management (WM) Programmatic Environmental Impact Statement (PEIS) (DOE 1996b) is being prepared to assess the potential environmental consequences of different alternatives for consolidating waste management facilities at the major DOE sites. The HLW alternatives to be considered in the WM PEIS are as follows:

- **No Action Alternative**
  - Store HLW canisters at Hanford, SRS, INEL, and WVDP in existing and approved storage facilities;
  - Continue current treatment approaches at each site;
  - Continue interim storage of liquid and calcine HLW at INEL; and
  - Continue activities necessary for ultimate disposal of HLW in a geologic repository.
- **Decentralized Alternative**
  - Continue storage of HLW at Hanford, SRS, INEL, and WVDP;
  - Continue current treatment approaches at the above sites;
  - Begin immobilization of INEL HLW in 2015.
  - Continue interim storage of stabilized (vitrified) HLW at each site; and
  - Continue activities necessary for ultimate disposal of HLW in a geologic repository.
- **Regionalization Alternative (Case 1)**
  - Same as Decentralized Alternative, except provide interim storage facilities for treated HLW at SRS for WVDP vitrified HLW canisters.

- **Regionalization Alternative (Case 2)**

  - Same as Decentralized Alternative, except provide interim storage facilities for treated HLW at Hanford for WVDP vitrified HLW canisters.

- **Centralization Alternative (Case 1)**

  - Same as Regionalization Case 1, except provide interim storage facilities for treated HLW at Hanford for WVDP, SRS, and INEL HLW canisters until acceptance of HLW canisters at a geologic repository. This case assumes that the repository opens on time in FY 2015.

- **Centralization Alternative (Case 2)**

  - Same as Case 1, except provide interim storage facilities for all treated HLW at Hanford for WVDP, SRS, and INEL HLW canisters. This case assumes the repository is not ready to accept delivery of HLW canisters in FY 2015 and all HLW is stored temporarily at Hanford.

The No Action Alternative includes only existing and approved waste management facilities and will provide a baseline against which the other alternatives can be assessed. High-level waste at Hanford, SRS, and WVDP is to be immobilized by glass-based vitrification. Currently, DOE is proceeding with operation of facilities for vitrification of HLW at Hanford, SRS, and WVDP. The HLW will be retrieved, pretreated to remove soluble and nonradioactive species, vitrified, and then poured into canisters to complete the immobilization process. Existing HLW canister storage facilities include the GWSB at SRS (for a total storage capacity of 2,286 canisters) and the CPC at WVDP (storage capacity of approximately 350 canisters). In accordance with the Record of Decision based on the previous EIS for the Hanford Site (DOE 1987), a storage facility will be available for storing approximately 750 HLW canisters (about 5 years of production from the original HWVP design). The interim storage facility at Hanford is expected to be operational by 2009. The more recent TWRS EIS (DOE 1996a) considers storage for a total of 12,200 HLW canisters. Because DOE has not issued a Record of Decision based on the TWRS EIS, the WM PEIS applies the storage capacity given in DOE (1987). No HLW canister storage facility exists or is approved for INEL (DOE 1995f).

The Decentralized Alternative is similar to the No Action Alternative except that the HLW at INEL undergoes immobilization. High-level waste immobilization at INEL is assumed to occur via vitrification with prior radionuclide partitioning. Existing calcine at INEL would be retrieved from underground calcine bins and separated into high- and low-activity fractions (and if necessary, heavy metals would be removed from the low-activity stream). On-site interim storage would be provided for all treated HLW awaiting shipment to a geologic repository for permanent disposal.

The Regional Consolidation Alternative is the same as the Decentralized Alternative except that the vitrified HLW canisters produced at one site (or sites) would be transported for interim storage at another. Two HLW sites exist in the western portion of the United States — Hanford and INEL. The majority of the radioactivity of INEL HLW is in the form of a solid calcine stored in high-integrity stainless steel bins with a design lifetime of 500 years. Because the calcine is physically and chemically stable, adequately stored, and would require further processing to put it in a physical form suitable for transport, establishing Hanford as a regional HLW site to receive INEL waste is not reasonable. These considerations indicate that establishing INEL as a regional site for Hanford's HLW is also not reasonable.

The two HLW sites in the eastern portion of the United States are WVDP and SRS. West Valley is by law a demonstration cleanup project and, therefore, not a reasonable location for a regional HLW interim storage site. However, SRS is a potential regional HLW interim storage site for the vitrified HLW canisters at WVDP because of the similarity in canister characteristics.

The vitrified HLW canisters that will be produced at WVDP are also similar to those that may be produced at Hanford and, therefore, if ultimate disposal of HLW canisters occurs in the West, then Hanford could be considered a reasonable storage site for WVDP vitrified HLW canisters. Because of the different physical forms and unavailability of current storage facilities for handling canisters from WVDP at INEL, INEL is not considered suitable to store either SRS or WVDP vitrified waste.

If ultimate disposal of HLW is to occur in the West, it is unreasonable to consider using SRS as a storage location for Hanford HLW. Hanford, however, is a potential site for regionalized storage of SRS HLW canisters.

The Centralized Consolidation Alternative is the same as the Regional Consolidation Alternative except that interim storage for all HLW canisters would be provided at one site. As discussed previously, it is unreasonable to consider WVDP or INEL for storage of HLW from Hanford and SRS, and to consider using SRS as a storage location for Hanford HLW if ultimate disposal is to occur in the West. Under the Centralized Alternative, the HLW canisters produced at SRS, WVDP, and INEL would be transported to the Hanford Site, where adequate storage capacity would be provided until the canisters are accepted at a geologic repository.

This study assumes that DOE will construct interim glass canister storage facilities based on a modular design. The model for future storage is taken to be the GWSB at SRS. The capacity of each module is assumed to range from 2,200 to 2,900 canisters. Therefore, storage of a total of 21,612 HLW canisters at a single centralized site would require 10 interim storage facilities, each with a storage capacity of approximately 2,160 canisters.

For conservatism, the loading rate during interim storage is taken to equal the average annual vitrification rate (i.e., 190 canisters per year for SRS and 790 canisters per year for Hanford).

4.2 DESCRIPTIONS OF HLW CASES

4.2.1 No Action Alternative

The existing glass canister storage facility at SRS has a current capacity of 2,268 canisters; design characteristics of an interim storage facility for the remaining 2,268 additional canisters to be produced at Savannah River are given in Table 3.21 (assuming a constant loading rate of 410 canisters per year). The characteristics of the CPC at WVDP are given in Table 1.2. A canister storage area for approximately 750 canisters is to be provided at the Hanford Site. The storage facility was expected to be modular in design to allow expansion if a storage period longer than 5 years was required. The HLW canisters at Hanford were to be stored in dry cells, with a negative pressure maintained through a monitored HEPA filter system (DOE 1987). The impacts associated with the construction and operation of the interim canister storage area are included in the resource requirements for the previous design for the HWVP (Table 3.11).

4.2.2 Decentralized Alternative

A comparison of the existing and projected interim storage facilities for this alternative is given in Table 1.2. It is assumed that six interim storage facilities, each with a storage capacity of 2,500 canisters, would be required for the 15,000 canisters estimated to be produced at Hanford. Design characteristics for one of the six interim storage facilities are given in Table 4.1 (assuming a constant loading rate of 790 canisters per year).

4.2.3 Regionalized Alternatives

The approach taken for the following cases assumes that the limited number of WVDP canisters (340 total) would not have a significant impact on the assumed loading rates for the various interim storage facilities (i.e., WVDP canisters are sent during periods when production is lower than the maximum canister loading rate).

TABLE 4.1 Decentralization Alternative (storage capacity of 2,500 canisters per facility, six facilities at Hanford)

Construction		Operations	
Parameter	Quantity	Parameter	Quantity
Resource Requirements			
Water (m^3)	98	Water (m^3/yr)	
Concrete (m^3)	275,000	Loading	11,000
Steel (t)	45,000	Storage	2.8
		Electricity (GWh/yr)	
		Loading	0.9
		Storage	Negligible
Land Requirements			
Occupied land (ha)	0.46	Number of floors	1
Disturbed land (ha)	0.84	Facility size (m^2)	4,600
Employment			
Total effort (worker-yr)	55	Loading (FTE)	53
Peak (persons)	430	Storage (FTE)	9
Construction duration (month)	54	Support (FTE)	18
Costs			
Capital cost ($ million)	≈ 45	Loading O&M ($ million/yr)	16
		Storage O&M ($ million/yr)	0.71
Secondary Waste Streams			
Hazardous	Negligible	Contact-handled LLW: loading (m^3/yr)	1,800
Sewage	Negligible	Contact-handled LLW: storage (m^3/yr)	220
Air Emissions			
Fugitive dust (t)	36	Radiological	Negligible
NO_x (t)	6.3	Hazardous	Negligible
SO_x (t)	0.5	Potential pollutant	Negligible

4.2.3.1 Case 1

This case calls for shipping WVDP vitrified HLW canisters to SRS for storage. The second interim storage facility at SRS would be expanded from its current projected capacity of 2,268 canisters to accommodate the 340 WVDP canisters (for a total of 2,626 canisters). Table 4.2 gives the design characteristics of the second glass storage facility at SRS (assuming a constant loading rate of 190 canisters per year).

4.2.3.2 Case 2

This case calls for shipping WVDP vitrified HLW canisters to Hanford for storage. The proposed glass canister storage facility design at Hanford would be expanded from its current projected capacity of 2,500 to 2,560 canisters to accommodate the 340 WVDP canisters (for a total of 15,340 canisters). Table 4.3 gives the design characteristics of one of the six interim storage facilities (assuming a constant loading rate of 790 canisters per year).

4.2.4 Centralized Alternatives

This case calls for shipping WVDP and SRS vitrified HLW canisters to Hanford for storage. In addition, the immobilized glass-ceramic HLW canisters at INEL would also be transported to Hanford for interim storage prior to ultimate disposal in a geologic repository. It is assumed that the difference in canister dimensions and thermal power between the three vitrification sites (Hanford, SRS, WVDP) and the glass-ceramic design at INEL would not significantly affect the design characteristics of the interim canister storage facility such as the spacing between the individual HLW canisters in the storage vault (taken to be approximately 2 ft).

4.2.4.1 Case 1

This case assumes the geologic repository begins acceptance of HLW canisters in FY 2015. A total of approximately 17,700 canisters is to be stored at Hanford in seven facilities (each with a storage capacity of approximately 2,530 canisters) prior to transport to the repository. Design characteristics of an interim storage facility with a storage capacity of 2,530 HLW canisters at the Hanford Site are given in Table 4.4; the loading/unloading rate of 980 canisters per year is based on Hanford's annual production rate of 790 canisters per year plus SRS's average annual production rate of 190 canisters per year.

TABLE 4.2 Regionalization Alternative: Case 1 (2,626 canisters at SRS)

Construction		Operations	
Parameter	Quantity	Parameter	Quantity
Resource Requirements			
Water (m^3)	100	Water (m^3/yr)	
Concrete (m^3)	280,000	Loading	5,700
Steel (t)	46,000	Storage	2.8
		Electricity (GWh/yr)	
		Loading	0.6
		Storage	Negligible
Land Requirements			
Occupied land (ha)	0.47	Number of floors	1
Disturbed land (ha)	0.85	Facility size (m^2)	4,700
Employment			
Total effort (worker-yr)	56	Loading (FTE)	41
Peak (persons)	440	Storage (FTE)	9
Construction duration (month)	54	Support (FTE)	19
Costs			
Capital cost ($ million)	≈ 45	Loading O&M ($ million/yr)	12
		Storage O&M ($ million/yr)	0.73
Secondary Waste Streams			
Hazardous	Negligible	Contact-handled LLW: loading (m^3/yr)	940
Sewage	Negligible	Contact-handled LLW: storage (m^3/yr)	230
Air Emissions			
Fugitive dust (t)	37	Radiological	Negligible
NO_x (t)	6.4	Hazardous	Negligible
SO_x (t)	0.5	Potential pollutant	Negligible

TABLE 4.3 Regionalization Alternative: Case 2 (storage capacity of 2,560 canisters per facility, six facilities at Hanford)

Construction		Operations	
Parameter	Quantity	Parameter	Quantity
Resource Requirements			
Water (m^3)	99	Water (m^3/yr)	
Concrete (m^3)	281,000	Loading	11,000
Steel (t)	46,000	Storage	2.8
		Electricity (GWh/yr)	
		Loading	0.9
		Storage	Negligible
Land Requirements			
Occupied land (ha)	0.47	Number of floors	1
Disturbed land (ha)	0.85	Facility size (m^2)	4,700
Employment			
Total effort (worker-yr)	56	Loading (FTE)	53
Peak (persons)	430	Storage (FTE)	9
Construction duration (month)	54	Support (FTE)	19
Costs			
Capital cost ($ million)	≈ 45	Loading O&M ($ million/yr)	16
		Storage O&M ($ million/yr)	0.73
Secondary Waste Streams			
Hazardous	Negligible	Contact-handled LLW: loading (m^3/yr)	1,800
Sewage	Negligible	Contact-handled LLW: storage (m^3/yr)	230
Air Emissions			
Fugitive dust (t)	36	Radiological	Negligible
NO_x (t)	6.4	Hazardous	Negligible
SO_x (t)	0.5	Potential pollutant	Negligible

TABLE 4.4 Centralization Alternative: Case 1 (storage capacity of 2,530 canisters per facility, seven facilities at Hanford)

Construction		Operations	
Parameter	Quantity	Parameter	Quantity
Resource Requirements			
Water (m^3)	99	Water (m^3/yr)	
Concrete (m^3)	283,000	Loading	17,000
Steel (t)	46,000	Storage	2.8
		Electricity (GWh/yr)	
		Loading	1.1
		Storage	Negligible
Land Requirements			
Occupied land (ha)	0.47	Number of floors	1
Disturbed land (ha)	0.85	Facility size (m^2)	4,700
Employment			
Total effort (worker-yr)	56	Loading (FTE)	63
Peak (persons)	440	Storage (FTE)	9
Construction duration (month)	54	Support (FTE)	19
Costs			
Capital cost ($ million)	≈ 45	Loading O&M ($ million/yr)	19
		Storage O&M ($ million/yr)	0.73
Secondary Waste Streams			
Hazardous	Negligible	Contact-handled: LLW loading (m^3/yr)	2,700
Sewage	Negligible	Contact-handled: LLW storage (m^3/yr)	230
Air Emissions			
Fugitive Dust (t)	37	Radiological	Negligible
NO_x (t)	6.4	Hazardous	Negligible
SO_x (t)	0.5	Potential pollutant	Negligible

4.2.4.2 Case 2

This case assumes that the geologic repository is unavailable to receive HLW canisters in FY 2015 and all HLW is stored temporarily at Hanford. A total of 21,612 canisters is to be stored at Hanford in 10 facilities (each with a storage capacity of approximately 2,161 canisters). Design characteristics of an interim storage facility for 2,160 HLW canisters at the Hanford Site are given in Table 4.5; the loading/unloading rate of 980 canisters per year is based on Hanford's annual production rate of 790 canisters per year plus SRS's average annual production rate of 190 canisters per year.

TABLE 4.5 Centralization Alternative: Case 2 (storage capacity of 2,160 canisters per facility, fourteen facilities at Hanford)

Construction		Operations	
Parameter	Quantity	Parameter	Quantity
Resource Requirements			
Water (m^3)	88	Water (m^3/yr)	
Concrete (m^3)	223,000	Loading	17,000
Steel (t)	37,000	Storage	2.2
		Electricity (GWh/yr)	
		Loading	1.1
		Storage	Negligible
Land Requirements			
Occupied land (ha)	0.42	Number of floors	1
Disturbed land (ha)	0.75	Facility size (m^2)	4,200
Employment			
Total effort (worker-yr)	46	Loading (FTE)	63
Peak (persons)	360	Storage (FTE)	9
Construction duration (month)	49	Support (FTE)	17
Costs			
Capital cost ($ million)	≈ 40	Loading O&M ($ million/yr)	19
		Storage O&M ($ million/yr)	0.59
Secondary Waste Streams			
Hazardous	Negligible	Contact-handled: LLW loading (m^3/yr)	2,700
Sewage	Negligible	Contact-handled: LLW storage (m^3/yr)	180
Air Emissions			
Fugitive Dust (t)	29	Radiological	Negligible
NO_x (t)	5.1	Hazardous	Negligible
SO_x (t	0.4	Potential pollutant	Negligible

5 COST ANALYSIS

5.1 METHODOLOGY AND ASSUMPTIONS

Waste management costs can be a very significant concern in any operation. To reach sound management decisions for handling both radioactive and nonradioactive wastes, a knowledge of the life-cycle costs (LCCs) is required (DOE 1983, 1986c, 1990a). The LCCs are the "cradle to grave" costs, that is, the costs incurred from the time the waste is generated to the end of its institutional control. The total LCC must include all costs associated with waste handling following its generation, current storage and treatment, transportation, future disposal, and monitoring. The WM PEIS will not analyze environmental impacts of HLW disposal at Yucca Mountain or alternative locations for a geologic repository. Because of the possibility of a prolonged delay of HLW disposal, alternatives for longer interim storage of HLW will be addressed in the WM PEIS.

Projecting the cost of the alternatives involves developing estimates of the individual cost components. The cost is divided into two components: capital investment and costs associated with annual operating charges. The capital cost of a facility includes process equipment, construction materials (e.g., steel and concrete), and labor, as well as indirect costs such as those for design, contingencies, and environmental compliance. The annual O&M costs are expenses for operation and maintenance staff, fixed and variable supplies, annual operating fees, administration and general expenses. These two cost components were estimated by updating available data on the cost of treatment, storage, and transportation of HLW. These costs cover all the major elements that contribute to the LCC. To facilitate comparison, data normalization was necessary but was limited principally to adjustments of all costs to beginning-of-year (BOY) 1994 dollars. Escalation factors used for these adjustments were obtained from the U.S. Department of Commerce Survey of Current Business (DOC 1994).

Given the current operations schedule of the Savannah River DWPF, it may be possible to transport the canisters directly from West Valley to SRS without ever placing the canisters in interim storage at West Valley, thereby reducing costs at West Valley. This action was not considered because it is not discussed in the Implementation Plan.

5.2 CURRENT HLW STORAGE

The unit cost of HLW storage was estimated on the basis of annual operating cost data from SRS, Hanford, and INEL. It was assumed that no capital investments would be required in the near-term, so that the only relevant costs are those from annual operations. Inclusion of tank farm upgrades, construction of new tanks, or the final D&D of the tank farms would increase the unit storage cost above that reported in this section.

The SRS has a large HLW inventory, with approximately 32 million gal in underground, double-walled tanks. The unit cost of operation of the HLW tank farms at SRS was estimated on the basis of an annual operating cost for the HLW tank farms ($162.901 million/yr [Street et al. 1992]), which was divided by the total inventory at the time the cost estimate was generated (32 million gal) to arrive at a value of $5.5/yr/gal HLW in BOY 1994 dollars.

Approximately 61 million gal of HLW are stored at Hanford. The total funding for the waste tank safety and operations program at Hanford was reported to be $166.8 million for FY 1992 (DOE 1992b); the HLW inventory at that time was 24.5 million gal of DST waste and 36.2 million gal of SST waste for a total of 60.8 million gal (ORNL 1992b). The unit storage cost is therefore $2.8/yr/gal HLW in BOY 1994 dollars. The Westinghouse Hanford Company completed a study in which it was determined that the funding requirements for operation of the tank farms in compliance with DOE and industry standards would require $262 million per year (DOE 1992b); this equates to a unit storage cost of $4.5/yr/gal HLW in BOY 1994 dollars. Therefore, the unit storage cost at Hanford ranges between $2.8 to $4.5/yr/gal HLW in BOY 1994 dollars.

Funding allocated during FY 1991 for HLW tank farm/calcine storage at INEL was reported to be $2.59 million (DOE 1991a). The inventory at INEL during 1991 was 6,800 m^3 (1.8 million gal) of liquid HLW and 3,600 m^3 (130,000 ft^3) of solid calcine, for a total of 10,400 m^3 (2.75 million gal). The unit storage cost at INEL is estimated to be $0.9/yr/gal HLW in BOY 1994 dollars. This assumes that the calcine requires the same level of management as liquid HLW, which may not be correct, because its more stable form may require less surveillance.

The 1991 inventory of HLW at WVDP included approximately 1,729 m^3 (460,000 gal) stored in two underground tanks. In this analysis, it is assumed that HLW pretreatment has been completed at West Valley, and, therefore, current storage and pretreatment are excluded from further analysis in the WM PEIS.

Table 5.1 compares the individual unit storage cost values. The unit storage cost ranges from $0.9 to $5.5/yr/gal HLW. This comparatively small range is remarkable given the differences in physical and chemical form, amount in storage, and number of storage vessels.

5.3 HLW PRETREATMENT/TREATMENT

The major facility involved in HLW treatment at SRS is the DWPF. Table 5.2 gives recent cost estimates for DWPF, along with some of the assumptions in making these estimates. The in-tank precipitation process is estimated to cost $97 million in BOY 1994 dollars. The treatment facility, the actual vitrification plant, is projected to cost $2.25 billion. Investment costs of $83 million are projected for the interim storage facility. Additional support facilities are expected

TABLE 5.1 Unit Storage Cost of HLW at the Three Main DOE Sites

HLW Site	Annual Storage Cost ($ million/yr)	HLW Inventory (million gal)	Unit Storage Cost (BOY 1994 $/yr/gal HLW)
SRS	162.091	32	5.5
Hanford	166.8 to 262	60.8	2.8 to 4.5
INEL	2.59	2.75	0.9
WVDP	NA[a]	NA	NA

[a] NA = not applicable.

TABLE 5.2 Cost Summary for HLW Pretreatment and Treatment at SRS

Capital Cost (millions of BOY 1994 dollars)[a]				
In-Tank Precipitation	DWPF	Interim Storage[b]	Other DWPF Support Facilities	Annual Operating Cost ($ million/yr)[c]
97	2,250[d]	83	1,730	168

[a] Source: GAO (1992).

[b] Covers storage facilities for canisters and for failed equipment, such as melters.

[c] Source: Buice (1992).

[d] This number includes construction plus start-up costs for the DWPF. An additional $1.8 billion will be needed for DWPF support facilities to bring the total to about $4 billion.

to add an additional $1.73 billion. The total capital investment, including the support facilities comes to about $4.2 billion in BOY 1994 dollars. Annual O&M costs are given for the whole facility (pretreatment, treatment, and interim storage) as $168 million.

Once the DWPF becomes operational, DOE estimates that it will take approximately 17 years before all waste is vitrified (ORNL 1992b). The (undiscounted) total LCC is estimated to be approximately $6.7 billion in 1992 dollars. A total of 5,242 vitrified-HLW canisters are projected to be generated during the 17 years of operations; each canister will contain approximately 9,014 gal (34.1 m^3) of SRS HLW (Choi and Fowler 1990). The unit cost of HLW treatment is approximately

$150/gal ($0.6/m^3) ($7.1 billion divided by 5,242 canisters, with 9,014 gal [34.1 m^3] HLW per canister).

A unit treatment cost of $26.29/gal ($0.1/m^3) HLW at SRS has been quoted in the literature (Street et al. 1992). This lower value results from the cost of the existing HLW treatment facilities at SRS being considered to be "sunk," that is, already incurred and not impacting current or future waste generation and/or treatment. However, exclusion of these sunk costs does not allow a full-cost recovery determination for the LCC.

The DOE plans to construct a facility similar to the DWPF at SRS for the Hanford site in Washington State; this facility is called the HWVP. Table 5.3 gives cost information for HLW pretreatment and treatment of the DST waste at Hanford. The total projected capital cost of the HWVP will be approximately $1.2 billion (1991 dollars) (LaRue and Cross 1991). This cost estimate is only for the vitrification plant itself and its auxiliary buildings and structures (including the CSB for glass canister interim storage) and does not include costs for pretreatment facilities. The pretreatment option to be used at Hanford has yet to be defined. Depending upon the pretreatment option(s) chosen, the capital cost may be increased by anywhere from $1.5 billion to over $3 billion (1991 dollars) (Grygiel 1991); the latest estimate is $1.7 billion for pretreatment of the DST waste using advanced actinide separation (Boomer 1992). This puts the total capital cost of the HWVP (including pretreatment facilities) from $2.7 billion to over $4.2 billion. The total annual operating costs (both fixed and variable) are estimated to be between $67 million (Boomer 1992) and $80 million (Stegen 1992), including the operating charges for the pretreatment facility. The largest component of the annual operating cost was attributable to fixed costs; the variable component consisted primarily of consumables such as frit, the cost of which is a few million dollars.

Only very broad estimates of the LCC for HLW pretreatment/treatment at Hanford can be found in the literature. For purposes of this cost analysis, all waste in the DSTs and SSTs at Hanford will be considered to be HLW. One recent report placed the LCC between $25 to $45 billion for all HLW at Hanford, including that contained within the SSTs; this cost estimate included research and development, waste characterization, waste pretreatment and facilities, vitrification facility design and construction, on-site canister storage facilities, and operational and other capital costs necessary to prepare and store the HLW pending shipment to a geologic repository (GAO 1993). Retrieval, however, was not mentioned. The cost to retrieve the HLW in the DSTs and SSTs has been projected to exceed $16 billion in BOY 1994 dollars. The latest numbers indicate that the total LCC could approach $53 billion in BOY 1994 dollars for the included research and development, waste characterization, waste pretreatment and facilities, vitrification facility design and construction, on-site canister storage facilities, and operational and other capital costs necessary to prepare and store the HLW pending shipment to a geologic repository (GAO 1993). Retrieval, however, was not mentioned. The cost to retrieve the HLW in the DSTs and SSTs has been projected to exceed $16 billion in BOY 1994 dollars. The latest numbers indicate that the total LCC could approach $53 billion in BOY 1994 dollars for the HLW at Hanford. The current HLW inventory at Hanford

TABLE 5.3 Cost Summary for HLW Pretreatment and Treatment of Double-Shell Tank Waste at Hanford

Capital Cost ($ million)[a]				Total Annual O&M Cost ($ million/yr)[b]	
Pretreatment	HWVP	Interim Storage[c]	Auxiliary Buildings[d]	Staff	Materials
1,700	1,060	35	105	65.5	1.5

[a] Source: Boomer (1992); costs given in 1992 dollars.

[b] Annual costs determined from Boomer (1992) assuming a 40-year pretreatment campaign.

[c] Interim storage cost taken from Boomer (1992) for a storage facility for 2,000 HLW canisters.

[d] Determined by subtracting costs of other facilities ($1,095 million) from total cost ($1,200 million).

includes 36 million gal (≈140,000 m^3) of SST waste and 25 million gal (≈95,000 m^3) of DST waste, for a total of 61 million gal (≈230,000 m^3). The unit cost of HLW treatment at Hanford is therefore $860/gal (≈$3.3/m^3) HLW ($53 billion per 61 million gal [≈230,000 m^3]) HLW.

The WVDP is being conducted at the WNYNSC in order to conduct a HLW management operation to immobilize the 660,000 gal (≈25,000 m^3) of HLW stored on-site. This analysis assumes that the pretreatment phase at West Valley will have been performed by 1994 and is therefore excluded from further analysis in the WM PEIS. Hot vitrification operations are expected to proceed from CY 1996 to 1998, during which a total of 300 HLW canisters will be produced. The funding support for fiscal years 1991 to 1997 is projected to approach $869 million in BOY 1994 dollars (DOE 1991a), with a FY 1997 operating budget of $142 million in BOY 1994 dollars during vitrification operations. On the basis of an estimated capital cost of $500 million and operations over three years at $135 million per year, the (undiscounted) LCC is approximately $0.9 billion. The unit cost of HLW treatment at West Valley is about $1,440/gal HLW in BOY 1994 dollars.

Pretreatment at INEL is divided into two steps: one is the on-going calcination of liquid HLW at the ICPP and the second is projected to start operations in FY 2014. Primary pretreatment is currently performed at INEL in the NWCF; the annual operating cost has been reported to be approximately $15.3 million per year (DOE 1991b). The secondary pretreatment process will be located in the IWIF. The treatment process for HLW at INEL is assumed to follow the glass-ceramic waste form since it appears to offer a promising technology for HLW immobilization at INEL;

however, other technologies are currently being considered. One estimate of the capital cost of the IWIF is between $500 million and $1 billion (Knecht 1992), less than that for the DWPF at SRS. One reason is that support facilities (such as the NWCF, calciner bins, Atmospheric Protection System for off-gas treatment, etc.) that can be utilized for the IWIF already exist at INEL, which is not the case at SRS. The existing support facilities may, however, reach the end of their operating lives before the presumed startup (FY 2014) of the IWIF; the cost to refurbish (or potentially replace) these facilities was not considered. The capital cost for a conceptual glass-ceramic immobilization plant that would process a total of 3,400 canisters has been reported to be $500 million in 1981 dollars (Grantham et al. 1983). The capital cost for an 8,500 total canister plant becomes approximately $1.4 billion when the "six-tenth" factor is used for scaling and the 1981 cost is escalated to 1994 dollars. The upper value of $1 billion given recently by Knecht (1992) appears to be more reasonable and has been used in this analysis. The total annual operating cost is estimated to be on the order of $250 million per year (Berreth and Knecht 1987). The LCC for a glass-ceramic immobilization plant operating a total of 35 years (FY 2015-2049) has been reported to be about $9.44 billion with a capital cost of about $1.6 billion (Cornelius 1994); these costs include the glass-ceramic facility, equipment, calcine conveyance system, support facilities, support facilities equipment, and balance of plant utilities upgrades. On the basis of a total glass-ceramic volume of 4,865 m^3 (171,783 ft^3), 0.47 m^3 (16.5 ft^3) calcine per m^3 liquid, and between 0.49 m^3 (17.3 ft^3) (aluminum calcine) to 0.62 m^3 (21.8 ft^3) (zirconium calcine) per m^3 of glass-ceramic, the unit treatment cost at INEL ranges from $1,700 to $2,100/gal liquid HLW. The higher unit cost in comparison with the other three sites reflects the additional costs of interim stabilization by calcination and calcine retrieval. Cost information is presented in Table 5.4. These values should be considered very preliminary because of the uncertainty of the immobilization process to be chosen (borosilicate glass or glass-ceramic), overall process design, starting date for immobilization, and other issues for development.

Table 5.5 compares the unit treatment cost as a function of DOE site. The unit treatment cost ranges from $147 to $2,100/gal HLW because of the differences in physical form, chemical constituents, amount to be treated, annual processing rate, and other variables. The high unit treatment cost (relative to other radioactive waste categories such as low-level, low-level mixed, etc.) is due to the remote operations from heavy protective shielding required to confine penetrating radiation. The treatment cost, however, when placed on an unit canister basis (Table 5.5) ranges from only $1.1 to $3.8 million per HLW canister.

5.4 INTERIM CANISTER STORAGE

The canisters of vitrified HLW from Hanford, SRS, and WVDP would be placed in an interim on-site storage facility awaiting transport to a geologic repository. The operating lifetime of the various interim storage facilities is assumed to be 50 years; the actual value will depend on the transportation schedule of the HLW canisters to a geologic repository for permanent disposal and on the transportation schedule of the WVDP canisters to Hanford or SRS.

TABLE 5.4 Cost Summary for HLW Pretreatment and Treatment at INEL[a]

Capital Cost ($ million)				Total Annual O&M Cost ($ million/year)		
NWCF[b]	IWIF	Interim Storage[c]	Auxiliary Buildings[b]	NWCF[d]	IWIF	Interim Storage[e]
NA	1,000	2	N/A	15.3	250	0.4

[a] An LCC of $9.44 billion has been reported for the glass-ceramic facility, equipment, calcine conveyance system, support facilities, support facilities equipment, and balance of plant utilities upgrades (Cornelius 1994).

[b] These facilities have been constructed and operated over a number of years. The capital cost to construct a similar facility today could not be estimated based on previous costs because of varying cost escalation, regulations, etc.

[c] Cost of HLW interim storage based on a total of 327 repository canisters at the INEL and a capital cost of $114.76 million for a total of 18,720 drywells ($6,100 per drywell) (Fletcher and Smith 1989). This reference also states that the D&D cost is 10% of initial capital cost.

[d] Annual operating charges assuming 184 full-time equivalents (FTEs) at a fully loaded annual cost of $73,000/yr and a materials cost of $1.8 million (DOE 1991b).

[e] Annual operations charges based on a dedicated crew of six FTEs to load a HLW repository canister into a drywell and a fully loaded cost of $73,000 per FTE.

Because of the lack of information on the shipping rate to be assumed for the various alternatives, the loading rate during interim storage is conservatively taken to equal the average annual vitrification rate (190 canisters per year for SRS, 790 canisters per year for Hanford). The various cost components for the different HLW alternatives are presented in Table 5.6. The (undiscounted) LCC is given by the following equation:

$$[\textit{Life-cycle cost}\ (\$)] = [\textit{Capital cost}\ (\$)] + [\textit{Loading/unloading duration}\ (\text{yr})] \times [\textit{Annual O\&M: loading}\ (\$/\text{yr})] + [\textit{Storage duration}\ (\text{yr})] \times [\textit{Annual O\&M: storage}\ (\$/\text{yr})]. \quad (5.1)$$

Because the loading/unloading duration is dependent on the shipping rate, the operation durations of the various interim glass canister storage facilities are unknown. The 50-year LCC was estimated assuming that the total time for unloading the HLW canisters prior to shipment to the national geologic repository would be equal to the total loading time.

TABLE 5.5 Cost Summary for HLW Pretreatment and Treatment

DOE Site	Total Projected Number of HLW Canisters	Unit Pretreatment/Treatment Cost (BOY 1994 $)	
		$/gal HLW	$ million/HLW canister
SRS	5,242	150	1.4
Hanford	14,000 - 37,000	860	1.5 - 3.8
INEL	8,500[a]	1,700 - 2,100	1.1
WVDP	300	1,440	3.2

[a] The number of HLW canisters is not given in Cornelius (1994) nor is the canister size waste loading. Preliminary estimates based on a 70% waste loading and a canister volume of 0.6 m^3 result in an approximate total of 8,500 canisters.

TABLE 5.6 Cost Characteristics of Interim Glass Canister Storage for the Various HLW Alternatives[a]

HLW Alternative	Facility Name	HLW Site	Facility Capacity (HLW) Canisters)	Total Number of Storage Facilities	Capital Cost ($ million)	Annual O&M Cost ($ million/yr)	
						Loading/ Unloading	Storage Only
Decentralization	MCSB	Hanford	2,500	6	≈ 45	16 [b]	0.71
Regionalization, Case 1	SGCSB	SRS	2,626	1	≈ 45	8.2 [c]	0.75
Regionalization, Case 2	MCSB	Hanford	2,560	6	≈ 45	16 [b]	0.73
Centralization, Case 1	MCSB	Hanford	2,575	7	≈ 45	17 [d]	0.73
Centralization, Case 2	MCSB	Hanford	2,160	10	≈ 41	17 [d]	0.62

[a] Abbreviations: SGCSB = second glass canister storage building and MCSB = modified canister storage building.

[b] Based on a (constant) loading/unloading rate of 790 HLW canisters per year.

[c] Based on a (constant) loading/unloading rate of 190 HLW canisters per year.

[d] Based on a (constant) loading/unloading rate of 980 HLW canisters per year (Hanford's annual production rate of 790 canisters per year plus SRS's annual production rate of 190 canisters per year).

Table 5.7 gives the LCCs based on a total operations period of 50 years for the various interim storage facilities. The differential LCC for the situation where both WVDP and SRS canisters are shipped to Hanford was determined by subtracting both the costs of current design facilities at Hanford and SRS. Neither the annual storage costs (on the order of $100,000/yr) nor the capital costs associated with construction of a receiving and shipping area at the WVDP for off-site canister shipments (on the order of $2 million, based on floor area of approximately 3,500 ft^2 at $500/$ft^2$ for a shipping rate of 100 canisters per year) were included in the comparison of storage costs at Hanford and SRS.

Analysis of the results in Table 5.7 indicates that the cost of interim storage of WVDP, INEL and SRS canisters at Hanford may not result in any appreciable cost savings. In addition, the difference in interim storage costs of the additional 300 WVDP canisters at either Hanford or SRS is not significant. Given the broad estimates of the capital and annual O&M costs, it does not appear possible to distinguish between the costs of storing the 300 WVDP canisters at Hanford and at SRS. However, it should be noted that the cost estimates given in this section are order-of-magnitude values based on similar previous cost data and as such may neglect design differences due to site characteristics. In addition, it was assumed that the difference in canister characteristics between INEL's and the other three HLW site's canister designs would not require any modification of the interim storage technology proposed for the Hanford and SRS canisters. Also, transportation costs have not been included in the results summarized in Table 5.7.

5.5 TRANSPORTATION OF HLW CANISTERS

Very limited cost information exists for shipment of HLW canisters. However, on the basis of numerous reports, there is general agreement that transportation costs for HLW would be similar to SNF, for which cost data are available. The total cost of transport is defined to be the sum of capital costs, maintenance costs, and shipping costs. A unit transportation cost (by truck) of $10,000 per canister has been quoted in the literature (Tang and Saling 1990; DOE 1990a). The unit transportation cost was developed on the basis of a total of 17,750 HLW canisters generated from defense operations; these canisters would be shipped from the three major HLW sites (Hanford, INEL, and SRS) to Yucca Mountain. Detailed cost algorithms exist to estimate the cost of shipping SNF in repository casks from a reactor site to another site, including the national geologic repository (DOE 1986a, 1991c; Tang and Saling 1990). In general, to calculate the total required number of transportation casks, average speeds, cask turnaround times, and availability of casks must be known or estimated. This then determines the required number of shipping casks to support the scheduled shipment of HLW canisters from one site to another and from a given site to the geologic repository.

This report utilizes a cost estimation procedure developed by DOE for the MRS study (DOE 1986a), as more recent cost models do not allow comparison of truck and rail costs on a

TABLE 5.7 Life-Cycle Cost Analysis for HLW Management Alternatives

Variable	Savannah River Site[a]		Hanford Site			
	Current Design[b]	Including 340 WVDP Canisters[c]	Current Design[b]	Including 340 WVDP Canisters[d]	Including 340 WVDP Canisters, and 2,373 SRS Canisters[e]	Including 340 WVDP Canisters, 4,572 SRS Canisters, and 1,700 INEL Canisters[f]
No. of canisters in new Interim Storage Facility(ies)	2,286	2,626	15,000	15,300	17,713	21,612
Capital cost ($ million)	≈43	≈45	≈45 × 6 ≈270	≈45 × 6 ≈270	≈45 × 7 ≈315	≈40 × 10 ≈410
Annual O&M, loading ($ million/yr)	≈8	≈8	≈16	≈16	≈17	≈17
Annual O&M, storage ($ million/yr)	≈0.66	≈0.75	≈0.71	≈0.73	≈0.72	≈0.62
LCC ($ million)	≈273	≈279	≈886	≈888	≈1,001	≈1,091
Δ LCC ($ million)[g]		≈6		≈2	≈115[h]	≈205[h]

[a] Excluding current interim storage capacity of 2,286 canisters. It is assumed that SRS will require additional canister storage space above that currently planned (2,286 canisters) because of the unavailability of a national repository at the time the current facility becomes full.

[b] Decentralization alternative.

[c] Regionalization alternative, Case 1. Assumes 24 total years for loading/unloading.

[d] Regionalization alternative, Case 2. Assumes 24 total years for loading/unloading.

[e] Centralization alternative, Case 1. Assumes 38 total years for loading/unloading.

[f] Centralization alternative, Case 2. Assumes 38 total years for loading/unloading.

[g] Change in LCC between current design and alternatives.

[h] Figure does not include the potential LCC savings associated with direct shipment of INEL and SRS canisters to Hanford, without ever placing the HLW canisters in interim storage at the INEL and SRS, respectively. The potential LCC savings could not be determined due to lack of information.

consistent basis. The LCC for HLW transportation can in general be calculated by summing over the following cost categories:

1. Shipping cost,
2. Security cost,
3. Cask capital and decommissioning cost,
4. Cask maintenance cost,
5. Inspection cost,
6. Demurrage cost,
7. Handling cost (loading and unloading), and
8. Transportation support system costs.

The following assumptions are made in this analysis:

1. The costs associated with the Inspection cost category are included in the annual operating charges for the various HLW interim storage facilities;
2. Demurrage is defined to be the charge for the detention of a freight car or truck by the shipper or receiver beyond the time allowed for loading, unloading, or shipping. It is assumed to be negligible in comparison with the other cost components (this component is actually not applicable to rail shipping);
3. The handling cost for loading and unloading at the HLW interim storage facility have already been considered; the handling cost at the repository is assumed to be out-of-scope;
4. Transportation support system costs include the costs to maintain the railcars and trailers which are assumed to be negligible as the average annual O&M cost is approximately $14,000 for a truck-trailer and $5,000 for a railcar.

The LCC therefore becomes the sum of the shipping, security, cask capital & decommissioning, and cask maintenance costs. The overall cost estimating methodology, assumptions, and cost data used in this analysis are the same as those utilized in the document *Environmental Assessment for a Monitored Retrievable Storage Facility*, Volume II [DOE 1986a]. The transportation cost estimates were however modified to reflect the above assumptions.

Other transportation cost models were reviewed for applicability (e.g., [DOE 1987; DOE 1991c]). These models either do not consider transportation of HLW canisters (i.e., designed for shipments of spent nuclear fuel) or consider only one mode of transportation. It should be noted that fair agreement in the various models exists: [DOE 1991c] lists the capital cost of a HLW rail cask to be approximately $3 million (presumably in 1990 dollars) which is equal to the value given in [DOE 1986a] ($2.5 million in 1986 dollars, which escalates to $3 million in 1990 dollars).

The rail and truck distances between the four HLW sites and the repository were provided by the ANL Transportation group. The average rate for rail shipments is not given in [DOE 1986a] (the text states that the "average rate . . . varies from 5 km/h for short hauls to 19 km/h for cross-country shipments"). It was however estimated by the following equation, which is based on the U.S. average for a dedicated train shipment [DOE 1991c]:

$$[\textit{Avg. Rail Speed}\ (\text{mph})] = [\textit{Distance traveled}\ (\text{one-way miles})] \,/\, (0.04204 \times [\textit{Distance traveled}\ (\text{one-way miles})] + 4)$$

The cost relationships assumed in this analysis are presented in Table 5.8. A factor of 1.3 (based on the GNP Deflator) is used to convert 1985 dollars to BOY 1994 dollars. The number of HLW canisters to be shipped from site-to-site and to the repository were taken from the HLW alternatives (Chapter 4).

The results of the transportation-cost estimates as a function of HLW alternative are presented in Table 5.9 for rail-based shipping, and in Table 5.10 for truck-based shipping. In general, cask maintenance costs contribute least to the total cost and represent about 4 to 6 percent of the total transportation estimate. The major difference in cost between truck and rail transport is the cost of security/safeguards; this is due to the regulations that require road shipments within heavily populated areas to be accompanied by armed escorts in separate vehicles, with two armed escorts required for rail shipments. Cask capital costs for truck transport are less than for rail due to the higher truck speed assumed in this study which in turn decreases the annual cask requirement.

Comparison of the estimated shipping costs for rail and truck transport are shown in Table 5.11. The estimated transportation cost for the Decentralized alternative is the lowest among the four alternatives studied. This reflects the fact that it is less costly to ship HLW canisters from the four HLW sites directly to the geologic repository than to ship the HLW canisters to a second HLW site and then on to the geologic repository, due to the shorter shipping distance. The highest cost is predicted for centralization at Hanford (due to the increased shipping distances) while it is not possible to distinguish between costs for Regionalization Alternative Cases 1 and 2.

TABLE 5.8 Assumed Relationships for the Four Transportation Cost Components

Cost Variable	Relationship Assumed in This Study	
	Rail-Based	Truck-Based
SPEED (mph)	DIS/(0.04204 × DIS + 4)[a]	35 (i.e., a constant value)[b]
Annual Cask Requirement (ACR)	$\sum$ {[(2 × DIS)/SPEED$_{rail}$]/24 + 2 × (5 days)}[c]	$\sum$ {[(2 × DIS)/SPEED$_{truck}$]/24 + 2 × (3 days)}[c]
Shipping Cost, less than 1,000 miles ($1985)	$\sum$ ([2.32+0.0067 × DIS] × 2,000) + ([2.15+0.0063 × DIS] × 1,800) × [No. of Canisters] / 5[c,d]	$\sum$ ([1.493+0.0033 × DIS] × 500) + ([0.428+0.0034 × DIS] × 475) × [No. of Canisters][c]
Shipping Cost, greater than 1,000 miles ($1985)	$\sum$ ([5.07+0.004 × DIS] × 2,000) + ([4.72+0.0037 × DIS] × 1,800) × [No. of Canisters] / 5[c,d]	$\sum$ ([-0.16+0.0049 × DIS] × 500) + ([-0.19+0.004 × DIS] × 475) × [No. of Canisters][c]
Security Cost ($1985)	$\sum$ { 291.65 × [DIS$^{-0.5987}$] × DIS } × [No. of Canisters] / 5	$\sum$ { 7.93 × [DIS$^{-0.1855}$] × DIS } × [No. of Canisters]
Cask Capital Cost ($1985)[e]	$\sum$ (ACR/300) × [No. of Canisters] × (2.5 × 10^5)	$\sum$ (ACR/300) × [No. of Canisters] × (1.5 × 10^5)
Cask Maintenance Cost ($1985)[f]	$\sum$ (ACR/300) × [No. of Canisters] × (2.5 × 10^4)	$\sum$ (ACR/300) × [No. of Canisters] × (1.5 × 10^4)

[a] DIS = distance traveled (one-way miles); is a function of WM PEIS alternative.

[b] Conservative value, based on DOE (1986a); a value of 40 mph is cited in DOE (1991c).

[c] The summations are to be performed over all shipping routes.

[d] Assumes five HLW canisters per rail shipping cask, one HLW canister per truck shipping cask (DOE 1986a).

[e] Assumes a capital cost of $2.5 million for rail cask, $1.5 million for truck cask (both in $1985 dollars) (DOE 1986a).

[f] Assumes an annual maintenance cost of $125,000 for rail cask, $75,000 for truck cask (both in $1985 dollars) (DOE 1986a).

This analysis predicts rail transport to be more costly than truck transport, which is in general agreement with the results of the MRS study (DOE 1986a). This study's cost estimate for the Decentralized alternative is consistent with the estimated cost for defense high-level waste transportation to a single repository in DOE (1990a).

TABLE 5.9 Rail Transportation Cost for HLW

Total Rail Transportation Cost — No Action

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Rail ($1985)	Total Rail ($BOY 1994)
WVDP	Yucca Mountain	340	2,540	4,088	3,799,323	461,032	4,141,172	414,117	8,815,644	1.14E+07
SRS	Yucca Mountain	4,572	2,839	4,569	55,097,849	6,482,663	57,682,060	5,768,206	125,030,778	1.62E+08
INEL	Yucca Mountain	0	756	1,217	1,850,514	0	0	0	1,850,514	2.40E+06
Hanford	Yucca Mountain	15,000	1,302	2,095	113,169,960	15,555,237	155,591,708	15,559,171	299,876,076	3.89E+08
	TOTAL:	19,912	33,373,508	53,706,965	173,917,646	22,498,932	217, 414,940	21,741,494	435,573,012	5.66E+08

Total Rail Transportation Cost—Decentralization

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Rail ($1985)	Total Rail ($BOY 1994)
WVDP	Yucca Mountain	340	2,540	4,088	3,799,323	461,032	4,141,172	414,117	8,815,644	1.14E+07
SRS	Yucca Mountain	4,572	2,839	4,569	55,097,849	6,482,663	57,682,060	5,768,206	125,030,778	1.62E+08
INEL	Yucca Mountain	1,700	756	1,217	1,850,514	1,417,398	16,278,813	1,627,881	21,174,606	2.75E+07
Hanford	Yucca Mountain	15,000	1,302	2,095	113,169,960	15,555,237	155,591,708	15,559,171	299,876,076	3.89E+08
	TOTAL:	21,612	34,658,708	55,775,198	173,917,646	23,916,329	233,693,753	23,369,375	454,897,103	5.91E+08

Total Rail Transportation Cost—Regionalization Case 1

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Rail ($1985)	Total Rail ($BOY 1994)
WVDP	SRS	340	1,217	1,958	2,480,451	343,161	3,484,559	348,456	6,656,627	8.64E+06
SRS	Yucca Mountain	4,912	2,839	4,569	59,195,239	6,964,751	61,971,627	6,197,163	134,328,780	1.74E+08
INEL	Yucca Mountain	1,700	756	1,217	1,850,514	1,417,398	16,278,813	1,627,881	21,174,606	2.75E+07
Hanford	Yucca Mountain	15,000	1,302	2,095	113,169,960	15,555,237	155,591,708	15,559,171	299,876,076	3.89E+08
	TOTAL:	21,952	35,174,148	56,604,680	176,696,164	24,280,547	237,326,708	23,732,671	462,036,089	6.00E+08

TABLE 5.9 (Cont.)

Total Rail Transportation Cost — Regionalization Case 2

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Rail ($1985)	Total Rail ($BOY 1994)
WVDP	Hanford	340	2,654	4,271	3,912,968	469,226	4,197,751	419,775	8,999,719	1.17E+07
SRS	Yucca Mountain	4,572	2,839	4,569	55,097,849	6,482,663	57,682,060	5,768,206	125,030,778	1.62E+08
INEL	Yucca Mountain	1,700	756	1,217	1,850,514	1,417,398	16,278,813	1,627,881	21,174,606	2.75E+07
Hanford	Yucca Mountain	15,340	1,302	2,095	115,735,146	15,907,822	159,118,454	15,911,845	306,673,267	3.98E+08
	TOTAL:	21,952	35,140,148	56,549,965	176,596,476	24,277,109	237,277,077	23,727,708	461,878,370	6.00E+08

Total Rail Transportation Cost—Centralization Case 1

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Rail ($1985)	Total Rail ($BOY 1994)
WVDP	Hanford	340	2,654	4,271	3,912,968	469,226	4,197,151	419,775	8,999,719	1.17E+07
SRS	Hanford	2,373	2,953	4,752	29,390,545	3,418,271	30,333,542	3,033,354	66,175,711	8.59E+07
INEL	Hanford	0	658	1,059	1,685,647	0	0	0	1,685,647	2.19E+06
Hanford	Yucca Mountain	17,713	1,302	2,095	133,638,633	18,368,660	183,733,062	18,373,306	354,113,662	4.60E+08
SRS	Yucca Mountain	2,199	2,839	4,569	26,500,474	3,117,974	27,743,406	2,774,341	60,136,194	7.81E+07
INEL	Yucca Mountain	1,700	756	1,217	10,104,446	1,417,398	16,278,813	1,627,881	29,428,538	3.82E+07
	TOTAL:	24,325	38,500,316	61,957,380	205,232,713	26,791,529	262,286,572	26,228,657	520,539,472	6.76E+08

Total Rail Transportation Cost—Centralization Case 2

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Rail ($1985)	Total Rail ($BOY 1994)
WVDP	Hanford	340	2,654	4,271	3,912,968	469,226	4,197,751	419,775	8,999,719	1.17E+07
SRS	Hanford	4,572	2,953	4,752	56,626,031	6,585,897	58,442,879	5,844,288	127,499,094	1.66E+08
INEL	Hanford	1,700	658	1,059	1,685,647	1,340,587	16,035,623	1,603,562	20,665,419	2.68E+07
Hanford	Yucca Mountain	21,612	1,302	2,095	163,055,278	22,411,985	224,176,533	22,417,653	432,061,450	5.61E+08
	TOTAL:	28,224	43,660,900	70,262,150	225,279,923	30,807,695	302,852,786	30,285,279	589,225,682	7.65E+08

TABLE 5.10 Truck Transportation Cost for HLW

Total Truck Transportation Cost — No Action

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Truck ($1985)	Total Truck ($BOY 1994)
WVDP	Yucca Mountain	340	2,407	3,874	3,502,068	1,530,962	1,994,262	267,426	7,294,718	9.61E+06
SRS	Yucca Mountain	4,572	2,448	3,939	47,907,931	20,872,108	27,040,114	3,618,411	99,438,564	1.31E+08
INEL	Yucca Mountain	0	746	1,201	0	0	0	0	0	0.00E+00
Hanford	Yucca Mountain	15,000	1,162	1,870	73,266,750	37,322,853	65,750,000	9,575,000	185,914,603	2.45E+08
	TOTAL:	19,912	29,440,636	47,377,914	124,676,749	59,725,923	94,784,376	13,460,838	191,647,885	3.86E+08

Total Truck Transportation Cost—Decentralization

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Truck ($1985)	Total Truck ($BOY 1994)
WVDP	Yucca Mountain	340	2,407	3,874	3,502,068	1,530,962	1,994,262	267,426	7,294,718	9.61E+06
SRS	Yucca Mountain	4,572	2,448	3,939	47,907,931	20,872,108	27,040,114	3,618,411	99,438,564	1.31E+08
INEL	Yucca Mountain	1,700	746	1,201	5,227,245	2,948,274	6,609,762	1,000,976	15,786,257	2.08E+07
Hanford	Yucca Mountain	15,000	1,162	1,870	73,266,750	37,322,853	65,750,000	9,575,000	185,914,603	2.45E+08
	TOTAL:	21,612	30,708,836	49,418,790	129,903,994	62,674,197	101,394,138	14,461,814	308,434,143	4.06E+08

Total Truck Transportation Cost—Regionalization Case 1

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Truck ($1985)	Total Truck ($BOY 1994)
WVDP	SRS	340	883	1,421	1,248,072	676,452	1,377,405	205,740	3,507,669	4.62E+06
SRS	Yucca Mountain	4,912	2,448	3,939	51,470,638	22,424,277	29,050,971	3,887,497	106,833,383	1.41E+08
INEL	Yucca Mountain	1,700	746	1,201	5,227,245	2,948,274	6,609,762	1,000,976	15,786,257	2.08E+07
Hanford	Yucca Mountain	15,000	1,162	1,870	73,266,750	37,322,853	65,750,000	9,575,000	185,914,603	2.45E+08
	TOTAL:	21,952	31,022,996	49,924,358	131,212,705	63,371,856	102,788,138	14,669,214	312,041,913	4.11E+08

TABLE 5.10 (Cont.)

Total Truck Transportation Cost — Regionalization Case 2

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Truck ($1985)	Total Truck ($BOY 1994)
WVDP	Hanford	340	2,556	4,113	3,722,439	1,607,720	2,054,571	273,457	7,658,188	1.01E+07
SRS	Yucca Mountain	4,572	2,448	3,939	47,907,931	20,872,108	27,040,114	3,618,411	99,438,564	1.31E+08
INEL	Yucca Mountain	1,700	746	1,201	5,227,245	2,948,274	6,609,762	1,000,976	15,786,257	2.08E+07
Hanford	Yucca Mountain	15,340	1,162	1,870	74,927,463	38,168,838	67,240,333	9,792,033	190,128,668	2.50E+08
	TOTAL:	21,952	31,154,576	50,136,106	131,785,078	63,596,940	102,944,781	14,684,878	313,011,677	4.12E+08

Total Truck Transportation Cost—Centralization Case 1

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Truck ($1985)	Total Truck ($BOY 1994)
WVDP	Hanford	340	2,556	4,113	3,722,439	1,607,720	2,054,571	273,457	7,658,188	1.01E+07
SRS	Hanford	2,373	2,727	4,388	27,745,591	11,828,685	14,822,775	1,956,878	56,353,928	7.42E+07
INEL	Hanford	0	599	964	0	0	0	0	0	0.00E+00
Hanford	Yucca Mountain	17,713	1,162	1,870	86,518,263	44,073,313	77,641,983	11,306,798	219,540,358	2.89E+08
SRS	Yucca Mountain	2,199	2,448	3,939	23,042,331	10,038,881	13,005,514	1,740,351	47,827,078	6.30E+07
INEL	Yucca Mountain	1,700	746	1,201	5,227,245	2,948,274	6,609,762	1,000,976	15,786,257	2.08E+07
	TOTAL:	24,325	34,574,069	55,638,991	146,255,869	70,496,874	114,134,606	16,278,461	347,165,809	4.57E+08

Total Truck Transportation Cost—Centralization Case 2

Route from	to	Canisters (total)	Distance (miles)	Distance (km)	Shipping Cost ($)	Security Cost ($)	Cask Capital Cost ($)	Mainten. Cost ($)	Total Truck ($1985)	Total Truck ($BOY 1994)
WVDP	Hanford	340	2,556	4,113	3,722,439	1,607,720	2,054,571	273,457	7,658,188	1.01E+07
SRS	Hanford	4,572	2,727	4,388	53,456,738	22,790,033	28,558,671	3,770,267	108,575,710	1.43E+08
INEL	Hanford	1,700	599	964	4,140,180	2,465,677	6,312,262	971,226	13,889,346	1.83E+07
Hanford	Yucca Mountain	21,612	1,162	1,870	105,562,733	53,774,767	94,732,600	13,795,660	267,865,760	3.53E+08
	TOTAL:	28,224	39,468,328	63,515,172	166,882,091	80,638,197	131,658,105	18,810,610	397,989,003	5.24E+08

TABLE 5.11 Comparison of Estimated Shipping Costs for Rail and Truck Transport

WM PEIS Alternative	Alternative Description	Transportation Cost ($BOY 1994)	
		Truck	Rail
No Action	Some canisters to Yuccan Mountain	3.86E+08	5.66E+08
Decentralization	All sites ship to Yucca Mountain	4.06E+08	5.91E+08
Regionalization, Case 1	WVDP ships to SRS	4.11E+08	6.00E+08
Regionalization, Case 2	WVDP ships to Hanford	4.12E+08	6.00E+08
Centralization, Case 1	WVDP, SRS, and INEL ship to Hanford	4.57E+08	6.76E+08
Centralization, Case 2	WVDP, SRS, and INEL ship to Hanford	5.24E+08	7.65E+08

6 UNCERTAINTIES ASSOCIATED WITH HLW

Uncertainties must be addressed in regulatory analyses, both for risk and cost measures. Uncertainties associated with risk assessments are generally difficult to estimate because best estimates of the relevant parameters are generally applied. The uncertainties in the source terms delivered to ORNL for calculation of health effects are in part due to uncertainties associated with the following tasks that were used to determine those source terms:

1. Waste characterization (inventory, isotopic and chemical composition, physical form),

2. Technology listing (technical information on existing and planned facilities), and

3. Facility assessment (for new facilities required for the various WM PEIS alternatives).

The uncertainties for the above components are addressed below.

6.1 WASTE CHARACTERIZATION

The reference case inventories used in this analysis are based on the most recent NEPA literature at the time of the Final WM PEIS (DOE 1996). However, the waste inventories assumed in this analysis may change due to the effect of changing regulatory and legal requirements on future waste generation (e.g., more rigid compliance or disciplines in operations than in the past).

This analysis assumes that the final waste form for the INEL HLW is based on a borosilicate glass contained within a "hybrid" spent fuel repository canister. It should be noted that the other three HLW sites assume a standard DWPF canister size of 0.62 m^3.

The projected startup date for the INEL immobilization facility is assumed in this study to be FY 2015; however, the recent INEL sitewide EIS included an alternative that considers start of HLW immobilization operations in FY 2008. This difference should not significantly impact the total number of canisters produced at the INEL, nor their radioactive content.

Other uncertainties concerning the total number of HLW canisters at the INEL are as follows:

- The technologies for HLW pretreatment are currently under investigation and may change radically in the future as a result of current R&D activities.

- The waste loading (i.e., how much high-activity waste can be placed in a single HLW canister without exceeding thermal power and activity limitations) is currently uncertain awaiting R&D activities and a performance assessment of the projected canister contents within the national geologic repository. A change in the waste loading from that assumed in the document *ICPP Radioactive Liquid and Calcine Waste Technologies Evaluation Interim Report* (WINCO 1994) will result in a revised number of canisters at the INEL.

HLW vitrification is underway at the DWPF at the SRS. Uncertainties concerning the total number of HLW canisters at the SRS are as follows:

- A number of recent EISs such as the *Interim Management of Nuclear Materials* (DOE 1995g) deal with potential reprocessing of existing nuclear materials resulting in additional HLW generation at the SRS. The EISs that could impact the future inventory of HLW at the SRS are listed in Chapter 1 of the Final WM PEIS.

- As operations continue to occur at the DWPF, changes to the current treatment process may be implemented which would impact the canister inventory. These operational changes may be enacted due to changes in the HLW inventory resulting from actions such as recovery of highly enriched uranium, processing of plutonium-238 materials, etc. which would change the chemical and radiological composition of the HLW inventory from that applied in the design of the DWPF.

The current Hanford approach is to blend the HLW retrieved from all 177 tanks prior to pretreatment. In this case, the canisters will have significantly lower values for radioactive content and thermal power than assumed in this analysis. However, because of uncertainty concerning tank retrieval and pretreatment, the data used in this study for the canister radioactivity are based on the HLW in the 10 DSTs at Hanford. This should result in conservative estimates for the risk analysis.

The specific disposal method for the cesium and strontium capsules at Hanford has not been selected. This study assumes that the capsules would be placed in overpacks for repository emplacement. Another option is to blend the cesium and strontium with the existing HLW liquid at Hanford prior to vitrification. The increase in the number of HLW canisters likely to be produced at Hanford, as well as the radionuclide content of the canisters, cannot be quantified due to lack of information on the waste loading that could be achieved.

Other uncertainties concerning the total number of HLW canisters at the Hanford Site are as follows:

- The dimensions of the HLW waste canister to be used at Hanford have not been finalized. Current legislation prohibits the placement in the first national geologic repository of spent fuel in excess of 70,000 metric tons, of which current planning allocates 10 percent for disposal of DOE-owned SNF and HLW. It is currently assumed that the 7,000 metric tons of DOE waste would be contained in approximately 18,000 standard-sized (i.e., 2-foot outside diameter by 10-foot long) canisters. There appears to be insufficient capacity in the first repository to accept all Hanford HLW. Waste package optimization may result in using a larger canister that would reduce the number of waste packages requiring handling, transport, and disposal at the national geologic repository.

- The amount and type of waste that would be removed from the tanks after retrieval is uncertain. The retrieval efficiency, which is the percentage of the tank waste that would be retrieved, is the objective of future R&D activities and may change from the values assumed in the *Draft Environmental Impact Statement for the Tank Waste Remediation System* (TWRS EIS) (DOE 1996a).

- The projected effectiveness of the chemical separations and vitrification technologies, as applied to the Hanford tank wastes, is the objective of future R&D activities and may change from the values assumed in the Draft TWRS EIS.

- The waste loading, which is the percentage of waste that is in the final vitrified form, is the objective of future R&D activities and may change from the values assumed in the Draft TWRS EIS.

The overall waste inventory at the WVDP can be considered to be well characterized. The radionuclide composition for the HLW canisters used in this study is taken from ORNL (1992a), which is based on a waste loading of 34%. The waste loading is now stated to be 40%, which would in turn increase the radionuclide content within the HLW canisters. This effect could not be incorporated due to a lack of detailed radionuclide breakdown for the three waste streams that comprise the HLW at WVDP (i.e., alkaline liquid, alkaline sludge, and acid liquid).

Other uncertainties concerning the total number of HLW canisters at the WVDP are as follows:

- The quantities and concentrations of waste projected to be HLW are uncertain, and therefore the number of HLW canisters becomes uncertain as a result. The "best estimate" of this future HLW is provided in the *Draft Environmental*

Impact Statement for Completion of the West Valley Demonstration Project and Closure or Long-Term Management of Facilities at the Western New York Nuclear Service Center (DOE and NYSERDA 1996).

6.2 TECHNOLOGY LISTING

Information about the characteristics of existing and planned facilities was taken from a variety of literature sources that tended to agree with each other. Uncertainty exists in the design of certain facilities as described below.

The DOE is considering building two vitrification plants at Hanford, one for LLW and the other for HLW (Kramer 1993). The processing rate of the HLW vitrification facility will be affected by the pretreatment option selected. Pretreatment reduces the amount of waste feed to the immobilization plant; therefore, the number of HLW canisters is dependent on the extent of the pretreatment performed. The previous DOE approach was to remove strontium, cesium, technetium, and TRUW elements from the soluble salts, which would be then combined with sludges and other streams containing high concentrations of fission products and TRUW elements. The vitrification facility design used in the WM PEIS was based on this approach. Pretreatment may now potentially include an ion-exchange process for extracting cesium and strontium from the liquid portion of the waste and sludge washing for dealing with the precipitate at the bottom of the tanks. The exact pretreatment option is to be assessed between 1994 and 1997; a design for the HLW pretreatment facility is to be completed by March 1998. The vitrification facility design used in this study, therefore, may not be the final design used to immobilize the HLW at Hanford. Due to the uncertainty in assessing which pretreatment approach will be taken in the future, it is not possible to quantify its effect on the resource requirements (both construction and operations) for the future Hanford High Level Waste Vitrification Facility.

6.3 FACILITY ASSESSMENT

The Facility Assessment of new HLW facilities in the WM PEIS focused primarily on interim storage of the vitrified HLW glass canisters. The various resource requirements for both construction (e.g., amount of structural steel) and operations (e.g., number of operating personnel) were determined by parametric scaling, based on an extensive literature review for similar facility designs. Modular sizing is dependent upon parameters such as loading capacity (HLW canisters/yr), storage capacity (number of HLW canisters), total heat release, and local thermal conditions. It was assumed in this study that the last two variables did not significantly affect various cost and performance characteristics of the interim storage facilities. The correlation coefficient (R^2) of the various correlations was always > 0.95, and generally > 0.99. Over the range in which the correlations were applied, it could be expected that some uncertainty could exist on the absolute

value for the different variables. This uncertainty should only be small (generally on the order of 10%) and can only be quantified on a variable-by-variable basis.

For comparison of interim storage facility designs for HLW with SNF, it was assumed that a HLW canister is equivalent to 0.5 MTHM of spent fuel. Analysis of the HLW canisters for SRS and Hanford indicates a heavy metal content greater than 0.5 MTHM (a difference < 10%). Given that the exponent for all correlations was always less than unity, this means that the resource requirements could be underestimated by a maximum of 10%.

The resource requirements during the loading/unloading phase of interim HLW glass canister storage were determined assuming a constant uniform shipping rate. The shipping rate of HLW canisters could be expected to vary over time due to variations in the production rate of the vitrification facility with time. The effect of this variation on the operations resource requirements cannot be quantified but could be expected to be not greater than one order of magnitude.

This study was directed to assume that DOE would construct interim glass canister storage facilities based on a modular design. The model for future storage was taken to be the GWSB at SRS. The capacity of each module was assumed to range from 2,200 to 2,900 canisters. Therefore, storage of a total of 21,612 HLW canisters at a single centralized site would require 10 interim storage facilities. The economies-of-scale associated with construction and operation of a single facility compared with a series of smaller facilities with lower capacities were not addressed, although they may be considerable (e.g., a total occupied land of 4.3 ha for 10 small facilities versus 1.5 ha for one large facility and a capital cost of $410 million for 10 facilities versus $140 million for one facility).

The number of canisters that may be placed on top of each other is determined by the thermal power of the canisters and the heat removal rate of the glass canister storage facility. The GWSB at SRS does not allow for stacking of HLW canisters on top of each other. The interim HLW canister storage design previously advanced at Hanford (Boomer 1992; Braun et al. 1993) would stack up to three canisters within a single storage sleeve. Exclusion of stacking for Hanford's canisters could increase the land occupied by the interim storage facility by up to three times.

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Wodrich, D.D., and E.A. Bracken, 1988, "Progress in Defense High-Level Waste Disposal," in *Proceedings of Waste Management '88,* Feb. 28-March 3, 1988, Tucson, Ariz., pp. 141-148.

Wodrich, D., et al., 1991, *Designation of Facility Usage Categories for Hanford Site Facilities,* WHC-SA-1145, Westinghouse Hanford Co., Richland, Wash.

Woods, W.D., et al., 1988, *MRS Action Plan Task B Report: Analyses of Alternative Designs and Operating Approaches for a Monitored Retrievable Storage Facility,* PNL-6770, Pacific Northwest Laboratory, Richland, Wash.

WSRC: See Westinghouse Savannah River Co.

WVDP: See West Valley Demonstration Project.

APPENDIX:

MODULAR VAULT DRY STORAGE TECHNOLOGY

APPENDIX:

MODULAR VAULT DRY STORAGE TECHNOLOGY

A.1 DESCRIPTION

The vitrified product from high-level waste (HLW) treatment will be placed in interim storage awaiting final disposal at a geologic repository. It is assumed that the interim canister storage facility at Hanford and at the Savannah River site (SRS) for the vitrified HLW would hold canisters in vertically sealed cavities within a concrete structure forming the storage vault. Each Canister Interim Storage Vault (CISV) will be an air-cooled, dry storage vault for vitrified HLW.

The storage vault is designed on the basis of the modular concept; that is, the storage vault is divided into subblocks to help isolate the various compartments in case of an emergency or accident. The main floor of the vault building supports an array of storage tubes that extend into a vault below the main floor and into a support plate mounted above the floor of the vault. Concrete plugs provide a cover for the tubes. The storage tubes are cooled (to dissipate the decay heat of the HLW canisters) by air that is drawn by natural convection into distributing ducts on the outside wall of the building, through the vault and around the storage tubes, and then up a stack. The thermosiphon ventilation system is self-regulating and can compensate for changes in heat load or weather conditions.

Activities at a given CISV include receipt and unloading of transportation casks containing canisters of vitrified waste, inspection of the canisters, and storage of the waste until transfer to a geologic repository. The canisters are to be stored in sealed sleeves so that the cooling air does not come directly into contact with the potentially contaminated surfaces of the HLW canisters. The air (or cover gas) in the interior of the storage tubes and air flow through the vault are continuously monitored to detect any leakage of radioactive material from the canisters; if a leak were detected, the control system would shut down all operating roof fans and the contaminated air would then be exhausted through a ventilation system equipped with high efficiency particulate air (HEPA) filtration.

The above interim storage technology is known as the Modular Vault Dry Storage (MVDS) technology. Example facilities are the Irradiated Fuel Storage Facility (IFSF) at Idaho National Engineering Laboratory (INEL) and the Independent Spent Fuel Storage Installation (ISFSI) at the Fort St. Vrain power station in Colorado. These facilities are similar to the vitrified HLW storage vault for the Defense Waste Processing Facility (DWPF) at SRS.

The following operations description is adapted from DOE (1989). The HLW glass canisters are transferred to the modular dry-storage system in a transportation cask using a shielded canister transporter (SCT). At the vitrification building, a concrete plug is lifted from a load-out hatch and a hoist and grapple arrangement is used to place the HLW canister into the shielded

transportation cask mounted on the SCT. The SCT is driven to the CISV where the SCT is positioned over a loading port in the floor. The shield plug is removed from the loading port by the hoist and grapple arrangement in the SCT, the HLW canister is lowered into the storage tube, and the shield plug is reinstalled. The storage tubes are sealed after loading, providing a redundant barrier between any leakage from the sealed canisters and the environment. The canister tube can be purged of air and filled with inert cover gas, if necessary (primarily to avoid corrosion and dry oxidation). The HLW canister can be moved from dry storage back to the SCT by reversing the sequence.

A parametric approach was used to determine correlations between the various dependent variables (e.g., capital cost and annual work force) and the performance parameters (e.g., canister storage capacity and annual loading rate); the primary performance parameters were determined from U.S. Department of Energy (DOE) (1986a) and Bonnet and Gallagher (1993).

A.2 RESOURCE CHARACTERISTICS FOR CONSTRUCTION OF AN MVDS FACILITY

Resource characteristics for construction (including variables such as the total capital cost, facility size, construction materials, work force, and schedule) are given in Table A.1; the following paragraphs, along with Tables A.2 through A.5, discuss and elaborate on the information in Table A.1. Table A.1 is designed to allow evaluation of the various HLW interim canister storage alternatives, given the total required storage capacity and the annual canister loading rate.

A.2.1 Anchor Facilities

The IFSF at INEL and the independent spent fuel storage installation (ISFSI) at the Fort St. Vrain power station in Colorado are anchor facilities. These facilities are similar to the vitrified HLW storage vault for the DWPF at SRS.

A.2.2 Occupied Land Area

The modular structure of the MVDS technology allows considerable flexibility of expansion as capacity requirements increase. The facility would be contained within a secured, fenced area to minimize intrusion. It is assumed in this study that a HLW canister is equivalent to 0.5 metric tons of heavy metal (MTHM) of spent nuclear fuel (SNF). The occupied land area was estimated using the literature results shown in Table A.2 for the facility area.

TABLE A.1 MVDS Resource Characteristics: Construction[a]

Parameter	Quantity	Comments or Assumptions	Reference(s)
		Land-Use (ha)	
Occupied (ha)	0.68 × [Capacity (HLW canister)]$^{0.54}$	ANL correlation (Section A.2.2)	See Table A.2.
Disturbed	1.8 × (Occupied Area)	Based on data supplied by PSCo for FSV facility.	PSCo (1990b)
		Facility Size	
Number of floors	1	Storage at grade level	NA
Size (m^2)	68 × [Capacity (HLW canister)]$^{0.54}$	ANL correlation (Section A.2.2)	See Table A.2.
		Capital Cost Data	
Total capital cost ($ million)	0.708 × [Capacity (HLW canister)]$^{0.53}$	ANL correlation (Section A.2.3)	See Table A.3.
		Construction Materials and Requirements	
Water (m^3)	117 × [Disturbed area (ha)]	Based on 250,000 gal per 8 ha disturbed land. Water assumed to be used only for dust control because concrete is batched off-site and trucked to construction site.	PSCo (1990a)
Concrete (m^3)	110 m^3 × [Capacity (HLW canister)]	Based on 3,670 m^3 concrete for the FSV ISFSI.	Nash (1993)
Steel (t)	≈ 18 t × [Capacity (HLW canister)]	Based on 618 t steel for the FSV ISFSI.	Nash (1993)
Electricity (MW-yr)	Negligible	Assumed to be negligible.	NA

TABLE A.1 (Cont.)

Parameter	Quantity	Comments or Assumptions	Reference(s)
		Construction Work force and Schedule	
Total effort (worker-yr)	7.1 + 0.019 × [Capacity (HLW canister)]	ANL correlation (Section A.2.4)	NA
Peak (persons)	7.8 peak workers / [Total effort (worker-yr)]	Based on the FSV ISFSI (peak of 60 workers for a total effort of ≈ 7.7 worker-years).	Based on PSCo (1990b) and personal communication with FSV staff.
Construction duration (months)	$1.36 \times [\text{Capacity (HLW canister)}]^{0.47}$	ANL correlation (Section A.2.5)	See Table A.5.
		Construction Impacts	
Hazardous (m^3)	Negligible	No hazardous chemicals to be used. A Chemical Control Program will govern the potential or actual use of any hazardous chemicals that could be used.	PSCo (1990a)
Nonhazardous liquids	0.0	Concrete for MVDS will be premixed, thus no water use or wastes from concrete batch operations will result.	PSCo (1990a)
Sewage	0.0	Assumed to be sanitary waste, chemical toilets will be provided with the wastes from toilets disposed off-site.	PSCo (1990a)
		Construction Emissions (t) (Section A.2.6)	
No_x (as NO_2)	$3 \times 10^{-3} \times$ [Capacity (HLW canister)	Based on MRS design.	Silviera et al. (1985)
CO	$6 \times 10^{-4} \times$ [Capacity (HLW canister)]	Based on MRS design.	Silviera et al. (1985)
Hydrocarbons	$2 \times 10^{-4} \times$ [Capacity (HLW canister)]	Based on MRS design.	Silviera et al. (1985)
Particulates	$2 \times 10^{-4} \times$ [Capacity (HLW canister)]	Based on MRS design.	Silviera et al. (1985)

TABLE A.1 (Cont.)

Parameter	Quantity	Comments or Assumptions	Reference(s)
SO_x (as SO_2)	$2 \times 10^{-4} \times$ [Capacity (HLW canister)]	Based on MRS design.	Silviera et al. (1985)
Fugitive dust (TSP)	$\approx 1.3 \times 10^{-4} \times$ [Concrete (m^3)]	Based on dust emissions of 0.2 lb/yd^3 concrete for concrete batching.	DOE (1986a)

[a] Abbreviations: ANL = Argonne National Laboratory, FSV = Fort St. Vrain, NA = not applicable, NRC = U.S. Nuclear Regulatory Commission, PSCo = Public Service Company of Colorado, TSP = total suspended particles/particulates, MRS = monitored retrievable storage.

TABLE A.2 Literature Results for MVDS Facility Land Area[a]

Facility	Capacity (HLW canisters)	Floor Area (m^2)	Reference(s)
Generic	500[b]	2,000	Baillif et al. (1986)
MRS MVDS facility	14,500[b]	13,725	Bosch and Carter (1991)
Japanese HLW storage	2,500	≈ 4,000	Jardine et al. (1989)
Generic	34.4[b]	410	Feizollahi and Shropshire (1993)
Generic	280[b]	1,650	Feizollahi and Shropshire (1993)
Proposed Wylfa facility	700[b]	≈ 2,000	IAEA (1988)
FSV ISFSI	33.4[b]	515	PSCo (1990b)
Proposed MRS	4,620[b]	5,591	Cundill et al. (1988)
SRS canister storage facility	2,565	4,340	Plodinec (1989)

[a] Abbreviations: IAEA = International Atomic Energy Agency, MRS = monitored retrievable storage.

[b] Based on 0.5 MTHM equivalent per HLW canister.

The facility designs in Table A.2 do not allow stacking of HLW canisters within the sealed sleeves of the MVDS design because of the expected high decay heat of the Savannah River HLW high-heat canisters. The facility size can be correlated by the following equation:

$$[\textit{Facility size}\ (m^2)]_{MVDS} = 70.8 \times [\textit{Capacity}\ (\text{HLW canister})]^{0.53}, \qquad (A.1)$$

with a correlation coefficient of approximately 99%. Figure A.1 compares the above correlation with the data.

A.2.3 Capital Cost

The cost of the MVDS technology has been stated to be relatively insensitive to the type of material stored (Fletcher and Smith 1989). For example, the capital cost differential for a MVDS containing spent fuel versus HLW canisters is approximately 7% (Triplett and Smith 1984). Given the similarity in facility designs, it was assumed in this study that cost data developed for spent fuel storage would be applicable for HLW interim storage.

Table A.3 gives the capital cost data available in the open literature. The capital cost of HLW interim storage of vitrified double-shell tank (DST) waste at Hanford is included in the overall cost for the Hanford Waste Vitrification Plant (HWVP). The capital cost of an Interim Storage Facility (ISF) for vitrified single-shell tank (SST) wastes at Hanford (2,004 canisters) has, however,

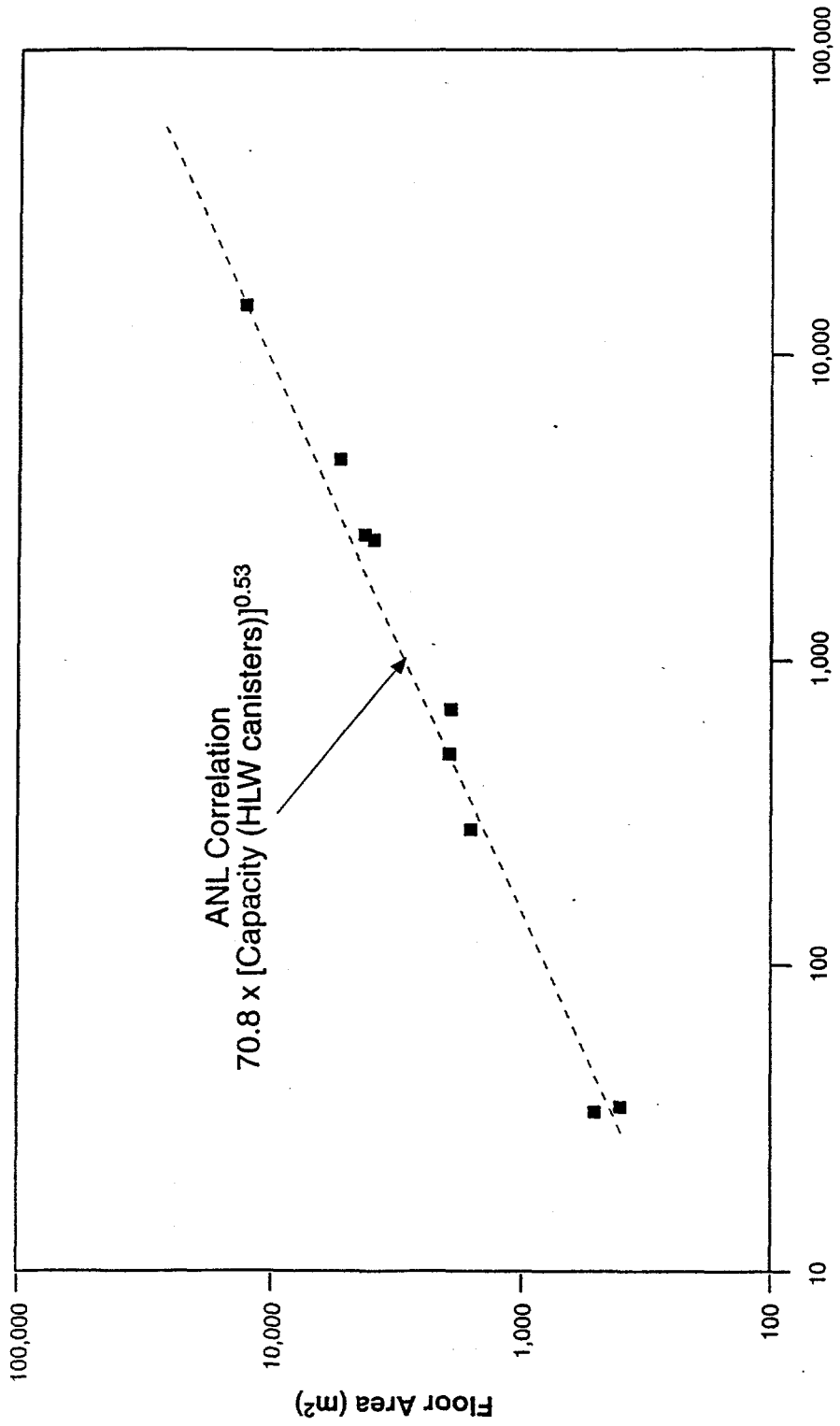

FIGURE A.1 Floor Area as a Function of Capacity: MVDS Interim Storage

TABLE A.3 Capital Cost Summary for SNF/HLW Interim Storage

Facility	Capacity (HLW canisters)	Reported Capital Cost ($ million)	Dollar Year	1992 Capital Cost ($ million)	Reference(s)
Generic	34.4[a]	5.2	1992	5	Feizollahi and Shropshire (1993)
Generic	280[a]	20	1992	18	Feizollahi and Shropshire (1993)
MRS	30,000[a]	156.562	mid-1988	173	Fletcher and Smith (1989)
MRS	200[a]	10.1	1987	10	Cundill et al. (1988)
MRS	1,000[a]	25.6	1987	26	Cundill et al. (1988)
MRS	2,000[a]	45	1987	46	Cundill et al. (1988)
SRS HLW	2,286	40	1992	40	Street et al. (1992)
Hanford HLW	2,004	35	1991	36	Boomer (1992)

[a] Based on 0.5 MTHM equivalent per HLW canister.

been reported to be $35 million (Boomer 1992); each HLW canister is assumed to contain 0.54 MTHM on the basis of a waste loading of 32%. A cost of $40 million has been projected for storage of 2,286 canisters at SRS (Street et al. 1992). The capital costs for the different SNF ISFs were decreased by 7% to allow equal comparison between facilities storing HLW and those storing SNF.

The cost values for the MRS were modified to subtract the cost components associated with site preparation (site-specific) and facility licensing. In addition, the capital costs quoted in Feizollahi and Shropshire (1993) were decreased by one magnitude after discussions with the authors. Correlation of the literature values with capacity yields the following equation:

$$[\textit{Capital cost}\ (\$\ \text{million})]_{MVDS} = 0.71\ [\textit{Capacity}\ (\text{HLW canister})]^{0.53}\ , \qquad \text{(A.2)}$$

with a (rather low) correlation coefficient of 98.5%. Figure A.2 compares the above correlation with the data summarized in Table A.3.

The decontamination and decommissioning (D&D) cost ranges from 8.9% of the construction cost (Woods et al. 1988) to approximately 10% (Fletcher and Smith 1989). The upper value of 10% will be used for conservatism. This approach assumes that a larger capital cost will result in a larger expected cost for decommissioning, which may not be true in all circumstances.

A.2.4 Total Construction Effort

The total construction labor was estimated by using available literature sources. The correlation of the construction site labor with storage capacity was estimated from the results of Cundill et al. (1988) for alternative MRS designs using the MVDS technology. Table A.4 presents the estimates for alternative MVDS designs.

The above results indicate that construction site labor is linearly proportional to the storage capacity.

$$[\textit{Construction effort}\ (\text{worker-year})]_{MVDS} = 7.1 + 0.019 \times [\textit{Capacity}\ (\text{HLW canister})]. \qquad \text{(A.3)}$$

Application of the above correlation to the reported construction effort for the Fort St. Vrain ISFSI ($\approx$ 7.7 person-yr for a 16.7 MTHM storage capacity) results in a difference of less than 1%.

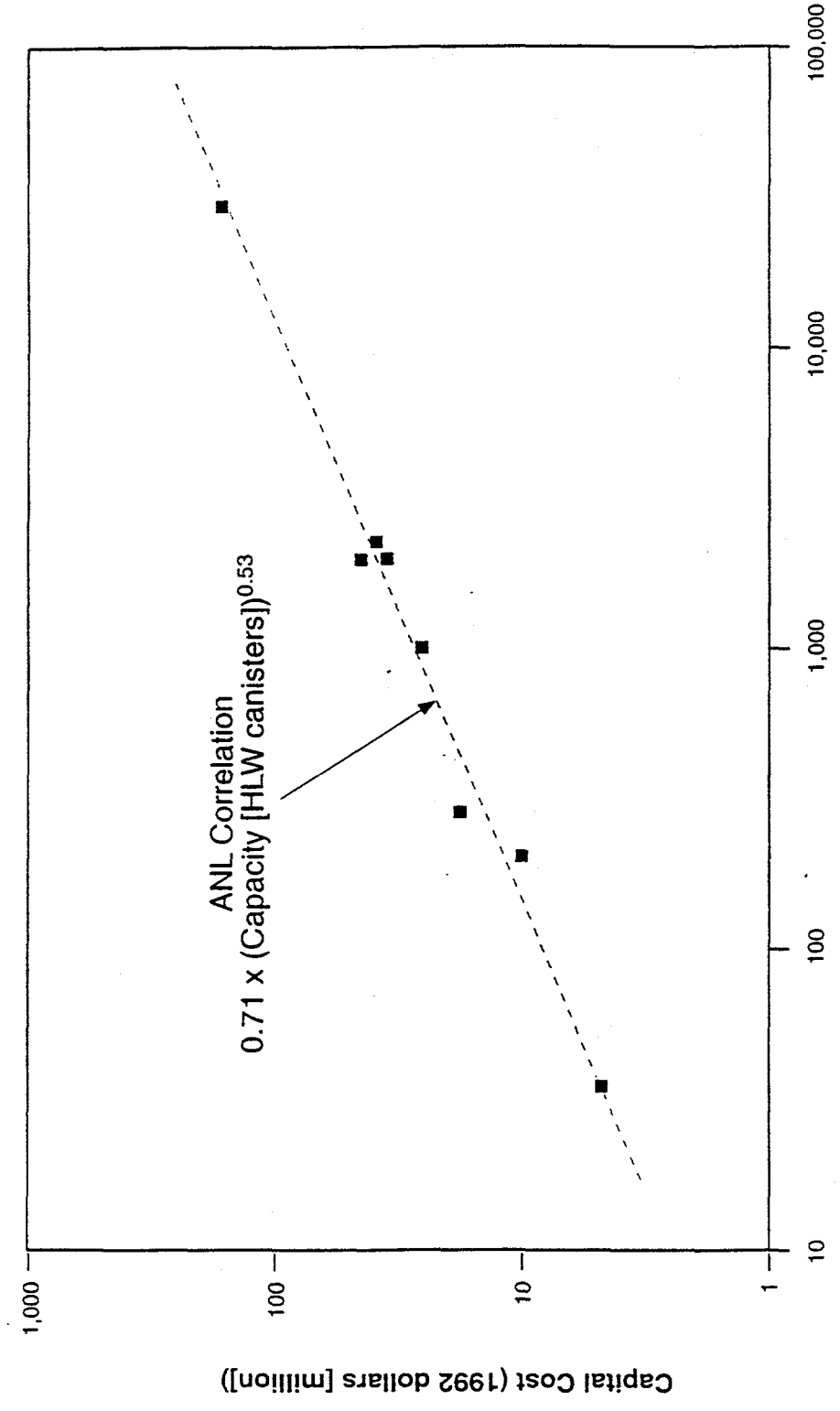

FIGURE A.2 Capital Cost as a Function of Capacity: MVDS Interim Storage

TABLE A.4 Construction Labor as a Function of MVDS Facility Storage Capacity

	Storage Capacity (HLW canister)[a]		
	200	1,000	2,000
Site labor cost ($ 1987)	604	1,441	2,500
Site labor (worker-yr)[b]	11	26.2	45.5

[a] Based on 0.5 MTHM equivalent per HLW canister.

[b] Based on an annual construction labor cost of $55,000 per construction worker.

A.2.5 Construction Duration

In general, the duration of construction depends on the level of site preparation, availability of construction materials and specialized construction labor, and facility licensing/environmental studies, among others. The construction duration for the MVDS technology is assumed to be primarily dependent on the storage capacity. Literature results for construction duration are shown in Table A.5. Data regression resulted in the following equation:

$$[\textit{Construction duration}\ (\text{month})]_{MVDS} = 1.36 \times [\textit{Capacity}\ (\text{HLW canister})]^{0.47}, \qquad \text{(A.4)}$$

with a correlation coefficient of 98.2%.

TABLE A.5 Literature Results for MVDS Facility Construction Duration

Facility	Capacity (HLW canisters)	Construction Duration (month)	Reference(s)
FSV ISFSI	33.4[a]	7 to 8	PSCo (1990b)
Proposed Wylfa facility	700[a]	24 to 30	IAEA (1988)
Proposed MRS	4,620[a]	84	Fletcher and Smith (1989)
PAMELA facility	1,500	40[b]	Demonie et al. (1987)

[a] Based on 0.5 MTHM equivalent per HLW canister.

[b] Includes 14 months for licensing.

A.2.6 Construction Emissions

Air pollutant construction emissions include dust generated during construction and construction equipment exhaust emissions. Source terms for equipment exhausts can, in general, be determined by first estimating the amount of construction equipment and thereby the quantity of diesel fuel. These estimates are usually based on review and comparison of equipment requirements for construction projects of similar magnitude. The particulate emissions from dust generated are calculated from the land area used for construction, the duration of the construction period, and the appropriate (site-specific) emission factor.

Estimated emissions during construction of a MRS storage facility (Silviera et al. 1985) were used for the MVDS and placed on a unit HLW canister basis (based on a maximum throughput of 3,600 canisters/yr for the MRS and assuming a three-year construction period).

Fugitive dust consists of particles with a diameter equal to or less than 30 mm. Fugitive dust emissions were estimated on the basis of dust emissions of 0.2 lb/yd^3 concrete for concrete batching (DOE 1986a).

A.3 RESOURCE CHARACTERISTICS FOR OPERATION OF AN MVDS FACILITY

Resource characteristics for operations (including variables such as annual operating charges, chemical requirements, work force, and annual waste generation) are shown in Table A.6; the following paragraphs, along with Tables A.7 through A.10, explain and elaborate on the information in Table A.6. Table A.6 is designed to allow evaluation of the various HLW interim canister storage alternatives, given the total required storage capacity and the annual canister loading rate.

A.3.1 Anchor Facilities

The IFSF at INEL and the ISFSI at the Fort St Vrain power station in Colorado are anchor facilities. These facilities are similar to the vitrified HLW storage vault for the DWPF at Savannah River.

A.3.2 Annual Operation and Maintenance Cost

Annual operation and maintenance (O&M) costs include the routine handling, storage, and retrieval of stored spent fuel. Costs during the storage of fuel are primarily associated with the O&M of the facility; most of these costs are for personnel. Additional fuel transfer costs are incurred for

TABLE A.6 MVDS Resource Characteristics: Operations[a]

Parameter	Quantity	Comments or Assumptions	Reference(s)
	Annual Operating Charges		
Annual O&M cost ($1,000/yr) loading/unloading	770 × [Throughput (HLW canister/yr)]$^{0.45}$	ANL correlation (Section A.3.2)	See Table A.7.
Annual O&M cost ($1,000/yr) storage only	38.6 + 0.27 × [Capacity (HLW canister)]	ANL correlation (Section A.3.2)	See Table A.7.
	Annual Resource Requirements during Operation (Section A.3.3)		
Water (m^3/yr) loading/unloading	14 × [Throughput (HLW canister/yr)]	Based on MRS design.	Smith (1983)
Annual electrical demand (GWh/yr) loading/unloading	0.031 × [Throughput (HLW canister/yr)]$^{0.5}$	ANL correlation	NA
Water (m^3/yr) storage only	1.1×10^{-3} × [Capacity (HLW canister)]	Assumes majority of water used for cask washdown.	PSCo (1990b)
Annual electrical demand (GWh/yr) storage only	Negligible	The MVDS system is totally passive, minimal power is required for monitoring devices and lighting.	PSCo (1990b)
	Hazardous Material Inventory (kg)		
Mass (kg_{HAZ}/kg_{HLW})	None	Assumes decontamination solution would be nontoxic (freon not used).	PSCo (1990b)
	Annual Work Force		
Number of workers (FTEs) loading/unloading	3.7 × [Throughput (canister/yr)]$^{0.4}$	ANL correlation (Section A.3.4)	See Table A.9.

TABLE A.6 (Cont.)

Parameter	Quantity	Comments or Assumptions	Reference(s)
Number of workers (FTEs) storage only	0.55 × [Capacity (HLW canister)]$^{0.36}$	ANL correlation (Section A.3.4)	See Table A.9.
Support (FTEs)	2 × (no. of workers)	ANL assumption	NA
	Annual Wastes (m^3/yr)		
HLW	0.0	No SNF reprocessing, etc., thus no HLW generated.	NA
CH-TRUW	0.0	None generated.	NA
RH-TRUW	0.0	None generated.	NA
CH-LLW: loading (Section A.3.5)	Liquid: 0.063 m^3/HLW canister	Average activity: 70 Ci/m^3	Adapted from PNL (1982)
	Compactible, combustible: 2 m^3/HL canister	Average activity: 0.02 Ci/m^3	NA
	Noncombustible: 0.22 m^3/HLW canister	Average activity: 0.02 Ci/m^3	NA
CH-LLW: storage (Section A.3.6)	Liquid: 0.0046 m^3/yr/HLW canister Solid: 0.084 m^3/yr/HLW canister	Average activity: 8.6E-02 Ci/m^3	Adapted from PNL (1982)
RH-LLW (Section A.3.7)	Assumed to be 0.0.	ANL assumption	NA
Mixed LLW	0.0	No hazardous chemicals used during operation, thus no possible contamination.	NA
Hazardous	0.0	No hazardous chemicals used during operation, thus no possible contamination.	NA
Sewage (Section A.3.8)	38 m^3/person-yr	Liquid rate is 33 m^3/person-yr and solid is 5 m^3/person-yr, for a total of 38 m^3/person-yr.	EG&G Idaho (1991)

TABLE A.6 (Cont.)

Parameter	Quantity	Comments or Assumptions	Reference(s)
	Air Emission Point		
Height (m)	To be modeled as a ground-level release.	(Section A.3.9)	NA
Flowrate (m^3/hr)	To be modeled as a ground-level release.		NA
Cross section (m^2)	To be modeled as a ground-level release.		NA
	Radioactive Annual Emissions (Ci/yr)		
Air emissions — activity as function of input (Ci_{out}/Ci_{in})	Assumed to be negligible.	(Section A.3.10)	NA
	Nonradioactive Annual Emissions (kg/yr)		
NO_x	Assumed to be negligible.	(Section A.3.11)	
SO_x	Assumed to be negligible.		EG&G Idaho (1991)
VOC	Assumed to be negligible.		EG&G Idaho (1991)

[a] Abbreviations: FTE = full-time equivalent, CH-LLW = contact-handled low-level waste, RH-LLW = low-level remote-handled waste, NO_x = nitrogen oxide, SO_x = sulfur oxide, PNL = Pacific Northwest Laboratory, CH-TRUW = contact-handled transuranic waste, RH-TRUW = remote-handled transuranic waste, and VOC = volatile organic compound.

TABLE A.7 Literature Results for MVDS Facility Operation and Maintenance

Facility	Capacity (HLW Canister)[a]	Annual O&M Cost during Storage ($1,000/yr) Reported	Dollar Year	Dollars (1992)	Reference(s)
Generic	34.4	58.8	1992	58.8	Feizollahi and Shropshire (1993)
Generic	280	121.8	1992	122	Feizollahi and Shropshire (1993)
MRS	200	67	1987	84	Cundill et al. (1988)
Generic	1,000	250	1988	297	Johnson (1990)
MRS	1,000	247	1987	308	Johnson (1990)
MRS	2,000	473	1987	592	Johnson (1990)

[a] Based on 0.5 MTHM equivalent per HLW canister.

TABLE A.8 Literature Results for MVDS Facility Loading/Unloading

Facility	Throughput (HLW canister/yr)[a]	Annual O&M Cost during Unloading/Loading ($1,000/yr) Reported	Dollar Year	Dollars (1992)	Reference(s)
Generic	10	2,316	1992	2,316	Feizollahi and Shropshire (1993)
Generic	100	5,431	1992	5,431	Feizollahi and Shropshire (1993)
MRS	5,400	37,100	1988	44,000	Woods et al. (1988)

[a] Based on 0.5 MTHM equivalent per HLW canister.

TABLE A.9 Literature Results for MVDS Facility Work Force

Interim Storage			Loading/Unloading		
Capacity (Canister)[a]	Work Force (FTEs)	Reference	Throughput (Canister/yr)[a]	Work Force (FTEs)	Reference
17.2	2	Feizollahi and Shropshire (1993)	5	10	Feizollahi and Shropshire (1993)
140	4	Feizollahi and Shropshire (1993)	50	24	Feizollahi and Shropshire (1993)
500	5	Cundill et al. (1988)	125	30	Jardine et al. (1989)
750	8	Baillif and Guay (1990)	1,800	105	Smith (1983)
1,000	9.5	Cundill et al. (1988)	3,000	128	Woods et al. (1988)

[a] Based on 0.5 MTHM equivalent per HLW canister.

TABLE A.10 Annual Waste Generation during MVDS Facility Loading/Unloading

LLW Treatability Category	Annual Generation Rate: Volumetric Basis (m^3/yr)	Annual Generation Rate: HLW Canister Basis[a] (m^3/HLW canister)	Average Activity (Ci/m^3)
Liquid (aqueous)	38	0.063	74
Compactible, combustible	1,168	2.0	0.021
Noncompactible, noncombustible	133	0.22	0.015

[a] Based on an annual loading rate of 600 HLW canisters per year.

conveying the fuel at the beginning and end of the storage period. In general, operating costs are dependent on the type of storage technology and the size or capacity of the storing facility. Cundill et al. (1988) state that the annual O&M cost during storage is a constant percentage (i.e., 1.4%) of the capital cost. Table A.7 gives the available annual O&M cost data for the storage phase.

The correlation of the annual operating costs for the storage period as a function of capacity is

$$[\textit{Annual O\&M cost}\ (\$1{,}000/\text{yr})]_{storage} = 38.6 + 0.27 \times [\textit{Capacity}\ (\text{HLW canister})]. \quad \text{(A.5)}$$

The correlation coefficient is very good (99.7%); however, this correlation should be very carefully applied to near-zero storage capacities because the above correlation does not approach a zero cost as the number of HLW canisters approaches zero. Very few cost data are available for the loading/unloading period (Table A.8).

Correlation of the cost values in Table A.8 yields

$$[\textit{Annual O\&M cost}\ (\$1{,}000/\text{yr})]_{loading} = 770 \times [\textit{Throughput}\ (\text{HLW canister/yr})]^{0.45}. \quad \text{(A.6)}$$

The above correlation has (unfortunately) only one degree of freedom.

A.3.3 Resource Requirements during Operation

During the loading/unloading phase, a total of 13 million gal (≈49,000 m^3) of water per year was estimated to be required for a maximum annual throughput of 1,800 MTHM/yr (Smith 1983). This is equivalent to approximately 14 m^3 (494 ft^3) per HLW canister.

The electrical requirement during the loading/unloading phase was determined by correlating the annual electrical demand for interim storage of SNF in a modular vault dry storage facility (Feizollahi and Shropshire 1993), assuming an average cost of electricity of 3.5 cents per kWh.

$$[\textit{Annual electrical demand}\ (\text{GWh/yr})]_{MVDS} = 0.031 \times [\textit{Throughput}\ (\text{HLW canister/yr})]^{0.5}. \quad \text{(A.7)}$$

The above correlation is expected to be accurate within the range of throughputs expected. However, it underestimates the electrical demand for a 1,800 MTHM/yr MRS design (9 GWh/yr) (Smith 1983).

The safety report for the Fort St. Vrain ISFSI estimated that approximately 10 gal/yr (0.038 m^3/yr) of liquid waste would be generated annually during the storage-only phase. Assuming the majority of liquid low-level waste (LLW) will result from water used for washdown of the shipping cask and facility decontamination, the annual water requirement can be given by

$$[\textit{Water}\ (m^3/\text{yr})]_{Storage} = (10\ \text{gal/yr}) \times (m^3/264.17\ \text{gal}) \times (0.5\ \text{MTHM/HLW canister})/(16.7\ \text{MTHM}) = 1.1 \times 10^{-3}\ m^3/\text{HLW canister}. \quad \text{(A.8)}$$

A.3.4 Estimated Work Force

The work force estimate includes only those personnel who spend an eight-hour day working in direct support of facility operations. The estimated work force includes analytical, health physics, maintenance, technical/production, and supervisory personnel. All workers are assumed to be exposed to radiation. The available literature was reviewed to establish the dependency of the MVDS facility work force on capacity/throughput (Table A.9).

The above values include maintenance and other direct support personnel. Assumptions made for the storage-only work force are an annual cost of $50,000 per FTE (Cundill et al. 1988) and a ratio of 1 to 3 operators-to-maintenance personnel for the PAMELA facility (Demonie et al. 1987). The correlation of the storage work force with capacity is

$$[\textit{Storage personnel}\ (\text{FTE})]_{MVDS} = 0.55 \times [\textit{Capacity}\ (\text{HLW canister})]^{0.36}, \quad \text{(A.9)}$$

with a (relatively low) correlation coefficient of 94%.

The value in Table A.9 for the storage work force from Smith (1983) did not include personnel for dry rod consolidation, on-site waste treatment, and security and other service functions; these were assumed to be supplied by the DOE site. The correlation of work force with throughput during loading/unloading can be given by

$$[\textit{Loading/unloading personnel}\ (\text{FTE})]_{MVDS} = 3.7 \times [\textit{Throughput}\ (\text{HLW canister/yr})]^{0.4}, \tag{A.10}$$

with a correlation coefficient greater than 99%.

The distribution of the work force is assumed to be approximately 50% operators, 30% maintenance, and 20% health physics (Deacon 1981).

A.3.5 CH-LLW: Loading Phase

Low-level waste in liquid and solid form would be generated during the loading phase. Wet waste streams would result from cask/vehicle washdowns, cask-flushes and cooling, and various decontamination operations performed at the MVDS facility. The majority of the liquid LLW produced during the loading phase is due to decontamination/washing of the incoming/exiting shipping casks. It has been estimated that about 4 gal (0.015 m^3) of decontamination solution would be required during wiping and rinsing of a single HLW shipping cask (Ralph M. Parsons Co. 1985). Each shipping cask is assumed to contain one HLW canister (i.e., shipment by truck).

Radioactive solid wastes would generally either be (1) dry active waste (compactible, combustible) or (2) noncompactible, noncombustible. Dry active wastes such as protective clothing, plastic sheeting, gloves, materials suspected of being contaminated, etc., are generated as a result of facility operations. Noncompactible, noncombustible LLW would consist of spent ventilation exhaust prefilters and HEPA filters and failed equipment. Solid waste would be packaged into drums and sent to other facilities for treatment and/or disposal.

In this analysis, the generation rate is assumed to be linearly proportional to the loading rate. Table A.10 gives the annual generation rate during the loading phase on the basis of a throughput of 600 HLW canisters per year (PNL 1982).

A.3.6 CH-LLW: Storage Phase

The CH-LLW wastes produced during the storage phase can be divided into solid (plastic, cloth, paper, etc.) and liquid (decontamination solutions, etc.) streams. The safety report for the Fort St. Vrain ISFSI estimated that approximately 10 gal/yr (0.038 m^3/yr) of liquid and about 25 ft^3/yr (0.71 m^3/yr) of solid waste would be produced during routine fuel storage. Assuming these values

depend on the amount in storage, then approximately 0.0046 m^3 liquid waste/HLW canister and 0.084 m^3 solid waste/HLW canister would be generated during interim storage. (The assumption that the waste generation is dependent on the storage capacity assumes that the majority of the waste is produced during routine equipment/facility activities such as maintenance, etc.).

It has been reported that the "annual waste volumes from a MVDS are the same as for a cask storage facility" (PNL 1982). Approximately 180 m^3 (6,355 ft^3) of solid wastes and 5 m^3 (176 ft^3) of wet wastes are expected to be generated annually, primarily from failed equipment and rare package failures. About 16 Ci of radioactive material are contained in these wastes annually. The total capacity of the cask storage facility is 3,000 MTHM. On the basis of a 0.5 MTHM equivalent per HLW canister, the annual generation rate during storage of liquid and solid LLW would be 0.0083 (0.29 ft^3) and 0.03 m^3 (1.1 ft^3) per HLW canister stored, respectively. The values estimated in PSCo. (1990b) were used in this study for conservatism.

The LLW activity during storage is estimated on the basis of a total of 16 Ci of radioactive material being produced out of a total of 185 m^3 (6,531 ft^3) generated annually from a 3,000-MTHM MVDS. The average activity of both streams is approximately 0.086 Ci/m^3.

A.3.7 RH-LLW

Remote-handled low-level waste generally includes spent HEPA filters and failed equipment. Available data indicate that the expected radioactivity of the spent HEPA filters is low, primarily because of the low contamination of the shipping casks and the low failure rate of the HLW canister packages. For these reasons, it was assumed that the amount of LLW-RH would be negligible.

A.3.8 Sewage

The sanitary wastes will be processed through sanitary waste treatment to produce a clarified liquid effluent. The amount of liquid sanitary waste (35 gal/d/person/shift) is based on EG&G Idaho (1991) and can be estimated by the following equation:

$$[\textit{Liquid sanitary waste}\ (m^3/yr)] = (35\ \text{gal/d/person/shift}) \times (168\ \text{working days/yr}) \times [\textit{Number of personnel}\ (\text{FTEs})] \times (3\ \text{shifts}) \times (3.785 \times 10^{-3}\ m^3/\text{gal}). \quad \text{(A.11)}$$

It is assumed that this conceptual storage facility would operate 24 hours per day, 5 days per week, 240 days per year, with an on-stream efficiency of 70%. The number of facility personnel is determined by correlation.

The amount of solid sanitary waste (5 m^3/person/yr) is based on EG&G Idaho (1991) and can be estimated by the following equation:

$$[\textit{Solid sanitary waste}\ (m^3/yr)] = (5\ m^3/person/yr) \times (FTEs). \quad (A.12)$$

The number of facility personnel is determined by correlation.

A.3.9 Stack Characteristics

Justifiable stack locations and heights are not possible because the design is preliminary and they cannot be defined until a more detailed conceptual design has been initiated and preliminary safety analysis completed. Stack characteristics will be affected by the location of adjacent facilities, local weather, and site conditions. These effects cannot be included at this time.

The stack height of a "generic" MVDS facility is given as 72 ft (≈22 m) (Cundill et al. 1988). The storage facility stack for the IFSF at INEL is located 65 ft (≈20 m) above ground (Stirrup 1993). Following the methodology employed by INEL, it may be assumed that the emissions from the MVDS may be modeled as a ground-level release because the release point is not greater than 2.5 times the associated building height (DOE 1993).

A.3.10 Radioactive Annual Emissions

The air circulation system of the MVDS is designed so that the cooling air does not come in contact with the HLW canisters and, therefore, should not become contaminated. Similarly, the concrete inner surfaces of the MVDS facility are not expected to be exposed to contaminated air and should remain clean throughout the lifetime of the facility (Block and Nash 1993). In addition, the cooling air for the vault is passed through prefilters and HEPA filtration prior to discharge for removal of essentially all particulates.

Annual radiological emissions of 4.8×10^{-6} Ci/yr were reported for the IFSF at the Idaho Chemical Processing Plant (ICPP) for 1992 (DOE 1993); much lower emission rates would be expected for HLW interim storage, because of the lower frequency of canister failures resulting in atmospheric releases. Under normal operating conditions, any releases of radionuclides would be limited to those from canister surface contamination. The surface contamination level of canisters produced from DST waste at Hanford are expected to be less than 370 Bq/m^2 for alpha contaminants and less than 3,700 Bq/m^2 for beta-gamma contaminants (Gurley and Minor 1985; Knecht and Berreth 1989). The amount of alpha contaminants adhering to a single HLW canister is approximately 5×10^{-8} Ci; the amount of beta-gamma contaminants is 5×10^{-7} Ci. On the basis of this data, radiological atmospheric emissions are assumed to be negligible.

A.3.11 Nonradioactive Annual Emissions

The standby emergency electrical generators have been identified as potential air pollutant emitters. Operation is assumed to be so infrequent as to be considered insignificant (PSCo 1990b). In agreement, negligible emissions have been reported for the 115-kW propane-fueled emergency generator available at the IFSF (DOE 1993).

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