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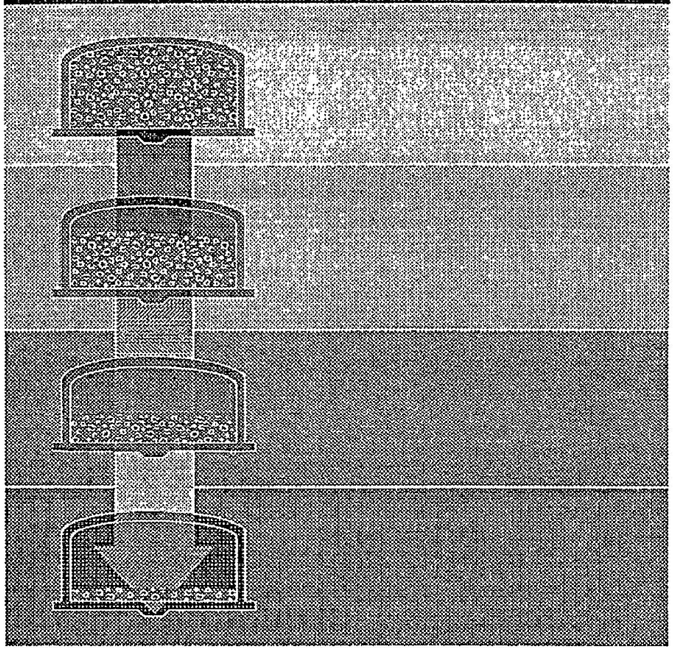
Tank Waste Remediation System, Hanford Site, Richland, Washington, Final Environmental Impact Statement

Volume Two



Appendix A: TWRS EIS Waste Inventory Data
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ACRONYMS AND ABBREVIATIONS

DST	double-shell tank
EIS	Environmental Impact Statement
LANL	Los Alamos National Laboratory
MUST	miscellaneous underground storage tank
SST	single-shell tank
TRAC	Track Radioactive Component
TWRS	Tank Waste Remediation System
VOC	volatile organic compounds

NAMES AND SYMBOLS FOR UNITS OF MEASURE, RADIOACTIVITY, AND ELECTRICITY/ENERGY

Length

cm	centimeter
ft	foot
in	inch
km	kilometer
m	meter
mi	mile

Area

ac	acre
ft^2	square foot
ha	hectare
km^2	square kilometer
mi^2	square mile

Volume

cm^3	cubic centimeter
ft^3	cubic foot
gal	gallon
L	liter
m^3	cubic meter
ppb	parts per billion
ppm	parts per million
yd^3	cubic yard

Mass

g	gram
kg	kilogram
lb	pound
mg	milligram
mt	metric ton

Radioactivity

Ci	curie
MCi	megacurie (1.0E+06 Ci)
mCi	millicurie (1.0E-03 Ci)
μCi	microcurie (1.0E-06 Ci)
nCi	nanocurie (1.0E-09 Ci)
pCi	picocurie (1.0E-12 Ci)

Electricity/Energy

A	ampere
J	joule
kV	kilovolt
kW	kilowatt
MeV	million electron volts
MW	megawatt
V	volt
W	watt

Temperature

°C	degrees Centigrade
°F	degrees Fahrenheit

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APPENDIX A
TWRS EIS WASTE INVENTORY DATA

A.1.0 INTRODUCTION

This appendix provides the inventory of waste addressed in this Environmental Impact Statement (EIS). The inventories consist of waste from the following four groups:

- Tank waste;
- Cesium (Cs) and strontium (Sr) capsules;
- Inactive miscellaneous underground storage tanks (MUSTs); and
- Anticipated future tank waste additions.

The major component by volume of the overall waste is the tank waste inventory (including future tank waste additions). This component accounts for more than 99 percent of the total waste volume and approximately 70 percent of the radiological activity of the four waste groups identified previously. Tank waste data are available on a tank-by-tank basis, but the accuracy of these data is suspect because they primarily are based on historical records of transfers between tanks rather than statistically based sampling and analyses programs. However, while the inventory of any specific tank may be suspect, the overall inventory for all of the tanks combined is considered more accurate. The tank waste inventory data are provided as the estimated overall chemical masses and radioactivity levels for the single-shell tanks (SSTs) and double-shell tanks (DSTs). The tank waste inventory data are broken down into tank groupings or source areas that were developed for analyzing groundwater impacts.

The waste inventory data in this appendix are from the following documents:

- Single-Shell and Double-Shell Tank Waste Inventory Data Package for the Tank Waste Remediation System Environmental Impact Statement (WHC 1995d);
- Disposition of Cesium and Strontium Capsules Engineering Data Package for the Tank Waste Remediation System Environmental Impact Statement (WHC 1995h); and
- Status Report on Inactive Miscellaneous Underground Storage Tanks (Rasmussen 1995).

A.2.0 WASTE INVENTORY DATA

A.2.1 TANK WASTE INVENTORY

The tank inventory data are presented in Tables A.2.1.1, A.2.1.2, and A.2.1.3. Table A.2.1.1 lists the current waste volumes stored in the SSTs, DSTs, and inactive MUSTs. Table A.2.1.2 lists the chemical constituents in the SSTs and DSTs, and Table A.2.1.3 lists the estimated radionuclide inventory for the SSTs and DSTs. The chemical inventory for the SSTs is categorized by waste types found in the tanks: sludge, saltcake, and liquid. The DST chemical inventory is presented as soluble and insoluble components. The soluble portion of the DST waste inventory was estimated using solubility factors, which were calculated using tank sampling and historical data. These solubility factors represent the amount of each component assumed to be soluble in water. The insoluble portion of the DST waste inventory is assumed to remain in a solid form during sludge washing operations.

Data showing the division of the constituents between the soluble and insoluble portion of the SST waste do not exist.

Based on estimates of tritium contained in the tank waste, the Effluent Treatment Facility is expected to process 242-A Evaporator condensate containing 2,360 curies (Ci) of tritium (reflecting decay to December 31, 1999) during its operational life. Processing the wastewater from the K and N Basins would add about 10 percent to this total (DOE 1994e).

A.2.1.1 Tank Aggregated Source Areas

The DSTs and SSTs represent 177 potential sources of contaminant release. These sources were grouped together into source areas (tank groupings) for groundwater modeling purposes. Each tank grouping contains between one and three tank farms. The tank farms were grouped together based on tank configuration, tank proximity, and groundwater flow direction. The inventory from the individual tank farms was then combined to create a waste inventory by source area (Pelton 1995). The SST and DST farms were maintained in separate source areas to support different release scenarios developed for the alternatives. Grouping the tank waste inventory together into source areas, based on tank configuration, geographic proximity, and groundwater flow direction, resulted in eight tank groupings, three in the 200 West Area (two SSTs and one DST) and five in the 200 East Area (three SSTs and two DSTs).

The tank farms were grouped into the source areas identified in Table A.2.1.4 and Figure A.2.1.1. The chemical species and estimated radionuclide inventory for the SST groups are shown in Tables A.2.1.5 and A.2.1.6. The chemical species and estimated radionuclide inventory for the DST groups are shown in Tables A.2.1.7 and A.2.1.8.

A.2.2 CESIUM AND STRONTIUM CAPSULE INVENTORY

The quantities, heat loading, and radioactivity levels for the Cs and Sr capsules are presented in Table A.2.2.1. The chemical form of the Cs in the capsules is cesium chloride (CsCl) and the chemical form of the Sr in the capsules is strontium fluoride (SrF_2). The combined total capsule volume is approximately 2 cubic meters (m^3) (70 cubic feet [ft^3]) (WHC 1995h).

The Cs content of the capsules is primarily Cs-137, which has a half-life of 30.17 years. Cesium-137 decays into the stable isotope barium-137. The Sr capsules contain mainly Sr-90, which has a half-life of 28.6 years. Strontium-90 decays to yttrium-90 and then to the stable isotope zirconium-90. The reduction in the number of curies, heat load, and concentration over time is due to the radioactive decay of the Cs and Sr into stable daughter products.

A.2.3 INACTIVE MISCELLANEOUS UNDERGROUND STORAGE TANK WASTE INVENTORY

Approximately 40 of the 60 total MUSTs in the Central Plateau that are associated with tank farm operations are inactive MUSTs with inventory that is included in the waste inventory subject to treatment and disposal under the Tank Waste Remediation System (TWRS) (Figures A.2.3.1 and A.2.3.2).

Table A.2.3.1 presents the volume of liquid and solids in the inactive MUSTs (Rasmussen 1995). The total volume of waste in these tanks approximately 448,000 liters (L) (118,000 gallons [gal]), which is less than one-half of 1 percent of the waste volume contained in the SSTs. Definitive characterization data do not exist for the inactive MUSTs, but because they received the same waste products that are contained in the tanks, the concentration of constituents is also expected to be approximately the same.

A.2.4 FUTURE TANK WASTE ADDITIONS

Waste projections for future tank waste additions are shown in Table A.2.4.1. This waste is expected to be added to DSTs after being reduced in water content in the 242-A Evaporator. The majority of the future waste additions would come from decontamination and decommissioning activities at inactive facilities on the Hanford Site. This waste would be classified as dilute, noncomplexed waste (does not contain complexing organics) that are low-level liquid waste. The 100 Area final (terminal) cleanout waste is classified as double-shell slurry feed, which is waste that is concentrated in the evaporator to a point just below the sodium aluminate saturation boundary (Hanlon 1995). Some future tank waste additions may be high-level waste or mixed waste that would come from cleanout of existing Site facilities. These future waste additions would be typical of the types of waste currently stored in the tanks.

The potential relocation of the K Basins sludge to the DSTs would result in the addition of approximately 54 m^3 (1,930 ft^3) of sludge to these tanks. The sludge contains spent nuclear fuel, corrosion products, small pieces of spent nuclear fuel (primarily uranium), iron oxides and aluminum oxides, concrete grit, fission and activation products from the spent nuclear fuel, and other materials such as sand and dust from the outside environment. The discovery of polychlorinated biphenyls in the sludge may affect the ability of the tank farms to accept the sludge. This waste would add approximately 11,000 Ci to the DSTs. This would include approximately 5,200 Ci of plutonium-241 (Pu-241), 260 Ci of plutonium-239 (Pu-239), 1,280 Ci of Sr-90, and 970 Ci of Cs-137. Following basin cleanout, the sludge plus about 1,200 m^3 (43,000 ft^3) of water would be transported to the DSTs for waste management, treatment, and disposition.

A.3.0 TANK INVENTORY DATA DISCUSSION

Obtaining representative sample data from the tanks is a very expensive and potentially hazardous activity because the tanks contain high levels of radioactive constituents and because the tank contents are heterogeneous. The SST chemical waste inventory data were derived using historical tank data based on the normalized Track Radioactive Component (TRAC) data. TRAC is a model that was developed to estimate tank waste radioactive inventories. The TRAC model output was later modified to account for known processing parameters and was then identified as normalized TRAC data.

The DST chemical and radiological waste inventories were developed using tank sample data in combination with historical tank data. DST radionuclide estimates were based on existing laboratory data and characterization reports. The isotopes presented in this appendix for DSTs were those consistently reported by laboratories, which is why the number of isotopes reported for DSTs is different than SSTs.

The waste inventory data used in developing the alternatives and their associated impacts were derived from model predictions and sample analysis. While the waste is currently undergoing additional characterization and the inventory may be revised as a result of ongoing analyses, the inventory used in the EIS is not expected to result in the discrimination for or against any of the alternatives presented.

There is considerable uncertainty associated with these inventory data. Additional tank characterization is required before final design of any alternative can take place. However, for the purposes of conceptual design, the concept of a nominal waste feed stream based on overall tank waste inventory can be used to develop plant capacities, project plant performance, and provide initial equipment sizing. The use of a nominal feed allows each of the proposed alternatives to be developed conceptually to a point where they can be analyzed in this EIS. This approach does not preclude the need for additional characterization.

A.3.1 OTHER TANK CHARACTERIZATION PROGRAMS

Several ongoing activities are involved with collecting and analyzing data on tanks contents. Each of these efforts is an attempt to provide more detailed and accurate tank waste inventory data.
The following are ongoing programs:

- Tank Characterization Program - Sampling and analysis of tank waste;
- Los Alamos National Laboratory (LANL) - Historical estimates based on observed waste stream data and process knowledge to develop inventory; and
- Historical Tank Content Estimates - Compiling available historical data.

The Tank Characterization Program, further addressed in Appendix B, gathers waste samples from each of the tanks for analysis. This program, which is based on data needs, is responsible for collecting and analyzing tank waste to satisfy the data requirements for tank safety issues and remediation process design. Ongoing waste characterization program activities to improve the estimates for tank waste inventory include 1) waste sampling and laboratory analysis; 2) data interpretation; and 3) historical review. The historical review provides a basis and background in data interpretations on waste management activities.

The LANL waste characterization effort consists of a series of spreadsheet-based computer models that derive composition estimates for the waste streams distributed to the tanks. When reconciled with the waste transaction records, these waste streams will provide an estimated accounting of the waste present in each tank as a function of time. Initial indications are that these model estimates, in their current form, are moderately successful in predicting certain bulk waste properties and inventories (WHC 1994f). Initial modeling results have been completed for all of the SSTs (solids inventory only) and DSTs. This program is ongoing, with plans to develop the model for the tank farm operations to track the tank waste inventory.

The Historical Tank Content Estimates are a series of documents being prepared by the current Management and Operations contractor that combine available historical tank data with the characterization data estimated by LANL (Agnew 1994). These documents will compile the tank waste volumes, photographs, temperatures, waste types, and waste inventory estimates over time (WHC 1994g, h, and WHC 1995b, o). Historical Tank Content Estimates have been initially released for all of the SSTs and DSTs. This is an ongoing program and current planning includes updating these documents during 1996.

A.3.2 LOS ALAMOS NATIONAL LABORATORY TANK WASTE CHARACTERIZATION DATA

The estimation of tank contents using the LANL model is expected to be completed by October 1996. At present, the LANL model has been used to estimate the composition of the solids (sludge plus saltcake) in the SSTs, and the composition of the solids and liquid in the DSTs. There are enough data from the LANL model to make a comparison with the inventory data package that is used in the EIS (WHC 1995d). Tables A.3.3.1 and A.3.3.2 compare the metric tons of chemicals and the metric tons or curies of radionuclides that are reported for the inventory data package and the LANL model (Agnew 1994, WHC 1994g, h, and WHC 1995b, o). The comparison of chemical constituents is limited to those chemicals that are common to both inventories. The comparison of radionuclides is restricted to those that are reported for the LANL model.

A general comparison of the amounts reported by the LANL model and the data package shows that the LANL model routinely reports amounts that are several times greater than the corresponding amounts from the data package. This result is observed for both chemicals and radionuclides. However, when the LANL model reports are complete, the total differences may be less. The derivation of the LANL model and the generation of the inventory data are both sufficiently complex that the source of the differences between the two are not readily explained. However, it is possible to address the two inventory sources in the light of their effect on the EIS. The EIS uses inventories as the basis for calculating risks, both during the remediation phase of the alternatives and during the post-remediation phase. Risks during remediation arise primarily from releases to the atmosphere. Risks during post remediation are caused by releases to groundwater.

Risks during remediation are caused primarily from exposure to Cs-137, Sr-90, iodine-129 (I-129), and carbon-14 (C-14) for radionuclides and volatile organic compounds (VOCs) for chemicals. The LANL model shows only Cs and Sr, so it cannot be used to calculate the risks for I-129 and C-14. The LANL model indicates a Cs content in the SSTs that is over four times that reported in the data package. In the case of Sr, the LANL model indicates twice as much in the DSTs than the data package reports.

Neither the LANL model nor the data package report VOCs, so another data source was used for these chemicals. If data from the LANL model were used for the EIS, calculations would show somewhat higher risks during remediation because of increased Cs and Sr quantities.

Risks during post remediation are caused by mobile elements migrating through groundwater. The mobile radionuclides of concern are C-14, I-129, technetium-99 (Tc-99), and uranium. The mobile chemical constituent of concern is the nitrate anion. The LANL model only indicates quantities for uranium and nitrate. Quantities are not shown for C-14, I-129, and Tc-99 for the LANL model, so no differences from currently projected impacts could be calculated. The LANL model indicates about 20 percent more uranium in the SSTs than the inventory data package shows. For the DSTs, the inventory data package does not indicate any uranium in the DSTs, while the LANL model shows 160 metric tons. For total uranium in both SSTs and DSTS, the LANL model indicates about 30 percent more uranium than the inventory data package shows. In the case of nitrate quantities, the inventory data package shows about twice as much in the SSTs than the LANL model shows. Both estimates are essentially equal for nitrate in the DSTs. The effect of using quantities estimated by the LANL model for the EIS would be to indicate marginally higher risks in post remediation caused by uranium and somewhat lower risks caused by nitrate.

A.3.3 TANK INVENTORY DATA ACCURACY AND ITS EFFECT ON THE EIS

The predicted inventories from different models will not necessarily be in agreement with regards to the kinds and quantities of substances that make up the tank wastes. There is an ongoing effort to compile a standard inventory estimate that would serve as a unified source of tank constituents (WHC 1995q). These best-basis estimates are to be incorporated into the existing Tank Characterization Database. However, this work is in its initial stages and completion is expected at a future date. Until this unified source has been completed and is universally used, other documents, such as the EIS, must use available inventory data and recognize the effects of inaccuracies in those data. This section presents the effects of inventory data accuracy on the various portions of the EIS.

An important point to keep in mind when considering inventory data accuracy is the ultimate significance of the data as they are used to calculate or predict environmental impacts. For a substance that is present in minute quantities and is not radioactive or toxic, high accuracy in reporting that substance in the tank inventory is not required. The effects of variation in the amount of such a benign substance would not be great. Conversely, if a substance is a major tank waste constituent, or is highly radioactive or very toxic, the accuracy in reporting that substance and the ultimate effect on environmental impacts must be recognized. For example, sodium is a major waste component and its quantities will affect the size of the low-activity waste facility for the ex situ alternatives. However, the pre-conceptual estimation of the size and cost of facilities for the EIS has a variation that is typically plus or minus 40 percent. This variation in size and cost estimation is based on factors that include the variability of the feed stock. A variation in sodium quantities by plus or minus 20 percent would not produce environmental effects that were unexpected.

Rather than discuss the effects of inventory accuracy on an element-by-element basis, this section presents the measures that were taken by each function or discipline to account for the variability of the tank waste inventory. These measures must strike a balance between understating environmental impacts and overstating these impacts by compounding conservatism upon conservatism. In addition to the discussion in this section, each appendix contains the major assumptions and uncertainties, which include other factors in addition to uncertainties in tank inventory data.

Engineering

To provide conservatism in generating inventory information for use by other disciplines, the engineering function used the inventory data package as the basis for conservative estimates of the releases during retrieval and subsequent processing; the dissolution of the residual materials remaining in the tanks and the low-activity waste vaults; and the effects of blending and composition on the volume of high-level waste glass or calcine. Releases from the tanks during ongoing current operations were obtained directly from analytical data, which do not involve concentration modeling. The data relating to these releases were used directly, with no additional conservative factors being applied.

Groundwater Modeling

The inventories generated by the engineering function were used without change by the groundwater modeling function. To ensure that groundwater effects were not understated, conservative values of distribution coefficients (K_d) were used. While this would not affect the inventory of contaminants, it would ensure that the travel times of contaminants were at the upper bound of the range that is generally accepted for these studies. While other assumptions were made to complete the groundwater modeling, they did not directly involve the contaminant inventory.

Air Modeling

The model inputs used by the air modeling function were the routine emissions from the tank farms and emissions from the remediation facilities. The air modeling function used the analytical results from ongoing current operations to predict the concentrations of contaminants that would be released from the tank farms. The emissions from the remediation facilities were provided by the engineering function (Jacobs 1996). The analytical results from current tank farm operations were obtained by direct measurement and were considered to be sufficiently accurate for use without modification. Emissions from remediation facilities are directly related to the tank inventories because it is the tank contents that are being processed. Because the models that predict air contaminant concentrations are considered sufficiently conservative, the calculated emissions from the remediation facilities were used without further modification.

Risk Assessment

Inventory data were used to calculate risks from routine exposures and accidents during remediation and post-remediation activities. The assessment of risk from routine exposures during remediation used the same inputs as the air modeling function. As explained in the previous paragraph, the analytical results from ongoing operations of the tank farms and the calculated emissions from the remediation facilities were used. Because the results of the groundwater modeling were used as input to the assessment of risk during post remediation, the conservatism employed by groundwater modeling was directly reflected in the risk assessment modeling. Consequently, further conservative assumptions concerning the contaminant concentrations were not postulated.

The accepted practice for assessing risks from accidents during remediation combines the overall inventory of contaminants, both modeled and analyzed, to form the contents of a so-called super tank. This is a unique use of the tank inventory and is intended to ensure that the consequences of accidents invariably involve exposures to the same quantities of contaminants. This concept is used solely for accident analysis and is consistent with current Hanford Site practice. The assessment of risks during post remediation uses the conservative estimate of the volume and inventory of the high-level waste glass or calcined product, which has been provided by the engineering function. The models that calculate the consequences of transportation accidents are considered sufficiently conservative, and the inventory provided by the engineering function is used without modification.

Figure A.2.1.1 Location of Tank Waste Source Areas

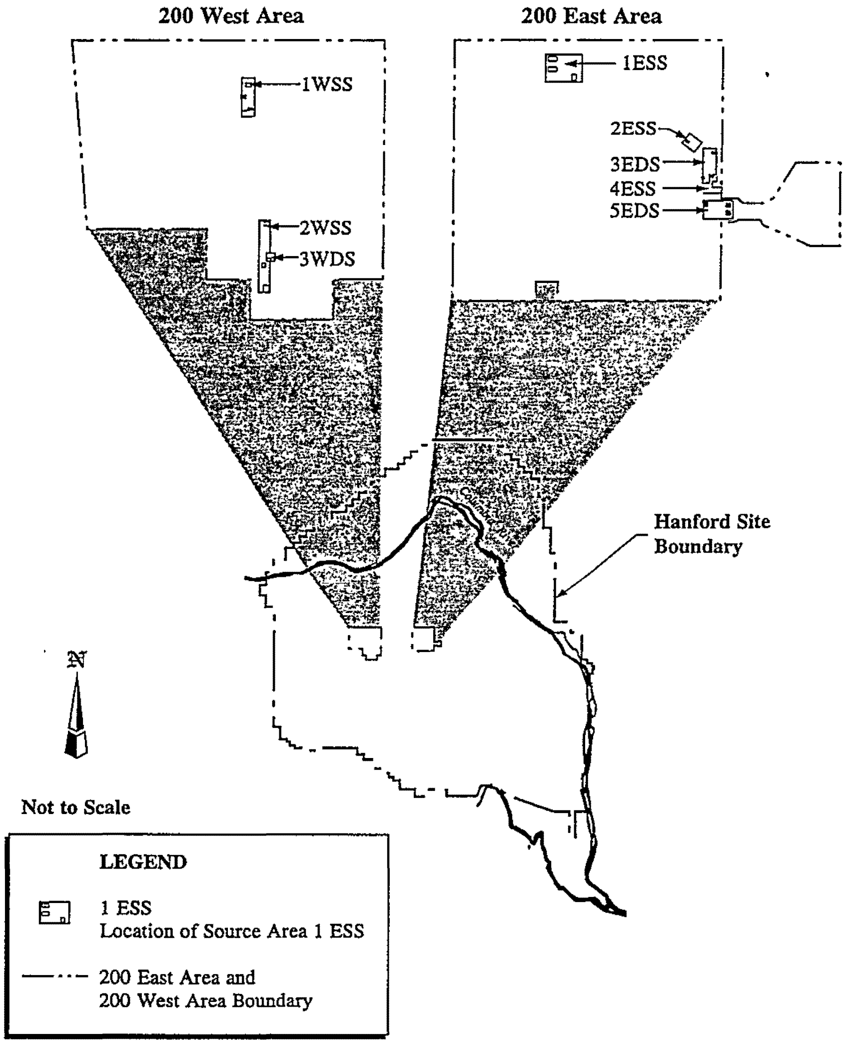
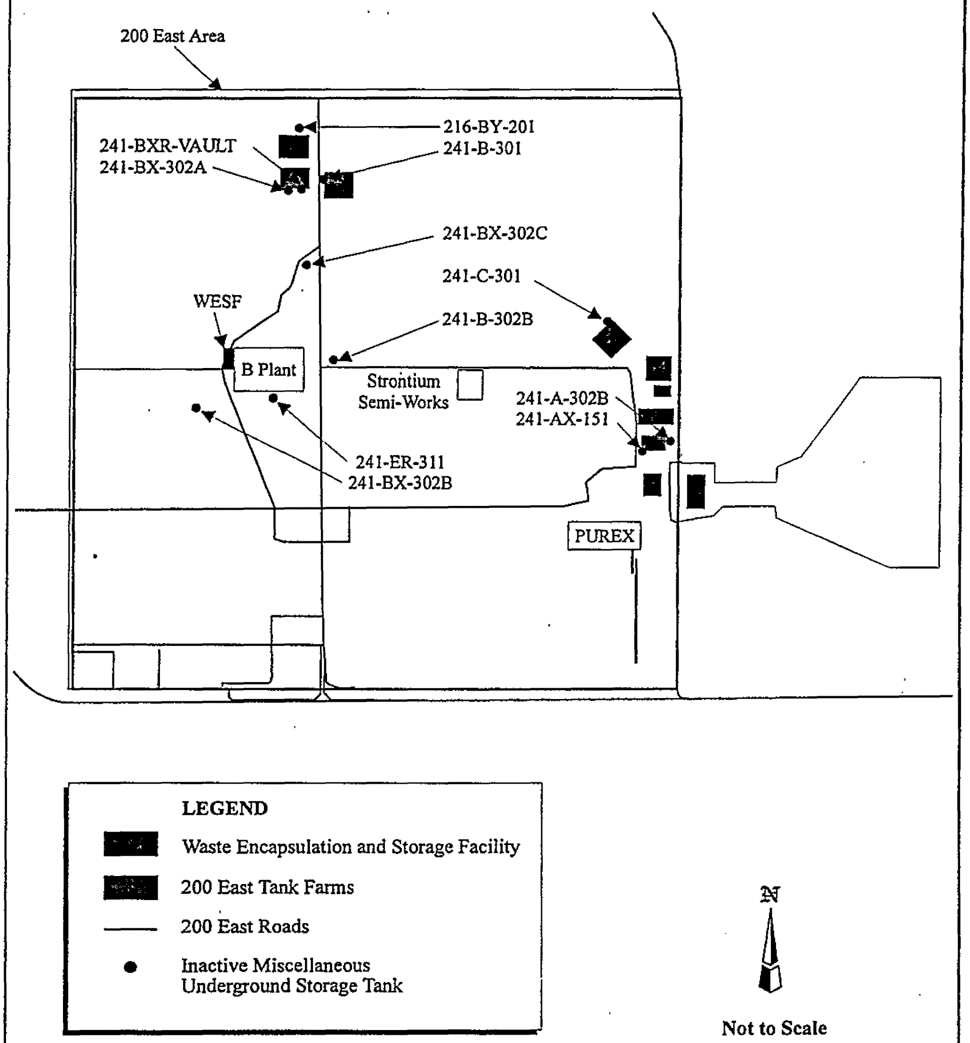
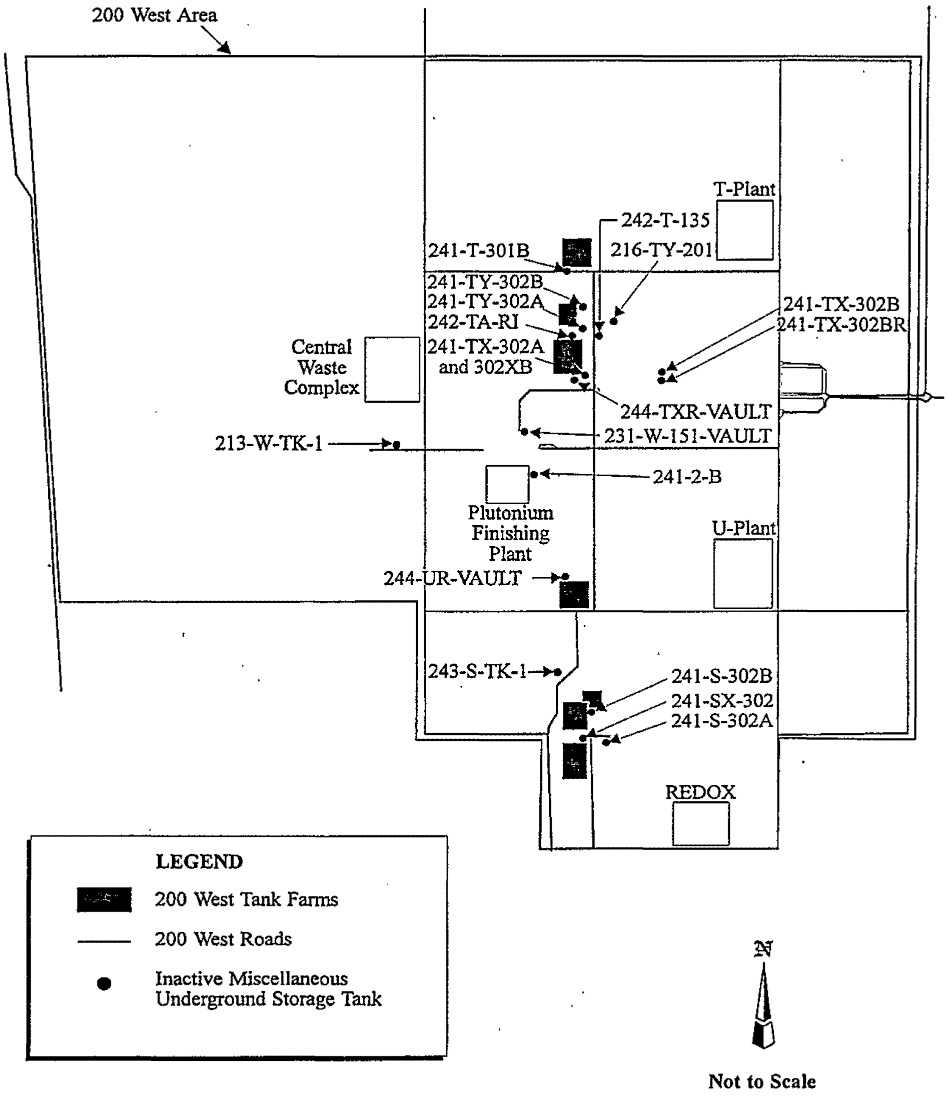


Figure A.2.3.1 Inactive Miscellaneous Underground Storage Tank Locations - 200 East Area



SOURCE: Adapted from Rasmussen 1995

Figure A.2.3.2 Inactive Miscellaneous Underground Storage Tank Locations - 200 West Area



SOURCE: Adapted from Rasmussen 1995

Table A.2.1.1 Tank Waste Volumes[1]

Waste Form	Inactive MUSTs [2] m^3 (gal)	SSTs [3] m^3 (gal)	DSTs [3] m^3 (gal)	Total m^3 (gal)
Liquid				
Supernatant	45 (12,000)	2,170 (551,000)	56,200 (14,842,000)	58,400 (15,400,000)
Solids [4]				
DST Slurry	N/A	0	7,720 (2,040,000)	7,720 (2,040,000)
Sludge	360 (95,000)	45,400 (11,997,000)	7,400 (1,955,000)	53,200 (14,050,000)
Saltcake	N/R	88,000 (23,242,000)	2,880 (760,000)	90,900 (24,000,000)
Total Waste	405 [5] (107,000)	135,600 (35,800,000)	74,200 (19,600,000)	210,200 (55,500,000)

Notes:

[1] Quantities with three or more significant digits do not imply a specific accuracy of the stated value.

[2] Source: Rasmussen 1995

[3] Source: Hanlon 1996

[4] Solids contain interstitial liquid that is contained within the interstitial spaces of the sludge and saltcake and is not added to the total waste volume. For SSTs the volume of interstitial liquid is 23,390 m^3 (6,051,000 gal), for DSTs the volume of interstitial liquid is 1,640 m^3 (439,000 gal), and for inactive MUSTs no interstitial liquid volume estimate was provided. The SSTs interstitial liquid remains in the tanks following interim stabilization.

[5] Total waste volume listed is greater than the sum of the liquid and solid waste forms listed and accounts for tanks where the total volume of waste is known but waste type is unknown.

N/A = Not applicable
N/R = Not Reported

Table A.2.1.2 Estimated Mass of Nonradioactive Chemical Components of SST and DST Waste in Metric Tons [1, 2]

Chemical Species	SSTs				DSTs			Overall SST and DST Total
	Sludge	Saltcake	Interstitial Liquid	Total	Soluble	Insoluble	Total	
Ag^+					3.28E-01	1.38E+00	1.7E+00	1.7E+00
$Al(OH)_4^-$	6.25E+02	1.25E+03	4.57E+02	2.33E+03	5.09E+03		5.09E+03	7.43E+03
Al^{+3} (3)	1.99E+03			1.99E+03		6.78E+01	6.78E+01	2.06E+03
As^{+5}					7.70E-01	4.98E-01	1.27E+00	1.27E+00
B^{+3}					5.19E-01	9.94E-01	1.51E+00	1.51E+00
Ba^{+2}					7.91E-01	3.09E+00	3.88E+00	3.88E+00
Be^{+2}					8.19E-02	7.61E-03	8.95E-02	8.95E-02
Bi^{+3}	2.61E+02			2.61E+02	2.26E+00		2.26E+00	2.64E+02
Ca^{+2}	1.28E+02			1.28E+02	1.03E+01	1.15E+01	2.18E+01	1.50E+02
Cd^{+2}	3.84E+00			3.84E+00	1.67E-01	6.01E+00	6.18E+00	1.00E+01
Ce^{+3}	2.35E+02			2.35E+02	2.26E-02	3.04E+00	3.07E+00	2.38E+02
Cl^-	4.00E+01			4.00E+01	2.73E+02	1.49E+00	2.74E+02	3.14E+02
CO_3^{-2}	1.15E+03	4.13E+02	3.96E+01	1.61E+03	1.92E+03	5.83E+01	1.98E+03	3.59E+03
Cr^{+3}	8.63E+01			8.63E+01	1.20E+02	3.41E+01	3.41E+01	1.20E+02
CrO_4^{-2}			2.14E+01	2.14E+01	1.77E-01		1.20E+02	1.41E+02
Cu^{+2}					3.25E+02	7.46E-01	9.23E-01	9.23E-01
F^-	8.00E+02		5.00E+01	8.05E+02		1.91E+01	3.71E+02	1.18E+03
$Fe(CN)_6^{-4}$	3.22E+02			3.22E+02	8.09E+00			3.22E+02
Fe^{+3}	6.27E+02			6.27E+02	5.84E-02	1.42E+02	1.50E+02	7.77E+02
Hg^+	9.00E-01			9.00E-01	5.46E+02		5.84E-02	9.58E-01
K^+					2.19E-01	2.02E+01	5.66E+02	5.66E+02
La^+					5.77E-03	2.10E+01	2.12E+01	2.12E+01
Li^+					9.65E-01	2.46E-02	3.04E-02	3.04E-02
Mg^{+2}					7.69E+00	1.10E+01	1.20E+01	1.20E+01
Mn^{+4}	1.20E+02			1.20E+02	4.87E+00	1.80E+01	2.57E+01	1.46E+02
Mo^{+6}					1.40E+04	8.01E-01	5.67E+00	5.67E+00
Na^+	1.58E+04	3.39E+04	2.30E+03	5.48E+04	4.07E+00	2.30E+02	1.43E+04	6.91E+04
Ni^{+2}	1.78E+02			1.78E+02	4.80E+03	6.57E+00	1.06E+01	1.89E+02
NO_2^-	2.00E+03	1.53E+03	1.27E+03	4.80E+03	1.03E+04	8.42E+00	4.81E+03	9.61E+03
NO_3^-	1.48E+04	8.03E+04	1.71E+03	9.68E+04	2.33E+03	3.91E+01	1.03E+04	1.07E+05
OH-	4.22E+03	8.51E+02	3.15E+02	5.39E+03	1.96E+00	1.23E+02	2.45E+03	7.84E+03
Pb^{+4}					3.29E+02	3.28E+00	5.24E+00	5.24E+00

Table A.2.1.2 Estimated Mass of Nonradioactive Chemical Components of SST and DST Waste in Metric Tons [1,2] (cont'd)

Chemical Species	SSTs				DSTs			Overall SST and DST Total
	Sludge	Saltcake	Interstitial Liquid	Total	Soluble	Insoluble	Total	
PO_4^{-3}	3.89E+03	6.43E+02	8.58E+01	4.62E+03	1.53E+01	2.16E+01	3.51E+02	4.97E+03
SiO_3^{-2}	1.21E+03			1.21E+03	3.86E+02	2.14E+02	2.29E+02	1.44E+03
SO_4^{-2}	5.01E+02	1.15E+03		1.65E+03		6.68E+00	3.93E+02	2.04E+03
Sr^{+2}	3.60E+01			3.60E+01	1.26E+03			3.60E+01
TOC(4)			2.00E+02	2.00E+02	3.54E+00	6.84E+01	1.33E+03	1.53E+03
UO_2^{+2}					6.20E-02	2.68E+01	3.03E+01	3.03E+01
V^{+5}					7.47E-01	1.88E-01	2.50E-01	2.50E-01
W^{+4}	1.44E+01			1.44E+01	3.59E+00		7.47E-01	1.52E+01
Zn^{+2}					4.48E-01	9.45E-01	4.54E+00	4.54E+00
Zr^{+4}	2.46E+02			2.46E+02		2.77E+02	2.77E+02	5.24E+02
Total w/o H_2O	4.93E+04	1.23E+05	6.40E+03	1.79E+05	4.18E+04	1.45E+03	4.32+04	2.22E+05
H_2O	2.62E+04	1.40E+04	5.16E+03	4.54E+04	8.59E+04		8.95+04	1.35E+05
TOTAL	7.55E+04	1.37E+05	1.16E+04	2.24E+05	1.31E+05	1.45E+03	1.33E+05	3.57E+05

Notes:

[1] One metric ton is equal to 1,000 kilograms (2,205 pounds mass).

[2] Values with three or more significant digits do not imply a specific accuracy of the stated value; no data entry provided for insignificant inventories.

[3] Al^{+3} includes the Al present in cancrinite and $Al(OH)_3$.

[4] Total organic carbon includes HEDTA, EDTA, hydroxyacetic acid, citric acid, and degradation products.

Table A.2.1.3 Estimated Radionuclide Inventory for SSTs and DSTs in Curies [1, 2]

Radionuclides	SSTs Total	DSTs		
		Soluble	Insoluble	Total
Ac-225	1.98E-05			
Ac-227	2.21E-02			
Am-241	3.30E+04	5.31E+03	6.54E+04	7.07E+04
Am-242	6.82E+01			
Am-242m	6.86E+01			
Am-243	3.32E+01			
At-217	1.98E-05			
Ba-137m	7.68E+06	2.48E+07	6.49E+05	2.54E+07
Bi-210	7.17E-08			
Bi-211	2.21E-02			
Bi-212	3.72E-14			
Bi-213	1.98E-05			
Bi-214	2.70E-07			
C-14 (3)	3.00E+03	3.45E+02	1.99E+03	2.34E+03
Cm-242	5.66E+01			
Cm-244	1.18E+02			
Cm-245	1.04E-02			
Cs-135	1.45E+02			
Cs-137	8.12E+06	2.61E+07	6.83E+05	2.68E+07
Eu-154		5.37E+04	1.44E+03	5.51E+04
Fr-221	1.98E-05			
Fr-223	3.06E-04			
I-129 (3)	1.60E+01	1.90E+01	3.30E+00	2.23E+01
Nb-93m	3.20E+03			
Ni-59	5.03E+03			
Ni-63	2.69E+05			
Np-237	6.97E+01			
Np-238	3.26E-01			
Np-239	3.32E+01			
Pa-231	3.80E-02			
Pa-233	6.97E+01			
Pa-234	7.69E-01			
Pa-234m	4.81E+02			
Pb-209	1.98E-05			
Pb-210	7.17E-08			
Pb-211	2.21E-02			

Table A.2.1.3 Estimated Radionuclide Inventory for SSTs and DSTs in Curies [1,2] (cont'd)

Radionuclides	SSTs Total	DSTs		
		Soluble	Insoluble	Total
Pb-212	3.72E-14			
Pb-214	2.70E-07			
Pd-107	8.65E+01			
Po-210	7.17E-08			
Po-211	6.04E-05			
Po-212	2.38E-14			
Po-213	1.94E-05			
Po-214	2.70E-07			
Po-215	2.21E-02			
Po-216	3.72E-14			
Po-218	2.70E-07			
Pu-238	1.08E+03			
Pu-239 (3)	1.80E+04	1.31E+03	7.05E+03	8.36E+03
Pu-240 (3)	4.30E+03	3.28E+02	2.07E+03	2.40E+03
Pu-241	3.55E+04	7.76E+02	3.86E+04	3.94E+04
Pu-242	4.32E-04			
Ra-223	2.21E-02			
Ra-224	3.72E-14			
Ra-225	1.98E-05			
Ra-226	2.70E-07			
Ra-228	7.42E-14			
Rh-106	3.79E-02			
Rn-219	2.21E-02			
Rn-220	3.72E-14			
Rn-222	2.70E-07			
Ru-106	3.79E-02			
Sb-126	8.78E+01			
Sb-126m	6.27E+02			
Se-79	9.11E+02			
Sm-151 (3)	6.30E+05			
Sn-126	6.27E+02			
Sr-90 (3)	4.36E+07	6.15E+05	9.47E+06	1.01E+07
Tc-99 (3)	1.10E+04	2.07E+04	3.99E+02	2.11E+04
Th-227	2.18E-02			
Th-228	3.72E-14			
Th-229	1.98E-05			

Table A.2.1.3 Estimated Radionuclide Inventory for SSTs and DSTs in Curies [1,2] (cont'd)

Radionuclides	SSTs Total	DSTs		
		Soluble	Insoluble	Total
Th-230	3.90E-05			
Th-231	2.06E+01			
Th-232	6.42E-13			
Th-234	4.81E+02			
Tl-207	2.21E-02			
Tl-208	1.34E-14			
Tl-209	4.28E-07			
U-233	1.21E-02			
U-234	2.12E-01			
U-235	2.06E+01			
U-236	2.88E-03			
U-237	8.69E-01			
U-238	4.81E+02			
Y-90	4.36E+07	6.15E+05	9.47E+06	1.01E+07
Zr-93	3.94E+03			
TOTAL	1.04E+08	5.23E+07	2.04E+07	7.27E+07

Notes:

[1] Values with three or more significant digits do not imply a specific accuracy of the stated value; no data entry provided for insignificant inventories.

[2] Radionuclides reflect decay and ingrowth to December 31, 1999.

[3] SST amounts adjusted from original Track Radioactive Component (TRAC) output to account for inventory adjustments based on sample analysis and waste transfers to DSTs.

Table A.2.1.4 Tank Source Areas

Source Area	Location	Tank Type	Tank Farms
1WSS	200 W	SST	T, TX, TY
2WSS	200 W	SST	U, S, SX
3WDS	200W	DST	SY
1ESS	200 E	SST	B, BX, BY
2ESS	200 E	SST	C
3EDS	200 E	DST	AN, AZ, AY
4ESS	200 E	SST	A, AX
5EDS	200 E	DST	AW, AP

Table A.2.1.5 Estimated Mass of Nonradioactive Chemical Components of SSTs by Aggregated Tank Grouping in Metric Tons [1, 2]

Chemical Species	SST Groupings					Total
	1WSS	2WSS	1ESS	2ESS	4ESS	
	40 tanks	43 tanks	40 tanks	16 tanks	10 tanks	149 tanks
$Al(OH)_4^-$	3.32E+02	1.11E+03	8.04E+02	3.87E+02	2.64E+01	2.66E+03
Al^{+3}	7.12E+01	7.24E+02	5.92E+02	5.80E+02	2.53E+01	1.99E+03
Bi^{+3}	1.66E+02	2.80E-01	9.43E+01	5.95E-01		2.61E+02
CO_3^{-2}	7.07E+02	1.60E+02	6.40E+02	6.20E+01	3.80E+01	1.61E+03
Ca^{+2}	1.98E-01	5.82E+00	5.47E+01	6.59E+01	1.54E+00	1.28E+02
Cd^{+2}	8.63E-01	1.02E+00	1.16E+00	6.11E-01	1.78E+01	3.48E+00
Ce^{+3}	1.16E+02	3.16E+01	8.82E+01	2.64E-01	1.75E+00	2.38E+02
Cl^-	2.19E+01	5.41E+00	1.26E+01	4.01E-02	6.03E-02	4.00E+01
Cr^{+3}	1.26E+00	8.33E+01	7.25E-01	1.65E-01	8.24E-01	8.63E+01
CrO_4^{-2}	3.12E-01	2.07E+01	1.80E-01	4.10E-02	2.04E-01	2.14E+01
F^-	1.44E+02	3.00E+01	2.32E+02	4.00E+02	3.08E-01	8.06E+02
Fe^{+3}	1.70E+02	8.18E+01	1.42E+02	5.55E+01	1.78E+02	6.27E+02
$Fe(CN)_6^{-4}$	2.19E+00	1.37E+00	2.48E+02	7.00E+01	5.30E-02	3.22E+02
Hg^+	2.02E-01	2.40E-01	2.73E-01	1.43E-01	4.17E-02	9.00E-01
Mn^{+4}	2.10E+01	1.13E+01	1.31E+01	5.12E+01	2.35E+01	1.20E+02
Na^+	1.45E+04	2.11E+04	1.16E+04	2.10E+03	3.78E+03	5.31E+04
Ni^{+2}	5.02E+00	3.33E+00	1.25E+02	4.40E+01	9.93E-01	1.78E+02
NO_2^-	1.76E+03	8.01E+02	2.06E+03	4.74E-01	2.32E+02	4.85E+03
NO_3^-	2.63E+04	4.55E+04	1.89E+04	2.59E+02	4.89E+03	9.59E+04
OH^-	4.53E+02	2.30E+03	1.04E+03	1.42E+03	1.74E+02	5.39E+03
PO_4^{-3}	2.67E+03	1.10E+02	1.81E+03	2.98E+01	8.61E-01	4.62E+03
SiO_3^-	5.60E+02	2.40E+02	4.04E+02	7.07E-01	2.09E+00	1.21E+03
SO_4^{-2}	6.34E+02	2.48E+02	5.53E+02	1.34E+02	8.35E+01	1.65E+03
Sr^{+2}	9.38E-04	6.98E-02	3.59E+01	2.53E-02	5.78E-02	3.60E+01
W^{+4}	3.42E+00	3.84E+00	4.38E+00	2.30E+00	6.68E-01	1.44E+01
Zr^{+4}	1.40E+01	2.31E+01	6.50E+00	2.03E+02	1.28E-01	2.46E+02

Notes:

[1] Minor differences exist between the aggregated totals for some chemical species and the total quantities reported in Table A.2.1.2. These differences are a result of updates made to the overall inventory that are not currently reflected in the aggregated inventory.

[2] Values with three or more significant digits do not imply a specific accuracy of the stated value.

Table A.2.1.6 Estimated Radionuclide Inventory for Aggregated SST Groupings in Curies [1,2,3]

Radionuclide	SST Groupings					Total
	1WSS	2WSS	1ESS	2ESS	4ESS	
	40 tanks	43 tanks	40 tanks	16 tanks	10 tanks	149 tanks
Ac-225	1.72E-06	2.88E-06	6.03E-06	2.25E-06	2.84E-06	1.57E-05
Ac-227	5.18E-03	4.26E-03	8.84E-03	1.55E-03	2.77E-04	2.01E-02
Am-241	1.70E+03	9.48E+03	7.49E+03	9.72E+03	4.62E+03	3.30E+04
Am-242	2.68E+00	1.94E+01	1.88E+01	2.00E+01	8.67E+00	6.95E+01
Am-242m	2.69E+00	1.95E+01	1.89E+01	2.01E+01	8.71E+00	6.98E+01
Am-243	1.02E+00	7.82E+00	9.57E+00	1.16E+01	3.12E+00	3.32E+01
At-217	1.72E-06	2.88E-06	6.03E-06	2.25E-06	2.84E-06	1.57E-05
Ba-137m	8.44E+05	3.78E+06	3.54E+06	1.26E+05	1.37E+05	8.42E+06
Bi-210	1.50E-08	1.16E-08	1.34E-08	7.03E-09	2.81E-09	4.99E-08
Bi-211	5.18E-03	4.26E-03	8.84E-03	1.55E-03	2.77E-04	2.01E-02
Bi-213	1.72E-06	2.88E-06	6.03E-06	2.25E-06	2.84E-06	1.57E-05
Bi-214	6.08E-08	5.03E-08	5.20E-08	2.98E-08	1.58E-08	2.09E-07
C-14	2.84E+02	4.90E+02	1.83E+03	2.15E+02	1.82E+02	3.00E+03
Cm-242	2.22E+00	1.61E+01	1.56E+01	1.66E+01	7.19E+00	5.76E+01
Cm-244	3.22E+00	2.39E+01	5.01E+01	5.34E+01	7.20E+00	1.38E+02
Cm-245	2.08E-04	1.74E-03	3.83E-03	4.09E-03	5.51E-04	1.04E-02
Cs-135	2.07E+01	6.79E+01	5.27E+01	1.61E+00	2.13E+00	1.45E+02
Cs-137	8.93E+05	3.99E+06	3.74E+06	1.33E+05	1.44E+05	8.90E+06
Fr-221	1.72E-06	2.88E-06	6.03E-06	2.25E-06	2.84E-06	1.57E-05
Fr-223	7.15E-05	5.88E-05	1.22E-04	2.14E-05	3.82E-06	2.78E-04
I-129	1.70E+00	4.39E+00	9.14E+00	5.97E-01	1.71E-01	1.60E+01
Nb-93m	8.54E+01	6.76E+02	3.71E+02	4.05E+02	1.50E+03	3.04E+03
Ni-59		1.71E+03	3.33E+03			5.03E+03
Ni-63	6.83E+03	4.86E+04	5.30E+04	5.87E+04	1.09E+05	2.76E+05
Np-237	8.26E+00	1.10E+01	4.96E+01	3.37E-01	4.48E-01	6.96E+01
Np-238	1.28E-02	9.26E-02	9.00E-02	9.55E-02	4.15E-02	3.32E-01
Np-239	1.02E+00	7.82E+00	9.57E+00	1.16E+01	3.12E+00	3.32E+01
Pa-231	9.61E-03	7.33E-03	1.53E-02	3.38E-03	6.37E-04	3.62E-02
Pa-233	8.26E+00	1.10E+01	4.96E+01	3.37E-01	4.48E-01	6.96E+01
Pa-234	2.75E-01	1.03E-01	2.61E-01	1.03E-01	2.64E-02	7.69E-01
Pa-234m	1.72E+02	6.47E+01	1.63E+02	6.45E+01	1.65E+01	4.81E+02
Pb-209	1.72E-06	2.88E-06	6.03E-06	2.25E-06	2.84E-06	1.57E-05

Table A.2.1.6 Estimated Radionuclide Inventory for Aggregated SST Groupings in Curies [1,2,3] (cont'd)

Radionuclide	SST Groupings					Total
	1WSS	2WSS	1ESS	2ESS	4ESS	
	40 tanks	43 tanks	40 tanks	16 tanks	10 tanks	149 tanks
Pb-210	1.50E-08	1.16E-08	1.34E-08	7.03E-09	2.81E-09	4.99E-08
Pb-211	5.18E-03	4.26E-03	8.84E-03	1.55E-03	2.77E-04	2.01E-02
Pb-214	6.08E-08	5.03E-08	5.20E-08	2.98E-08	1.58E-08	2.09E-07
Pd-107	9.03E+00	2.33E+01	4.95E+01	3.66E+00	9.74E-01	8.65E+01
Po-210	1.50E-08	1.16E-08	1.34E-08	7.03E-09	2.81E-09	4.99E-08
Po-211	1.41E-05	1.16E-05	2.41E-05	4.24E-06	7.56E-07	5.49E-05
Po-213	1.69E-06	2.81E-06	5.90E-06	2.20E-06	2.78E-06	1.54E-05
Po-214	6.08E-08	5.03E-08	5.20E-08	2.98E-08	1.58E-08	2.09E-07
Po-215	5.18E-03	4.26E-03	8.84E-03	1.55E-03	2.77E-04	2.01E-02
Po-218	6.08E-08	5.03E-08	5.20E-08	2.98E-08	1.58E-08	2.09E-07
Pu-238	2.10E+02	2.96E+02	1.85E+02	1.99E+02	2.24E+02	1.11E+03
Pu-239	2.08E+03	3.59E+03	2.90E+03	4.84E+03	4.59E+03	1.80E+04
Pu-240	4.09E+02	7.91E+02	6.94E+02	1.24E+03	1.17E+03	4.30E+03
Pu-241	3.92E+03	6.36E+03	8.46E+03	1.32E+04	1.11E+04	4.30E+04
Pu-242	1.33E-05	9.59E-05	9.32E-05	9.89E-05	4.29E-05	3.44E-04
Ra-223	5.18E-03	4.26E-03	8.84E-03	1.55E-03	2.77E-04	2.01E-02
Ra-225	1.72E-06	2.88E-06	6.03E-06	2.25E-06	2.84E-06	1.57E-05
Ra-226	6.08E-08	5.03E-08	5.20E-08	2.98E-08	1.58E-08	2.09E-07
Rh-106	6.11E-05	1.70E-02	8.25E-02	9.73E-02	3.97E-01	5.94E-01
Rn-219	5.18E-03	4.26E-03	8.84E-03	1.55E-03	2.77E-04	2.01E-02
Rn-222	6.08E-08	5.03E-08	5.20E-08	2.98E-08	1.58E-08	2.09E-07
Ru-106	6.11E-05	1.70E-02	8.25E-02	9.73E-02	3.97E-01	5.94E-01
Sb-126	7.94E+00	2.33E+01	6.71E+00	1.39E+01	3.59E+01	8.78E+01
Sb-126m	5.67E+01	1.66E+02	4.79E+01	9.96E+01	2.57E+02	6.27E+02
Se-79	9.71E+01	2.51E+02	5.21E+02	3.23E+01	9.51E+00	9.11E+02
Sm-151	6.24E+04	1.84E+05	5.47E+04	1.01E+05	2.48E+05	6.50E+05
Sn-126	5.67E+01	1.66E+02	4.79E+01	9.96E+01	2.57E+02	6.27E+02
Sr-90	1.50E+06	1.43E+07	8.32E+06	4.90E+06	1.90E+07	4.80E+07
Tc-99	1.17E+03	3.03E+03	6.29E+03	3.93E+02	1.15E+02	1.10E+04
Th-227	5.11E-03	4.20E-03	8.72E-03	1.53E-03	2.73E-04	1.98E-02
Th-229	1.72E-06	2.88E-06	6.03E-06	2.25E-06	2.84E-06	1.57E-05
Th-230	9.00E-06	7.86E-06	7.51E-06	4.51E-06	3.11E-06	3.20E-05

Table A.2.1.6 Estimated Radionuclide Inventory for Aggregated SST Groupings in Curies [1, 2, 3] (cont'd)

Radionuclide	SST Groupings					Total
	1WSS	2WSS	1ESS	2ESS	4ESS	
	40 tanks	43 tanks	40 tanks	16 tanks	10 tanks	149 tanks
Th-231	7.22E+00	2.93E+00	6.92E+00	2.81E+00	6.98E-01	2.06E+01
Th-232	1.85E-14	3.57E-14	3.13E-14	5.59E-14	5.27E-14	1.94E-13
Th-234	1.72E+02	6.47E+01	1.63E+02	6.45E+01	1.65E+01	4.81E+02
Tl-207	5.16E-03	4.25E-03	8.82E-03	1.55E-03	2.76E-04	2.01E-02
Tl-209	3.73E-08	6.21E-08	1.30E-07	4.85E-08	6.13E-08	3.39E-07
U-233	1.20E-03	1.75E-03	5.91E-03	8.07E-04	1.18E-03	1.08E-02
U-234	4.91E-02	4.59E-02	4.31E-02	3.04E-02	2.52E-02	1.94E-01
U-235	7.22E+00	2.93E+00	6.92E+00	2.81E+00	6.98E-01	2.06E+01
U-236	2.16E-04	4.18E-04	3.67E-04	6.54E-04	6.17E-04	2.27E-03
U-237	9.60E-02	1.56E-01	2.07E-01	3.23E-01	2.71E-01	1.05E+00
U-238	1.72E+02	6.47E+01	1.63E+02	6.45E+01	1.65E+01	4.81E+02
Y-90	1.51E+06	1.45E+07	8.41E+06	4.95E+06	1.92E+07	4.85E+07
Zr-93	4.54E+01	8.00E+02	2.42E+02	5.63E+02	2.29E+03	3.94E+03

Notes:

[1] Minor differences exist between the aggregated totals for some radionuclides and the total quantities reported in Table A.2.1.3. These differences are a result of updates made to the overall inventory that are not currently reflected in the aggregated inventory.

[2] Radionuclides reflect decay to 12/31/95 and were back calculated from 12/31/99 data. No decay chains were used in back calculating inventories.

[3] Values with three or more significant digits do not imply a specific accuracy of the stated value; no data entry provided for insignificant inventories.

Table A.2.1.7 Estimated Mass of Nonradioactive Chemical Components by Aggregated DST Grouping in Metric Tons [1, 2]

Chemical Species	DST Groupings						Total	
	5EDS		3EDS		3WDS			
	Soluble	Insoluble	Soluble	Insoluble	Soluble	Insoluble	Soluble	Insoluble
Ag^{+}	1.18E-01	1.32E+00	2.10E-01			5.80E-02	3.28E-01	1.38E+00
Al^{+3}		4.29E+01		1.05E+01		1.44E+01		6.78E+01
As^{+5}	7.44E-01	2.97E-01	2.60E-02			2.01E-01	7.70E-01	4.98E-01
B^{+3}	1.01E-01	9.17E-01	6.64E-02		3.52E-01	7.66E-02	5.19E-01	9.94E-01
Ba^{+2}	7.01E-01	2.41E+00	6.16E-02	6.80E-01	2.93E-02	3.26E-03	7.91E-01	3.09E+00
Be^{+2}	5.97E-02	7.47E-03	2.22E-02	1.39E-04			8.19E-02	7.61E-03
Bi^{+3}	1.74E+00		5.21E-01				2.26E+00	
Ca^{+2}	5.19E+00	7.96E+00	6.02E-01	1.79E+00	4.48E+00	1.74E+00	1.03E+01	1.15E+01
Cd^{+2}	9.68E-02	5.77E+00	7.05E-02			2.42E-01	1.67E-01	6.01E+00
Ce^{+3}	2.26E-02	2.78E+00				2.64E-01	2.26E-02	3.04E+00
Cr^{+3}		9.31E+00		7.84E-01		2.40E+01		3.41E+01
Cu^{+2}	7.32E-02	4.05E-01	1.04E-01			3.42E-01	1.77E-01	7.46E-01
Fe^{+3}	3.10E+00	1.32E+02	8.18E-01	1.92E+00	4.17E+00	8.27E+00	8.09E+00	1.42E+02
Hg^{+}	5.75E-02		9.43E-04				5.84E-02	
K^{+}	1.48E+02	1.79E+00	3.60E+02	1.80E+01	3.72E+01	3.76E-01	5.46E+02	2.02E+01
La^{+3}	2.19E-01	1.96E+01		1.35E+00		9.89E-02	2.19E-01	2.10E+01
Li^{+}	2.13E-03	2.26E-02			3.64E-03	1.96E-03	5.77E-03	2.46E-02
Mg^{+2}	5.26E-01	9.53E+00	3.92E-01	1.00E+00	4.67E-02	4.62E-01	9.65E-01	1.10E+01
Mn^{+4}	6.11E+00	1.48E+01	1.53E-01	9.50E-01	1.42E+00	2.29E+00	7.69E+00	1.80E+01
Mo^{+6}	3.82E+00	2.09E-01	2.72E-01	5.25E-01	7.85E-01	6.71E-02	4.87E+00	8.01E-01
Na^{+}	6.00E+03	6.51E+01	2.80E+03	1.36E+02	2.13E+03	2.85E+01	1.09E+04	2.30E+02
Ni^{+2}	3.34E+00	5.03E+00	1.81E-01	4.30E-01	5.54E-01	1.11E+00	4.07E+00	6.57E+00
Pb^{+4}	5.63E-01	2.95E+00	1.39E+00			3.34E-01	1.96E+00	3.28E+00
SiO_3^{-2}	1.03E+01	1.99E+02	5.13E+00	9.32E+00	7.15E-02	6.01E+00	1.55E+01	2.14E+02
UO_2^{+2}	2.19E+00	7.91E-01	1.36E+00	2.60E+01			3.54E+00	2.68E+01
V^{+5}	6.20E-02	8.09E-03		1.66E-01		1.37E-02	6.20E-02	1.88E-01
W^{+6}	7.47E-01						7.47E-01	
Zn^{+2}	3.50E-01	4.20E-01	2.14E+00	2.80E-01	1.10E+00	2.45E-01	3.59E+00	9.45E-01
Zr^{+4}	2.30E-01	1.85E+01	2.18E-01	2.58E+02		1.85E-01	4.48E-01	2.77E+02
$Al(OH)_4^{-}$	2.44E+03		8.34E+02		1.19E+03		4.47E+03	
CO_3^{-2}	9.41E+02	5.17E+01	8.43E+02	3.00E+00	8.39E+01	3.61E+00	1.87E+03	5.83E+01
Cl^{-}	1.47E+02	5.75E-01	5.24E+01	3.60E-02	7.41E+01	8.77E-01	2.73E+02	1.49E+00

Table A.2.1.7 Estimated Mass of Nonradioactive Chemical Components by Aggregated DST Grouping in Metric Tons [1, 2] (cont'd)

Chemical Species	DST Groupings						Total	
	5EDS		3EDS		3WDS			
	Soluble	Insoluble	Soluble	Insoluble	Soluble	Insoluble	Soluble	Insoluble
$CrOH_4^{-2}$	4.06E+01		1.14E+01		6.01E+01		1.12E+02	
F^-	3.56E+01	6.39E-01	3.02E+02	1.81E+01	8.63E+00	3.63E-01	3.46E+02	1.91E+01
SO_4^{-2}	2.77E+02	1.96E+00	7.10E+01	1.25E+00	3.83E+01	3.46E+00	3.86E+02	6.68E+00
NO_3^-	4.43E+03	2.04E+01	2.19E+03	8.28E+00	1.03E+03	1.04E+01	7.65E+03	3.91E+01
NO_2^-	1.94E+03	5.06E+00	8.88E+02	8.94E-01	2.44E+02	2.46E+00	3.07E+03	8.42E+00
PO_4^{-3}	7.13E+01	1.40E+01	7.44E+01	4.57E-01	6.63E+01	7.16E+00	2.12E+02	2.16E+01
OH^-	9.99E+02	2.45E+01	6.97E+02	5.96E+01	2.03E+02	3.87E+01	1.90E+03	1.23E+02
TOC	8.08E+02	6.25E+01	5.48E+01	4.60E+00	1.28E+02	1.29E+00	9.90E+02	6.84E+01

Notes:

[1] Minor differences exist between the aggregated totals for some chemical species and the total quantities reported in Table A.2.1.2. These differences are a result of updates made to the overall inventory that are not currently reflected in the aggregated inventory.

[2] Values with three or more significant digits do not imply a specific accuracy of the stated value; no data entry provided for insignificant inventories.

Table A.2.1.8 Estimated Radionuclide Inventory for Aggregated DST Groupings in Curies [1, 2, 3]

Radionuclide	DST Groupings							
	3WDS		3EDS		5EDS		Total	
	Soluble	Insoluble	Soluble	Insoluble	Soluble	Insoluble	Soluble	Insoluble
C-14		8.06E-01	3.40E+02	1.98E+03	5.23E+00		3.45E+02	1.98E+03
Sr-90	2.16E+04	3.95E+04	6.41E+05	1.03E+07	1.34E+04	5.62E+04	6.76E+05	1.04E+07
Y-90	2.16E+04	3.95E+04	6.41E+05	1.03E+07	1.34E+04	5.62E+04	6.76E+05	1.04E+07
Tc-99	3.66E+03		1.54E+04	3.99E+02	1.60E+03		2.07E+04	3.99E+02
I-129	5.62E-01	1.71E+00	1.49E+01	1.43E+00	3.55E+00	1.59E-01	1.90E+01	3.30E+00
Cs-137	3.63E+06	3.91E+04	2.13E+07	5.44E+05	3.73E+06	1.66E+05	2.87E+07	7.49E+05
Ba-137m	3.63E+06	3.91E+04	2.13E+07	5.44E+05	3.73E+06	1.66E+05	2.87E+07	7.49E+05
Eu-154	4.34E+02	4.38E+00	7.14E+04	1.93E+03	5.33E+02		7.24E+04	1.93E+03
Np-237		3.79E-01	6.76E+00	3.83E+01	1.02E-01	3.98E-02	6.86E+00	3.87E+01
Pu-238		1.05E+03	1.65E+02	5.12E+00	1.40E+02	5.46E+01	3.05E+02	1.11E+03
Pu-239	2.02E+01	2.12E+03	1.22E+03	3.72E+03	6.27E+01	1.22E+03	1.30E+03	7.06E+03
Pu-240	5.06E+00	7.59E+02	3.07E+02	9.64E+02	1.57E+01	3.44E+02	3.28E+02	2.07E+03
Pu-241	9.47E+00	1.85E+04	7.43E+02	1.00E+04	2.86E+01	1.03E+04	7.81E+02	3.88E+04
Am-241	1.63E+02	1.17E+04	5.03E+03	5.37E+04	1.47E+02	3.51E+02	5.34E+03	6.58E+04
Total	8.60E+06	2.11E+05	4.99E+07	2.18E+07	1.01E+07	5.93E+05	5.88E+07	2.24E+07

Notes:

[1] Minor differences exist between the aggregated totals for some radionuclides and the total quantities reported in Table A.2.1.3. These differences are a result of updates made to the overall inventory that are not currently reflected in the aggregated inventory.

[2] Radionuclides reflect decay to 12/31/95 and were back calculated from 12/31/99 data. No decay chains were used in back calculating inventories.

[3] Values with three or more significant digits do not imply a specific accuracy of the stated value; no data entry provided for insignificant inventories.

Table A.2.2.1 Characteristics of Existing Capsules

Characteristics	Strontium (601 capsules) [1]				Cesium (1,328 capsules) [1]			
	As filled	Dec. 31, 1994	Dec. 31, 1999	Dec. 31, 2019	As filled	Dec. 31, 1994	Dec. 31, 1999	Dec. 31, 2019
Cumulative (MCi)	32.66	23.01	20.40	12.70	73.90	53.40	47.40	29.90
Cumulative (kW)	220.80	154.20	136.90	85.10	355.30	256.40	228.40	143.90
Average (kCi)	54.36	38.47	34.14	21.16	55.70	40.10	35.75	22.58
Average (W)	367.43	260.07	230.78	143.08	267.60	192.59	171.69	108.44
Highest curies loading (kCi)	146.60	93.27	82.76	51.31	74.50	54.38	48.48	30.62

Notes:

[1] The values for megacuries and highest curies loading reflect only parent radionuclide activity for the Sr-90/Y-90 decay chain and the Cs-137/Ba-137 decay chain.

Table A.2.3.1 Inactive MUSTs Estimated Current Waste Volumes in Liters [1]

Tank Designation	Nominal Tank Capacity	Solids Volume	Liquid Volume	Total Waste Volume	Comments
231-W-151-001	15,000	0	5,400	5,400	Settling tank, O/S since 1974
231-W-151-002	3,600	40	3,600	3,600	Settling tank, O/S since 1974
241-Z-8	58,000	1,900	0	1,900	Settling tank, O/S since 1962
241-A-302B	51,000	no data	no data	13,600	Interim Isolated. Monitored
241-B-301	136,000	82,000	2,200	84,000	Catch tank, O/S since 1984
241-B-302B	67,000	2,600	16,000	18,600	Catch tank, O/S since 1985
241-BX-302A	67,000	3,200	0	3,200	Catch tank, O/S since 1985
241-BX-302B	43,000	3,600	300	3,900	Catch tank, O/S since 1985
241-BX-302C	43,000	2,400	900	3,300	Catch tank, O/S since 1985
241-C-301	136,000	34,000	5,600	39,600	Catch tank, O/S since 1983
241-S-302A	67,000	no data	no data	19,000	Catch tank, O/S since 1991
241-S-302B	54,000	0	0	0	Emptied
241-SX-302	67,000	4,000	1,100	5,100	Catch tank, O/S since 1983
241-T-301B	136,000	82,000	2,200	84,200	Catch tank, O/S since 1985
241-TX-302A	67,000	9,300	100	9,400	Catch tank, O/S since 1982
241-TX-302B	67,000	no data	no data	5,000	Catch tank, stabilized and isolated in 1954
241-TX-302BR	45,000	no data	no data	no data	Catch tank, contents unknown
241-TX-302XB	50,000	400	900	1,300	Catch tank, O/S since 1985
241-TY-302A	67,000	1,700	0	1,700	Catch tank, O/S since 1981
241-TY-302B	54,000	0	0	0	Emptied
244-BXR-001	190,000	27,000	0	27,000	Uranium recovery tank, O/S since 1957
244-BXR-002	57,000	6,800	1,400	8,200	Uranium recovery tank, O/S since 1957
244-BXR-003	57,000	5,500	1,400	6,900	Uranium recovery tank, O/S since 1957
244-BXR-011	190,000	26,000	400	26,400	Uranium recovery tank, O/S since 1956
244-TXR-001	190,000	8,700	200	8,900	Uranium recovery tank, O/S since 1956
244-TXR-002	57,000	11,000	0	11,000	Uranium recovery tank, O/S since 1956

Table A.2.3.1 Inactive MUSTs Estimated Current Waste Volumes in Liters [1] (cont'd)

Tank Designation	Nominal Tank Capacity	Solids Volume	Liquid Volume	Total Waste Volume	Comments
244-TXR-003	57,000	24,600	0	24,600	Uranium recovery tank, O/S since 1956
244-UR-001	190,000	7,000	1,500	8,500	Uranium recovery tank, O/S since 1957
244-UR-002	57,000	8,700	2,200	10,900	Uranium recovery tank, O/S since 1957
244-UR-003	57,000	5,900	0	5,900	Uranium recovery tank, O/S since either 1957 or 1976
244-UR-004	31,000	Minimal	Minimal	Minimal	Volumes unknown. Used for temporary storage of 60 percent nitric acid solutions.
244-ER-311A	N/R	N/R	N/R	N/R	Southwest of B Plant
241-AX-151	N/R	N/R	N/R	N/R	Diverter station with several tanks inside
216-TY-201	N/R	N/R	N/R	N/R	Flush tank, located east of TY Tank Farm
216-BY-201	N/R	N/R	N/R	N/R	Flush tank, located north of BY Tank Farm
242-TA-R1	N/R	N/R	N/R	N/R	Receiver tank for Z Plant
242-T-135	N/R	N/R	N/R	N/R	Outside the 242-T Evaporator, decontamination tank
243S-TK-1	N/R	N/R	N/R	N/R	Decontamination tank
213-W-TK-1	7,100	N/R	N/R	7,100	Decontamination tank, tank may not have received waste in the past.

Notes:

[1] Values with three or more significant digits do not imply a specific accuracy of the stated value.

N/R = Not Reported

O/S = Out of Service

Table A.2.4.1 Future Post Evaporator DST Waste Projections

Source Facility	Waste type	Volume (m^3)	Duration of Accumulation
PUREX: Deactivation waste	DN[1]	5,700	FY 1994 - FY 1997
B Plant: Terminal cleanout waste (concentrated)	DN	2,100	FY 1997 - FY 2001
100 Area: Terminal cleanout waste (concentrated)	DSSF[2]	2,200	FY 1995 - FY 1999
100 Area: Sulfate waste	DN	140	Not reported
300 Area: Fuel supply cleanout	DN	45	Not reported
105-F, 105-H: Basin cleanout	DN	850	Not reported
Tank 107-AN: Caustic addition	DN	190	Not reported
100-KE, 100-KW: Basin cleanout	DN	1,200	Not reported
TOTAL		12,400	

Notes:
[1] Dilute noncomplexed waste.
[2] Double-shell slurry feed.
FY = Fiscal Year

Table A.3.3.1 Comparison of Reported Quantities of Chemicals in Metric Tons

Chemical	SST - Sludge and Saltcake		DST - Soluble and Insoluble	
	LANL	Data Package	LANL	Data Package
Na^+	2.97E+04	4.97E+04	1.07E+04	1.43E+04
Al^{+3}	5.56E+03	1.99E+03	1.53E+03	6.78E+01
Fe^{+3}	2.59E+03	6.27E+02	2.23E+02	1.50E+02
Cr^{+3}	7.47E+02	8.63E+01	7.75E+01	3.43E+01
Bi^{+3}	6.63E+02	2.61E+02	1.05E+01	2.26E+00
La^{+3}	4.01E+01	0	4.75E-02	2.12E+01
Ce^{+3}	1.70E-04	2.35E+02	9.41E-01	3.07E+00
$ZrO(OH)_2$	0	3.81E+02 [1]	1.94E+02	4.29E+02 [1]
Pb^{+2}	1.32E+01	0 [2]	8.39E-01	5.24E+00 [2]
Ni^{+2}	2.10E+02	1.78E+02	2.22E+01	1.06E+01
Sr^{+2}	1.57E+02	3.60E+01	3.20E-02	0
Mn^{+4}	1.75E+01	1.20E+02	2.75E+01	2.57E+01
Ca^{+2}	5.36E+02	1.28E+02	1.08E+02	2.18E+01
K^{+1}	1.30E+02	0	2.05E+02	5.66E+02
OH^-	1.70E+04	5.07E+03	4.71E+03	2.45E+03
NO_3^-	4.30E+04	9.51E+04	1.04E+04	1.03E+04
NO_2^-	5.57E+03	3.53E+03	4.00E+03	4.18E+03
CO_3^{-2}	2.72E+03	1.56E+03	1.44E+03	1.98E+03
PO_4^{-3}	3.91E+03	4.53E+03	7.25E+02	3.51E+02
SO_4^{-2}	4.32E+03	1.65E+03	1.17E+03	3.93E+02
SiO_3^{-2}	3.73E+02	1.21E+03	7.73E+01	2.29E+02
F^-	5.38E+02	8.00E+02	3.59E+02	3.71E+02
Cl^-	3.48E+02	4.00E+01	2.74E+02	2.74E+02
$Fe(CN)_6^{-4}$	1.39E+02	3.22E+02	0	0

Notes:
[1] Converted from Zr^{+4}.
[2] Shows as Pb^{+4}.
LANL = Los Alamos National Laboratory

Table A.3.3.2 Comparison of Reported Quantities of Radionuclides

Element	SST - Sludge and Saltcake		DST - Soluble and Insoluble	
	LANL	Data Package [1]	LANL	Data Package
Pu (MT)	5.37E-01	3.10E-01 [2]	1.59E-01	1.46E-01 [2]
U (MT)	1.70E+03	1.43E+03 [2]	1.60E+02	0
Cs (Ci)	3.79E+07	8.12E+06	2.82E+07	2.68E+07
Sr (Ci)	3.85E+07	4.36E+07	2.13E+07	1.01E+07

Notes:
[1] Total = sludge + saltcake + liquid (assumed saltwell pumping is completed).
[2] Converted from curies.
LANL = Los Alamos National Laboratory

APPENDIX A
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Appendix B
Description of Alternatives

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ACRONYMS AND ABBREVIATIONS

APM	ammonium phosphomolybdate
CFR	Code of Federal Regulations
CMPO	N-diisobutylcarbamoylmethylphosphine oxide
DOE	U.S. Department of Energy
DST	double-shell tank
DTPA	diethylenetriaminepentaacetic acid
Ecology	Washington State Department of Ecology
EDTA	ethylenedianimetetraacetic acid
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
FR	Federal Register
HEDTA	hydroxyethylanediamivetriacetic acid
HEPA	high-efficiency particulate air
HI	hazard index
HLW	high-level waste
HMPC	Hanford Multi-Purpose Canister
HVAC	heating, ventilating, and air conditioning
LAW	low-activity waste
MUST	miscellaneous underground storage tank
NCRW	neutralized cladding removal waste
NEPA	National Environmental Policy Act
NRC	U.S. Nuclear Regulatory Commission
PFP	Plutonium Finishing Plant
PUREX	Plutonium-Uranium Extraction
RCRA	Resource Conservation and Recovery Act
REDOX	Reduction-Oxidation
SIS	Safe Interim Storage
SST	single-shell tank
TBP	tributyl phosphate
TFCF	tank farm confinement facility
Tri-Party Agreement	Hanford Federal Facility Agreement and Consent Order
TRUEX	transuranic extraction
TSLCC	Total System Life Cycle Cost
TWRS	Tank Waste Remediation System
WAC	Washington Administrative Code
WESF	Waste Encapsulation and Storage Facility
WHC	Westinghouse Hanford Company
WNP-2	Washington Public Power Supply System Nuclear Plant 2

NAMES AND SYMBOLS FOR UNITS OF MEASURE, RADIOACTIVITY, AND ELECTRICITY/ENERGY

Length

cm	centimeter
ft	foot
in	inch
km	kilometer
m	meter
mi	mile

Area

ac	acre
ft^2	square foot
ha	hectare
km^2	square kilometer
mi^2	square mile

Volume

cm^3	cubic centimeter
ft^3	cubic foot
gal	gallon
L	liter
m^3	cubic meter
ppb	parts per billion
ppm	parts per million
yd^3	cubic yard

Mass

g	gram
kg	kilogram
lb	pound
mg	milligram
mt	metric ton

Radioactivity

Ci	curie
MCi	megacurie (1.0E+06 Ci)
mCi	millicurie (1.0E-03 Ci)
μCi	microcurie (1.0E-06 Ci)
nCi	nanocurie (1.0E-09 Ci)
pCi	picocurie (1.0E-12 Ci)

Electricity/Energy

A	ampere
J	joule
kV	kilovolt
kW	kilowatt
MeV	million electron volts
MW	megawatt
V	volt
W	watt

Temperature

°C	degrees Centigrade
°F	degrees Fahrenheit

APPENDIX B
DESCRIPTION OF ALTERNATIVES

This appendix describes the Hanford Site tank waste and cesium (Cs) and strontium (Sr) capsules and the alternatives that are addressed in this Tank Waste Remediation System (TWRS) Environmental Impact Statement (EIS). The detailed description of each alternative includes 1) an overview of the alternative; 2) the facilities to be constructed; and 3) the processes involved. Because certain post-remediation activities are common to each alternative, they are discussed in a separate section followed by a discussion of specific applicable technologies that are not now but could be included in the alternatives. The appendix concludes with a compilation of comparison data such as resource requirements, staffing requirements, and emissions data for each alternative.

This appendix provides specific details about the alternatives to supplement the more general discussion in Volume One, Section 3.0 of the EIS. The data used in generating this appendix came from a series of engineering data packages that were compiled separately by Westinghouse Hanford Company (WHC) and Jacobs Engineering Group Inc. (WHC 1995a, c, d, e, f, g, h, i, j, n and Jacobs 1996).

B.1.0 EXISTING FACILITIES AND OPERATIONS

B.1.1 TANK WASTE

From 1943 to 1988, the primary purpose of the Hanford Site was to produce weapons-grade plutonium (Pu) and other defense-related material to support the national defense mission. Plutonium production occurred in a nuclear reactor when a uranium-238 (U-238) atom in a fuel rod absorbed a neutron released from the splitting of another atom. After the fuel rods spent the required length of time in the reactor, the fuel was removed and processed to recover the Pu. The first processes to recover Pu were developed to exclusively separate Pu from the other elements in the fuel rods. Later, processes were developed to also recover U, which was then recycled back into the reactor fuel process. Processing fuel elements involved performing chemical separations to isolate and recover the Pu and U from the spent fuel elements. Chemical waste, the by-product of these separations, created the need for large-capacity, onsite storage. The tank farms, which are a group of interconnected underground storage tanks, were designed and built to accommodate the chemical waste. The first 149 storage tanks built were single-shell tanks (SSTs), which are reinforced-concrete tanks with a single steel tank. The last 28 tanks built were double-shell tanks (DSTs), which are reinforced-concrete tanks with two steel tanks. The locations of the tanks are shown in Figures B.1.1.1 and B.1.1.2.

Chemical separations processing generated approximately 1.5E+09 liters (L) (4.0E+08 gallons [gal]) of waste. More than 1.1E+09 L (3.0E+08 gal) of waste was sent to the SSTs and DSTs throughout the production period. Volume reduction practices were used to maintain waste volumes within the available tank space. Through liquid evaporation, waste concentration, and decanting (liquid removal following solids settling) dilute waste to the ground, this waste volume has been reduced to approximately 2.1E+08 L (5.6E+07 gal) (Hanlon 1996). The decanting, or discharging, of settled

Figure B.1.1.1 Hanford Site

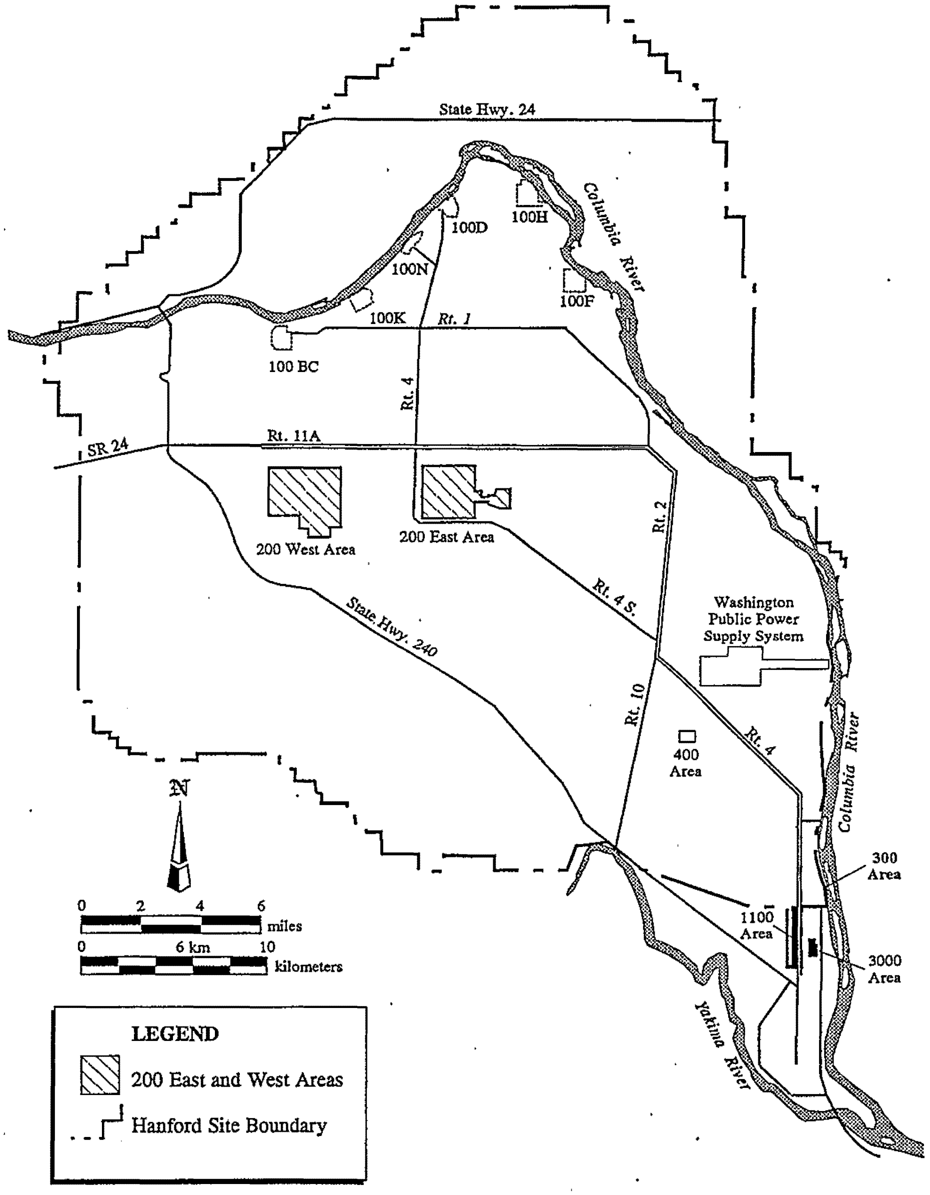
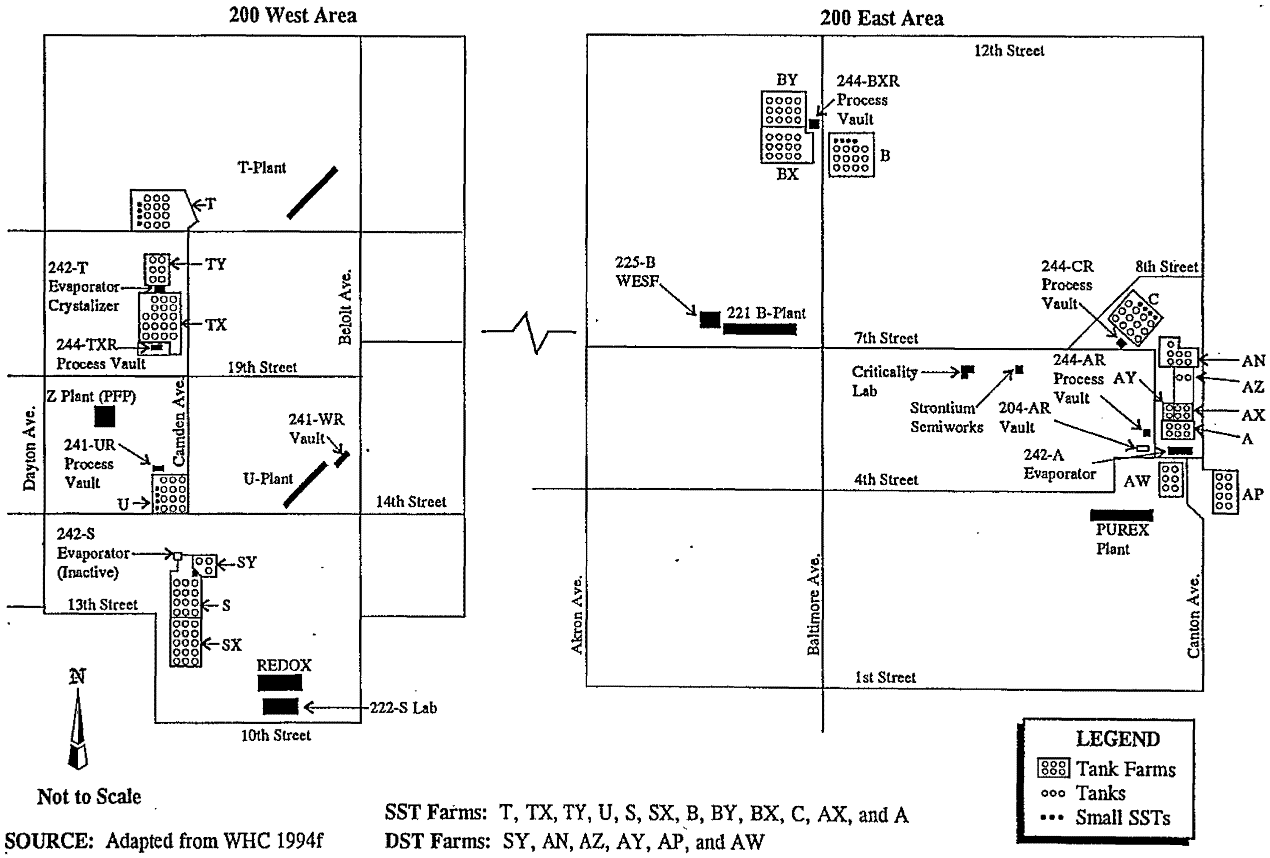


Figure B.1.1.2 Tank Farm Locations in 200 East and West Areas



SST Farms: T, TX, TY, U, S, SX, B, BY, BX, C, AX, and A
DST Farms: SY, AN, AZ, AY, AP, and AW

SOURCE: Adapted from WHC 1994f

SST waste to the ground was stopped in 1966; no tank waste from SSTs or DSTs has been intentionally discharged to the ground since that time. Liquid discharged to the ground was sent to cribs (drain fields) to drain into the soil. This practice resulted in soil and groundwater contamination. These cribs are past practice units that are not within the scope of the TWRS EIS.

Underground transfer lines (pipelines) transferred liquid waste from the processing plants to the tank farms. Routing the liquid waste from a plant to a specific tank farm was controlled by valve pits and diversion boxes. Diversion boxes, which are concrete-walled pits located in the ground with a removable top at ground level, allowed a jumper or spool piece to be installed to control the routing of the waste and minimize the number of pipelines. After the waste transfer was completed, the volume change in the tank was logged for future reference.

B.1.1.1 Description of Single-Shell Tanks

The SSTs were the first large volume tanks constructed. The 149 SSTs at the Hanford Site vary in size from 2.1E+05 to 3.8E+06 L (5.5E+04 to 1.0E+06 gal). Figure B.1.1.3 shows a typical SST. The SSTs consist of a reinforced-concrete shell surrounding a carbon steel tank. Each of the larger tanks has multiple access points called risers that provide access to the tank from the surface. The risers are either sections of pipe or square concrete pits that connect to the top of the tanks, which are 1.8 to 2.5 meters (m) (6 to 8 feet [ft]) below grade. The risers, between 10 and 110 centimeters (cm) (4 and 42 inches [in.]) wide, are used for monitoring instruments, camera observation, tank ventilation systems, and sampling. Wells drilled into the ground around the tanks are used for monitoring and detecting leaks in the SST farms.

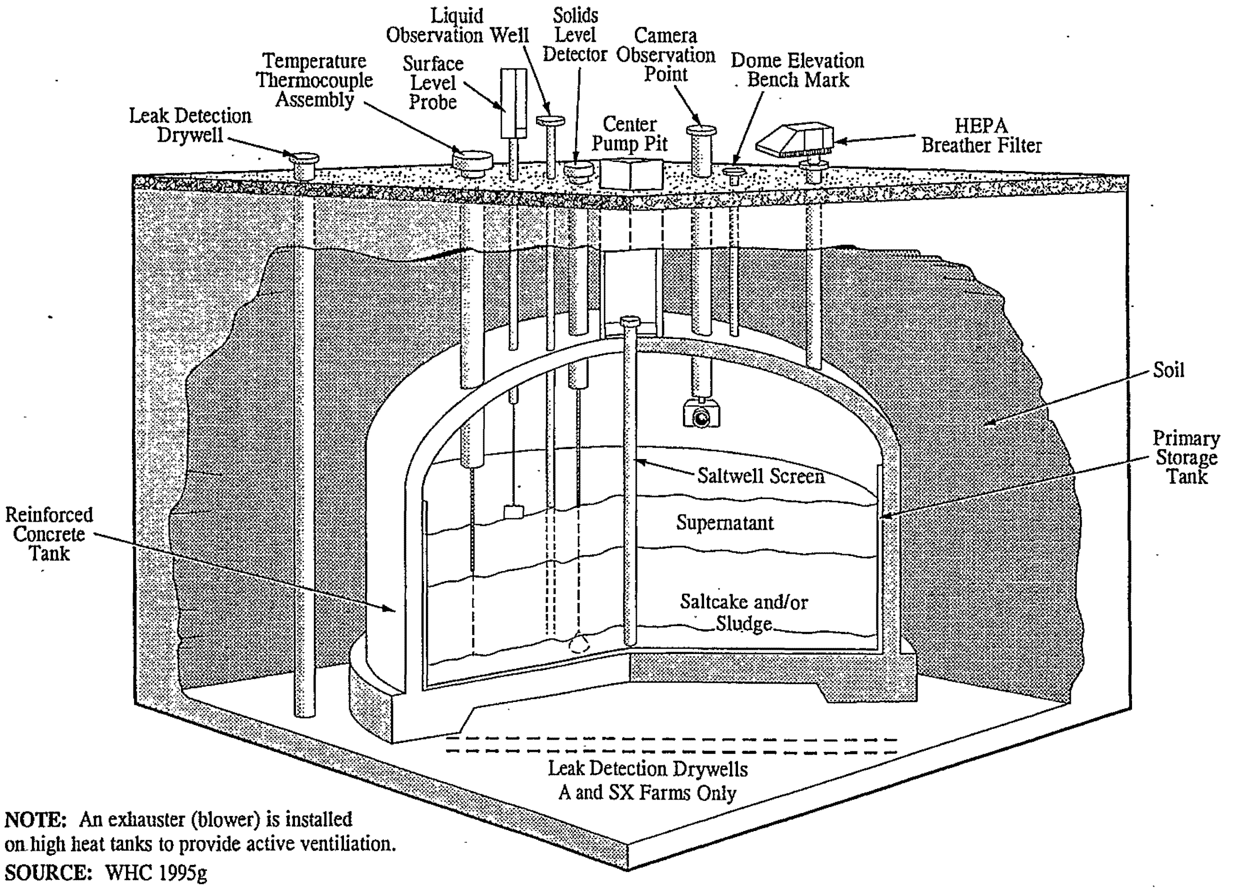
The sizes and quantities of SSTs that were built in the 200 Areas are shown in Table B.1.1.1.

Table B.1.1.1 Single-Shell Tank Summary

Quantity	Capacity in liters (gallons)	Size in meters (feet)
16	2.1E+5 L (55,000 gal)	6.1 m diameter by 7.9 m high (20 ft diameter by 26 ft high)
60	2.0E+6 L (530,000 gal)	22.9 m diameter by 9.1 m high (75 ft diameter by 30 ft high)
48	2.9E+6 L (758,000 gal)	22.9 m diameter by 11.3 m high (75 ft diameter by 37 ft high)
25	3.8E+6 L (1,000,000 gal)	22.9 m diameter by 14.6 m high (75 ft diameter by 48 ft high)

The tank farms are located close to the center of the 1,450-square-kilometer (km^2) (560-square-mile [mi^2]) Hanford Site (shown in Figure B.1.1.1) in the 200 Areas. The 200 Areas are specific areas of operation and are divided into the 200 East Area and the 200 West Area, which are approximately equal in size. The tank farms are approximately 8 kilometers (km) (5 miles [mi]) from the Columbia

Figure B.1.1.3 Single-Shell Tank Configuration



NOTE: An exhauster (blower) is installed on high heat tanks to provide active ventilation.

SOURCE: WHC 1995g

River at their closest point. There are 66 SSTs in the 200 East Area and 83 SSTs in the 200 West Area. These tanks are arranged in groups called farms, which range from 4 to 18 tanks. Building the tanks in farms allowed the tanks to be interconnected within the farm, thereby reducing the number of pipelines between the processing plants and the tank farms. The tank farm concept also allowed the use of cascades, in which the first tank overflowed into the second tank, the second into the third, and so on within the tank farms to allow solids settling. The solids contain a majority of the radionuclides, except for Cs-137, iodine-129 (I-129), and technetium-99 (Tc-99), which are more prevalent in the liquid phase.

The 200 West Area has 83 SSTs in six tank farms. These tanks supported operations of T Plant, U Plant, the Plutonium Finishing Plant (PFP), and the Reduction-Oxidation (REDOX) Plant as described in Section B.1.1.6.

There are 66 SSTs in the 200 East Area associated with operations of the Plutonium-Uranium Extraction (PUREX) Plant and B Plant. North of the PUREX Plant are three tank farms with a total of 26 tanks. North of B Plant on the northern edge of the 200 East Area are three tank farms with a total of 40 SSTs.

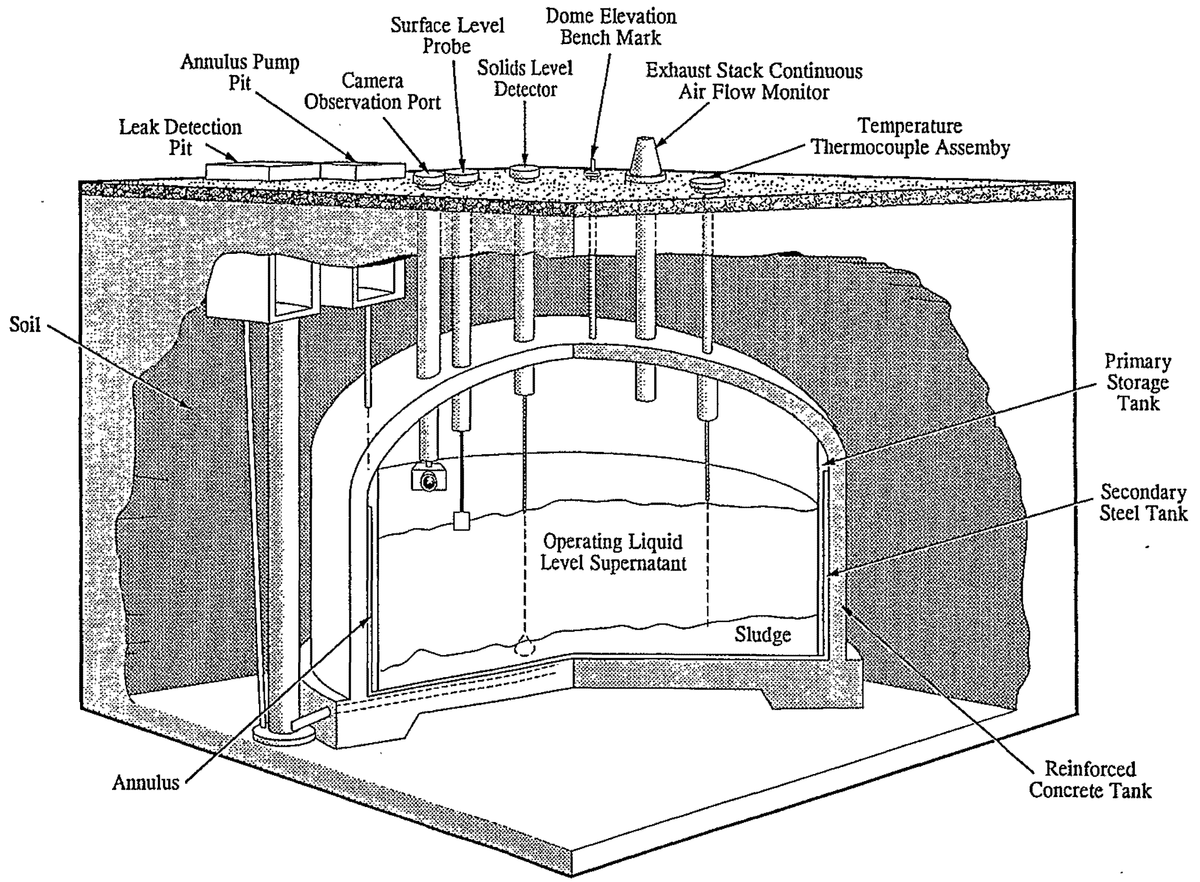
B.1.1.2 Description of Double-Shell Tanks

The DSTs were developed as a design improvement over the SSTs. The DSTs have double-carbon steel tanks inside a reinforced-concrete shell, as shown in Figure B.1.1.4. There is an annulus or space between the two steel tanks with equipment to detect and recover waste in the event that the inner tank develops a leak. Each tank has multiple risers connecting the tank with the surface above. These risers are different diameters or sizes depending on their intended use. The risers for the DSTs are used for the same purpose as the SST risers (i.e., monitoring instruments, camera observation tank ventilation systems, and sampling). Each DST tank has its own leak detection pit that is connected to the bottom of the tank and monitored for tank leaks.

The DSTs are approximately 23 m (75 ft) in diameter, 15 m (48 ft) tall, and cylindrical in shape with a concrete-domed top. All of the tanks are buried in the ground with the tops of the domes located approximately 2 m (7 ft) below the surface. Twenty-four of the tanks have a capacity of 4.4E+06 L (1.2E+06 gal), while four of the tanks have a capacity of 3.8E+06 L (1.0E+06 gal). Tank farm operations restrict the total volume allowed in the tanks to approximately 76,000 L (20,000 gal) below maximum capacity.

The 25 DSTs in the 200 East Area are located just north of the PUREX Plant. Twenty-one of the DSTs have a capacity of 4.4E+06 L (1.2E+06 gal) and four of the DSTs have a capacity of 3.8E+06 L (1.0E+06 gal). The 21 DSTs in service are operating between 75 and 97 percent of allowable volume capacity. The waste in the DSTs is primarily liquid with small volumes of sludges and saltcakes. There have been no leaks from the DSTs.

Figure B.1.1.4 Double-Shell Tank General Configuration



B.1.1.3 Miscellaneous Underground Storage Tanks

In addition to the 177 underground storage tanks previously discussed, there are approximately 20 active and 40 inactive miscellaneous underground storage tanks (MUSTs). The EIS alternatives also address the disposition of the waste in the MUSTs. The inactive MUSTs were used during processing and waste transfer operations and were not intended for use as long-term storage tanks. The MUSTs that were used primarily for solids settling, adding caustic, and catch tanks are also currently inactive. The characteristics of the waste contained in the inactive MUSTs is expected to be similar to the SST waste. The active MUSTs still are used as receiver tanks during waste transfer activities or as catch tanks to collect potential spills and leaks.

Most of the inactive MUSTs were interim stabilized and isolated before September 1985. The MUSTs range in size from 3,400 to 190,000 L (900 to 50,000 gal). There is a wide range in the amount of waste currently in the MUSTs. While most of the MUSTs are empty or nearly empty, several inactive MUSTs contain residual sludges and liquid. The volume of waste in all the MUSTs combined is less than one-half of 1 percent of the total tank inventory (WHC 1995n).

B.1.1.4 Existing Transfer Lines

When the tank farms were constructed, they were connected to the process facilities by underground transfer lines. Associated with these transfer lines are subgrade valve pits and diversion boxes. Valve pits and diversion boxes provide a means to route waste to specific tank farms with a minimum number of transfer lines. In addition, there is an existing cross-site transfer system to transfer waste between the 200 East and 200 West Areas. Some of the older transfer lines are blocked or plugged up and cannot be used for waste transfers. All of the existing transfer lines are buried below grade to use the natural radiation shielding of the ground. Most of the transfer lines installed during early operations are single-wall carbon-steel pipe lines, while later lines are double-wall pipe lines with a stainless-steel inner pipe encased in an outer carbon-steel pipe. The valve pits and diversion boxes are below grade concrete structures that are covered with removable concrete panels. A new replacement cross-site transfer system is under construction and scheduled to begin operations in 1998.

B.1.1.5 Support Facilities

Support facilities provide utilities and other operations to help manage the tanks and tank waste. The following is a list of the primary existing support facilities required to continue managing the tank waste.

- Steam is provided by the 284-East Steam Plant. The Steam Plant was built in 1943 with a design life of approximately 20 years. The boilers operate below capacity and require a high level of maintenance.
- Water, both sanitary and process, is delivered to the 200 Areas by the Hanford Site Water System.
- Electrical power is delivered to the Hanford Site by the Bonneville Power Administration. The 200 Areas have one substation with two independent transformers.

- Road and rail access is established to the 200 Areas.
- Tank waste and new waste undergo evaporation at the 242-A Evaporator to reduce waste volume requiring storage. The 242-A Evaporator has recently been upgraded.
- Evaporator condensate is treated at the Effluent Treatment Facility to remove contaminants before being discharged.

B.1.1.6 Tank Waste

Sources of the Waste

Several different chemical separations processes were used in the past for separating and recovering Pu and U from irradiated reactor fuels at the Hanford Site. Common steps to the different recovery processes included chemically removing the fuel element cladding, dissolving the fuel in nitric acid, chemically processing the fuel to separate the Pu, and in some instances separating the U from the dissolved fuel mixture.

The first processing for Pu recovery started in 1944 at T Plant and 1945 at B Plant using the bismuth phosphate process. Both plants used bismuth phosphate to precipitate Pu from dissolved spent fuel solutions. The extraction waste was classified as a metal waste and contained 90 percent of the fission products and 99 percent of the U. This waste was sent to specific SST tank farms in the 200 East and 200 West Areas.

In January 1952, the REDOX Plant began operating as the world's first nuclear solvent extraction plant using the REDOX process. The REDOX process extracted Pu and U into a hexone solvent in a continuous solvent extraction process.

In January 1956, the PUREX Plant began operating. PUREX used tributyl phosphate (TBP) in a kerosene base as a solvent to extract U and Pu from the fuel elements that had been previously dissolved in a nitric acid solution. Both the REDOX and PUREX process recovered Pu, U, and neptunium (Np) from spent reactor fuel.

All of the acidic aqueous waste was made alkaline by adding sodium hydroxide or calcium carbonate before storing in the underground storage tanks.

The PFP took the plutonium nitrate product from PUREX Plant and REDOX Plant and further refined it into Pu metal. The PFP used a process similar to PUREX Plant to further purify the Pu and produce a finished Pu product from the PUREX Plant output. The PFP sent waste to the tank farms that was low in radioactivity and high in metallic nitrates. Before PFP was operating, the plutonium nitrate paste was transported to Los Alamos National Laboratories for processing.

Because U was not recovered in the bismuth phosphate process it was sent to the tank farms during B Plant and T Plant operations. The U Plant was built and operated to recover the U from B and T

Plant tank waste. This U recovery operation required the recovery of B and T Plant waste from the tank farms.

Midway through U Plant operations the process of scavenging or precipitating Cs with ferrocyanide was started to remove the Cs from the liquid waste. This scavenging operation precipitated the Cs in the tanks as solids, allowing the liquid to be decanted and sent to the cribs. This practice allowed for the discharge of clarified liquid and provided additional tank space. This process was completed in 1957 (WHC 1995b).

B Plant was also operated as a waste fractionization plant in the 1960's to early 1980's. Cesium and Sr were recovered as waste by-product and the secondary waste containing complexants (ethylenediaminetetraacetic acid [EDTA] and hydroxyethylenediaminetriacetic acid) were sent to the tank farms.

As a result of using the tanks to hold waste from such a variety of operations, the tank contents have changed as time passed. While records were kept as transfers were made, the inter-tank piping allowed the tank contents to cascade from one tank to another. Consequently, the tanks now contain a variable mixture of sludge, precipitated salts (saltcake), and liquid. Characterization on a tank-by-tank basis would be required to determine the actual contents of any given tank.

Waste Types

The waste stored in DSTs is reported by waste type stored in individual tanks. There are seven waste types associated with DSTs.

- Concentrated complexant waste is concentrated product from evaporating dilute complexed waste.
- Concentrated phosphate waste is waste originating from the decontamination of the N Reactor in the 100-N Area.
- Dilute complexed waste is characterized by a high content of organic carbon including organic complexants. The main source of dilute complexed waste in the DSTs is the liquid-removal operations from the SSTs.
- Dilute noncomplexed waste is low-activity liquid waste.
- Double-shell slurry is waste that exceeds the sodium aluminate saturation boundary in the evaporator without exceeding receiver tank composition limits.
- PUREX Plant neutralized cladding removal waste (NCRW) is the solids portion of the PUREX Plant NCRW. This NCRW waste was sent to the tank farms as a slurry and is classified as transuranic (TRU) waste.
- PFP TRU solid is solid TRU waste from PFP operations.

B.1.1.7 Current TWRS Activities

The TWRS program was established in 1991 to safely manage and dispose of radioactive and chemical or mixed waste that has been generated at the Hanford Site. The current TWRS program mission is to dispose of the radioactive tank waste (includes current and future tank waste) and the Sr/Cs capsules in an environmentally sound, safe, and cost-effective manner.

Continued Operations of Tank Farm System

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, drywell monitoring around tanks for increases in radioactivity levels, and groundwater monitoring.

TWRS 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. This program includes job preplanning and identification of new technologies such as low volume hazardous waste decontamination practices to limit the generation of new waste. A waste minimization support program for non-TWRS waste generators is used to encourage waste minimization practices;
- 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 treating evaporator condensate at the Effluent Treatment Facility.

These activities are not within the scope of this EIS because they were addressed in previous National Environmental Policy Act (NEPA) documents: the Safe Interim Storage of Hanford Tank Waste EIS (SIS EIS) (DOE 1995i), Waste Tank Safety Program Environmental Assessment (DOE 1993h), Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes EIS (DOE 1987).

Tank Monitoring and Maintenance

As part of its routine operations, the Hanford Site has an extensive tank farm surveillance program in which tanks are monitored for temperature, surface level, and interstitial liquid level (in tanks having low-activity waste [LAW]) as required to safely manage and operate the tank farms. There are pressure and gas monitors on some tanks. The surface level inside the tanks is monitored either manually with an installed tape or with automated instrumentation.

Watchlist tank temperatures are monitored with automated equipment where installed, and manually where required. The automated systems allow temperature monitoring on a continuous basis. Watchlist tanks that require manual readings are done on a weekly or monthly basis. All Watchlist tanks are reviewed for increasing temperature trends. Non-Watchlist tank temperatures are monitored at 6-month intervals.

Fifty-eight of the SSTs and two of the DSTs have liquid observation wells installed for monitoring the level of interstitial liquid within the waste. Liquid observation wells are installed in SSTs that are known to have, or may have, greater than 1.95E+05 L (50,000 gal) of drainable liquid. The liquid observation wells are fiberglass or plastic pipe, sealed at the bottom, extending from the ground level down into the tank and through the waste to within 2.5 cm (1 in.) of the tank bottom. Gamma and neutron probes are used to monitor changes in the interstitial liquid level. Changes in liquid level would indicate fluid leakage either into or out of the tank, or could be an indication of the presence of gas within the waste if the observed liquid level changes are consistent with atmospheric pressure changes. The two steel liquid observation wells that are installed in the DSTs are used only for special monitoring purposes.

Radiation measurements are taken in the drywells surrounding the SSTs, in the leak detection pits, and the space between the liners of the DSTs. An increase in the radiation levels in any of the monitoring wells or pits would indicate a possible tank leak.

Safety Issues

All U.S. Department of Energy (DOE) facilities that store hazardous or radioactive materials have documented authorization bases that establish a range of operating parameters (e.g., temperature, pressure, concentration) within which routine operations are conducted. These authorization bases also evaluate the effects of potential accidents, abnormal events, and natural disasters.

The possibility of driving heavy equipment over an unstabilized tank during construction or operations, which potentially could result in a tank closure collapse was considered. To reduce the potential for this accident, engineered features would be installed and administrative controls used to prevent large vehicles from driving on top of the tank domes. These engineered barriers would be mechanical barriers such as closely spaced posts installed around the tanks or tank farms.

Watchlist Tanks

Concern over waste tanks having the potential for releasing high-level radioactive waste to the environment resulted in the passing of Public Law 101-510, Section 3137, Safety Measures for Waste Tanks at Hanford Nuclear Reservation, also known as the Wyden Amendment. In response to this law, DOE developed a set of criteria to identify tanks with potential safety concerns as Watchlist tanks. Current published information indicates that there are 50 Watchlist tanks, with 10 tanks listed in more than one of four different Watchlist categories based on specific safety concerns. The four different Watchlist categories include flammable gas, ferrocyanide, high organic content, and high-heat generation. The tanks in each category are shown in Table B.1.1.2 (Hanlon 1995 and Cowan 1996). As safety issues are resolved or mitigated, the number of tanks on the Watchlist is expected to change.

Table B.1.1.2 Watchlist Tanks

Watchlist Tank Category	Number of Tanks	Concern
Flammable Gas	25 [1]	Potential for explosion due to generation of flammable gas inside the tank dome above the flammability limit.
Ferrocyanide	14	Potential for explosion due to heat generation during postulated ferrocyanide oxidation reaction.
High Organic	20	Potential for explosion due to heat generation during postulated organic oxidation reaction, and/or for combustion of a separated lighter than water organic solvent phase.
High Heat	1	Failure of a tank due to loss of cooling water and subsequent high-heat induced structural damage.

Notes:

[1] Recently, internal Site controls have been applied to all 177 tanks for possible flammable gas generation/retention. A final determination of how many tanks represent a flammable gas issue has not been determined.

Source: Hanlon 1995

The flammable gas Watchlist identifies those tanks whose contents have the potential to generate/retain and release hydrogen gas at levels above the flammability limit, which is approximately 4 percent hydrogen by volume. Hydrogen and ammonia are generated within the tanks through radiolysis or radiation-induced decomposition and chemical reactions. If flammable concentrations are reached and an ignition source is present, the potential reaction could cause a radioactive release or provide an energy source to facilitate other reactions within the tank. Currently there are 25 hydrogen-generating tanks in this category. Tank 101-SY is currently being mitigated by using mixer pumps to stir the waste and allow hydrogen gas to be released gradually to prevent episodic releases of hydrogen that are above the lower explosive limit. Other tanks are being screened and evaluated to assess their magnitude of the risk from flammable gas generation, storage, and intermittent release.

The ferrocyanide Watchlist tanks are a concern because of the potential for self-propagating reactions if ferrocyanide in sufficient concentration comes in contact with an oxidizer (nitrates and nitrites) at a high temperature. The measured temperatures in all the ferrocyanide tanks are at or below 60 °C (140 °F), well below the 180 to 200 °C (360 to 390 °F) temperature required for self-propagating reactions to occur. The list of tanks with ferrocyanide was developed based on assessments of tank contents using process information. As tank characterization progresses, tanks with insufficient quantities of ferrocyanide for self-propagating reactions will be removed from the Watchlist. Currently, there are 14 tanks listed in this category.

There are 20 tanks in the high-organic Watchlist category. These are tanks that are estimated or have the potential to contain 3 percent total organic carbon on a dry weight basis. The concern with these tanks is that at elevated temperatures above 180 °C (360 °F), the organics in the tanks could result in self-propagating reactions with the nitrate and nitrite. These tanks are checked for the presence of an entrained or floating organic solvent layer that might pose a risk from a slow pooled or wicked fuel burn. Studies are underway to gain a better understanding of the high-organic safety issues. The differences between the measured tank temperatures and the temperatures required to sustain a reaction are large; therefore, the probability of a reaction is considered very low.

Currently one tank, tank C-106, is in the high-heat Watchlist category because of its content of heat-generating sludge. The heat generation is caused by decaying Cs and Sr in the sludge. The concern with the high-heat tank is that the heat-generating sludge could boil off or evaporate the liquid from the tank, which would raise the sludge temperature. If the temperature within the tank rises above the allowable limit for the tank materials, structural failure of the tank and collapse of the tank dome may result. While the tank currently is considered sound, water must be added periodically to keep the sludge wet and provide evaporative cooling.

Unreviewed Safety Questions

DOE has a formal administrative program to identify, communicate, and establish corrective actions for known or suspected operating conditions that have not been analyzed or that fall outside of the established authorization bases as an Unreviewed Safety Question. Following the identification of an Unreviewed Safety Question, a review is conducted, and corrective action is taken if applicable. Following the review process, the Unreviewed Safety Questions may be closed from an administrative standpoint, which means that conditions surrounding the safety issue have been analyzed. However, the conditions upon which the safety issue is based may still exist and may require mitigation, controls, or corrective action. In this way, safety issues and Unreviewed Safety Questions are related. The safety issues that were identified under the Watchlist program were also analyzed as Unreviewed Safety Questions. Those issues that had not been addressed in the documentation authorization basis were established as Unreviewed Safety Questions. Following the review processes, the Unreviewed Safety Question can be closed while the tank remains on the Watchlist for resolution of the safety issue. The Hanford Federal Facility Agreement and Consent Order (Tri-Party Agreement) (Ecology et al. 1994) requires the resolution of all Unreviewed Safety Questions by September 1998.

Technical evaluation and mitigative actions have resulted in closing the following Unreviewed Safety Questions: ferrocyanide (closed in March 1994); floating organic layer in tank C-103 (closed in May 1994); and criticality (closed in March 1994). Criticality was addressed on a tank farm basis and did not result in identifying any individual tanks to be added to the Watchlist tanks. Criticality would be an issue during tank waste retrieval and transfer, and would be evaluated on a tank-by-tank basis during final design. Closure of the Unreviewed Safety Questions was accomplished by defining the parameters (e.g., concentrations and temperature) of potential reactions that could lead to an uncontrolled release, collecting physical and chemical data on the waste, and establishing safety operating specifications.

The remaining Unreviewed Safety Questions are undergoing resolution. Mitigative action has been implemented for tank SY-101, the most widely known flammable-gas generating tank. This mitigative action involved installing a mixer pump to control the periodic release of flammable hydrogen gas and provide for more frequent and gradual releases of hydrogen. This mitigative action reduces the maximum concentration of flammable gas that can exist in the tank and greatly reduces the potential for an uncontrolled gas burn.

There is a safety screening and characterization program ongoing to determine if any additional tanks should be placed under special controls. Recently all 177 tanks, Watchlist and non-Watchlist, were placed under flammable gas controls, which means that flammable gas generation/retention may exist in all 177 tanks and special safety measures will be taken during maintenance, monitoring, and waste transfer activities. Until the necessary characterization data are obtained, the tank farm system will continue to operate under a conservative management program to maintain a safe operating envelope. Additional data may allow for relaxed operating procedures, where appropriate. Volume Four, Appendix E contains a more detailed description of the tank safety issue.

Interim Stabilization to Prevent Further Leakage

DOE removed all SSTs from service in November 1980 and initiated a program to remove all pumpable liquid and stabilize the tank waste until final disposition. This effort, known as interim stabilization, is currently ongoing. Approximately 30 tanks remain to be interim stabilized and these will be complete by the year 2000.

There are 67 confirmed or assumed leaking SSTs in the 200 Area tank farms. Over the years, these tanks have leaked an estimated 2.3E+06 to 3.4E+06 L (600,000 to 900,000 gal) of liquid to the soil column. All but five of the SSTs that are assumed leakers have been interim stabilized to minimize potential releases to the environment (Hanlon 1996).

An ongoing vadose zone characterization program that was initiated in April 1995 (DOE 1995t) is providing new baseline characterization data on the potential contaminant distribution in the vadose zone sediments beneath and in the vicinity of the SSTs. This has resulted in some recent information for the SX Tank Farm. The characterization effort relies on geophysical logging of existing drywells using a spectral gamma logging system with a high-purity intrinsic germanium detection device to provide assays of gamma-emitting radionuclides near the drywells (Brodeur 1996).

Ten of the 15 tanks in the SX Tank Farm are assumed or verified as leaking, as discussed in Volume Five, Appendix K. Ninety-five drywells ranging in depth from 23 m (75 ft) to 38 m (125 ft) from ground surface were logged with the Spectral gama logging system in the SX Tank Farm. The most abundant and highest concentration radionuclide detected was cesium-137, which was detected in "virtually every borehole" (Brodeur 1996). Cesium-137 was detected at the following depths in several drywells: 23 m (75 ft) in drywells 41-09-03 and 41-08-07, 32 m (105 ft) in 41-09-04, 27 m (90 ft) in 41-11-10, and 38 m (125 ft) in 41-12-02.

Other gamma-emitting radionuclides detected include cobalt-60, europium-152, and europium-154, which generally were found near the surface and are believed to be the result of spills (Brodeur 1996). Cobalt-60 was found in drywell 41-14-06 only. It was detected at a depth of 17 to 23 m (55 to 76 ft) below ground surface. The data are unclear as to whether relatively immobile contaminants such as cesium-137 would be found dispersed laterally within the vadose zone (i.e., at observed concentrations laterally several meters from the drywells) at the depths of over 30 m (100 ft) based on ambient conditions and vadose zone contaminant transport via advective flow in interstitial pore spaces. This suggests that there may be other transport mechanism(s) occurring such as those discussed in Volume Five, Section K.4.1.3. The viability of any other potential transport mechanism has not yet been demonstrated but is one of the objectives of the ongoing investigations.

Interim stabilization consists of saltwell pumping and is intended to reduce the volume of free waste liquid in the SSTs and minimize potential liquid losses to the environment. Interim stabilization is accomplished by reducing the supernatant liquid content of a tank to less than 190 m^3 (50,000 gal). The jet-pump system used to remove pumpable liquid continues operating until the pumping rate falls below 0.19 L/min (0.05 gal/min). The pumping effort may use the LR-56(H) cask truck for emergency pumping of leaking SSTs. Liquid removed from the SSTs is transferred to a DST. Interstitial liquid (within the solid pores) remains in the SSTs following interim stabilization. The 30 tanks that require saltwell pumping are scheduled to be completed by the year 2000.

Waste Characterization

The tank waste characterization process involves determining the physical, radiological, and chemical properties of the waste. Considerable historical data are available that have been used to estimate the contents of the storage tanks. Historical data, which are based on invoices for the purchase of chemicals and waste transfer and processing records, provide a basis for an overall inventory of the waste in the tanks. Historical tank content estimates have been completed for the DSTs and the solid waste in the SSTs (WHC 1995b). These estimates provide an inventory of the radioactive and mixed waste stored in the SSTs and DSTs.

Waste characterization is performed to help resolve safety issues, allow for the safe storage of the waste until waste treatment operations begin, and support planning and design decisions for implementing the remedial alternative selected. A considerable amount of inventory information is available from process records and past sampling activities. However, this information is not considered adequate to characterize the waste in individual tanks to support safety, treatment, and design activities.

There is an ongoing waste characterization program that is using waste sampling and analysis, in situ measurements, monitoring, surveillance, and waste behavior modeling to provide more detailed and accurate characterization data for the contents of each tank. Current agreements between DOE, the Washington State Department of Ecology (Ecology), and the U.S. Environmental Protection Agency (EPA) require that all characterization reports be issued by September 1999. Prior to disposal system final design, additional data requirements may be generated.

The tank waste is classified as liquid, sludges, or saltcake. Liquid is made up of water and organic compounds (e.g., solvents that are both heavier and lighter than water) with dissolved salts. Sludges are mixtures of insoluble (will not dissolve in tank liquid) metal salt compounds that settle out of solution after the waste is made alkaline for storage. A majority of the radioactive elements are contained in the sludges. However, radionuclides such as I-129, Tc-99, and Cs-137 are more prevalent in the liquid phase. Salts or saltcake are primarily sodium and aluminum salts that crystallize out of solution following evaporation. These three types of waste exist in the tanks in numerous combinations and proportions resulting in complex combinations of waste with varied physical and chemical properties. Sludges have been found with consistencies from mud to hardened clay. Layers of organic compounds have been found in some tanks floating on the top of solid waste, and crusts have formed in some tanks where a layer of solids has formed on top of the tank liquid.

Present data indicate that the SSTs as a group have on a volume basis 65 percent saltcake, 33 percent sludges, and 2 percent liquid, although the percentages of these differ greatly between tanks.
The DSTs have more than 77 percent liquid with 9 percent sludges, 10 percent double-shell slurry, and 4 percent saltcake (Hanlon 1995). These percentages may change as additional data become available and as waste transfers take place. SST and DST chemical inventory estimates, based on historical data, are provided in Volume Two, Appendix A of the EIS.

Evaporating Liquid in the 242A-Evaporator

The 242-A Evaporator is used to manage waste volume by evaporating the water from the tank waste. Recent evaporation campaigns have removed several million gallons of water from the tank waste. This water would be transferred to the Effluent Treatment Facility for treatment and release to the State-approved land disposal site. Following evaporation, concentrated waste would be returned to the DSTs.

B.1.1.8 Proposed TWRS Activities

Several tank waste activities are planned for implementation in the near future. These activities will address urgent safety or regulatory compliance issues.

Newly Generated Waste

At present, the DSTs are used to store waste generated from ongoing site activities. Future DST additions are expected to come from routine operations. These waste additions would involve loading the waste as liquid or slurry into a tank truck or railcar at the generating facility, transporting the waste to the tank farms, and unloading and transferring the waste into existing DSTs for storage. This waste would be transferred using existing rail or tanker truck systems. Section B.9.2 contains a description of the LR-56(H) truck. Facilities generating waste requiring transport to the tank farms include:

- 300 Area laboratory and facility cleanout;
- Cleanout waste from PUREX Plant, PFP, and B Plant;
- Decontamination waste from T Plant;
- Routine laboratory waste; and
- Cleanout of K Basins.

Additional information on newly generated waste is contained in Volume Two, Appendix A.

Safe Interim Storage

One issue that requires action is the safe storage of tank waste in the interim period before implementing actions for the permanent remediation of tank waste. To address this issue, the SIS EIS was prepared to consider alternatives for maintaining safe storage of Hanford Site tank waste (DOE 1995i). The actions considered in the SIS EIS include interim actions to 1) mitigate the generation of high concentrations of flammable gases in tank 101-SY; and 2) contribute to the interim stabilization of older SSTs, many of which have leaked.

The most pressing interim need identified by DOE and Ecology was for a safe, reliable, and regulatory compliant replacement cross-site transfer capability to move waste between the 200 West and 200 East Area tank farms. This transfer capability is needed because the 200 West Area has far less useable DST capacity than there is waste in SSTs. The replacement waste transfer capability would provide a safe, reliable, and regulatory compliant means to move waste from the 200 West Area to the available DST capacity located in the 200 East Area.

Based on tank waste management and operation activities when the SIS EIS was prepared, the following needs were addressed:

- Removing saltwell liquid from older SSTs to reduce the likelihood of liquid waste escaping from corroded tanks into the environment. Many of these tanks have leaked, and historically, new leaks, either known or assumed, have developed in these tanks at a rate of more than one per year;
- Providing the ability to transfer the tank waste via a regulatory compliant system to mitigate any future safety concerns and use current or future tank space allocations;
- Providing adequate tank waste storage capacity for future waste volumes associated with tank farm operations and other Hanford Site facility operations; and
- Mitigating the flammable gas safety issue in tank 101-SY.

The alternatives evaluated in the SIS EIS provide DOE with the ability to continue safe storage of high-level tank waste and upgrade the regulatory compliance status with regard to Resource Conservation and Recovery Act (RCRA) (40 Code of Federal Regulations [CFR] 260) and the Washington Administrative Code (WAC) Dangerous Waste Regulations (WAC 173-303).

On December 1, 1995, DOE and Ecology published their Record of Decision for the SIS EIS in the Federal Register (FR) (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;
- Continue operating the mixer pump in tank 101-SY to mitigate the unacceptable accumulation of hydrogen and other flammable gases; and
- Perform activities to mitigate the loss of shrub-steppe habitat.

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. The design and operation of the replacement cross-site transfer system will meet the requirements of RCRA and WAC for secondary containment and Tri-Party Agreement Milestone M-43-07, which required construction of the replacement cross-site transfer system to commence by November 1995. Construction of the cross-site transfer system has begun and the system is scheduled to be operational in 1998.

DOE will continue to use the existing cross-site transfer system until the replacement cross-site transfer system is operational to provide access to 200 East Area DSTs for storage of 200 West Area facility waste and retrieved liquid waste from SSTs. Saltwell liquid retrieval will continue to reduce the risk to the environment from leaking SSTs. Operational procedures will ensure the integrity of the existing cross-site transfer system before any waste transfers. The current planning base estimates that the existing cross-site transfer system will operate for approximately 625 hours during 5 transfers before the replacement cross-site transfer system is operational in 1998.

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 1 year of operation. DOE and Ecology revised their preferred alternative between release of the Draft and Final EIS, based on the demonstrated success of the mixer pump, and 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 since the issuance of the Final SIS EIS, DOE has decided to defer a decision on the construction and operation of a retrieval system in tank 102-SY. Through an ongoing safety evaluation process, DOE recently revisited its operational assumptions regarding the potential for the occurrence of a nuclear criticality event during waste storage and transfers. Changes to the Tank Farm Authorization Basis for Criticality approved in September 1995 were rescinded by DOE in October 1995, pending the outcome of a criticality safety evaluation process outlined for the Defense Nuclear Facility Safety Board on November 8, 1995. Until these criticality safety evaluations are completed, the Hanford Site will operate under the historic limits, which maintain reasonable assurance of subcritical conditions during tank farm storage and transfer operations. Of the actions evaluated in the Final SIS EIS, only the retrieval of solids from tank 102-SY was affected by the technical uncertainties regarding a criticality. Based on the quantities of Pu 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. As a result, a decision on retrieval of solids from tank 102-SY was deferred in the SIS EIS Record of Decision. Also, pending the outcome of the technical initiative to resolve the tank waste criticality safety issue, transfers of waste (primarily saltwell liquid) through tank 102-SY will be limited to noncomplexed waste. Tank 101-SY mixer pump operations, interim operations of the existing cross-site transfer system, operation of the replacement cross-site transfer system, saltwell liquid retrievals, and 200 West Area facility waste generation all would occur within the applicable criticality limits and would be subcritical.

Privatization of Tank Farm Activities

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 has issued a TWRS Privatization Request for Proposal and has received two bids to treat tank wastes (Briggs 1996). 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 LAW immobilization facilities, one of which may include a high-level waste (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. The planning process for these privatization activities is not complete. This planning process is subject to the final decision concerning remediation of the tank waste, which is the subject of this EIS.

Tank Farm Upgrades

Upgrades to the tank farms are planned to improve the reliability of safety-related systems, minimize onsite health and safety hazards, upgrade the regulatory compliance status of the tank farms, and place the tank farms in a controlled, stable condition until disposal is complete. Upgrades planned include 1) instrumentation including the automatic tank data gathering and management control system and the closed-circuit television monitoring to minimize personnel exposure; 2) tank ventilation to replace outdated ventilation systems; and 3) an electrical system to provide electrical power service with sufficient capacity and in compliance with current electrical codes (WHC 1996c). These three components of the tank farm upgrades are not addressed in the TWRS EIS but will be the subject of other analyses.

Upgrades to the existing waste transfer system that would be used in conjunction with the replacement cross-site transfer system also are planned. Waste transfer system upgrades are included in the TWRS EIS and discussed in Section B.3.0.2.

Initial Tank Retrieval System

This project would provide systems for retrieval of waste from up to 10 DSTs. Initial tank retrieval capabilities also would allow consolidation of compatible tank waste to create additional DST storage capacity and support passive mitigation such as diluting hydrogen-gas-generating Watchlist tanks should that become necessary. Retrieval of waste and transfer from all tanks is addressed in this EIS so the Initial Tank Retrieval System project is a subset of the actions included in this EIS and is not addressed separately.

Waste transfer system upgrades are an element of the Tank Farm Upgrades Project included in the TWRS EIS. Waste transfer system upgrades are discussed in Section B.3.0.2.

Hanford Tanks Initiative

Under this program, several waste retrieval activities discussed in the TWRS EIS would be demonstrated in support of the ex situ alternatives. This program would reduce the uncertainties associated with waste retrieval by developing and demonstrating the technologies required to meet retrieval requirements. The Hanford Tanks Initiative includes activities associated with waste retrieval and tank closure. Those activities associated with waste retrieval are covered under this EIS while activities associated with the closure would be the subject of future NEPA analysis.

This program would demonstrate equipment and systems for removal of tank residuals from tank 241-C-106 that are expected to remain following initial retrieval by sluicing. The objective would be to retrieve sufficient waste to meet waste retrieval requirements. This program also would attempt to develop technologies and criteria to retrieve waste from known or assumed leaking SSTs.

B.1.2 CESIUM AND STRONTIUM CAPSULES

B.1.2.1 Background

The cesium chloride (CsCl) and strontium fluoride (SrF_2) capsule program separated the heat-generating Cs and Sr from the tank waste. To reduce the heat being generated in the tanks, a portion of the tank waste was recovered and processed to isolate the Cs and Sr. Removing the heat-generating isotopes from the waste allowed safe storage of the waste. Cs and Sr were removed from existing tank waste through the waste retrieval and treatment program or by treating the waste as it came out of the processing facility before it was put into the waste storage tanks. The Cs and Sr capsule inventory now stored at the Waste Encapsulation and Storage Facility (WESF) is the result of separating Cs and Sr from other waste. The Cs and Sr were converted to chloride and fluoride salts, respectively, and encapsulated for storage. The retrieval and processing activities started in 1967 and lasted until 1985. The storage of the capsule inventory at WESF is an ongoing activity. The capsules are currently designated as waste by-product, which means they are available for productive uses if uses can be found. If and when they are determined to have no potential productive uses, they would be managed and disposed of as HLW consistent with the TWRS EIS alternative selected for implementation.

The majority of the Sr was removed from tank waste sludges obtained from eight tanks in the A and AX Tank Farms. Additional Sr was recovered directly from PUREX Plant waste. Cs is relatively soluble in the tank liquid, which allowed Cs recovery from tank liquid from numerous tanks. The majority of the Cs was recovered from liquid waste produced at the PUREX or REDOX Plants using an ion exchange recovery process.

A capsule configuration was selected for containing the stabilized CsCl and SrF_2 salts because it provides a physical form suitable for long-term storage. Details of capsule construction are shown in Figure B.1.2.1. Of the 1,577 Cs capsules initially fabricated, 249 have been subjected to destructive testing or repackaged into smaller sources and will not be returned. Similarly, of the 640 Sr capsules that were initially fabricated, 39 have been subjected to destructive testing or repackaging and will not be returned. At present, approximately 1,328 Cs and 601 Sr capsules are either stored onsite or will be returned to be stored at WESF by the end of 1997. The number of capsules could increase if any existing capsule or cut-up capsule contents are repackaged.

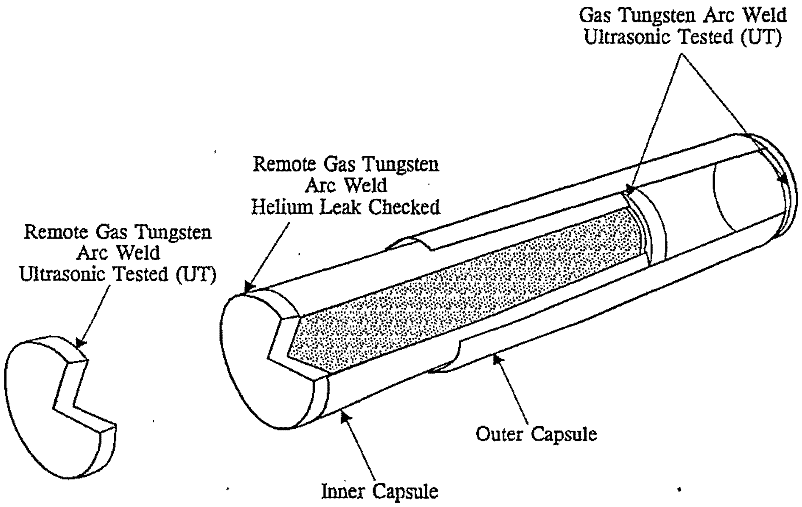
Once recovered, the Cs was converted to CsCl, which was melted and poured into a type 316-L stainless-steel capsule, which was then capped and sealed by welding. This capsule was placed inside another capsule and sealed by welding on an outer cap. Figure B.1.2.1 illustrates the general configuration and original design dimensions of the capsules. Later design revisions incrementally increased the inner and outer wall thicknesses. The majority of the capsules produced have the thicker walls. The Cs content of the capsules is primarily Cs-137, with a half-life of 30.17 years, releasing 8.7E-2 watts per gram (W/g) of initial Cs. This decay emits a beta ray 5.4 percent of the time with a maximum energy of 1.2 million electron-volts (MeV), and a beta ray 94.6 percent of the time with a maximum energy of 0.5 MeV. The less-frequent decay mode creates stable barium-137 (Ba-137). The more-frequent decay mode creates Ba-137m, a metastable isotope that decays to the stable Ba-137 through a gamma ray of energy 0.66 MeV.

The Ba-137m has such a short half-life (2.5 minutes) that it can be thought of as occurring simultaneously with the decay of Cs-137. The second decay adds 3.4E-1 W/g of initial Cs. The curie and thermal loading of the Cs capsules at various time periods is provided in Volume Two, Appendix A.

Figure B.1.2.1 Capsule Details

Chemical Form	Physical Form	Curie Loading	Percent of Theoretical Density Based on Total Void Space of Capsule	Capsule Temperature			
				In Air		In Water	
				Center Line	Surface	Center Line	Surface
Strontium Fluoride	Compacted Powder	150 kCi (Max)	68	860° C	430° C	660° C	71° C
Cesium Chloride	Melt-Cast	70 kCi (max)	65	450° C	200° C	327° C	58° C

Note: The curie loading for the capsules followed a broad nearly normal distribution.



	Capsule									
	Inner					Outer				
	Material	Wall[1] Thickness	Outside Diameter	Total Length	Total Cap Thickness	Material	Wall[1] Thickness	Outside Diameter	Total Length	Total Cap Thickness
Strontium Fluoride	Hastelloy C-276 (UT)	0.305 (UT)	5.72	48.39	1.02	Stainless Steel 316-L (UT)	0.277 (UT)	6.67	51.05	1.02
Cesium Chloride	Stainless Steel 316-L (UT)	0.241 (UT)	5.72	50.10	1.02	Stainless Steel 316-L (UT)	0.277 (UT)	6.67	52.77	1.02

Note: All dimensions are in cm.

[1] *Wall thicknesses shown represent original design values.*

SOURCE: WHC 1995h

The Sr was converted to SrF_2 salt and was physically packed into a metal capsule. The metal alloy used for the SrF_2 inner capsules was Hastelloy C276™, which is a high-temperature corrosion-resistant alloy. After welding a cap on the inner capsule, the entire capsule was placed into a type 316-L stainless-steel outer capsule and an outer cap was welded in place. The Sr content of the capsules is primarily Sr-90, which has half-life of 28.6 years. The Sr-90 decay emits a beta ray with a maximum energy of 0.5 MeV releasing 1.6E-1 W/g of initial Sr. This creates yttrium-90 (Y-90), which decays to stable zirconium-90. The Y-90 has such a short half-life (3 hours), that it can be thought of as occurring simultaneously with Sr-90. The second decay in this chain manifests itself in the emissions of a beta ray with maximum energy of 2.3 MeV, releasing an additional 7.7E-1 W/g of initial Sr. The curie and thermal loading of the Sr capsules at various time periods is provided in Volume Two, Appendix A. The high-temperature corrosion-resistant alloy is required for the SrF_2 capsules, because the Sr-90 decay chain results in higher capsule temperatures than experienced with the CsCl capsules.

The Cs capsules, which are strong emitters of penetrating gamma radiation, were shipped offsite in limited numbers and used for commercial irradiation purposes. The Sr capsules were used as heat sources because the primary radiation emitted by Sr is contained within the metallic capsule, which in turn heats the capsule. The capsules have also been used by DOE programs for fabricating radioactive sources and various research activities at Pacific Northwest National Laboratory, Sandia National Laboratory, and Oak Ridge National Laboratory. Several studies have been performed that document the integrity of the Cs and Sr capsules and their ability to continue safe storage. Corrosion data indicate that attack on the capsule walls from the CsCl would be very low.

The Cs capsule program was terminated, and the approximately 778 CsCl capsules that were at commercial facilities are in the process of being returned to the Hanford Site. Current plans call for all Cs capsules to be returned to the Site by the end of 1997. The commercial uses of the Cs capsules varied, with the majority of them used for sterilizing medical equipment and supplies. The offsite commercial uses of the CsCl capsules are shown in Table B.1.2.1

Table B.1.2.1 Offsite Commercial Uses of Cesium Chloride Capsules

Commercial Facility	Cesium Chloride Capsule Uses
Applied Radiant Energy Corporation	Strengthen wood products
Radiation Sterilizers, Inc.	Medical supplies and saline solution sterilization
IOTECH, Inc.	Medical product sterilization
Rocketdyne	Irradiation

B.1.2.2 Description of Cesium and Strontium Capsules

The Cs and Sr capsule program was performed between 1974 and 1985 at WESF to remove the heat-generating Cs and Sr isotopes from the tank waste because they generated sufficient decay heat to evaporate the water from the tank waste. Hypothetically, after all the tank wastewater had evaporated, the waste would continue to heat and had the potential to initiate a self-propagating reaction or destroy the structural integrity of the tank. Between 300 and 400 °C (570 and 750 °F), the oxidizing chemicals present (such as sodium nitrate) could have reacted with the organic chemicals remaining in the tank. This possibility was initially avoided by replacing the water that had evaporated; a more permanent solution was to substantially decrease the concentration of the heat source. The program to decrease the tank concentration and package the Cs and Sr was carried out between 1974 and 1985 at WESF, which is annexed to B Plant in the 200 East Area. The program timeline is shown in Figure B.1.2.2.

A capsule consists of a sealed inner metallic tube containing the radioactive material inside an outer metallic capsule providing secondary containment. The double-walled capsule is used to provide added safety for confinement (see Figure B.1.2.1).

Current and Planned Activities

The only ongoing and planned activities for the capsules are the continued storage of the capsules in WESF, return of the remaining capsules to WESF, and attempts to find productive uses for the Cs and Sr capsules. The Cs and Sr capsules are currently stored in water-filled basins at WESF in the 200 East Area. WESF is directly adjacent to B Plant in the 200 East Area, and is approximately 5,600 square meters (m^2) (60,000 square feet [ft^2]), approximately one-fifth the size of B Plant.

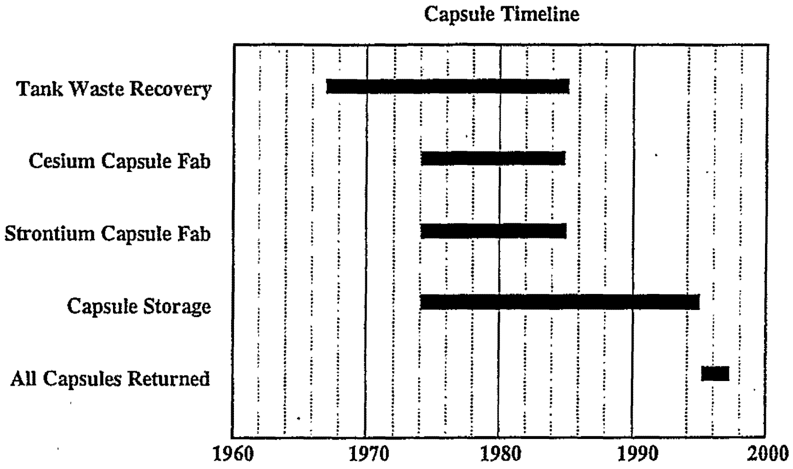
The capsules are stored, in a retrievable manner, in racks at the bottom of the pool cells, which are filled with water to a depth of 4 m (13 ft). The storage racks provide for controlled capsule storage locations within the pools. WESF has a total of eight pools, five that are active and used for capsule storage, one that is used for temporary storage, and two that are not used but are maintained. Storing the capsules under water cools the capsules and provides radiation protection for WESF workers. All of the storage basins are monitored for radiation, which would indicate a capsule leak.

Currently, B Plant is scheduled for deactivation by the year 2001. DOE currently is upgrading WESF to operate independently of B Plant because in the past, operation of WESF was dependent on the operation of B Plant.

DOE is in the early planning stages of considering whether the capsules should remain in WESF or be placed in alternative locations for storage. Among the possible alternatives that may be considered are placing the capsules in the proposed Canister Storage Building originally planned to store HLW.

No decisions have been made to proceed with any alternative storage options. For purposes of analyzing impacts in the TWRS EIS, it is assumed that the capsules will remain in WESF until disposal. If DOE decides to change the method or location for the interim storage of the capsules, an appropriate NEPA review would be performed. A Cs and Sr capsule management program will provide for management of the capsules until final disposition has been implemented.

Figure B.1.2.2 Capsule Program Timeline



SOURCE: Adapted from WHC 1995b

Capsule safety concerns have not been broken down into specific categories. However, the dominant safety issue for the capsules is the integrity of the storage facility. As it currently exists, the storage facility at WESF has no provision for handling a situation in which the cooling water is lost. If a catastrophic event such as an earthquake were to occur and cause a failure in the basin or its water supply, there is no engineered system to provide secondary containment or an alternate water supply, although efforts are underway to resolve this issue. The impacts of such an event are discussed in Volume Four, Appendix E of the EIS.

DOE is pursuing alternative uses for the Cs and Sr capsules. If no future uses for these capsules are found, the capsules eventually would be designated as HLW and managed and disposed of consistent with the Tri-Party Agreement and the TWRS EIS alternative selected for implementation.

B.1.2.3 Volume and Activity Comparison Between Capsules and Tank Waste

The volume of the material in all of the Cs and Sr capsules combined is approximately 2 cubic meters (m^3) (70 ft^3), which is very small in comparison to the 2.1E+5 m^3 (7.5E+06 ft^3) in the waste storage tanks. Although the amount of material in the capsules is small, the amount of radioactivity contained in the capsules is approximately 35 percent of the total activity of the waste storage tanks and the capsules combined. Thus, separating and encapsulating the Cs and Sr from the other tank waste resulted in containing a large portion of the radioactivity in a small volume.

B.1.2.4 Current Monitoring and Maintenance

Monitoring and maintenance activities for the capsules involve calculating the annual inventory, physically verifying that the inner capsule can still move independently of the outer capsule, and using online radiation monitors to detect pool cell water contamination. The annual inventory provides the exact storage location and accountability for all of the Cs and Sr capsules stored at WESF.

The Cs capsules are "clunk-tested" on a quarterly basis. This involves physically grasping one end of a capsule with a pool tong and rapidly moving the capsule vertically approximately 15 cm (6 in.). This allows the inner capsule to slide within the outer capsule, making a "clunk" sound that is easily heard and felt by the operator performing the test. This test verifies that the capsule has not bulged.

B.2.0 DEVELOPMENT OF ALTERNATIVES

This section explains the process that was followed to develop alternatives for remediating the tank waste, implementing the alternatives for remediating the tank waste, and remediating the cesium and strontium capsules. This section also discusses the TWRS activities that are not included in the EIS.

B.2.1 TANK WASTE

B.2.1.1 TWRS Elements

Final remediation of TWRS involves three distinct activities: remediating the tank waste; dispositioning the tanks and all associated equipment (a process called closure); and decontaminating and decommissioning any new facilities constructed to remediate the tank waste. These activities are described in the following text.

Remediating Tank Waste

Remediating the tank waste involves those activities associated with remediating the waste in 177 underground tanks and approximately 60 MUSTs. The activities required to remediate this waste is the subject of this EIS. Volume One, Section 3.3. describes the process followed to select alternatives for inclusion in this EIS. The remainder of the EIS provides information relative to the environmental impacts of the alternatives addressed.

Dispositioning the Tanks (Closure)

The final disposition of the tanks and associated equipment and the remediation of contaminated soil and groundwater associated with leaks from the tanks is a process called closure. Closure is not within the scope of this EIS because there is insufficient information concerning the amount of contamination that would need to be remediated. For purposes of comparing the alternatives, a single and consistent method of closure was assumed for all of the alternatives. Closure as a landfill was chosen as the representative closure method for purposes of analysis and is included in all of the alternatives (except the No Action and Long-Term Management alternatives). This does not mean that closure as a landfill is proposed or necessarily would be selected in the future. It is included to allow a meaningful comparison of the in situ and ex situ alternatives and to provide information to the public and the decision makers of the total cost and impacts of final restoration of the Site.

Decontamination and Decommissioning
Decontamination and decommissioning of any new facilities constructed to implement any of the alternatives is not evaluated in detail in this EIS because the decisions on the appropriate method to accomplish decontaminating and decommissioning the facilities would not be required until the treatment and disposal of waste was completed, which is up to 30 years in the future. Insufficient information is available at present to provide meaningful evaluation; however, decontamination and decommissioning of these facilities is foreseeable. Therefore, the costs, personnel requirements, and volume of contaminated and noncontaminated materials resulting from decontamination and decommissioning are included in each alternative to show how tank waste remediation and decontamination and decommissioning are interrelated.

B.2.1.1.1 Development of Alternatives

A wide range of potentially applicable technologies exists for treating tank waste. One of the challenges for DOE and Ecology is to eliminate technologies that are not viable and develop a range of reasonable alternatives for presentation in the TWRS EIS. This section describes how the alternatives were developed.

There is a distinction between technologies and alternatives. Technologies are specific processes (e.g., cesium ion exchange) that relate to a component (e.g., retrieval or treatment) of an alternative. Alternatives include a set of technologies, or building blocks, that have been engineered to work together, forming complete systems for accomplishing the purpose and need for action. Alternatives are made up of a number of technologies linked together.

The first step in developing alternatives was to screen out technologies that were not viable. The full range of available technologies for each component of the proposed action was evaluated, and technologies that were not viable were eliminated from further consideration. The technologies eliminated by this screening process are described in Volume One, Section 3.8 and Volume Two, Appendix C.

After rejecting technologies that were not viable, a large number of potential technologies remained for inclusion in the EIS. It would not be possible to develop alternatives that include all of the potential combinations of technologies. In accordance with NEPA, representative alternatives were developed to bound the full range of reasonable alternatives (10 CFR 1500). Upper, lower, and intermediate bounding alternatives were developed in terms of cost, risk, and technologies for the two primary decisions that affect environmental impacts: the amount of waste to be retrieved from the tanks and the degree of separation of retrieved waste into HLW and LAW. The full range of applicable technologies and alternatives is included in the EIS.

Because representative alternatives were developed to support detailed analysis in the EIS, there are many other viable technologies for individual components of the alternatives that could not be included in the detailed analysis. These technologies are included in Section B.9 and could be substituted for one of the technologies that is included in an alternative without a substantial change in the impacts of that alternative. An evaluation was performed of each of the technologies identified in this appendix. Most of these technologies would have little change on impacts. Where there were changes in impacts, the changes were discussed. The level of analysis was dependent on the magnitude of the change on impacts.

The alternatives developed for presentation in the EIS were chosen to be representative of many of the possible variations of the alternative. The design information for all alternatives is at an early planning stage, and the details of the alternative that ultimately is selected and implemented may change as the design process matures. Therefore, the alternatives are intended to represent an overall plan for remediation at a level of detail sufficient for impact analysis and alternative comparisons.

NEPA requires that an EIS include a No Action alternative, which addresses the alternative of not taking the proposed action (i.e., not initiating the project). For the TWRS project, there is a management program in place to continue the safe management of the tank waste and the capsules; therefore, the No Action alternative addressed in this EIS (continue the current waste management program) consists of the activities currently being conducted to safely manage the waste. Further, under the No Action alternative, no new facilities would be constructed other than those for which decisions already have been made based on other NEPA reviews (e.g., the SIS EIS).

Since the late 1950's, there have been numerous studies analyzing alternatives for tank waste treatment and disposal. The technologies that have contributed to the development of the alternatives presented in the EIS come from different sources. One of the main sources of information is the Tank Waste Technical Options Report (Boomer et al. 1993). The initial set of technologies used in the report was obtained by reviewing literature for processing radioactive, hazardous, and mixed waste. The literature review was supplemented by several DOE-sponsored workshops on treatment technologies for Hanford Site tank waste. Objectives and technologies were also proposed for consideration in the EIS during the public scoping process.

Four general categories of response actions have emerged through the alternative identification process. These categories are 1) continued safe management of the tank waste; 2) waste treatment and disposal in the tanks, referred to as in situ treatment; 3) waste treatment outside of the tanks in a processing facility, referred to as ex situ treatment; and 4) a combination of in situ and ex situ treatments. In situ waste treatment would not involve removing the waste from the tanks. In situ alternatives eliminate the need for any waste retrieval and would result in leaving all of the waste onsite following treatment. Ex situ treatment would require that the waste be removed from the tanks for treatment and disposal. Ex situ alternatives provide the opportunity for separating the waste into HLW and LAW components. The purpose of separating the waste is to minimize the volume of HLW requiring offsite disposal. Combination alternatives provide the opportunity to selectively retrieve waste for ex situ treatment based on waste type to achieve acceptable post-remediation risk levels.

Ex situ alternatives allow for geologic disposal of HLW at a potential geologic repository. Solely for the purpose of analysis, the potential geologic repository at Yucca Mountain, Nevada was assumed to be the final destination because it is currently being characterized to determine its suitability as a repository. It was assumed that the potential geologic repository would be operational and accept HLW generated by the ex situ alternative (see Section B.10.0 for additional details).

In January 1994, DOE, Ecology, and EPA renegotiated the Tri-Party Agreement, which led to a new proposed technical strategy for remediating the tank waste. This technical strategy provides the basis for the TWRS EIS Ex Situ Intermediate Separations alternative and includes the following activities:

- Retrieve present and future waste from all DSTs and SSTs;
- Separate the waste into high-level and low-activity streams to the extent required to meet onsite disposal requirements for LAW and maintain an acceptable volume of HLW for offsite disposal;
- Vitrify the LAW and dispose of it onsite in a near-surface disposal facility in a retrievable form; and
- Vitrify the HLW and store it onsite at a designated interim storage facility for future disposal at the potential geologic repository.

B.2.1.1.2 Implementation of Alternatives

There are many technical uncertainties associated with all of the alternatives for remediating the tank waste. These uncertainties include the types of waste contained in the tanks and the effectiveness of the retrieval techniques, waste separations, waste immobilization, and cost of implementing the alternatives. These uncertainties exist because some of the technologies that may be implemented are first-of-a-kind technologies, and have not previously been applied to the TWRS tank waste, or have not been applied on a scale as large as would be required for the TWRS tank waste.

Because of these uncertainties, DOE considered different approaches to implementing the alternatives to reduce the financial risk involved if one or more of the technical uncertainties could not be readily resolved. DOE identified two approaches to implementing the alternatives: full-scale implementation and phased implementation. Under full implementation, either DOE or a private contractor would design, construct, and operate full-scale facilities to remediate the tank waste. Under phased implementation, DOE or a private contractor would design, build, and operate demonstration-scale facilities to prove that the remediation concept would function adequately before constructing and operating a full-scale facility. All calculations performed for this EIS are based on DOE implementing the alternatives through the existing Management and Operations Contractor system. This phased implementation approach has the potential to prove that the technologies work before committing large capital expenditures that could not be recovered.

A phased approach could be developed for any of the alternatives but not all phased approaches would involve changes to environmental impacts from the full-scale approach. Therefore, not all phased approaches need to be addressed in the EIS. To decide which of the full-scale alternatives would need to have an associated phased implementation alternative addressed in this EIS, the following two criteria were used.

- Would the full-scale alternative involve large front-end expenditures of funds that could be lost if an unproven technology did not function adequately?
- Would the environmental impacts of the phased implementation approach be different than those of the full-scale alternative?

If either criterion was met, a phased approach would be included in the EIS.

Applying these criteria showed that most alternatives did not warrant a separate analysis of a phased implementation approach. A phased implementation approach to the No Action and Long-Term Management alternatives would not involve changes in environmental impacts, large front-end expenditures, or unproven technologies, so no phased approach was included in the EIS for these alternatives. A phased implementation approach to the In Situ Fill and Cap alternative would involve the simple process of filling several tanks as a demonstration, and therefore would not involve different environmental impacts or large front-end expenditures of funds that could be lost, so no phased approach was included in the EIS. Similarly, a phased approach to the In Situ Vitrification alternative would involve testing the in situ vitrification process first on MUSTS, then small tanks, and then large tanks. Although this technology previously has not been performed on the tank waste, it could be tested gradually without any differences in environmental impacts or large expenditures of funds that could be lost if the process did not function adequately. Therefore, the In Situ Vitrification alternative did not warrant a separate phased implementation alternative, and no phased approach was included in the EIS.

All of the ex situ alternatives involve the application of technologies that have not been applied to the tank waste, and all would involve large front-end expenditures of funds to construct large, complex separations and immobilization facilities. The phased implementation approach for these alternatives would involve constructing and operating demonstration-scale facilities prior to constructing the full-scale facilities, and therefore would result in environmental impacts substantially different than the full-scale implementation alternative. Therefore, a Phased Implementation alternative has been included in the EIS to bound the impacts for the ex situ alternatives.

The Phased Implementation alternative consists of two phases: a proof of concept or demonstration phase (Phase 1) and a full-scale treatment phase (Phase 2). Phase 1 would include the construction and operation of one combined separations and LAW vitrification facility and one combined separations, LAW vitrification, and HLW vitrification facility. Enough waste would be remediated to prove that the many waste types in the tanks could be remediated effectively. Phase 2 would include completing tank waste remediation by constructing and operating new full-scale separations, LAW immobilization, and HLW vitrification facilities. The degree of separations into LAW and HLW was assumed to be similar to the Ex Situ Intermediate Separations alternative and includes additional processes to separate out the Sr, Tc, and TRU elements.

The following tank waste alternatives are addressed in this EIS:

- No Action;
- Long-Term Management;
- In Situ Fill and Cap;
- In Situ Vitrification;
- Ex Situ Intermediate Separations;
- Ex Situ No Separations;
- Ex Situ Extensive Separations;
- Ex Situ/In Situ Combination 1;
- Ex Situ/In Situ Combination 2; and
- Phased Implementation (preferred alternative).

The alternatives developed for detailed analysis cover the full range of actions as well as the No Action alternative. The tank waste alternatives range from waste containment with the Long-Term Management alternative to extensive processing (separating HLW from LAW fractions) and immobilization using new technologies with the Ex Situ Extensive Separations alternative. The relationship among the alternatives is shown in Figure B.2.1.1.

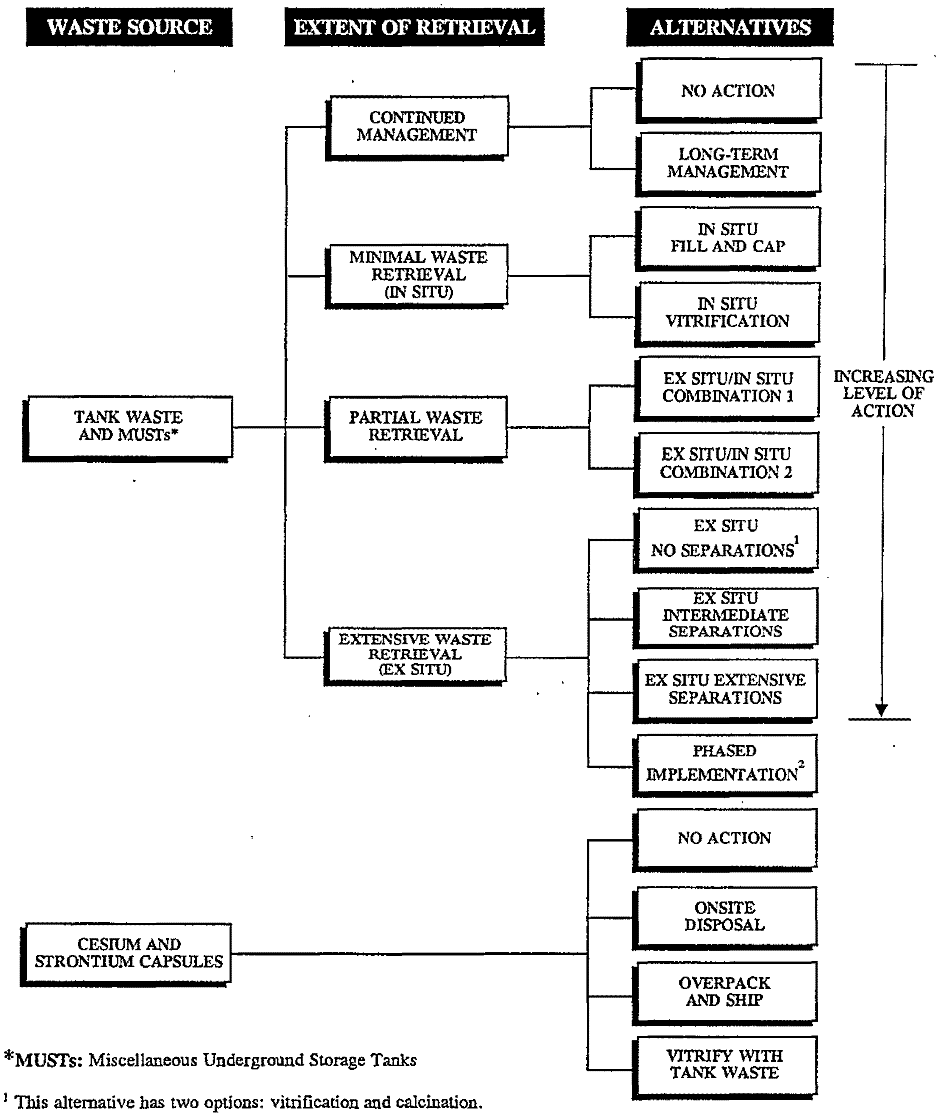
B.2.1.2 Cesium and Strontium Capsules

The Cs and Sr capsules are currently classified as waste by-product and this EIS is addressing only measures to remediate the capsules when an if they are determined to have no potential productive uses. The development of alternatives to remediate the Cs and Sr capsules is much less technically complicated than the tank waste. There are two distinct activities related to remediation of the capsules: the disposition of the capsules, which is the subject of this EIS; and decontamination and decommissioning of the current storage location of the capsules in WESF, which is part of B Plant. Decontamination and decommissioning of WESF would be performed with the remainder of B Plant and is not within the scope of this EIS.

The alternatives for remediating the capsules include No Action, disposal on the Hanford Site, or disposal off the Hanford Site either with or separately from the tank waste. None of these involve unproven technologies or the construction of major facilities. The following capsules alternatives are addressed in this EIS:

- No Action;
- Onsite Disposal;
- Overpack and Ship; and
- Vitrify with Tank Waste.

Figure B.2.1.1 Relationship Among TWRS EIS Alternatives



*MUSTs: Miscellaneous Underground Storage Tanks

[1] This alternative has two options: vitrification and calcination.

[2] A phased approach could be taken to all alternatives except No Action and Long-Term Management. The phased approach would have the same impacts as the full implementation approach for all alternatives except the ex situ alternatives. A single phased alternative called Phased Implementation was included in the EIS to be representative of implementing the phased approach for any of the ex situ alternatives and to bound impacts.

B.3.0 TANK WASTE ALTERNATIVES

The following sections describe each of the tank waste alternatives. Elements common to all tank waste alternatives are described in Section B.3.0. The discussion includes a general description of the alternative followed by a description of the construction activities that would be involved if the alternative would be implemented. The discussion continues with a description of the process/operations and ends with a discussion of key issues associated with implementing the alternative. Engineering data for each alternative may be found in Section B.11.0. Each alternative includes the continuation of routine operations discussed in Section B.1.1.7.

B.3.0.1 Current Operations

Included in all of the alternatives are the continued operations activities necessary to maintain the tanks and associated facilities until they are no longer required for waste management. Current operations include the following activities:

- Managing operations;
- Operating and maintaining facilities and equipment;
- Monitoring tanks to gather information including data on waste temperatures, liquid levels, and tank status;
- Monitoring leak detection equipment, including drywells around the tanks for increases in radioactivity, groundwater monitoring, and in-tank liquid level monitoring;
- Adhering to regulatory compliance and reporting;
- Conducting security and surveillance of facilities and grounds;
- Performing interim stabilization of SSTs by saltwell pumping;
- Operating the 242-A Evaporator to concentrate waste;
- Maintaining tank safety including diluting tank waste as necessary and maintaining adequate storage capacity; and
- Characterizing MUST waste associated with TWRS.

The 242-A Evaporator is an existing facility located in the 200 East Area. This facility has been recently upgraded and is used to concentrate liquid waste to maintain adequate tank space for the addition of planned waste additions to the DSTs. This also involves maintaining spare tank space for unplanned or emergency response waste transfers. The 242-A Evaporator would be used during all of the alternatives and during current operations to concentrate all of the DST waste for the In Situ Fill and Cap alternative and the In Situ Vitrification alternative. The 242-A Evaporator would be used up through the year 2005, after which there is expected to be limited requirements for waste evaporation. The 242-A Evaporator would require major upgrades to continue operations beyond the year 2005.

The functions and activities for current operations are the same for each alternative but the costs, schedule, and staffing levels vary between alternatives according to the schedule for completing waste treatment and subsequent closure of the tank farms. The impacts of these routine operations are included in the impacts presented for each alternative in Volume One, Section 5.0.

B.3.0.2 Waste Transfer System Upgrades

Included in all of the alternatives except for No Action are upgrades to the existing waste transfer system under Project W-314. Waste transfer system upgrades would involve constructing buried waste transfer pipelines in 200 East Area. The new transfer lines would all be double-walled with leak detection and corrosion prevention systems.

These upgrades would provide for reliable and compliant waste transfers from waste generating facilities into and between the DST farms. Selected valve pits and diversion boxes would be upgraded by installing liners to provide secondary containment in the event of a leak or spill. Also included are new jumper and cover installations for selected valve pits and diversion boxes. The various flow-path combinations would be indicated on the new cover blocks. The replaced burial transfer lines would be abandoned in place; whereas the replaced items such as valve and diversion box jumpers and box covers would be removed and disposed of accordingly (WHC 1996c).

B.3.0.3 Major Assumptions and Uncertainties for Ex Situ Alternatives

The major assumptions used for the ex situ alternatives are summarized in Table B.3.0.1. Additional information regarding major assumptions and uncertainties are provided in Section B.8.0.

B.3.0.4 Multi-Purpose Canister

For comparison purposes it has been assumed that each of the ex situ alternatives would use the conceptual Hanford Multi-Purpose Canister (HMPC) for interim onsite storage of HLW and subsequent transportation to the potential geologic repository. The HMPC would be approximately 4.6 m (15 ft) long and 1.4 m (4.5 ft) in diameter. The sizing of the HLW canisters and the decision to use a multi-purpose overpack canister have not been finalized. There may be potential economic and handling benefits to using an HMPC for the TWRS Program. There also may be potential additional cost in using HMPCs if future evaluations determine that the HMPCs were not acceptable for disposal or the HMPCs require costly changes in repository design and operations.

B.3.0.5 Liquid Effluent Processing

Liquid effluent processing of secondary radioactive waste streams for all of the alternatives would be provided by the secondary radioactive liquid-waste processing system. This system, which is a combination of a storage, treatment, and disposal facility, was completed in 1994 and would be permitted and operational in time to support each of the alternatives. The secondary radioactive liquid-waste processing system consists of the Liquid Effluent Retention Facility, the Effluent Treatment Facility, and the State-approved land disposal site.

The Liquid Effluent Retention Facility provides for interim storage for dilute, radioactive aqueous waste streams. The facility provides up to 4.9E+07 L (1.3E+07 gal) of temporary storage capacity for liquid waste prior to treatment at the Effluent Treatment Facility. This storage capacity is provided by two 2.5E+07 L (6.5E+06 gal) lined and covered basins. An additional storage basin is provided for emergency use.

Table B.3.0.1 Ex Situ Alternatives Major Assumptions

Parameter	Ex Situ Intermediate Separations	Ex Situ No Separations Vitrification	Ex Situ Extensive Separations	Ex Situ/In Situ Combination 1 (Ex Situ Portion)	Ex Situ/In Situ Combination 2 (Ex Situ Portion)	Phased Implementation
Retrieval Efficiency (% recovered from the tanks)	99%	99%	99%	99%	99%	99%
Treatment Plant Size						
HLW	20 mt/day	200 mt/day	1 mt/day	8 mt/day	5 mt/day	Phase 1: 1 at 1 mt/day HLW 2 at 20 mt/day LAW
LAW	200 mt/day	N/A	200 mt/day	120 mt/day	70 mt/day	Phase 2: 1 at 10 mt/day HLW 2 at 100 mt/day LAW
Operating Efficiency	60%	60%	60%	60%	60%	Phase 1: 60% Phase 2: 70%
HLW Canister Size	1.17 m^3 (41 ft^3)	10 m^3 (360 ft^3)	1.17 m^3 (41 ft^3)	1.17 m^3 (41 ft^3)	1.17 m^3 (41 ft^3)	1.17 m^3 (41 ft^3)
Waste Loading [1] weight % (before blending)						
HLW	20%	20% [2]	20%	20%	20%	20%
LAW	15%	N/A	15%	15%	15%	15%
Blending Factor						
HLW	1.2	1.2	1.2	1.2	1.2	1.2
LAW	1.0	N/A	1.0	1.0	1.0	1.0

Notes:

[1] HLW loading is in terms of weight percent waste oxides excluding the sodium and silica in the tank waste. LAW loading is in terms of weight percent sodium oxide.

[2] HLW loading is in terms of weight percent sodium oxide.

N/A = Not applicable

The Effluent Treatment Facility provides the final liquid effluent processing step prior to disposal. This facility uses "best available technology" to reduce the concentrations of radioactive and hazardous waste constituents to meet water quality discharge regulatory limits. The treatment process includes a combination of filtration, reverse osmosis, ultraviolet oxidation for organic destruction, evaporation, and ion exchange.

The treated effluents from the Effluent Treatment Facility would be transferred to the State-approved land disposal site for final discharge to the soil column. This facility is located north of the 200 West Area. Treated effluents are verified for composition at the Effluent Treatment Facility and transferred to the State-approved land disposal site where they are discharged to the soil column through a piping manifold.

B.3.0.6 Waste Compositions

Vitrification or glassmaking is a waste stabilization and solidification technology that incorporates radioactive and hazardous waste into a glass matrix. This process involves blending the waste material with glass formers or additives and heating the mixture to glass-forming temperatures. The types of glass formers added to the waste define the resulting glass type.

Borosilicate glass is based on a composition of silicon dioxide, boron trioxide, sodium oxide, and lithium oxide. Borosilicate glass is the standard final waste form for treating high-level radioactive waste because of its durability and ability to accommodate a varied range of waste feeds (DOE 1990). Additionally, borosilicate glass is currently identified as the only standard HLW form that will be accepted at the potential geologic repository (DOE 1994g).

Other types of glass could be selected for the vitrification of HLW or LAW; however, they would have to meet the repository or performance assessment criteria. One example is the soda-lime glass that would be produced by the Ex Situ No Separations (vitrification) alternative. Soda-lime glass consists of mainly silicon dioxide, sodium oxide, and calcium oxide.

Two types of vitrified waste forms described in the alternatives are monoliths and cullet. Monoliths would be produced by casting the molten glass into canisters resulting in a single piece of glass. The cullet would be produced by quenching the molten glass in water following vitrification, resulting in gravel-sized pieces of glass.

Cullet would provide processing and material handling advantages for the high-capacity processing facilities. The disadvantage of cullet as a waste form is its high surface area-to-volume ratio, which results in lower long-term performance. Matrices or coating material can be used in conjunction with the cullet to improve the waste-form performance. All of the ex situ alternatives that produce vitrified LAW for onsite disposal have assumed cullet in a matrix material as the waste form for onsite LAW disposal. This provides for a conservative analysis of the long-term impacts resulting from onsite disposal of LAW.

Grouting of the retrieved tank waste is a technology that could be applied to any of the ex situ alternatives in place of vitrifying the waste. Grout is a common solidification and stabilization technology used in the management of hazardous and radioactive waste. Grout is a general term that refers to a waste form obtained by mixing waste with chemical additives to stabilize and immobilize the hazardous constituents. The grouting process applied to the ex situ treatment of the tank waste would involve waste retrieval and transfer to a grout facility where the waste would be mixed with appropriate mixtures of grout formers. After the grout is mixed, it could be placed into containers or pumped into large vaults for solidification and disposal.

Grouting of tank waste has been studied extensively at the Hanford Site for use as a technology for LAW disposal. Grouting of the LAW was selected as the LAW treatment method in the Hanford Defense Waste EIS (DOE 1987). The LAW described in the Hanford Defense Waste EIS included liquid waste from the tanks (after separation of HLW components) and secondary waste from the HLW vitrification facility, which would consist of waste from canister decontamination, drying of feed material, and off-gas treatment.

Each of the alternatives that involve treating the waste would involve collecting small sample quantities (up to about 2.5 L [0.65 gal] per sample) of tank waste and shipping the samples to offsite locations for bench-scale waste treatment and immobilization performance demonstration and testing purposes. The general approach would include collecting grab and/or core samples from the tanks, verifying that the samples and sample contents meet appropriate specifications (using existing onsite laboratory facilities, including necessary laboratory preparatory work [e.g., preparing composite samples]), and appropriately packaging and transporting the samples to other DOE facilities or to private contractor facilities. These activities are covered under NEPA categorical exclusions defined in NEPA Implementing Procedures (10 CFR 1021).

B.3.0.7 Waste Minimization

Each alternative would involve waste minimization practices for primary, secondary, and tertiary waste. Primary waste is the treated tank waste and capsule contents requiring disposal. Primary waste minimization practices would be used to control the volume of HLW and LAW requiring disposal. Secondary waste is generated during waste handling and processing and includes items such as contaminated filters, spent ion exchange resins, and liquid effluents. Tertiary waste typically consists of items such as contaminated personnel protective clothing and equipment.

Secondary waste minimization would involve practices such as using metal high-efficiency particulate air (HEPA) filters that could be washed in-place and reused. In some process configurations, spent ion exchange resin would be fed into the waste treatment process to reduce the volume of secondary waste. Liquid effluents from all alternatives would be treated at the Effluent Treatment Facility in the 200 East Area prior to release. The amount of tertiary waste generated would be primarily a function of the number of operating personnel. Secondary and tertiary waste would be divided into LAW and TRU waste based on characterization. LAW would be disposed of at the onsite low-level waste burial grounds. TRU waste would be retrievably stored for future packaging at the Waste Receiving and Processing Facility. Current plans call for disposing TRU waste at the Waste Isolation Pilot Plant.

Each tank waste alternative that uses high-temperature processing (vitrification or calcination) would make extensive use of recycle streams to recycle back into the treatment process volatile radionuclide and chemical constituents captured in the off-gas systems. These recycle streams would minimize the generation of secondary waste. It has been determined that a bleed stream would be required for each alternative to avoid a continuous buildup of certain volatile radionuclides and chemical constituents, namely Tc-99 and mercury (Hg), in these recycle streams. For comparison purposes for each alternative, it has been assumed that the bleed stream would be 1 percent of the recycle stream. This additional secondary waste stream would then require low temperature stabilization, such as grouting, prior to storage or disposal. The stabilized recycle bleed stream waste would be transported to the existing Hanford Site solid waste handling system.

B.3.0.8 Cost Estimates

Complete life-cycle cost estimates are presented for each alternative. These estimates are based on conceptual designs and have a certain level of uncertainty associated with them. This uncertainty is accounted for in the estimating process by adding a contingency or percentage increase to the capital cost estimates. The contingency applied to the capital cost estimates for each alternative is in the range of 30 to 50 percent with the variation used to account for differences in the levels of design details developed for the different alternatives. A cost uncertainty analysis has been completed for the tank waste alternatives that resulted in an estimated cost range for each alternative as shown in Section B.8.3.

Capital cost as used in presenting cost data for the alternatives represents the total installed cost for the treatment facilities and includes materials and equipment, labor, construction management, project management, engineering, and contingency.

Capital Cost = DC + CM + PM +E + C
DC = Direct cost of materials, equipment, and labor
CM = Construction management
PM = Project management
E = Engineering
C = Contingency (equal to [DC+CM+PM+E]*Contingency Factor)

Cost associated with current operations are included in each of the cost estimates. Research and development cost is included in the cost estimates provided for each alternative that requires development. This cost is assumed to develop the technologies required to implement an alternative. The resolution of implementability issues identified for each alternative would be a part of the development work and thus, the research and development costs partially reflect the implementation uncertainties. To account for the conceptual nature of the designs for the alternatives, the research and development cost was taken as a percentage of the estimated capital cost.

Repository fees for alternatives that include shipment of HLW to the potential geologic repository are discussed in Section B.10.0.

B.3.0.9 Plant Sizing

The design capacities for the full-scale ex situ processing facilities were developed using a consistent approach. Each vitrification plant was sized using a 60 percent total operating efficiency. This assumed operating efficiency was used in conjunction with the waste inventory and operating schedule to size the treatment facilities. The Ex Situ No Separations Calcination alternative was developed using the same throughput rates and schedule as the Ex Situ No Separations Vitrification alternative resulting in an operating efficiency lower than 60 percent. The overall efficiency used to develop the Phased Implementation (Phase 2) treatment facilities was assumed to be higher than the full implementation alternatives. This efficiency improvement would result from lessons learned and processing experience gained during Phase 1.

Each of the ex situ alternatives include sufficient interim onsite storage capacity to store all of the immobilized HLW produced while awaiting shipment to the potential geologic repository. This allows any of the alternatives to operate independently of the potential geologic repository. To address concerns regarding the uncertainty in the geologic repository schedule, the impacts associated with interim onsite storage of the vitrified waste have been assessed for a 50-year period. This is longer than the schedules developed for each alternative for shipment of HLW to the potential geologic repository.

B.3.1 NO ACTION ALTERNATIVE (TANK WASTE)

B.3.1.1 General Description

The No Action alternative for the tank waste would consist of continuing to manage the tank farms consistent with current waste management programs. This would include waste retrieval or other actions for which decisions have already been made. No remediation of the tank waste would occur. DOE would continue to monitor and maintain the tanks and support facilities.

Maintenance activities would include monitoring the tanks and upgrading instrumentation and ventilation equipment. Administrative controls would be maintained to prevent inadvertent human intrusion. Because it is not reasonable to assume that administrative controls can be maintained forever, a time must be assumed when the management of the tank farms would end. For the purpose of assessing impacts, it was assumed that administrative controls would be effective for 100 years. The 100-year administrative control period is being used to provide a consistent basis for assessing potential human health and environmental impacts. DOE and Ecology currently have no policies or plans that would permit the loss of administrative control for radioactive and hazardous materials.

Saltwell pumping of the SSTs is an ongoing operation that is scheduled to be completed in the year 2000. Because the majority of the free liquid in the SSTs will have been removed, this waste will represent a minimum threat of releasing liquid to the groundwater until a point in the distant future. Therefore, no additional management action other than monitoring and maintaining the tanks is needed for the SSTs during the 100-year administrative control period. Existing MUSTs would be left in place and monitored similarly to the SSTs.

The DSTs have an estimated design life of 50 years. Continued management would include maintaining spare DST space to accommodate leak recovery in the event of a DST leak. Tank conditions would be continually monitored, and those tanks determined to be leaking would require recovery of the leakage from the tank annulus. The recovered waste would be transferred to another DST (Figure B.3.1.1).

B.3.1.2 Facilities to be Constructed

Under the No Action alternative, no new facilities would be constructed, including the W-314 feeder lines discussed in Section B.3.0.2.

B.3.1.3 Description of the Process

For a period of 100 years, the No Action alternative would continue monitoring and providing safe storage of the tank waste that are currently in the SSTs, DSTs, and inactive MUSTs.

SST Process

Saltwell pumping would be completed, and the remaining waste, sludge, and saltcake would remain in place. SSTs would remain in an interim stabilized condition and would be monitored following the completion of saltwell pumping and intrusion prevention activities, as described in Section B.1.1.7. The SSTs would contain less than 190 m^3 (50,000 gal) of interstitial liquid and less than 19 m^3 (5,000 gal) of drainable or free liquid.

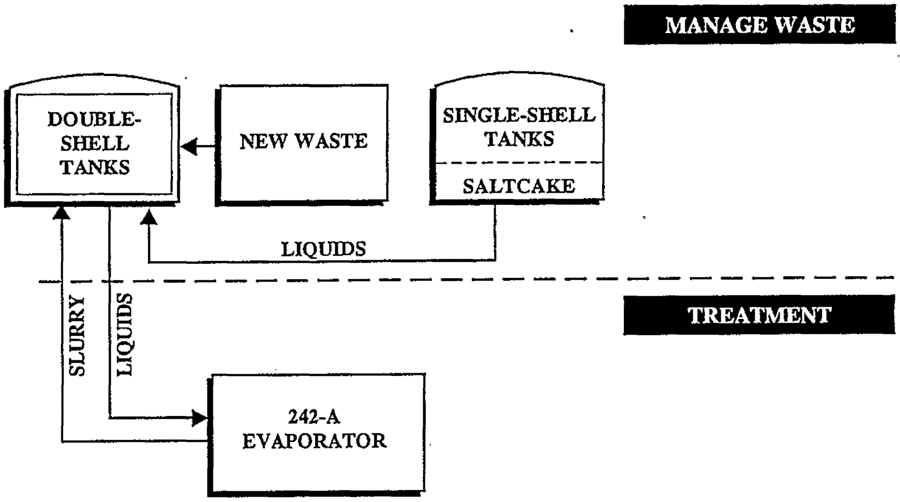
Structural analysis of tank design and laboratory testing of concrete samples from SSTs showed the probability of tank dome failure before loss of institutional control from deterioration or earthquake-induced forces to be slight. SST monitoring activities would include continued monitoring of tank dome elevations. Maintenance on the tank and support structures would continue, and risers and other openings into the tanks would continue to be capped in an effort to isolate the tanks. Drywell monitoring would continue and upgrades to the drywells would be made as necessary.

Surveillance under the No Action alternative would be provided appropriate to the degree of isolation of the tanks. Thus, surveillance would continue at the current level until the adequacy of isolation procedures could be confirmed. Site services (security, fire protection, environmental monitoring, and utilities) would be maintained at current levels.

DST Process

Surveillance and monitoring activities would continue at current levels. Spare DST space would be managed to receive waste from other DSTs in the event of a tank failure. Site services (security, fire protection, environmental monitoring, and utilities) would continue during this 100-year period.

Figure B.3.1.1 No Action Alternative



B.3.1.4 Implementability

This alternative would be a continuation of current operations and as such would not present any specific process uncertainties. There is some uncertainty in estimating the corrosion life of the DSTs. Their current design life is 50 years, but in practice some tanks may last longer.

This alternative would not comply with Federal and State requirements for storing hazardous waste. When administrative control is assumed to be discontinued after 100 years, the waste left in place would not comply with State and Federal (including DOE Order 5820.2A) requirements for disposal of hazardous, radioactive, or mixed waste (Volume One, Section 6.2). Extensive additional characterization would be required to address RCRA land disposal requirements if waste was left in place.

B.3.2 LONG-TERM MANAGEMENT ALTERNATIVE

B.3.2.1 General Description

The Long-Term Management alternative for the tank waste would consist of continued safe management of the tank farms. While no remediation of the tank waste would occur, DOE would continue to monitor and maintain the tanks and support facilities and perform those measures necessary for long-term safe storage of the waste. This alternative is similar to the No Action alternative except that DSTs would be replaced as needed to prevent the release of liquid waste.

Maintenance activities would include monitoring the tanks and upgrading instrumentation and ventilation equipment. Administrative controls would be maintained to prevent inadvertent human intrusion. Because it is not reasonable to assume that safe management can be maintained forever, a time must be assumed when the management of the tank farms would end. For the purpose of assessing impacts, it was assumed that administrative controls would be effective for 100 years. The 100-year administrative control period is being used to provide a consistent basis for assessing potential human health and environmental impacts. DOE and Ecology currently have no policies or plans that would permit the loss of administrative control for radioactive and hazardous materials.

Saltwell pumping of the SSTs is an ongoing operation that is scheduled to be completed in the year 2000. Because the majority of the free liquid in the SSTs will have been removed, this waste will no longer represent a threat of releasing liquid to the groundwater until a point in the distant future. Therefore, no additional management action other than monitoring and maintaining the tanks is needed for the SSTs during the 100-year administrative control period.

The DSTs have an estimated design life of 50 years. The tanks would need to be replaced to prevent leaks and continue the safe management of the tank waste. Tank conditions would be continually monitored, and those tanks determined to be at the risk of failure would be replaced. Existing MUSTs would be left in place and monitored similar to the SSTs. For evaluation purposes, it is assumed that the existing DSTs would be replaced at the end of their existing design life (in approximately 50 years) and again 50 years after that, just prior to the end of the administrative controls (Figure B.3.2.1).

For evaluation purposes, it is assumed that 1 percent of the existing waste volume in each tank would remain in the old DSTs. A permanent marker would be erected around the empty tanks, and security and facility controls would be maintained to protect workers and the public for 100 years.

B.3.2.2 Facilities to be Constructed

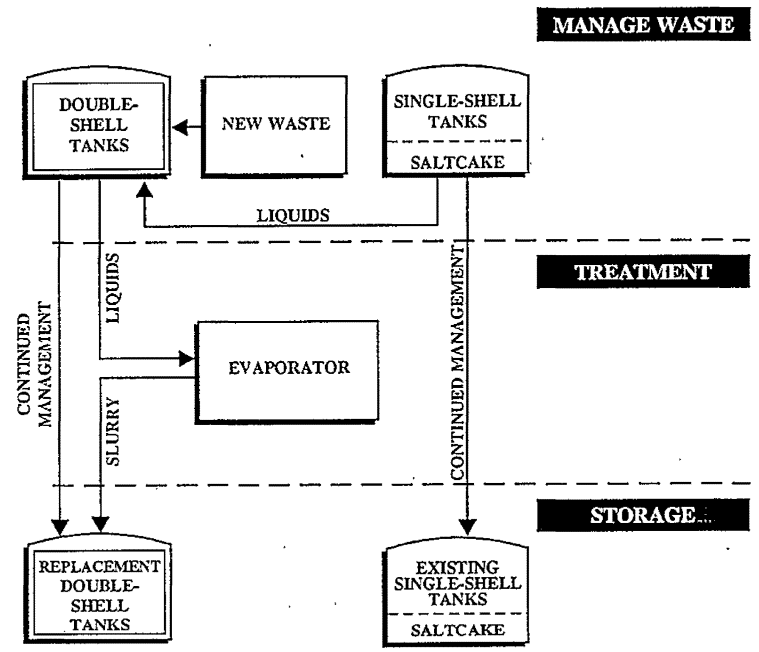
SST Facilities

Under the Long-Term Management alternative, no new facilities would be required for the SSTs. Saltwell pumping would be completed and the remaining waste, sludge, and saltcake would remain in place.

DST Facilities

Upgrades to the waste transfer system under Project W-314 would be constructed as described in Section B.3.0.2.

Figure B.3.2.1 Long-Term Management Alternative



Twenty-six new DSTs would be constructed at the end of the design life of the existing tanks in approximately 50 years. This includes one spare tank for emergency use. Each new DST would consist of a primary steel tank to contain the tank contents and an outer secondary steel shell to contain any potential leaks. The double-shelled system would be supported by an outer concrete shell designed to sustain all loads. The annular space between the primary and the secondary steel shells would be equipped with ventilation piping, pumping equipment, leak detection devices, and inspection equipment. A supporting pad under the primary tank would be slotted for ventilation air flow and leak detection. Instrumentation would be added to monitor for temperature, corrosion rate, pressure, gaseous content in the headspace, and radiation (WHC 1995g).

The primary tank ventilation system would maintain a negative pressure differential in the tank and exhaust gases from the tank vapor space to the atmosphere following treatment. In sequence the gases would pass through a condenser, mist eliminator, heater, HEPA filters, and gas adsorption filter.

The annulus ventilation system would remove heat from the primary tanks walls and floor. The exhaust gases would pass through HEPA filters before being released to the atmosphere. A continuous air monitor would indicate leaks of any radioactive material. Support facilities would include an administration building, gas sampling facilities, and stack monitoring facilities.

Two mixing turbine pumps and one transfer pump would be installed in each DST. A mixer pump is a specially-designed, vertical submersible pump. The pump suction would draw liquid from the 6-m (20-ft) level of the tank and re-inject the liquid at the 0.7-m (2-ft) level of the tank. Operating at 1,000 revolutions/minute, the pump would inject 8,300 L/min (2,200 gal/min) of waste slurry through two opposed nozzles.

A transfer pump would be a vertical long-shaft slurry pump. The pump suction would draw slurry from the bottom of the tank, and the pumping action would force the slurry through the discharge piping, which would exit through the tank riser. Other required items would include booster pumps, jumpers, dilution system, tank cooling equipment, and instrumentation.

Separate incoming and outgoing transfer lines would connect the new DSTs with existing facilities and new evaporators. All process piping, drain and sample lines, and condensate lines would be encased in double-walled piping to collect and detect leaks from the primary piping. Pipelines would be insulated to minimize temperature drops during process transfers. Valve pits would have appropriate shielding and leak detection capability. Instrumentation would provide automatic shutdown to guard against potential releases.

The new evaporators associated with the retankings would be vertical thermosiphon evaporators equipped with mechanical blowers to provide vacuum. The vacuum within the evaporator would allow it to concentrate liquid at a lower temperature.

B.3.2.3 Description of the Process

The Long-Term Management alternative would continue the monitoring and safe storage of the tank waste that are currently in the SSTs, DSTs, and MUSTs for a period of 100 years. The process that would be used for the SSTs and MUSTs would be different from that used for the DSTs because the DSTs contain large volumes of liquid waste, whereas the SSTs contain small volumes of free liquid.

SST Process

SSTs would remain in an interim stabilized condition and would be monitored following the completion of saltwell pumping and intrusion prevention activities, as described in Section B.1.1.7. The SSTs would contain less than 190 m^3 (50,000 gal) of interstitial liquid and less than 19 m^3 (5,000 gal) of drainable or free liquid.

Structural analysis of tank design and laboratory testing of concrete samples from SSTs showed the slight probability of tank dome failure before loss of institutional control from deterioration or earthquake-induced forces. SST monitoring activities would include continued monitoring of tank dome elevations. Maintenance on the tank and support structures would continue, and risers and other openings into the tanks would continue to be capped in an effort to isolate the tanks. Drywell monitoring would continue, and upgrades to the drywells would be made as necessary.

Surveillance under the Long-Term Management alternative would be provided appropriate to the degree of isolation of the tanks. Thus, surveillance would continue at the current level until the adequacy of isolation procedures could be confirmed. Site services (security, fire protection, environmental monitoring, and utilities) would be maintained at current levels.

DST Process

Surveillance and monitoring activities would continue at current levels. Spare DST space would be managed to receive waste from other DSTs in the event of a tank failure.

The DST waste is mainly liquid, and consequently, a tank leak to the gravel from a DST (both shells failing) would represent a greater threat to the environment than a tank leak from a SST due to the potential for a greater volume of waste to migrate to the groundwater. The DST waste would be removed and transferred into new DSTs at intervals corresponding to the 50-year design life of the tanks. The design life corresponds to a minimum length of service time that a tank would be expected to remain functional. The DSTs could remain functional for more than 50 years. The first retanking campaign would begin in the year 2037, which corresponds to 50 or more years of service for the 28 existing DSTs. The DSTs were put into service between 1971 and 1987, and the first retanking campaign would correspond to using the full 50-year service life of the DSTs placed into service in 1987. Monitoring and maintenance activities would continue to ensure safe storage of waste in those DSTs that would exceed the 50-year design life. This would include maintaining spare DST space and leak recovery from the annulus (space between the inner and outer liner) of the DSTs. For each retanking campaign, 26 new 3.8E+06-L (1.0E+06-gal) DSTs would be required to replace the existing tanks. A total of two retanking campaigns would be required during the 100-year administrative control period.

DST waste would be transferred from old DSTs to new DSTs after construction was completed. Supernate would be transferred directly to new tanks. The slurry (sludge diluted with liquid at a 3-to-1 ratio) would be concentrated in an evaporator before being transferred to the new DSTs to maintain waste volumes. The evaporator condensate would be used as a dilution liquid for other DST retrieval operations. At the end of the retrieval period the evaporator condensate would be treated and released. Site services (security, fire protection, environmental monitoring, and utilities) would continue during this 100-year period.

B.3.2.4 Implementability

This alternative would be a continuation of current operations and previously performed activities (tank construction) and as such would not present any specific process uncertainties. There is some uncertainty in estimating the corrosion life of the DSTs. The design life of the current DSTs is approximately 50 years. Many tanks are expected to exceed their design life; however, a structural integrity assessment has not been completed to date.

This alternative would not comply with Federal and State requirements for storing hazardous waste. When administrative control is assumed to be discontinued after 100 years, the waste left in place would not comply with State and Federal (including DOE Order 5820.2A) requirements for disposal of hazardous, radioactive, or mixed waste (Volume One, Section 6.2). Extensive additional characterization would be required to address RCRA land disposal requirements if waste was left in place.

B.3.3 IN SITU FILL AND CAP ALTERNATIVE

B.3.3.1 General Description

The In Situ Fill and Cap alternative would consist of evaporating all pumpable liquid from DSTs in the 242-A Evaporator, filling tank voids in SSTs and DSTs with gravel, and constructing Hanford Barriers over the tank farm areas. Reducing the volume of liquid waste in the DSTs would be accomplished by sending the pumpable liquid to the existing 242-A Evaporator for concentration and return to the DSTs. All SSTs would have all pumpable liquids removed and would not require liquid evaporation. The tanks would then be filled with gravel using a centrifugal thrower. Gravel fill serves several purposes. In the event of a catastrophic collapse of the tank, gravel would minimize subsidence, reduce impact on the barrier system, and reduce exposure of the waste. In addition, gravel would provide an isolating layer between the surface barrier and waste.

As part of closure, a multi-layer barrier (such as a Hanford Barrier) consisting of layers of basalt riprap, gravel, and soil would be constructed over the tanks to reduce the infiltration of precipitation and inhibit intrusion by humans, plant roots, and burrowing animals. Surface and subsurface markers would be used to mark the location. Security and administrative controls would be implemented and maintained indefinitely to protect workers and the public. For the purpose of calculating the maximum potential impacts it is assumed that the controls will be terminated after 100 years. This alternative was developed from technologies identified during the scoping process.

B.3.3.2 Facilities to be Constructed

Evaporator Facilities

The evaporator facilities would consist of the existing 242-A Evaporator, which would require upgrading the heating, ventilating, and air conditioning (HVAC) system and replacing the pump before evaporating the volume of liquid in the DSTs.

Gravel Plant

Gravel would be excavated from the potential Pit 30 borrow site (between the 200 West and 200 East Areas). The gravel would be stored in four stockpiles, each 60 by 60 m (200 by 200 ft). Each stockpile would serve the following clusters of tank farms:

200 East Area	1. 241-A, 241-AX, 241-AW, 241-AP, 241-AY, 241-AZ; and 2. 241-B, 241-BX, 241-BY, 241-C, 241-AN.
200 West Area	3. 241-T, 241-TX, 241-TY; and 4. 241-S, 241-SX, 241-U, 241-SY.

Each stockpile would be serviced by a loader and a conveyor network that would service one tank at a time. Because the time required to fill a tank would be no more than a few operating days, most conveyor runs would be assembled from mobile sections and repositioned as required.

Two portable gravel distributing systems would be constructed. One system would consist of a centrifugal thrower, a feed hopper, a feed conveyor, a 30-m^3/min (1,000-ft^3/min) exhauster, a dust collection and HEPA filtration system, and a containment enclosure measuring 4 by 4 by 3 m (13 by 13 by 10 ft). Figure B.3.3.1 shows how the gravel would be handled. Contaminated equipment would be decontaminated and placed in a low-level waste burial ground.

B.3.3.3 Description of the Process

242-A Evaporator

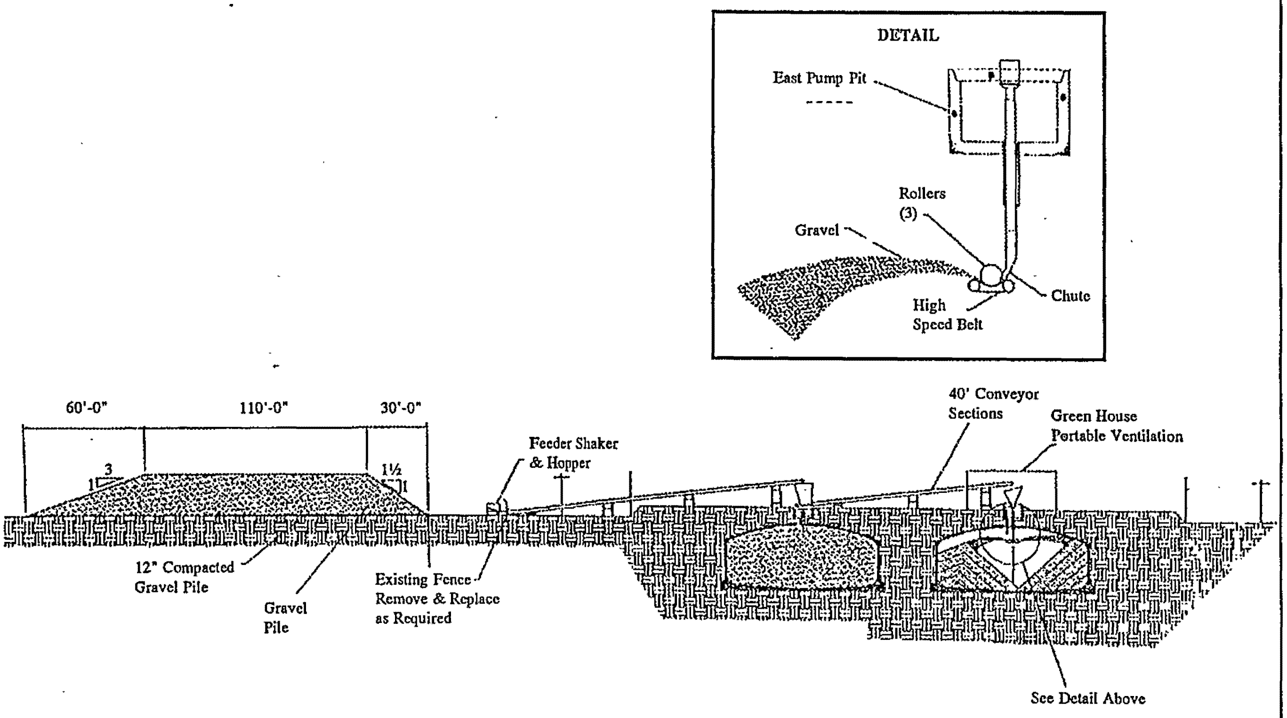
Pumpable liquid from all DSTs and the liquid removed from the SSTs during saltwell pumping would be pumped to the 242-A Evaporator where excess water would be removed by vacuum evaporation. The resulting slurry would be routed to an underground DST for settling. Supernatant liquid would be recycled to the 242-A Evaporator until the desired product quality was attained. The concentrated waste then would be returned to the DST tank farm at an elevated temperature as a slurry. Excess condensate from the process would be routed to the 200 Area Effluent Treatment Facility. Noncondensable gases would be passed through a HEPA filter system and discharged to the atmosphere. All effluents would meet discharge limits.

Gravel Fill

Each tank would be filled with gravel to prevent future collapse of the tank dome. Uniformly-graded basalt gravel with a particle size of 1 to 2 cm (0.4 to 0.8 in.) would be used to fill the tank domes for the following reasons.

- Uniform particle size is desirable because particle trajectories and velocities would vary with particle mass. A poorly sorted (nonuniform) material would result in nonuniform material distribution.
- Material distribution becomes more difficult with increasing particle size.
- The gravel fill process would distribute uniform-sized crushed rock throughout the void space of the tank. Gravel filling is a commercially-proven technology for subsidence control, and has been tested at the Hanford Site to verify using the technology with local materials in a tank-like environment.

Figure B.3.3.1 In Situ Fill and Cap Arrangement Schematic



SOURCE: Adapted from WHC 1995f

Tanks currently contain a variety of equipment such as purge tubes, suspended and anchored air lift circulators, and failed pumps. All in-tank equipment would be evaluated regarding its potential to impede the distribution process or create undesirable voids. If unacceptable, equipment would be removed or multiple distributors would be required to fill around obstacles. However, using multiple distributors could require additional risers in the tank dome. Additional risers could also be needed for monitoring equipment. Installing the distributor(s) would require modifying existing pits and risers. Monitoring equipment and instrumentation would have to be placed within the tank before filling. All tank preparation work would occur before fill activities. Because gravel fill operations would displace air from the tanks, a portable confinement structure, approximately 4 by 4 by 3 m (13 by 13 by 10 ft), would be installed to control air emissions.

Gravel would be distributed within the tank by a thrower or slinger. A fast-moving horizontal belt would capture and throw the gravel. The throwing mechanism, which would be suspended within the tank typically from the center riser, would rotate to throw the gravel. The belt speed, belt angle, gravel-feed rate, and rotational speed would be the primary controlling parameters. A hopper mounted directly above the distributor would be fed from the conveyor system. The hopper would in turn feed the distributor through a quick-acting isolation valve. The valve would not control feed flow to the distributor but would isolate the distributor from the ambient environment, should the tank differential pressure become unstable.

Feed to the hopper would work in conjunction with the isolation valve. An enclosure would be placed around the distributor and hopper assembly to serve as a confinement buffer; however, it would not be considered a confinement zone. There could be conditions that require using more than one distributor in a tank, such as installed hardware that could not be removed. The availability of existing risers versus the difficulty of installing new risers could also drive the decision to use more than one distributor. These distributors would be somewhat smaller but operate similarly to the larger center-mounted unit. It is presumed that all tanks would use center-mounted enclosed distributors. Sacrificial material vibrators may be strategically placed within the tanks to ensure maximum fill in critical areas.

The HVAC system would provide sufficient capacity and controls to ensure that process operations could not upset the tank air pressure differential. Because gravel filling would generate a considerable volume of airborne particulates, a series of cyclone separators would be used to separate and remove the particulates from the exhaust stream before passing through prefilters and a dual-stage HEPA filter system. The collected particulates would be removed, placed in drums, and considered low-level waste.

The success of the filling operations would depend on the ability to demonstrate and verify that the required fill distribution and uniformity was achieved. The fill-monitoring system would have the capability to remotely view interior tank operations. In addition, fill surface elevation measurements and mapping would be available in real time to document the progress of the operation and verify results. To verify fill integrity, density and compaction would be measured.

Post Remediation

Post remediation would involve decontaminating and decommissioning all equipment and facilities constructed for the alternative. MUSTs and ancillary tank farm equipment would be filled with grout. The final step of in situ disposal would be installing Hanford Barriers to reduce the infiltration of precipitation and inhibit intrusion by humans, plant roots, and burrowing animals. The closure barriers would be horizontal above grade engineered structures that isolate the waste site from the accessible environment.

B.3.3.4 Implementability

The primary issues associated with implementing the In Situ Fill and Cap alternative are as follows.

- The moisture content in the remaining tank solids is not known at the present time. Following tank filling operations, oxidizing chemicals would still be in contact with organics, though there is generally a positive effect observed with aging of the chemical materials. The possibility of spontaneous reactions over long periods of time or the presence of compounds that could be initiators or catalysts would need to be resolved by further investigations.
- Other slow decomposition reactions can occur in the absence of water. The possibility of generating hydrogen or other flammable gases is an issue that would require further investigation. Each tank would need to be evaluated individually to determine that this alternative could be safely implemented.

This alternative would not meet the land disposal requirements of RCRA for hazardous waste. Near-surface disposal of HLW would not meet DOE Order 5820.2A requirements for disposal of readily retrievable HLW in a potential geologic repository (see Volume One, Section 6.2).

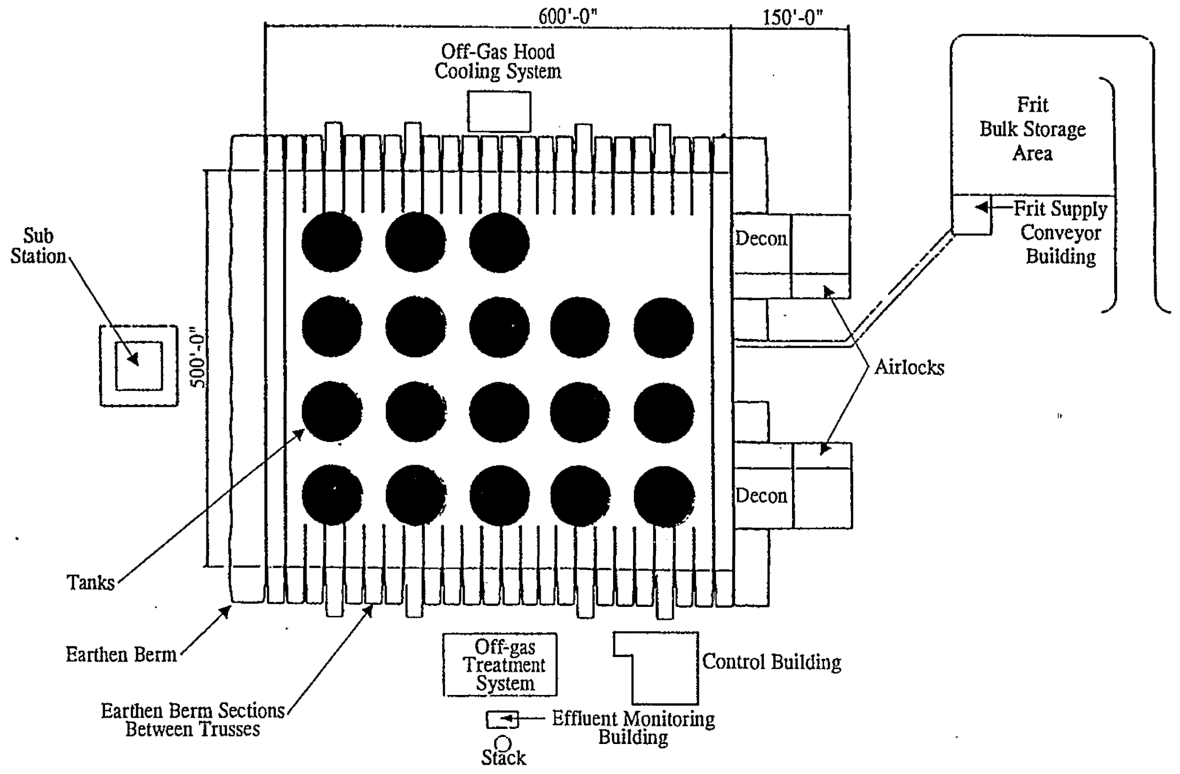
B.3.4 IN SITU VITRIFICATION ALTERNATIVE

B.3.4.1 General Description

The In Situ Vitrification alternative would involve immobilizing the waste in the tanks. Large structures would be built over the tank farms to provide containment and control airborne releases during the in situ vitrification process. Silica in the form of sand would be added to the waste and electrodes would be inserted into the waste. Electrical current would be applied until the waste and silica are vitrified (melted). The vitrified waste subsequently would cool into a glass-like material (estimated temperature range from 1,450 to 1,600 °C [2,600 to 2,900 °F]). The in situ vitrification process would include pollution abatement controls to ensure that all effluents and emissions are within regulatory standards (WHC 1995f).

Figures B.3.4.1 and B.3.4.2 show three views of a typical in situ vitrification facility layout and Figure B.3.4.3 shows the flow diagram for in situ vitrification.

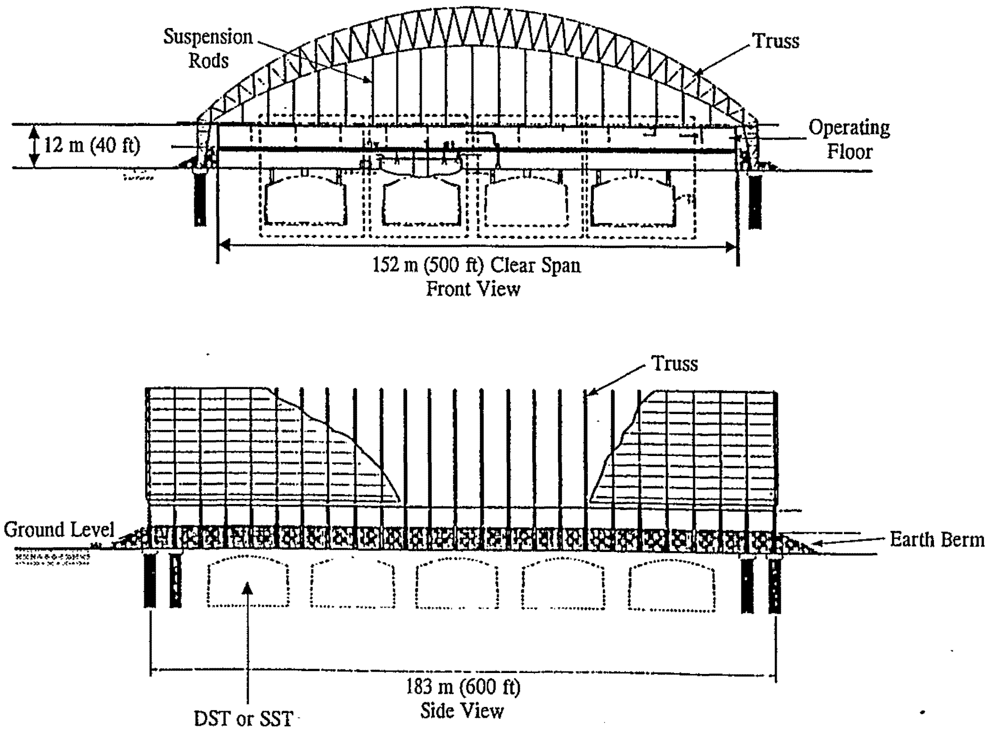
Figure B.3.4.1 Tank Farm Confinement Facility Plan



Not to Scale

SOURCE: WHC 1995f

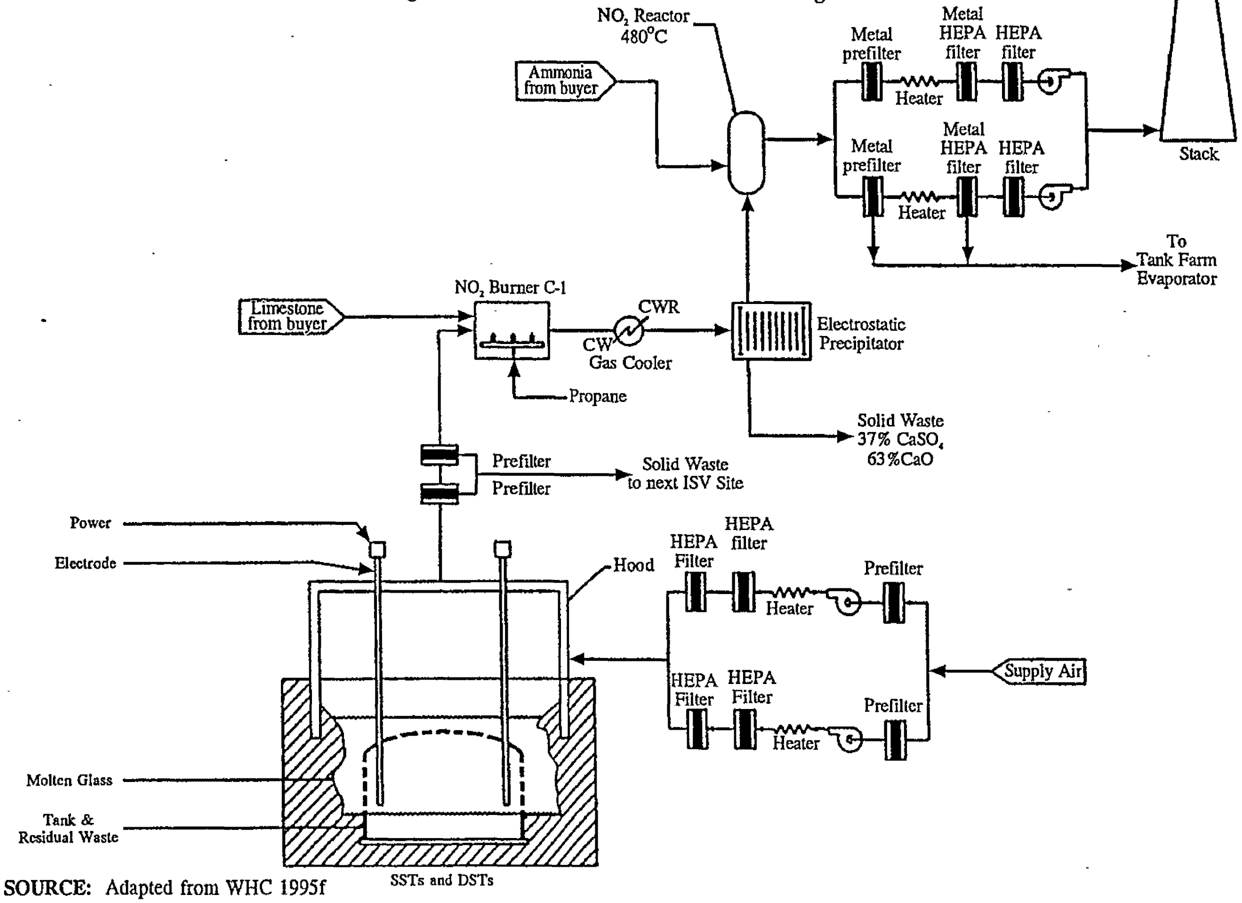
Figure B.3.4.2 Tank Farm Confinement Facility Elevations



Not to Scale

SOURCE: WHC 1995f .

Figure B.3.4.3 In Situ Vitrification Flow Diagram



SOURCE: Adapted from WHC 1995f

The vitrification process would be a large-scale in situ vitrification process capable of vitrifying an entire tank at a time. The largest scale in situ vitrification equipment currently available produces a melt 15 m (50 ft) in diameter, 6 m (20 ft) deep, and requires 3.5 megawatts (MW) of power. A development and testing period would be necessary before the in situ vitrification option could be implemented. Development and testing have been estimated to take 7 years. Current research and development efforts are addressing depth-enhancement techniques that would make it possible to reach the 15- to 18-m (50- to 60-ft) depths required for the tanks (WHC 1995f).

Vitrification would densify the soil and waste by eliminating the interstitial space between particles creating a depressed area over the vitrified mass. Interstitial space typically ranges between 5 to 30 percent of the volume depending on the initial compaction of the soil or waste. After vitrification the depressed area would be filled with soil and covered with a Hanford Barrier, which would isolate the vitrified waste from the environment.

As part of closure, a multi-layered barrier (such as a Hanford Barrier) consisting of layers of basalt riprap, gravel, and soil would be constructed over the tanks to isolate them. Surface and subsurface markers would be used to mark the location of the tanks. Security and administrative controls would be implemented and maintained for 100 years to protect workers and the public. A more detailed description of the Hanford Barrier is contained in Section B.6.0.

B.3.4.2 Facilities to be Constructed

Tank Farm Confinement Facility

A tank farm confinement facility (TFCF) would be constructed over an entire tank farm so that the tank farm area, including the space between the tanks, could be vitrified. The TFCF would consist of structural steel members comprising a dome-like structure that would provide structural support for in situ vitrification equipment. The TFCF would support the flux or frit handling system and the electrode system, and would contain and collect the off-gas, including the volatile components of the waste. The TFCF would provide a large confinement volume for controlling any releases from operations in the tanks. This facility would provide unrestricted overhead access to all parts of a tank farm and the ability to conduct multiple operations simultaneously within a single confinement structure.

The TFCFs would be sized for five typical farms with tank-by-tank arrays of 2 by 2, 2 by 3, 3 by 4, 3 by 5, and 4 by 5. For the largest tank farm (4 by 5), the support structure would span the width of the farm, 165 m (540 ft). The suspended confinement structure would be 12 m (40 ft) high, 150 m (500 ft) wide, and 180 m (600 ft) long. The 2 by 2 TFCFs would be used for those tanks farms with four or fewer tanks.

The confinement structure would contain a single operating deck, 6 m (20 ft) above grade, made of reinforced-concrete panels. The operating deck, which would be used as a platform for most activities, would support moveable shielding, materials, and personnel. Operating equipment would be suspended beneath the deck where it could be moved to perform a variety of tasks. The roof of the confinement structure would support operating equipment not required on the operating deck. Partial decks or subdecks would be located just below the roof to carry services and utilities.

Moveable-walled (or buffer) areas, each 30 by 30 m (100 by 100 ft) would provide the operating environment for tank-specific activities. Similar walled areas would be located below deck at ground level. No more than three buffer areas would be erected at any given time. Their locations would be determined by operation sequencing and building load control rules. Utility distribution systems and services needed for operations would be flexible (trays, subdecks/floors, and plugs) to service the entire operating deck. Hard points or mounts would be located below the roof and operating deck for installing cranes, trolleys, hoists, rails or other load-bearing equipment.

While the TFCF would be designed to allow all materials exposed to radioactivity to be kept below the operating deck level, provisions would be made to allow some contaminated equipment properly confined and shielded to be brought through the operating deck and transferred out of the facility. Other contaminated materials, effluents, and off-gases would be handled by collection systems suspended below the operating deck. Personnel access to ground level would be provided through airlocks and personnel lifts.

The TFCF would contain two primary confinement zones. The area below the operating deck would be Zone 1, at the highest negative pressure differential with respect to the atmosphere; the area above the operating deck would be Zone 2, at a lower differential pressure. Air to both Zone 1 and 2 would be supplied by separate inlet supply units located adjacent to the structure. Exhaust from Zones 1 and 2 would be conditioned and filtered in facilities adjacent to the structure.

The walled or buffer areas in Zone 2 would contain a vent system to condition the air for occupational use while no differential pressure between the buffer areas and Zone 2 would be maintained. The exhaust would pass through HEPA filters and be monitored before being released to the atmosphere. The walled areas in Zone 1 would contain a vent system that would draw air from Zone 1, and filter, condition, and exhaust it through HEPA filters back to Zone 1. Although the buffer exhaust would pass through HEPA filters, the buffer areas would not be considered separate confinement zones. Each buffer area would operate at the pressure of the parent zone. The volumes of the proposed HVAC system zones include the following:

- Zone 1 system - 1,100-m^3 (40,000-ft^3) HEPA-filtered supply and exhaust;
- Zone 2 system - 570-m^3 (20,000-ft^3) HEPA-filtered exhaust;
- Buffer system - Three required, 280 m^3 (10,000 ft^3) with air conditioning for occupied areas in Zone 2 and HEPA exhaust; and
- Buffer system - Three required, 280 m^3 (10,000 ft^3) HEPA exhaust, no conditioning in Zone 1.

The arched truss system and foundation, which would support the confinement structure, would not have fire protection. However, all load-carrying structural members located in the confinement structure and connecting the confinement structure to the truss system would be wrapped with insulation to provide a 2-hour fire rating. All electrical conductors on the roof deck and above would be similarly wrapped.

Fire protection for the interior of the confinement structure would be provided by separate systems along the lines of confinement zones. Zone 1 (ground level) and Zone 2 (operating deck) would be protected by dry-foam water-spray systems rated for a 4-hour fire. Each zone or level would be serviced by independent systems. Water and foam delivery systems including reservoir, pumps, and piping would be sized for a 4-hour fire. Support facilities would be serviced by conventional wet pipe systems.

Shielding would be provided to ensure that operational exposure limits for operations personnel would be satisfied. Baseline exposure range would be calculated using a bounding case tank source at various elevations (0, 3, and 6 m [0, 10, and 20 ft]) aboveground.

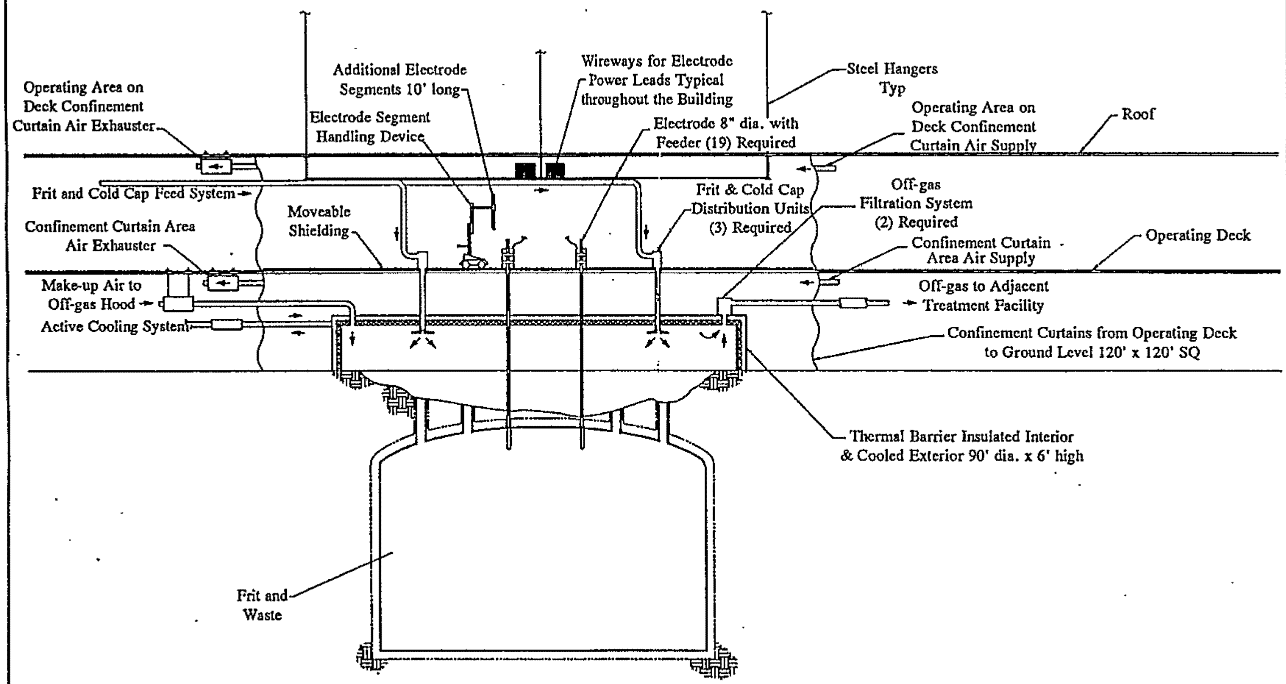
The confinement structure would be designed to accommodate the static and dynamic loadings of all equipment and related operations and would not impose any static or dynamic load on the underground tank structure. In no way would the tanks be relied on for confinement structure support. Equipment handling provisions would be integrated with the confinement structure so that contaminated operations equipment could be safely transferred to and from the facility. All contaminated equipment would be transported in containers and shielded overpacks, which would provide two levels of containment.

The area under the operating floor would be equipped with thermal barriers within the lower buffer zone to absorb radiant energy and to control gases from the melt. This multi-purpose thermal barrier, which would measure 27.5 m (90 ft) in diameter and 1.8 m (6 ft) high, would be placed between the in situ vitrification site and the TFCF. The thermal barrier exterior would be equipped with an active cooling system, and the thermal barrier interior would be lined with insulation to provide a thermal differential from the melt's surface to the operating deck. Figure B.3.4.4 illustrates the layout of the thermal barrier and the integration with its services.

A number of systems would be integrated with the thermal barrier. The electrodes would pass through the barrier into the melt, and a material feed system would penetrate the barrier for the dome-fill process. The fill materials would be distributed within the TFCF via conveyor, then conveyed via ducts through the operating deck, and fed by rotary valves through the barrier into the tank.

The thermal barrier would be serviced by an off-gas treatment system comprised of a supply and exhaust system, local particulate filters integrated with the barrier, an insulated duct network for off-gas to traverse the facility, and a treatment system to treat the air stream prior to release. The geometry of the barrier would help collect off-gas and entrained particulate directly from the melt. The supply

Figure B.3.4.4 In Situ Vitrification Arrangement



SOURCE: WHC 1995f

system would draw from the confined zone above the tank into the barrier. The exhaust and treatment systems would be located adjacent to the TFCF. Local filters would be replaceable with spent filters disposed of into the melt, and replacements would be passed through ports in the operating deck. Similar pass-through concepts would be used to replace other consumables or failed components.

Because of the conceptual nature of the engineering data package for in situ vitrification (WHC 1995f), the actual methods for decontaminating and decommissioning the process facilities has not yet been determined. In general, processing equipment will be decontaminated and removed to a low-level waste burial ground. The remaining processing facilities will be decontaminated to the extent possible and entombed in place, with a material to be decided upon.

Support Facilities

Support facilities would be located outside the TFCF. A 23- by 46-m (75- by 150-ft) off-gas processing system is the largest of the support facilities. Other support facilities would include:

- Frit bulk storage area - 80 by 60 m (260 by 200 ft);
- Frit supply conveyor building - 15 by 15 m (50 by 50 ft);
- Frit supply conveyor - 105 by 3 m (345 by 10 ft);
- Electrical substation - 16 by 18 m (53 by 60 ft) to bring in the overhead high voltage line;
- Effluent monitoring building - 12 by 6 m (40 by 20 ft);
- Tank farm retrieval unit annex (control room and operations support) - 31 by 23 m (100 by 75 ft);
- Two off-gas hood decontamination stations - 23.2 by 22.9 m (76 by 75 ft) each;
- Two off-gas hood repair and loadout stations; and
- Eight HVAC exhaust systems - 6.1 by 18.3 m (20 by 60 ft) each.

Crushing Facility in Pit 30

Excavating and crushing equipment in the potential Pit 30 borrow site, located between 200 East and 200 West Areas, are assumed to currently be in place from other Hanford Site operations. Consequently, no additional facilities would be necessary to excavate the 540,000 m^3 (714,000 cubic yard [yd^3]) of sand required for filling the tank void spaces.

B.3.4.3 Description of the Process

The in situ vitrification process consists of 1) removing all pumpable liquid from the tanks; 2) completely filling the tank dome spaces with Hanford Site sand; 3) vitrifying the waste in place; and 4) installing protective barriers over the vitrified waste.

Evaporation

Pumpable liquid from all DSTs and the liquid removed from the SSTs during saltwell pumping (9.8E+07 L [2.6E+07 gal] total) would be pumped to the existing 242-A Evaporator where excess water would be removed by vacuum evaporation. The resulting slurry would be routed to an existing

DST. Supernatant liquid would be recycled to the 242-A Evaporator. The concentrated waste then would be returned to the tank farm DST at an elevated temperature as a slurry. Excess condensate from the process would be routed to the 200 Area Effluent Treatment Facility. Noncondensable gases would be passed through a HEPA filter system and discharged to the atmosphere. All effluents would meet Federal and State discharge limits for controlled areas.

It is estimated that the 242-A Evaporator would require 5 years to process 9.8E+07 L (2.6E+07 gal) of waste. This is based on the 242-A Evaporator's previous operating rate of 2.3E+07 L (6.0E+06 gal) per year. An allowance has been included in the operating cost estimate for upgrading the 242-A Evaporator HVAC system and replacing the existing pump.

Vitrification

In situ vitrification would use joule-heating to melt the waste into a vitreous monolithic mass. High temperatures during vitrification would cause a number of the waste constituents (nitrates and organics) to decompose, leaving oxides of nonvolatile compounds in the matrix. In situ vitrification would provide a stable matrix and an extremely insoluble waste form for the long-term disposal of tank waste. Figure B.3.4.3 shows the flow diagram for in situ vitrification of a tank farm.

Before beginning the vitrification operation, each tank would be electrically isolated from all support systems. This would include disconnecting and removing the piping, instrumentation wiring, and ventilation systems that are shared with other tanks. This would prevent potential accidents and damage because of stray electrical current.

An array of electrodes would be inserted through the off-gas containment hood and would contact the soil surface above a tank. A conductive mixture such as glass frit (or sand) and graphite would be placed between the electrodes to act as a starter path, because soil has a low electrical conductivity when its moisture has been driven off. An electrical potential would be applied to the electrodes to establish an electrical current in the starter path. The resultant power would heat the starter path and surrounding soil well above the soil-melting temperatures of 1,100 to 1,400 °C (2,000 to 2,500 °F). The starter path would quickly be consumed by oxidation, and the current would then be transferred to the molten soil, operating between 1,450 and 1,600 °C (2,600 to 2,900 °F). As the molten zone grows, it would incorporate any radionuclides and nonvolatile hazardous elements that may be present. The high temperature of the process would destroy organic components by pyrolysis. The pyrolyzed by-products would migrate to the surface of the vitrified zone where they would combust in the presence of air. A hood placed over the area being vitrified would direct the gaseous effluents to an off-gas treatment system.

Void spaces in the tanks would disrupt the vitrification process. Therefore, the void space above the waste would be filled with sand excavated onsite. Testing would determine the particle size distribution of the sand. About 540,000 m^3 (714,000 yd^3) of fill material would be added, compared to 230,000 m^3 (304,000 yd^3) of waste currently in the tanks.

A schematic arrangement for in situ vitrification is shown in Figure B.3.4.4. For the standard 23-m (75-ft) -diameter tank, 19 electrodes approximately 30 cm (1 ft) in diameter would be used. They would be mounted at corner points in an array of multiple triangles approximately 5.5 m (18 ft) apart per leg. Similar arrays would be used for the soil areas between tanks. A 28-m (90-ft) -diameter off-gas hood would be placed over the tank, and electrical and off-gas connections would be made. The electrodes would feed through the hood as the melt progressed. Each electrode would be lowered through sealed penetrations in the operating deck at a controlled rate using a feeder mechanism. Power would be provided to the electrodes through a collar that would be part of the feeder. The power and feeder controls would be moveable to all in situ vitrification locations within the TFCF. Transformers would be centrally located adjacent to the TFCF. Vitrification would continue until a depth of about 20 m (66 ft) is reached. The maximum depth of the tanks is 15 m (49 ft). Each tank farm would have two in situ vitrification setups and share an off-gas system. Four in situ vitrification systems would be onsite, with at least two systems operating at all times.

The tank melt zone width and depth would normally be 24 m (80 ft) and 18 m (60 ft), respectively, while plume melt zone width and depth could reach 43 m (140 ft) and 30 m (100 ft). Plume treatment would require a greater width to efficiently treat contamination. The depth may increase if testing shows a need for deeper treatment. However, melting an entire tank farm would result in a vitreous cap over any underlying contaminated soil.

Venting the tanks to the TFCF during the in situ vitrification process would be controlled at least two ways. One method would create penetrations in the tank walls and floor so that gases and vapors generated inside the tank could be vented up and around the melt. This approach has been successfully implemented in commercial in situ vitrification operations by using a vibrating beam technique to puncture arrays of buried drums and rupture the walls of a buried concrete vault. A second method would offset the melts so that only a portion of the tank is processed in a single melt setting. The gases and vapors generated beneath the melt would be vented up around the melt to the surface. The second method would leave a portion of the tank untreated initially but subsequently treat the remainder in adjacent melts.

The large-scale in situ vitrification application would require about 7E-01 kWh/kg of melt. A 225-mt/hour melter would require about 160 MW (14 percent of the Washington Public Supply System Nuclear Plant Two [WNP-2] design rating). If two melter systems are used, the in situ vitrification system would require 320 MW (27.8 percent of WNP-2 design rating). The transformer system would have variable taps to deliver constant power to accommodate the changing resistive load as the melt progresses. It is assumed that power would come from the existing electrical grid but new feed lines would be required. If at any time the grid would not be capable of handling the load, the power requirements could be reduced by operating only one in situ vitrification system rather than two.

The off-gas would contain the reaction products resulting from the thermal destruction of the nitrates, nitrites, organic compounds, and some of the more volatile radionuclides contained in the waste.

The high temperature in the melt zone would destroy the organic constituents and disassociate the bulk of the nitrogen oxides to oxygen and nitrogen. Destruction efficiencies of greater than 99 percent have been assumed for the destruction of organic compounds and nitrates with the in situ vitrification process. A thermal oxidizer would be included in the off-gas treatment train to further destroy organic compounds that would not be completely destroyed in the melt prior to atmospheric release.

The I-129 and carbon-14 (C-14) present in the tank waste would volatilize during the in situ vitrification process and would not be captured in the off-gas treatment system. The I-129 would be volatile gaseous diatomic iodine (I_2) and would pass through the prefilters and HEPA filtration system. The C-14 would oxidize into gaseous carbon dioxide (CO_2) and would also pass through the prefilters and HEPA filtration system. Other semivolatile radionuclides (e.g., Tc-99, Sr-90, and Cs-137) would escape the molten region into the off-gas but would be in particulate form and thus captured in the HEPA filters.

The in situ vitrification off-gas would be cooled through a water quench system, a venturi scrubber, a solids separator, a chiller, and a mist eliminator. The hot gases would be cooled from approximately 1,350 °C (2,460 °F) to a temperature of 30 °C (86 °F) leaving the mist eliminator. Most of the semivolatile radionuclides such as Tc-99, Sr-90, and Cs-137 would be captured from the off-gas by this scrubbing action. Even though it was assumed that all the I-129 would be volatile I_2 and would leave the system in the off-gas, the gas temperature of 30 °C (86 °F) is considerably lower than the I_2 melting point of 113 °C (238 °F). This indicates the potential for much of the I to be captured in solid form in the quench water. Additional I and other radionuclides would be captured on the metallic fiber HEPA filters, where they would be recovered in the filter wash water and become a portion of the quench tank condensate stream. The residual iodine in the off-gas exiting the filters would be removed by use of an activated charcoal bed.

The nitrogen oxides, NO and NO_2 (referred to as NO_x), which are not absorbed and recovered in the quench water condensate stream, would be converted to elemental nitrogen (N_2) and water in a NO_x catalytic reactor using ammonia. The sulfur oxides (So_x)would be removed as calcium sulfate and calcium sulfite, which would be subsequently removed from the off-gas stream by an electrostatic precipitator. This process would result in a secondary waste stream of approximately 7,300 mt (8,050 tons) of LAW that would require handling and disposal.' The condensed liquid stream would be transported to an evaporator for concentration. The concentrated waste from the evaporator would be immobilized using a low-temperature process such as grouting.

Because the tanks contain material that may react violently when heated, safely treating these materials is one major issue that will determine the applicability of in situ vitrification as a method of in situ treatment. Testing would establish acceptable concentration limits for sodium nitrate, ferrocyanide, and other reactive compounds. These compounds would be tested separately and in combination at the actual levels found in the tanks.

MUSTs and ancillary equipment located within existing tank farm boundaries would be vitrified. The remaining MUSTs and ancillary equipment would be filled with grout.

The in situ vitrification facility would incorporate remote operation and one-use maintenance concepts. If equipment fails or is no longer operable, it would be disconnected and dropped into the tank to be vitrified. Figures B.3.4.1 and B.3.4.2 show the proposed in situ vitrification facility site plan, structural elevation and sections, above and below operating deck plans, elevation, and the electrical site plan.

Post Remediation

Following vitrification, all facilities constructed for the alternative would be decontaminated and decommissioned, and ancillary equipment that was not vitrified would be filled with grout. The final step of post remediation would be installing the Hanford Barriers. These barriers would be horizontal, multi-layered, abovegrade engineered structures whose function would be to limit intrusion of rainfall through the barrier, thereby reducing the dissolution of contaminants and their subsequent migration to the groundwater. Section B.6 describes the multi-layered barrier (Hanford Barrier), which may be placed over each of the tank farms.

B.3.4.4 Implementability

Implementability of a remedial alternative is determined by two factors: the history of the demonstrated performance of a technology and the ability to conduct and operate it given the existing conditions at the site. The primary issues associated with implementing the In Situ Vitrification alternative are described as follows.

- This alternative is more conceptual in design and development than the ex situ vitrification alternatives discussed in Section B.3.0 and thus has a higher degree of uncertainty associated with costs, schedules, resource requirements, and air emissions.
- This process has not been performed at the scale described here and the equipment types and sizes are the result of estimates and engineering judgement. Consequently, there is a higher degree of uncertainty for the exact equipment that would be required. This uncertainty would be reduced by using smaller vitrification systems and multiple melts to vitrify the large tanks.
- Because in situ vitrification would be accomplished one tank at a time, knowledge of the waste composition in each tank would be necessary. The contents of each tank would be analyzed and any required fluxes would be determined to ensure that the glass product produced met specification. Similarly, the off-gas components and volume would be within design accurately estimated for each tank. The system would need to be able to safely process the bounding case tank for the process to be viable. Consequently, vitrifying an entire tank farm as postulated could occur only after all the tanks in that farm have been characterized.
- The safety of drying some of the waste types is uncertain. It is possible that hot spots created by drying may cause a self-propagating reaction between an oxidizer, such as

sodium nitrate, and any organics present. These issues would need to be evaluated and resolved before the In Situ Vitrification alternative could be implemented for all tanks.

- The capability of the off-gas handling system appears to be underestimated when compared to the more sophisticated off-gas processing postulated for the Ex Situ Intermediate Separations alternative. It is possible that the off-gas treatment system for in situ vitrification would require considerably more gas-processing equipment than proposed in the engineering data package (WHC 1995f). Disposing of the calcium sulfate that would result from this method of gas treatment has not been completely addressed. Should further development work indicate that the calcium sulfate from the off-gas system cannot simply be placed in the next tank to be vitrified, a secondary waste stream may also be generated. The generation of secondary waste is not currently considered in the alternative evaluation.
- While current commercial experience is limited to melting areas 15 m (49 ft) in diameter by 6 m (20 ft) deep, this alternative assumes an entire tank that is 23 m (75 ft) in diameter by 18 m (60 ft) deep can be vitrified. For this alternative to be viable, even using multiple melts per tank, further development work would need to be performed to demonstrate conclusively that vitrification can be completed to the required depth.
- The TFCF design is conceptual and further development would be required for it to comply with the requirements of General Design Criteria, DOE Order 6430.1A. A structural analysis of the TFCF has not been made, and the truss and beam sizes and the resulting costs required to build the facility are uncertain. The TFCF may be difficult to construct because of the atypical nature of the design and the restrictions associated with working in and around the tank farms. The conceptual TFCF design could be reduced to a size covering no more than two or four tanks and could be a portable or movable facility. Should the design fail to meet Site standards, it would need to be modified to meet Site standards or a new concept would be selected.
- Inspecting the final waste form to confirm that all of the waste is stabilized and the waste form is acceptable for disposal would be difficult to perform. Because the technology is not sufficiently mature, the performance assessment would rely heavily on assumptions and be difficult to actually perform. At present, insufficient development work has been done to provide the criteria for successful vitrification. Systems or methods would need to be developed to sample the vitrified material to ensure quality.
- The physical size of the off-gas system to handle the proposed rate of processing, which is 225 mt/hr, would be about 27 times the size of the off-gas system for the ex situ vitrification process at a rate of 200 mt/day. Several of these units would be required because of the physical separations between the various tank farms. The physical size would preclude using a trailer mounted design for the off-gas processing. The full impact of construction costs are not known and cannot be incorporated at this time.

- Reprocessing waste that fails to meet disposal criteria would require remelting the waste. Remelting a mass as large as a waste tank could be expected to require as much time as the initial melting. Remelting is not currently included in the alternative. One effect would be to extend the schedule proposed in the engineering data package (WHC 1995f).
- Reprocessing may not be effective if deep portions of the vitrified mass need additional flux. Further development work is required to ensure that remelting will generate sufficient thermal agitation to stir the melt zone and mix the required flux throughout the melt.

This alternative may meet the RCRA land disposal requirements if hazardous waste is adequately treated during vitrification. Near-surface disposal of HLW may not meet DOE Order 5820.2A (DOE 1988) requirements for disposal of readily retrievable HLW in a potential geologic repository (Volume One, Section 6.2).

B.3.5 EX SITU INTERMEDIATE SEPARATIONS ALTERNATIVE

B.3.5.1 General Description

Ex situ alternatives would require removing the waste from the tanks for treatment and separating the waste into high- and low-level components. The benefit of separating the waste would be to minimize the volume of HLW requiring offsite disposal and reduce the amount of radioactivity for disposal in near-surface vaults onsite. Ex situ alternatives would dispose of HLW at a potential geologic repository, which is assumed to be at Yucca Mountain, Nevada (see Section B.10 for further discussion).

This alternative involves retrieving as much of the waste as practicable from the tanks and separating it into HLW and LAW streams. Each waste stream would be vitrified into glass. The HLW would be transported offsite to the potential geologic repository and the LAW would be placed in retrievable near-surface disposal vaults at the Hanford Site (WHC 1995j).

It should be noted that the design information for all of the alternatives is at an early planning stage. The details of implementing the selected alternative(s) are likely to change as the planning and design process matures. Therefore, these alternatives are intended to represent an overall plan for remediation rather than a definitive design. Any aspect of the alternative could change as the design process optimizes details of the plan; however, the overall plan for the alternative would not change. This alternative would involve the actions described in the following text.

Retrieval

Slurry pumping would be used to extract DST waste. Hydraulic sluicing would be used to remove SST waste. If hydraulic sluicing does not meet waste retrieval requirements, robotic arm-based retrieval methods would be used. Robotic-arm removal of solid waste saltcake within the tanks would require using a crusher to produce fine particulate material that could be slurried and pumped from the waste tank to the receiving or blending tank(s). Once the waste is removed and converted to a slurry form it would be pumped via pipe line(s) from the tank farms to a pretreatment facility.

In addition, the robotic arm would be used to remove solid waste such as piping and instrument trees from the tank. This type of solid waste would require remote mechanical handling for separate treatment prior to disposal as low-level waste.

Pretreatment
Pretreatment would consist of performing sludge washing, enhanced sludge washing, solid/liquid separation, and ion exchange to separate the waste into HLW and LAW streams. The solids in the waste would be washed to dissolve salts to the extent practical, and the salt solutions would be added to the supernatant for Cs removal. The sludge remaining in the tanks would be transferred to the HLW vitrification facility. The Ex Situ Extensive Separations alternative includes using multiple pretreatment modules designed to minimize the volume of HLW.

Immobilization
The LAW would be pumped into a LAW vitrification facility where it would be concentrated and mixed with glass formers (e.g., borosilicate and silica) and vitrified. Vitrification is a high-temperature process in which the waste is blended with additives and fused into a glass-like form suitable for disposal. The vitrification facility would have pollution abatement controls to ensure that effluents and emissions are within regulatory standards.

The washed sludges mixed with the separated Cs would be routed from a temporary storage facility to a HLW vitrification facility where they would be mixed with glass formers and fused into glass. The HLW glass would be sent to an onsite interim storage facility where it would be stored before shipment to a permanent potential geologic repository. The HLW vitrification facility would include pollution abatement controls to ensure that all effluents and emissions are within regulatory standards.

Disposal
The disposal of radioactive waste is regulated by DOE and the U.S. Nuclear Regulatory Commission (NRC). DOE's guidance for classifying waste is contained in DOE Order 5820.2A, Radioactive Waste Management (DOE 1988). The Order classifies waste into HLW, low-level waste, and TRU waste. Specific guidance includes near-surface disposal of low-level waste and deep geologic disposal of HLW and TRUs. The NRC regulates and licenses the disposal of radioactive materials from non-DOE facilities and the disposal of HLW for DOE facilities through regulations contained in 10 CFR 60. The Nuclear Waste Policy Act provides the statutory framework for NRC regulation of HLW disposal. The NRC guidance on waste classification is contained in 10 CFR Part 61. Currently, DOE disposal of low-level waste is not regulated by the NRC, although NRC rulings regarding waste treatment and waste feed limitations would affect classifying waste that is subject to HLW disposal requirements.

The vitrified LAW glass would be put into large disposal containers and placed into a near-surface retrievable disposal facility on the Hanford Site. Retrievable disposal means that the design of the disposal facility would be for permanent disposal but the waste could be retrieved from the disposal facility within a certain amount of time (assumed to be 50 years) if a different method of disposal was determined to be necessary. A Hanford Barrier would be constructed over the retrievable LAW disposal site to inhibit migration of contaminants and intrusion by humans, plant roots, or burrowing animals. Markers would be used to identify the location of the storage disposal facility. Security and administrative controls would be implemented and maintained indefinitely to protect workers and the public. For the purpose of calculating the potential impacts, it is assumed that the controls will be terminated after 100 years.

The vitrified HLW glass, following canister packaging into HMPCs, would be placed in an aboveground interim storage facility at the Hanford Site. The glass would then be shipped to the potential geologic repository for permanent disposal.

B.3.5.2 Facilities to be Constructed

The alternative includes constructing a tank waste retrieval and transfer facility, a sludge washing (separations) facility, the vitrification and process support facilities, onsite LAW disposal facilities, and temporary HLW storage facilities.

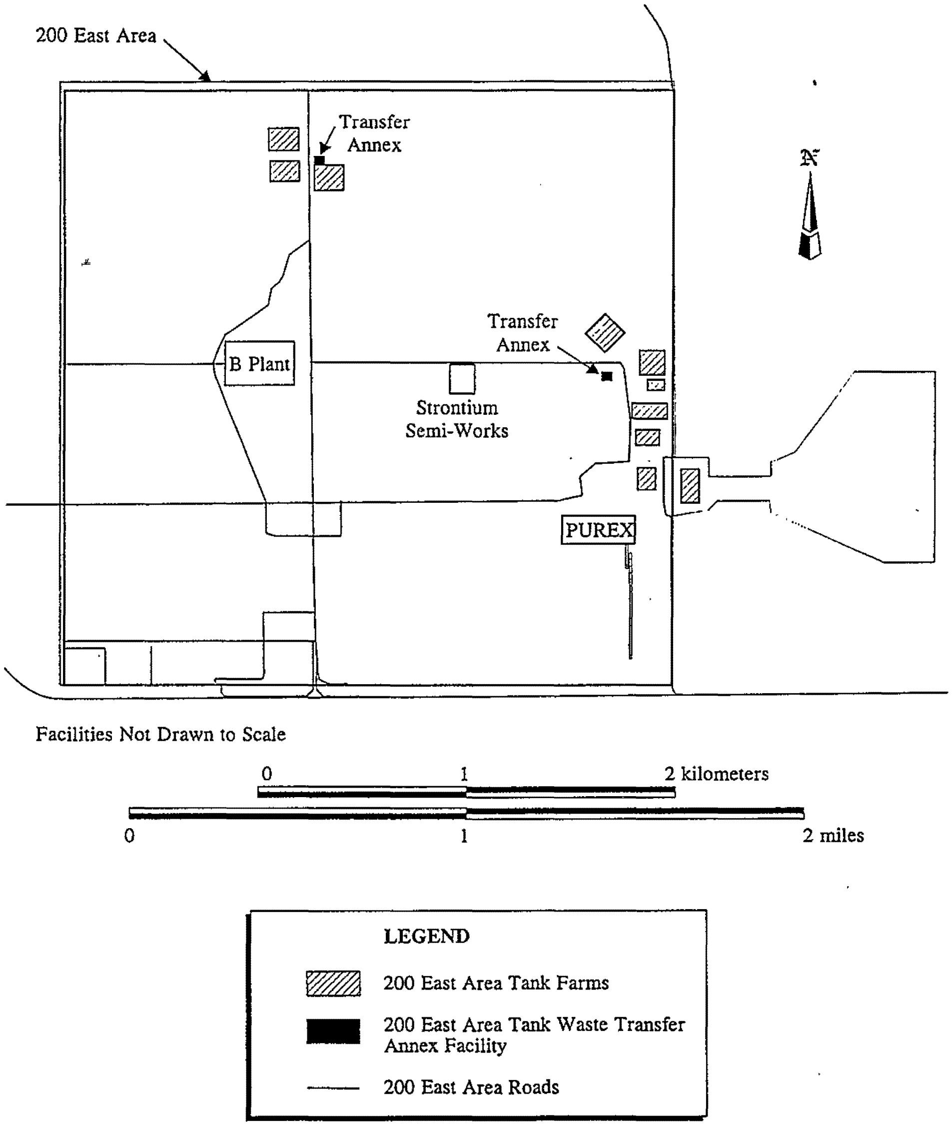
Tank Waste Retrieval and Transfer Facilities

The retrieval and transfer facilities would include bridging structures over the tanks to support the equipment, the off-gas treatment systems, four transfer annex buildings, one waste staging and sampling facility, and the transfer piping system (WHC 1995n). The bridge structures would span the tanks to transfer the equipment loads to foundations outside the perimeter of the tank. The structures, which would be movable or relocatable from tank to tank, would include a vertical, 24-m (80-ft) -high container to house equipment withdrawn from the tank while the entire assembly was relocated to another tank. Operating areas in the structures would be provided with HEPA ventilation equipment to maintain the pressure gradient required between the process, operating, and uncontrolled areas.

After being retrieved from the tanks, the waste would be transferred to the sludge washing tanks. The waste transfer system would include two transfer annexes in the 200 East Area and two transfer annexes and a waste staging and sampling facility in the 200 West Area. The transfer annexes and waste staging and sampling facility are shown in Figures B.3.5.1 and B.3.5.2. The inter farm and cross-site transfer piping would also be part of the system.

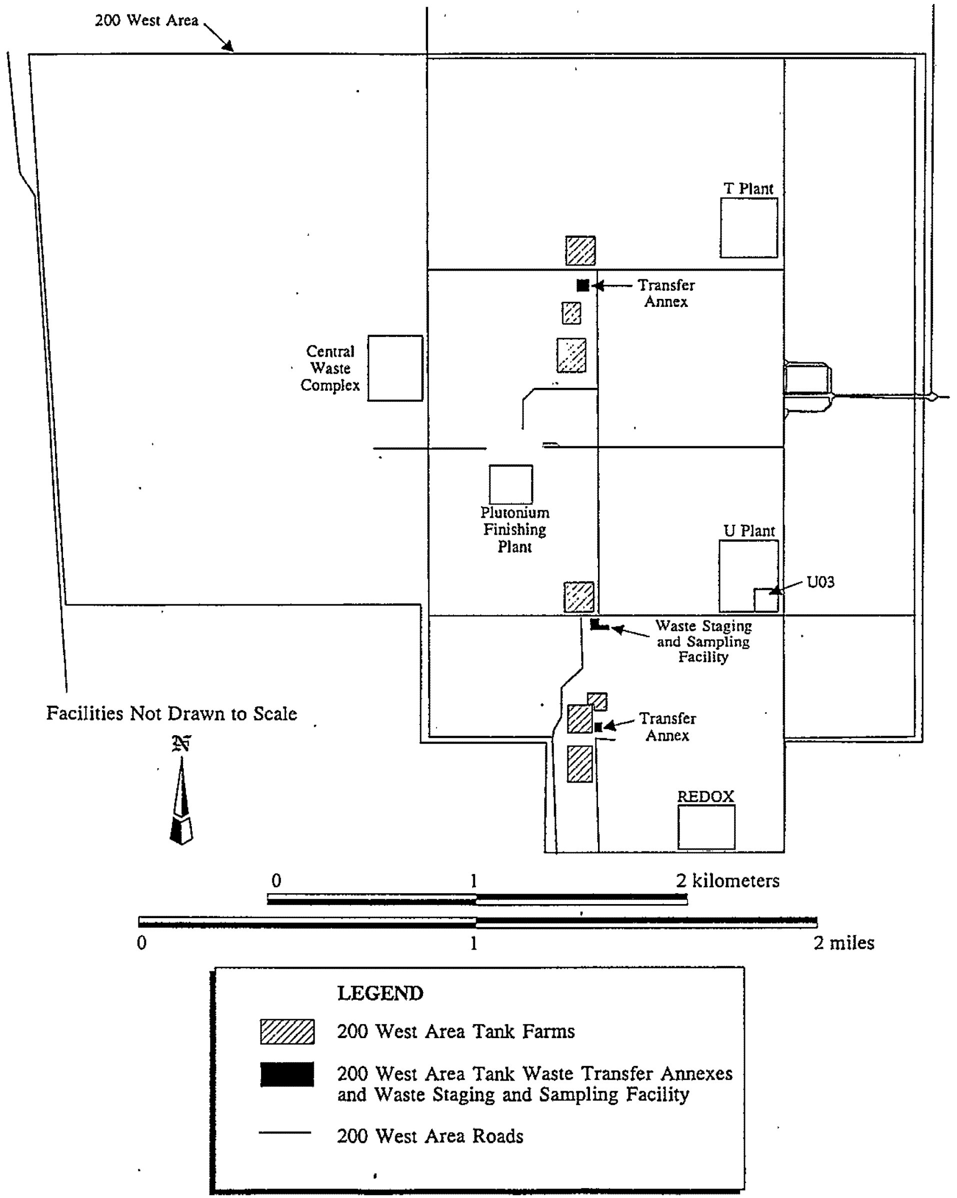
The transfer annexes would include multi-story facilities that contained tanks to store, blend, and dilute the slurry, equipment to crush oversize solids (saltcake and hardened sludge), and pumps to transfer the slurry to the processing facility or, in the 200 West Area, to the waste staging and sampling facility.

Figure B.3.5.1 200 East Area Tank Waste Transfer Facilities



Facilities Not Drawn to Scale

Figure B.3.5.2 200 West Area Tank Waste Transfer Facilities



The buildings would be built of concrete, approximately 25 m (80 ft) on each side, 11 m (35 ft) high, and extend 5 m (16 ft) below grade to allow the earth to serve as shielding. The facility would include the process equipment, a maintenance bridge crane, a decontamination area, an HVAC system with HEPA filters, a control room, and other features necessary for facility operations. The waste staging and sampling facility would pump the waste from the 200 West Area to the replacement cross-site transfer system for transfer to the processing facility in the 200 East Area. This facility would also be built of concrete, but would be larger than the transfer annexes. It would be approximately 73 m (240 ft) long, 23 m (75 ft) wide, 12 m (40 ft) high, and extend approximately 12 m (40 ft) belowgrade. Process equipment would include six agitated slurry tanks and two transfer pumps.

The processing facility would include a maintenance bridge crane with a repair bay, a decontamination bay, an HVAC system with HEPA filters, and an attached structure for emission/effluent monitoring. Except for the initial installation, tank farm piping would be rearranged during operation to accommodate the needs of the operation. Transfer piping between the tank farms and the transfer annexes and between the waste staging and sampling facility would be constructed as part of this alternative.

The replacement cross-site transfer line between the 200 West and 200 East Area lines would consist of two 8-cm (3-in.) -diameter stainless-steel pipes, each encased in a 15-cm (6-in.) -carbon-steel outer pipe to provide secondary containment as required by Federal and State regulations and DOE design criteria (see discussion in Section B.1.1.8). The lines would be sloped (at least 0.25 percent to preclude accumulation of solids) and buried, bermed, or appropriately shielded for radiation and freeze protection. The pipeline would be designed to prevent corrosion from the metal pipes contacting the soil. Both pipelines would be insulated with polyurethane foam and covered with a fiberglass jacket. A diversion box would connect the new transfer line to existing pipelines to facilitate liquid waste transfer between the 200 West and 200 East Areas. A booster pump located in the diversion box would provide the power to transfer waste slurries at the minimum required velocity to prevent the lines from clogging. A vent station would be located at the high point of the transfer system. The function of the vent station would be to introduce air into the lines after a transfer to allow draining the primary containment pipes. Both the diversion box and the vent station would be equipped with stainless-steel liners and would have provisions for washing down radioactive contamination, collecting accumulated liquid, and routing the liquid back to the tank farms. All process piping would have sufficient earth cover to reduce personnel exposure to as low as reasonably achievable, and would not exceed 0.05 millirem per hour (mrem/hr) at grade. The diversion box and cover would attenuate radiation levels to 0.05 mrem/hr at the surface.

Separations Facility

Separations would consist of two major process steps, sludge washing and Cs ion exchange. Other radionuclides would be removed, if required, to conform to the limits for LAW. The Cs ion exchange would be performed in the low-activity vitrification facility. The general arrangements for separations and low-activity vitrification are shown in Figures B.3.5.3 and B.3.5.4; however, the final design decision about washing the tank sludges has not yet been made. It is possible that an alternate method such as washing on crossflow filters may be used. Because in-tank washing represents a bounding condition for sludge washing, it will be described in detail in this appendix. Sludge washing would be done in DSTs that would be modified to accommodate the process. A mixer, decant pumps, and sludge transfer pumps would be added to the tanks through existing risers in the tank dome. New surface tanks would be installed for process chemicals, and surface piping would be rearranged to accomplish the objectives of the washing operation. Surface facilities would include three 20-m^3 (700-ft^3) process tanks, a tank ventilation system with HEPA filters to isolate the tank atmosphere, pump service and decontamination facilities, and an operations building. The ventilation system would allow the tanks to breathe as the waste level varied during transfer and mixing operations. The tank ventilation system, which would use HEPA filters, would be centrally located to serve the sludge washing system.

The pump service and decontamination facilities would be arranged around a central chamber mated to tank nozzles and would contain equipment removed from the tank. The equipment would be flushed with fresh water as it was removed, and the central chamber would finally be filled with lead shot before the entire assembly was transferred to a LAW disposal facility. Each internal tank pump would have a dedicated confinement chamber.

The operations building would be a 590-m^2 (6,400-ft^2) single-story block structure that would house a motor control center and a control room for the washing operation. Change rooms, operations offices, and a lunch room would also be included.

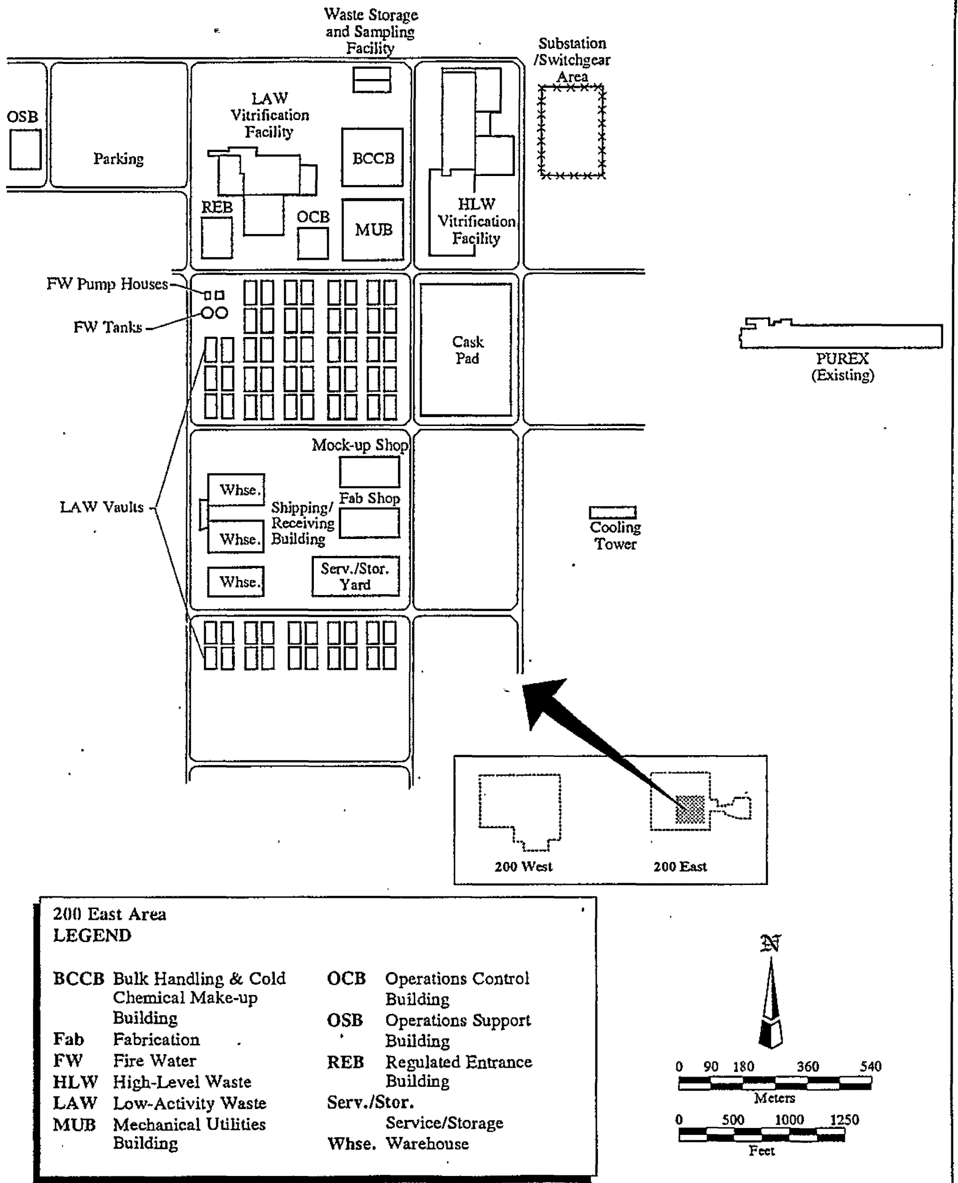
Low-Activity Waste Vitrification Facility

The LAW waste vitrification facility would be sized to produce 200 mt (220 tons) of vitrified waste per day in two production trains. It would contain seven operational areas, including feed receipt and sampling, Cs ion exchange, melter operations, cullet processing, cullet matrix operations, cold chemical makeup, and off-gas treatment areas.

The facility would have an overall footprint of 90 m (290 ft) wide by 75 m (250 ft) long with an overall height of 40 m (130 ft), of which 20 m (65 ft) would be belowgrade. In addition to the process level, which would be belowgrade, the facility would have two other levels, one at grade and the other at +9 m (30 ft). Overall, the facility would have a total area of approximately 6,800 m^2 (73,000 ft^2).

The process level would include feed receipt and sampling, Cs ion exchange, process evaporation, and cullet processing areas. Feed receipt and sampling would occur in six 200-m^3 (7,100-ft^3) tanks that would receive feed from six 400-m^3 (14,300-ft^3) tanks external to the building. The Cs ion exchange area would include a single stage of 12.5-m^3 (440-ft^3) columns and supporting tanks. The cullet processing area would consist of quench tanks below the evaporator and 18 cullet storage tanks.

Figure B.3.5.3 Ex Situ Intermediate Separations Layout

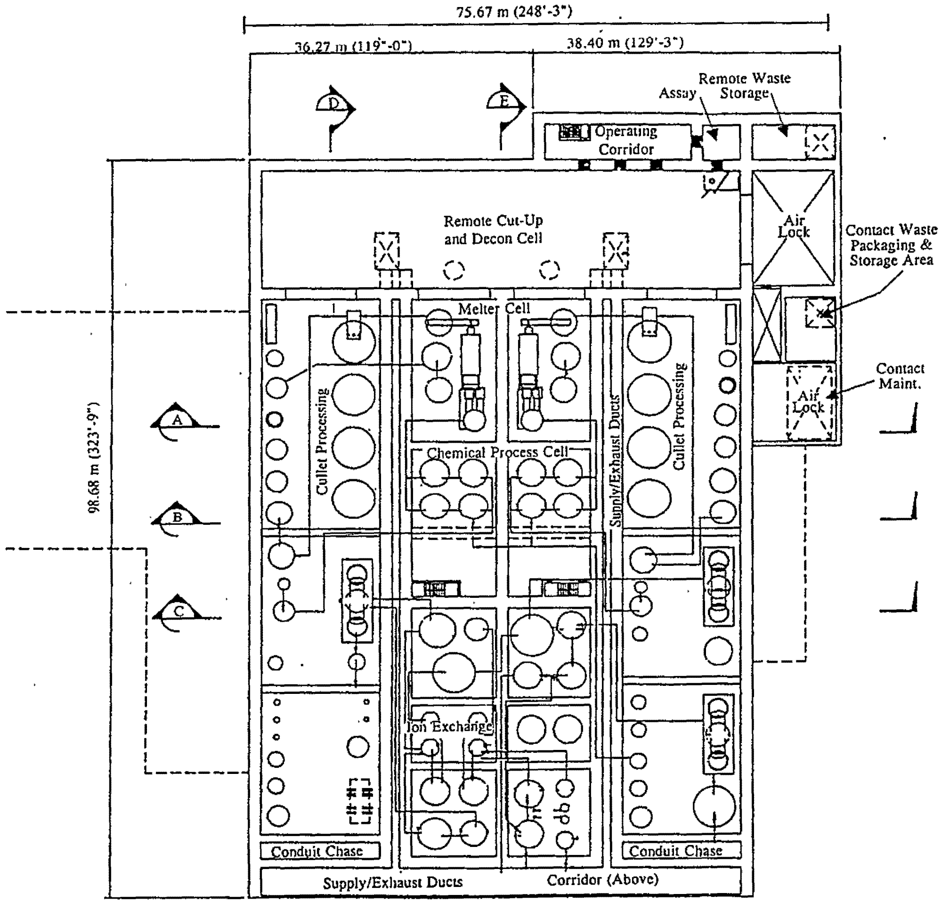


200 East Area
LEGEND

BCCB	Bulk Handling & Cold Chemical Make-up Building	OCB	Operations Control Building
Fab	Fabrication	OSB	Operations Support Building
FW	Fire Water	REB	Regulated Entrance Building
HLW	High-Level Waste	Serv./Stor.	Service/Storage
LAW	Low-Activity Waste	Whse.	Warehouse
MUB	Mechanical Utilities Building		

SOURCE: Adapted from WHC 1995j

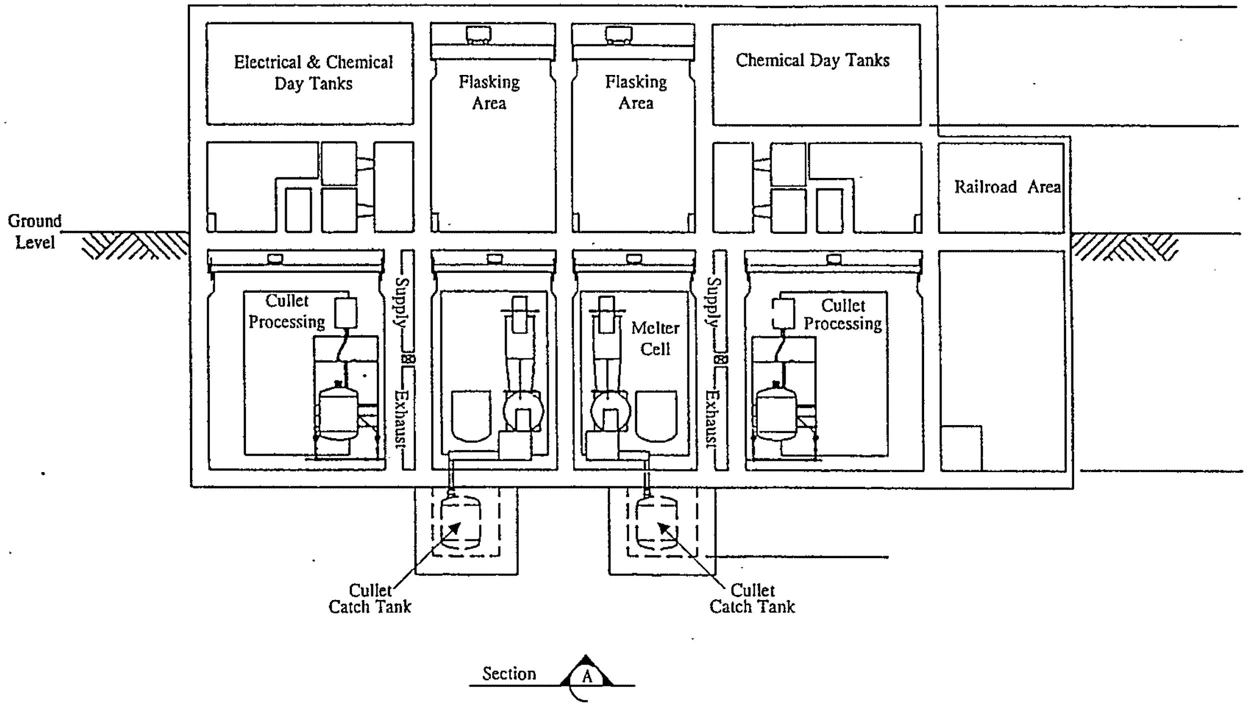
Figure B.3.5.4 Pretreatment/Low-Activity Waste Facility Layout (Sheet 1 of 7)



Plan View
6.1 m (20 ft) Below Grade

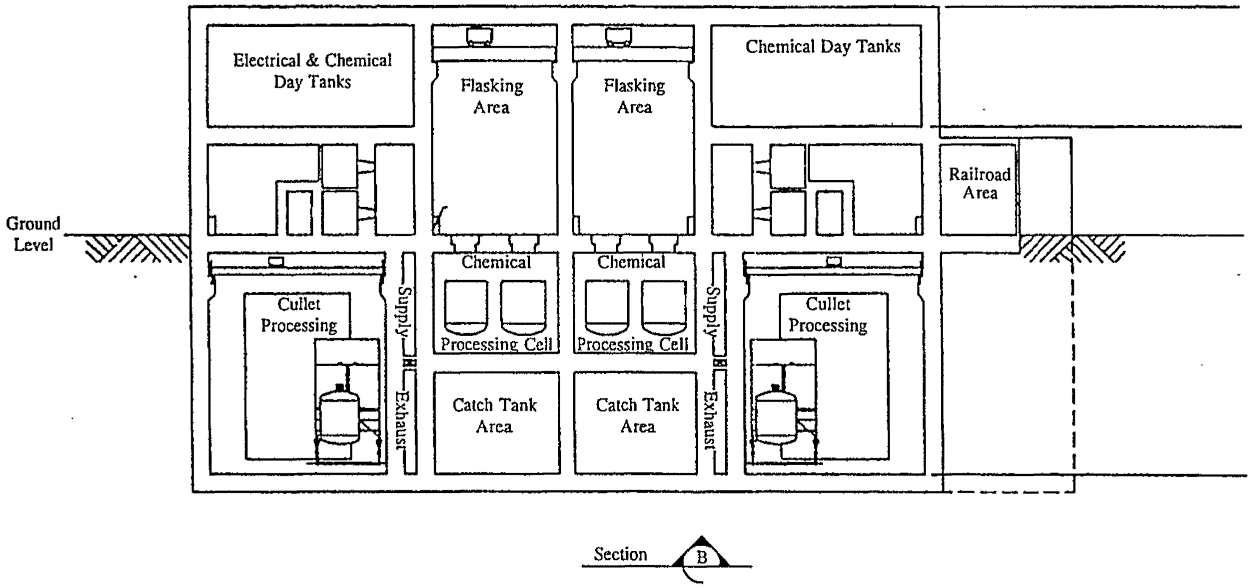
SOURCE: WHC 1995j

Figure B.3.5.4 Pretreatment/Low-Activity Waste Facility Layout (Sheet 2 of 7)



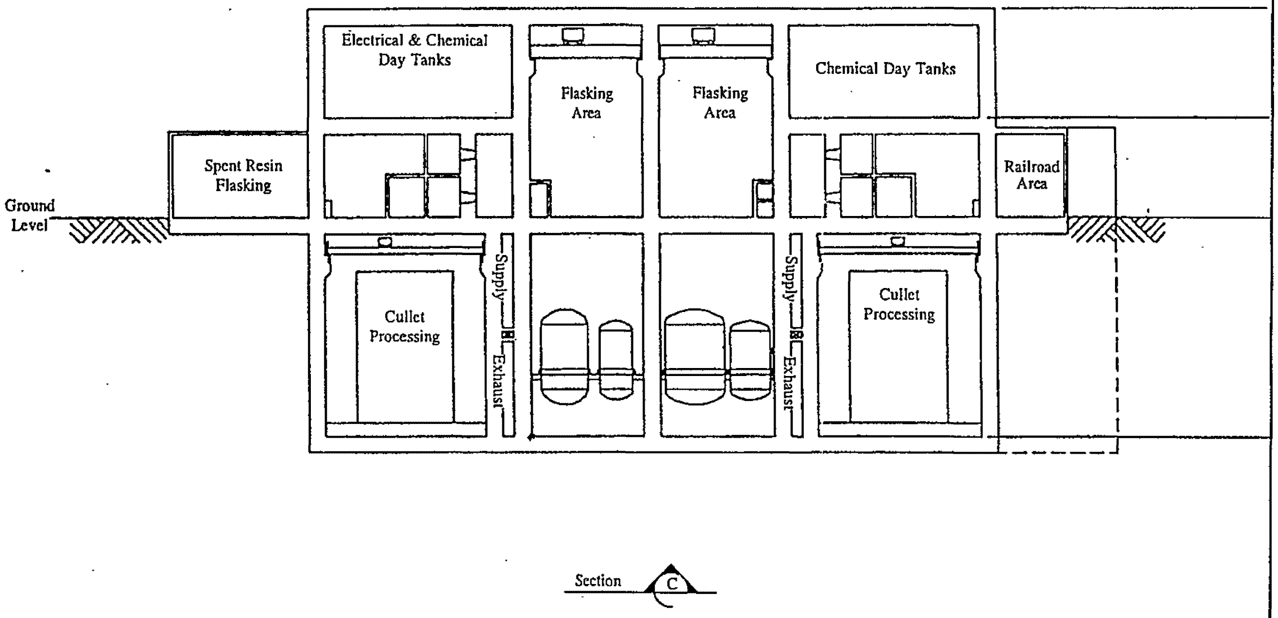
SOURCE: WHC 1995j

Figure B.3.5.4 Pretreatment/Low-Activity Waste Facility Layout (Sheet 3 of 7)



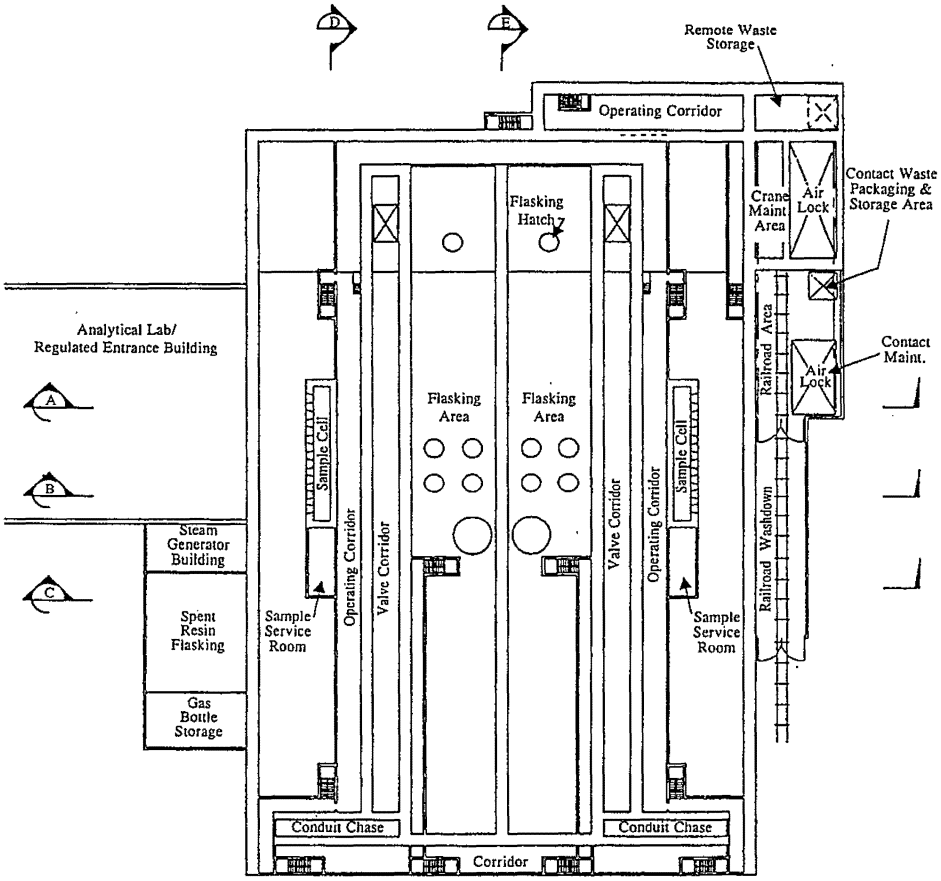
SOURCE: WHC 1995j

Figure B.3.5.4 Pretreatment/Low-Activity Waste Facility Layout (Sheet 4 of 7)



SOURCE: WHC 1995j

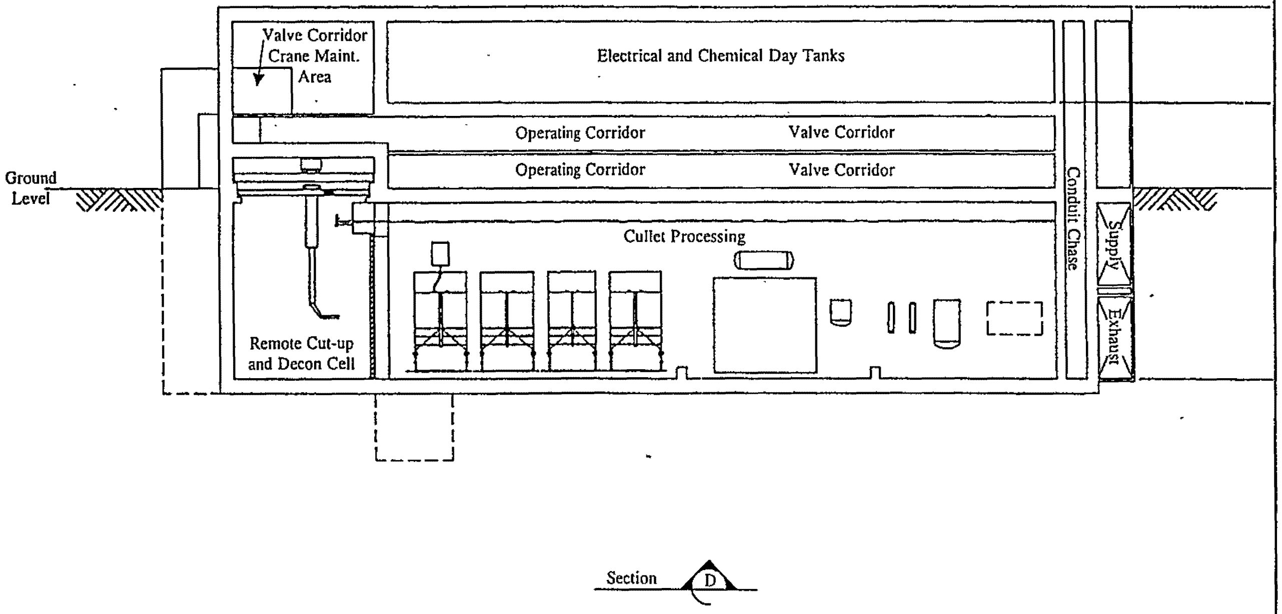
Figure B.3.5.4 Pretreatment/Low-Activity Waste Facility Layout (Sheet 5 of 7)



Plan View
6.1 m (20 ft) Above Ground

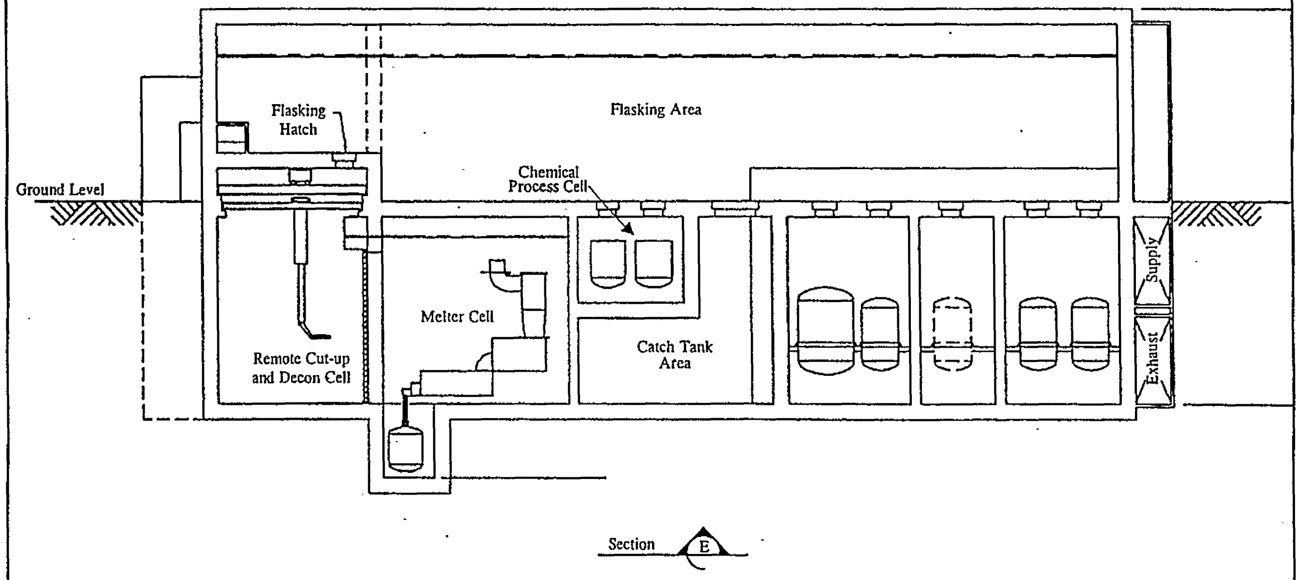
SOURCE: WHC 1995j

Figure B.3.5.4 Pretreatment/Low-Activity Waste Facility Layout (Sheet 6 of 7)



SOURCE: WHC 1995j

Figure B.3.5.4 Pretreatment/Low-Activity Waste Facility Layout (Sheet 7 of 7)



SOURCE: WHC 1995j

The facility's other two levels would provide space for support services and additional process equipment. The grade level of the facility would provide space to support canister filling operations, instrumentation for the process equipment, melter operations, process evaporator for LAW melter feed, maintenance areas, and sulfur operations. The +9-m (30-ft) level would provide electrical services and cold chemical makeup systems.

Low-Activity Waste Cullet Disposal

Under the Ex Situ Intermediate Separations alternative, LAW cullet would be disposed onsite. The cullet would be mixed with a matrix material in the vitrification facility and placed into disposal containers (approximately 2.6 m^3 [92 ft^3]), which would then be transported to onsite disposal vaults. A total of 66 vaults would be constructed. Each vault would be an estimated 37 m (120 ft) long by 15 m (50 ft) deep. The vaults would be engineered concrete structures.

The requirements for using a matrix material and specific matrix material requirements have not been established. The use of a matrix material for the LAW waste form has been included as being representative of waste form matrices for bounding the transportation and resource impacts.

When all of the LAW glass has been placed in final storage, a Hanford Barrier would be constructed over the storage site. Hanford Barrier performance objectives are discussed in Section B.6.0.

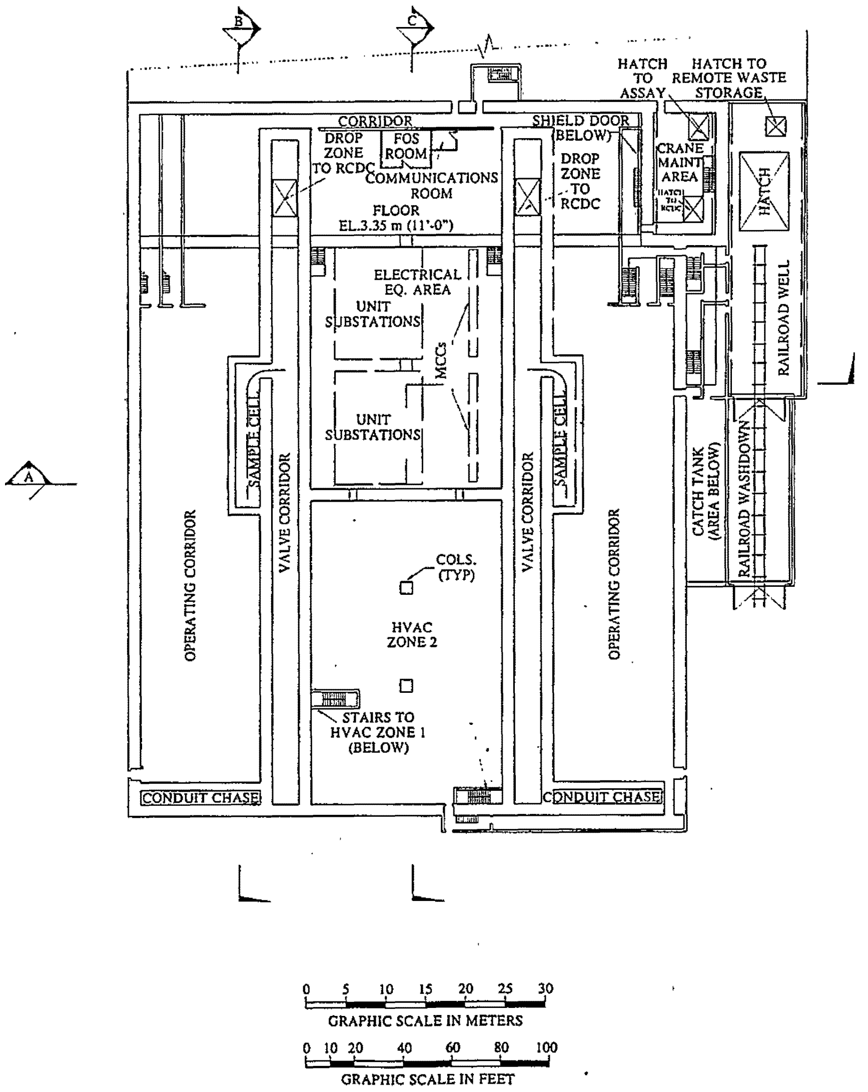
High-Level Waste Vitrification Facility

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 (Figure B.3.5.5). The facility would have an overall height of 45 m (150 ft), of which 13 m (45 ft) would extend belowgrade. In addition to the process level, the facility would have three other levels at -13 m (-45 ft), +13 m (+45 ft), and +20 m (+65 ft). 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.

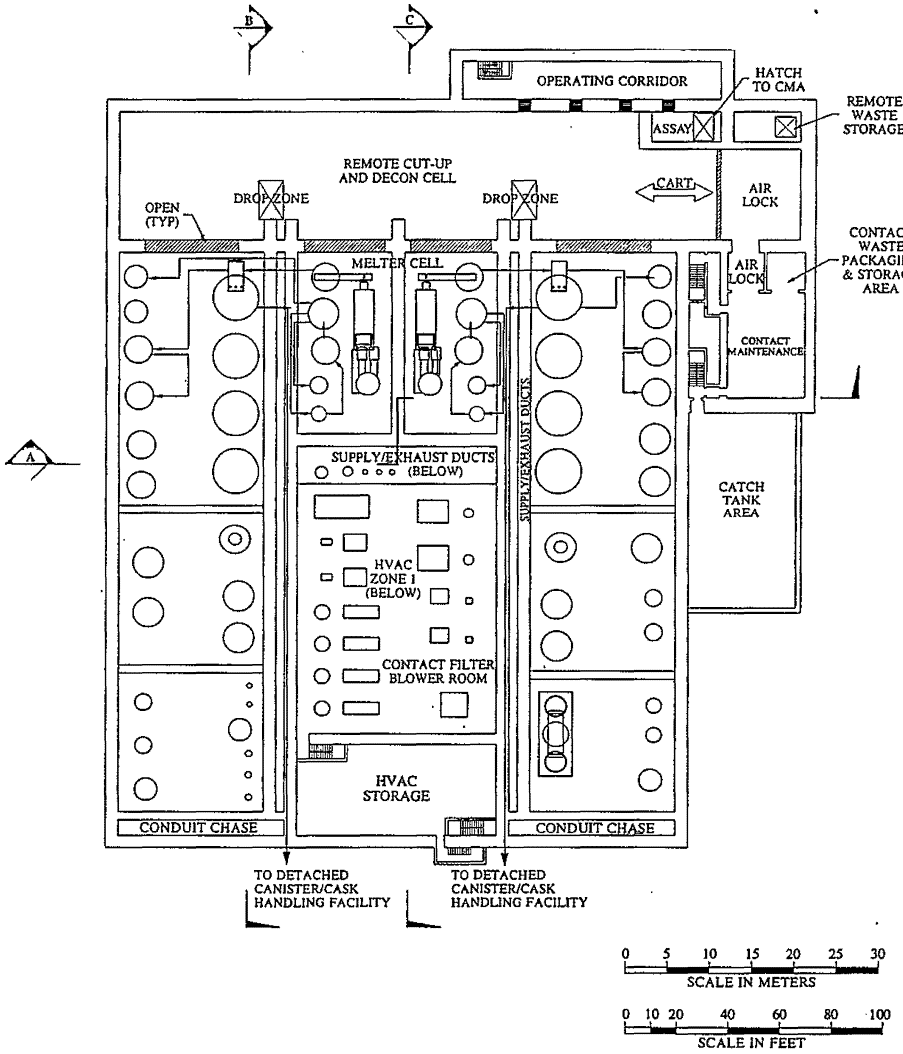
The final HLW glass form would be a glass canister measuring 0.61 by 4.57 m (2.0 ft by 15.0 ft). The HLW interim onsite storage facility would allow enough interim storage space for all of the HLW glass produced. After the HLW campaign concluded, the canisters would be transported in the HMPCs (four canisters per HMPC) to the potential geologic repository for final disposal.

Figure B.3.5.5 High-Level Waste Vitrification Facility Plant Plan View 0.00 Meters (at Grade) (Sheet 1 of 5)



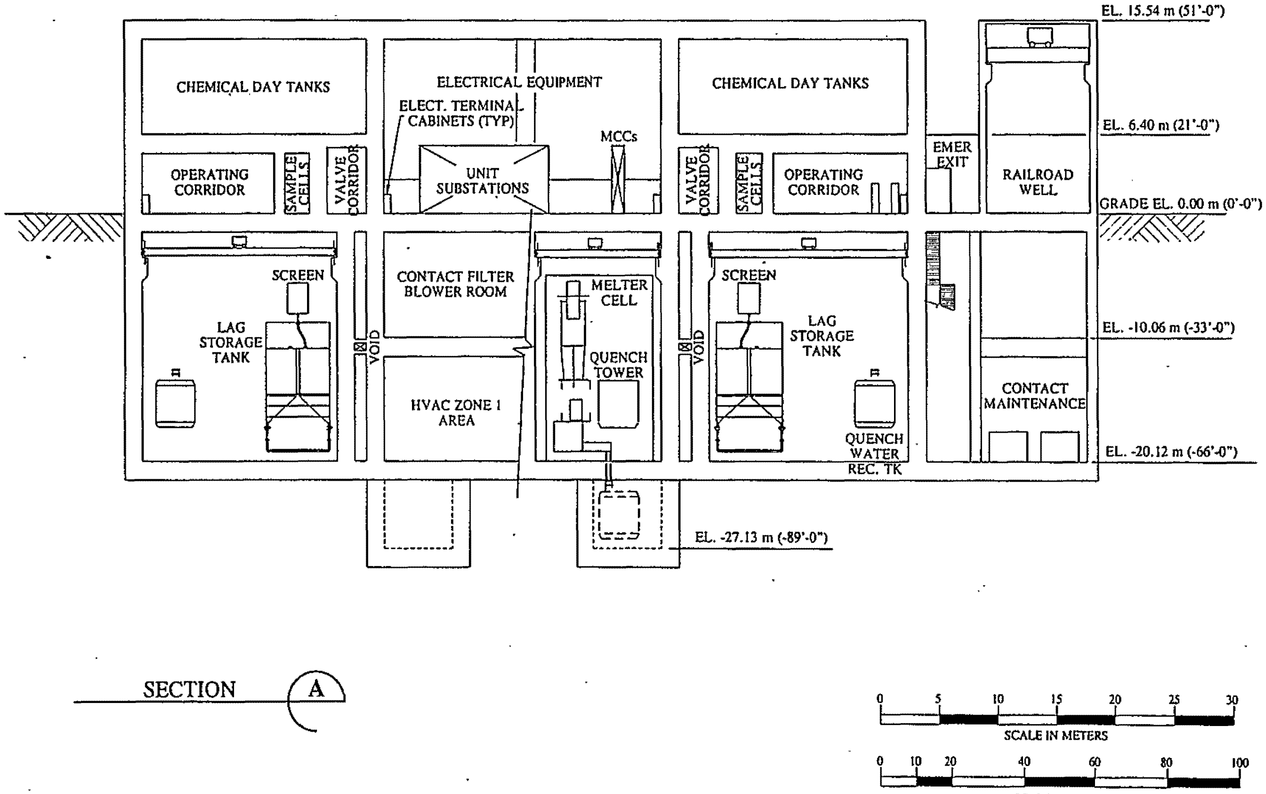
SOURCE: WHC 1995j

Figure B.3.5.5 High-Level Waste Vitrification Facility Plant - 20 Meters (Sheet 2 of 5)



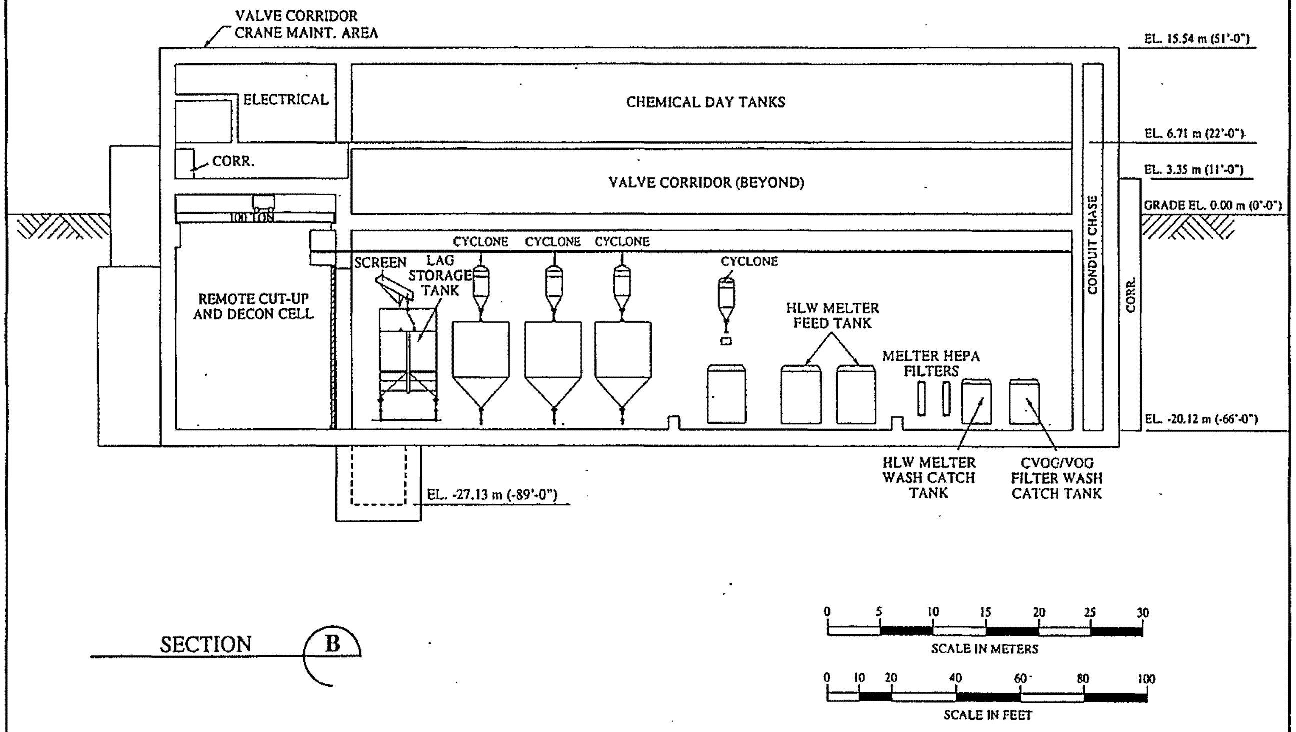
SOURCE: WHC 1995j

Figure B.3.5.5 High-Level Waste Vitrification Facility Plant - Section A (Sheet 3 of 5)



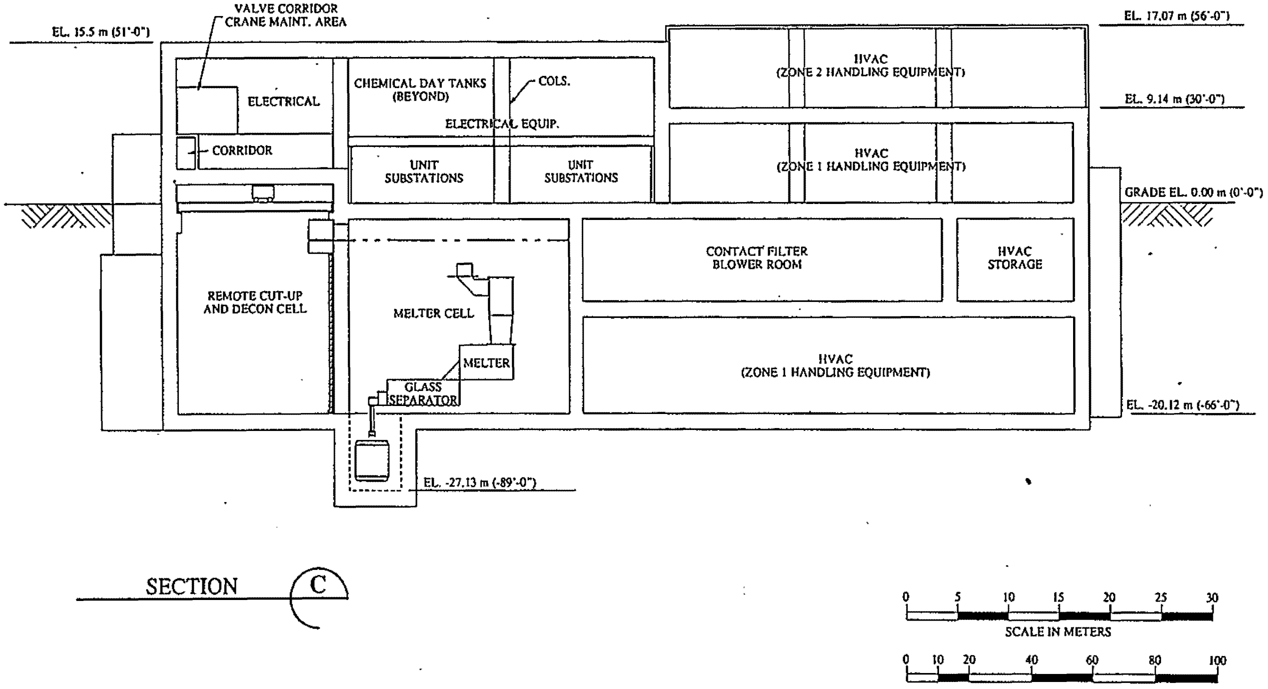
SOURCE: WHC 1995j

Figure B.3.5.5 High-Level Waste Vitrification Facility Plant - Section B (Sheet 4 of 5)



SOURCE: WHC 1995j

Figure B.3.5.5 High-Level Waste Vitrification Facility Plant - Section C (Sheet 5 of 5)



SOURCE: WHC 1995j

Support Facilities

Each of the process facilities would provide its own process support equipment. Common utilities and cold chemical areas would provide headers for service to support the process systems in the plants. These common services would include:

- Medium pressure steam and condensate;
- Instrument and plant compressed air;
- Cooling water;
- Sanitary water;
- Process water;
- Demineralized water;
- Raw water and fire water;
- Sanitary sewer;
- Nonradioactive liquid waste processing;
- Cold chemical bulk storage and makeup;
- Oxygen; and
- Electrical power.

Support facilities that would provide for nonprocess and personnel activities would include the following.

- The Operations Support Building would serve as the administration building for the complex. It would have 19,000 m^2 (21,000 ft^2) of floor space with approximately 40 percent dedicated to offices and the remaining 60 percent dedicated to office support functions (e.g., conference rooms, lunch rooms, utility rooms, equipment areas, storage rooms, and supply rooms).
- The Regulated Entrance Building would be the single point of entry into the facility for maintenance and operation personnel. The building would provide 6,500 m^2 (70,000 ft^2) of space for security operations, health physics, change rooms, lunch rooms, and a first aid clinic.
- The 2,100-m^2 (22,500-ft^2) Operations Control Building would house the central control room for the entire TWRS Treatment Complex as well as space for control support functions.
- The Bulk Cold Chemical Building would be a one-story building approximately 90 by 90 m (300 by 300 ft) providing 8,360 m^2 (90,000 ft^2) of floor area. The building would store anhydrous ammonia, kerosene, nitric acid, LAW form matrix materials, sodium hydroxide, and sulfur. Chemical makeup would also be located in this building.
- The Switch Gear/Generator Building would be a 90- by 90-m (300- by 300-ft), single-story structure. It would house switch gear and be unoccupied.
- The Mechanical Utilities Building would be a single-story, 90- by 90-m (300- by 300-ft) building. It would house plant air compressors, an instrument air system, chillers, a demineralized water system, and a process steam and condensate system.

- Four small pumphouses external to the Mechanical Utilities Building would pressurize fire-water and cooling-water systems.

Other support facilities would include a cooling tower (60 by 90 m [200 by 300 ft]), a fabrication shop (45 by 90 m [150 by 300 ft]), mock-up shops (45 by 90 m [150 by 300 ft]), three warehouses (45 by 90 m [150 by 300 ft]), and a switchyard. The switchyard would include a 120 by 150 m (400 by 500 ft) substation consisting of incoming 230-kilovolt (kV) dead-end towers feeding a double-ended bus with a single tie breaker. The bus would feed redundant transformers rated 230 to 13.8 kV, with a capacity of approximately 100,000 kV-ampheres. The 13.8-kV transformer secondaries would feed a double-ended switchgear, located in a switchgear building that would include utility monitoring and control equipment.

B.3.5.3 Description of the Process

Overview

The overall tank waste treatment process would include 1) retrieving the waste; 2) separating the LAW from the HLW; 3) vitrifying each waste stream separately; 4) disposing of the LAW onsite; 5) temporarily storing the HLW; and 6) transporting the HLW to the potential geologic repository at a future date. Separating the HLW from the LAW would be accomplished with a liquid/solid separation process (many of the HLW constituents are insoluble) and a subsequent ion exchange step to recover Cs (which is partially soluble and has allowable concentration limits in the LAW) from the liquid phase. Other radionuclides would be removed, if required, to conform to the limits for LAW. The HLW and LAW would be vitrified in separate but similar processes. The vitrification process would include feed-preparation systems, the vitrification process itself, off-gas treatment systems, wastewater processing systems, glass-handling systems, and a number of utility and support systems. Figures B.3.5.6 and B.3.5.7 illustrate the process.

Sludge washing would be performed with approximately four modified DSTs. Sludge washing may also be done on filters or in centrifuges. The supernatant aqueous phase would be pumped to the LAW vitrification facility where the first operation would be Cs recovery. The sludge from sludge washing would be transferred to the freestanding HLW vitrification facility as would be the Cs recovered in the LAW facility. The vitrified LAW cullet would be placed in containers and transported to vaults for onsite disposal. The HLW would be temporarily stored in casks on a pad near the HLW facility before being shipped to the potential geologic repository for permanent disposal.

Tank Waste Retrieval and Transfer

The Tri-Party Agreement (Ecology et al. 1994) includes a milestone that directly impacts the TWRS program. Milestone M-45-00 requires tank waste residues not exceeding 10.2 m^3 (360 ft^3) in each 100 series tank, and tank residues not exceeding 0.85 m^3 (30 ft^3) in each 200 series tank. Thus, this milestone provides the basis for the 99 percent removal requirement.

Figure B.3.5.6 Ex Situ Intermediate Separations Alternative - Separations and LAW Vitrification Process

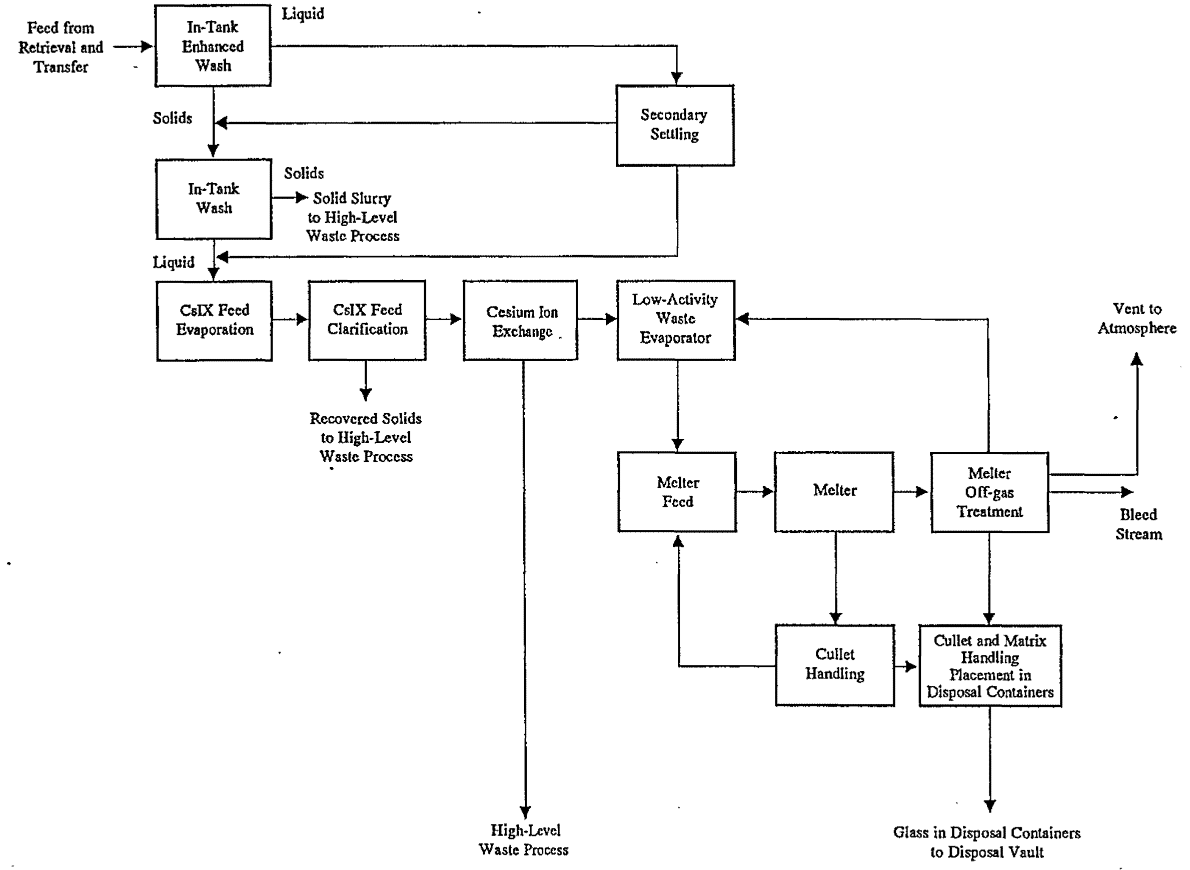
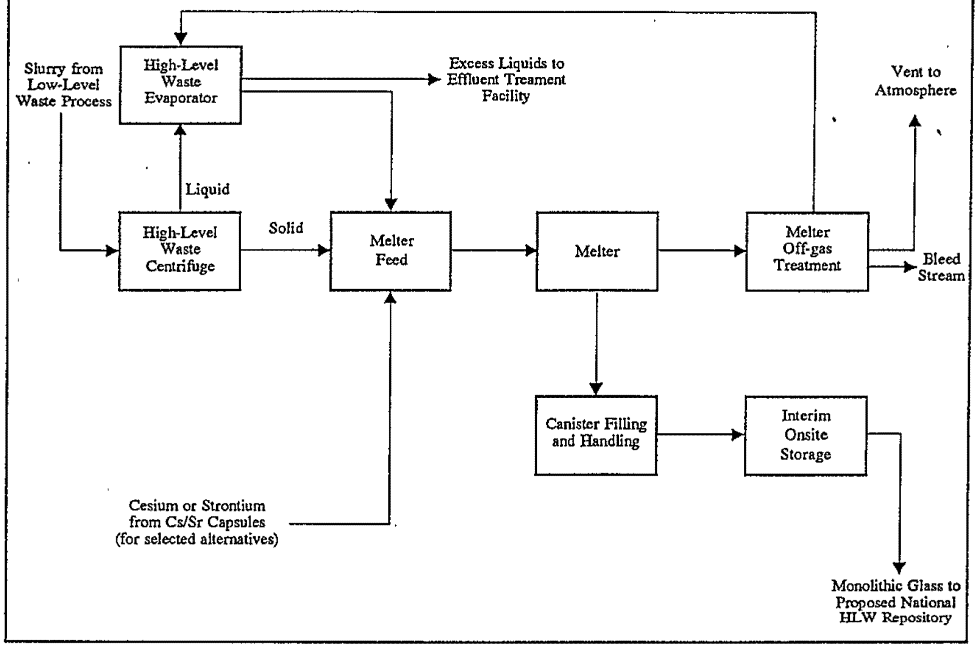


Figure B.3.5.7 Ex Situ Intermediate Separations Alternative - HLW Vitrification Process

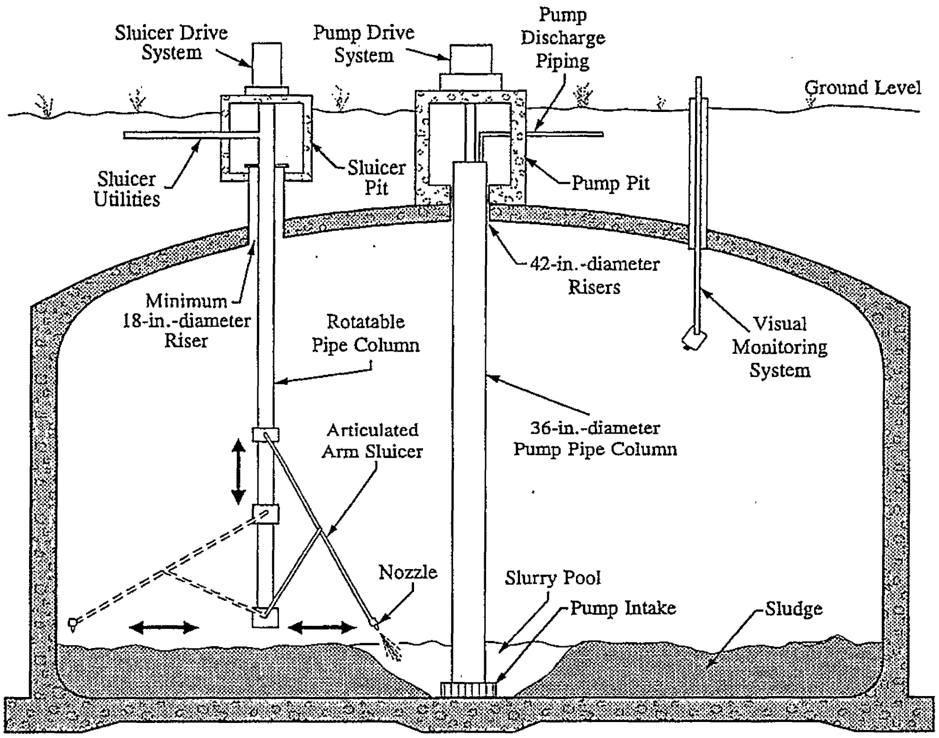


Most of the SST waste would be removed by reslurrying the waste with a hydraulic jet. This process, referred to as sluicing, would remove the slurry with a pump to remove all but a 1 percent heel of waste from the tanks. The sluicers would dislodge and erode the sludges and dislodge, dissolve, and/or breakup the saltcake creating a slurry, which would be pumped to a DST where it would be allowed to settle. The supernate would be recycled to the sluicing jets to continue the recovery process. Reusing the saturated supernate would minimize saltcake dissolution and reduce the liquid volume in the process. Controlling the liquid volumes would be important because virtually all of the water added to recover and transfer the waste would need to be removed by evaporation before vitrification.

Currently, there are several technologies available for use in sluicing systems, one of which is presented in Figure B.3.5.8. Considerable experience on tank sluicing on which a design can be based exists, as the SSTs were previously sluiced to recover U sludges from 1952 to 1957 and again to recover Sr sludges from 1962 to 1978.

In some instances, the sluicing operation may not be able to remove sufficient waste to meet the removal requirement. A recovery system based on a robotic arm would be used as a backup for the SSTs. A robotic arm would provide additional flexibility to position sluicing jets and pumps and

Figure B.3.5.8 Sluicing Arrangement for Single-Shell Tank Waste Retrieval



Not to Scale

SOURCE: Adapted from WHC 1995n

extended capability to recover additional waste by using tools and equipment. Arm-based systems would also provide for dismantling and recovering internal tank hardware that would otherwise interfere with sludge retrieval. Figure B.3.5.9 is a conceptual view of the robotic arm. It is estimated that 24 sluicing systems and 12 robotic arm systems would be required. This estimate is based on the proposed retrieval and transfer schedule, the life and reliability of the equipment, and the amount of sludges that will be difficult to retrieve.

Because the solids in the DSTs may not be compacted into the dense material that occurs in the SSTs, the principal technology used for retrieving the DST waste would be mixer pumps. The pumps would be installed in existing DST risers. The pump's rotating hydraulic jets would breakup and mobilize the sludge, and vertical turbine pumps would transfer the slurry. Unlike the SST equipment that would be moved from tank to tank, each DST would be permanently equipped with two to four mixer pumps. A sluicing system would be provided as a backup to the mixer pumps. It is assumed that six of the 28 DSTs (20 percent) would use sluicing to retrieve waste that could not be retrieved with mixer pumps (WHC 1995n).

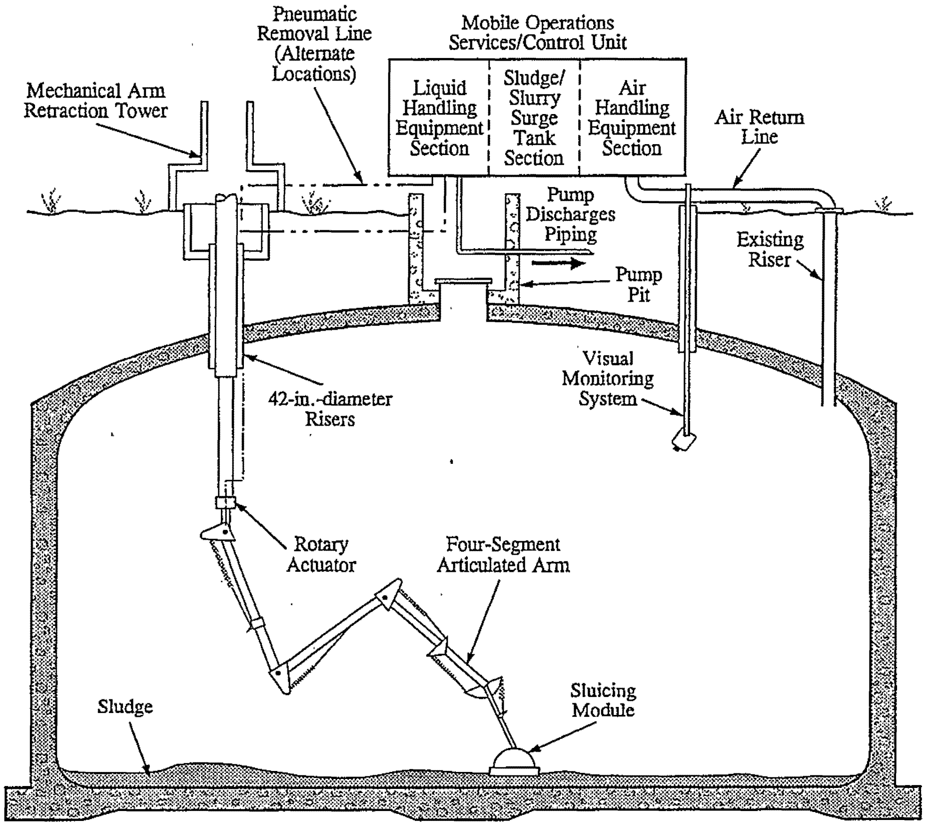
After retrieval, the waste from the SSTs would be either transferred directly to process facilities or the DSTs for interim storage. The waste transfer annexes would be the primary means for transferring waste, but container transport could be selectively used for small waste volumes and tank heels. Four waste transfer annexes would be constructed, two in the 200 West Area and two in the 200 East Area. In addition, a waste staging and sampling facility would be provided in the 200 West Area.

The waste transfer annexes would be located close to clusters of SST farms to receive waste slurry from the SSTs, condition the slurry, and pump it within the 200 East Area to DST storage or the processing facility. In the 200 West Area, the waste transfer annexes would pump to the staging and sampling facility that in turn would pump the waste to the 200 East Area processing facilities. Slurry conditioning would include dissolving, diluting, and reducing the size of entrained solids.

Waste would be recovered from approximately 60 MUSTs by sluicing and then transported by the LR-56(H) truck or a containerized transfer system to the transfer annexes for discharge to the process. Approximately 120 trips or more with the 3,800-L (1,000-gal) LR-56(H) truck would be required to nearly empty the tank waste volumes tabulated (see Appendix A, Table A.2.3.1) for 28 of the MUSTs.

For purposes of this EIS, it is assumed that an average, or nominal, feed would be the input to the processing plant. The concept of nominal feed is an averaging of the feed during the duration of waste treatment operations. For the ex situ alternatives, the retrieval function would be designed to deliver a nominal feed to the processing plant. The actual feed would vary depending on tank inventories and retrieval sequences. The Facility Configuration Study (Boomer et al. 1994) identified the following five design feed streams that would be addressed in the engineering design of the proposed treatment facility:

Figure B.3.5.9 Robotic Arm-Based Arrangement for Single-Shell Tank Waste Retrieval



Not to Scale

SOURCE: Adapted from WHC 1995a

- Nominal feed, average feed over plant life;
- Shielding basis feed, highest radionuclide concentration feed used for shielding design;
- Safety/regulatory assessment feed, bounding radionuclide feed used for accident analysis;
- Criticality assessment feed, feed with bounding fissile material content used to define criticality controls; and
- Variability assessment feed and range of feed compositions that might be expected during plant operation.

Sludge Washing

One of the primary purposes of the sludge washing step would be to dissolve constituents that limit the waste loading of the HLW such as aluminum, chromium, and phosphorous. Sodium hydroxide solutions would be used during enhanced sludge washing to solubilize aluminum, chromium, and phosphorous, which have limited solubility in water alone. Approximately 85 percent of the aluminum, 75 percent of the chromium, and 70 percent of the phosphorous would be recovered from the HLW and sent to the LAW vitrification facility. The supernatant solutions from the sludge washing process would be forwarded to the separations facility for Cs recovery.

Feed to the sludge washing process would be a slurry of insoluble sludges suspended in an aqueous solution of soluble waste. The solids would contain most of the HLW and, except for Cs and some complexed waste, the solution would contain limited HLW. The HLW (solids slurry at approximately 50 percent by weight would be separated from the liquids in a counter-current decantation operation that would use existing DSTs. Sludge washing could also be done outside the tanks on filters or in centrifuges. The waste slurry would be allowed to settle (separation by mechanical means may be required) to 50 percent total solids by weight, the supernatant would be transferred to the separations facility, and supernatant from a previous wash would be added to the solids remaining in the tank. After two washes with successively cleaner water the aqueous phase of the slurry would contain fewer soluble salts and the slurry could be transferred to the HLW vitrification facility.

Low-Activity Waste Processing

Cesium Recovery

For purposes of the EIS, it has been assumed that the only soluble radionuclide that would be removed would be Cs. It may become necessary to provide further liquid processing to remove additional radionuclides from the LAW to the extent required to meet onsite disposal requirements. This additional liquid processing could include organic destruction and Sr and Tc removal. The impacts to be expected as a result of additional liquid processing would be a small decrease in the amount of LAW and a small increase in the amount of HLW.

The Cs is soluble in alkaline solutions and in sufficient concentrations is a HLW. A minimum of 85 percent of the Cs would be removed from the feed to the LAW melter by an ion exchange process. Four ion exchange columns preceded by a submicron prefilter would be arranged so that three of the columns would load in series while the fourth column was being regenerated. Nitric acid would be used to elute the Cs and sodium hydroxide, and wash water solutions would be used to regenerate the resin in the fourth column. The fourth column would be returned to service once the first column was loaded. The columns would be sized for continuous operation. The Cs solution would be characterized, concentrated by evaporation, and transferred to the HLW vitrification facility.

The LAW remaining following the separations processes would contain approximately 17 million curies (MCi) of radioactivity including 10 MCi Cs and Ba, 6.8 MCi of Sr and Y, 2.59E-02 MCi of Tc-99, and a total of 1.22E-02 MCi of TRU isotopes.

Feed Conditioning System

The primary functions of the feed conditioning system would be to 1) mix and concentrate the LAW feed; 2) provide for chemical adjustment and sampling; and 3) supply a controlled and monitored feed to the melter. The feed conditioning system would be made up of the following:

- Six 380,000-L (100,000-gal) sample/holding tanks located in an underground vault adjacent to the vitrification building;
- One 36,000-L (9,500-gal) evaporator feed tank, which would also be used to collect the various aqueous plant recycle streams;
- One steam-heated evaporator; and
- Four 36,000-L (9,500-gal) melter feed adjustment tanks.

The feed would be held in the six sample/holding tanks for sampling and analysis before being forwarded to the single evaporator feed tank and evaporator, which would be in continuous operation. The evaporator concentrate would be divided into four streams and continuously forwarded to two pairs (four) of melter feed adjustment tanks. At this point, the evaporator concentrate would be sampled and analyzed before being transferred to two pairs of melter feed tanks. Each pair of melter feed tanks would supply a melter in a staggered cycle so that the melters would receive a continuous feed.

The LAW evaporator feed tank would provide a place for blending various recycle waste streams from the LAW vitrification building, such as melter off-gas quench liquid, cullet fines slurry, and filter wash from the six off-gas HEPA filters. The tank would have an agitator to ensure complete mixing. From this tank the blended stream would be pumped to the evaporator.

The steam-heated LAW evaporator would continuously receive the blended stream containing about 2 weight percent suspended solids and about 18 weight percent dissolved solids. Evaporated water would rise to the overhead condenser through mist-eliminators to minimize the carry-over of contamination. The evaporator overhead would generate condensate that would be sent to the process condensate recycle tanks. The evaporator bottoms would contain about 5 weight percent suspended solids and about 47 weight percent dissolved solids. This bottoms stream would be split in half to serve the two melter trains, and would be continuously pumped to one of the available LAW melter feed adjustment tanks in each of the two trains. The LAW melter feed adjustment tanks and the downstream melter feed tanks would be located in the chemical process cell. The tanks, cooling coils, and piping in the chemical process cell would have a no-maintenance design. The tanks, pumps, and agitators would be located below ports in the cell roof so that they could be removed into shielded flasks for transport to maintenance.

Bulk Flux System

Bulk fluxes include silica, alumina, borate, and calcium oxide. The bulk flux system would include receiving and storage silos with a pneumatic loading system; a conveyor discharge system with batching capabilities; a batch mixer; and pneumatic transfer to a day bin, which would feed the melter. The fluxes would be selected, proportioned, and blended to complement the analysis of the waste feed tank so that the desired vitreous product is produced.

Oxygen Plant

The final design selection of the LAW melter has not yet been made. For purposes of analysis this alternative would include a vortex melter that would be fuel-fired, although ultimately other melter designs could be chosen. The melter would be fired with oxygen to reduce the volume of flue gas and minimize the formation of NO_x. The oxygen would be supplied by a pressure-swing adsorption unit with stored liquid oxygen available as backup.

Vitrification System

The vitrification system would combine the waste and flux in the desired proportions, heat them to the temperature required for vitrification, and evaporate the aqueous phase of the solution in the waste slurry. In the vortex melter this process would happen very rapidly. There would be two parallel vitrification systems, each with a 100-mt/day capacity for glass product. Each vitrification system would include a flux-feed system, waste injection system, burner system, vortex melter, glass separator, and a glass quench system. The flux-feed system would include a pneumatically supplied day bin and a weighfeeder with air locks.

The waste slurry would be fed to the combustion chamber of each melter at a 0.061-m^3/min (16-gal/min) rate. There the semi-volatiles and volatiles would be burned off and the remaining solids and waste oxides would be combined with glass forming oxides. The combustion fuel, kerosene, would be pumped to the combustion melter along with 100 percent gaseous oxygen. The incoming waste slurry from the LAW melter feed tank would be mixed uniformly with the glass forming oxides in a mixing and injection valve mounted on top of the combustion chamber. The glass oxides would be gravity fed from head bins and carefully metered by weigh feeders into the mixing and injection valve. The hot combustion gases and by-products would flow axially through the cyclone, creating a rotating gas flow. The heavier premelted glass solids would be deposited along the refractory wall by the action of the rotating gases (centrifugal force) and form a thin film as they flowed axially through the cyclone. The hot combustion gases and by-products would continue heating the glass while in the cyclone to finish dissolving the waste oxides into the glass matrix. The hot combustion gases would be approximately 50 °C (120 °F) hotter than the glass melt film, which would be 1,300 °C (2,370 °F).

The glass separators would function as reservoirs to refine the glass and remove entrained gases. Each glass separator would be close-coupled to a quench flume where the molten glass would be fractured into cullet. The final design decisions concerning the LAW glass form have not yet been made. While this alternative is based on the concept of glass cullet, ultimately, other forms such as canisters or monoliths could be chosen. Also close-coupled to each glass separator would be a quench tower that would cool the melter off-gas and collect the condensables.

Cullet Slurry Handling

The cullet slurry handling system would include a quench flume, wet roll crusher, cullet catch tank, slurry catch tank, washing trommel screen, and a transfer pump to recycle fines and quench water back to the evaporator feed tank. There would be two production trains in the cullet handling system for the LAW plant. Molten glass from the glass separator would be discharged to the quench flume where it would make contact with water and fracture into cullet. The cullet would pass through a wet roll crusher to break up any oversized pieces and drop into the cullet catch tank. Steam from the quenching operation would be condensed and recycled to the quench tank. The cullet slurry would be pumped from the cullet catch tank to a trommel screen where it would be dewatered and washed to remove adhering fines. The fines would be returned to the cullet catch tank with any excess water and recycled to the feed conditioning system.

Dry-Cullet Handling

The product handling system, which would fill the casks, would include a combination dryer/storage bin and a pneumatic transfer system. In the dry-cullet handling system, the cullet would be dried and transported via a pneumatic conveyor to the cullet storage bin, where it would be sampled and held until analyses are complete. Accepted cullet would be pneumatically transferred to the day bin, while rejected material would be recycled to the melter feed tank as off-specification material. The cullet in the day bin would be fed forward to the waste form matrix mixer.

Cullet Transfer to Vaults

This alternative is based on mixing the cullet with matrix material and placing the mixture into disposal containers. The disposal containers would provide a means for handling LAW and retrieving them at a future time if required. The LAW disposal containers would be transported using a specialized transporter and placed into the disposal vaults.

Off-Gas Systems Description

Overview

The main off-gas systems would be the melter off-gas system, vessel off-gas system, condenser vessel off-gas system, bin vent off-gas system, and the pneumatic vessel off-gas system. Each of the tank waste alternatives would make extensive use of recycle streams in the process to recycle back into the treatment process volatile radionuclide and chemical constituents captured in the off-gas systems. These recycle streams would be used to minimize the generation of secondary waste. It has been determined that a bleed stream would be required for each alternative to avoid a continuous buildup of certain volatile radionuclide and chemical constituents (e.g., Tc-99 and mercury [Hg]) in these recycle streams. For comparison purposes, it has been assumed for each alternative that the bleed stream percentage would be 1 percent of the recycle stream and that this secondary waste stream would be stabilized by some low-temperature process.

The melter off-gas system would receive the hot combustion gases from the glass separator. The gases leaving the melter would contain products of combustion, steam, volatilized radionuclides from the feed, and entrained particulates from the rapid water evaporation in the feed slurry. This gas would also contain nitrogen and sulfur dioxide (SO_2) from the decomposition of process feed constituents. The off-gas would be first quenched with scrub water, which would condense the water vapor and remove particulates, and water-soluble contaminants. Excess condensate from the melter off-gas system would be recycled to the HLW evaporator feed tank. The scrubbed melter off-gas would undergo further cooling and successive stages of HEPA filtration to remove radionuclide particulates, after which SO_2 would be adsorbed from the gas and subsequently converted into elemental sulfur by a Claus unit. Finally, the partially treated gas would pass through a catalytic de-NO_x reactor, where NO_x would be converted into nitrogen and water vapor before passing through another HEPA filter and discharging to the atmosphere.

Melter Off-Gas System

The primary functions of the melter off-gas system would be to 1) cool and quench the melter off-gas; 2) remove radionuclides and certain chemical constituents; 3) catalytically destroy SO_2; and 4) recover elemental sulfur to permit the release of these emissions to the atmosphere consistent with regulatory requirements. An additional function would be to provide a differential pressure confinement boundary for the melter.

The gas cooling and quenching portion of the melter off-gas system would consist of two identical parallel trains, each dedicated to a single melter. Each train would consist of a quench tower, a venturi scrubber and separator, and a mist eliminator. Each train would include a dedicated cooler, chiller, scrub solution tank, scrub solution recirculating pump, and scrub solution transfer pump.

The radionuclide removal portion of the melter off-gas system would include two operating trains and one standby train of sub-micron particulate filtration and blowers. The emissions abatement portion of the melter off-gas system would consist of a single operating train of catalytic NO_x destruction, SO_2 removal, and sulfur recovery equipment.

Melter off-gas flow from each of the two melters would be quenched from 1,360 to 75 °C (2,480 to 170 °F) by direct, counter-current contact with 32 °C (90 °F) water in a refractory-lined quench tower. Entrained particulates would also be scrubbed from the off-gas in the quench tower. The scrub water and condensed moisture from the bottom of the tower would drain by gravity back to the scrub solution tank for re-use. The quenched off-gas would be contacted with scrub water in a venturi scrubber to further remove entrained particulates. The separator would receive the venturi scrubber discharge and separate the off-gas from the scrub water, which would gravity drain to the scrub solution tank.

A chiller would cool the off-gas leaving the separator to 30 °C (86 °F) before it would enter the mist eliminator. The mist eliminator would use glass fiber candle elements to remove mist and particulates from the off-gas stream. A continuous water spray would help clean condensate and particulates from the candle elements. The rinse from the mist eliminator would gravity drain to the scrub solution tank. The liquid mixture from the scrub solution tank would be cooled to 32 °C (90 °F) in the cooler and would be recycled back to the quench tower and venturi scrubber. A purge of excess process condensate plus associated solids would be continuously discharged from each scrub solution tank and collected in the scrub filter tank. The solution would then be recycled from the tank back to the evaporator feed tank for treatment. A small bleed stream would be taken from this recycle stream to prevent a buildup of certain volatile radionuclides and chemical constituents. This secondary waste stream would be stabilized by an appropriate low temperature process (such as grout).

The off-gas from each mist eliminator would flow to one of three identical parallel trains of filters. Two of the three trains would be in operation with the third train on standby. Each train would consist of a heater, two back-washable metal HEPA filters in series, and a blower. These metal HEPA filters would be high-efficiency metal fiber filters that would be back-washable for removal of radioactive particulates. The heater and washable metal HEPA filters would be remotely maintainable and located inside a hot cell. The blowers would be located in a contact-maintenance room. The heater would raise the off-gas temperature to prevent any condensation of moisture, which would increase filter pressure drop, reduce filter efficiency, and cause acid gas corrosion in the equipment and piping. The back-washable metal HEPA filter would remove submicron radioactive particulates from the off-gas stream. The blowers would draw the off-gas through the system and provide a pressure confinement boundary for all of the equipment, including the melter relative to the remote cells.

The filtered off-gases discharged from the blowers would be combined and then processed to remove SO_2 and catalytically destroy NO_x. The combined melter off-gas stream would first be blended with pure oxygen and the recycled tailgas from the downstream Claus unit before entering the tube side of the melter off-gas heat exchanger. Oxygen addition would help SO_2 absorption and catalytic NO_x destruction. In the exchange, the melter off-gas would be heated to 400 °C (750 °F) by exchange with the hot effluent gas from the NO_x catalytic reactor. The melter off-gas would then be sent to one of three copper oxide (CuO) bed absorbers containing CuO-impregnated alumina sorbent. Approximately 90 percent of the SO_2 would be absorbed and converted to copper sulfate in the presence of oxygen in the SO_2 absorber.

With one CuO bed serving as an SO_2 absorber, the remaining two CuO beds would be in the sulfate reduction mode and the SO_2 absorber regeneration mode, respectively. For sulfate reduction, a reducing gas stream containing hydrogen would reduce the copper sulfide and liberate gaseous hydrogen sulfide. The hydrogen would be produced by catalytically cracking ammonia to nitrogen gas and hydrogen. The hydrogen sulfide rich effluent would be sent to the Claus unit, which would recover the sulfur in its elemental form. The tailgas from the Claus unit would be recycled to join the melter off-gas downstream of the blowers. The SO_2 absorber regeneration would prepare the CuO bed for SO_2 absorption service by passing air across the absorber bed to oxidize the copper to CuO. Air leaving absorber regeneration would be sent to the vessel off-gas system for treatment.

From the SO_2 absorber, the melter off-gas would be preheated to 500 °C (932 °F) in an electric heater before entering the NO_x reactor. The NO_x reactor would contain a catalyst bed for the selective catalytic reduction of NO_x to produce nitrogen and water vapor in the presence of ammonia. The treated off-gas stream would be cooled to 66 °C (150 °F) or less as it passed first through the shellside of the melter off-gas heat exchanger, and then through the water-cooled melter off-gas discharge cooler prior to release to the process exhaust system.

Process Area Ventilation (Other Off-Gas Systems)

The primary function of the vessel off-gas, condenser vessel off-gas, bin vent off-gas, and pneumatic vessel off-gas systems would be to decontaminate vessel vent gases to meet regulatory requirements for stack release. An additional function of these systems would be to provide a pressure differential on process areas relative to the surrounding cells or vaults to prevent the out-migration of radioactive materials. Each of these systems would consist of a vent collection header, filter preheaters, metal HEPA filters, and blowers. The off-gases from the process vessels would be collected by the vent header and routed to one of two identical parallel trains of filtration. Each train would consist of a heater, two back-washable metal fiber HEPA filter and a blower. Both of these back-washable metal HEPA filters would be high-efficiency metal fiber filters that would be remotely maintainable and would be located inside a hot cell. The blower would be located in a contact maintenance room. The heater would raise the off-gas temperature to prevent the downstream condensation of moisture, which would increase filter pressure drop and reduce filter efficiency. The back-washable metal HEPA filters would remove submicron radioactive particulated from the gas stream. Following filtration, the vent gases would be pressurized by the two 100-percent capacity blowers before being discharged to the HVAC exhaust system.

In the process exhaust system, the melter off-gas would be combined with processed gas streams from other portions of the process and a stream of supply air. The combined flow of supply air and process gas streams would be exhausted through a high-efficiency metal fiber HEPA filter followed by a conventional paper HEPA filter and blower prior to being exhausted to the stack and discharged to the atmosphere. The metal HEPA filter would be remotely maintainable and located inside a hot cell. The conventional HEPA filter and blower would be located in a contact maintenance room.

Process Liquid Waste System

All of the process liquid waste from the LAW vitrification facility would be in the form of process condensate from contaminated process streams. The process condensate recycle tanks and the pH adjustment tank would be located inside an underground vault near the LAW vitrification building.

The process condensate recycle tanks would accumulate the continuously generating condensate, and sequester the contents while awaiting the analytical results of sampling. On-specification liquid would be transferred to the pH adjustment tank, but off-specification liquid would be returned at a controlled rate to the HLW evaporator feed tank for rework. To accommodate the occasional need to recycle off-specification liquid waste, the condensate recycle tanks would be sized so that two of the three tanks would be used to process the normal forward flow of on-specification liquid. The third tank would be used for short-term storage of off-specification waste.

Each process condensate recycle tank would have a 295,000-L (78,800-gal) capacity, and a working capacity of 274,000 L (73,000 gal). About 18 hours would be required to fill a single tank. With the two operating tanks alternately receiving the incoming feed, the time available for sampling, analysis, and pump-out of a tank would also be about 18 hours. Each tank would be agitated to ensure complete mixing. Each tank would have a sampling device and two motor-driven transfer pumps.

From a filled recycle tank, on-specification condensate would be transferred in batches to the pH adjustment tank every 18 hours. The adjustment tank would have the same capacity and type of associated equipment as the recycle tank. In the adjustment tank, a measured volume of sodium hydroxide would be added, based on previous sampling and analyses. The contents of the tank would then be sampled and analyzed prior to being transferred out of the facility. The normal destination for effluent that would meet acceptance criteria would be the Liquid Effluent Retention Facility from which the liquid waste would be transferred to the Effluent Treatment Facility for treatment and final disposal. On nonroutine occasions, off-specification liquid from the pH adjustment tank could be transferred to the tank farms.

Process Steam and Condensate System

The process steam and condensate system would provide 1,000 kilopascals (150 pounds per square inch-gauge) steam for the heating requirements of closed-loop process steam users. To minimize the amount of potentially radioactive material leaving the area, the process steam and condensate building would be located in the vitrification building. The process steam and condensate system would include the process steam generator, process steam condensate condenser and cooler, process condensate pumps, process condensate collection tank, particulate filter and ion exchange unit, and distribution piping for process steam and condensate. The HEPA filters would be provided on the process condensate collection tank vent discharge.

Process Cooling-Water System

The process cooling-water system would be capable of maintaining process tanks at 50 °C (122 °F) or less, during normal process operations and idle or shutdown periods. The process cooling water system would include heat exchangers, recirculation pumps, distribution piping, an expansion tank with HEPA filters on the tank vent, and a chemical addition tank. To minimize the amount of potentially radioactive material leaving the area, the process cooling water system would be located in the vitrification building.

Melter Cooling-Water System

The melter cooling-water system would remove heat from the melter during normal process operation. It would include heat exchangers, recirculation pumps, distribution piping, an expansion tank with HEPA filters on the tank vent, and a chemical addition tank. To minimize the potential for radioactive contamination outside of the facility, the process cooling water system would be located in the vitrification building.

Process Chilled-Water System

The process chilled-water system would remove heat from process streams, which would be cooled to below 27 °C (80 °F). This system would include a process water chiller, a process chilled-water expansion tank with HEPA filters on the tank vent, and a process chilled-water pump. To minimize the amount of potentially radioactive material leaving the area, the process chilled-water system would be located in the vitrification building.

Cold Chemical Vent System

The cold chemical vent system would provide vapor control on vents from cold chemical feed and decontamination tanks, drain catch tanks, and other potentially radioactive sources throughout the vitrification building. This system would include HEPA filters, blowers, and piping.

Breathing Air System

The breathing air system would provide breathing quality air for respirators. The source of this air would be breathing air bottles that would be located outside of the vitrification building. The breathing air stations, which would be the distribution system for breathing air, would be located inside the building. The building could also be served by portable breathing air carts.

Health Physics System Vacuum System

The health physics system vacuum system would provide a dedicated central vacuum system to support health physics monitoring and sampling systems. This system would provide constant flow rates for the monitors and samplers at various locations in the vitrification, regulated entrance, and operations control buildings, and the vitrification building annex. Each location would include HEPA filters, blowers, and piping. Buildings external to the process facilities would have their own dedicated health physics system. The health physics system vacuum system provided in buildings external to the process facilities would be located with the other shared facilities.

Potentially Radioactive Liquid Waste Processing System

The potentially radioactive liquid waste processing system would collect and store liquid waste from potentially contaminated areas. This waste would be analyzed for radioactivity. If the waste was determined to be radioactive, it would be transferred to radioactive waste processing for further treatment. If the waste was not radioactive, it would be transferred to nonradioactive waste processing. Facilities within the vitrification building would include drain catch tanks, pumps, transfer pumps, and HEPA filters. An externally located part of the potentially radioactive liquid waste collection system would convey potentially radioactive waste from the regulated entrance building and the repair shops to the main part of the system in the vitrification building.

Cold Chemicals System

The cold chemicals receipt, makeup, and distribution system would include all facilities required to receive, store, prepare, and feed cold chemicals to the process, neutralization, and decontamination facilities. The portion of the system that would be located within the vitrification building would include the cold chemical feed and decontamination tanks, their associated transfer pumps, and distribution piping.

High-Level Waste Processing

The HLW vitrification facility would be a freestanding, single train plant designed to produce 20 mt/day (22 tons/day) of of HLW glass. It would be essentially a small-scale version of the LAW vitrification facility performing similar processing and requiring similar support and utility systems.

Feed Conditioning System

The HLW vitrification facility would receive HLW slurry from the sludge washing operation and Cs 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. As in the LAW vitrification facility, 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.

Vitrification System

A cold cap melter would be included in the alternative for HLW vitrification as the most thoroughly researched melter in the size required for this production level. The melter would use joule heating, in which 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 that would receive 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 entered the melt.

At this stage the HLW vitrification process would deviate from the LAW vitrification process. Instead of producing cullet as in the LAW process, 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. The quench flume, trommel, pneumatic transfer equipment, and a number of bins proposed for LAW vitrification would not be required to support HLW vitrification.

Canister Fill Operations

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 dried, the crane would transport it to the smear test cell, where the canisters would be smear-tested for surface contamination. If the canisters failed the test they would be returned to the decontamination cell. If the canisters passed the test they would be forwarded to the load-out cell. Canisters would enter the load-out tunnel on a transfer cart. The tunnel would have a crane that would remove the canister from the cart and place it into an HMPC overpack container (four canisters per overpack).

Full HMPCs would be removed from the load-out well with the cask staging building crane. The cask lids would then be bolted on. The casks would be smear-tested, inspected, and then transferred to temporary storage pads pending shipment to the potential geologic repository for disposal.

Post Remediation

When processing of the tank waste has been completed, the processing facilities would be decontaminated and decommissioned in the following manner.

- Processing equipment would be decontaminated sufficiently to allow onsite disposal in a low-level waste burial ground.
- Processing facilities would be decontaminated to the extent possible and then entombed in place. The exact materials that would be used to cover processing facilities have not been decided.

B.3.5.4 Implementability

Issues related to implementing this alternative can be grouped into the following categories.

- Some of the technologies involved in this alternative are first-of-a-kind and thus do not have a performance history. In particular, the robotic-arm concept for retrieval and the fuel-fired melter for producing LAW glass have been used as applicable concepts. In neither case is there performance history, particularly with the radioactive waste.
- Processes for retrieving, separating, and immobilizing waste often have been based on engineering judgement and assumptions. Performance of key processes (e.g., sludge washing) has been assumed in the absence of extensive quantitative data. Quantitative performance requirements have not been established for many of the processes and functions. Further process testing to determine equipment sizes is necessary before plant engineering could proceed.
- Cost estimates for this alternative have a high degree of uncertainty because many processes are first-of-a-kind systems.

- Retrieval criteria specifying recovery of 99 percent of the waste volume in each tank may not be achievable. Recovery of less tank waste would have a direct bearing on classifying the waste remaining in the tank.
- While the robotic arm being considered for backup to the sluicing operation has been designed and built, it has not been tested and therefore may not perform as assumed.
- Facility requirements for shielding have not been generated and exposure during retrieval is based on engineering judgement.
- Recovery of DST waste by agitating with turbine pumps has not been demonstrated. If the turbine pumps do not perform as expected, then additional retrieval methods would be necessary.
- The vortec melter, which has been selected as a concept for this alternative, has been demonstrated on generic glass-making feedstock but not tested on the actual feeds that will be used in this process. The off-gases from a fuel fired melter may contain elevated levels of Cs, sodium, or radionuclides. The capture of large amounts of impurities in the scrubbers may result in a large quantity of liquid to be recycled or treated in a separate facility. The magnitude of the recycles stream has not been completely evaluated.
- The proposed LAW waste form is unique and has not been used before.
- The engineering data that served as a basis for this alternative were developed using cullet in a matrix material as a LAW form for onsite disposal.
- A performance assessment has not been completed defining the LAW waste form requirements for retrievable storage and disposal at the Hanford Site, and DOE and NRC have not yet completed negotiations on what constitutes "incidental waste" for disposal of LAW at Hanford. Additional separations steps may therefore be required to meet LAW disposal criteria. The laboratory data now available on enhanced sludge washing are limited. There may be a need to evaluate additional alternate pretreatment methods for certain classes of waste.

The following development or demonstration activities would be necessary if this alternative is selected for implementation:

- Design and test tank retrieval systems;
- Evaluate sludge washing;
- Evaluate the Cs ion exchange;
- Evaluate separable phase organic treatment;
- Test and evaluate the HLW melter;
- Test and evaluate the LAW melter;
- Evaluate melter off-gas treatment systems;
- Balance and determine the flowsheets size of recycle streams to accurately estimate equipment size and costs;
- Conduct performance assessment activities;

- Evaluate alternative approaches to durability testing; and
- Evaluate acceptance strategies for LAW and HLW waste forms.

This alternative would meet all applicable regulations for disposal of hazardous, radioactive, or mixed waste assuming that the hazardous waste components are adequately treated during waste processing or vitrification.

B.3.6 EX SITU NO SEPARATIONS ALTERNATIVE

B.3.6.1 General Description of the Alternative

The Ex Situ No Separations alternative is similar to the Ex Situ Intermediate Separations alternative except that there would be no separation of the waste into LAW and HLW; all waste would be handled as HLW. All of the waste would be vitrified or calcined without any pretreatment and placed in interim storage before being shipped to the potential geologic repository for final disposal. Consequently, there would be no LAW to be disposed of onsite.

Under the calcination option of this alternative, the waste would be calcined rather than vitrified. Calcination is the process of heating precipitates or residues to a temperature that is sufficiently elevated to decompose chemical compounds such as hydroxides or nitrates. Calcination differs from vitrification in that calcination temperatures would not necessarily cause the waste to melt and form a glass. Instead, the primary reaction product would be sodium carbonate. All of the waste would be retrieved from the tanks and calcined without any pretreatment. The calcined product (a dry powder) would be placed in large canisters for interim onsite storage before being shipped to the potential geologic repository for final disposal. For this alternative, no LAW would be disposed of onsite.

B.3.6.2 Facilities to be Constructed

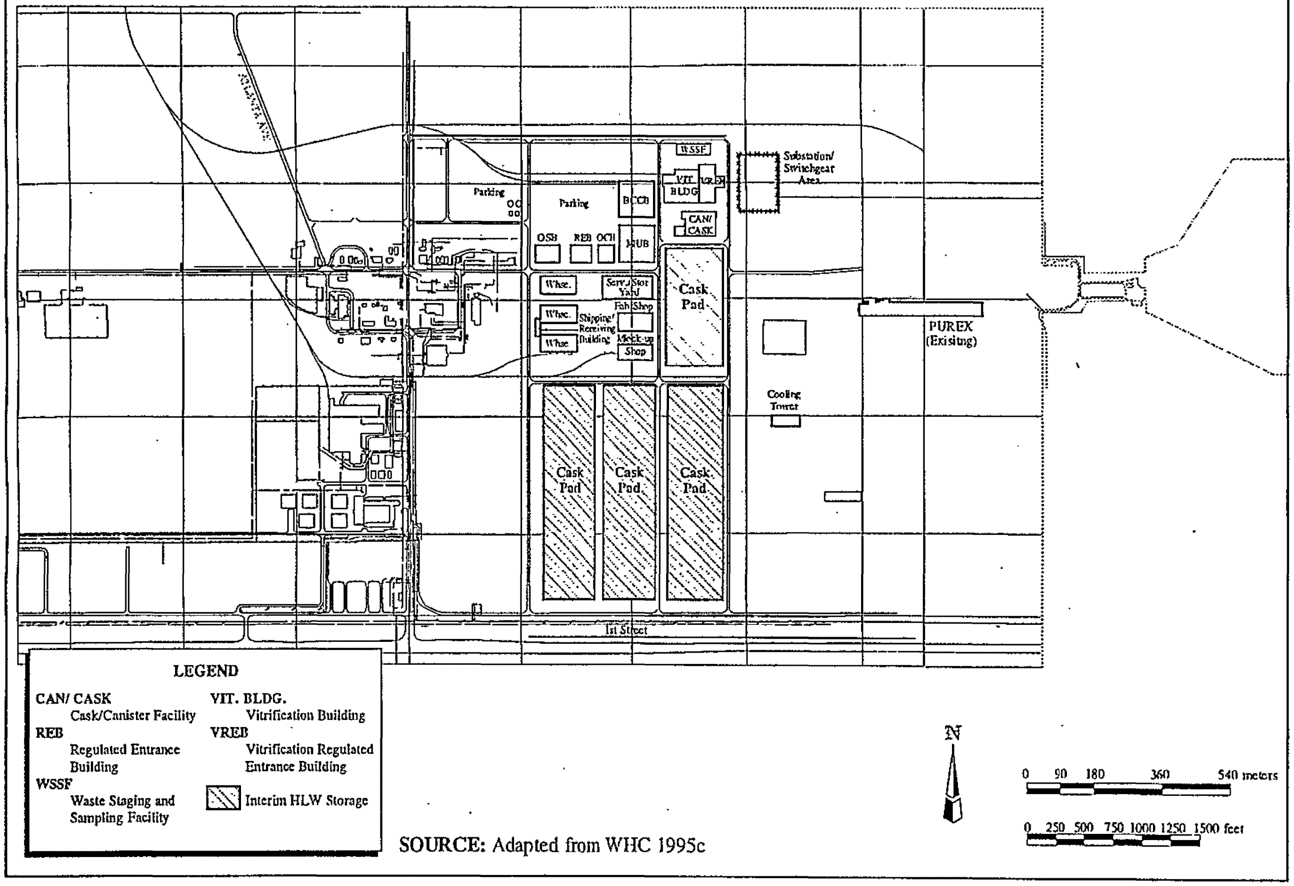
Tank Waste Retrieval and Transfer Facilities

The facilities that would be constructed for recovering and transferring tank waste to the calcination or vitrification facility are exactly the same for both alternatives with one exception. There would be no requirement for sludge washing for the No Separations alternative. The waste would be pumped directly from the Transfer Annex or the Waste Staging Facility to the receipt and sampling system at the processing plant (vitrification or calcination facilities).

Vitrification Facility

If vitrification is chosen for this alternative, a vitrification facility would be constructed. The single vitrification facility for the Ex Situ No Separations alternative would be similar to the Ex Situ Intermediate Separations alternative LAW vitrification facility with a few exceptions. The No Separations (Vitrification) facility would not have Cs ion exchange columns or LAW vaults for onsite near surface disposal. In place of the matrix and cullet mixing and containerization system it would have a system for packaging the cullet in canisters and overpacking them into HMPCs, which would be placed on interim storage pads to await offsite transport (Figure B.3.6.1).

Figure B.3.6.1 Ex Situ No Separations Facility Layout



SOURCE: Adapted from WHC 1995c

Because all of the waste would be considered high-level, separate HLW and LAW vitrification facilities would not be required. All of the waste would be vitrified in a single facility that would be virtually the same size as the LAW facility in the Ex Situ Intermediate Separations alternative (Section B.3.5). The off-gas treatment facilities would be identical in function to those described for the Ex Situ Intermediate Separations alternative.

Calcination Facility
If calcination is chosen for this alternative, a calcination facility would be constructed instead of a vitrification facility. The calcination facility would have a receiving and sampling system as in the Ex Situ Intermediate Separations facility. The calcination facility would not have a Cs ion exchange circuit, nor would the facility form cullet. Instead, it would have a system for processing the hot calcine and placing it in canisters. The canisters would be overpacked into HMPCs and placed on an interim storage pad and subsequently transported to the potential geologic repository for disposal. All of the waste would be calcined in a single facility. Because no engineering has been done for this alternative, the size of the facility has been estimated using engineering judgement. It is estimated that the calcination facility would be approximately the same size as the LAW facility in the Ex Situ Intermediate Separations alternative.

Support Facilities
All of the support facilities required for the Ex Situ Intermediate Separations alternative (Section B.3.5) would also be required in the same size and the same quantity for the Ex Situ No Separations alternative. As stated previously, there would be no LAW vaults for onsite waste disposal, but an increased area would be required for interim storage of the shipping casks for the HLW produced by the Ex Situ No Separations process.

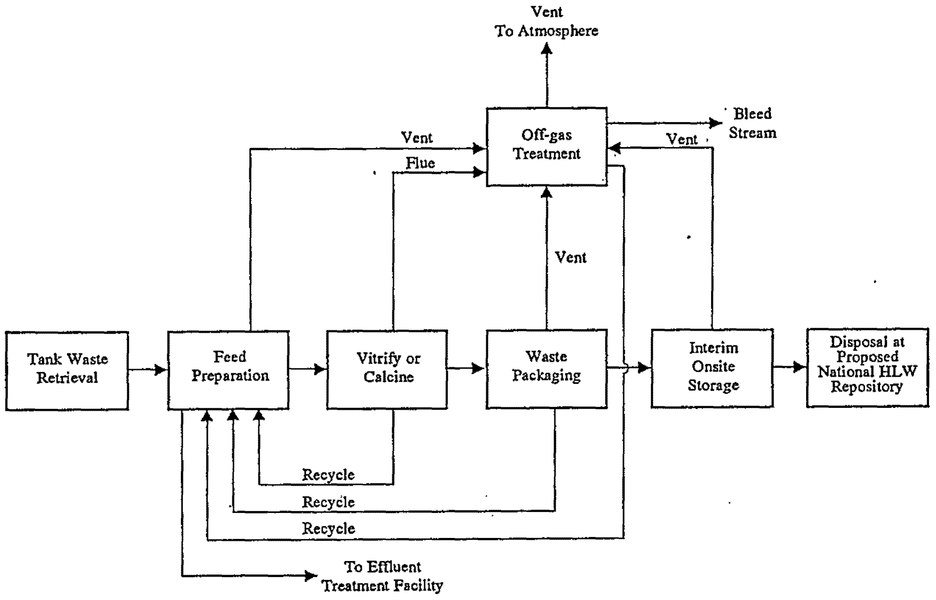
The support systems for the calcination process would be essentially the same as those for the other ex situ alternatives. These would include:

- Fuel receipt and storage area;
- Process steam and condensate;
- Cooling water supply and return;
- Sugar receipt and storage area;
- Breathing air and other bottled gases;
- Electrical supply;
- HVAC and process ventilation; and
- Health protection facilities.

B.3.6.3 Process Description

The process for the Ex Situ No Separations alternative is similar to the Ex Situ Intermediate Separations alternative except that the waste would not be separated into LAW and HLW, because all waste would be HLW. This HLW would be vitrified or calcined and transported to the potential geologic repository for disposal. Figure B.3.6.2 illustrates the process.

Figure B.3.6.2 Ex Situ No Separations Alternative - Process Flow Diagram



Vitrification Process

In the Ex Situ No Separations alternative, there would be no sludge washing or Cs extraction process. The waste recovered from the tanks would be pumped via the Waste Transfer Annexes or the Waste Staging Facility directly to the receiving area of the vitrification facility. Other than deleting the Cs extraction process, there would be no change to the receiving process. The main process flow would be identical with the Ex Situ Intermediate Separations alternative from the evaporator through the day bin, which feeds the equipment that mixes the molten sulfur with the glass cullet in that process. In the Ex Situ No Separations alternative, the day bin would feed a cullet containerization system. The recycle systems, off-gas systems, liquid waste systems, and utility support systems would also be functionally identical to those of the Ex Situ Intermediate Separations alternative. Each of the tank waste alternatives would make extensive use of recycle streams in the process to recycle volatile radionuclide and chemical constituents captured in the off-gas systems back into the treatment process. These recycle streams would be used to minimize the generation of secondary waste. A bleed stream would be required for the off-gas system for vitrification and calcination to avoid a continuous buildup of certain volatile radionuclide and chemical constituents, namely Tc-99 and Hg, in these recycle streams. For comparison purposes, it has been assumed that the bleed stream percentage would be the same (at 1 percent of the recycle stream) and that this secondary waste stream would be stabilized by some low temperature process (such as grout).

The canister-filling process would be similar to the canister-filling operation in the HLW facility in the Ex Situ Intermediate Separations alternative although with larger equipment. The Ex Situ Intermediate Separations alternative would produce 20 mt/day of HLW glass. The Ex Situ No Separations alternative would produce 200 mt/day of HLW glass. Other differences are that the container would be filled with loose cullet. The container would be a 1.67-m (5.5-ft) diameter by 4.57-m (15-ft) long canister that would be overpacked in an HMPC, which would be the same type of container used to overpack the HLW canisters described in the Ex Situ Intermediate Separations alternative.

Calcination Process

Calcination is the process of heating precipitates or residues to a temperature that is sufficiently elevated to decompose chemical compounds such as hydroxides or nitrates. It differs from vitrification in that calcination temperatures do not necessarily cause the reacting materials to melt and form a glass. Consequently, the final product of calcination is a solid or semi-solid, if certain products have been partially fused during the calcination process. Calcination techniques for solidifying radioactive waste similar to the TWRS waste have been studied previously, but no recent results are available.

Sugar calcination refers to a process in which sugar is mixed with the tank waste prior to calcination. The calcination process would consist of evaporating the remaining feed liquid water content and the sodium nitrate, nitrite and hydroxide salts reacting with sugar and oxygen to form sodium carbonate salt, nitrogen oxide, carbon dioxide, and water vapor. Pure oxygen would be supplied to the calciner for these reactions. The oxygen would also combust the organic materials present in the feed to produce carbon dioxide gas and water vapor. Because sodium carbonate has a sufficiently high melting point, 850 °C (1,560 °F), it would remain as a solid in the calcining process rather than melting. Without reacting with sugar, sodium nitrate melts at 308 °C (586 °F) and sodium nitrite melts at 271 °C (520 °F).

Feed Preparation

Because all of the tank waste would be calcined, the waste feed to this process would be identical to the feed to the HLW vitrification melter in the Ex Situ Intermediate Separations alternative (Section B.3.5). Because the feed components would not be separated, all of the calcined product would be considered HLW. The primary function of the feed preparation system would be to mix measured amounts of sugar with the tank waste prior to calcination. Each batch of tank waste would be analyzed to determine the sugar requirements. A weighed amount of bulk dry sugar would then be added, and the mixture would be agitated until the sugar was dissolved.

Calcination

The prepared feed, after first being screened to separate small amounts of coarse solids and foreign objects, would be pumped to the feed nozzles of a spray calciner. The calciner would be an indirectly fired vessel consisting of a number of 20-cm (8-in.) diameter vertical tubes. The vessel is a box design approximately 9 by 9 m (30 by 30 ft) with an approximate height of 4.6 m (15 ft). This particular configuration would limit the reacting mass within the calciner as the reaction of the sugar could be very rapid and large quantities of sugar and nitrates could react violently. The calcination reactions would take place inside the tubes. The tubes would be heated by combustion of kerosene fuel with oxygen outside the tubes and the resulting hot off-gases exhausted directly to the atmosphere, probably after some indirect heat recovery operation. These gases would consist of only products from the combustion of kerosene with oxygen and should require no treatment as they would contain only very low levels of SO_x and NO_x due to the presence of small amounts of sulfur and nitrogen in the kerosene.

The feed for the calciner would consist of a slurry containing approximately 50 percent by weight solids (dissolved and suspended). Atomizing steam at the rate of approximately one-half of the feed rate (on a mass basis) would be added to ensure proper dispersion of the spray inside the calciner tubes. The atomized waste droplets would lose their water by evaporation and be heated to reaction temperature by the indirectly heated tubes as they fell through the length of the tube. The chemical reactions of the waste with the sugar would take place with the release of NO_x gases and the formation of solids, which would be collected at the bottom of the calciner. The calciner would operate at a temperature between 700 and 800 °C (1,300 and 1,470 °F).

The evaporated water and injected steam for atomization along with the gaseous products from calcination would be exhausted to a ceramic candle filter where particulates would be removed from the hot gases, and then processed similar to vitrification off-gas treatment. The solids removed from the ceramic filter would be collected with the solids from the calciner for further processing and compaction. The ceramic filter equipment envelope would be approximately 9 m (30 ft) in diameter by 18 m (60 ft) high.

Compaction

The calcined solids would consist of a hot, fine powder with a low bulk density, and would require compaction to increase its bulk density. This fine powder would be hot processed in a roll-type compactor machine to produce small pellets or briquettes of high bulk density. The bulk density of the briquettes would be approximately 90 percent of the theoretical density of the solids. After compaction, the product briquettes would be screened to remove fines, air cooled, and transferred to the HLW cyclone bin for feeding into the canisters. The fines collected from screening the briquettes would be returned to the feed bin for recycle to the compactor machine.

Canister Operations

After the calcined product briquettes were transferred to the HLW cyclone bin, the vitrification process canister filling operation flowsheet would be used. The calcined briquettes would be placed in 1.67-m (5.5-ft) diameter by 4.57-m (15-ft) long canisters identical to the canisters used for glass cullet for Ex Situ No Separations alternative. A major difference is the quantity of calcine briquettes to be disposed. The Ex Situ No Separations alternative would produce 92 mt/day (100 tons/day) of HLW calcine briquettes. The number of canisters required for calcine briquettes would be 10,300, approximately 65 percent less than the 29,100 required for vitrification.

Off-Gas Treatment

Off-gas processing for calcination would be the same as that used for off-gas processing for vitrification. The HLW off-gas system would receive hot gases from the HLW calciner ceramic candle filter. The gases would be cooled and scrubbed with water to remove most of the remaining particulates and water soluble materials, which would be recycled to the process feed tanks. A small bleed stream from this recycle stream would be required to prevent a buildup of certain volatile radionuclides and chemical constituents. This secondary waste stream would be stabilized by some low temperature process (such as grout). The scrubbed off-gas would pass through a mist eliminator to remove fine water droplets and then through metal HEPA filters to remove the majority of the radionuclide particulates. The off-gas would then flow to an SO_2 adsorption process and a catalytic NO_x reactor before being discharged to the atmosphere. The amount of NO_x emissions estimated for the calcination process would be approximately five times larger than estimated for the vitrification process. The difference is caused by the assumption that reaction products of nitrites and nitrates for the calcination process would be NO_x, whereas for the vitrification process the assumption also includes a large quantity of nitrogen as a reaction product.

Post Remediation

When tank waste processing has been completed, the processing facilities would be decontaminated and decommissioned in the following manner.

- Processing equipment will be decontaminated sufficiently to allow onsite disposal in a low-level waste burial ground.
- Processing facilities will be decontaminated to the extent possible and then entombed in place. The exact materials that would be used to cover processing facilities have not been decided.

B.3.6.4 Implementability

Issues associated with implementing this alternative include the following.

- The Ex Situ No Separations (Vitrification) option has the same uncertainties as those listed for the Ex Situ Intermediate Separations alternative (Section B.3.5.4). In addition, this option would result in a large volume of vitrified HLW (2.91E+05 m^3 [1.0E+07 ft^3]). The calcination option would also produce a large volume of calcined HLW (1.0E+05 m^3 [3.7E+06 ft^3]); however it is approximately 65 percent less than the volume of vitrified HLW.
- The calcination step using sugar as a reductant has had limited laboratory testing and the proposed facilities are conceptual. Calcination as a unit operation has been in use for many years on an industrial scale. No design or engineering has been completed for the process or support facilities. Consequently, the processing steps have been based on experience and engineering judgement. It is estimated that the consumption of fuel (kerosene) for calcination would be approximately 10 percent of that required for vitrification. Steam use for calcination would be higher than for vitrification due to

the atomization steam required for feeding the calciner. Electrical power for calcination would be approximately 70 percent of that required for vitrification.

- The process design parameters for calcining, such as feed rate, temperature, reagent addition, and mass and energy balances remain conceptual in nature. A substantial part of the flowsheet for calcination and vitrification would be the same; implementation of the calcination and vitrification options is estimated to be of approximately the same size and complexity. As a result of this similarity, the nature of most support services is estimated to be similar for calcination and vitrification. Exceptions to this are that raw water use for the calcination option is estimated to be approximately 10 percent of that for vitrification, and sanitary water use for calcination is estimated to be approximately 70 percent of that for vitrification.
- It is estimated that the calcination and vitrification options would be approximately the same in size and complexity and therefore would have approximately the same costs for capital, monitoring and maintenance, decontamination and decommissioning, and research and development. Differences in cost occur in the operating category due to reduced cost of HLW casks/canisters and HLW disposal fees for calcination relative to vitrification (see Section B.10.0 for further discussion). The operating costs for calcination are estimated to be approximately 60 percent of that for vitrification, resulting in an estimated overall cost for calcination that is approximately 60 percent of that for vitrification.
- Further laboratory and pilot-plant testing is required for calcining, particularly for analyzing reaction products including the nature of the gas streams and off-gas treatment methods. The calciner and off-gas processing may require different sizes and types of equipment from the ones conceptualized for the EIS.
- Processes for retrieving, pretreating, and immobilizing waste often have been based on engineering judgement and assumptions, performance of processes (e.g., sludge sluicing, robotic arm solids removal, and producing HLW glass with a high waste loading) has been assumed in the absence of extensive quantitative data. Further process testing (vitrification or calcination) to determine equipment size would be necessary before plant engineering could proceed.
- Retrieval criteria that specifies recovering 99 percent of the waste volume in each tank may not be achievable. Recovering less tank waste would have a direct bearing on classifying the waste remaining in the tank.
- Performance requirements for shielding have not been generated. Exposure during retrieval is based on engineering judgement.
- Recovery of DST waste by agitating with turbine pumps has not been demonstrated. If the turbine pumps do not perform as expected, then additional retrieval methods would be necessary.

This alternative would meet all applicable regulations for disposal of hazardous, radioactive, or mixed waste assuming that the hazardous waste components would be adequately treated during waste processing and vitrification or calcining. However, the HLW forms (soda-lime glass or calcine) may not meet the current standard waste form (borosilicate glass) specified in the waste acceptance requirements (see Volume One, Section 6.2). The vitrified cullet waste form, with its high surface-area-to-volume ratio may not be acceptable for disposal at the potential geologic repository. These wastes forms might not be acceptable and would require acceptance criteria resolution, which could result in delayed acceptance. The compacted powder calcine also might not meet the waste acceptance requirement for immobilization of particulates.

B.3.7 EX SITU EXTENSIVE SEPARATIONS ALTERNATIVE

B.3.7.1 General Description of the Alternative

The Ex Situ Extensive Separations alternative is similar to the Ex Situ Intermediate Separations alternative but involves performing additional complex chemical separations processes to separate the HLW components from the recovered tank waste. The purpose of the Ex Situ Extensive Separations alternative is to process tank waste to produce a minimum number of vitrified HLW canisters, and reduce the curie loading of LAW to NRC Class A or as low as reasonably achievable, which ever is lower (WHC 1995c). Under the Ex Situ Extensive Separations alternative, the waste would be recovered from the tanks and a complex series of processing steps would be performed during pretreatment to separate HLW from LAW. A series of chemical processing operations would be used to separate HLW elements such as U, Pu, Np, thorium, americium, lanthanide (rare earth metals) series elements, Cs, Sr, and Tc from the waste. Under this alternative, the activities to be performed following pretreatment would be very similar to those included in the Ex Situ Intermediate Separations alternative. The HLW would be vitrified, stored onsite, and disposed of at the potential geologic repository. The LAW would be vitrified and placed in retrievable containers in a near-surface, disposal facility at the Hanford Site. This alternative would create a smaller volume of HLW being sent to the potential geologic repository. The resulting LAW requiring onsite disposal would be approximately the same volume but would have a lower radionuclide concentration than the Ex Situ Intermediate Separations alternative (WHC 1995e).

B.3.7.2 Facilities to be Constructed

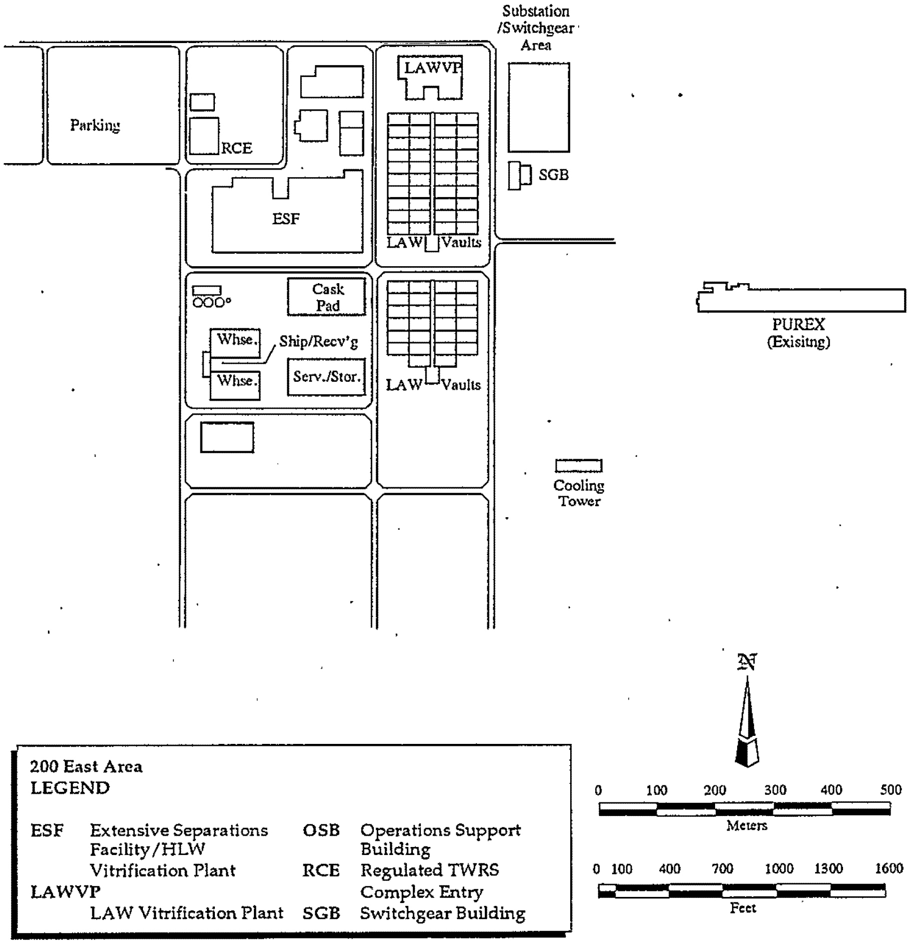
The main processing facilities would consist of an integrated pretreatment (chemical processing) and HLW vitrification facility and a detached LAW vitrification facility. The integrated pretreatment-HLW vitrification facility would be operated and maintained remotely, and the detached LAW vitrification facility would be contact operated and maintained (Figure B.3.7.1).

Integrated Pretreatment - High-Level Waste Vitrification Facility

The integrated facility would have an overall size of 94 by 230 m (310 by 770 ft) and a height of 40 m (130 ft), of which 20 m (65 ft) would extend belowgrade. The facility would be divided into three levels: a processing level, a level at grade, and a level at 12 m (40 ft) abovegrade. It would contain:

- Sludge washing and dissolution;
- Alkaline liquid processing;
- Acidic liquid processing;

Figure B.3.7.1 Ex Situ Extensive Separations Facility Layout



200 East Area
LEGEND

ESF	Extensive Separations Facility/HLW Vitrification Plant	OSB	Operations Support Building
LAWVP	LAW Vitrification Plant	RCE	Regulated TWRS Complex Entry
		SGB	Switchgear Building

SOURCE: Adapted from WHC 1995c

- Destruction, recovery, and recycle of bulk chemicals;
- Feed receipt and sampling;
- Chemical makeup; and
- HVAC.

The HLW vitrification portion of the integrated facility would have an overall size of 30 by 140 m (100 by 460 ft) and a height of 28 m (92 ft), of which 11 m (36 ft) would extend belowgrade. The facility would include:

- Melter operations;
- Maintenance areas;
- Canister loading;
- Cold chemical makeup; and
- HVAC.

Detached Low-Activity Waste Vitrification Facility

The LAW vitrification facility would have an overall size of 24 by 75 m (80 by 250 ft) with a height of 21 m (70 ft). The building would be aboveground and would include a process level, a grade level, and a level at +9 m (+30 ft). The facility would be divided into the following areas:

- Feed receipt and sampling;
- Melter operations;
- Cullet processing;
- Cold chemical makeup; and
- HVAC.

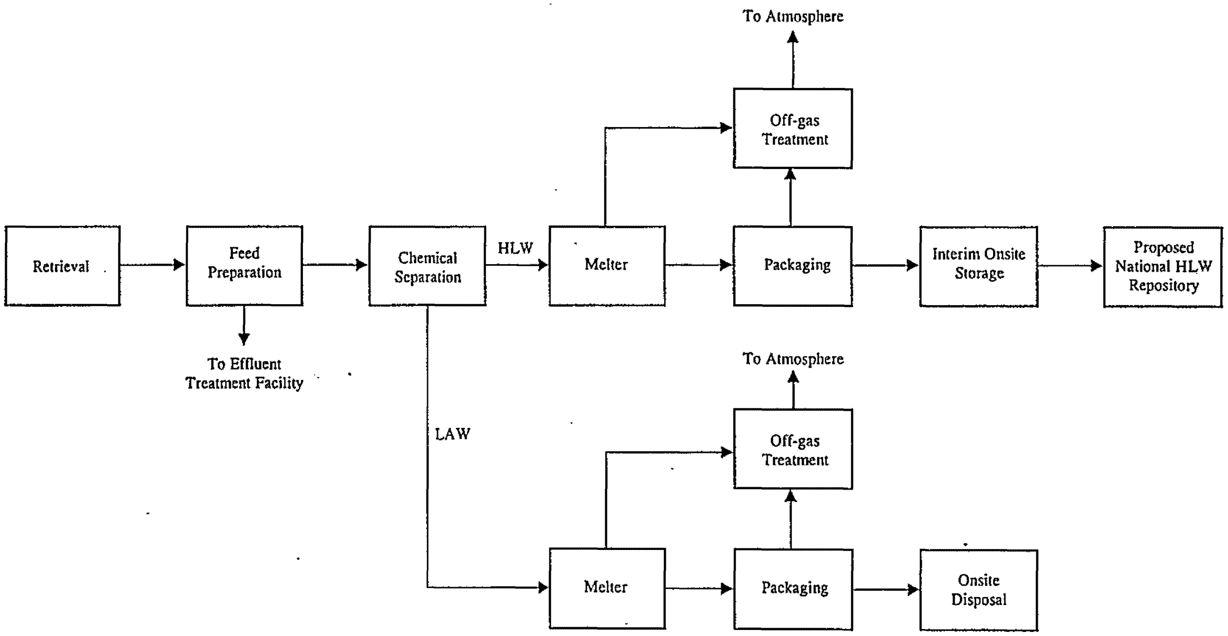
B.3.7.3 Process Description

The overall waste treatment process would include recovering and transferring the waste from the tanks, separating the HLW from the LAW, vitrifying the HLW, vitrifying the LAW, shipping the HLW offsite, and disposing of the LAW in onsite vaults in retrievable containers. The separation processes would include sludge washing, caustic and acid leaching, solvent extraction and ion exchange of acidic solutions, ion exchange of alkaline solutions, and recycling water, nitric acid, and sodium hydroxide to reduce HLW volumes. A process flow diagram is provided in Figure B.3.7.2.

Tank Waste Retrieval and Transfer

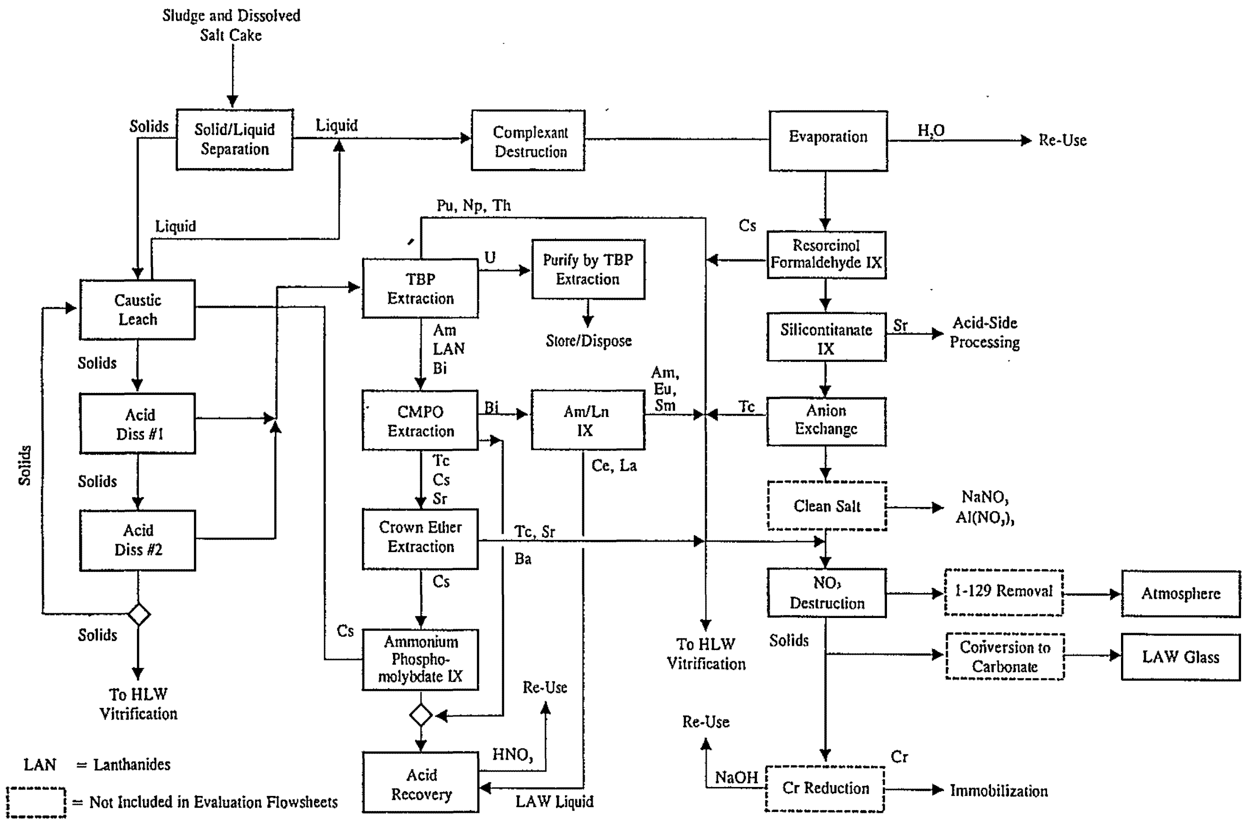
Recovering waste from SSTs and DSTs would not change from one ex situ process to another. Tank waste retrieval and transfer would be dependent on the content of the tanks, but would not be dependent on the processing of the waste. The recovery and transfer of the tank waste for the Ex Situ Extensive Separations alternative would be the same as that for the Ex Situ Intermediate Separations alternative. A full discussion of tank waste retrieval and transfer can be found in Section B.3.5.

Figure B.3.7.2 Ex Situ Extensive Separations Alternative - Process Flow Diagram (Sheet 1 of 2)



NOTE: Recycle streams not shown

Figure B.3.7.2 Ex Situ Extensive Separations Alternative - Process Flow Diagram (Sheet 2 of 2)



SOURCE: Adapted from WHC 1995e

Solids Separation and Dissolution

Liquid/Solid Separation

The Ex Situ Extensive Separations process would use centrifuges for separating liquid and solids in various stages of processing. These separations would occur after tank retrieval, complexing agent destruction, caustic dissolution, acid dissolution, and the chromium removal step. Several stages of liquid and solid separation would be used because supernate entrainment in the solids from the centrifuge would be assumed to make up about 12 percent of the centrifuge feed.

Destruction of Complexing Agents

The liquid resulting from liquid and solid separation would be treated by a wet air oxidation process to destroy organics, including complexing agents and ferrocyanides. The use of an organic destruction process is considered essential to break down the complexing agents that hold metal ions (such as Sr) in solution and prevent their extraction by subsequent processing. The wet air oxidation process has previous commercial application. In this process the liquid would be held at 325 °C (620 °F) and 14,000 kPa (2,000 psi) for 1 hour. The metals that would be released from their complexes would precipitate as hydroxides upon cooling. Hydroxides of Sr, nickel, calcium, and iron would occur along with coprecipitated TRU elements and lanthanides. Oxygen and hydroxide would react with organic constituents to form carbonates, oxalates, nitrogen, ammonia, and hydrogen.

Caustic Leach

Caustic leach is the first of three dissolution steps that would be used to reduce the amount of insoluble sludge that would ultimately be processed as HLW. Several hours of digestion at approximately 90 °C (200 °F) in appropriately designed reactors would be used to dissolve the desired elements. The caustic leach would be 4 molar in sodium hydroxide to solubilize aluminum, nickel ferrocyanide, and cancrinite. The liquid from caustic leaching would be added to the liquid from the initial liquid and solid separation, and the combined stream would be sent to the complexing agent destruction process. The solids from caustic leaching would then be sent to the first acid leach.

First Acid Leach

The first acid leach would be in a mixture that is 4.5 molar in nitric acid and approximately 0.3 molar in oxalic acid. This leaching operation would be expected to solubilize about 90 percent of the following substances: Cr^{+3}, Fe^{+3}, $Fe(CN)_6^{-3}$, Mn^{+2}, MnO_2, Ni^{+3}, PO_4^{-3}, Pu^{+4}, SO_4^{-2}, and Zr^{+4}. The solids from the first acid leach would then be sent to the second acid leach, while the liquid from both acid leaches would be combined and sent to solvent extraction of acidic liquid.

Second Acid Leach

The second acid leach would be in a mixture that is 4.5 molar in nitric acid and approximately 1 molar in hydrofluoric acid. This leaching operation would solubilize the remaining solids to the maximum extent possible. Most of the undissolved material from the second acid leach would be recycled to the caustic leaching operation, while a minor fraction of the undissolved solids would be sent to the HLW vitrification operation.

Purification of Acid Soluble Radionuclides

Tributyl Phosphate Extraction of Transuranic Compounds

The active extractant for this solvent extraction process would be the same as used in the PUREX process, which is 30 percent TBP in a hydrocarbon diluent. In the first extraction, U, Pu, and Np would be extracted into the organic phase. The extracted Pu and Np would be selectively stripped into an aqueous phase and sent to the HLW vitrification process. The U would be stripped separately in a third processing step, and recovered for reuse by re-extraction and re-stripping. This U (approximately 1,400 mt [1,500 tons]) would be available for reuse if a market for the U could be found.

N-diisobutylcarbamoylmethylphosphine Oxide Solvent Extraction

The raffinate from the first TBP cycle would be sent to a N-diisobutylcarbamoylmethylphosphine oxide (CMPO) solvent extraction process to remove trivalent lanthanides, americium, and bismuth.
The solvent would be 0.2 molar CMPO and 1.4 molar TBP in a hydrocarbon diluent, which has also been proposed for the transuranic extraction (TRUEX) process. Americium and trivalent lanthanides would be stripped from the organic phase into dilute nitric acid. Bismuth would be removed separately in a separate wash step with a sodium carbonate and EDTA solution.

Am and Lanthanide Ion Exchange

This separation would be accomplished by band displacement cation exchange using cation exchange resins loaded in sequence. Concentrate from CMPO stripping would be loaded on the resin in preparation for separation by displacement cation exchange. Elution of the resin would be with diethylenetriaminepentaacetic acid (DTPA) onto a second zinc-loaded resin. Continued elution would occur through a series of columns established discrete bands of metal ions in sequence depending on the formation constants of the metal ion DTPA complexes. The elution effluent would be divided into three portions. The first and third portions would be sent to the LAW process stream. The second portion would be sent to the HLW process stream.

Crown Ether Solvent Extraction

The raffinate from CMPO extraction would contain Cs, Sr, and Tc, and would require further processing to remove these elements. This raffinate would be concentrated by evaporation, and subsequently contacted with a crown ether solvent (0.2 molar in diluent) to remove these elements. They would be stripped in a second contact, and the strip solution would be concentrated by evaporation and then sent to the HLW process stream.

Ammonium Phosphomolybdate Ion Adsorption

The final acidic processing step would use ammonium phosphomolybdate (APM) to remove Cs from the raffinate from the crown ether extraction process. The adsorbent would be 10 percent APM on an alumina substrate. Because Cs cannot readily by eluted from APM, the loaded sorbent would be transferred to the caustic leach step of the sludge dissolution process. The caustic leach would dissolve 90 percent of the sorbent, releasing Cs into the basic leach liquid.

Removal of Radionuclides From Alkaline Liquid

Cesium Ion Exchange

The combined liquid from caustic leach and complexant destruction would be evaporated to 7 molar sodium hydroxide and put through ion exchange columns containing a resorcinol-formaldehyde ion exchange resin that removes Cs from basic solutions. Four ion exchange columns would be used, with three used for extraction, and the fourth undergoing elution with 1 molar formic acid. The eluted Cs would be sent to HLW process stream.

Strontium Removal by Silicotitanate

The basic stream from Cs ion exchange would be sent to a column containing crystalline silicotitanate, where the Sr in solution would be adsorbed irreversibly. The Pu and Cs could also be adsorbed on the silicotitanate. Because elution is not possible, the loaded adsorbent would be transferred to the acid dissolution reactors, where the silicotitanate would be dissolved, releasing Sr into acid solution.

Technetium Ion Exchange

The raffinate from Sr removal would be sent to strong base anion exchange columns where Tc would be removed as the pertechnetate ion (TcO4-). Elution from the ion exchanger would be by 6 molar nitric acid. The eluant would be concentrated by evaporation and sent to HLW treatment. Nitric acid would be recovered from the evaporator overheads and recycled. The raffinate from Tc removal would be sent to the LAW process stream.

High-Level Waste Concentration and Denitration

From the separation steps described previously, the HLW streams would be combined and concentrated by vacuum evaporation to remove nitric acid until the remaining liquid was a 3 molar nitric acid. The dilute overheads from the evaporator would be sent to acid recovery for reuse as a bulk chemical. The 3 molar nitric acid and the raffinate from the Cs ion exchange process would be combined and undergo denitrification by reaction with sucrose. Sufficient sucrose would be supplied to achieve 0.5 molar nitric acid in the liquid after sucrose conversion. This liquid would be fed to a HLW centrifuge process along with undissolved solids from the final acid dissolution step. The NO_x produced would be sent to the acid recovery system for conversion to nitric acid.

Low-Activity Waste Concentration

The LAW streams from the previously described sections would be combined and concentrated by evaporating water to a 7 molar sodium hydroxide solution. The evaporator bottoms product would form the feed to the LAW calcination process, while the evaporator overheads would be used for dilution water or recycled to wash operations in the various separation processes. The LAW remaining following the separations processes would contain approximately 0.32 MCi of radioactivity including: 7.0E-02 MCi of Cs and Ba, 1.10E-02 MCi of Sr and Y, 1.56E-04 MCi of Tc-99, and 1.22E-03 MCi of TRU isotopes.

Recovery and Reuse of Bulk Chemicals

The recovery and reuse of bulk chemicals would take place in four major unit operations. These would be water evaporation and reuse, nitric acid distillation, nitrate destruction, and recovery and recycle of sodium hydroxide. Water from the various process evaporators would ultimately be routed to a wash water tank, where the recycled water would be used to meet the dilution requirements of other parts of the process. Acidic evaporator overheads would be contacted with the NO_x streams from the denitration and calcination steps. Hydrogen peroxide and air would be added to convert the NO_x to nitric acid. The resulting dilute acid would be concentrated to recover the nitric acid, and the water formed would be used as recycle. The caustic slurry produced in the calcination operation would be evaporated to produce a strong sodium hydroxide solution, which would be recycled to meet process requirements. Excess caustic slurry would be disposed of with the LAW.

Removal of Heavy Metals

A chromium-reduction process would be included to reduce chromium $(Cr)^{+6}$, which is mobile in groundwater, to Cr^{+3}, which precipitates as the hydroxide and does not have a high mobility. While this would keep the Cr from entering the groundwater, the process would result in a Cr product that would require disposal as a mixed waste. The process would employ the addition of 1.5 molar ammonium hydroxide as a reductant, which would be expected to reduce 99 percent of the Cr. Nitrogen gas would evolve during the reduction reaction and would be vented to the process stack. Insoluble Cr would be removed after reduction by centrifuging, and the solids would be sent to a separate waste processing step.

Clean Salt Process

This process represents a concept that potentially would reduce the LAW volume. The primary salts produced by the process would be sodium nitrate and aluminum nitrate. There is a concern that Cs-137 would also be extracted by the process and cause Cs-137 to enter the LAW stream. Varying degrees of decontamination could be achieved by increasing the number of recrystallization stages that are used on the waste stream.

B.3.7.4 Description of Immobilization and Off-Gas Treatment

High-Level Waste Melter

The evaporator bottoms from the HLW evaporator would be routed to the melter feed section. After sampling, cooling, and adjusting the slurry, it would be transferred to the melter feed system. This would be a batch system, which would mix the slurry and glass-forming frit. This mixture would then be continuously fed to the HLW melter system. The high temperature of the melter would convert the incoming feed slurry to molten glass containing 20 percent waste oxides. The HLW melter would be joule-heated and operate at a temperature of 1,200 °C (2,200 °F). Volatilized melter feed components would form a separate off-gas stream that would pass to off-gas processing. Periodically the molten glass would be poured into cylindrical stainless steel canisters. The glass-filled canisters would be plugged and welded shut before being decontaminated to remove surface decontamination.

The cooled canisters would be taken to interim onsite storage before final transportation to the potential geologic repository.

High-Level Waste Off-Gas Processing

The HLW off-gas processing for the Ex Situ Extensive Separations alternative is similar to the HLW off-gas processing for the Ex Situ Intermediate Separations alternative. Each of the tank waste alternatives that uses high-temperature processing (vitrification or calcination) would make extensive use of recycle streams to recycle back into the treatment process volatile radionuclide and chemical constituents captured in the off-gas systems. The recycle streams would be used to minimize the generation of secondary waste. For this alternative, it has been determined that a bleed stream would be required for each alternative to avoid a continuous buildup of certain volatile radionuclide and chemical constituents, namely Tc-99 and Hg, in these recycle streams. For comparison purposes, it has been assumed for this alternative that the bleed stream percentage would be the same one percent of the recycle stream and that this secondary waste stream would be stabilized by some low temperature process (such as grout). The HLW off-gas system would receive hot gases from the HLW melter. The gases would be first cooled and scrubbed with water to remove most of the particulates and water soluble materials. The quenched off-gas would pass through a mist eliminator to remove fine water droplets and then through HEPA filters to remove the majority of the radionuclide particles. The scrubbed off-gases would flow to an SO_2 adsorption process and a catalytic NO_x reactor before being discharged. The SO_2 would be removed by adsorption on CuO beds prior to NO_x destruction. The desorbed sulfur as hydrogen sulfide would be converted into elemental sulfur by a Claus Unit, which would discharge its sulfur product to the LAW vitrification facility for use in LAW cullet disposal.

Low-Activity Waste Melter

Evaporator bottoms from the LAW evaporator would be sampled, cooled, and adjusted before being transferred to the LAW melter feed system. The melter feed and dry-glass formers would be fed into a combustion melter where they would combine and form molten LAW glass. The LAW glass would exit the melter and pass through a quenching and crushing stage resulting in pea-sized fractured glass known as cullet. The final design decision concerning the form of the LAW glass has not yet been made. While this alternative is based on the concept of glass cullet, ultimately other forms such as canisters or monoliths could be chosen. For purposes of calculating impacts for this EIS, it was assumed that the cullet would be analyzed to ensure that it meets product specifications, mixed with a matrix material, placed into large disposal containers, and transported to onsite vaults for disposal. The final waste form matrix for the cullet has not been specified. Various types of waste form matrices available are discussed in Section B.9.3.

Low-Activity Waste Off-Gas Processing

The LAW off-gas processing for the Ex Situ Extensive Separations alternative is similar to the LAW off-gas processing for the Ex Situ Intermediate Separations alternative. The LAW off-gas system would receive hot gases from the LAW melter. The gases would be first cooled and scrubbed with water to remove most of the particulates and water soluble materials. The quenched off-gas would pass through a mist eliminator to remove fine water droplets and then through HEPA filters to remove the majority of the radionuclide particles. The scrubbed off-gases would flow to an SO_2 adsorption process and a catalytic NO_x reactor before being discharged to the atmosphere. The recovered SO_2 would be converted into elemental sulfur by a Claus Unit, which would discharge its sulfur product to the LAW vitrification facility for use in LAW cullet disposal.

Low-Activity Waste Calcination

The bottoms from the LAW and other feed streams would be fed to a modified plasma arc calcination process for destroying nitrate and recovering sodium hydroxide. The main modification would be using ammonia as the combustion fuel. The calciner feed would be heated to 800 °C (1,470 °F) under atmospheric pressure that would vaporize the contained water and destroy sodium nitrate. The calciner off-gases would be quenched, water scrubbed, reacted to remove NO_x, filtered, and sent to the process stack. The calciner molten salt stream would then be redissolved in a water quench. The quench solution would be expected to contain the majority of the Cs and Tc.

Post Remediation

When processing of the tank waste has been completed, the processing facilities would be decontaminated and decommissioned in the following manner.

- Processing equipment will be decontaminated sufficiently to allow onsite disposal in a low-level waste burial ground.
- Processing facilities will be decontaminated to the extent possible and then entombed in place. The exact materials which will be used to cover processing facilities have not been decided.

B.3.7.5 Implementability

The Ex Situ Extensive Separations alternative has the same uncertainties for retrieving and transferring the waste as those listed for the Ex Situ Intermediate Separations alternative (Section B.3.5.4). In addition, this alternative consists of concepts that are intended to reduce the volume of HLW. Many of these concepts have no testing to affirm their applicability. The key issues relating to this alternative are:

- The performance of key processes has been assumed in the absences of substantive data. Further testing and development would be required to ensure that the processes would function as intended and make the required separations; and
- Quantitative performance requirements have not been established for many of the processes and functions. Further engineering would be dependent on developing a process that will meet the quantitative performance requirements.

The HLW canisters produced under this alternative would have a higher thermal loading than other alternatives and the assumed method of interim onsite storage, which relies on dry storage with passive cooling, would require further evaluation. This alternative may require using a storage facility with active cooling to remove decay heat generated by the vitrified HLW.

This alternative would meet all applicable regulations for disposal of hazardous, radioactive, or mixed waste assuming that the hazardous waste components are adequately treated during waste processing and vitrification.

B.3.8 EX SITU/IN SITU COMBINATION ALTERNATIVES

B.3.8.1 General Description of the Alternatives

The Ex Situ/In Situ Combination 1 and 2 alternatives were developed to assess the impacts that would result if a combination of two or more of the tank waste alternatives were selected for implementation. Because the tank waste differs greatly in the physical, chemical, and radiological characteristics, it may be appropriate to implement different alternatives for different tanks. There is a wide variety of potential combinations of alternatives that could be developed, and there are many potential criteria that could be used to select a combination of alternatives for implementation. The Ex Situ/In Situ Combination 1 and 2 alternatives described in the following text were developed to bound the impacts that could result from a combination of alternatives, and are intended to represent a wide variety of potential alternatives that could be developed to remediate the tank waste.

The Ex Situ/In Situ Combination alternatives represent a combination of the In Situ Fill and Cap and Ex Situ Intermediate Separations alternatives. Under the approach used to represent this alternative, tanks would be evaluated on a tank-by-tank basis to determine the appropriate remediation method based on the contents of the tank. The objective would be to effectively treat the tank waste in a manner that has acceptable risk and less overall cost than the Ex Situ Intermediate Separations alternative. This objective could be achieved by selecting tanks for ex situ treatment based on their contribution to post-remediation risk. Those tanks that are not selected for ex situ treatment would be treated in situ by filling and capping. Waste from tanks selected for ex situ treatment would be retrieved from the tanks and transferred to processing facilities for treatment. Closure activities would consist of filling those tanks selected for ex situ treatment with gravel and constructing a Hanford Barrier over all tank farms as well as the LAW retrievable disposal vaults from ex situ treatment.

Ex Situ/In Situ Combination 1 Alternative

Approximately one-half of the volume of the tank waste would be treated by the ex situ method and one-half would be treated by the in situ method. By selectively retrieving tanks for ex situ treatment, approximately 90 percent of the contaminants that contribute to long-term risks would be disposed of ex situ while retrieving only 50 percent of the waste.

Ex Situ/In Situ Combination 2 Alternative

Approximately 30 percent of the volume of the tank waste would be treated by the ex situ method and 70 percent would be treated by the in situ method. By selectively retrieving tanks for ex situ treatment, approximately 85 percent of the contaminants that contribute to long-term risks would be disposed of ex situ while retrieving only 30 percent of the waste.

B.3.8.2 Selection Process

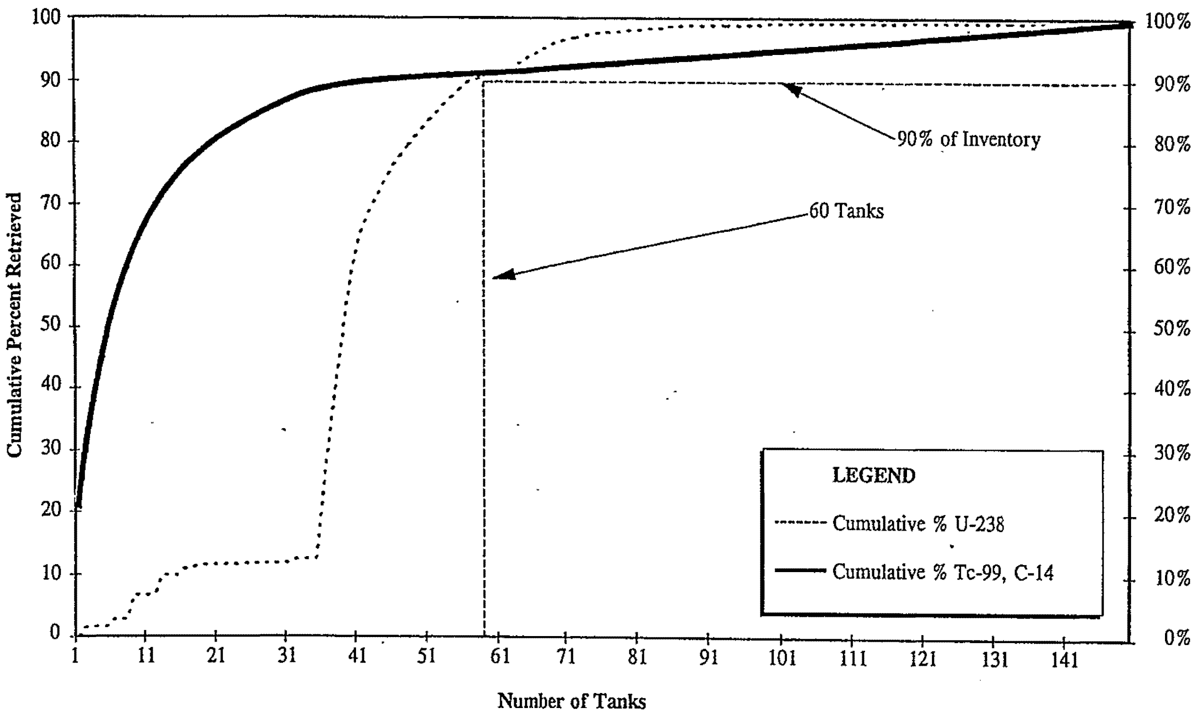
Ex Situ/In Situ Combination 1 Alternative

There are many potential criteria that could be used to develop a selection process. Additional waste characterization and analysis would be necessary to implement this alternative. The Ex Situ/In Situ Combination 1 alternative presented in the EIS is an alternative that was developed to represent the numerous alternatives that could be chosen. For example purposes, this EIS has examined a selection process based on retrieving those tanks containing substances that represent the greatest risk to human health. This example selection process may not be the exact selection criteria that would be chosen, but it illustrates the impacts of the Ex Situ/In Situ Combination 1 alternative.

The objective of the selection process was to examine the published characteristics of the radionuclides in the tanks and select the minimum number of tanks to be retrieved that would result in a risk of contracting cancer to a hypothetical onsite farmer in the future that would be comparable to the ex situ alternatives. Examining the risk calculations results for the Ex Situ Intermediate Separations alternative demonstrated that recovering 90 percent of the mobile constituents from the tanks would meet the established criteria and result in residual risks that fall between those for Ex Situ Intermediate Separations and In Situ Fill and Cap alternatives. The risk calculations showed that the long-term risks were caused by the mobility of four tank waste constituents: U-238, Tc-99, C-14, and I-129. Consequently, the selection process chosen was one in which 90 percent of these mobile constituents would be retrieved, assuming that 99 percent of the contents of any given tank could be retrieved. The selection process for the DSTs and SSTs was based on the same principle. Similarly, risk calculations showed that only a single chemical constituent, the nitrate anion, resulted in a Hazard Index (HI) value of greater than 1.0 for the hypothetical onsite farmer in the future. Because nitrate is present in all the tanks in amounts far exceeding those of the radionuclides (107,00 mt), virtually all of the tanks would have to be retrieved to recover 90 percent of this constituent.

The tank inventory for the SSTs showed the following amounts for the mobile species: U (1,423 mt); Tc-99 (1.64 mt); I-29 (0.24 mt); and C-14 (0.004 mt). The U is present in amounts almost 1,000 times greater than the remaining mobile elements. The selection process started by assuming retrieval of the tank with the greatest published U content, tank TX-113. The next tank selected was the one with the second highest U content, tank BY-104. The selection process was repeated until the cumulative U recovery was 90 percent, and was then repeated for the remaining three mobile elements until their cumulative recovery reached 90 percent. The results of this procedure, as displayed in Figure B.3.8.1, show that the cumulative retrieval of the constituents of concern would be 90 percent by the time that 60 SSTs have been retrieved. This procedure would also recover approximately 85 percent of the Cs and 65 percent of the Sr remaining in the SSTs.

Figure B.3.8.1 Ex Situ/In Situ Combination 1 - Single-Shell Tanks



For the DSTs, the procedure was similar, but modified slightly because the published data (WHC 1995d) do not report U in the DSTs. The selection process for the DSTs was to retrieve the tanks based on their Tc-99 content until the cumulative recovery was 90 percent, then retrieve additional tanks as required until the cumulative recovery of C-14 was 90 percent. The results of this modified procedure, as displayed in Figure B.3.8.2, show that the cumulative retrieval of Tc-99 and C-14 would be 90 percent when 10 selected tanks have been retrieved. This process would recover approximately 85 percent of the Cs and Sr in the DSTs. While the selection process was directed towards retrieving mobile groundwater radionuclides, an additional benefit was retrieving 69 percent of the nitrate and waste from 25 of the 50 current Watchlist tanks.

Ex Situ/In Situ Combination 2 Alternative

The selection process described for Ex Situ/In Situ Combination 1 was modified to provide for the ex situ treatment of the largest contributors to long-term risks (Tc-99, C-14, I-129, and U-238) while limiting the waste to be processed. This modified selection process also included Np-237 in the tank selection process. This modified selection criteria resulted in 25 tanks selected for ex situ treatment instead of 70 tanks, based on the currently available characterization data. The actual number of tanks selected would be based on future characterization of the tanks.

By selecting the appropriate tanks for ex situ treatment, up to 85 percent of the constituents that are the greatest contributors to long-term risk would be disposed of ex situ while retrieving approximately 30 percent of the waste. Under the Ex Situ/In Situ Combination 2 alternative with the retrieval of approximately 25 selected tanks, 85 percent of Tc-99, 80 percent of C-14, 80 percent of I-129, and 50 percent of the U-238 would be retrieved rather than 90 percent as with the Ex Situ/In Situ Combination 1 alternative (Figure B.3.8.3).

B.3.8.3 Facilities to be Constructed

Ex Situ/In Situ Combination 1 Alternative

Construction activities required for this alternative would involve constructing all of the facilities identified in the Ex Situ Intermediate Separations alternative and In Situ Fill and Cap alternative, but at a reduced scale. For the ex situ portion, the volume of waste requiring treatment and immobilization would come from 70 tanks instead of 177 tanks. In situ treatment would be required for the remaining tanks.

The following list identifies the major activities that would take place during the construction phase for the ex situ component of the Ex Situ/In Situ Combination 1 alternative:

- Install retrieval and transfer facilities;
- Construct separations facilities;
- Construct a HLW vitrification facility;
- Construct a LAW vitrification facility; and
- Construct a LAW disposal facility (vaults).

Figure B.3.8.2 Ex Situ/In Situ Combination 1 - Double-Shell Tanks

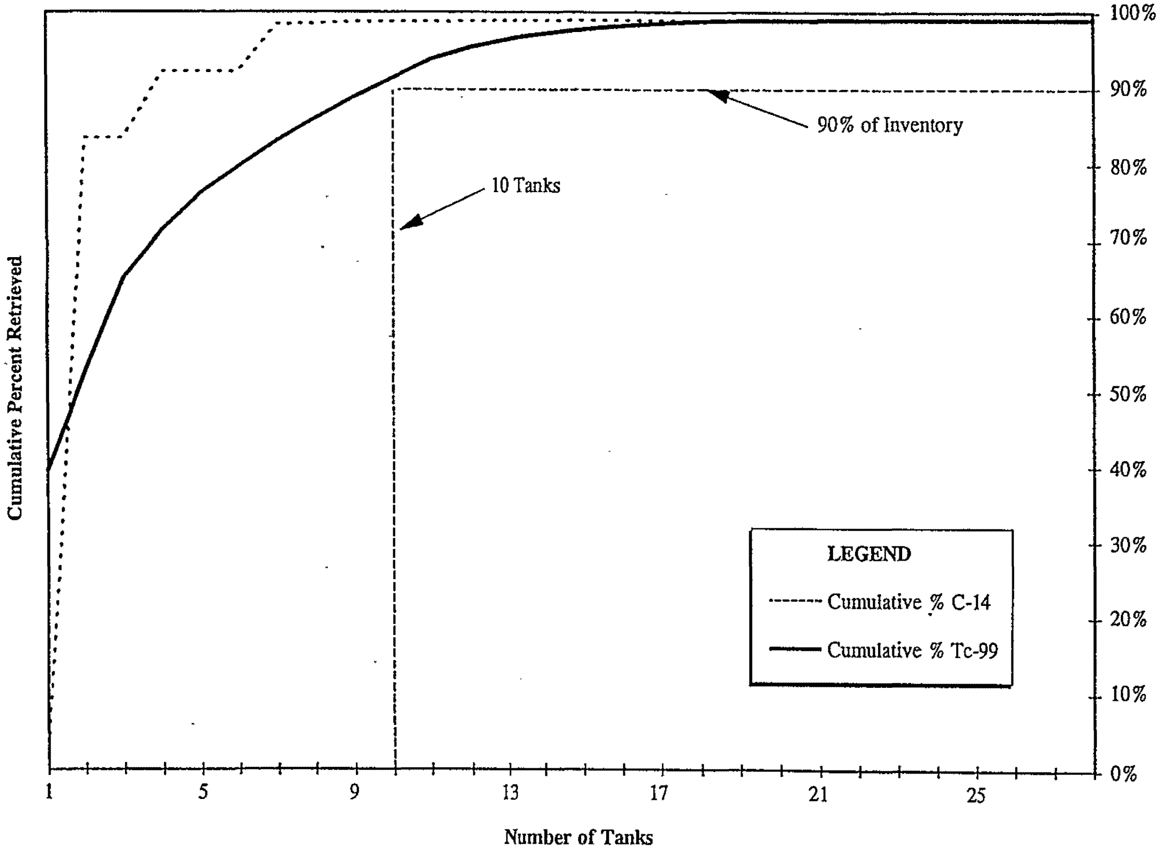
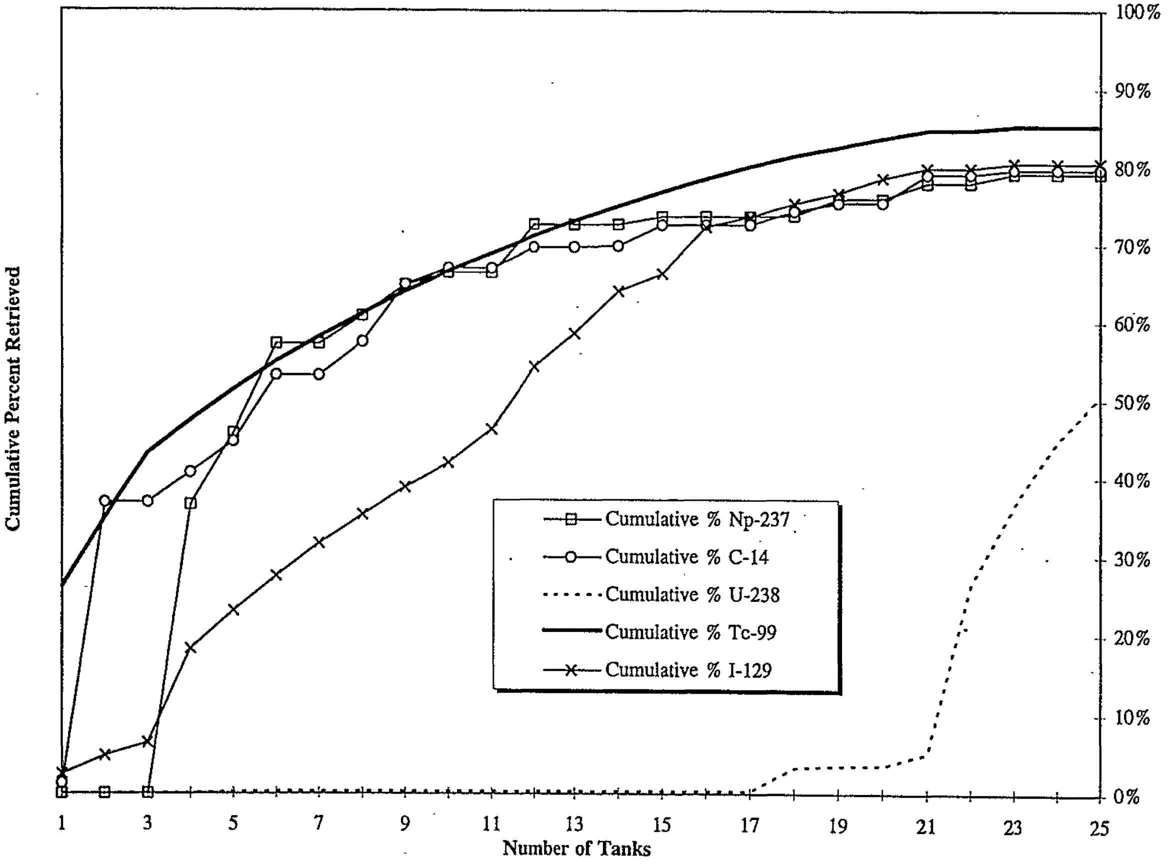


Figure B.3.8.3 Ex Situ/In Situ Combination 2 - Single- and Double-Shell Tanks



For the in situ component of this alternative, the following construction activities would take place:

- Install gravel handling systems; and
- Construct gravel storage sites for stockpiles.

A detailed description of facilities to be constructed for the Ex Situ Intermediate Separations alternative is included in Section B.3.5.2. A description of facilities to be constructed for the In Situ Fill and Cap alternative is included in Section B.3.3.2.

Ex Situ/In Situ Combination 2 Alternative

Construction activities required for this alternative would involve constructing all of the facilities identified in the Ex Situ Intermediate Separations alternative and In Situ Fill and Cap alternative, but at a reduced scale (Figure B.3.8.4). For the ex situ portion, the volume of waste requiring treatment and immobilization would come from 25 tanks instead of 177 tanks. In situ treatment would be required for the remaining tanks.

The following list identifies the major activities that would take place during the construction phase for the ex situ component of the Ex Situ/In Situ Combination 2 alternative:

- Install retrieval and transfer facilities;
- Construct separations facilities;
- Construct a HLW vitrification facility;
- Construct a LAW vitrification facility; and
- Construct a LAW disposal facility (vaults).

For the in situ component of this alternative, the following construction activities would take place:

- Install gravel handling systems; and
- Construct gravel storage sites for stockpiles.

A detailed description of facilities to be constructed for the Ex Situ Intermediate Separations alternative is included in Section B.3.5.2. A description of facilities to be constructed for the In Situ Fill and Cap alternative is included in Section B.3.3.2.

B.3.8.4 Description of the Process

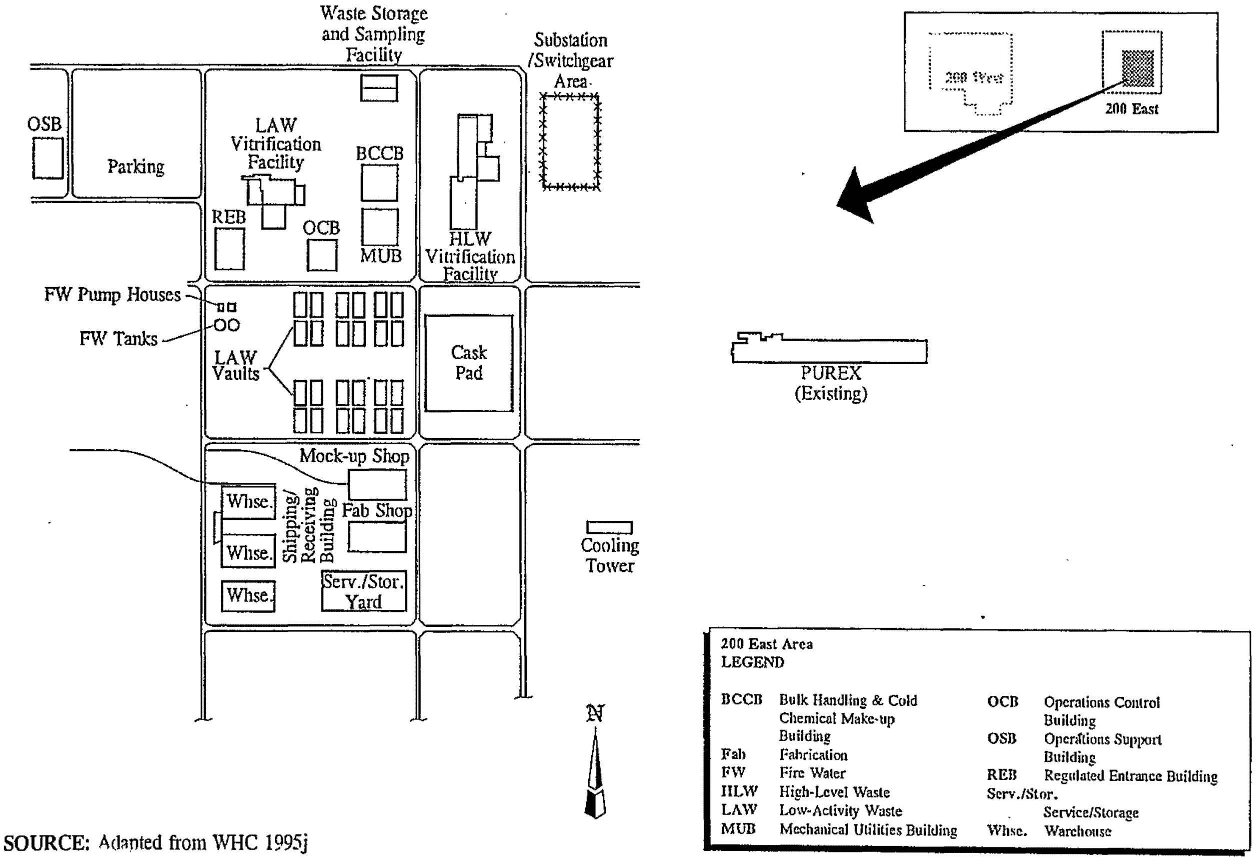
Processing Retrieved Waste

The waste that would be retrieved under either of the combination alternatives would be treated using the process identified for the Ex Situ Intermediate Separations alternative. For further details of the process, see Section B.3.5.

Processing Nonretrieved Waste

Tanks that would not be selected for retrieval under either of the combination alternatives would be treated in situ using the methods identified in the In Situ Fill and Cap alternative. For further details of this alternative, see the In Situ Fill and Cap alternative in Section B.3.3.

Figure B.3.8.4 Ex Situ/In Situ Combination 2 Facility Layout



SOURCE: Adapted from WHC 1995j

Post Remediation

After remediation, tank farm closure and decontamination and decommissioning would take place. Tank farm closure would involve the following activities:

- Retrieved tanks would be stabilized with gravel (in situ tanks would have been stabilized during in situ operations);
- Tank farm structures such as MUSTs, pump pits, valve boxes, and diversion boxes would be stabilized with grout; and
- Hanford Barriers would be constructed over SSTs, DSTs, and LAW retrievable disposal vaults.

Decontaminating and decommissioning equipment and processing facilities would include disposing of noncontaminated material by entombing in place onsite and disposing of contaminated equipment and materials at onsite low-level waste burial grounds.

B.3.8.5 Implementability

Because these alternatives represent a combination of alternatives, the implementability is also a combination of those issues identified in discussing the implementability of both the In Situ Fill and Cap alternative and the Ex Situ Intermediate Separations alternative (Sections B.3.3.4 and B.3.5.4, respectively). However, developing acceptable tank selection criteria is unique to the Ex Situ/In Situ Combination 1 and 2 alternatives and would require more complete and accurate waste characterization than currently exists. There are numerous ways to fully develop these alternatives. The final selection criteria would be based on tank characterization program results, short-term versus long-term risks, and additional development of the Ex Situ Intermediate Separations and In Situ Fill and Cap alternatives.

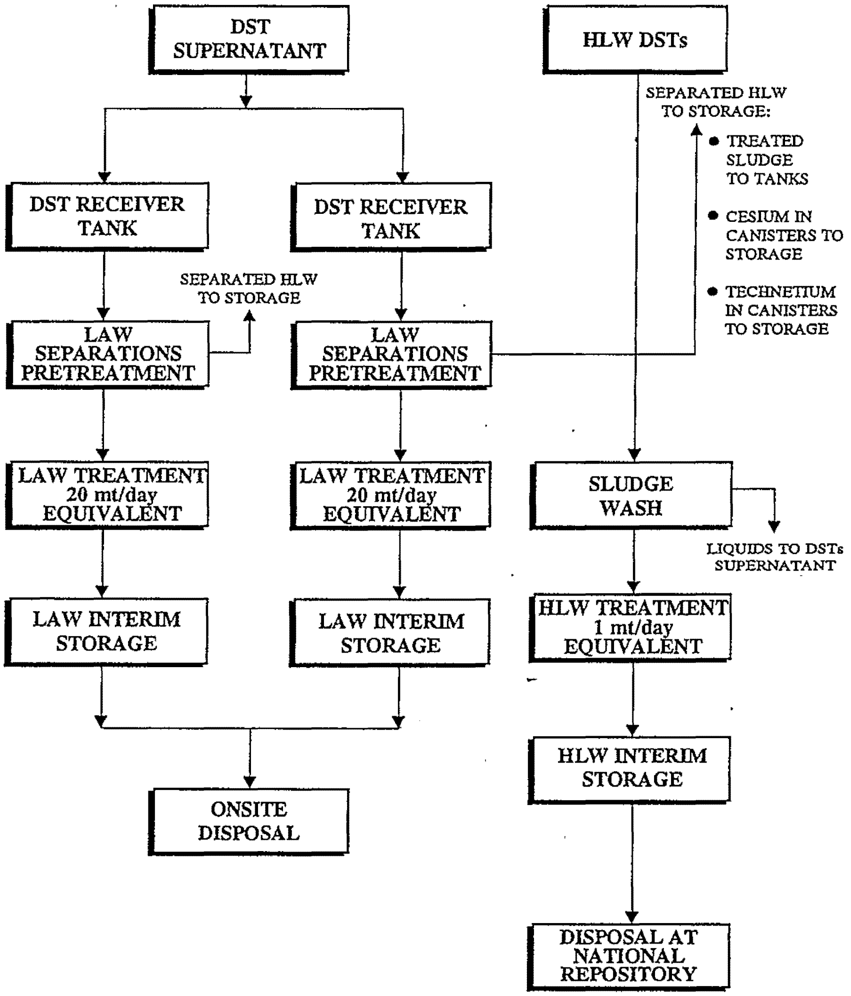
The in situ portion of these alternatives would not meet the RCRA land disposal requirements for hazardous waste or DOE policy to dispose of readily retrievable HLW in a geologic repository. The ex situ portion of these alternatives would meet all regulations for disposal of hazardous, radioactive, or mixed waste assuming that the hazardous waste components would be adequately treated during processing or vitrification.

B.3.9 PHASED IMPLEMENTATION ALTERNATIVE

B.3.9.1 General Description

The Phased Implementation alternative would provide a mechanism to implement tank waste remediation in a two-step process. The first phase would be a proof-of-concept demonstration phase of the separations and immobilization processes for selected tank waste. The first phase would use demonstration-scale treatment facilities. The second phase would involve scaling up or replacing the demonstration-scale processes to treat the remaining tank waste. The Phased Implementation approach could be applied to any of the tank waste alternatives involving waste treatment; however, for purposes of analysis the Ex Situ Intermediate Separations alternative with some additional separations was selected as a representative alternative for analysis (Figure B.3.9.1). The description of the Phased Implementation alternative and the estimates for resources and emissions were developed from the

Figure B.3.9.1 Phased Implementation



Ex Situ Intermediate Separations alternative. This basis included vitrified LAW glass cullet as a LAW form and vitrified borosilcate glass as a HLW form. Other types of glass or waste forms could be selected for HLW or LAW treatment; however, they would have to meet the repository acceptance criteria or performance assessment criteria.

B.3.9.1.1 Phase 1

Under Phase 1, readily retrievable, well-characterized waste from the DSTs (including SST saltwell liquids transferred to DSTs) would be retrieved and processed in two separate demonstration facilities. One of the facilities would process liquid waste to produce an immobilized LAW, while the other facility would produce an immobilized LAW and vitrified HLW. The facility with both LAW and HLW immobilization could be constructed as separate facilities.

Retrieval

Liquid waste retrieval for LAW treatment would be accomplished by using existing waste transfer systems currently installed in the DSTs. The waste identified for HLW processing would be retrieved from selected tanks containing higher concentrations of HLW constituents. The waste identified for HLW processing would be sludge washed to reduce the volume of vitrified HLW. The washed sludges would be transferred directly to the HLW treatment facility for vitrification. The HLW that would be conditioned and retrieved under currently planned demonstrations for retrieval as sludge washing would be used as feed for HLW processing.

Separations

Separations would consist of performing a solid-liquid separation followed by additional chemical processing steps on the liquid stream to remove HLW and TRU constituents to the extent required to meet specifications for the immobilized LAW.

Immobilization

The LAW would be processed using a technology that would meet LAW acceptance specifications. The acceptance specifications would have specific requirements for size, chemical composition limits, isotopic content, and physical parameters. The immobilized LAW waste would be placed into containers for interim storage as future onsite near-surface disposal. For purposes of analysis in this EIS, vitrification was selected as the immobilization process.

The HLW would be processed into a borosilicate glass form that would meet the established waste form acceptance criteria at the potential geologic repository. The HLW would be placed into canisters and overpacked into HMPCs for handling and transport. The HMPCs would be transported to an onsite interim storage facility pending offsite disposal at the potential geologic repository.

Disposal

There would be no disposal component for Phase 1 of the Phased Implementation alterative. The immobilized LAW and HLW would be packaged and stored onsite in interim storage facilities and disposed of during the implementation of Phase 2.

B.3.9.1.2 Phase 2

Following the successful implementation of Phase 1, Phase 2 would be implemented to complete the tank waste remediation. Under Phase 2, the waste remaining in the tanks and MUSTs would be retrieved and processed in new full-scale facilities. The new full-scale facilities would be two 100 mt/day (110 ton/day) LAW facilities and one 10 mt/day (11 ton/day) HLW facility.

Retrieval

Waste retrieval for Phase 2 would involve constructing and operating a full-scale retrieval system that would be capable of retrieving as much waste as practicable (assumed to be 99 percent) from all SSTs, DSTs, and MUSTs. The waste retrieval systems and processes used for Phase 2 would be the same as those described for the Ex Situ Intermediate Separations alternative.

Separations

Separations would consist of the same processes described for the Ex Situ Intermediate Separations alternative, followed by additional chemical processing steps on the LAW stream to remove HLW and TRU constituents to the extent required to meet specifications for the immobilized LAW.

Immobilization

The HLW and LAW immobilization processes used during Phase 2 would be the same processes demonstrated during Phase 1.

Disposal

The disposal of immobilized HLW and LAW would be the same for the Ex Situ Intermediate Separations alternative. The immobilized LAW would be placed into disposal containers at the treatment facility and transported to an onsite near-surface retrievable disposal facility.

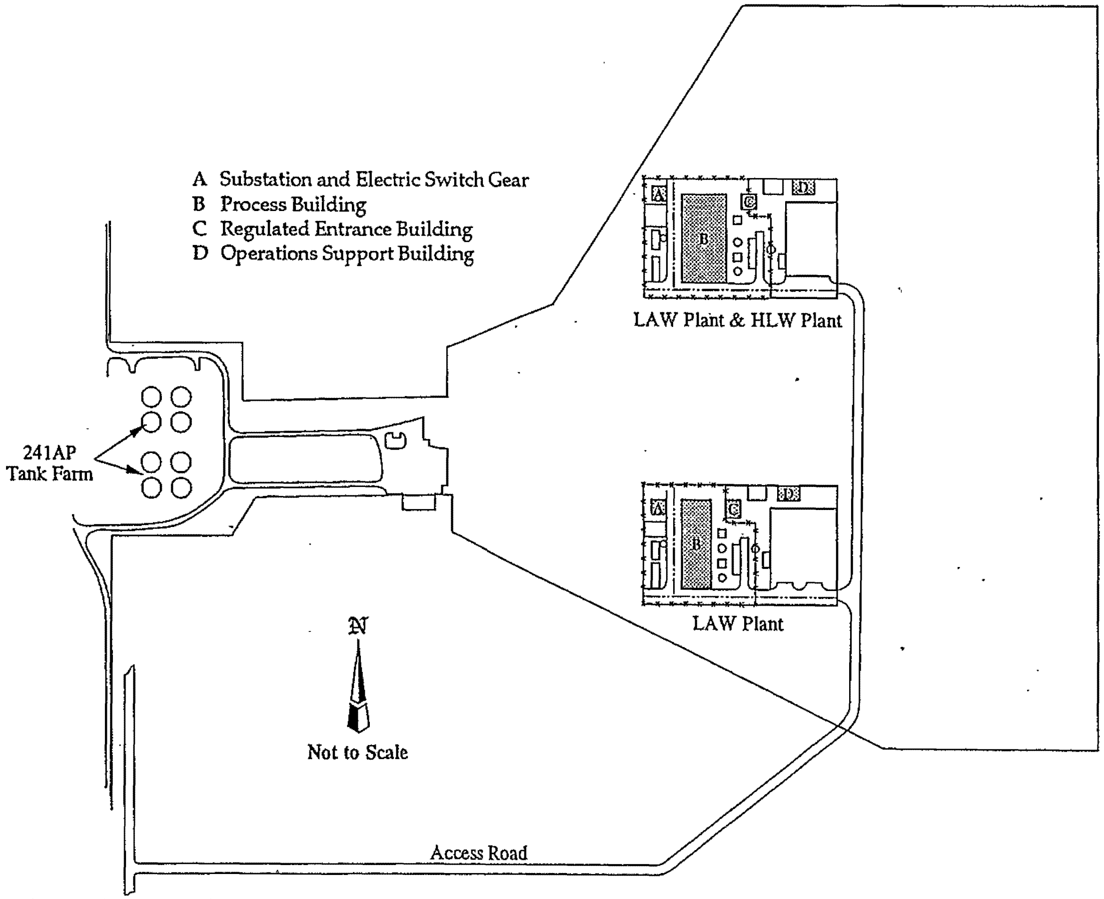
The vitrified HLW would be placed into canisters, packaged into HMPCs, and placed in an aboveground storage facility. The canisters would then be shipped to the potential geologic repository for permanent disposal.

B.3.9.2 Facilities to be Constructed

B.3.9.2.1 Phase 1

This alternative would involve constructing two independent waste treatment facilities. One facility would produce immobilized LAW and the other facility would produce immobilized LAW and vitrified HLW. Each treatment facility would be constructed with support facilities as required to support each operation, as shown in Figure B.3.9.2.

Figure B.3.9.2 Phased Implementation Facility Layout



Necessary pipelines would be constructed from the designated tanks in the 241-AP Tank Farm to the treatment facilities. Additional pipelines would be constructed between the existing waste transfer system and the HLW processing facility. These pipelines would either be buried or constructed on grade inside a shielded pipe run.

The existing Canister Storage Building would be modified to accommodate interim storage of HLW canisters. This would include modifying the underground vaults and ventilation system to accommodate the physical and thermal leaching associated with interim storage of all HLW produced during Phase 1.

The existing grout vaults would be modified to accommodate interim LAW storage of the containerized LAW during Phase 1 operations. This would include modifications to the existing vaults to allow placement and interim storage of the LAW disposal containers pending future retrieval and disposal during Phase 2.

Separations Facilities

Each of the waste treatment facilities would include an integral separations and immobilization facility. The separations facilities would include the processing equipment to filter solids and remove selected radionuclides from the waste stream.

Low-Activity Waste Immobilization

Both of the waste treatment facilities, which would include LAW immobilization facilities, would be sized to produce the equivalent of 20 mt/day (22 tons/day) of vitrified waste at a sodium oxide loading of 15 weight percent. This basis was used to estimate the required facility size and resource requirements.

The facility that only treated LAW would be smaller than the combined LAW plus HLW facility and would have an overall footprint of 40 by 120 m (130 by 390 ft). The facility that treated LAW and HLW would have an overall footprint of 60 by 120 m (200 by 390 ft).

High-Level Waste Immobilization

The HLW immobilization facility would be sized to produce the equivalent of 1 mt/day of vitrified waste at a waste oxide loading of 20 weight percent.

Support Facilities

Each of the processing facilities would require its own support facilities. These facilities would include:

- Cold chemical storage, supply, and makeup;
- Substation and electrical distribution;
- Cooling tower;
- Operations control;

- Regulated entrance building;
- Emergency generator;
- Emergency response center;
- Operations support buildings;
- Process chemical storage; and
- Process water and potable water lines. These would be installed to connect the sites with existing distribution lines in the 200 East Area.

B.3.9.2.2 Phase 2

Construction activities required for Phase 2 would involve constructing all of the facilities identified in the Ex Situ Intermediate Separations alternative, but with reduced scale waste treatment and support facilities (Figure B.3.9.3). Because Phase 1 operations would produce up to 13 percent of the immobilized LAW volume, the size of the treatment facilities required for the Phase 2 would be approximately the same as the ex situ treatment described for the Ex Situ Intermediate Separations alternative. The facilities that would be constructed for Phase 2 operations would include:

- Waste retrieval and transfer facilities as described for Ex Situ Intermediate Separations;
- Two separations and LAW treatment facilities that would be similar to the vitrification facility described for the Ex Situ Intermediate Separations alternative, each with a 100-mt/day (110-ton/day) capacity;
- A 10-mt/day (11-ton/day) HLW vitrification facility that would be similar to the HLW vitrification facility described for the Ex Situ Intermediate Separations alternative;
- Support facilities that would provide utilities, resources, and personnel support to the Phase 2 treatment facilities;
- A LAW disposal facility that would provide for retrievable disposal of LAW produced throughout Phase 1 and Phase 2 (this facility would be the same as the LAW disposal facility described for the Ex Situ Intermediate Separations alternative);
- A HLW interim storage facility for interim storage of the HMPCs; and
- Hanford Barriers over the LAW retrievable disposal facility and tank farms following waste remediation.

B.3.9.3 Description of the Process

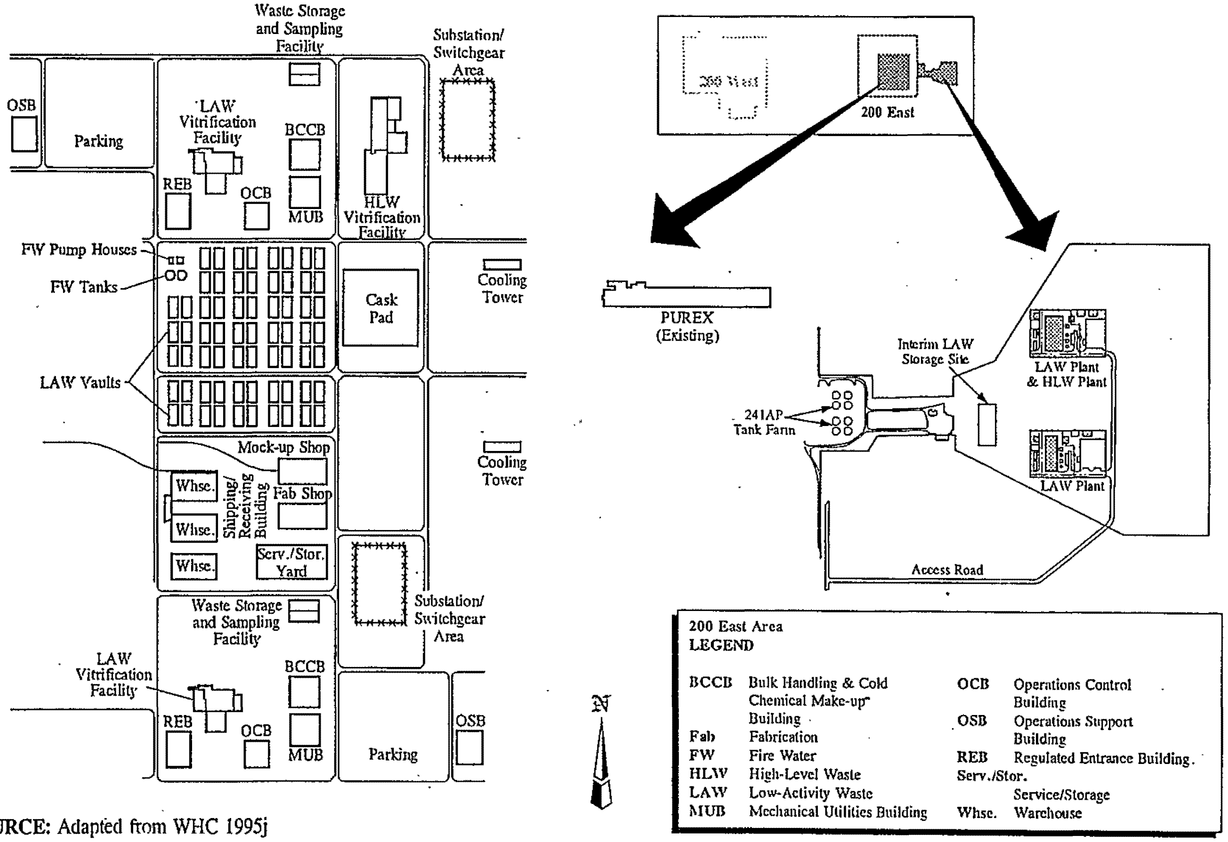
B.3.9.3.1 Phase 1

Overview

The following processes would be included to treat tank waste under Phase 1:

- Retrieve selected waste for LAW treatment;
- Retrieve selected waste for HLW treatment;
- Transfer liquid waste for LAW treatment to a receiver tank;
- Following sludge washing, transfer selected waste for HLW processing directly to the HLW plant;
- Perform separations to remove Cs, Tc, Sr, TRU elements, and sludges from the LAW feed stream;

Figure B.3.9.3 Phased Implementation (Total Alternative) Facility Layout



200 East Area
LEGEND

BCCB	Bulk Handling & Cold Chemical Make-up Building	OCB	Operations Control Building
Fab	Fabrication	OSB	Operations Support Building
FW	Fire Water	REB	Regulated Entrance Building
HLW	High-Level Waste	Serv./Stor.	Service/Storage
LAW	Low-Activity Waste	Whse.	Warehouse
MUB	Mechanical Utilities Building		

SOURCE: Adapted from WHC 1995j

- Store the separated Cs and Tc produced during separations at the treatment facilities, or package and transport the separated Cs and Tc to onsite interim storage for future waste treatment;
- Return sludges containing Sr and TRU waste separated during LAW treatment to the DSTs for storage;
- Vitrify both the LAW and HLW;
- Place the vitrified HLW into canisters;
- Place the vitrified LAW into containers; and
- Transport the immobilized waste to onsite interim storage facilities.

Each waste treatment facility would be designed, built, and operated separately. It is assumed that the technologies selected for the separations and immobilization processes would produce a waste form that meets DOE specifications. Therefore, the process description for the Phased Implementation alternative has been developed using the Ex Situ Intermediate Separations alternative, with additional separations processing as a basis. This approach provides for analyzing the alternative using representative technologies.

Tank Waste Retrieval and Transfer

The first step in waste processing would be to recover and transfer waste to be treated at LAW facilities from the tanks to the DST feed tanks. The waste feed to the LAW facilities would be retrieved and transferred in batches from selected DSTs into two existing DSTs designated as feed tanks. Each LAW facility would have one designated DST as a feed tank. The waste feed stream for LAW treatment would be primarily DST liquid waste but could include SST saltwell liquids or SST waste recovered during retrieval demonstrations. The waste feed to the HLW plant would be retrieved and transferred separately. The selected waste for HLW treatment would be sludge washed and the washed solids would be routed directly to the HLW processing facility. The waste treated at the HLW facility would be HLW recovered directly from selected tanks and may or may not include the HLW that would be separated at the LAW treatment facilities.

Liquid waste retrieval and transfer would use equipment and systems currently in place in the DST farms. Sludge washing and slurry pumping using techniques identified for the Ex Situ Intermediate Separations alternative would be used to retrieve waste for treatment at the HLW facility.

Separations

For purposes of analysis, the separations processes described for the Ex Situ Intermediate Separations and Ex Situ Extensive Separations alternatives were used.

The specific technologies used for separations and immobilization have not been defined, and therefore are not specifically identified or discussed for this alternative. The separations and immobilization technologies used for waste immobilization would be controlled by waste product specifications, which would control the physical properties, chemistry, radionuclide content , and volume of the immobilized LAW and HLW.

Separations prior to LAW immobilization would be performed to remove the Cs, Tc, Sr, TRU elements, and entrained sludge particles from the waste stream. The separated Cs and Tc radionuclides would either be stored at the treatment facilities or packaged in canisters for onsite dry storage, the treated sludges along with the Sr and TRU elements would be returned to the DSTs for storage, and the treated liquid waste stream would then be immobilized.

Immobilization

The LAW waste stream would be immobilized using a technology to treat the waste that would yield a stabilized waste product similar to vitrified glass with regard to waste performance characteristics. Vitrification was assumed for purposes of evaluation. The immobilized LAW would be placed into canisters approximately 1.8 m long by 1.2 m wide by 1.2 m high (6 ft long by 4 ft wide by 4 ft high).

The immobilized LAW would be sealed in steel containers at the Phase 1 treatment facilities for interim storage and eventual onsite disposal. The sealed LAW containers would be transported to the four existing grout vaults nearby for temporary storage until the Phase 2 LAW onsite disposal vaults are ready to receive the containerized LAW material. The Phase 1 immobilized LAW would be transported to this new disposal vault site to be entombed with the Phase 2 LAW waste.

The DST waste would be retrieved and transferred to the receiver tanks or in the case of the HLW, directly to the HLW processing facility. Waste from the receiver tanks would be transferred to the treatment facilities on an as-needed basis. The HLW would be vitrified into borosilicate glass.
The HLW plant would be designed to produce the equivalent of 1 mt/day of HLW glass at a 20 weight percent waste oxide loading. The vitrified HLW would be placed directly into canisters. The HLW canisters (0.61 m [2 ft] in diameter by 4.57 m [15 ft] long) would be placed in transportation casks and transported to the canister storage building for interim storage. The canisters would be removed from the transportation casks and placed into storage tubes at one of the canister storage building vaults.

Each of the waste treatment facilities would operate off-gas treatment systems that would include control technologies for priority pollutants and radionuclides. The treatment of the off-gas would be similar to the processes and equipment as described for the Ex Situ Intermediate Separations alternative.

B.3.9.3.2 Phase 2

Overview

The tank waste treatment process for Phase 2 would include 1) retrieving the waste from tanks; 2) separating the LAW from the HLW; 3) immobilizing the LAW stream; 4) vitrifying the HLW stream; 5) disposing of the LAW onsite; 6) temporarily storing the HLW; and 7) transporting the HLW to the potential geologic repository at a future date. The processes used for waste treatment during Phase 2 would be the same processes demonstrated during Phase 1 operations.

Tank Waste Retrieval and Transfer

The process used for waste retrieval and transfer during Phase 2 would be the same as the process described for retrieval under the Ex Situ Intermediate Separations alternative. Waste retrieval during Phase 1 mainly would consist of removing liquid waste from DSTs. These DSTs would require additional waste retrieval during Phase 2 to remove sludges and meet requirements for waste residuals.

Separations

Separations processes used during Phase 2 would be the same processes that were developed and demonstrated during Phase 1. The LAW remaining following the separations process would contain approximately 17 MCi of radioactivity, including 10 MCi of Cs and Ba, 6.8 MCi of Sr and Y, 2.59E-02 MCi of Tc-99, and a total of 1.22E-02 MCi of TRU isotopes.

Immobilization

Immobilization of the HLW and LAW streams during Phase 2 would use the same processes that were developed and demonstrated during Phase 1. The HLW treatment during Phase 2 would also include vitrifying the Cs and Tc waste that was separated to produce the LAW during Phase 1 operations. The operation of the Phase 1 treatment processes would allow for optimizing the processes used during waste treatment at the new Phase 2 facilities.

Post Remediation

The post-remediation process for this alternative would be the same as that described for the Ex Situ Intermediate Separations alternative. When tank waste processing has been completed, the processing facilities would be decontaminated and decommissioned in the following manner:

- Processing equipment would be decontaminated sufficiently to allow onsite disposal in a low-level waste burial ground.
- Processing facilities would be decontaminated to the extent possible and then entombed in place. The exact materials that would be used to cover processing facilities have not been defined.

B.3.9.4 Implementability

Because the Phased Implementation alternative is only a demonstration-scale facility, many of the implementability issues surrounding the ex situ alternatives are reduced in complexity. Issues relating to implementing this alternative can be grouped into the following categories:

- Capability to produce immobilized waste within the waste form specifications developed; and
- Successful operation of the Phased Implementation alternative (Phase 1) is critical to the follow-on implementation of Phase 2 (the completion of retrieval treatment and disposal activities).

Phase 1 shares some of the same implementability issues as the Ex Situ Intermediate Separations alternative and the Ex Situ Extensive Separations alternative because several of the separations and treatment processes that would be used during Phase 1 were assumed to be similar to the processes described for those alternatives. Performance of key processes has been assumed in the absence of substantive data. Cost estimates have a high degree of uncertainty because some of the processes are unproven.

The phased implementation approach provides the opportunity for significantly improving the process design and facility configuration for Phase 2. Lessons learned and processing experience gained during Phase 1 would be applied to the construction and operation of Phase 2 facilities. This approach would allow for increased operating efficiency during Phase 2.

During Phase 2, the waste would be retrieved from the tanks using the same processes as the other ex situ alternatives, and thus Phase 2 shares the same implementability issues regarding retrieval as the other ex situ alternatives.

This alternative would meet all applicable regulations for disposal of hazardous, radioactive, or mixed waste assuming that the hazardous waste components are adequately treated during waste processing or vitrification.

B.4.0 CAPSULES

The following sections describe each of the capsule alternatives. The capsules are currently defined as waste by-product, which means they are available for productive uses if uses can be found. If and when the capsules are determined to have no potential productive uses, it is assumed they would be subject to management and disposal as HLW under TWRS. The discussion includes a general description of the alternative followed by a description of the construction activities that would be included if the alternative were implemented. The discussion continues with a description of the process/operation and ends with a discussion of key issues associated with implementing the alternative. Engineering data for each alternative can be found in Section B.11.0.

B.4.1 NO ACTION ALTERNATIVE (CAPSULES)

B.4.1.1 General Description

The No Action alternative for the capsules would consist of continued safe management. Currently, the capsules are stored in water basins in WESF. Additional capsules are being returned to the Hanford Site and would be stored in the water basins. The capsules and basins would be maintained and administrative controls would prevent inadvertent human intrusion. WESF is scheduled to be decontaminated and decommissioned within the next 10 years, and administrative controls would be assumed to be effective until an alternative waste storage facility could be constructed. If this alternative is selected, within the next 10 years DOE and Ecology would need to decide on a strategy for continued storage elsewhere or select a disposal alternative for the capsule contents. This will be considered in the Cs and Sr capsule management plan.

Monitoring and maintenance activities for the capsules involve calculating the annual inventory, physically verifying that the inner capsule can still move independently of the outer capsule (Cs capsules only), and using online radiation monitors to detect pool cell water contamination. The annual inventory provides the exact storage location and accountability for all of the Cs and Sr capsules stored at WESF.

The Cs capsules are clunk tested on a quarterly basis. This involves physically grasping one end of a capsule with a pool tong and rapidly moving the capsule vertically approximately 15 cm (6 in.). This allows the inner capsule to slide within the outer capsule, making a clunk sound that is easily heard and felt by the operator performing the test. This test verifies that the capsule has not bulged. A capsule that failed the clunk test would be removed from the storage basins and placed into a hot cell for additional evaluation.

Leak detection in the storage basin would be performed by online beta monitors that would be set to alarm when the activity present in the pool water exceeded a set level.

Maintenance of the storage facility includes maintaining the electrical and mechanical systems required to safely operate the facility. This would include life extension or replacement for failed or aging equipment.

This alternative would meet all applicable regulations (Volume One, Section 6.2).

B.4.2 ONSITE DISPOSAL ALTERNATIVE

B.4.2.1 General Description

This alternative would consist of packaging the capsules into sealed canisters and placing them in a newly constructed subsurface disposal facility in the 200 Area. This alternative would be similar to the in-place stabilization and disposal alternative addressed in the Hanford Defense Waste EIS for Cs and Sr capsules (DOE 1987).

The Cs and Sr capsules would remain in storage in a series of water-filled storage pools at WESF until the modified capsule packaging facility was completed. They would be retrieved from the storage pools and inspected for surface contamination, corrosion, structural defects, and heat content before placing them in a capsule vault. The capsules would be stored in the vault until they would be transferred to the canister-packaging facility, also a part of WESF.

At the WESF canister-packaging facility, the capsules would be placed in a seal-welded canister, which would be placed in drywells for onsite disposal. Two to four capsules would be placed in a canister depending on heat load. The sealed canister package would be leak tested, ultrasonically scanned, checked for surface contamination, and decontaminated before being transported to the subsurface disposal facility. A shielded transporter would place the canister in the drywell.

For this alternative, it was assumed that the capsules would remain in dry-storage with administrative controls in effect (WHC 1995h). For the purpose of calculating the potential impacts, it is assumed that the controls would be terminated after 100 years.

B.4.2.2 Facilities to be Constructed

The capsule packaging operation would be performed in the existing WESF Building located in the 200 East Area, next to B Plant. Figure B.4.2.1 provides a plant layout diagram of WESF. The approximate dimension of the WESF Building is 90 by 120 m (300 by 400 ft).

Modifying existing hot cells and/or constructing new hot cells would provide the capabilities required for the capsule-packaging operation. There are currently eight hot cells: A, A Cell Hood, B, C, D, E, F, and G. Each cell has a viewing window and ports for two manipulators, except for G Cell, which has two viewing windows (each window has two ports for manipulators). Three additional hot cells would be constructed for the capsule-packaging facility for inspection, weld stations, weld integrity tests, contamination checking, and decontamination. In addition, facilities would be modified and/or constructed for capsule disposal vaults, canister storage and testing, and canister packaging operations.

A drywell disposal facility would also be constructed. The ground surface of the storage area would be graded flat and nearly level with only enough slope to provide for surface drainage. A total of 672 drywells (584 canisters plus 15 percent contingency) would be drilled to a depth of 4.6 m (15 ft). They would be arranged in a grid pattern (5 m [16.4 ft] center-to-center) occupying a surface area of 3.8E+04 m^2 (195 by 195 m) (640 by 640 ft) with a 30-m (100-ft) buffer. The site selected for the drywell disposal facility is near the western boundary of the 200 East Area. Figure B.4.2.2 illustrates the drywell disposal facility casing assembly, and Figure B.4.2.3 is a representation of a drywell disposal array.

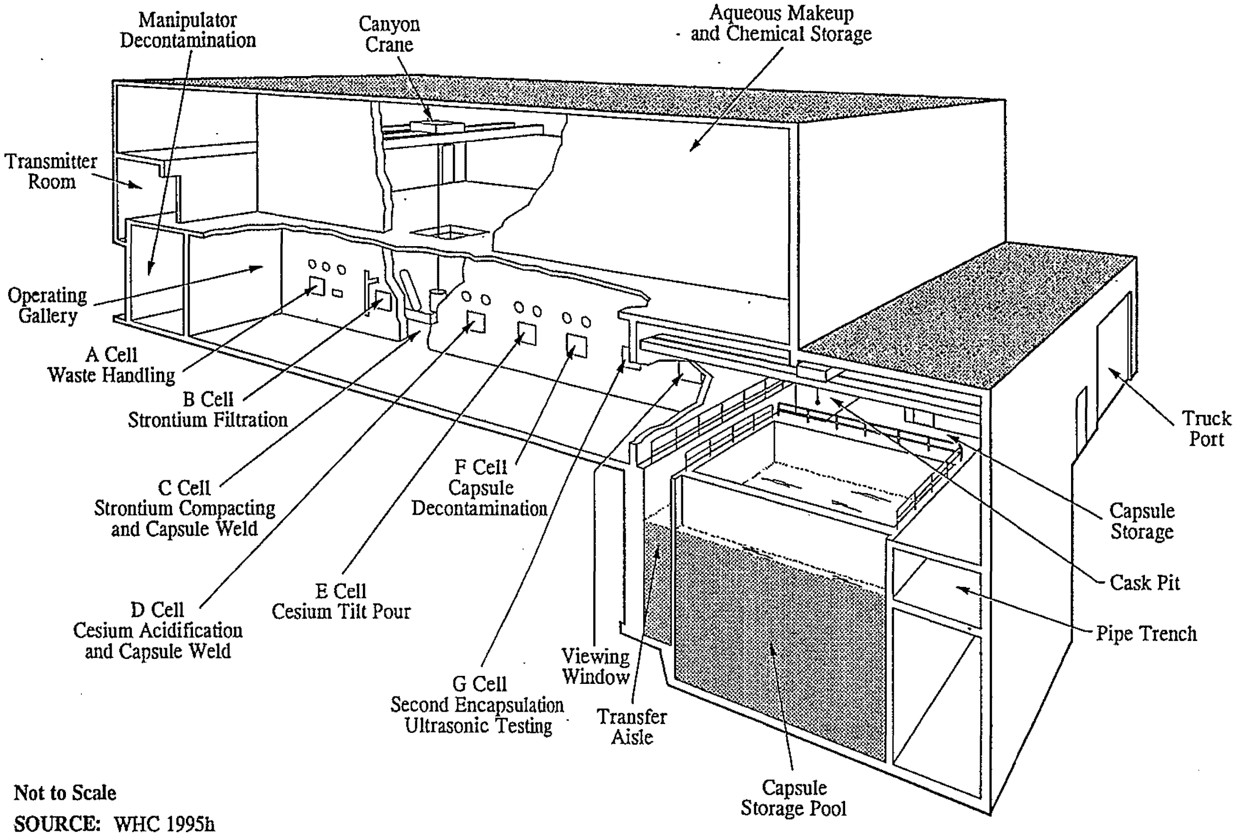
B.4.2.3 Process Description

The process activities for the Onsite Disposal alternative are divided into four major operations. A process flowsheet is provided in Figure B.4.2.4.

Waste Encapsulation and Storage Facility

The Cs and Sr capsules would be stored in the water-filled storage pools at WESF until a capsule-packaging facility is completed. When the capsule-packaging facility is completed, the capsules would be remotely removed from the pool and placed in an inspection cell where they would be checked for surface contamination, corrosion, structural defects, and heat content before being moved to the capsule-packaging facility. Capsules that fail inspection would undergo decontamination, rework, and testing until the capsules meet the requirements for the canister-packaging operation. After passing inspection they would be stored in the capsule vault until being transferred to the canister-packaging operation.

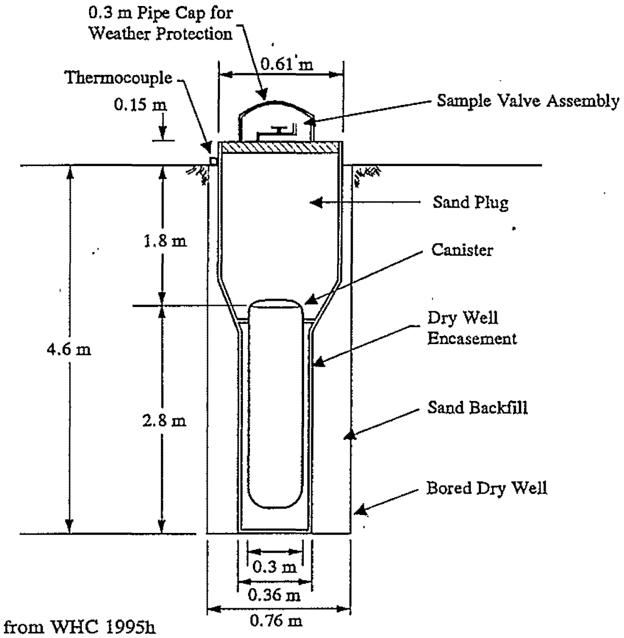
Figure B.4.2.1 Waste Encapsulation and Storage Facility



Not to Scale

SOURCE: WHC 1995h

Figure B.4.2.2 Capsule Dry-Well Disposal Assembly



SOURCE: Adapted from WHC 1995h

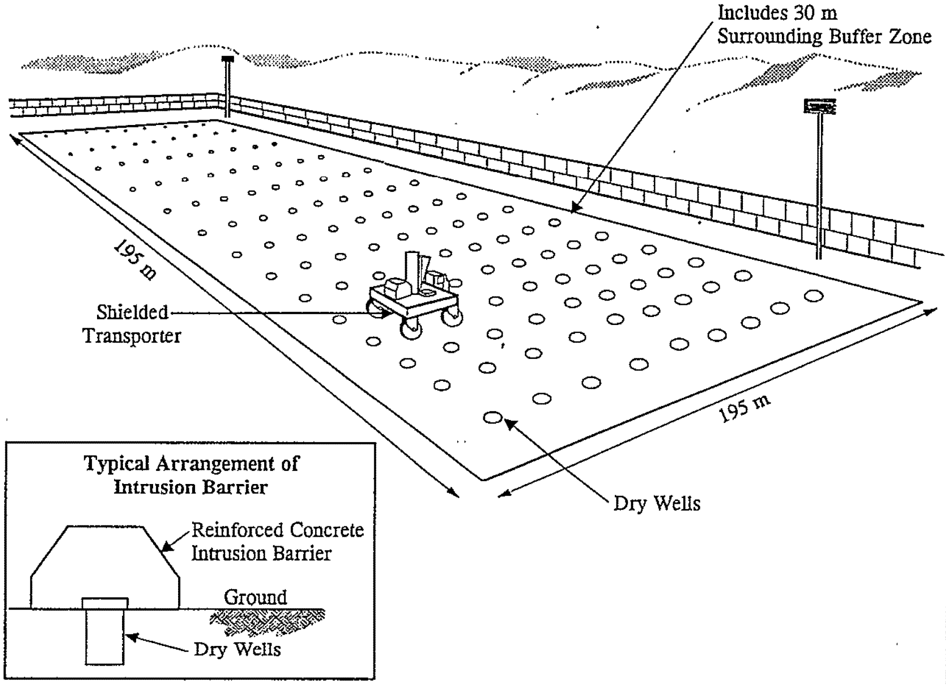
Capsule Packaging Facility

The capsules would be remotely removed from the vaults and then would be placed in racks and inserted into canisters. The loaded canister would then be remotely moved to a weld station where the lid would be welded in place. The canister would then undergo leak testing, ultrasonic scanning, and examination for surface contamination.

The canisters (3 m [10 ft] long) used for Onsite Disposal would be smaller than the canisters (4.5 m [15 ft] long) used for packaging and shipping to an offsite potential geologic repository. The canisters used for Onsite Disposal would be 0.3 m (1 ft) in diameter and 3 m (10 ft) long.

The allowable heat load for Onsite Disposal would be smaller than the allowable heat load for disposal at a potential geologic repository. The drywell heat load limit would be 0.55 kW per canister, which is estimated to be one to four capsules per canister (WHC 1995h). The canisters are expected to contain about three Sr capsules or four Cs capsules. Table B.4.2.1 summarizes the estimated capsules and canisters required for onsite disposal.

Figure B.4.2.3 Capsules Onsite Disposal Arrangement (Conceptual)



Not to Scale

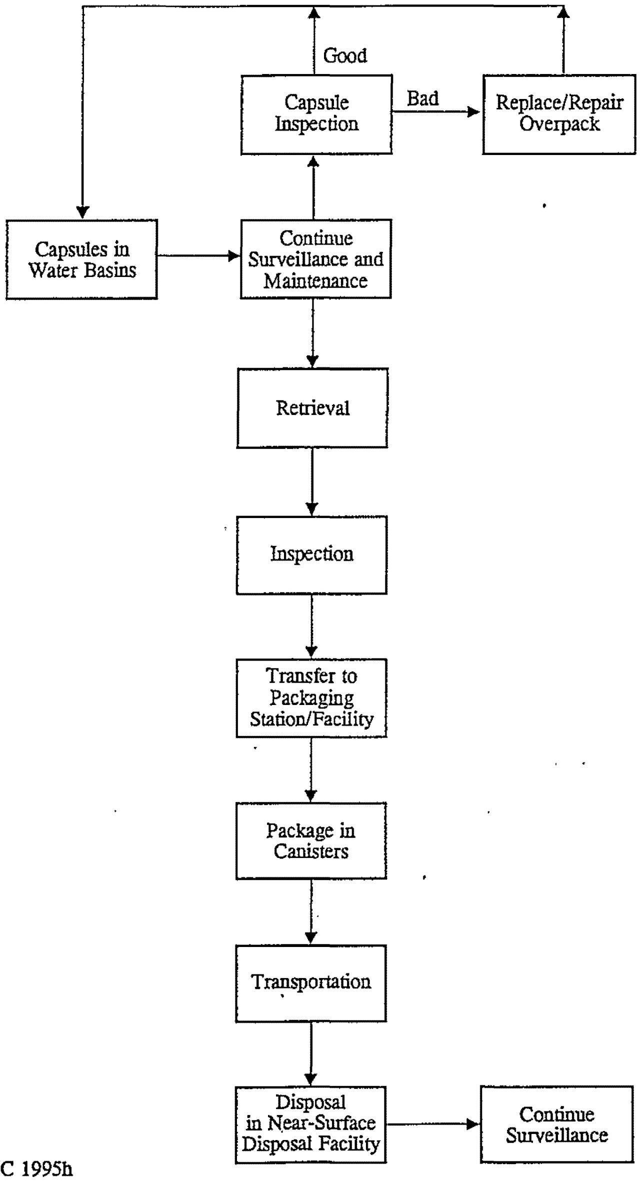
Disposal

After placing the capsules into canisters, the canisters would be transported by a shielded vehicle for placement in near-surface drywells to provide long-term, passively-cooled storage. There would be one canister per drywell. After placing the sealed canisters into the drywells an intrusion prevention barrier would be placed over each drywell.

Monitoring and Maintenance

All of the canisters in the drywell disposal facility would be closely monitored for radiological and nonradiological emissions. All associated equipment, instrumentation, and controls would be maintained. Continuous security and monitoring and maintenance operations would be performed for a period of 100 years, at which time institutional control would cease.

Figure B.4.2.4 Onsite Disposal Alternative - Process Flow Diagram



SOURCE: WHC 1995h

Table B.4.2.1 Estimated Capsules, Sealed Canisters, and Multi-Purpose Canisters

Waste Type	Overpack and Ship Alternative					
	Number of Capsules	Heat Load (kW) (2010)	Average kW per Capsule (2010)	Average Number of Capsules per Canister [2]	Number of Canisters [1]	Number of HMPCs to Repository [3]
Strontium Fluoride	601	107.9	0.18	5 to 9	67 to 121	17 to 31
Cesium Chloride	1,328	181.3	0.13	5 to 9	148 to 266	37 to 67
Total	1,929	289.2	N/A [5]	N/A	215 to 387	54 to 98
	Onsite Disposal Alternative					
Strontium Fluoride Capsules	601	107.9	0.18	2.8	217	N/A
Cesium Chloride Capsules	1,328	181.3	0.13	3.8 [4]	367	N/A
Total	1,929	289.2	N/A [5]	N/A [5]	584	N/A

Notes:
[1] Estimates are based on heat loads only.
[2] The higher estimates are based on conservative thermal limits of 1.17 kW/canister (strontium fluoride) and 0.8 kW/canister (cesium chloride) for the potential geologic repository heat load limits. A half-life of 30 years is assumed for cesium and strontium. Impacts were assessed based on five capsules per canister.
[3] Current assumption is that each HMPC will hold no more than four packaged and sealed canisters of either height.
[4] Numbers are based on drywell storage facility heat load of 0.5 kW/canister. The canister for this option is 3 m (10 ft) long and 0.3 m (1 ft) in diameter.
[5] Averages have not been calculated for the capsule total.
HMPC = Hanford Multi-Purpose Canister
kW = kilowatt

B.4.2.4 Implementability

Implementing the alternative would involve mechanical handling of the capsules and canisters and thus presents no new technology uncertainties that would require extensive research and development. One issue that would require evaluation would be the corrosion of the drywell casing and the performance of the disposal configuration.

This alternative would not meet the land disposal requirements of RCRA for hazardous waste. Near-surface disposal of HLW may not meet DOE Order 5820.2A requirements for disposal of readily retrievable HLW in a potential geologic repository (Volume One, Section 6.2).

B.4.3 OVERPACK AND SHIP ALTERNATIVE

B.4.3.1 General Description

For this alternative, the capsules would continue to be stored in a series of water-filled storage pools at WESF until a modified WESF capsule-packaging facility is completed. The capsules would be retrieved from the water-filled storage pools, inspected for surface contamination, corrosion, structural defects, and heat content, and temporarily placed in a capsule vault. The capsules would be stored in the capsule vault until they could be packaged into sealed canisters in the canister-packaging operations.

At the capsule-packaging facility, the sealed canisters would be packaged into HMPCs and placed in the onsite HLW interim storage facility. Monitoring and maintenance would be performed at the onsite interim storage facility while HMPCs are in temporary storage (WHC 1995h).

B.4.3.2 Facilities to be Constructed

The capsule-packaging operation would be performed in the existing WESF Building, whose location and size are described in the Onsite Disposal alternative (Section B.4.2.). While the building modifications would be almost identical, areas for overpacking the canisters into HMPCs would be constructed only for this alternative. Temporary storage for the HMPCs loaded with canisters would be on an engineered storage pad either in place of or near the interim storage of vitrified HLW with the approximate dimensions of 130 by 150 m (430 by 500 ft). The pad would have a stormwater collection and monitoring system, which would provide for collecting and decontaminating spills.

B.4.3.3 Process Description

The process activities for this are divided into five major operations. The process flowsheet for this alternative is provided in Figure B.4.3.1. Final design of the canister packaging would include design criteria for waste acceptance at the potential geologic repository.

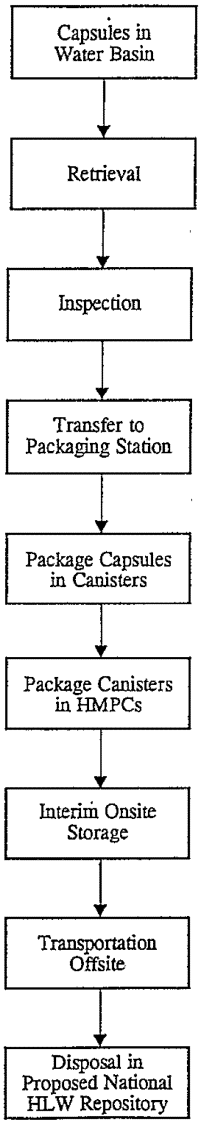
Waste Encapsulation and Storage Facility

As in the Onsite Disposal alternative, the Cs and Sr capsules would be stored in the water-filled storage pools in WESF until a capsule-packaging facility is completed. When the facility is in operation, the capsules would be remotely removed from the pools and placed in an inspection cell where they would be checked for surface contamination, corrosion, structural defects, and heat content before transferring them to the capsule-packaging facility. Capsules that fail the inspection would undergo decontamination, rework, and testing until the capsules meet the requirements for the canister-packaging operation. After passing inspection they would be stored in the capsule vault until they could be transferred to the canister-packaging operation. The capsule vault is a shielded storage room that is used for storing the inspected capsules prior to loading into canisters, which are 4.57 m (15 ft) long.

Capsule Packaging Facility

In this operation, the capsules that were stored in the vaults would be transported to the capsule-packaging area, and placed in racks that would be loaded into canisters. Depending on the heat emitted by each canister, five to nine capsules would be loaded in one canister. After loading, the canisters would be moved to a weld station where the lid would be welded in place.

Figure B.4.3.1 Overpack and Ship Alternative - Process Flow Diagram



SOURCE: WHC 1995h

The seal-welded canisters would undergo leak testing, ultrasonic scanning, and checking for surface contamination. If the canister is found to be contaminated, it would go to electropolishing decontamination before overpacking in HMPCs. The HMPC can hold a maximum of four canisters.

The canisters used for packaging and shipping the capsules to an offsite potential geologic repository would be larger than the canisters used for drywell storage. The canisters used for packaging the capsules under this alternative would be 0.61 m (2 ft) in diameter and 4.57 m (15 ft) long. The allowable heat load for onsite disposal is 1.17 kW per canister for Cs and 0.80 kW per canister for Sr. Each canister would be expected to hold five to nine Cs or Sr capsules. Table B.4.2.1 summarizes the number of capsules, canisters, and HMPCs required to implement this alternative.

Onsite Interim Storage

The loaded HMPCs would be stored ready for transport on a separate engineered pad with dimensions of 130 by 150 m (430 by 500 ft) until the potential geologic repository is available. Loading of the capsules into the canisters and loading of the canisters into the HMPCs would be expected to be accomplished near the scheduled time for transport to the repository so that loaded HMPC interim storage would be minimized.

Monitoring and Maintenance

All the HMPCs would be closely monitored for radiological and nonradiological emissions. All associated equipment, instrumentation, and controls would also be maintained. Continuous monitoring and maintenance would be performed at the onsite interim storage facility until the HMPCs would be transported to the potential geologic repository.

Transport to the Potential Geologic Repository

When the potential geologic repository is ready to accept processed HLW the HMPCs would be removed from the onsite interim storage facility and transported by railcar to the repository.

B.4.3.4 Implementability

Implementability of this alternative could be affected by the acceptability of the packaged capsules at the potential geologic repository. The acceptability issue involves the waste form. The solubility of this waste may ultimately exceed the HLW acceptance criteria. The Cs and Sr salts would not be immobilized under this alternative but instead would be packaged to provide two additional barriers for containing the capsules. If it is determined that the salt form of these waste would not meet the Waste Acceptance Criteria, the capsule contents would have to be removed and processed appropriately to meet the Waste Acceptance Criteria. Further evaluation would be required to resolve technical and programmatic concerns associated with disposal of the Cs and Sr capsules in the potential geologic repository.

This alternative may not meet the land disposal restrictions of RCRA because of the characteristic corrosivity of the CsCl and SrF_2. Assuming the waste is mixed waste, it would not meet the DOE restriction against disposal of mixed waste in the first potential geologic repository.
Also, the powder waste form of the SrF_2 may not meet the waste acceptance requirement to immobilize particulate waste (Volume One, Section 6.2).

B.4.4 VITRIFY WITH TANK WASTE ALTERNATIVE

B.4.4.1 General Description

This alternative would consist of continued storage of capsules in water-filled storage pools inside WESF until the HLW vitrification facility is completed. Then the capsules would be retrieved from the storage pools and transferred to the HLW vitrification facility, which would include equipment to chemically process, if necessary, and blend the Cs and Sr with the tank waste feed to the HLW vitrification process. The remainder of the process would be similar to the process described for vitrifying HLW under the Ex Situ Intermediate Separations alternative (Section B.3.5).

As part of the HLW glass, the Cs and Sr would be monitored in temporary storage and transported by railcar to the potential geologic repository.

B.4.4.2 Facilities to be Constructed

The dismantling of Cs and Sr capsules and the processing of Cs and Sr salt would be integrated with the HLW vitrification facility. For this alternative, the Cs and Sr capsules dismantling facility would be built as part of the HLW vitrification facility.

The capsule processing facility would include hot cells to open the double-walled capsules, mixing and storage tanks for CsCl, a pulverizer and slurry tank for SrF_2, chemical processing facilities if required, pumps for blending Cs or Sr compounds with HLW slurry prior to vitrification, and decontamination facilities for the empty capsules.

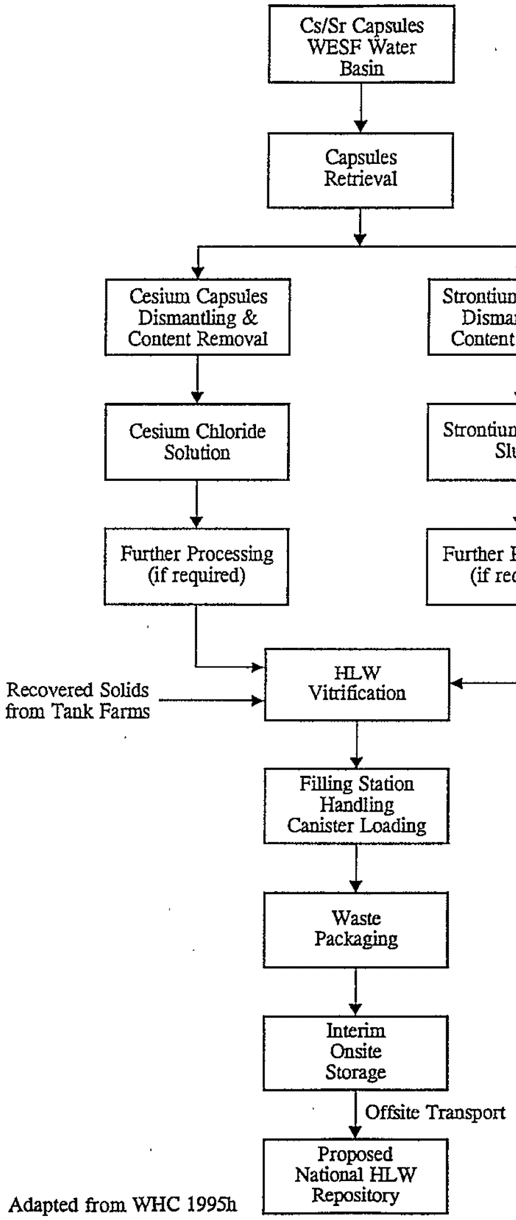
B.4.4.3 Process Description

The process activities for the extensive immobilization option are divided into four major operations, as shown on the flowsheet in Figure B.4.4.1.

Capsules Retrieval From Waste Encapsulation and Storage Facility Storage Pool

The Cs and Sr capsules would be stored in water-filled storage pools at WESF until the HLW vitrification facility is completed and ready for operation. The capsules would then be remotely retrieved, loaded in casks, and transported by truck to the capsule-dismantling hot cells that would be part of HLW vitrification facility.

Figure B.4.4.1 Vitrify with Tank Waste Alternative - Process Flow Diagram



SOURCE: Adapted from WHC 1995h

Dismantling and Removal of Capsules Content

At the dismantling facility, the outer and inner walls of the capsules would be remotely cut open to remove the CsCl and SrF_2 salts, and the empty Cs and Sr capsules would be decontaminated and disposed of with other low-level metallic waste.

Blending Cesium Chloride or Strontium Fluoride with HLW Slurry

The CsCl would be dissolved in water, blended with the HLW slurry from the tank farms, and used as feed to the vitrification facility. The SrF_2 would be pulverized and then water would be added to make a slurry with a solids content of less then 4 volume percent. The SrF_2 slurry would then be mixed with the HLW slurry and used as feed to the vitrification facility. An alternative treatment for the halides would be to convert them to nitrates prior to vitrification if the halide salts cannot be directly fed to vitrification.

Chemical Processing of Capsule Contents

This processing converts the halides to nitrates. The dissolved CsCl would be processed through ion exchange columns where the chloride ion would exchange for a nitrate ion, resulting in a cesium nitrate solution. Two ion exchange columns would be used to allow alternate processing and regeneration cycles. Regeneration would be with 1 molar nitric acid.

The pulverized SrF_2 would be dissolved in sulfuric acid to produce a precipitated strontium sulfate and gaseous hydrofluoric acid that would be sent to the off-gas processing facility. The strontium sulfate would be reacted with sodium carbonate to form strontium carbonate. The last processing step would be to react the strontium carbonate with nitric acid to form a solution of strontium nitrate.

High-Level Waste Vitrification

The Cs and Sr salts would be blended with the tank waste and fed to the HLW melter feed section. The HLW would be stored onsite until the potential geologic repository is ready to accept HLW. When the potential geologic repository is ready to accept processed HLW, the Cs and Sr (as part of the HLW glass) would be transferred to the repository.

B.4.4.4 Implementability

This alternative could only be implemented if one of the tank waste ex situ alternatives or the Ex Situ/In Situ Combination alternative were selected. Chemical processing could be required to remove the chloride and fluoride from the Cs and Sr salts so that they meet the feed specifications that would be developed for the HLW vitrification feed stream. Further study would be required to determine if the capsule contents could be successfully treated as part of the calcination feed stream. Regenerating the Cs ion exchange media produces hydrochloric acid. Neutralizing the hydrochloric acid may produce a secondary waste product requiring further treatment and disposal. The production of hydrofluoric acid during strontium processing would require additional off-gas processing and would produce magnesium fluoride, which would require disposal as a secondary waste.

This alternative would meet all applicable regulations for disposal of hazardous, radioactive, or mixed waste assuming that the hazardous waste components are adequately treated during waste processing or vitrification.

B.5.0 TANK CLOSURE

This section describes the representative tank closure process that has been included in the alternatives to allow an equitable comparison of alternatives. Closure is a term that refers to the final disposition of the tanks and associated piping, any residual waste that remains in the tanks following remediation, equipment that may be left in the tanks, and any soil or groundwater contamination associated with the tank farm operations.

Under the Tri-Party Agreement, both SSTs and DSTs are RCRA hazardous waste management units that will be eventually closed under State Dangerous Waste regulations (WAC 173-303). Three options exist for this closure: 1) clean closure, involving removal of all waste and waste constituents, including tank, debris, contaminated equipment, and contaminated soil and groundwater; 2) modified closure, which involves a variety of closure methods but requires periodic (at least once after 5 years) assessments to determine if modified closure requirements are being met; and 3) closure as a landfill with waste remaining in-place and corrective action taken for contaminated media under post-closure requirements. All three options require the submittal and approval of closure plans by Ecology. There is currently insufficient information available to make a decision on how to close the tanks, so closure is not within the scope of this EIS. However, decisions (such as the percent of waste recovery) on how to treat and dispose of the tank waste may impact the level of closure activities in the future.

To provide information on how closure activities would be affected by remediating the tank waste, a representative approach to tank closure (closure as a landfill) has been included in each of the TWRS alternatives to allow an equitable comparison of the alternatives. This is described in the following text.

Closure would address 149 SSTs, 28 DSTs, and approximately 60 MUSTs and includes the other ancillary equipment associated with waste tank activities. Closure would apply as follows.

- Both the SSTs and DSTs would be stabilized to prevent dome collapse by gravel filling for all ex situ vitrification alternatives and the In Situ Fill and Cap alternative. The gravel-fill process would involve the uniform distribution of sized, crushed rock throughout the tank including the tank dome, using a gravel slinger. This commercially-proven technology is used in filling ship holds and silos with materials such as grain or cement. Tests performed at the Hanford Site have verified the use of this technology with local materials in a tank-like environment.
- Ancillary equipment and MUSTs would be grout filled for stabilization in all treatment alternatives with the exception of the No Action and Long-Term Management alternatives. Ancillary equipment and MUSTs would not be excavated or packaged. Ancillary equipment would include diversion boxes, catch tanks, valve and pump pits,

process pits, diverter stations, receiver vaults, condensate tanks, risers, transfer piping, and piping encasements associated with SST operations. Pipelines would include lines between tanks and process facilities, air and steam supply lines, raw water lines, and drains.

- During tank farm closure, ancillary equipment items and MUSTs would be stabilized in place with an appropriate grout material. The physical immobilization of contaminants provided by grout could be augmented by using sequestering agents, such as zeolites, that would be capable of chemical bonding with contaminants. If ancillary equipment was plugged at one or more points, several access ports would have to be installed to ensure complete grout filling.
- For purposes of assessing the environmental impacts associated with dispositioning of ancillary equipment and MUSTs as part of closure, it was assumed that the entire void volume within the ancillary equipment would be filled with grout and that no ancillary equipment or Inactive MUSTs would be excavated, packaged, or disposed of as LAW or mixed waste (WHC 1995i).
- Surface barriers (Hanford Barriers) would be placed over SSTs and DSTs for all alternatives except the No Action and Long-Term Management alternatives. Barriers would also be placed over the LAW vaults described in the Ex Situ Intermediate Separations, Ex Situ No Separations, Ex Situ Extensive Separations, Ex Situ/In Situ Combination 1 and 2, and the Phased Implementation alternatives.

B.6.0 THE HANFORD BARRIER

This section describes the multi-layered barrier, or Hanford Barrier, which is included in all alternatives that include closure activities as a representative surface barrier (cap) for closure as a landfill.

The Hanford Barrier would be a horizontal, multi-layered above grade engineered soil structure whose function would be to isolate the waste site from the environment by preventing or reducing the likelihood of wind erosion, water infiltration, and plant, animal, and human intrusion. It would be composed of 10 layers, with a combined thickness of 4.5 m (14.8 ft), and placed over the top of the stabilized tanks and the LAW disposal sites. Each Hanford Barrier would extend an additional 9 m (30 ft) beyond the perimeter of the area to be protected. Performance objectives of the Hanford Barrier system would include the following:

- Function in a semi-arid to sub-humid climate;
- Limit the amount of water migration through the waste to near zero amounts;
- Be maintenance free;
- Minimize the likelihood of intrusion by plants, animals, or people;
- Limit the amount of gases released;
- Minimize erosion;
- Meet or exceed RCRA cover performance requirements;

- Isolate waste for a minimum of 1,000 years; and
- Be acceptable to regulators and the public.

B.6.1 DESCRIPTION OF THE BARRIER LAYERS

The layers of the barrier are described from the top down. The Hanford Barrier design is provided in Figure B.6.1.1.

Top Vegetative Cover

The top vegetative cover would be for water retention and removal. Five species of perennial grasses would be planted across the barrier top. Seeding would include disking the soil, applying granular fertilizer, and seeding with a perennial grass mixture. To help establish cover grass, the site would be mulched with straw, which would be crimped into the soil to minimize wind erosion until the vegetation cover is developed.

Top (First and Second) Barrier Layers

The first barrier layer would consist of topsoil with a pea-gravel mixture; the second layer would consist of topsoil without pea-gravel. The first layer would be 1 m (3.3 ft) of sandy silt to silt loam soil with a 15 percent by weight admixture of pea-gravel. This layer would be placed loosely with a bulk density of 1.46 grams(g)/cm^3 (18.7 pounds [lb]/ft^3). The second layer would have the same type of topsoil; however, the bulk density would be approximately 1.38 g/cm^3 (86 lb/ft^3). These two layers would manage water by storing precipitation, providing a media for the growth of cover vegetation, and allowing evaporation and transpiration by the cover plants. The proposed topsoil would be obtained from a borrow site.

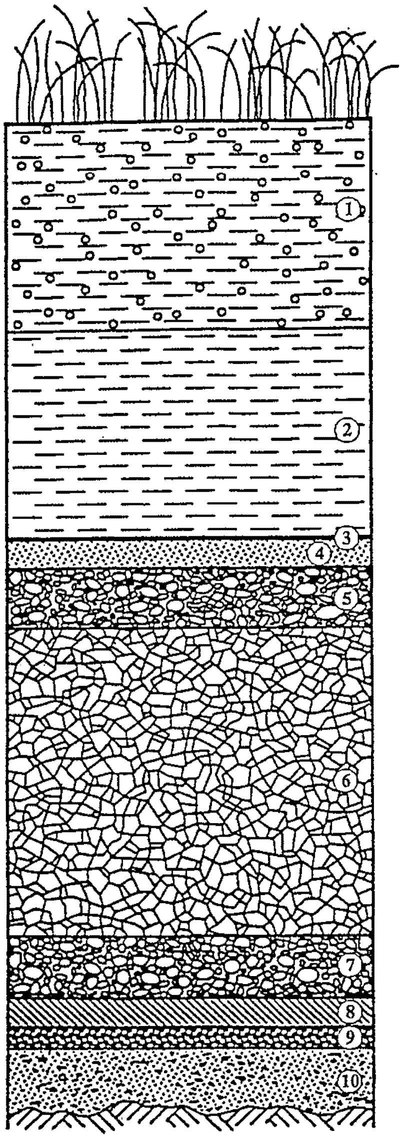
Third Barrier Layer

The third layer would be a geotextile, used primarily to separate topsoil layers from the sand filtration layer. After construction is completed, this geotextile would no longer have a specific function and therefore its long-term durability is not an issue.

Fourth and Fifth Barrier Layers

The fourth layer would be a sand filter, and the fifth layer would be a gravel filter. The purpose of these two layers would be to prevent migration and accumulation of fine-textured topsoil in the basalt layer. A capillary barrier, which occurs when a layer of fine-textured soil overlays a layer of coarser-textured soil (e.g., sand, gravel, or rock), would be created at the interface between the geotextile and the fourth layer (sand filter). Surface tension effects within the pore space of fine-textured soil would exert a negative pressure on the contained soil moisture. For moisture to drain out of fine-textured soil, surface tension would have to be overcome by developing gravitational pressure (hydraulic head) within the layer. In effect, some portion of the full thickness of this fine-soil layer will have to become completely saturated before drainage could occur. The sand filter would be 0.15 m (0.5 ft) deep, and the gravel filter would be 0.3 m (1 ft) deep. Both layers would be obtained from a borrow site.

Figure B.6.1.1 Hanford Barrier



Cover Vegetation: Mixed perennial grasses

Layer 1: (100 cm; 40 in.) Silt loam topsoil with pea gravel admixture

Layer 2: (100 cm; 40 in.) Silt loam topsoil without pea gravel

Layer 3: (0.1 cm; 0.04 in.) Geotextile filter fabric

Layer 4: (15 cm; 6 in.) Sand filter layer

Layer 5: (30 cm; 12 in.) Gravel filter layer

Layer 6: (150 cm; 60 in.) Coarse, fractured basalt

Layer 7: (30 cm; 12 in.) Lateral drainage layer (drainage gravel)

Layer 8: (15 cm; 6 in.) Low-permeability asphalt layer

Layer 9: (10 cm; 4 in.) Asphalt base course

Layer 10: (variable thickness) Grading fill

SOURCE: WHC 1995i

Sixth Barrier Layer

The sixth layer would be constructed of coarse basalt smaller than 25 by 5 cm (10 by 2 in.). The basalt layer would control biointrusion from plant roots, burrowing animals, and humans. The basalt would impede exploratory drilling. A subsurface layer consisting of loose fractured rock would pose a particularly adverse drilling condition for the following reasons:

- Circulation could not be maintained;
- Cuttings could not be adequately removed from the hole;
- The drill bit could not receive adequate lubrication; and
- Firm contact could not be maintained between the bit and the rock.

All of these factors would contribute to high bit wear and minimum advance of the drill hole. In addition, the layer would prevent moisture retention because large void spaces will enable water to drain into the seventh layer.

Seventh Barrier Layer

The seventh layer would be for lateral drainage. It would consist of screened material having a diameter of 1 millimeter (mm) (0.04 in.) or greater, which would give a hydraulic conductivity of at least 1 cm/sec (0.4 in./sec). This layer is part of contingency planning; any water draining to the seventh layer would be collected and/or diverted to the edge of the cover because of the 2 percent slope. This layer would be approximately 4 m (13 ft) below final grade to protect against frost penetration.

Eighth Barrier Layer

The eighth layer would consists of asphalt that would serve as a low-permeability barrier and as a secondary biointrusion barrier. The asphalt would be a durable asphaltic concrete mixture consisting of double-tar asphalt with added sand as a binder material. This layer would be 0.15 m (0.5 ft) thick with a hydraulic conductivity of approximately 1.0E-8 cm/sec. Natural analog studies estimate that this asphalt could remain functional for a period of 5,000 years or more as long as the asphalt remains covered and protected from ultraviolet radiation and freeze and thaw activity. To provide additional protection against leakage, the asphaltic concrete would be coated with a sprayed asphaltic coating material, which would be puncture-resistant, flexible, and easy to apply. The asphaltic coating material would have a permeability value of about 1.0E-11 cm/sec.

Ninth Barrier Layer

The ninth layer would be an asphalt base course that would provide a stable base for constructing the asphalt layer.

Tenth Barrier Layer

The tenth layer would contain grading fill that would establish a smooth, planar base surface for constructing the barrier layers. The sites covered by the Hanford Barrier would be contoured and graded for a uniform slope of 2 percent.

Locations to be Covered

The following locations would be covered with a Hanford Barrier:

- Tank Farm A (SST 6 tanks): 0.86 hectares (ha) (2.13 acres [ac]);
- Tank Farm AN (DST 7 tanks): 1.30 ha (3.21 ac);
- Tank Farm AP (DST 8 tanks): 1.17 ha (2.89 ac);
- Tank Farm AW (DST 6 tanks): 0.91 ha (2.26 ac);
- Tank Farm AX (SST 4 tanks): 0.63 ha (1.55 ac);
- Tank Farm AY (DST 2 tanks): 0.37 ha (0.92 ac);
- Tank Farm AZ (DST 2 tanks): 0.37 ha (0.92 ac);
- Tank Farm B (SST 16 tanks): 1.85 ha (4.57 ac);
- Tank Farm BX (SST 12 tanks): 1.54 ha (3.80 ac);
- Tank Farm BY (SST 12 tanks): 1.55 ha (3.84 ac);
- Tank Farm C (SST 16 tanks): 1.89 ha (4.68 ac);
- Tank Farm S (SST 12 tanks): 1.57 ha (3.89 ac);
- Tank Farm SX (SST 15 tanks): 1.91 ha (4.72 ac);
- Tank Farm SY (DST 3 tanks): 0.65 ha (1.61 ac);
- Tank Farm T (SST 16 tanks): 1.85 ha (4.57 ac);
- Tank Farm TX (SST 18 tanks): 2.46 ha (6.09 ac);
- Tank Farm TY (SST 6 tanks): 0.87 ha (2.16 ac);
- Tank Farm U (SST 16 tanks): 1.89 ha (4.67 ac); and
- LAW disposal vault.

B.6.2 SUMMARY OF BORROW SITES AND BORROW MATERIALS

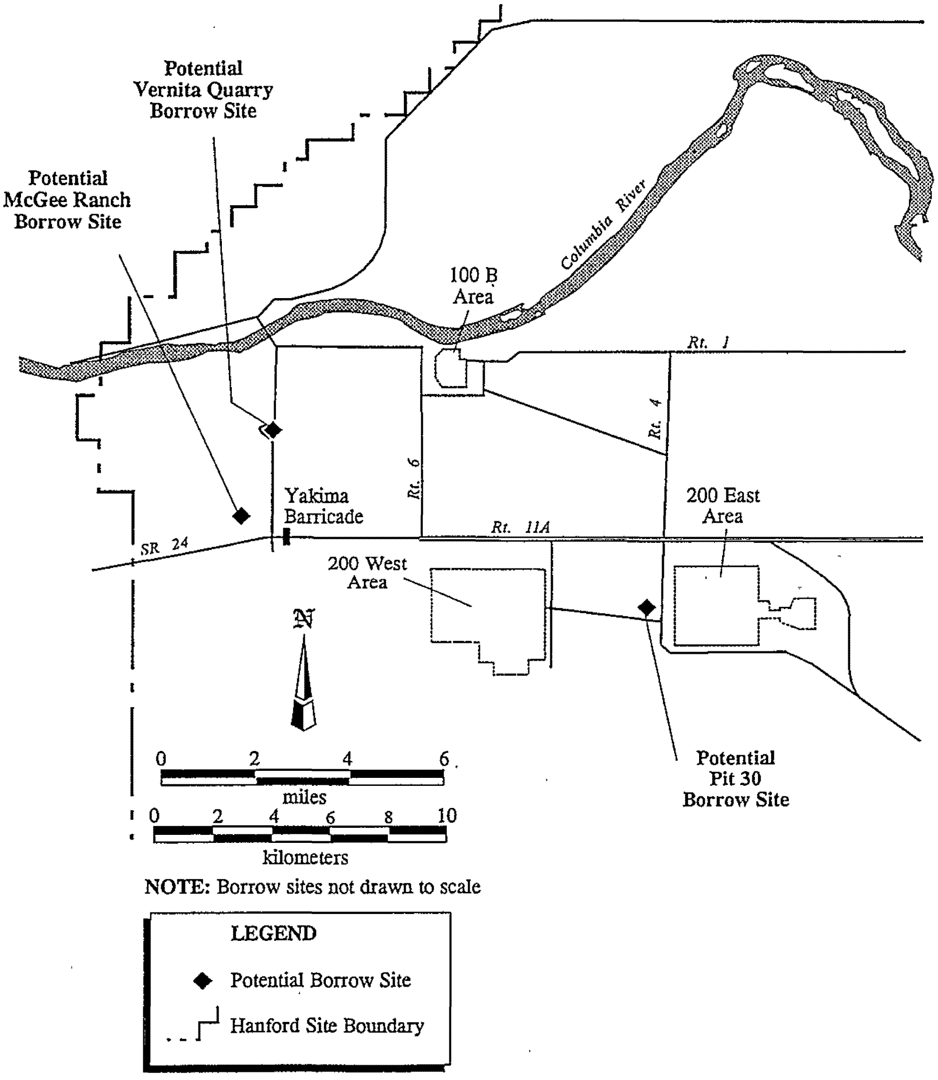
There are three sites assumed in the engineering data packages for borrow materials. These are Pit 30, which would supply sand and aggregate; the Vernita Quarry, which would supply riprap; and McGee Ranch, which would supply silt. These areas are also potential borrow sites. A decision on exactly which borrow site would be used and to what extent they would be used would be made through future NEPA analysis. The following is a brief description of the location and estimated distance of the potential borrow sites used for calculation purposes.

These potential borrow sites have been evaluated previously (BHI 1995) with respect to site proximity, availability of borrow material, transportation, safety, and land reclamation. This preliminary analysis indicated that all the potential borrow sites were suitable sources of borrow material. Figure B.6.2.1 shows the location of the proposed borrow sites.

Pit 30

The potential Pit 30 borrow site is an existing and established borrow pit, located approximately centrally with respect to the 200 Areas. The estimated haul distance is 10 km (6 mi) round-trip to the 200 East and 200 West Areas.

Figure B.6.2.1 Potential Borrow Sites for TWRS



NOTE: Borrow sites not drawn to scale

SOURCE: Adapted from DOE 1993e

Vernita Quarry

The potential Vernita Quarry borrow site is located east of State Route 24 near the Vernita Bridge. It was probably originally used to support highway construction. Distance estimates (one-way) are 12 km (7 mi) from the 200 West Area and 19 km (12 mi) to the 200 East Area.

McGee Ranch

The potential McGee Ranch borrow site is located west of State Route 24 and north of Route 11A. Distance estimates (one-way) are 11 km (6.5 mi) to the 200 West Area and 18 km (11 mi) to the 200 East Area.

Borrow Material Quantities

The estimated quantities of borrow materials taken from the various engineering data packages are shown on the following three tables. Table B.6.2.1 shows borrow materials used during construction and operation for the alternatives. Table B.6.2.2 shows borrow materials that are estimated for backfilling the empty tanks for all of the ex situ alternatives. Table B.6.2.3 shows the data for borrow materials used in the construction of the multi-layered barriers (Hanford Barriers), which may be placed over the tanks and vaults. In this table, each group of alternatives uses the same quantity of borrow materials.

Table B.6.2.1 Borrow Site Summary - Materials Used During Construction and Operations

Alternative	Borrow Type	Amount in m^3 (ft^3)	Potential Location
No Action	N/A	N/A	N/A
Long-Term Management	Aggregate/Sand	3.8E+05 (1.34E+07)	Pit 30
In Situ Fill and Cap [1]	Aggregate/Sand	7.54E+05 (2.67E+07)	Pit 30
In Situ Vitrification [1]	Aggregate/Sand	6.79E+05 (2.40E+07)	Pit 30
Ex Situ Intermediate Separations	Aggregate/Sand	1.66E+06 (5.87E+07)	Pit 30
Ex Situ No Separations			
- Vitrification	Aggregate/Sand	2.90E+06 (1.03E+08)	Pit 30
- Calcination	Aggregate/Sand	1.33E+06 (4.71E+07)	Pit 30
Ex Situ Extensive Separations	Aggregate/Sand	9.50E+05 (3.36E+07)	Pit 30
Ex Situ/In Situ Combination 1 [1]	Aggregate/Sand	1.03E+06 (3.65E+07)	Pit 30
Ex Situ/In Situ Combination 2 [1]	Aggregate/Sand	1.03E+06 (3.65E+07)	Pit 30
Phased Implementation (Phase 1 only)	Aggregate/Sand	3.20E+04 (1.13E+06)	Pit 30
Phased Implementation (Phase 2)	Aggregate/Sand	1.20E+06 (4.24E+07)	Pit 30

Notes:
[1] Includes material to fill tanks that would be treated in situ.
N/A = Not applicable

Table B.6.2.2 Borrow Site Summary - Materials Used for Backfill of Empty Tanks for all Ex Situ Alternatives

Alternative	Method	Kind	Amount in m^3 (ft^3)	Potential Location
Ex Situ (all)	Tank Stabilization	Aggregate	7.58E+05 (2.70E+07)	Pit 30
Ex Situ/In Situ Combination 1	Tank Stabilization	Aggregate	3.30E+05 (1.17E+07)	Pit 30
Ex Situ/In Situ Combination 2	Tank Stabilization	Aggregate	1.18E+05 (4.16E+06)	Pit 30
Phased Implementation (Phase 2)	Tank Stabilization	Aggregate	7.58E+05 (2.70E+07)	Pit 30

Table B.6.2.3 Borrow Site Summary - Materials Used for Construction of Hanford Barriers

Alternative	Borrow Kind	Amount in m^3 (ft^3)	Potential Location
In Situ Fill and Cap, In Situ Vitrification, or Ex Situ No Separations [1]	Silt	3.77E+05 (1.30E+07)	McGee Ranch
	Riprap	6.38E+05 (2.30E+07)	Vernita Quarry
	Aggregate/Sand	4.24E+05 (1.50E+07)	Pit 30
Ex Situ Intermediate Separations, Ex Situ Extensive Separations, or Phased Implementation (Phase 2) [2]	Silt	5.66E+05 (2.0E+07)	McGee Ranch
	Riprap	9.85E+05 (3.48E+07)	Vernita Quarry
	Aggregate/Sand	5.81E+05 (2.06+07)	Pit 30
Ex Situ/In Situ Combination 1 [2]	Silt	4.72E+05 (1.67E+07)	McGee Ranch
	Riprap	7.85E+05 (2.8E+07)	Vernita Quarry
	Aggregate/Sand	5.16E+05 (1.83E+07)	Pit 30
Ex Situ/In Situ Combination 2 [2]	Silt	4.25E+05 (1.50E+07)	McGee Ranch
	Riprap	7.18E+05 (2.54+07)	Vernita Quarry
	Aggregate/Sand	4.82E+05 (1.71E+07)	Pit 30

Notes:

[1] Includes materials to construct barriers over tank farms.

[2] Includes materials to construct barriers over tank farms and the LAW disposal vaults.

N/A = Not applicable

B.7.0 SITING OF FACILITIES

This section describes the preliminary siting study that was performed to develop a representative site for impact assessment purposes.

The site optimization process would be implemented to ensure that new facilities would be located at a site that meets facility requirements and minimizes the impacts associated with construction and operations. The site optimization process would involve identifying and evaluating sites based on selection criteria that incorporate stakeholder values.

The site optimization process for the TWRS sites is an ongoing program whose function is to identify a site that best meets the selection criteria. The in situ alternatives would be sited at the existing tank farms and would require site selection for support facilities. For the ex situ alternatives, the area proposed for potential sites has been restricted to in and around the 200 East Area. The 200 Areas have been heavily used for fuel reprocessing and waste management and disposal activities. The 200 East Area location was selected for the following reasons.

- Based on the TWRS Facility Configuration Study (Boomer et al. 1994) and the TWRS Process Flowsheet (Orme 1994), pretreating tank waste (if done in an existing tank) would be done by the in-tank sludge washing process in the 200 East A Farm Tank Complex. Tank waste from the 200 West Area would be retrieved to the SY Tank Farm and transferred cross-site to the AW Tank Farm where in-tank sludge washing would be performed. Waste in the 200 East Area would be retrieved to the AN Tank Farm where it would be washed and separated into HLW and LAW streams. The LAW streams would be pumped to the AP Tank Farm and then to the pretreatment and LAW vitrification facilities. The HLW streams would be pumped directly from the AN and AW Tank Farms to the HLW vitrification facility or to interim storage.
- The Hanford Site has consolidated activities over the past 20 years in the 200 East Area, as opposed to the 200 West Area, which has placed much of the necessary facilities and infrastructure in and around the 200 East Area.
- There is more available, useable land in the 200 East Area than the 200 West Area (i.e., land that is unused or is not reserved for other use).

B.7.1 SELECTION CRITERIA

Hanford Site evaluation criteria used for evaluating potential sites considered stakeholder values, regulatory compliance issues, costs, and risks. In a site selection study the selection criteria described in the following sections have been based on stakeholder values, regulatory compliance, and cost and risk reduction (Shord 1995 and Jacobs 1996).

B.7.1.1 Protect the Environment

Cultural, Archeological, and Historical Sites

The TWRS remediation site shall not have any areas of cultural, archeological, or historical significance that cannot be reasonably mitigated.

Ecological

The TWRS remediation site shall not have any areas of ecological impact that cannot be reasonably mitigated.

Groundwater Protection

The Columbia River shall be protected, and groundwater contamination will be dealt with realistically and forcefully. This issue concerns the ability of the Hanford Site to meet Federal, State, and local requirements for protecting groundwater. Factors include the 1) impact of previous Hanford Site practices (e.g., liquid effluent discharges, SST leaks, disposal actions) on groundwater under the Site; 2) hydrology of the Site; and 3) the impact of the Site on proposed future Hanford Site disposal operations (e.g., LAW disposal).

Harm During Cleanup
Establishing the TWRS complex (on the particular site) shall cause no irreparable harm to the environment.

Natural Resource Damage
The TWRS remediation site shall minimize and avoid any impacts to natural resources.

B.7.1.2 Protect Public/Worker Health and Safety

Transportation
Waste will be transported safely, and measures will be taken to prepare for emergencies.
The transportation of radioactive and hazardous waste and material through populated areas will be kept to a minimum.

Exposures
Exposures will be as low as reasonably achievable. The TWRS remediation site shall minimize the adverse impacts on the health and safety of personnel. The concept of reducing the exposure of workers to radiological and hazardous substances to as low as reasonably achievable principles will be considered.

Accidents on the TWRS Complex
The TWRS remediation site will minimize the effects of possible accidents at adjacent facilities on the TWRS complex.

Accidents from the TWRS Complex
The TWRS remediation site will minimize the effects of possible accidents at the TWRS complex and its associated facilities (e.g., transfer lines) on adjacent facilities.

B.7.1.3 Use the Central Plateau Wisely for Waste Management

Land use planning for the TWRS remediation site should be in concert with and not conflict with other land use planning documents.

B.7.1.4 Promote Local Economic Development

The TWRS remediation site will capture economic development opportunities locally by being conducive to privatization of facilities.

B.7.1.5 Support the Tri-Party Agreement

The TWRS remediation site will support meeting the Tri-Party Agreement schedule and get on with cleanup to achieve substantive progress in a timely manner.

B.7.1.6 Consider Cost Impacts

The following cost impacts shall be considered.

Construction Costs

Utilities

The installation/upgrade costs of electricity, raw water, sanitary water, steam, and telecommunications. Existing and planned utilities will be considered.

Railroads

The installation/upgrades costs of rail and roads.

Liquid Effluent Disposal

The installation of liquid effluent disposal lines from the complex to the liquid effluent disposal system.

Sanitary Sewer

The installation costs of a sanitary sewer to tie into the planned 200 East Area sanitary sewer system (Project L-116).

Storm Water Runoff

The installation costs of a system to channel stormwater away from the site.

Construction Proximity

The ability to locate temporary construction support facilities close to the facilities being constructed and the availability of adequate laydown and construction support areas.

Construction Commonality

Maximize the use of common construction support needs (laydown areas, utilities, parking, batch plant, offices, shops, warehouse, and change rooms) between project or construction phases of multiple facilities of the same project.

Site Preparations

Costs associated with earth-moving activities necessary to complete construction. Factors include topography, site irregularities, and finish grade elevation. The removal/relocation of existing structures are additional factors.

Operating Costs

Operating costs between the various sites shall be qualitatively assessed and shall include items such as facility and feed/waste transfer costs of flushing, diluting waste, concentrating diluted waste (evaporating waste to manage DST space), and line drain back.

B.7.1.7 Provide Flexibility

Provide flexibility in the following areas.

Site Expansion
Adequate expansion area should be available for future TWRS facility needs. Although the expansion area cannot be quantified at this point, more potential expansion area is preferable to less.

Facility Relationships
The TWRS remediation site should allow the interacting of process facilities to maximize use of common support facilities and utilities and facilitate flows (tank waste transfers, raw materials, effluent disposal, process waste streams) between process facilities and related operations.

Compatibility
The TWRS remediation site should be compatible with ongoing programs, current construction projects, and planned projects.

Proximity
The TWRS remediation site should possess the ability to 1) move the vitrified waste to HLW interim storage and subsequently to final storage offsite; and 2) retrieve LAW from onsite disposal for repackaging for offsite shipment.

Contracting Flexibility
The TWRS remediation site should be conducive to the use of innovative contracting concepts such as 1) fixed-price contracts for design, construction, startup, and initial operations; and 2) privatization. Ease of access, interfaces with site operations, and the potential to encounter unforeseen conditions are to be considered.

B.7.1.8 Reduce Risks

Reduce risks (technical, regulatory, operational, construction, and planning) in the following areas.

Hydraulics
The potential for transfer line plugging should be minimized to the extent possible. Factors to be considered should include waste transfer system configuration (i.e., number of process pits), line traps, quantity of flush water after each transfer, line drain back to low point, number of low points in system, dilution requirements to mitigate plugging of transfer system, pumping requirements (to minimize the use of pump booster stations), and siphoning effect between the shipping location and the processing facilities. In essence, the inner tank/facility piping should be free draining (to the extent practical) to the transfer destination.

Proximity to Existing Facilities
The distance between the processing facilities for pretreatment/LAW treatment and HLW, and the DSTs existing in the 200 East Area (A Farm Complex) shall be kept to a practical minimum.

Interferences and Contamination
Minimize potential problems to be encountered during construction and operation due to existing above or belowground structures or radioactive/hazardous contamination.

Seismic
The distance to known earthquake faults shall be taken into consideration.

Site Activities
The impact on other Hanford Site activities and operating facilities during construction and operation should be kept to a minimum.

Decontamination and Decommissioning
The decontamination and decommissioning activities in the 200 East Area should be considered in siting the TWRS complex. This would include the decontamination and decommissioning impact of other facilities in the area on the TWRS complex and the ultimate decontamination and decommissioning of the TWRS complex.

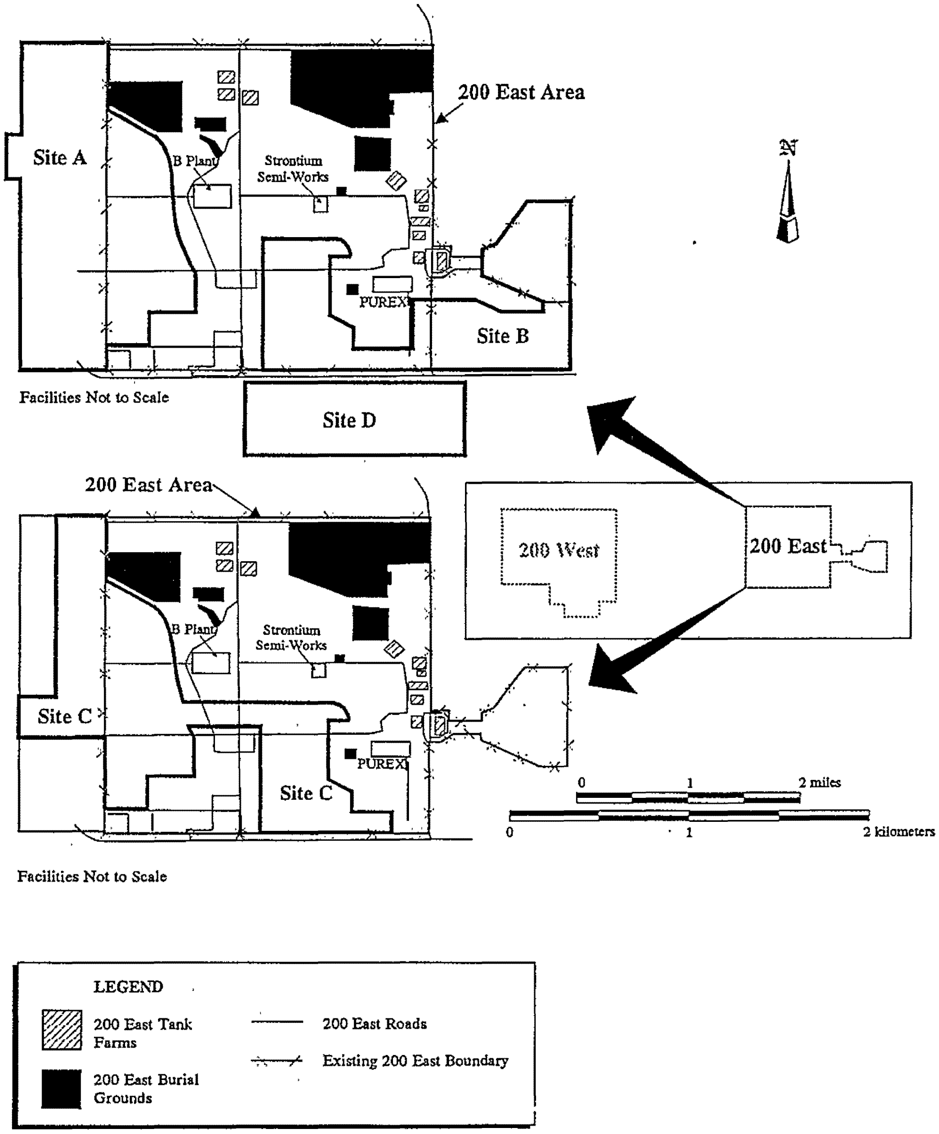
Design
The need for new technology/design complexity should be minimized.

B.7.2 RECOMMENDATION

The final site selection for the facilities associated with the ex situ alternatives has not been made. However, a recommended site has been nominated based on the applicability of the eight criteria that were given previously and adopted for use in this EIS. The selection process focused on six alternate layouts in the 200 East Area. Each layout was evaluated and given a numerical ranking for each of the eight criteria. Comparison matrix was then constructed to compare the ranking of each layout. The location and size of the highest ranking layout are shown as Site C in Figure B.7.2.1. Sites A, B, and D included alternate layouts that did not score as high for locating the full-scale treatment facilities.

For purposes of the EIS, a combination of Site B and Site C has been assumed to be a representative site capable of accommodating the full-scale processing facilities, LAW disposal, and HLW temporary storage for all ex situ alternatives. Site B has been assumed to be a representative site for locating the Phase 1 treatment facilities under the Phased Implementation alternative (WHC 1996). These sites are considered to be representative sites for the purpose of alternative evaluation. This does not preclude other sites from ultimately being selected and appropriate NEPA analysis will be completed prior to final site selection. To support the analysis of environmental impacts in this EIS, the representative site is used as the location where each of the ex situ alternatives would be located. All of the ex situ alternatives will be treated as if they were located on the representative site.

Figure B.7.2.1 Potential Site Location



SOURCE: Shord 1995

B.8.0 MAJOR ASSUMPTIONS

To develop engineering data required to perform impact analyses for each of the alternatives discussed in the EIS, assumptions were made regarding the technologies that have been configured to create a remediation alternative. These assumptions were based either on the best information available, applications of a similar technology, or engineering judgement. By definition when an assumption is made there is some level of uncertainty associated with it that can be expressed as a range for the assumed value that reasonably could be expected. This section identifies the major assumptions used for the alternatives and the uncertainties associated with the cost estimates. Uncertainties associated with the engineering data are discussed in Volume Five, Appendix K.

B.8.1 IN SITU ALTERNATIVES

It was assumed that there would be no leaks from the SSTs or DSTs during the administrative control period for the No Action, Long-Term Management, or In Situ Fill and Cap alternatives. This assumption is based on ongoing SST interim stabilization to remove pumpable liquids, the ability to detect and recover leaks from the space between the inner and outer liners of the DSTs, and ongoing monitoring activities within the tank farms. The SSTs and DSTs were assumed to maintain their structural integrity throughout the administrative control period under the No Action and Long-Term Management alternatives.

The In Situ Vitrification, In Situ Fill and Cap, and the in situ portion of the Ex Situ/In Situ Combination alternatives were assumed to require additional characterization data to evaluate the acceptability of in-place disposal and address RCRA land disposal requirement considerations. This requirement would be in addition to the current characterization requirements for the ex situ alternatives. These additional characterization efforts could involve extensive laboratory analysis of additional tank samples and may require modifications to the tanks to install additional risers for sampling access.

In Situ Vitrification

The in situ vitrification system was assumed to be capable of vitrifying each of the tanks to the required depth resulting in a consistent waste form. It was also assumed that the variation in waste composition and inventory from tank to tank would not impact the ability to produce an acceptable waste form.

In Situ Fill and Cap

The concentrated liquid waste contained in the DSTs was assumed to be acceptable for gravel filling. Under the In Situ Fill and Cap alternative, the DST liquids would be concentrated using the 242-A Evaporator to remove as much water from the waste as possible but would still contain substantial volumes of liquid. It has been estimated that concentration by the 242-A Evaporator would reduce the current liquid volumes contained in the tanks by approximately one-third (WHC 1995f).

B.8.2 EX SITU ALTERNATIVES

Waste Retrieval Efficiency

The waste retrieval function described for the ex situ alternatives was assumed to remove 99 percent of the waste volume contained in each tank during waste retrieval. Under this assumption, 1 percent of the tank volume would be left in-tank as a residual. It was further assumed that the 1 percent waste volume represented 1 percent of the waste inventory on a chemical and radiological basis.

The amount and type of waste that would remain in the tanks after retrieval is uncertain. The Tri-Party Agreement (Ecology et al. 1994) set a goal for the SSTs that no more than 1 percent of the tank inventory would remain as a residual following waste retrieval activities. The engineering data for the waste retrieval and transfer function common to all ex situ alternatives was developed using 99 percent retrieval as a goal. However, achieving this level of tank waste retrieval may require extraordinary effort and cost, and it may not be practicable to achieve 99 percent retrieval from all tanks.

Releases During Retrieval

Retrieval of SST waste under each of the ex situ alternatives was assumed to result in the release of 15,000 L (4,000 gal) from each SST to the soils surrounding the tank during retrieval operations. It was also assumed that the contaminant concentrations in the liquids released were at maximum predicted concentrations using the congruent dissolution model. See Volume Four, Section F.2.2.3 for a discussion on the congruent dissolution model. No leakage was assumed to occur from the DSTs during retrieval operations because DSTs have provisions for leak containment and collection. This assumption is based on having 67 known or suspected SSTs that have leaked in the past (Hanlon 1995). Most of the SSTs were built in the 1940's and now are about 50 years old. The leakage volume estimate was based on current information from the waste retrieval program and on the assumption that the average leakage from an SST would be one order of magnitude lower than the maximum release estimated for tank 241-C-106 during sluicing operations. The maximum leak estimated from tank 241-C-106 during sluicing operations was 150,000 L (40,000 gal). The leak estimate for tank 241-C-106 assumes that the leak occurs early in the sluicing operation, leak detection devices and controls fail, sluicing operations proceed without these leak detection devices, the leak(s) occur at the bottom of the tank, and the remaining sludge does not plug any leaks (DOE 1995d).

The assumption that each of the 149 SSTs leaks 15,000 L (4,000 gal) during retrieval is conservative and provides an upper bound of 2,260,000 L (596,000 gal) on the calculated impacts from tank leakage during retrieval. Total leakage from all SSTs during retrieval operations would be expected to be lower than the bounding values used because of the following assumptions.

- Seventy-five percent of the tanks that are known or suspected leakers are assumed to have leaked at the air-water interface on the sidewall of the tank and would remain above the liquid level during sluicing (51 tanks).
- Twenty-five percent of the tanks that are known or suspected leakers are assumed to have leaked at or near the tank bottom and would be retrieved using a robotic arm

based system (16 tanks). The robotic arm based system would not use the large volumes of liquids required for sluicing operations.
- Leak detection systems would be used during waste retrieval operations, and indications of tank leakage during retrieval would result in actions taken to minimize leakage. These actions could include switching to robotic arm based systems or limiting the amount of sluicing liquid in the tank.
- Administrative controls would be used to monitor liquid inventories.
- There is a tendency for solids in the sludge to plug any leaks.
- The free liquid in the tanks during sluicing could be pumped out in a short time using the transfer pumps.

The most probable occurrence of a leak during sluicing would involve the sluicers opening a plugged leak in the tank wall. The waste leakage during sluicing would be any free-standing liquid above the level of the leak point and the sluicing stream as it impacts the tank wall. Based on historical leak rates of other SSTs, the actual leaked volume is expected to be on the order of a few thousand liters (a few thousand gallons) (DOE 1995d). DOE currently is working with Ecology to define the operating envelope for allowable leakage during retrieval. Final design of the waste retrieval systems would include measures to detect control leakage.

Tank Residuals

The residual contaminants left in the tanks would either be insoluble and hardened on the tank walls and bottom or be of a size that could not be broken up and removed from the tanks. In either case, the residual would have low solubility because the retrieval technologies proposed would use substantial quantities of liquid in an attempt to dissolve or suspend the waste during retrieval. Because of the uncertainties regarding the amount and type of residual waste that would remain in the tanks, a conservative assumption was made to bound the impact of the residual waste. For purposes of the analysis, it was assumed that 99 percent recovery would be achieved for ex situ alternatives, and the residual waste left in the tanks would contain 1 percent of all the original tank inventory, including the water-soluble contaminants.

The assumption that the 1 percent tank residuals following retrieval represent 1 percent of the original tank inventory is conservative because it assumes that soluble and insoluble constituents would remain as residuals in the same proportions as the original tank inventory. The effect of retrieving less than 99 percent of the waste volumes from the tanks during retrieval would be an increase in the amount of waste left in the tanks and corresponding increases in groundwater contaminant concentrations and post-remediation risk. The in situ and combination alternatives leave substantially more waste onsite for disposal and provide an upper bound on the impacts associated with the amount and type of waste that is disposed of onsite.

A nominal case retrieval release and residual tank inventory was developed to assess the impacts that would result from nominal, as compared to bounding, assumptions for tank releases during retrieval and the residual waste left in the tanks following retrieval. Additional information on the nominal case is provided in Section B.3.0.

The nominal retrieval release inventory was developed by assuming that the waste would be diluted by one-third by adding water during waste retrieval. Possible dilution ratios that would be used during waste retrieval range from 3:1 to 10:1. Thus, the dilution factor of one-third assumed for the nominal case is a conservative assumption and is substantially lower than the dilution factions that would be obtained using 3:1 or 10:1 dilution ratio. These dilution ratios represent the amount of liquid required to mobilize the waste solids and would be made up of existing tank liquids and water additions.
The nominal case retrieval release volume was assumed to be 15,000 L (4,000 gal) from each SST and the contaminant concentrations were assumed to be two-thirds of the bounding case. The average volume of waste released from each SST during retrieval was not reduced for the nominal case because insufficient information is available to support a lower average release volume. The volume of waste released during retrieval would depend on the ability to detect a leak and take corrective action.

The nominal tank residual inventory was developed by modifying the bounding tank residual inventory to reduce the mobile constituents of concern based on solubility. The mobile constituents of concern were evaluated because of their contribution to post-remediation risk. The isotopes C-14, Tc-99, and I-129 were reduced for the nominal case tank residual inventory to 10 percent of the bounding tank residual inventory. This is based on the assumption that 90 percent of the residual inventory of these isotopes would be soluble in the retrieval liquids and would be retrieved from the tanks for ex situ treatment. Typical sludge wash factors representing the solubility in water for each of these isotopes are as high as 99 percent. The nominal case residual was limited to 90 percent to account for conditions where the scale and hardened sludges were not exposed to the sluicing liquid during retrieval. Table B.8.2.1 shows the nominal and bounding residual inventories for select mobile constituents.

Table B.8.2.1 Tank Residual Inventory in Curies

Isotope	Existing Tank Inventory	Bounding Residual Inventory	Nominal Case Residual Inventory
C-14	5,340	53	5.3
Tc-99	32,100	320	32
I-129	38	0.38	0.038

Assumptions Affecting HLW Volume

The major factors that affect the volume of HLW produced by any of the ex situ alternatives include waste inventory, waste loading (glass specifications), blending, and the efficiency of the separations processes.

The waste inventory that has been used for all alternatives is provided in Volume Two, Appendix A along with a discussion on data accuracy and uncertainty.

Waste loading is the mass fraction of the nonvolatile waste oxides in the vitrified waste. The waste oxide loading would be controlled by the amount of glass formers that are added during the vitrification process. The higher the waste loading, the more waste that would be contained in the vitrified glass and the lower the waste volume.

Blending is the mixing of the waste from different tanks during retrieval to obtain an average waste feed stream for treatment. Because there are 177 tanks that contain waste and the waste composition varies from tank to tank, it would be difficult to achieve a completely uniform blending of the waste during retrieval.

Separating the waste into HLW and LAW streams for treatment would involve various processes to physically or chemically separate specific constituents in the waste stream. The separations efficiency would be a measure of how well these processes work and would define the amount of each constituent that would be processed in the HLW and LAW treatment facilities.

The assumptions used for each of the factors described previously and their combined affect on the overall volume of HLW and LAW are discussed in the following sections.

Waste Loading

The waste loading for all ex situ treatment alternatives except for Ex Situ No Separations was assumed to be 20 weight percent waste oxides for the HLW and 15 weight percent sodium oxide for the LAW. The waste loading for the Ex Situ No Separations alternative was assumed to be 20 weight percent sodium oxide.

Waste loading was assumed to be 20 weight percent waste oxides (this includes all waste constituents that would be converted to oxides in the vitrified waste form, excluding the sodium and silica contained in the tank waste) for HLW glass for each alternative that would involve separating the HLW and LAW. Because the No Separations alternative would not separate the HLW and LAW, all of the sodium in the waste inventory would be converted into the HLW glass and the methodology described for the other alternatives would not be valid. The 20 weight percent sodium oxide loading for the No Separations alternative would result in a glass that would be equivalent to established glass compositions defined in the Waste Acceptance System Requirements Document (DOE 1995s). The Waste Acceptance System Requirements Document does not set specific limits for the different constituents that make up waste loading, but instead requires that for acceptance a waste form must be equal to or better than the reference glass.

The waste loading would affect the volume of waste that would be produced from a given amount of waste. This volume, along with the operating schedule and the assumed operating efficiency, would determine the size of the processing facilities and the operating resource requirements required to support the process. A decrease in waste loading would then translate into a larger volume of vitrified waste, larger treatment facilities or longer operating schedules, increased resource requirements, and higher disposal cost.

Waste loading may typically range from 20 to 40 weight percent waste oxides with 30 to 35 weight percent loading used as a target value. The Defense Waste Processing Facility glass has a design basis waste loading of 25 weight percent and a maximum waste loading of 38 percent (DOE 1995s).

The waste loading for all alternatives that would produce LAW was assumed to be 15 weight percent sodium oxide. The volume of LAW produced affects the size and number of LAW disposal vaults that would be built onsite.

Waste Blending

Each of the ex situ alternatives that use vitrification as an immobilization technology have assumed a waste blending factor of 1.2 for the HLW to account for variations in the composition of the waste during retrieval operations. Variations in the waste feed composition would not affect the calcined product that would be produced by the Ex Situ No Separations (calcination) alternative. Uniform blending would require simultaneous retrieval from specific groups of tanks to deliver a uniform average feed stream to the treatment facilities. The blending factor would be multiplied by the volume of HLW produced under uniform blending conditions to calculate the volume of HLW expected due to variations in the waste feed. Because variations in the waste feed composition would not be expected to affect the LAW vitrification process, a blending factor of 1.0 was assumed for LAW. One of the major sources of uncertainty associated with developing a retrieval sequence that would achieve a uniform blending was the lack of accepted tank-by-tank inventory data. The HLW blending factor for Hanford tank waste was the recommendation of an independent technical review team (Taylor-Lang 1996).

Separations Efficiencies

The volume of vitrified HLW produced would be a function of the waste loading and the mass of waste to be vitrified. Reducing the HLW volume through separations processes would therefore require separating the nonradiological constituents from the HLW constituents during the pretreatment process. The lower bound on the number of canisters that could be produced would be controlled by the heat-generating limit of 1,500 W per canister (DOE 1995q). This heat-generating limit would provide a lower bound on the number of 1.2-m^3 (41-ft^3) canisters of 177 for the tank waste and 298 for the tank waste combined with the Cs and Sr capsules (WHC 1995e). The following flowsheet assumptions would affect the volume of HLW produced:

Intermediate Separations, Phased Implementation, and ex situ portion of the Ex Situ/In Situ Combination 1 and 2 alternatives:

- The enhanced sludge washing process would solubilize 85 percent of the aluminum, 75 percent of the Cr, and 70 percent of the phosphate into the liquid phase, and following solid-liquid separations these would be included in the LAW feed;
- Solid-liquid separation would assume gravity settling in the tanks followed by decanting of the liquid. The solids settling process was assumed to achieve 50 weight percent solids.

Extensive Separations:

- Solid liquid separations would use centrifuges capable of achieving 0.1 percent solid in clarified liquids.
- Acid-side dissolution of the solid phase species would assume between 50 and 90 percent dissolution in a two-step dissolution process. This would include recycling 95 percent of the undissolved solids from the second acid dissolution step back to the caustic leaching step to begin another dissolution cycle. The remaining 5 percent of the undissolved solids would be sent to the HLW process. There is uncertainty in the optimistic acid-side dissolution assumptions that are critical to the volume of HLW produced by the extensive separations process.

The volume of HLW produced would directly impact the number of HLW packages requiring disposal at the potential geologic repository, which in turn would affect the cost associated with disposal. The number of HLW packages produced would also determine the number of offsite shipments required to transport the immobilized HLW to the potential geologic repository. The waste loading would also determine the concentration of radiological contaminants in the waste form. There would be a relationship between the waste loading, number of shipments (probability of an accident), and the concentration of contaminants in the waste form (consequence of an accident). As the waste loading increased, the probability of an accident would go down because there would be fewer trips required to transport the waste, but the consequences of an accident would go up because there would be a higher concentration of contaminants in the waste form (see Appendix E, Section E.15.0 for a discussion of accident uncertainties).

Canister Size and Type

Two sizes of HLW canisters were assumed for the ex situ alternatives. All of the ex situ alternatives except the No Separations alternative assumed a canister size of 0.6-m inside diameter by 4.57-m long (2-ft inside diameter by 15 ft long) with a net volume capacity of 1.17 m^3 (41 ft^3). The Ex Situ No Separations alternative (both vitrification and calcination) assumed a canister size of 1.7-m diameter by 4.6 m long (5.5-ft diameter by 15 ft long) with a net volume capacity of 10 m^3 (360 ft^3).

It is recognized that these sizes are larger than the 0.62-m^3 (22-ft^3) standard size canister that is identified for canistered HLW in the current waste acceptance requirements at the potential geologic repository (DOE 1995q). However, the DOE Office of Civilian Radioactive Waste Management has recently acknowledged the technical acceptability of a longer canister (e.g., 0.6-m diameter by 4.6 m long [2-ft diameter by 15 ft long]) for Hanford HLW (Milner 1996). The larger 10-m^3 (360-ft^3) canister assumed for the No Separations alternative was not evaluated for acceptance by the Office of Civilian Radioactive Waste Management. The large canister would occupy the same space as a standard waste package. The standard waste package would consist of four 1.2-m^3 (41-ft^3) canisters within a large disposal container. The design of the waste package and canister sizing has not been finalized.

B.8.3 COST UNCERTAINTY

Cost uncertainty for the various tank waste treatment alternatives has been evaluated using Decision Science Corporation's Range Estimating Program for personal computers. The Range Estimating Program has been applied to thousands of diverse problems by thousands of users. The Range Estimating Program inputs allow the user to specify a simple range rather than require selection of a probability density function. The Range Estimating Program outputs identify, quantify, and rank the risks.

The upper level of the cost range for new technologies was estimated such that there was a high certainty that its capital or operating cost would not be exceeded. This upper level (as a percent of the estimated cost) varied up to a high of plus 200 percent based upon the degree of uncertainty and complexity of the technology. The use of this high-range level addressed the concerns expressed in the System Requirements Review, Hanford Tank Waste Remediation System Final Report issued April 1995, which indicated that actual costs of new technology facilities of the type under consideration herein can often exceed estimated costs by a factor of two or more (DOE 1995s).

The information presented in Table B.8.3.1 identifies a range for the total estimated cost of each alternative. This range represents the calculated variation in estimated cost that could occur for any of the alternatives. This range is a function of input parameters such as the level of design development, uncertainties associated with implementability, and assumptions made for the relative uncertainty of different cost components. The total estimated cost range is statistically based and was obtained through a Monte Carlo simulation. The input parameters are based on the alternatives described in the EIS; however, major changes to the waste inventory, conceptual designs, or major assumptions would change the estimated cost range.

Input to the Range Estimating Program was based on best available information, conceptual cost estimates, and engineering judgement (Jacobs 1996).

B.9.0 TECHNOLOGIES

As discussed in Section B.2.0, there are numerous technologies that could be used for remediating tank waste. Technologies are specific processes that form the building blocks of the alternatives. Alternatives are then made up of a set of technologies that have been designed to function together.

Table B.8.3.1 Comparison of Tank Waste Alternatives Cost Uncertainty

Alternative	Treatment Cost [1]		Total Alternative Cost [1] (Treatment + Repository Fee)	
	Estimated Cost	Estimated Cost Range [2]	Estimated Cost	Estimated Cost Range [2]
No Action	$14,300	$12,555 - $16,083	$14,300	$12,555 - $16,083
Long-Term Management	$20,770	$18,962 - $23,208	$20,770	$18,962 - $23,208
In Situ Fill and Cap	$7,885	$6,973 - $8,816	$7,885	$6,973 - $8,816
In Situ Vitrification	$16,478	$15,933 - $27,434	$16,478	$15,933 - $27,434
Ex Situ Intermediate Separations	$24,837	$23,887 - $29,950	$30,117	$29,037 - $35,145
Ex Situ No Separations Vitrification	$23,634	$22,646 - $27,717	$62,534	$59,184 - $74,538
Ex Situ No Separations Calcination	$21,700	$20,776 - $25,642	$36,330	$35,185 - $42,524
Ex Situ Extensive Separations	$27,725	$26,935 - $37,464	$27,995	$27,205 - $37,734
Ex Situ/In Situ Combination 1	$19,676	$18,322 - $23,266	$23,386	$22,132 - $26,983
Ex Situ/In Situ Combination 2	$15,432	$14,442 - $17,733	$17,582	$16,617 - $19,870
Phased Implementation	$27,199	$25,091 - $33,010	$32,479	$30,388 - $38,148

Notes:

[1] Cost is in millions of 1995 dollars.

[2] Cost as determined by Decision Science Corporation Range Estimating Program for personal computers (in millions of 1995 dollars).

Technologies that were not included in the alternatives that were developed for impact analysis, but are still viable as potential components of a remediation alternative are discussed in this section. For example, the technology selected for inclusion in the alternatives for immobilizing the LAW was vitrification. However, the ceramic waste form may also be viable and could be substituted as a LAW immobilization process.

B.9.1 IN SITU WASTE TREATMENT TECHNOLOGIES

In Situ Grout

Grout is a common solidification and stabilization technology used in managing hazardous waste. Stabilization is a process in which additives are mixed with the waste to minimize the rate of contaminant migrating from the waste form. Solidification is a process in which additives are mixed with the waste to yield a physical waste form, as measured by properties such as permeability and compressive strength, that is acceptable for waste storage or disposal. Performance measures used to evaluate solidification and stabilization technologies are obtained through leaching tests that provide data on the rate at which contaminants are released from the waste form under the action of water.

In situ grout is a technology that could be used to immobilize the waste and stabilize the tanks as an option to the waste drying and gravel filling operations described in the In Situ Fill and Cap alternative. Applying this technology would involve adding a grout mixture to each of the tanks, mechanically mixing the waste with the grout mixture, and stabilizing the tanks by filling the dome space with grout. Using this technology would leave the waste its current locations for disposal as in the In Situ Fill and

Cap alternative, except that the waste would be solidified in a grout matrix instead of dried. After completing grouting operations, a Hanford Barrier would be installed over each of the tank farms.

A pozzolan-based grout formulation made up of sand, flyash, water, cement, and air entrainment additive could also be used (WHC 1995f). Pozzolanic materials can react with lime in the presence of water to produce a solid cement-like material. Flyash is the most commonly used pozzolanic material. Other types of grout formulations include cement-based thermoplastics and organic polymer-based grouts. Implementation of this technology would require the following actions:

- Reduce the volume of liquid in the DSTs by evaporation;
- Construct a TFCF over each tank farm;
- Remove the soil covering the top of each tank;
- Remove the top of each tank (dome) for access by the grout mixer; and
- Mix the waste mechanically in each tank with the grout mixture.

Grouting the tank waste in situ would result in a waste form with lower contaminant leachability compared to drying the waste and filling the tanks with gravel. However, if the rate of water infiltration to the waste form is controlled by using an effective surface barrier, the infiltration rate becomes the controlling factor in contaminant flux. Thus, the difference in performance of the in situ grout waste form in the In Situ Fill and Cap alternative is expected to be minor when a Hanford Barrier is used.

The impacts associated with implementing the in situ grout technology would be bounded by the impacts associated with the In Situ Vitrification alternative and the In Situ Fill and Cap alternative. In situ grouting would require a TFCF during operations, which would greatly increase the capital cost requirements and construction personnel levels over levels estimated for the In Situ Fill and Cap alternative. The capital costs and construction staffing requirements would approach those estimated for the In Situ Vitrification alternative. In addition, there would be an increase in offsite transportation associated with in situ grouting to bring the grout forming materials onsite.

In Situ Vitrification of Individual Tanks

The In Situ Vitrification alternative is based on the assumption that during operations, because of overlapping melt regions, the entire tank farm would be vitrified. In situ vitrification is a technology that could be applied to vitrify individual tanks or selected areas within the tank farms. Because the molten region would expand during vitrification, some overlapping of vitrified areas between tanks would be expected.

Minimal impacts would be associated with vitrifying the individual tanks and minimizing the vitrified region between the tanks as opposed to vitrifying the entire tank farm area. Using alternative confinement concepts that provided confinement and off-gas collection for an individual tank may reduce the construction and resource impacts compared to those associated with building a TFCF over each tank farm. This technology is not mature enough to accurately define the limits of the vitrified zone.

Use of Previously Contaminated Materials

To assess impacts and estimating costs, this EIS has assumed that all fill and borrow material is uncontaminated. However, it may be possible to use slightly contaminated material for glass formers in the in situ vitrification process. This alternate material must be characterized so that it would be added in the correct proportions and potential exposures would be within Site and DOE limits. The impacts associated with using previously contaminated materials would include a slight increase in groundwater contamination and potentially higher costs associated with added characterization and personnel protection. The amount of LAW from other areas would be reduced.

B.9.2 WASTE RETRIEVAL AND TRANSFER TECHNOLOGIES

The function of waste retrieval and transfer technologies is to remove the waste from the tank and transfer the waste to a treatment facility. Waste retrieval and transfer technologies are applicable to all ex situ alternatives where waste treatment will occur outside of existing storage tanks.

Retrieval Criteria

The current waste retrieval criteria is assumed to be capable of removing 99 percent of the existing waste volume from each tank during retrieval operations. This assumption is based on judgement and waste retrieval operations performed at the Hanford Site in the past. The current physical form of the waste stored in some of the SSTs appears to have dried and aged to the point that waste retrieval assumptions based on past practices may not be valid, and the criteria of 99 percent waste retrieval from each tank may be impractical or impossible using current retrieval concepts.

Retrieving 99 percent of the tank waste would leave a residual waste inventory of 1 percent in each tank. This 1 percent residual would be treated as a source of contamination that would, after a long period of time, migrate out of the tanks and become available for transport through the vadose zone. The rate of migration and transport of the contaminants would be highly dependent on the rate at which water infiltrates the residual waste, which would be controlled by installing a Hanford Barrier over the tank farms following retrieval.

Retrieving less than 99 percent of the tank waste would result in a larger residual inventory being left in the tanks for disposal. In turn, this large tank waste residual inventory would result in increased levels of long-term risk associated with the release and migration of contaminants associated with the larger residual inventory.

Retrieval Using Alkali Solutions

Retrieving alkali soluble residuals is a technology that could be used during retrieval operations for any of the ex situ alternatives. Retrieving alkali soluble residuals would involve washing the tanks with an alkali (sodium hydroxide) solution to remove the alkali soluble portion of the remaining waste solids for additional processing. Retrieving alkali soluble waste could allow increased retrieval for certain types of tank waste. The impacts of using this technology would be increased chemical additions to the waste inventory and potentially lower residual waste inventory left in the tanks following retrieval.

Retrieval Using Acid Solutions

Dissolving tank residuals in acid is a technology that could be used during the retrieval operations for any of the ex situ alternatives. This technology could be used to dissolve hardened sludges and waste that could not otherwise be retrieved, which would help achieve a specific retrieval criteria.

The dissolving action of the acid on the residual waste would also act on the interior of the tank and could open or enlarge an existing leak path. This technology would be most applicable to DSTs because the outer tank shell would contain any leakage developed by the inner shell. Implementing this technology would require controls to minimize the potential for increased tank leakage. The impacts of using this technology would be increased chemical additions to the waste inventory and potentially lower residual waste inventory left in the tanks following retrieval.

Tank Waste Retrieval Technologies

Many different technologies to retrieve the tank waste have been identified and evaluated (Boomer et al. 1993). The function of a retrieval technology is to remove the waste from the underground storage tanks in a safe, effective, and efficient manner that meets a defined retrieval criteria for the volume of waste retrieved. Retrieval technologies that have been identified and could be used to retrieve tank waste during any of the ex situ alternatives include.

- Mechanical retrieval would use a mechanical device like a back-hoe bucket or skip hoist to mobilize the waste and remove it from the tank. Mechanical retrieval would require an arm-based maneuvering device that would permit remote operation of the retrieval system.
- The Houdini™ waste retrieval system is a small, remotely-controlled robotic crawler type vehicle that is being evaluated at other DOE sites for waste retrieval operations. This type of technology could be selectively applied following other retrieval technologies to achieve retrieval criteria. The Houdini™ system being developed would collapse to fit through existing tank openings and would have mechanical attachments that would be used to break up and mobilize waste.
- Pneumatic retrieval is similar to hydraulic retrieval methods except that air would be used to move the waste as opposed to liquid.

Subsurface Barriers

Subsurface barrier technology could be used during retrieval operations for any of the ex situ alternatives. Subsurface barriers are most suitable for use in conjunction with hydraulic retrieval technologies, which have a higher potential for SST leakage. Subsurface barriers would not stop a leak but would provide containment to control the migration of tank leakage. Subsurface barriers are impermeable layers that would be installed in the soil surrounding a tank to contain any leakage that might occur during waste retrieval operations. The possibility of using subsurface barriers derived from concerns about using hydraulic sluicing for retrieval, and because some of the SSTs are either confirmed or assumed leakers. The function of the subsurface barriers would be to prevent tank leakage from migrating beyond the barrier into the vadose zone. This would help leak cleanup by minimizing the volume of contaminated soil.

A study titled Feasibility Study of Tank Leakage Mitigation Using Subsurface Barriers (Treat et al. 1995) has been completed in support of Tri-Party Agreement Milestones M-45-07A (Ecology et al. 1994). This feasibility study assessed:

- The potential environmental impacts of waste storage and retrieval activities without the application of subsurface barriers;
- Functional requirements of subsurface barriers to minimize the impacts associated with waste storage and retrieval activities; and
- The application of existing subsurface barrier technologies and the potential of existing technologies to meet functional requirements for SST waste storage and retrieval activities.

Fourteen different tank waste retrieval alternatives were analyzed in the feasibility study. The alternatives ranged from a No Action alternative, in which none of the waste was retrieved, to clean closure, where waste retrieval activities were assumed to remove 100 percent of the tank waste. The alternatives analyzed represented combinations of technologies for waste retrieval, subsurface barrier containment, tank stabilization, and surface barriers. The 14 alternatives analyzed included 8 alternatives with subsurface barriers and 6 alternatives without subsurface barriers.

The following subsurface barrier technologies were screened in the feasibility study as potential technologies that could be used for subsurface barriers:

- Chemical jet grout encapsulation;
- Freeze walls;
- Jet grout curtains;
- Permeation chemical grouting;
- Wax emulsion permeation grouting;
- Silica, silicate permeation grouting;
- Polymer permeation grouting;
- Formed-in-place horizontal grout barriers;
- Circulating air barriers;
- Radio-frequency desiccating subsurface barriers;
- Sheet metal piling subsurface barriers;
- Close-coupled injected chemical barriers;
- Induced liquefaction barriers;
- Slurry walls;
- Deep soil mixing;
- Soil fracturing longwall mining;

- Modified sulfur cement;
- Sequestering agents;
- Reactive barriers;
- Impermeable coatings;
- Microtunneling;
- In situ vitrification; and
- Soil saw (uses reciprocating high-pressure jets of grit or bentonite to create a vertical barrier).

Screening of the potential technologies resulted in selecting the following five barrier technologies for detailed analysis:

- Close-coupled injected chemical barrier. This would involve injecting chemicals (e.g., portland cement) directly adjacent to the tank sides and bottom. The term close coupled indicates that the barrier would be right next to the tank walls;
- Box-shaped chemical wall. A low-permeability basin would be formed beneath the level of existing soil contamination. This is a stand off type of barrier in which the bottom of the barrier would be sloped to a low point to help collect tank leaks. The barrier would be constructed of a low-permeability material such as portland cement;
- V-shaped chemical barrier. This stand off type of barrier would use angle drilling techniques to construct a V-shaped barrier that would start at the surface on each side of a tank farm and angle down to meet in the middle. The slope of the angled barrier walls would facilitate liquid collection and removal;
- Freeze wall. The V-shaped freeze wall would be similar to the V-shaped chemical barrier except that ice would be used instead of chemicals to create the barrier; and
- Circulating air barrier. The circulating air barrier would rely on water evaporating from the soil, limiting the ability of a leak to migrate through the vadose zone.

A comparative risk assessment and cost estimate was made for each of the alternatives evaluated in the feasibility study. This analysis provided an evaluation of the impacts of waste storage and retrieval with and without the use of subsurface barriers. The following conclusions were drawn from the subsurface barrier study.

- All functional requirements can potentially be satisfied using any of the subsurface barrier options evaluated. This conclusion is clarified with the observations that 1) little data on the performance of subsurface barriers exist; and 2) the draft functional requirements are largely and appropriately qualitative at this early state of development.
- Using any of the subsurface barrier concepts in general applications to tank farms would result in relatively small incremental reductions in the risk level achievable using baseline retrieval technologies (traditional sluicing, empty tank stabilization, and surface barriers).

- The cost-effectiveness of the subsurface barriers, calculated by the method most favorable to subsurface barriers, is about 1E-04 times that of surface barriers, and 1E-02 times that of the set of baseline technologies. Uncertainty in the performance of subsurface barriers is high, but because the impact of subsurface barriers on risk and cost-effectiveness is low, even the best-case assumptions of subsurface barrier performance have a relatively small effect on overall risk and cost-effectiveness of SST disposal options.

Waste Transfer Technologies

The function of waste transfer technologies in each of the ex situ alternatives would transport the waste as it was retrieved from the tanks to a nearby processing facility. The method of waste transfer would be through a pipeline. An alternate transfer technology would be containerized waste transfer. Containerized transfer of the waste would involve placing the waste into a container as it came from the retrieval system and transporting the containers to the waste treatment facility. Containerized waste transfer is better suited to mechanical and pneumatic transfer methods than hydraulic retrieval methods. Containerized transfer would avoid the potential mixing of incompatible tank waste. The impact of containerized waste transfer between the tanks and the treatment facility would include:

- Increased radiological exposure;
- Increased onsite transportation; and
- No construction of the waste retrieval annexes described in the ex situ alternatives.

Truck Transfer

Truck transfer of waste using a modified tanker trailer truck or an LR-56(H) truck (specially designed vehicle for onsite transfers) is a technology that could be used as an alternative to the transfer of waste through pipelines. It could also be used to support various characterization activities and pretreatment/treatment activities. This waste transfer technology would use trucks to transport liquid waste between permanent or portable loading facilities. Waste transfer using trucks is better suited to limited waste volumes and intermittent transfers.

Truck transfer of waste was evaluated in the SIS EIS (DOE 1995i) as an alternative to constructing a replacement cross-site transfer system to transfer waste from 200 West to 200 East Area. A modified tanker trailer with a capacity of 19,000 L (5,000 gal) and the LR-56(H) with a capacity of 3,800 L (1,000 gal) were evaluated as options to pipeline transfer for an estimated 2E+07 L (5E+06 gal) of waste from the 200 West Area to the 200 East Area.

The analysis performed for the SIS EIS concluded that the environmental impacts associated with truck transfer of waste were not appreciably different from those associated with pipeline transfer except in the area of worker exposure. Worker exposure would be higher due to increased exposure for the truck driver and the workers involved with load and unload facility operations.

Table B.9.2.1 summarizes the number of LR-56(H) truck trips estimated to transfer waste from T Plant and PFP. The number of trips associated with using the modified tanker trailer would be fewer because of the larger capacity. These estimates were developed using the T Plant and PFP waste volume projections.

Table B.9.2.1 Estimated Truck Trips Required for T Plant and PFP Waste Transfers

Source (Year)	Waste Volume	Number of Trips
PFP (FY 1995 thru 2006)	265,000 L (70,000 gal)	70
T Plant (1996)	97,000 L (25,700 gal)	26
T Plant (1997)	97,000 L (25,700 gal)	26
T Plant (1998)	193,000 L (51,000 gal)	51
T Plant (1999)	291,000 L (77,000 gal)	77
T Plant (2000)	390,000 L (103,000 gal)	103
T Plant (2001)	488,000 L (129,000 gal)	129
T Plant (2002)	583,000 L (154,000 gal)	154
T Plant (2003 to 2023) [1]	681,000 L/yr (180,000 gal/yr)	3,600

Notes:

[1] T Plant is used as a decontamination facility and would be used throughout the continued operations period for any of the alternatives. The year 2023 was selected as a representative date for evaluation purposes only.

The impacts from the transfer of the projected PFP and T Plant waste were estimated to be similar to the impacts associated with implementing the replacement of transfer lines. Implementing truck transfer to transport waste from T Plant and PFP to the DSTs in the 200 East Area would require constructing or upgrading loading facilities and improving Site roads to accommodate the trucks. The worker exposure associated with truck transfer of the waste would be higher than the exposure associated with pipeline transfer of the same waste.

LR-56(H) Truck for Transporting Liquid Radioactive Waste

The LR-56(H) truck is a specifically designed vehicle for transporting liquid radioactive waste between areas on the Hanford Site. The vehicle is designed to U.S. Department of Transportation standards and regulatory standards specific to the Hanford Site. The design includes lead shielding around a tank (capacity approximately 3,800 L [1,000 gal]) with redundant level and temperature monitors, alarms, and pumps for waste transfer. The truck can use either portable or permanent waste loading facilities at the point of origin and at the destination point.

Liquid waste could be transferred from such locations as PFP, T Plant, the 300 Area facilities, 100 Area, and the 400 Area to waste processing facilities or to the DST system. Other uses of the truck to transfer liquid waste could include transferring the following waste into the TWRS management system:

- 100 Area cleanout waste from the 100 Area facilities;
- 300 Area fuel supply cleanout, waste from the 340 Building, and other 300 Area facilities;
- Miscellaneous transfers within the 200 Areas where pipeline transfer would not be an option due to failure, nonexistence, or lack of compliance status of existing lines;
- MUSTs cleanout across the Hanford Site; and
- Accumulations of contaminated rainwater (not greater in activity than HLW contained in DSTs/SSTs) from areas such as diversion boxes or tank vaults as needed to prevent spillage or leakage to ground.

B.9.3 EX SITU WASTE TREATMENT TECHNOLOGIES

Ceramic Waste Forms

Ceramic materials encompass a broad group of nonmetallic, inorganic solids with a wide range of compositions and properties. Their structure may be either crystalline or glassy. The ceramic form is often achieved by high-temperature treatment (burning or firing). Ceramics are stable, durable, and considered very leach resistant. Ceramics could be used in place of vitrified glass as an immobilization treatment for either HLW or LAW in any of the ex situ alternatives.

Immobilizing the tank waste using ceramic technologies would involve 1) retrieving the waste from the tanks; 2) potentially separating the waste into HLW and LAW components; and 3) performing waste pretreatment, which could include calcining, adding ceramic formers, and thermally treating in the range of 1,200 °C (2,200 °F) to obtain the desired properties.

Tailored ceramics have been identified and evaluated for immobilization of tank waste. Tailored ceramics refer to a mixture of different types of ceramic formers developed to immobilize a waste stream. Each of the different types of formers used would have the ability to chemically bind a specific waste element. Additional strength and chemical durability can be designed into the waste form when adding an excess of the tailoring species.

The ceramic form evaluated for immobilizing HLW was an aluminosilicate compound, Synroc D, which consists of zirconolite, perovskite, spinel, and nepheline. Sodium would be immobilized in this compound as nepheline. The theoretical sodium oxide loading based on all formulation assumptions would be 22 weight percent. For application at the Hanford Site, the ceramic form assumed to be produced would consist of nepheline, monazite, and corundum.

Ceramics could be formed into different physical forms including monoliths or pellets. Pellets could be manufactured in a continuously vertical shaft kiln while the ceramic monoliths would require a hot isostatic pressing operation to form the ceramic. Hot isostatic pressing is a commercial process in which the canister containing the waste and ceramic formers is evacuated and placed in a vessel that is pressurized between 15 to 70 MPa (2,000 to 10,000 psi) at a temperature of approximately 1,200 °C

(2,200 °F). With similar waste loadings, the hot isostatic pressed ceramic technology and the vitrification technology would yield similar volumes of waste for disposal.

The impacts of using ceramic-forming technologies to process the tank waste would be approximately the same as those impacts associated with vitrifying the tank waste. Both technologies are ex situ waste treatments used to immobilize the waste. Ceramic technologies would require the following facilities to process the waste:

- Retrieval and transfer systems;
- Separations facilities if required;
- Waste processing facilities;
- Interim storage facilities for HLW; and
- Disposal facilities for LAW.

Vitrification Technologies

Vitrification is a molten glass process in which the waste would be combined with glass-formers and heated to glass-forming temperatures. The melter is the piece of equipment that would take the waste material and glass-formers, heat the feed material to a glass-forming temperature of approximately 1,200 °C (2,200 °F) where chemical and organic destruction occurs, and output a molten glass product containing the waste.

Vitrification melters vary by their methods of heating the waste, feeding the waste, and the glass product produced. In addition, glass melters can operate in a batch or continuous mode. Some of the melter types identified for potential application to waste vitrification include the following:

- Joule-heated ceramic lined melters;
- Induction melters;
- Microwave melters;
- Plasma-arc melters;
- Transferred plasma melters;
- Fuel-fired melters; and
- Cold-crucible melters that use a cooled-glass skull on the melter walls to prolong melter operating life.

Melters that require a dry waste feed stream would require calcining before being fed. The calcining step would remove excess water, destroy some of the chemical compounds, and convert the major constituent in the feed (i.e., sodium nitrate) into an oxide or a carbonate.

The French have developed and operated vitrification processes using a rotary calcine and metal melter to vitrify waste that resulted from reprocessing spent nuclear fuel from light-water reactors.
This process calcines the acidic waste and continuously feeds an induction-heated metal susceptor and crucible. The borosilicate glass product formed is then poured into canisters approximately 1.3 m (4.2 ft) high and 0.43 m (1.4 ft) in diameter (DOE 1990).

The process developed for waste vitrification at the West Valley Demonstration Project in New York State and at the DOE Savannah River Site in South Carolina is the liquid-fed ceramic-lined melter. The liquid-fed ceramic-lined melter is a joule-heated melter developed from commercial ceramic-lined melters for use in vitrifying defense waste (DOE 1990).

The impacts associated with selecting a different melter type for the ex situ vitrification alternatives would involve potential changes in volume, composition, and treatment for the melter off-gas, changes in the resources required to fire the melter, and possible facility impacts required to accommodate the space requirements for the melter and off-gas equipment. For example, fuel-fired melters would generate a larger volume of off-gas than other melter types. This larger off-gas volume would require larger treatment equipment in the off-gas train for emissions control. One potential benefit of using a fuel-fired melter would be the higher throughputs that could be achieved. Some melter types might not be suitable for scaling up to high capacity and would require multiple melters operating in parallel to achieve high capacity production rates, which may increase the size of the facility.

Calcination Technologies

Calcination is the process of removing water and heating the waste to a temperature sufficiently elevated to decompose some of the chemical compounds such as hydroxides or nitrates. Calcination differs from vitrification in that calcination temperatures would not necessarily cause the reacting materials to melt and form a glass. The calciner is the piece of equipment that would heat the feed material to a calcination temperature of approximately 700 °C (1,300 °F) where the chemical and organic destruction occurs and output a solid waste product.

Calciners can vary by their methods of heating and feeding the waste, and the solid characteristics of the waste produced. Some of the calciner types identified for potential application to waste calcination include the following:

- Spray calciners;
- Rotary calciners;
- Fluid bed calciners;
- Indirect fired calciners; and
- Electrically heated calciners.

The impacts associated with selecting a different calciner type for the calcination alternative would involve potential changes in: volume, composition, and treatment for the calciner off-gas; changes to or elimination of the compaction step required for the solid produced; changes in the resources required to fire the calciner; and possible facility modifications required to accommodate the space requirements for the calciner, compactor, and off-gas equipment.

Alternate Glass Compositions

Borosilicate glass is based on a composition of silicon dioxide, boron trioxide, sodium oxide, and lithium oxide. Borosilicate glass has been chosen by most countries as the standard final waste form for either HLW or LAW disposal. For possible use at the Hanford Site, borosilicate glass was chosen over other waste forms for its durability, ability to accommodate a varied range of waste feeds, and its adaptability for radioactive waste processing at an industrial level (DOE 1990).

Other types of glass, including the soda lime glass that would be produced by the Ex Situ No Separations alternative, could be selected as glass types for the final waste form for vitrified tank waste. The type of glass selected for use in the vitrification process is controlled by the types and proportions of glass formers used. The driving factors for selecting a glass type include waste loading, leachability, processability, and waste acceptance criteria at the potential geologic repository.

The impacts associated with changing the composition of glass produced in the vitrification process would be minimal for any of the ex situ vitrification alternatives provided the waste loading remained approximately the same. The glass waste loading limitations control the volume of final waste product requiring disposal. This in turn could have substantial impacts associated with transportation of the glass and charges assessed by the repository.

Separations Technologies

Separations refers to a broad range of technologies for removing or separating selected chemical constituents from other constituents. Application of separations processes would typically be designed to remove specific constituents from material flow streams within a processing plant and could be carried out in either a continuous or batch process. These processes fall into the general categories of chemical, physical, or a combination of chemical and physical.

New separations processes that show potential benefits in the areas of improved separations efficiencies, economic benefits, reduced secondary waste generation, superior performance, or environmental impacts are continually being identified and developed for potential application. One example is the application of amorphous silica gels that can be tailored to sequester selected elements at a specific pH.

The process described for the Ex Situ Extensive Separations alternative contains many but not all of the concepts that potentially could be used to extract specific components from the waste. Other concepts have been proposed that would potentially enhance the separation of other HLW components. However, adding other processes to the flowsheet would have a negligible effect on the impacts of this alternative. The quantity of HLW sent to the repository would not be materially decreased.

Off-Gas Treatment Technologies for Radionuclides

The design of off-gas treatment systems for each alternative would ensure that emissions of radionuclides would be below regulatory limits. For the In Situ Vitrification alternative, the probability of a cancer fatality to the maximally-exposed individual in the general public from exposure to routine off-gas emissions would be 1.6E-11. For the Ex Situ Intermediate Separations alternative, the probability of a cancer fatality to the maximally-exposed individual in the general public from exposure to routine off-gas emissions would be 3.3E-06. Volume Three, Appendix D of the EIS provides further discussion of the risk associated with each alternative. Should it be determined that radionuclide emissions from the stack gases were to be reduced to levels more restrictive than current regulations, specific treatment technologies would be examined on a case-by-case basis.

The I-129 in the tank waste would be volatilized as I_2 during thermal treatment processes. Gaseous iodine would not be captured using traditional HEPA filtration. Two technologies that could be used to capture gaseous iodine would be adsorption on activated carbon and reaction with silver to form silver iodide. Recovering iodine in minute amounts is expected to be inefficient.

The control of C-14 emissions from any of the thermal treatment processes would be difficult. During vitrification the C-14 would be oxidized to CO_2 along with all other nonradioactive carbon in the waste stream. The CO_2 containing the C-14 would make up a small percentage of the total CO_2 in the off-gas stream. However, any treatment technology used to capture the C-14 would have to capture all of the CO_2. This potentially could be done by passing the off-gas through a recovery system in which CO_2 is precipitated as calcium carbonate via reaction with a lime scrubbing solution. This process would generate a substantial secondary waste stream that would require further processing and disposal.

For the Ex Situ No Separations (Calcination) alternative process, the majority of C-14 present would be incorporated into the waste product in the form of solid carbonate salts. Only a small percentage of C-14 would be released as CO_2 gas.

Grouting of Retrieved Tank Waste

Grouting of the retrieved tank waste is a technology that could be applied to any of the ex situ alternatives. As previously described, grout is a common solidification and stabilization technology employed in the management of hazardous waste. Grout is a general term that refers to a waste form obtained by mixing waste with chemical additives to stabilize and immobilize the hazardous constituents. The grouting process applied to the ex situ treatment of the tank waste would involve waste retrieval and transfer to a grout facility where the waste would be mixed with appropriate mixtures of grout formers. After the grout was mixed, it would be placed into containers for solidification and disposal.

Grouting of tank waste has been extensively studied at the Hanford Site for use as a technology for LAW disposal. Grouting of the LAW was selected as the LAW treatment method in the Hanford Defense Waste EIS (DOE 1987). The LAW described in the Hanford Defense Waste EIS included liquid waste from the tanks (after separation of HLW components) and secondary waste from the HLW vitrification facility, which would consist of waste from canister decontamination, drying of feed material, and off-gas treatment. As a result of a revised technical strategy and stakeholder input, grouting of LAW was replaced by vitrification of LAW as the proposed waste treatment technology. Even with this strategy, there still will be a requirement to grout the LAW generated as secondary process waste from the HLW vitrification facility and the additional LAW vitrification facility. However, this grouting facility would be greatly reduced in size.

The impacts associated with grouting the tank LAW for onsite disposal instead of vitrifying the LAW would include the following:

- Potentially increased volume of waste requiring disposal. The estimated volume of grouted LAW would be approximately three times the volume of vitrified LAW. This would increase the number of vaults and the permanent land use commitment for disposal vaults by 14 ha (35 ac);
- Increased contaminant flux out of the waste form during groundwater leaching because of a higher leachability of grout compared to glass. This would result in some increase in the long-term risk. Leachability and long-term impacts could be reduced by additional treatment such as calcination before grouting. However, calcination of the LAW would be necessary, which would result in emissions and short-term risk approximately equal to vitrification; and
- Reduced complexity of the processing facility resulting in potential reduced capital cost requirements and reduced resource requirements. A grout facility (transportable grout facility) was constructed and operated in the 200 East Area in the late 1980's. It is currently in standby and could be restarted, which would avoid some capital cost. Capacity of the plant is about 500 tons per day.

Low-Activity Waste Disposal Technologies

There are a number of disposal technologies being used or developed for LAW. These technologies use a multiple barrier system, which include the solidified LAW form itself as well as primary and secondary containment methods for the solidified LAW.

The primary containment for the solidified LAW form could be metal, concrete, or a hybrid fiber-reinforced concrete. These containers, which would be made in various shapes and sizes, are commonly referred to as drums, canisters, or containers. The primary container would be placed in a belowgrade or abovegrade secondary containment vault constructed of concrete and/or an engineered soil structure. Alternately, the vaults would be the primary and only containment for the solidified LAW.

The most important protection against releases of contaminants after disposal in a multiple barrier system is considered to be the solidified waste form itself. Because complete isolation by land disposal is difficult, the practicality of minimizing releases through improved waste forms is now recognized as both desirable and necessary. The primary function of a waste form is the retention of its hazardous and radioactive components. Also important is its structural stability for handling, transportation, storage, and disposal. Numerous materials are being used or developed for the solidification of LAW. A short description of the main categories of these materials is given as follows. Some of the following categories (e.g., a modified sulfur cement to bond a LAW glass cullet) can be combined.

- Hydraulic cements are binders that harden by chemical reactions with water. The major types of cement of interest to waste immobilization are portland, blast furnace slag, pozzolanic, aluminous, and masonry.
- Modified sulfur cement is a recently developed material that is commercially produced in the United States. The basic raw material is elemental sulfur reacted with a small percentage (5 percent) of polymer to improve physical properties. Sulfur cement is highly resistant to alkaline and acidic environments. Sulfur cement has been proposed as a waste form matrix for vitrified LAW cullet in previous engineering studies and in the engineering data packages developed for this EIS. The stability of sulfur as a matrix has not been demonstrated. The reaction of modifiers with sulfur to form a linear polymer is exothermic and requires 24 hours to complete (Boomer et al. 1993). Further investigation would be required during the design phase to determine the viability of the cullet in sulfur waste form.
- Glasses are high-melting-point materials, generally inorganic oxides, which on cooling, form an amorphous structure. For solidification, waste solids are generally incorporated into the glass structure as oxides produced during the high-temperature (1,200 °C [2,200 °F]) processing conditions.
- Organic polymers consist of large molecules built up by the repetition of small simple chemical units. Although there are a large number of polymeric materials suggested for the solidification of LAWs, the most prominent systems are epoxies, polyethylene, and unsaturated polyesters.
- Asphalt (or bitumen) is a complex mixture of high-molecular-weight hydrocarbons containing both aliphatic and aromatic constituents. Waste solids are mixed in and coated with liquid asphalt and mechanically held in a solid asphalt matrix after cooling.
- Ceramics encompass a broad group of nonmetallic, inorganic solids with a range of compositions and properties. Waste forms can be crystallized, glass, or chemically-bonded ceramics.

Future evaluations of LAW disposal technologies may result in the selection of other solidified LAW forms or primary/secondary containment methods.

B.10.0 HIGH-LEVEL WASTE DISPOSAL AT THE POTENTIAL GEOLOGIC REPOSITORY

B.10.1 WASTE ACCEPTANCE

The Nuclear Waste Policy Act of 1982 established a national policy for disposal of HLW and commercial spent nuclear fuel in a geologic repository and required the President to evaluate the use of commercial repository capacity for the disposal of defense high-level nuclear waste. In February 1985, the Secretary of Energy submitted a memorandum to the President recommending that DOE proceed with plans and actions to dispose of defense waste in a commercial repository. In an April 1985 Presidential Memorandum, the President approved proceeding on the basis of the recommendation. Subsequently, in September 1988, DOE issued DOE Order 5820.2A, which stated requirements to process and dispose of DOE's new and readily retrievable HLW in a potential geologic repository and to consider options such as in-place stabilization or retrieval, processing and disposal in a potential geologic repository for permanent disposal of a singly contained tank waste.

The Nuclear Waste Policy Act Amendments of 1987 ordered termination of activities for all potential geologic repository candidate sites other than Yucca Mountain site and required that the Secretary of Energy report to the President and Congress between January 1, 2007 and January 1, 2010 on the need for a second repository. The Nuclear Waste Policy Act prohibits emplacement in the first repository of a quantity of spent fuel containing in excess of 70,000 mt (77,000 tons) of heavy metal of a quantity of solidified high-level radioactive waste resulting from the reprocessing of spent nuclear fuel until such a time as a second repository is in operation. It is recognized that current projections for spent nuclear fuel and HLW exceed the 70,000-mt (77,000-ton) limit for the first repository; however, this issue will not be resolved prior to addressing the need for a second repository. The EIS is based on the assumption that the potential geologic repository or a second repository could accommodate all of the HLW produced by any of the alternatives. Therefore, the current planning basis for disposal of DOE's new or readily retrievable HLW is for disposal at a geologic repository, which may be Yucca Mountain should that site be shown to be acceptable and approved as a potential geologic repository.

In support of the potential first geologic repository, DOE has issued a Waste Acceptance Systems Requirements Document (DOE 1994g) describing functions and technical requirements for a system that would accept HLW and spent nuclear fuel into the Civilian Radioactive Waste Management System. The Waste Acceptance Systems Requirements Document sets forth the criteria established for waste forms reviewed and judged acceptable for disposal. All radioactive waste (both spent nuclear fuel and HLW) that would be accepted into the Civilian Radioactive Waste Management System would be required to meet either existing waste acceptance criteria or waste acceptance criteria developed for a specific waste form. The current waste form acceptance criteria include the following requirements:

- Radioactive waste shall be in solid form;
- Particulate waste forms shall be consolidated (e.g., by incorporating the waste into an encapsulating matrix) to limit the availability and generation of particulates;
- Combustible radioactive waste shall be reduced to noncombustible form unless it can be demonstrated that a fire involving the waste packages containing combustibles will not adversely affect other waste packages, any structures, systems, and components important to safety, or the repository's ability for waste isolation.

The three criteria previously listed are in response to requirements of 10 CFR 60.135(c).

- The waste form shall not contribute to free liquid in the waste packages to an amount that could compromise the ability of the waste package to achieve the performance objectives related to containment of the waste form or result in spillage and spread of contamination in the event of waste package perforation during the period through permanent closure. This criterion is in response to the requirements of 10 CFR 60.135(b) (2).

- The waste form shall not contain explosive, pyrophoric, or chemically reactive materials in an amount that could compromise the repository's ability for waste isolation or the repository's ability to satisfy the performance objective. This criterion is in response to the requirements of 10 CFR 60.135(b)(1).
- The waste shall not exceed the repository limit for defense waste in terms of metric tons of equivalent heavy metal (DOE 1995q).

Establishing acceptance criteria for other HLW products (waste form plus the packaging system) would involve identification of the candidate waste product. The candidate waste form product would then be judged for acceptability. If the candidate waste form product is judged to be an acceptable candidate for repository disposal, waste acceptance criteria would be established. All HLW waste sent to the repository would meet a set of waste acceptance requirements defined for that product.

At present, the Waste Acceptance Systems Requirement Document (DOE 1995q) assumes that the standard HLW form to be accepted will be vitrified borosilicate glass. The borosilicate glass is to be sealed inside an austenitic stainless-steel canister. The assumption of the standard form is intended to provide guidance to proceed with waste acceptance activities. It is based on informed technical opinion, preliminary study results, and accumulated institutional experience. The standard form assumption is subject to further resolution in subsequent revisions of the Waste Acceptance Systems Requirements Document.

Throughout the TWRS EIS, Yucca Mountain is referred to as the potential geologic repository. Currently, Yucca Mountain is the only site being characterized as a geologic repository for HLW. If selected as the site for development, it would be ready to accept HLW no sooner than 2015. The potential environmental impacts that would occur at the geologic repository from the disposal of HLW from TWRS are not addressed in this EIS. Potential impacts at the repository are being addressed in an EIS that DOE will prepare to analyze the Site-specific environmental impacts from construction, operation, and eventual closure of a potential geologic repository for spent nuclear fuel and HLW at Yucca Mountain. Detailed evaluations to support decisions on the disposal of HLW from the Hanford Site would be made following the completion of the repository EIS. The repository EIS will also assess the impacts of transporting spent nuclear fuel and HLW from various storage locations to the potential geologic repository.

Each of the ex situ alternatives addressed in this EIS include sufficient interim onsite storage facilities to store all of the immobilized HLW produced while awaiting offsite transport and disposal at the potential geologic repository. This would allow each of the alternatives to operate independent of the acceptance schedule for the potential geologic repository. Schedules for shipping HLW to the potential geologic repository were developed for each alternative. The assumed shipment schedule would begin at approximately 2020. This is five years after the scheduled opening of the repository and would allow DOE to ship a backlog of HLW from other sites. To address concerns regarding the scheduled acceptance of the Hanford Site's HLW at the repository, the impacts of interim onsite storage have been assessed for a 50-year period.

The range in number of canisters that would be produced under the different alternatives varies widely based on the amount of separations and does not agree with the current technical planning basis for the geologic repository. The current geologic repository design is based on acceptance of approximately 7,100 standard sized canisters (1,800 HMPCs) of HLW from the Hanford Site. The number of canisters and waste packages that would be produced under the different alternatives is subject to change during waste package design and optimization. Using the larger canisters would reduce the number of waste packages requiring storage, transportation, and disposal at the potential geologic repository.

Subsequent to issuing the current Waste Acceptance Systems Requirements Document (DOE 1995q), DOE determined that the potential first geologic repository will accept only spent nuclear fuel and HLW that does not include components regulated as hazardous waste under RCRA. As most of the Hanford HLW contains hazardous or characteristic components, the HLW would have to be treated and/or delisted to be disposed of in the potential first geologic repository.

B.10.2 HIGH-LEVEL WASTE DISPOSAL COST

Repository fees for alternatives that include shipping HLW to the potential geologic repository are based on analysis performed by the Office of Civilian Radioactive Waste Management in support of the TWRS EIS (Milner 1996a). This analysis was performed using a consistent methodology as used by the Civilian Radioactive Waste Management Program in developing the Analysis of the Total System Life Cycle Cost (TSLCC) of the Civilian Radioactive Waste Management Program of September 1995 (DOE 1995u). Lifecycle cost estimates for four alternative scenarios were provided for disposal of vitrified HLW from the Hanford Site. The four alternatives varied the volume and HLW canister sizes from the 1995 TSLCC estimate basis. The analysis included estimates for two new HLW waste package designs, two new transportation cask designs, and estimates of changes to repository surface facilities, subsurface impacts, transportation, and other program cost elements. The analysis provided scoping level detail scaled from the detailed point estimate reported in the 1995 TSLCC analysis.

Estimates of the total defense share, based on application of the 1987 Federal Register methodology, were provided in the cost estimate report. Allocating the defense share between the Hanford Site and other defense sites was estimated by multiplying the defense share by the ratio of the number of Hanford Site waste packages to the total number of defense waste packages. A waste package consists of up to four canisters of HLW and is equivalent to an HMPC for Hanford Site waste from a numerical standpoint. Repository fees for alternatives that were not addressed in the Office of Civilian and Radioactive Waste Management report were estimated by extrapolating data from the estimate (Jacobs 1996). The estimated disposal fees for placement of HLW in the potential geologic repository are shown in Table B.10.2.1.

Table B.10.2.1 HLW Disposal Fees

Alternative	Canister Size	Number of Canisters	Number of HMPCs [1]	Hanford Site Disposal Fee [2]
Ex Situ Intermediate Separations	1.17 m^3 (41 ft^3)	12,200	3,050	$5,280
Ex Situ No Separations Vitrification	10 m^3 (360 ft^3)	29,100	29,100	$38,900
Ex Situ No Separations Calcination	10 m^3 (360 ft^3)	10,300	10,300	$14,630
Ex Situ Extensive Separations	1.2 m^3 (41 ft^3)	570	143	$270
Ex Situ/In Situ Combination 1	1.2 m^3 (41 ft^3)	8,500	2,130	$3,710
Ex Situ/In Situ Combination 2	1.2 m^3 (41 ft^3)	4,900	1,230	$2,150
Phased Implementation	1.2 m^3 (41 ft^3)	12,200	3,050	$5,280
Capsules - Overpack and Ship	1.2 m^3 (41 ft^3)	400	100	$124
Capsules - Vitrify with Tank Waste	1.2 m^3 (41 ft^3)	342	86	$111

Notes:

[1] HMPCs are the assumed packaging configuration for placement of canistered HLW in a geologic repository. One HMPC would be placed into a disposal waste package at the geologic repository and would contain four 1.17 m^3 (41 ft^3) canisters or one 10 m^3 (360 ft^3) canister.

[2] Cost in millions of 1995 dollars.

The 1995 TSLCC forms the baseline for comparing disposal cost between alternatives and for allocating the defense share for each alternative. The 1995 TSLCC assumed 2,465 waste packages of HLW from the Hanford Site would be disposed of with approximately 2,050 waste packages of HLW from other DOE sites and the West Valley Demonstration Project. These waste packages were assumed to be commingled with waste packages of commercial spent nuclear fuel containing approximately 84,000 mt of U. The 1995 TSLCC assumed disposal in a single repository, with the Yucca Mountain site in Nevada serving as a surrogate to allow estimation of TSLCC. The design concepts assume emplacement of waste packages containing HLW canisters in the spaces between commercial spent nuclear fuel packages in a special arrangement with a high thermal load.

The methodology collects direct cost, allocates certain indirect cost elements based on piece count and areal dispersion factors, and then assigns remaining cost based on factors derived from relative direct and allocated cost. Unassigned cost comprises a significant portion of the total system cost due to high development and evaluation cost compared to construction and operation.

A methodology to specifically evaluate extreme variations from the 1995 TSLCC basis of 2,465 waste packages from the Hanford Site was not developed. For these reasons, there is a higher level of uncertainty in the estimates for the Ex Situ Extensive Separations and Ex Situ No Separations alternatives.

All of the ex situ alternatives except Ex Situ No Separations (vitrification and calcination) were able to maintain 1995 TSLCC design assumptions for the repository thermal loading approach and emplacement of HLW waste packages in the space between hot spent nuclear fuel packages. The Ex Situ No Separations alternative would require an additional area that would have a low thermal loading, dedicated to excess Hanford Site HLW. The number of HLW packages produced by the Ex Situ No Separations alternative exceeds the number of available openings in the high thermal load repository.

Development, evaluation, and other program costs were evaluated and were assumed to be essentially constant for all ex situ alternatives except for Ex Situ No Separations. Significant increases in development, evaluation, and other program costs would occur for Ex Situ No Separations due to additional repository area and licensing and significant extension of waste acceptance and transportation operations.

The estimated repository fees are at a scoping level of detail, scaled from TSLCC data and estimated through use of TSLCC models. The estimates are consistent with the 1995 TSLCC. Results are not based on engineering studies of the specific alternatives and do not represent detailed point estimates. Changes in the repository system baseline will have system impact and will affect cost estimates.

B.11.0 ALTERNATIVES DATA

The following statistical section provides a direct comparison of the various alternatives. The data are grouped in the following categories and arranged so the alternatives can be compared:

- Schedule for each alternative (Table B.11.0.1);
- Cost summary for tank waste and capsule alternatives (Tables B.11.0.2 and B.11.0.3);
- Resource summary for tank waste and capsule alternatives (Tables B.11.0.4 and B.11.0.5);
- Radiological emissions summary (Table B.11.0.6);
- Nonradiological emissions summary (Table B.11.0.7); and
- Transportation summary for tank waste and capsule alternatives (Tables B.11.0.8 and B.11.0.9).

In addition to the tables, effort-power requirements in effort years are presented in graphs to provide a direct comparison of the alternatives (Figures B.11.0.1, B.11.0.2, and B.11.0.3). The graphs do not extend beyond the years 2040, after which staffing requirements are minimal.

Figure B.11.0.1 Staffing Levels, Tank Waste Ex Situ Alternatives

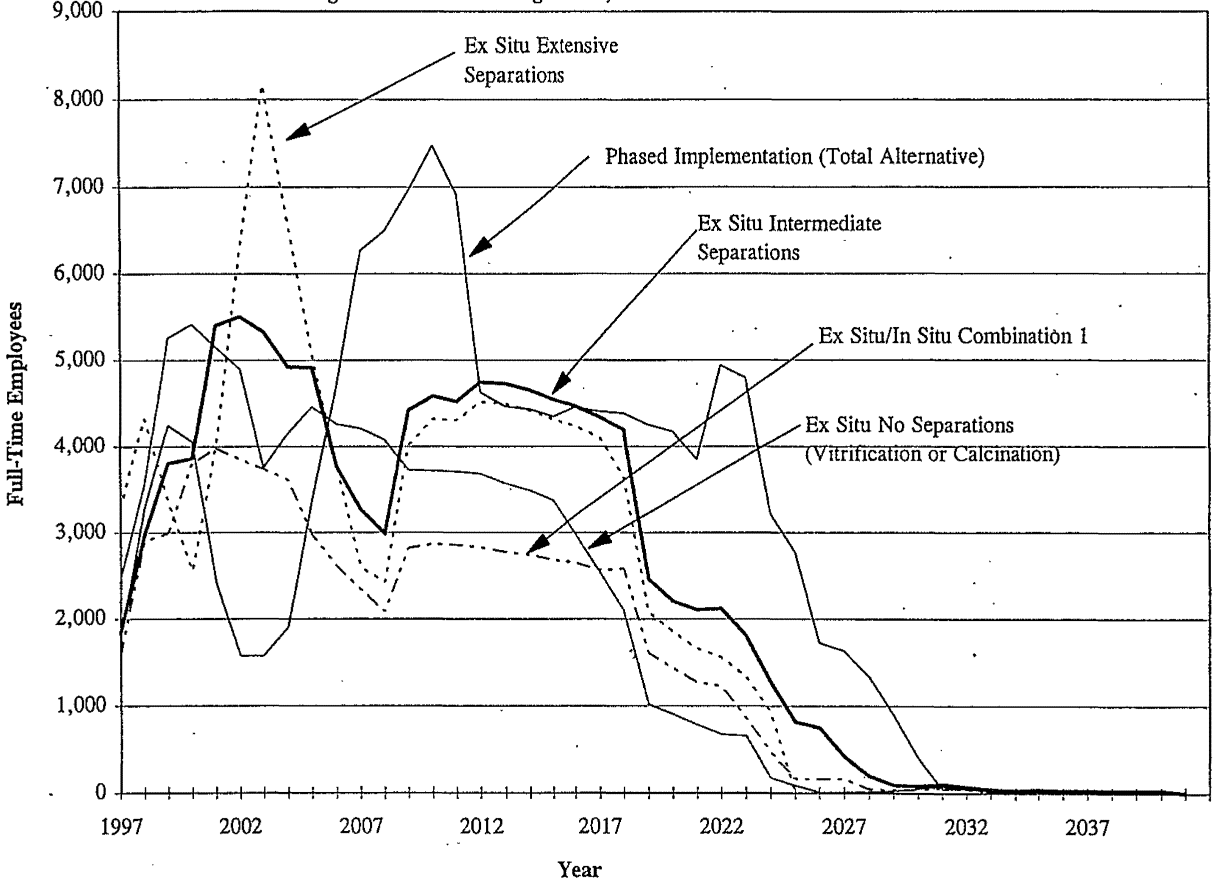


Figure B.11.0.2 Staffing Levels, Tank Waste In Situ and No Action Alternatives

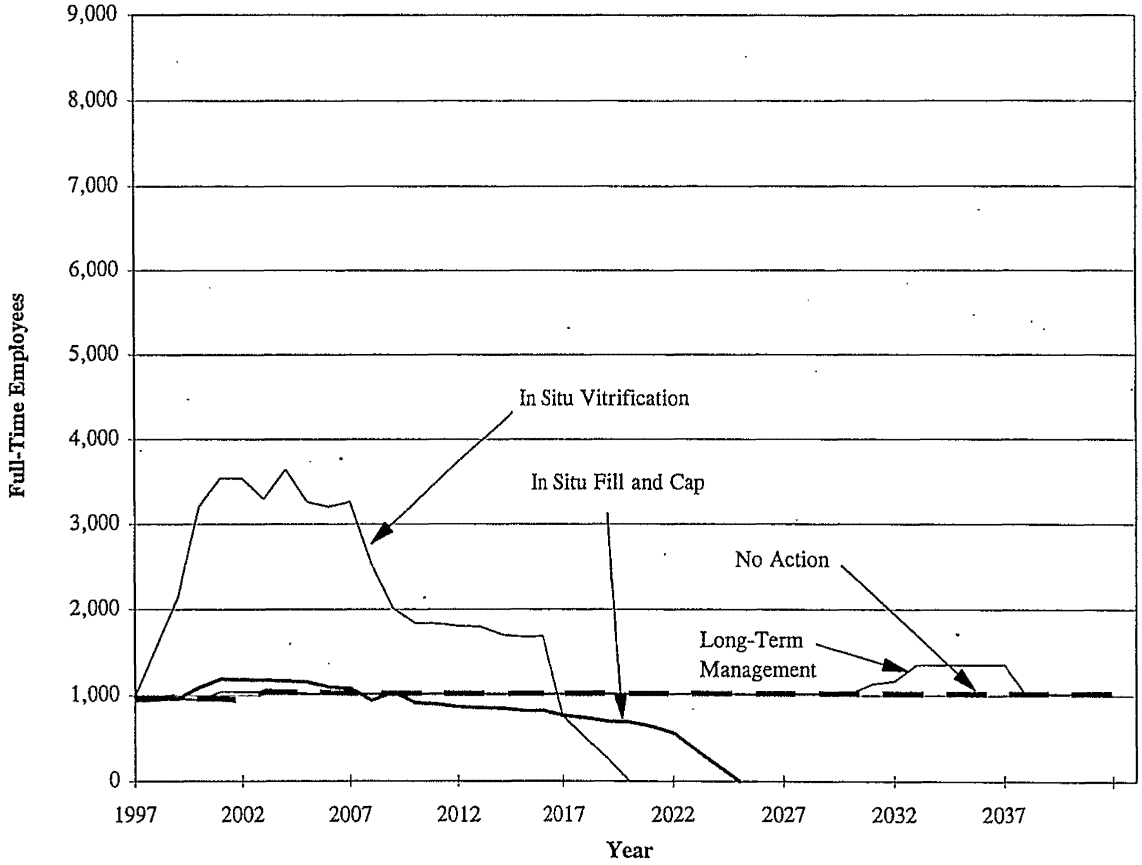


Figure B.11.0.3 Staffing Levels, Capsule Alternatives

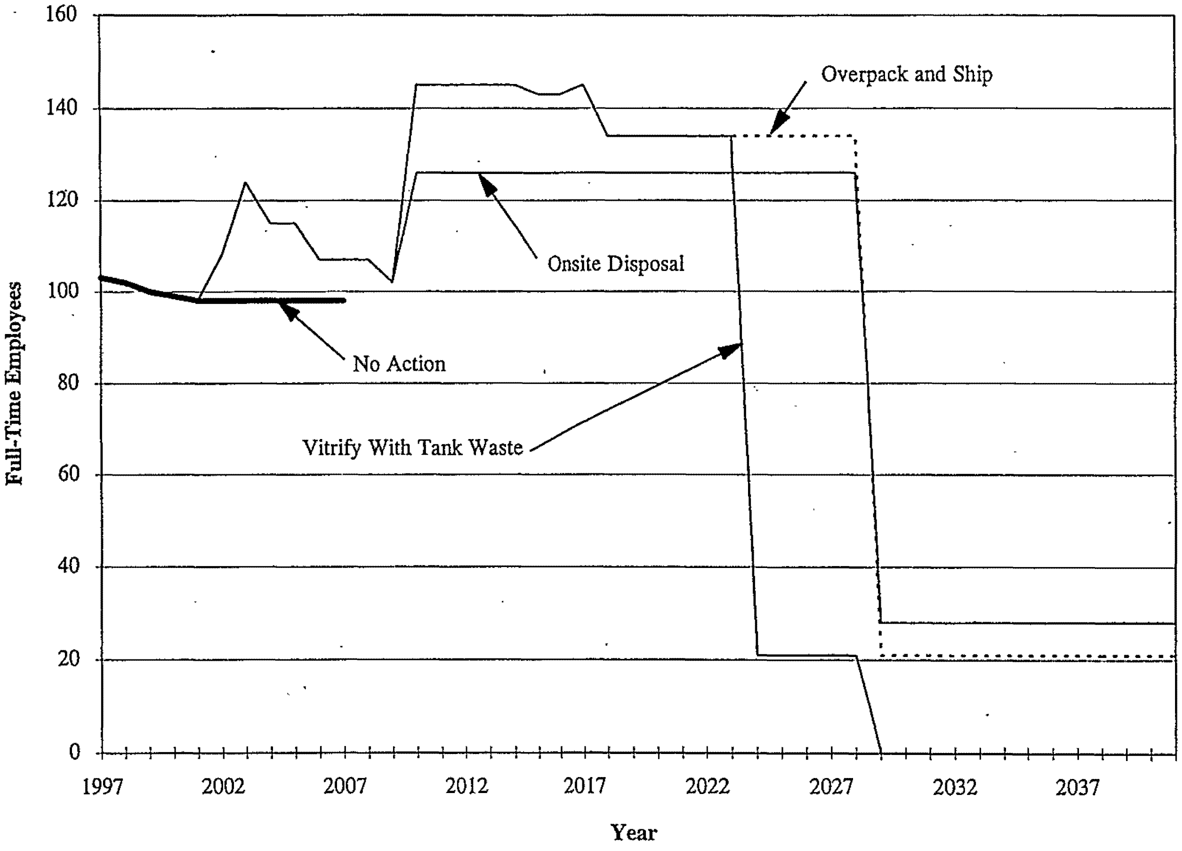


Table B.11.0.1 Schedule - Tank Waste and Capsule Alternatives

Alternative/Activity	Schedule
No Action Alternative (Tank Waste)	
Continued Operations	1997-2097
Long-Term Management Alternative	
Continued Operations	1997-2097
Construction	
Waste Retrieval	2033-2037/2083-2087
New Tanks	2033-2037/2083-2087
Operations	2037-2042/2087-2092
Decontamination and Decommissioning	2042-2047/2092-2097
Monitoring and Maintenance	1997-2097
In Situ Fill and Cap Alternative	
Continued Operations	1997-2009
Research and Development	N/A
Construction	2003-2005
Operation	2000-2009
Decontamination and Decommissioning	2009-2012
Monitoring and Maintenance	2009-2029
Closure	2012-2029
Post-Closure Monitoring	2029-2129
In Situ Vitrification Alternative	
Continued Operations	1997-2013
Research and Development	1997-2002
Construction	1999-2016
Operation	2005-2013
DST Evaporation	2000-2005
In Situ Vitrification Start Up	2005-2008
In Situ Vitrification Operation	2008-2013
Decontamination and Decommissioning	2013-2016
Monitoring and Maintenance	2016-2033
Closure	2016-2033
Post-Closure Monitoring	2033-2133
Ex Situ Intermediate Separations Alternative	
Continued Operations	1997-2022
Research and Development	1997-2018

Table B.11.0.1 Schedule, Tank Waste and Capsule Alternatives (cont'd)

Alternative/Activity	Schedule
Construction	
Waste Retrieval and Transfer	1998-2017
Waste Treatment	1999-2009
HLW Treatment Facility	2003-2009
LAW Treatment Facility	1999-2004
Operation	
Waste Retrieval	2001-2022
Pretreatment	2004-2022
Low-Level Waste Vitrification	2005-2022
HLW Vitrification	2011-2022
Low-Level Waste Disposal	2005-2022
HLW Transportation and Disposal [1]	2022-2040
Decontamination and Decommissioning	
Waste Retrieval Facilities	2013-2025
Treatment Facilities	2022-2031
HLW Monitoring and Maintenance [2]	2022-2040
Closure	2010-2034
Post-Closure Monitoring	2034-2134
Ex Situ No Separations (Vitrification and Calcination) Alternative	
Continued Operations	1997-2020
Research and Development	1997-2007
Construction	
Waste Retrieval and Transfer	1998-2017
Waste Treatment	1999-2005
Operation	
Waste Retrieval	2003-2020
HLW Vitrification/Calcination	2006-2020
HLW Transportation and Disposal [1]	2020-2040
Decontamination and Decommissioning	
Waste Retrieval Facilities	2015-2023
Treatment Facilities	2020-2025
HLW Monitoring and Maintenance [2]	2020-2040
Closure	2011-2025
Post-Closure Monitoring	2025-2125
Ex Situ Extensive Separations Alternative	
Continued Operations	1997-2023

Table B.11.0.1 Schedule, Tank Waste and Capsule Alternatives (cont'd)

Alternative/Activity	Schedule
Research and Development	1997-2018
Construction	
Waste Retrieval and Transfer	1998-2016
Waste Treatment Facility	1999-2011
HLW Treatment Facility	2001-2006
LAW Treatment Facility	1999-2004
Operation	
Waste Retrieval	2003-2023
Treatment	2004-2023
Pretreatment	2004-2023
Low-level waste Vitrification	2006-2023
HLW Vitrification	2013-2023
Low-Level Waste Disposal	2006-2023
HLW Transportation and Disposal [1]	2022-2025
Decontamination and Decommissioning	
Waste Retrieval Facilities	2014-2024
Treatment Facilities	2023-2028
HLW Monitoring and Maintenance [2]	2023-2025
Closure	2012-2032
Post-Closure Monitoring	2032-2132
Ex Situ/In Situ Combination 1 Alternative	
Continued Operations	1997-2023
Research and Development	1997-2018
Construction	
Waste Retrieval and Transfer	1998-2016
Waste Treatment Facilities	1999-2009
HLW Treatment Facility	2003-2009
LAW Treatment Facility	1999-2005
Fill and Cap	2003-2005
Operation	
Waste Retrieval	2002-2022
Waste Treatment	2005-2023
Pretreatment	2005-2023
Low-Level Waste Vitrification	2006-2023
HLW Vitrification	2010-2023
Low-Level Waste Disposal	2006-2023
HLW Transportation and Disposal [1]	2022-2040

Table B.11.0.1 Schedule, Tank Waste and Capsule Alternatives (cont'd)

Alternative/Activity	Schedule
Fill and Cap Operations	2000-2009
Decontamination and Decommissioning	
Waste Retrieval Facilities	2013-2024
Treatment Facilities	2023-2032
Fill and Cap Equipment	2022-2025
HLW Monitoring and Maintenance [2]	2023-2040
Closure	2011-2035
Post-Closure Monitoring	2035-2135
Ex Situ/In Situ Combination 2 Alternative	
Continued Operations	1997-2022
Research and Development	1997-2018
Construction	
Waste Retrieval and Transfer	1998-2016
Waste Treatment Facilities	1999-2008
HLW Treatment Facility	2002-2008
LAW Treatment Facility	1999-2004
Fill and Cap	2003-2005
Operation	
Waste Retrieval	2002-2022
Waste Treatment	2004-2022
Pretreatment	2004-2022
Low-Level Waste Vitrification	2005-2022
HLW Vitrification	2009-2022
low-Level Waste Disposal	2005-2022
HLW Transportation and Disposal [1]	2022-2040
Fill and Cap	2000-2009
Decontamination and Decommissioning	
Waste Retrieval Facilities	2013-2024
Treatment Facilities	2022-2031
Fill and Cap Equipment	2022-2025
HLW Monitoring and Maintenance [2]	2022-2040
Closure	2010-2032
Post-Closure Monitoring	2032-2132
Phased Implementation (Phase 1) Alternative	
Continued Operations (Phase 1 and Phase 2)	1997-2028
Research and Development	N/A

Table B.11.0.1 Schedule, Tank Waste and Capsule Alternatives (cont'd)

Alternative/Activity	Schedule
Construction	
Treatment Facilities	1998-2002
Operation	
LAW Treatment Facility	2002-2012
HLW Treatment Facility	2002-2008
Phased Implementation (Phase 2) Alternative	
Construction	
Waste Retrieval and Transfer	2004-2020
Treatment Facilities	2006-2012
HLW Treatment Facility	2007-2012
LAW Treatment Facility	2006-2011
Operation	
Waste Retrieval	2008-2028
Treatment	
HLW	2012-2028
LAW	2011-2024
HLW Transportation [1]	2022-2040
Decontamination and Decommissioning	2015-2031
HLW Monitoring and Maintenance [2]	2028-2040
Closure	2016-2040
Post-Closure Monitoring and Maintenance	2040-2140
No Action (Capsules)	
Continued Operation	1997-2007
Research and Development	N/A
Construction	N/A
Operation	N/A
Decontamination and Decommissioning	N/A
Monitoring and Maintenance	N/A
Onsite Disposal Alternative	
Continued Operation	1997-2028
Research and Development	N/A
Construction	2002-2009
Operation	2010-2029
Decontamination and Decommissioning	N/A
Monitoring and Maintenance	2029-2129
Overpack and Ship Alternative	
Continued Operation	1997-2028

Table B.11.0.1 Schedule, Tank Waste and Capsule Alternatives (cont'd)

Alternative/Activity	Schedule
Research and Development	N/A
Construction	2002-2009
Operation	2010-2028
HLW Transportation and Disposal [1]	2028-2029
Decontamination and Decommissioning	N/A
Monitoring and Maintenance [2]	2010-2029
Vitrify with Tank Wastes Alternative	
Continued Operation	1997-2023
Research and Development	N/A
Construction	2007-2012
Operation	2012-2028
HLW Transportation to Repository [1]	2022-2040
Decontamination and Decommissioning	2028-2033
Monitoring and Maintenance [2]	2028-2040

Notes:

[1] Interim storage could be extended up to 50 years (2070).

[2] HLW Monitoring and Maintenance could be extended up to 50 years (2070).

Table B.11.0.2 Cost Summary for Tank Waste Alternatives [1]

Cost Component	No Action Alternative	Long-Term Management Alternative	In Situ Fill and Cap Alternative [4]	In Situ Vitrification Alternative [4]	Ex Situ Intermediate Separations Alternative	Ex Situ No Separations Alternative (Vitrification)
Current Operations [2]	$14,300	$14,300	$7,469	$8,652	$8,600	$8,325
Research and Development	N/A	$100	0	$70	$820	$470
Capital Cost	N/A	$5,930	$25	$4,900	$5,880	$4,890
Operating Cost [3]	N/A	$440	$275	$2,740	$9,368	$9,797
Closure Costs	N/A	N/A	$116	$116	$169	$152
Total Treatment Cost	$14,300	$20,770	$7,885	$16,478	$24,837	$23,634
Repository Fee	N/A	N/A	N/A	N/A	$5,280	$38,900
Total Estimated Cost	$14,300	$20,770	7,885	$16,478	$30,117	$62,534

Cost Component	Ex Situ No Separations Alternative (Calcination)	Ex Situ Extensive Separations Alternative	Ex Situ/ In Situ Combination 1 Alternative [4]	Ex Situ/ In Situ Combination 2 Alternative [4]	Phased Implementation Alternative
Current Operations [2]	$8,325	$8,600	$9,142	$9,080	$8,600
Research and Development	$470	$1,490	$816	$813	$190
Capital Cost	$4,890	$7,482	$3,314	$2,353	$7,580
Operating Cost [3]	$7,863	$9,983	$6,267	$3,061	$10,618
Closure Costs	$152	$170	$137	$125	$211
Total Treatment Cost	$21,700	$27,725	$19,676	$15,432	$27,199
Repository Fee	$14,630	$270	$3,710	$2,150	$5,280
Total Estimated Cost	$36,330	$27,995	$23,386	$17,582	$32,479

Notes:

[1] Costs are reported in millions of 1995 dollars.

[2] Current operations includes: program management, operations and maintenance, tank farm safety, tank farm upgrades, and characterization.

[3] Operating costs include start-up, decontamination and decommissioning, and monitoring and maintenance costs.

[4] Includes additional money for additional characterization requirements required for in situ disposal.

N/A = Not applicable

Table B.11.0.3 Cost Summary for Capsule Alternatives [1]

Cost Component	No Action Alternative	Onsite Disposal Alternative	Overpack and Ship Alternative	Vitrify with Tank Waste Alternative
Current Operations [2]	$112	$377	$377	$315
Capital Cost	N/A	$64	$32	$36
Operating Cost [3]	N/A	$232	$40	$53
Closure Costs	N/A	$5	N/A	N/A
Research and Development	N/A	$19	$14	$5
Repository Fee	N/A	N/A	$124	$111
Total Cost	$112	$697	$587	$511

Notes:

[1] Costs are reported in millions of 1995 dollars. Cost uncertainties were not estimated for the capsule alternatives.

[2] Current operations includes: program management, operations and maintenance, tank farm safety, tank farm upgrades, and characterization.

[3] Operating costs include start-up, decontamination and decommissioning, and monitoring and maintenance costs.

N/A = Not applicable

Table B.11.0.4 Resource Summary, Tank Waste Alternatives

Resource	Phase	No Action Alternative	Long-Term Management Alternative	In Situ Fill and Cap Alternative	In Situ Vitrification Alternative
Land, permanently committed (hectares) (Long-term commitment of radiologically contaminated area)	Total	17	25	25	25
	Construction/Operation	0	8	17	17
	Closure	17	17	8	8
Land, incremental temporarily committed (hectares) (area disturbed in the 200 Area during construction and operations)	Total	0	50	21	111
	Construction/Operation	0	50	1	91
	Closure	0	0	20	20
Borrow Pit Disturbed Area (hectares) based on an excavation depth of 3 meters	McGee Ranch	N/A	N/A	16	16
	Pit 30	N/A	16	39	37
	Vernita Quarry	N/A	N/A	21	21
Water, Total (cubic meters)	Sanitary plus Raw Water	1.5E+06	1.5E+06	2.43E+06	1.71E+08
Water, Sanitary (cubic meters)	Total	N/A	N/R	2.43E+06	1.71E+08
	Construction	N/A	N/R	5.9E+05	1.7E+08
	Operation	N/A	N/A	1.8E+06	1.2E+06
	Closure	N/A	N/A	4.13E+04	3.8E+04
Water, Raw (cubic meters)	Total	1.5E+06	1.5E+06	N/R	N/R
	Construction	N/A	N/A	N/A	N/A
	Operation	1.5E+06	1.5E+06	N/R	N/R
	Closure	N/A	N/A	N/A	N/A
Energy Electricity (GWh)	Total	1.1E+03	1.1E+03	9.5E+02	4.46E+03
	Construction	N/A	4.4E-01	1.8E+00	5.6E+01
	Operation	1.1E+03	1.1E+03	9.5E+02	4.4E+03
	Closure	N/A	N/A	N/A	N/A
Gasoline (cubic meters)	Total	N/A	8.6E+04	3.98E+02	1.53+04
	Construction	N/A	8.6E+04	9.7E+01	1.5E+04
	Operation	N/A	N/A	N/A	N/A
	Closure	N/A	N/A	3.01E+02	3.01E+02

Table B.11.0.4 Resource Summary, Tank Waste Alternatives (cont'd)

Resource	Phase	No Action Alternative	Long-Term Management Alternative	In Situ Fill and Cap Alternative	In Situ Vitrification Alternative
Diesel (cubic meters)	Total	2.2E+04	8.5E+04	5.96E+04	7.11E+04
	Construction	N/R	6.3E+04	1.3E+04	2.0E+04
	Operation	2.2E+04	2.2E+04	6.10E+03	6.10E+03
	Closure	N/A	N/A	5.22E+04	4.5E+04
Kerosene (cubic meters)	Operation	N/A	N/A	N/A	N/A
Materials: Concrete (cubic meters)	Total	N/A	3.5E+05	1.9E+04	1.3E+05
	Construction	N/A	3.5E+05	N/R	1.3E+05
	Operation	N/A	N/A	N/A	N/A
	Closure	N/A	N/A	1.9E+04	N/R
Carbon Steel (metric tons)	Total	1.0E+03	1.4E+04	N/R	2.6E+05
	Construction	N/A	1.3E+04	N/R	2.6E+05
	Operation	1.0E+03	1.0E+03	N/R	N/R
	Closure	N/A	N/A	N/A	N/A
Stainless Steel (mt)	Total	N/A	2.2E+01	N/A	N/A
	Construction	N/A	2.2E+01	N/A	N/A
	Operation	N/A	N/A	N/A	N/A
Hastelloy/Inconel (mt)	Construction	N/A	N/A	N/A	N/A
Glass Formers (mt)	Operation	N/A	N/A	N/A	1.1E+04
Process Chemicals (mt)	Operation	N/A	N/A	N/A	6.8E+03
Bulk Sulfur Cement Sulfur (mt)	Operation	N/A	N/A	N/A	N/A
Dicyclopentadine (mt)		N/A	N/A	N/A	N/A
Oligomer (mt)		N/A	N/A	N/A	N/A
Ion exchange media (cubic meters)	Operation	N/A	N/A	N/A	N/A
Borrow Site Silt (cubic meters) [1]	Closure	N/A	N/A	3.77E+05	3.77E+05
Borrow Site Sand and Gravel (cubic meters)	Total	N/A	4.65E+05	1.18E+06	1.104E+06
	Construction	N/A	4.65E+05	N/A	1.39E+05
	Operation	N/A	N/A	N/A	5.4E+05
	Closure	N/A	N/A	1.18E+06	4.24E+05
Borrow Site Basalt (cubic meters) [1]	Closure	N/A	N/A	6.38E+05	6.38E+05
Asphalt (cubic meters)	Closure	N/A	N/A	6.25E+05	6.25E+05

Table B.11.0.4 Resource Summary, Tank Waste Alternatives (cont'd)

Resource	Phase	Ex Situ Intermediate Separations Alternative	Ex Situ No Separations Alternative (Vitrification)	Ex Situ No Separations Alternative (Calcination)	Ex Situ Extensive Separations Alternative
Land, permanently committed (hectares) (Long-term commitment of radiologically contaminated area)	Total	46	28	28	44
	Construction/ Operation	35	20	20	33
	Closure	11	8	8	11
Land, incremental temporarily committed (hectares) (area disturbed in the 200 Area during construction and operations)	Total	112	88	90	112
	Construction/ Operation	88	68	70	88
	Closure	24	20	20	24
Borrow Pit Disturbed Area (hectares) based on an excavation depth of 3 meters	McGee Ranch	17	16	16	19
	Pit 30	81	138	84	77
	Vernita Quarry	29	21	21	32
Water, Total (cubic meters)	Sanitary plus Raw Water	1.86E+07	4.05E+06	2.45E+06	6.7E+07
Water, Sanitary (cubic meters)	Total	6.37E+06	3.36E+06	2.36E+06	7.0E+06
	Construction	5.02E+05	2.39E+05	2.39E+05	5.29E+05
	Operation	5.8E+06	3.08E+06	2.08E+06	6.42E+06
	Closure	6.33E+04	4.13E+04	4.13E+04	4.33E+04
Water, Raw (cubic meters)	Total	1.22E+07	6.91E+05	8.58E+04	6.0E+07
	Construction	N/R	N/R	N/R	N/R
	Operation	1.22E+07	6.91E+05	8.58E+04	6.0E+07
	Closure	N/R	N/R	N/R	N/R
Energy Electricity (GWh)	Total	1.09E+04	7.05E+03	4.72E+03	3.33E+04
	Construction	7.38E+01	3.0E+01	3.0E+01	8.9E+01
	Operation	1.08E+04	7.02E+03	4.69E+03	3.32E+04
	Closure	N/R	N/R	N/R	N/R
Gasoline (cubic meters)	Total	8.1E+03	5.43E+03	5.43E+03	1.02E+04
	Construction	7.63E+03	5.13E+03	5.13E+03	9.73E+03
	Operation	5.1E+01	N/R	N/R	4.4E+01
	Closure	4.12E+02	3.01E+02	3.01E+02	4.09E+02

Table B.11.0.4 Resource Summary, Tank Waste Alternatives (cont'd)

Resource	Phase	Ex Situ Intermediate Separations Alternative	Ex Situ No Separations Alternative (Vitrification)	Ex Situ No Separations Alternative (Calcination)	Ex Situ Extensive Separations Alternative
Diesel (cubic meters)	Total	9.95E+04	6.55E+04	6.53E+04	1.08E+05
	Construction	2.11E+04	1.19E+04	1.19E+04	2.9E+04
	Operation	8.16E+03	1.39E+03	1.17E+03	8.17E+03
	Closure	7.02E+04	5.22E+04	5.22E+04	7.08E+04
Kerosene (cubic meters)	Operation	6.58E+04	8.48E+04	1.05E+04	1.87E+05
Materials: Concrete (cubic meters)	Total	1.04E+06	2.78E+06	1.26E+06	9.08E+05
	Construction	5.41E+05	2.76E+05	2.66E+05	5.0E+05
	Operation	4.75E+05	2.48E+06	9.8E+05	3.89E+05
	Closure	1.9E+04	1.9E+04	1.9E+04	1.9E+04
Carbon Steel (metric tons)	Total	2.69E+05	2.58E+05	1.32E+05	3.02E+05
	Construction	1.07E+05	6.17E+04	6.12E+04	1.4E+05
	Operation	1.62E+05	1.96E+05	7.04E+04	1.62E+05
	Closure	N/R	N/R	N/R	N/R
Stainless Steel (mt)	Total	3.26E+04	1.78E+05	7.1E+04	2.16E+04
	Construction	1.9E+04	1.26E+04	1.26E+04	2.1E+04
	Operation	1.36E+04	1.65E+05	5.84E+04	6.41E+02
Hastelloy/Inconel (mt)	Construction	2.3E+03	1.51E+03	1.51E+03	1.8E+03
Glass Formers (mt)	Operation	5.65E+05	4.29E+05	N/A	5.23E+05
Process Chemicals (mt)	Operation	2.71E+05	2.87E+05	3.06E+05	1.07E+06
Bulk Sulfur Cement	Operation				
Sulfur (mt)		1.48E+05	N/A	N/A	2.83E+05
Dicyclopentadine (mt)		3.86E+03	N/A	N/A	7.42E+03
Oligomer (mt)		3.86E+03	N/A	N/A	7.42E+03
Ion exchange media (cubic meters)	Operation	2.2E+02	N/A	N/A	4.9E+03
Borrow Site Silt (cubic meters) [1]	Closure	5.19E+05	3.77E+05	3.77E+05	5.66E+05
Borrow Site Sand and Gravel (cubic meters)	Total	2.43E+06	4.13E+06	2.54E+06	2.31E+06
	Construction	5.8E+05	2.97E+05	2.85E+05	5.35E+05
	Operation	5.08E+05	1.67E+06	6.61E+05	2.63E+05
	Closure	1.34E+06	1.2E+06	1.2E+06	1.36E+06
Borrow Site Basalt (cubic meters) [1]	Closure	8.78E+05	6.38E+05	6.38E+05	9.58E+05
Asphalt (cubic meters)	Closure	8.6E+04	6.26E+04	6.26E+04	9.39E+04

Table B.11.0.4 Resource Summary, Tank Waste Alternatives (cont'd)

Resource	Phase	Ex Situ/In Situ Combination 1 Alternative	Ex Situ/In Situ Combination 2 Alternative	Phased Implementation Alternative (Phase 1)	Phased Implementation Alternative (Phase 2)
Land, permanently committed (hectares) (Long-term commitment of radiologically contaminated area)	Total	41	34	3	46
	Construction/ Operation	31	25	3	35
	Closure	10	9	0	11
Land, incremental temporarily committed (hectares) (area disturbed in the 200 Area during construction and operations)	Total	107	84	32	145
	Construction/ Operation	85	62	32	121
	Closure	22	22	0	24
Borrow Pit Disturbed Area (hectares) based on an excavation depth of 3 meters	McGee Ranch	16	14	N/A	19
	Pit 30	63	55	1	86
	Vernita Quarry	26	24	N/A	32
Water, Total (cubic meters)	Sanitary plus Raw Water	1.31E+07	7.38E+06	2.81E+06	1.85E+07
Water, Sanitary (cubic meters)	Total	5.35E+06	2.91E+06	6.1E+05	6.48E+06
	Construction	4.22E+05	2.95E+05	3.9E+05	6.78E+05
	Operation	4.42E+06	2.56E+06	2.2E+05	5.7E+06
	Closure	5.09E+04	4.95E+04	N/A	6.33E+04
Water, Raw (cubic meters)	Total	7.71E+06	4.47E+06	2.2E+06	1.2E+07
	Construction	N/R	N/R	N/R	N/R
	Operation	7.71E+06	4.47E+06	2.2E+06	1.2E+07
	Closure	N/R	N/R	N/A	N/R
Energy Electricity (GWh)	Total	1.01E+04	7.57E+03	1.65E+03	1.07E+04
	Construction	3.72E+03	2.60E+03	5.0E+01	1.0E+02
	Operation	6.34E+03	4.97E+03	1.6E+03	1.06E+04
	Closure	N/R	N/R	N/R	N/R
Gasoline (cubic meters)	Total	5.26E+03	3.55E+03	6.5E+03	1.09E+04
	Construction	4.88E+03	3.27E+03	6.5E+03	1.04E+04
	Operation	2.73E+01	2.18E+01	N/A	5.01E+01
	Closure	3.5E+02	2.57E+02	N/A	4.12E+02

Table B.11.0.4 Resource Summary, Tank Waste Alternatives (cont'd)

Resource	Phase	Ex Situ/In Situ Combination 1 Alternative	Ex Situ/In Situ Combination 2 Alternative	Phased Implementation Alternative (Phase 1)	Phased Implementation Alternative (Phase 2)
Diesel (cubic meters)	Total	9.85E+04	7.84E+04	1.8E+04	1.07E+05
	Construction	3.13E+04	1.61E+04	1.8E+04	2.84E+04
	Operation	7.17E+03	3.88E+03	N/A	9.14E+03
	Closure	6.04E+04	5.84E+04	N/A	6.97E+04
Kerosene (cubic meters)	Operation	4.04E+04	2.34E+04	9.8E+02	6.48E+04
Materials: Concrete (cubic meters)	Total	5.76E+05	3.83E+05	3.01E+04	1.14E+06
	Construction	3.07E+05	1.91E+05	2.4E+04	7.15E+05
	Operation	2.50E+05	1.73E+05	6.07E+03	4.7E+05
	Closure	1.9E+04	1.9E+04	N/A	1.9E+04
Carbon Steel (metric tons)	Total	1.49E+05	9.73E+04	8.64E+04	2.97E+05
	Construction	5.36E+04	3.75E+04	8.3E+04	1.42E+05
	Operation	9.52E+04	5.98E+04	3.43E+03	1.55E+05
	Closure	N/R	N/R	N/A	N/R
Stainless Steel (mt)	Total	1.91E+04	1.22E+04	1.55E+04	3.76E+04
	Construction	9.5E+03	6.65E+03	1.5E+04	2.44E+04
	Operation	9.62E+03	5.56E+03	4.8E+02	1.32E+04
Hastelloy/Inconel (mt)	Construction	1.15E+03	8.05E+02	2.6E+03	3.13E+03
Glass Formers (mt)	Operation	3.47E+05	2.02E+05	1.3E+05	5.58E+05
Process Chemicals (mt)	Operation	1.60E+05	9.30E+04	1.64E+04	2.67E+05
Bulk Sulfur Cement	Operation				
Sulfur (mt)		8.91E+04	5.17E+04	2.96E+04	1.46E+05
Dicyclopentadine (mt)		2.33E+03	1.35E+03	7.72E+02	3.81E+03
Oligomer (mt)		2.33E+03	1.35E+03	7.72E+02	3.81E+03
Ion exchange media (cubic meters)	Operation	1.1E+02	6.38E+01	4.4E+01	2.17E+02
Borrow Site Silt (cubic meters) [1]	Closure	4.72E+05	4.25E+05	N/A	5.66E+05
Borrow Site Sand and Gravel (cubic meters)	Total	1.89E+06	1.65E+06	3.2E+04	2.56E+06
	Construction	3.29E+05	2.04E+05	2.55E+04	7.65E+05
	Operation	2.68E+05	1.86E+05	6.49E+03	4.36E+05
	Closure	1.29E+06	1.26E+06	N/A	1.36E+06
Borrow Site Basalt (cubic meters)	Closure	7.85E+05	7.18E+05	N/A	9.58E+05
Asphalt (cubic meters)	Closure	7.83E+04	7.04E+04	N/A	9.39E+04

Notes:
GWh = gigawatt-hours
N/A = Not applicable
N/R = Not reported

Table B.11.0.5 Resource Summary, Capsule Alternatives

Resource	Phase	No Action	Onsite Disposal	Overpack and Ship	Vitrify With Tank Waste
Land Permanently Committed (hectares)	Total	1	1.8	N/A	N/A
Land Temporarily Committed (hectares)	Total	N/A	4	2	1
Water (cubic meters)	Total	6.0E+03	6.4E+03	1.5E+04	2.3E+04
	Construction	N/A	3.9E+02	3.9E+02	5.5E+02
	Operation	6.0E+03	6.0E+03	1.50E+04	2.25E+04
Electricity (GWh)	Total	4.0E-01	1.29E+02	4.4E-01	7.0E-01
	Construction	N/A	8.0E-02	1.0E-02	1.5E-02
	Operation	4.0E-01	1.2E+02	4.3E-01	7.0E-01
Gasoline (cubic meters)	Total	N/A	2.0E+02	5.0E+01	5.0E+01
	Construction		2.0E+02	5.0E+01	5.0E+01
	Operation		N/R	N/A	N/A
Propane (cubic meters)	Construction	N/A	5.8E+02	1.4E+02	N/A
Diesel (cubic meters)	Total	2.4E+01	4.8E+01	4.6E+01	4.6E+01
	Construction	0	0	0	0
	Operation	2.4E+01	4.8E+01	4.6E+01	4.6E+01
Concrete (cubic meters)	Total	N/A	4.1E+03	2.3E+03	3.45E+03
	Construction		4.1E+03	2.3E+03	3.45E+03
	Operation		N/A	N/A	N/A
Carbon Steel (metric tons)	Total	N/A	1.03E+03	3.8E+02	6.0E+02
	Construction		8.1E+02	2.2E+02	3.7E+02
	Operation		2.2E+02	1.6E+02	2.3E+02
Stainless Steel (metric tons)	Total	N/A	2.26E+01	1.96E+01	2.0E+01
	Construction		2.26E+01	1.96E+01	2.0E+01
	Operation		N/A	N/A	N/A
Silica Sand (metric tons)	Operation	N/A	5.30E+02	N/A	N/A

Notes:
GWh = gigawatt-hours
N/A = Not applicable
N/R = Not reported

Table B.11.0.6 Radiological Emissions (Curies), Tank Waste Alternatives [1, 3]

Emission Type	Phase	No Action Alternative	Long-Term Management Alternative	In Situ Fill and Cap Alternative	In Situ Vitrification Alternative
Am-241	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	0	0	1.5E-05	1.0E-06
	Water	0	0	N/R	N/R
C-14	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	0	0	7.5E-07	5.3E+03
	Water	0	0	N/R	N/A
Cs-137	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	2.1E-03	2.1E-03	4.9E-03	3.5E-04
	Water	1.8E-02	1.8E-02	N/R	N/R
I-129	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	4.6E-03	4.6E-03	2.2E-09	3.8E+01
	Water	2.7E-02	2.7E-02	N/R	N/A
Pu-239, -240	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	0	0	4.6E-06	3.3E-07
	Water	0	0	N/R	N/R
Ru-106	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	0	0	5.3E-12	3.8E-13
	Water	0	0	N/R	N/R
Sm-151	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	0	0	8.8E-05	6.3E-06
	Water	0	0	N/R	N/R

Table B.11.0.6 Radiological Emissions (Curies), Tank Waste Alternatives [1,3] (cont'd)

Emission Type	Phase	No Action Alternative	Long-Term Management Alternative	In Situ Fill and Cap Alternative	In Situ Vitrification Alternative
Tc-99	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	0	0	4.5E-06	3.2E-07
	Water	0	0	N/R	N/R
Zr-93	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	0	0	5.5E-07	3.9E-08
	Water	0	0	N/R	N/R
Total Alpha [2]	Retrieval:	N/A	N/A	N/R	N/R
	Air				
	Water				
	Operations:				
	Air	2.1E-05	2.1E-05	7.1E-04	7.1E-04
	Water	N/A	N/A	N/A	N/A
Total Beta [2]	Retrieval:	N/A	N/A	N/A	N/A
	Air				
	Water				
	Operations:				
	Air	1.2E-05	1.2E-05	4.0E-04	4.0E-04
	Water	0	0	0	0

Table B.11.0.6 Radiological Emissions (Curies), Tank Waste Alternatives [1,3] (cont'd)

Emission Type	Phase	Ex Situ Intermediate Separations Alternative	Ex Situ No Separations Alternative (Vitrification)	Ex Situ No Separations Alternative (Calcination)	Ex Situ Extensive Separations Alternative
Am-241	Retrieval:				
	Air	0	0	0	0
	Water	3.5E+03	3.5E+03	3.5E+03	3.5E+03
	Operations:				
	Air	5.2E-02	5.3E-02	5.3E-02	5.2E-02
	Water	4.2E-04	2.3E-04	2.3E-04	4.2E-04
C-14	Retrieval:				
	Air	0	0	0	0
	Water	1.0E+02	1.0E+02	1.0E+02	1.0E+02
	Operations:				
	Air	5.3E+03	5.3E+03	8.1E+02	5.3E+03
	Water	N/A	N/A	N/A	N/A
Cs-137	Retrieval:				
	Air	2.0E-02	2.0E-02	2.0E-02	2.0E-02
	Water	1.2E+05	1.2E+05	1.2E+05	1.2E+05
	Operations:				
	Air	2.5E+01	1.8E+01	1.8E+01	1.7E+01
	Water	6.9E-02	3.9E-02	3.9E-02	6.9E-02
I-129	Retrieval:				
	Air	3.0E-02	3.0E-02	3.0E-2	3.0E-02
	Water	1.0E-02	1.0E-02	1.0E-02	1.0E-02
	Operations:				
	Air	3.8E+01	3.8E+01	3.8E+01	3.8E+01
	Water	3.5E-04	1.9E-04	1.9E-04	3.5E-04
Pu-239, -240	Retrieval:				
	Air	0	0	0	0
	Water	2.2E+03	2.2E+03	2.2E+03	2.2E+03
	Operations:				
	Air	1.7E-02	1.7E-02	1.7E-02	1.6E-02
	Water	4.2E-04	2.3E-04	2.3E-04	4.2E-04
Ru-106	Retrieval:				
	Air	0	0	0	0
	Water	3.8E-02	3.8E-02	3.8E-02	3.8E-02
	Operations:				
	Air	1.9E-08	1.9E-08	1.9E-08	1.9E-08
	Water	1.0E-02	5.8E-03	5.8E-03	1.0E-02
Sm-151	Retrieval:				
	Air	0	0	0	0
	Water	4.0E+04	4.0E+04	4.0E+04	4.0E+04
	Operations:				
	Air	3.2E-01	3.2E-01	3.9E-01	3.2E-01
	Water	N/A	N/A	N/A	N/A

Table B.11.0.6 Radiological Emissions (Curies), Tank Waste Alternatives [1,3] (cont'd)

Emission Type	Phase	Ex Situ Intermediate Separations Alternative	Ex Situ No Separations Alternative (Vitrification)	Ex Situ No Separations Alternative (Calcination)	Ex Situ Extensive Separations Alternative
Tc-99	Retrieval:				
	Air	0	0	0	0
	Water	2.3E+02	2.3E+02	2.3E+02	2.3E+02
	Operations:				
	Air	1.6E-02	1.6E-02	1.6E-02	1.6E-02
	Water	N/A	N/A	N/A	N/A
Zr-93	Retrieval:				
	Air	0	0	0	0
	Water	2.2E+02	2.2E+02	2.8E+02	2.2E+02
	Operations:				
	Air	1.8E-01	1.8E-01	1.8E-01	2.0E-03
	Water	N/A	N/A	N/A	N/A

Table B.11.0.6 Radiological Emissions (Curies), Tank Waste Alternatives [1,3] (cont'd)

Emission Type	Phase	Ex Situ/ In Situ Combination 1 Alternative	Ex Situ/ In Situ Combination 2 Alternative	Phased Implementation Alternative (Phase 1)	Phased Implementation Alternative (Phase 2)
Am-241	Retrieval:				
	Air	0	0	0	0
	Water	2.7E+03	2.2E+02	0	3.5E+03
	Operations:				
	Air	4.8E-02	2.9E-02	1.4E-03	5.1E-02
	Water	3.8E-04	2.3E-04	6.4E-05	3.7E-04
C-14	Retrieval:				
	Air	0	0	0	0
	Water	6.8E+01	6.7E+00	0	1.0E+02
	Operations:				
	Air	4.8E+03	4.3E+03	8.2E+02	4.5E+03
	Water	N/A	N/A	N/A	N/A
Cs-137	Retrieval:				
	Air	8.0E-03	4.8E-03	2.3E-03	1.8E-02
	Water	5.0E+04	1.2E+04	0	1.0E+05
	Operations:				
	Air	2.2E+01	1.3E+01	1.1E+00	2.3E+01
	Water	6.2E-02	3.7E-02	1.0E-02	7.2E-02
I-129	Retrieval:				
	Air	2.7E-02	1.6E+02	3.5E-03	2.7E-02
	Water	2.1E-01	6.4E-02	0	1.0E-02
	Operations:				
	Air	3.4E+01	3.1E+01	4.4E+00	3.4E+01
	Water	3.1E-04	1.9E-04	3.6E-05	2.7E-04
Pu-239, -240	Retrieval:				
	Air	0	0	0	0
	Water	1.6E+03	4.4E+01	0	2.2E+03
	Operations:				
	Air	8.5E-03	5.1E-03	3.4E-03	1.4E-02
	Water	1.6E+03	4.4E+01	8.4E-05	3.4E-04
Ru-106	Retrieval:				
	Air	0	0	0	0
	Water	2.7E-02	1.4E-03	0	3.8E-02
	Operations:				
	Air	6.8E-09	4.1E-09	7.8E-10	6.0E-09
	Water	3.6E-03	2.2E-03	0	3.6E-03
Sm-151	Retrieval:				
	Air	0	0	0	0
	Water	2.8E+04	1.8E+03	0	4.0E+04
	Operations:				
	Air	2.2E-01	1.3E-01	2.5E-02	2.0E-01
	Water	N/A	N/A	N/A	N/A

Table B.11.0.6 Radiological Emissions (Curies), Tank Waste Alternatives [1,3] (cont'd)

Emission Type	Phase	Ex Situ/ In Situ Combination 1 Alternative	Ex Situ/ In Situ Combination 2 Alternative	Phased Implementation Alternative (Phase 1)	Phased Implementation Alternative (Phase 2)
Tc-99	Retrieval:				
	Air	0	0	0	0
	Water	1.4+02	2.6E+01	0	2.3E+02
	Operations:				
	Air	1.4E-02	8.4E-03	1.1E-04	1.4E-02
	Water	N/A	N/A	N/A	N/A
Zr-93	Retrieval:				
	Air	0	0	0	0
	Water	1.7E+02	9.5E+00	0	2.2E+02
	Operations:				
	Air	1.2E-01	7.2E-02	1.4E-02	1.1E-07
	Water	N/A	N/A	0	N/A

Notes:

[1] Water emissions estimated during retrieval are conservatively based on an estimate of 15,000 L (4,000 gal) of liquid waste leakage from each SST during sluicing. This estimate was made for purposes of assessing groundwater impacts and does not imply that the proposed SST retrieval method would result leakage of the estimated volumes from each SST during retrieval.

[2] Total alpha and total beta are reported for the No Action alternative based on existing tank farm monitoring data for evaporator operation. The emissions reported for No Action and Long-Term Management represent annual emissions from periodic evaporator operation.

[3] Radiological emissions taken from WHC 1995c, e, f, g, i, j, n, and Jacobs 1996. Emissions shown for the in situ and ex situ alternatives do not include routine operating emissions. The emissions shown for the operations phase of the No Action and Long-Term Management alternatives are routine tank farm emissions and when annualized, these emissions would apply to the routine operations phase of each alternative.

N/A = Not applicable

N/R = Not reported

Table B.11.0.7 Nonradiological Emissions, Tank Waste Alternatives [1]

Emission Type	Phase	No Action Alternative	Long-Term Management Alternative	In Situ Fill and Cap Alternative	In Situ Vitrification Alternative
Particulate (kg)	construction	N/A	3.70E+02	3.3E+03	3.6E+04
	operation	2.0E+01	2.0E+01	<1E+03	6.0E+04
	closure	N/A	N/A	1.56E+05	1.56E+05
VOC (kg)	construction	N/A	N/R	N/R	N/R
	operation	6.8E+01	6.8E+01	N/R	N/R
	closure	N/A	N/A	N/R	N/R
Fugitive Dust (metric tons)	construction	N/A	1.35E+01	N/R	2.2E+05
	operation	N/A	N/A	N/R	N/R
	closure	N/A	N/A	3.1E+03	3.1E+03
NO_x (kg)	construction	N/A	1.91E+02	3.6E+04	3.5E+05
	operation	7.7E+01	7.7E+01	26	<3.6E+05
	closure	N/A	N/A	2.54E+06	2.54E+06
SO_x (kg)	construction	N/A	5.18E+02	4.7E+03	7.3E+04
	operation	1.2E+01	1.2E+01	0	0
	closure	N/A	N/A	2.76E+05	2.76E+05
Toxic Air Pollutant (kg)	construction	N/R	N/R	N/R	N/R
	operation	Benzene: 1.9 kg/yr Ammonia: 241 kg/yr Acetone: 82 kg/yr CO: 33 kg/yr	Benzene: 1.9 kg/yr Ammonia: 241 kg/yr Acetone: 82 kg/yr CO: 33 kg/yr	N/R	Ammonia: 5.6E+04
	closure	N/A	N/A	N/R	N/R
Hazardous Air Pollutant (kg)	construction	N/A	N/R	N/R	N/R
	operation	N/R	N/R	N/R	N/R
	closure	N/A	N/A	N/R	N/R
CO (kg)	construction	N/A	1.04E+04	3.0E+04	1.1E+06
	operation	7.1E+02	7.1E+02	0	0
	closure	N/A	N/A	1.1E+06	1.1E+06
Hydrocarbons (kg)	construction	N/A	1.07E+03	6.1E+03	2.9E+05
	operation	N/R	N/R	<1.1E+01	1.1E+01
	closure	N/A	N/A	1.19E+05	1.19E+05

Table B.11.0.7 Nonradiological Emissions, Tank Waste Alternatives [1] (cont'd)

Emission Type	Phase	No Action Alternative	Long-Term Management Alternative	In Situ Fill and Cap Alternative	In Situ Vitrification Alternative
Aldehydes (kg)	construction	N/A	50	7.7E+02	8.2E+01
	operation	N/R	N/R	N/A	N/A
	closure	N/A	N/A	6.19E+02	6.19E+02
Organic Acids (kg)	construction	N/A	5.0E+01	2.6E+02	4.0E+03
	operation	N/R	N/R	N/A	N/A
	closure	N/A	N/A	N/R	N/R

Emission Type	Phase	Ex Situ Intermediate Separations Alternative	Ex Situ No Separations Alternative (Vitrification)	Ex Situ No Separations Alternative (Calcination)	Ex Situ Extensive Separations Alternative
Particulate (kg)	construction	6.9E+04	5.0E+04	5.0E+04	8.3E+04
	operation	1.5E+04 kg/yr (95% cooling tower emissions)	8.8E+03 kg (95% cooling tower emissions)	8.8E+03 kg (95% cooling tower emissions)	3.08E+05 (0.003% PM-10)
	closure	2.06E+05	1.56E+05	1.56E+05	2.09E+05
VOC (kg)	construction	N/R	N/R	N/R	N/R
	operation	7.7E+02	6.0E+02	6.0E+02	9.8E+03
	closure	N/R	N/R	N/R	N/R
Fugitive Dust (metric tons)	construction	5.0E+02	4.07E+02	4.07E+02	6.0E+02
	operation	N/A	N/R	N/R	N/A
	closure	4.09E+03	3.1E+03	3.1E+03	4.1E+03
NO_x (kg)	construction	9.5E+05	6.9E+05	6.9E+05	1.14E+06
	operation	1.1E+04	1.4E+05	7.29E+05	1.47E+07
	closure	3.36E+06	2.54E+06	2.54E+06	3.4E+06
SO_x (kg)	construction	1.29E+04	9.48E+03	9.48E+03	1.55E+04
	operation	1.5E+04	2.13E+05	2.13E+05	1.33E+06
	closure	3.64E+05	2.76E+05	2.76E+05	3.69E+05

Table B.11.0.7 Nonradiological Emissions, Tank Waste Alternatives [1] (cont'd)

Emission Type	Phase	Ex Situ Intermediate Separations Alternative	Ex Situ No Separations Alternative (Vitrification)	Ex Situ No Separations Alternative (Calcination)	Ex Situ Extensive Separations Alternative
Toxic Air Pollutant (kg)	construction	N/R	N/R	N/R	N/R
	operation	HCl: 2.6E+03 HF: 6.1E+03 NH_3: 2.4E+04 HNO_3: 1.4E+03	HCl: 2.5E+05 HF: 6.2E+05 NH_3: 2.0E+04 HNO_3: 2.0E+03	HCl: 2.5E+05 HF: 6.2E+05 NH_3: 2.0E+04 HNO_3: 2.0E+03	Formic Acid: 1.54E+03 HF: 1.26E+03 Hydrogen Peroxide: 2.8E+01 HNO_3: 2.16E+04 Sodium Hydroxide: 9.8E+01
	closure	N/R	N/R	N/R	N/R
Hazardous Air Pollutant (kg)	construction	N/R	N/R	N/R	N/R
	operation	<100 kg/yr	1.4E+03	1.4E+03	Cr: 4.5E+01
	closure	N/R	N/R	N/R	N/R
CO (kg)	construction	1.0E+07	6.93E+06	6.93E+06	2.01E+07
	operation	<1.0E+05 kg/yr	2.12E+05	2.12E+05	2.44E+06
	closure	1.46E+06	1.11E+06	1.11E+06	1.48E+06
Hydrocarbons (kg)		5.5E+05	4.01E+05	4.01E+05	1.16E+06
	operation	N/R	N/R	N/R	N/R
	closure	1.57E+05	1.19E+05	1.19E+05	1.59E+05
Aldehydes (kg)	construction	5.1E+02	3.45E+02	3.45E+02	1.09E+03
	operation	N/R	N/R	N/R	N/R
	closure	8.17E+02	6.19E+02	6.19E+02	8.29E+02
Organic Acids (kg)	construction	N/A	N/A	0	N/A
	operation	N/R	N/R	N/R	N/R
	closure	N/R	N/R	N/R	N/R

Table B.11.0.7 Nonradiological Emissions, Tank Waste Alternatives [1] (cont'd)

Emission Type	Phase	Ex Situ/In Situ Combination 1 Alternative	Ex Situ/In Situ Combination 2 Alternative	Phased Implementation Alternative (Phase 1)	Phased Implementation Alternative (Phase 2)
Particulate (kg)	construction	3.5E+04	3.5E+04	2.0E+04	3.5E+04
	operation	8.1E+03	4.9E+03	1.6E+03	9.7E+03
	closure	1.81E+05	1.81E+05	N/R	2.06E+05
VOC (kg)	construction	N/R	N/R	N/R	N/R
	operation	3.9E+02	2.3E+02	4.3E+02	8.2E+02
	closure	N/R	N/R	N/R	N/R
Fugitive Dust (metric tons)	construction	2.49E+02	2.49E+02	1.64E+02	2.49E+02
	operation	N/R	N/R	N/R	N/R
	closure	3.6E+03	3.6E+03	N/R	4.09E+03
NO_x (kg)	construction	4.8E+05	4.8E+05	2.8E+05	4.8E+05
	operation	5.5E+03	3.3E+03	2.17E+04	2.7E+04
	closure	2.95E+06	2.95E+06	N/R	3.36E+06
SO_x (kg)	construction	6.5E+03	6.5E+03	3.8E+03	6.5E+03
	operation	7.5E+03	4.5E+03	3.15E+04	3.9E+04
	closure	3.25E+05	3.25E+05	N/R	3.64E+05
Toxic Air Pollutant (kg)	construction	N/R	N/R	N/R	N/R
	operation	HCl: 1.3E+03 HF: 3.1E+03 NH_3: 1.2E+04 HNO_3: 7.0E+02	HCl: 7.8E+02 HF: 1.86E+03 NH_3: 7.2E+03 HNO^3: 4.2E+02	HCl: 2.41E+02 HF: 8.94E+03 NH_3: 2.99E+03 HNO_3: 2.6E+03	HCl: 1.5E+03 HF: 1.2E+04 NH_3: 1.5E+04 HNO_3: 3.3E+03
	closure	N/R	N/R	N/R	N/R
Hazardous Air Pollutant (kg)	construction	N/R	N/R	N/R	N/R
	operation	<50 kg/yr	<30 kg/yr	<100 kg/yr	<150kg/yr
	closure	N/R	N/R	N/R	N/R
CO (kg)	construction	5.0E+06	5.0E+06	2.8E+06	5.0E+06
	operation	<5.0E+03	<3.0E+03	<1.0E+05 kg/yr	<1.0E+05 kg/yr
	closure	1.28E+06	1.28E+06	N/R	1.46E+06
Hydrocarbons (kg)	construction	2.7E+05	2.7E+05	1.6E+05	2.7E+05
	operation	<7	<4.2	N/R	<7
	closure	1.38E+05	1.38E+05	N/R	1.57E+05

Table B.11.0.7 Nonradiological Emissions, Tank Waste Alternatives [1] (cont'd)

Emission Type	Phase	Ex Situ/In Situ Combination 1 Alternative	Ex Situ/In Situ Combination 2 Alternative	Phased Implementation Alternative (Phase 1)	Phased Implementation Alternative (Phase 2)
Aldehydes (kg)	construction	2.55E+02	2.55E+02	138	2.55E+02
	operation	N/R	N/R	N/R	N/R
	closure	7.19E+02	7.19E+02	N/R	8.17E+02
Organic Acids (kg)	construction	1.6E+02	1.6E+02	N/A	1.6E+02
	operation	N/R	N/R	N/R	N/R
	closure	N/R	N/R	N/R	N/R

Notes:

[1] Nonradiological emissions taken from WHC 1995c, e, f, g, i, j, n, and Jacobs 1996. Emissions shown for the in situ and ex situ alternatives do not include routine operating emissions. The emissions shown for the operations phase of the No Action and Long-Term Management alternatives are routine tank farm emissions and when annualized, these emissions would apply to the routine operations phase of each alternative.

CO = carbon monoxide
HCl = hydrochloric acid
HF = hydrofluoric acid
HNO_3 = nitric acid
N/A= Not applicable
NH_3 = ammonia
NO_x = nitrogen oxide
N/R= Not reported
PM-10 = particulate matter nominally less than 10μm
SO_x = sulfur oxide
VOC = volatile organic compound

Table B.11.0.8 Transportation Summary by Tank Waste Alternative

Alternative	Rail Transport, Distance (km)		Truck Transport, Distance (km)		Employee Vehicle Distance (km)
	Onsite	Offsite	Onsite	Offsite	
No Action	N/A	N/A	7.86E+04	1.73E+05	2.81E+09
Long-Term Management	0	1.04E+06	4.35E+06	1.61E+06	2.91E+09
In Situ Fill and Cap	0	3.04E+04	3.78E+06	1.38E+06	7.05E+08
In Situ Vitrification	N/A	2.33E+05	3.93E+06	2.66E+06	1.32E+09
Ex Situ Intermediate Separations	N/A	4.81E+05	1.15E+07	7.72E+06	2.31E+09
Ex Situ No Separations (Vitrification)	N/A	6.66E+05	1.01E+07	5.15E+06	1.82E+09
Ex Situ No Separations (Calcination)	N/A	3.21E+05	7.30E+06	3.80E+06	1.82E+09
Ex Situ Extensive Separations	N/A	5.76E+05	1.14E+07	2.88E+07	2.20E+09
Ex Situ/In Situ Combination 1	N/A	2.88E+05	7.81E+06	5.32E+06	1.97E+09
Ex Situ/In Situ Combination 2	N/A	1.79E+05	6.67E+06	3.68E+06	1.45E+09
Phased Implementation (Phase 1)	N/A	1.38E+05	3.74E+05	1.72E+06	4.76E+08
Phased Implementation (Phase 2)	N/A	5.07E+05	1.26E+07	7.51E+06	3.56E+09

Notes:
N/A = Not applicable

Table B.11.0.9 Transportation Summary by Capsule Alternative

Alternative	Rail Transport, Distance (km)		Truck Transport, Distance (km)		Employee Vehicle Distance (km)
	Onsite	Offsite	Onsite	Offsite	
No Action	N/A	N/A	N/A	N/A	2.70E+07
Onsite Disposal	N/A	N/A	6.89E+03	3.92E+04	3.49E+07
Overpack and Ship	N/A	1.47E+04	N/A	2.80E+04	6.50E+06
Vitrify with Tank Waste	N/A	N/A	5.89E+03	2.80E+04	6.50E+06

Notes:
N/A = Not applicable

APPENDIX B
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Appendix C Alternatives Considered but Rejected from Further Evaluation

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ACRONYMS AND ABBREVIATIONS

DOE	U.S. Department of Energy
Ecology	Washington State Department of Ecology
EIS	Environmental Impact Statement
HLW	high-level waste
LAW	low-activity waste
PUREX	Plutonium-Uranium Extraction
TRU	transuranic
TWRS	Tank Waste Remediation System

NAMES AND SYMBOLS FOR UNITS OF MEASURE, RADIOACTIVITY, AND ELECTRICITY/ENERGY

Length		**Area**		**Volume**	
cm	centimeter	ac	acre	cm^3	cubic centimeter
ft	foot	ft^2	square foot	ft^3	cubic foot
in	inch	ha	hectare	gal	gallon
km	kilometer	km^2	square kilometer	L	liter
m	meter	mi^2	square mile	m^3	cubic meter
mi	mile			ppb	parts per billion
				ppm	parts per million
				yd^3	cubic yard

Mass		**Radioactivity**		**Electricity/Energy**	
g	gram	Ci	curie	A	ampere
kg	kilogram	MCi	megacurie (1.0E+06 Ci)	J	joule
lb	pound	mCi	millicurie (1.0E-03 Ci)	kV	kilovolt
mg	milligram	μCi	microcurie (1.0E-06 Ci)	kW	kilowatt
mt	metric ton	nCi	nanocurie (1.0E-09 Ci)	MeV	million electron volts
		pCi	picocurie (1.0E-12 Ci)	MW	megawatt
				V	volt
				W	watt

Temperature	
°C	degrees Centigrade
°F	degrees Fahrenheit

APPENDIX C
ALTERNATIVES CONSIDERED BUT REJECTED FROM FURTHER EVALUATION

C.1.0 INTRODUCTION

This appendix describes the alternatives that were considered but rejected as inappropriate for detailed evaluation in the Tank Waste Remediation System (TWRS) Environmental Impact Statement (EIS). Discussion of additional alternatives, which were suggested by the public during the Draft EIS comment period, is contained in Volume Six, Appendix L. The initial range of technology options potentially applicable for remediating the tank waste and cesium (Cs) and strontium (Sr) capsules was developed by the U.S. Department of Energy (DOE) and the Washington State Department of Ecology (Ecology). The full range of alternatives was evaluated by DOE and Ecology and options that were not appropriate for detailed evaluation in the EIS were rejected. In addition, a number of potential alternatives were suggested by the public during the EIS scoping meetings. These alternatives were also evaluated by DOE and Ecology. The alternatives that were determined to be viable were included as alternatives in the EIS and those alternatives determined to be inappropriate for detailed evaluations were rejected from further consideration. The following criteria were used to determine the appropriateness of an alternative.

- Is the alternative relevant to the purpose and need for agency action in this EIS? If not, then the alternative recommended involves a topic or subject that is not part of this EIS and is not relevant or appropriate for inclusion in this EIS.
- Is the alternative technically viable and practicable?
- Can the alternative be designed to be protective of human health and the environment with practicable mitigative measures?
- Is the technology sufficiently mature to allow detailed evaluation? This criteria refers to technologies that are purely theoretical in their potential application to the TWRS project, and the costs and the time required to develop the technology would be exorbitant.
- Is the technology appreciably different than an alternative already included in the EIS or does it offer potential advantages in terms of effectiveness, costs, or impacts to human health and the environment?

If the answer to any of these questions was no, the alternative was rejected from further consideration in the EIS.

The rejected alternatives are divided into two main categories. The categories are 1) alternatives or technologies identified as potential technology options by DOE and Ecology that did not meet one of the criteria identified previously; and 2) alternatives or technologies proposed by members of the public that did not meet one or more of the criteria identified previously. The following sections discuss the content of the rejected alternative or technology and the reason for rejecting it.

C.2.0 ALTERNATIVES AND OPTIONS DEVELOPED BY DOE AND ECOLOGY

The following alternatives were initially identified by DOE and Ecology as being potentially applicable for remediating the tank waste and capsules; however, they did not meet one or more of the criteria identified in Section C.1.0.

C.2.1 RETRIEVAL AND TRANSFER

Open Tank Mining

This retrieval method pertains to an array of potential technologies that rely on mobile surface- or subsurface-based equipment to penetrate the tank, retrieve the waste, and remove the tank. Because this method of waste retrieval would need to be adapted to a radioactive environment, the extensive redesign of existing equipment and further development would result in exceedingly complex and potentially impractical systems. Consequently, the complexity would defeat the perceived benefits. This alternative was rejected from further consideration because it was not technically viable and practicable.

Drift Tunneling

The drift tunneling concept would insert mining equipment into tunnels bored in the side or bottom of the tank. The waste would be loaded into cars that would transport the waste to the treatment facility (DOE 1995a). This concept had the following disadvantages: 1) it would require a hole in the tank below the surface of the waste; 2) it would not be likely that mining equipment could operate across the full distance of a tank; 3) a tunnel would be dug in contaminated soil; 4) the concept is more complex than a mechanical system; 5) it would be difficult to provide confinement for contaminated soil and waste; and 6) loading, transporting, and decontaminating the cars would be impractical. This alternative was rejected from further consideration because it could not be designed to be protective of human health and the environment with reasonable mitigation measures and was not technically viable or practical.

Drag Arm

The drag arm concept would consist of a chopper pump, with a cutter head used to chop up the waste, operating on a blanket of water above the waste (DOE 1995a). This concept had the following disadvantages: 1) it would require a blanket of water over waste, which would increase the potential of large leaks from the tanks; 2) it would not remove waste that has hardened on the sides and bottoms of the tanks; 3) it would not operate in tanks where equipment was disposed; 4) it would not operate in a tank with numerous risers or in-tank debris; 5) it would not remove waste around stiffening angles at sides of tank; and 6) it would be difficult to operate. This alternative was rejected from further consideration because it was not technically viable and practicable.

Mechanical Dredge

The mechanical dredge concept would consist of a floating dredge device used to scoop up the waste as it was pulled along a positioning arm by a drag cable. The device would operate on a blanket of water positioned over the waste (DOE 1995a). This concept had the following disadvantages: 1) it would not operate in tanks with numerous risers or in-tank debris; 2) it would not remove waste near debris; 3) it would require a blanket of water over the waste, which would increase the potential of large leaks from the tanks; 4) it would not remove waste that has hardened on the sides and bottoms of the tanks; 5) it would not remove waste from around stiffening angles at the sides of tanks; and 6) it would be difficult to operate. This alternative was rejected from further consideration because it was not technically viable and practicable.

Load, Haul, Dump, Elevate

The load, haul, dump, elevate concept would use a self-propelled front loader-type device to scoop up the waste and transport it to a bucket or belt conveyor that would transport it out of the tank (DOE 1995a). This concept had the following disadvantages: 1) it would not operate on an uneven waste surface; 2) it would sink below the surface on soft waste; 3) the use of buckets and belt conveyors would not be suited for remote operation; and 4) it would have difficulty operating around tank risers and other debris. This alternative was rejected from further consideration because it was not technically viable and practicable.

Continuous Miner and Elevator

The continuous miner and elevator concept would use a self-propelled mining system introduced into the tank through a large opening in the top of the tank. The miner mechanism would propel itself around the inside of the tank, mechanically chewing and cutting up the waste then transporting the waste out of the tank with a bucket or belt conveyor (DOE 1995a). This concept had the following disadvantages: 1) a self-propelled vehicle would not work well on an uneven surface of tank waste; 2) a miner would sink below the surface of soft waste; 3) mechanical conveyors would not work remotely; and 4) a continuous miner would have difficulty operating around tank risers. This alternative was rejected from further consideration because it was not technically viable and practicable.

C.2.2 SEPARATIONS (Boomer et al. 1993)

Radio-Frequency Plasma Torch and Plasma Centrifuge

This method of processing would involve separating an ionized plasma stream into heavy and light fractions. The system would consist of a radio-frequency-induced plasma torch dissociator and an electromagnetic plasma centrifuge. The torch would use ionized inert gas to create a plasma dissociation zone where compounds in the feed stream would ionize into their constituent elements. Heavy mass particles would be separated from lighter mass particles in the plasma centrifuge. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation.

Selective Leaching Processes

This process represents an intermediate position between simple water washing and dissolution of the sludge and would involve the selective removal of chemical components or groups of components. Because testing is still in the laboratory phase, this alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation.

Sodium Nitrate Crystallization

This technique would involve partitioning acidified waste solutions into a small volume of sludge and a much larger volume of sodium nitrate. If used, this technology would be applied to aqueous solutions of saltcake. The solution would be adjusted to a pH level of 1 to 2, and the solution would be thermally concentrated to exceed the solubility of sodium nitrate, which is removed by filtration. One perceived technical disadvantage would be the creation of additional sodium nitrate when the solution pH is adjusted. Because laboratory-scale development is currently underway, this alternative was rejected from further consideration because it was not sufficiently mature to allow detailed evaluation.

Precipitation Removal of Transuranic Elements, Strontium-90, and Technetium-99 from Alkaline Solution

This process would involve removing transuranic (TRU) elements, strontium-90 (Sr-90), and technetium-99 (Tc-99) from the alkaline waste by such techniques as hydroxide adjustment, sulfide precipitation, or formation of insoluble phosphates. Because initial laboratory scouting tests are just underway, this alternative was rejected from further consideration because it was not sufficiently mature to allow detailed evaluation.

Nickel Ferrocyanide Precipitation of Cesium-137

This process would co-precipitate cesium-137 (Cs-137) with the addition of nickel salts and ferrocyanide. In the 1950's, Cs-137 was removed on a large scale from alkaline bismuth phosphate waste. The process was later adapted to precipitate Cs-137 from the Plutonium-Uranium Extraction (PUREX) Plant high-level waste (HLW). This alternative was rejected from further consideration because it did not appear to offer a substantial processing advantage over conventional ion exchange techniques.

Sodium Titanate Precipitation from Alkaline Solutions

This process would consist of removing Sr-90 and TRU elements by co-precipitation with sodium titanate in alkaline solutions. This process has been demonstrated on a laboratory scale at the Savannah River Site. The disadvantage of this alternative was that initial test work indicated complexed species are not co-precipitated, meaning that Sr-90 and TRU elements would remain in solution unless the complexing agents were previously destroyed. As a result, this alternative was rejected from further consideration because it was not technically viable and practicable.

Bismuth Phosphate Precipitation of Transuranic Elements

Bismuth phosphate was one of the first processes used in acidic solutions to co-precipitate plutonium and neptunium. The disadvantage of this alternative was that the process would not function properly in alkaline media and would not remove trivalent americium (Am^{+3}) even from acidic solutions. This alternative was rejected from further consideration because it was not technically viable and practicable.

Zirconium Phosphate Sorption

This process would use zirconium phosphate in a manner similar to an ion exchange resin. Zirconium phosphate would form a gelatinous amorphous solid of variable composition, which would adsorb cations because of an electrostatic charge formed at the surface. At present, there is only laboratory experience on this process; however, it is known that zirconium phosphate is unstable in the alkaline solutions such as the tank waste. This alternative was rejected from further consideration because it was not technically viable and practicable.

Molecular Recognition Removal of Transuranic Elements, Technetium, Strontium, and Cesium

This process would consist of extracting TRU elements, Tc, Sr, and Cs by a crown ether fixed on a solid substrate similar to an ion exchange media. This process would be a theoretical adaptation from using crown ethers in liquid-liquid extraction systems. This alternative was rejected from further consideration because it was not sufficiently mature to allow detailed evaluation.

Zeolites

This concept is based on using inorganic ion exchangers to remove Cs-137 from solution. The zeolite would be employed in columns similar to that of conventional ion exchange resins. Because the zeolite could not be eluted by nitric acid, which would destroy the loading capacity, it would be used once and then added to the feed to HLW vitrification. Because of the large increase in volume of HLW glass that would be produced, this alternative was rejected from further consideration because it was not considered technically viable and practicable.

Removal of Cesium-137 and Technetium-99 by Solvent Extraction

Various solvent extraction processes have been demonstrated on a bench scale and in some cases on a pilot scale for removing Cs-137 and Tc-99 from highly basic solutions. This concept had the following disadvantages: 1) the tendency to form aqueous-organic emulsions in alkaline media would lead to incomplete phase separation; 2) the polar solvents required to give acceptable phase separation are often toxic and possibly carcinogenic; and 3) large amounts of nitric acid would possibly be needed for elution. This technology was rejected because it is not considered technically viable and practicable.

Steam Reforming of Volatile Organic Compounds

This process would use the reaction of methane and steam with volatile organics at high temperatures and pressures to produce gaseous products such as carbon monoxide and hydrogen. The organics would be volatilized in fluid bed reactors. This concept had the following disadvantages: 1) many of the complexing agents in the waste would not be volatile and would remain in solution; and 2) high temperatures and flow problems with the waste would possibly cause problems in fluid bed reactors. This alternative was rejected from further consideration because it was not technically viable and practicable.

Oxalate Precipitation

The oxalate ion could be used to precipitate trivalent and quadravalent actinides and trivalent lanthanides from dilute nitric acid solution. The precipitated oxalates would be removed by mechanical means such as filtration. This technology was rejected from further consideration because it was not appreciably different and better than methods addressed in the EIS.

Lanthanum Fluoride Precipitation

This process would be used to precipitate TRU elements and lanthanides by adding hydrofluoric acid to acidified tank waste. The precipitate would subsequently dissolve in a mixture of nitric acid and aluminum nitrate. This alternative was rejected from further consideration because it was not appreciably better than the methods addressed in the EIS.

Antimonic Acid Sorption of Strontium-90

In this process, crystalline antimonic acid would selectively sorb Sr-90 from highly acidic nuclear waste solutions. This concept has not been developed further because laboratory testing has shown that no suitable eluting reagent has been identified. In addition, only small quantities of antimonic acid have been produced. This alternative was rejected from further consideration because it was not technically viable and practicable.

Phosphotungstic Acid Precipitation of Cesium-137

Phosphotungstic acid would precipitate Cs-137 in nitric acid solutions. Plant-scale recovery of Cs-137 from PUREX Plant waste has been routinely performed. The precipitated product has been recovered and subsequently purified. Because this method of precipitation would only remove 95 percent of the Cs-137, leaving 5 percent to be recovered by routine ion exchange methods, this alternative was rejected from further consideration because it was not technically viable and practicable.

Actinide Extraction Using Diamides

This process would consist of solvent extraction methods using diamides, which are bifunctional organic molecules that will extract +3, +4, and +6 actinides from strong nitric acid solutions. This concept is still in the laboratory experimentation phase. Other extractants are expected to provide superior performance. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation.

Actinide Extraction Using Carbamoylmethyl Phosphonate

The carbamoylmethyl phosphonate reagent would extract the same elements as the diamides (i.e., +3, +4, and +6 actinides). However, a more preferred extractant would be carbamoylmethyl phosphine oxide. Carbamoylmethyl phosphine oxide would be a stronger extractant and has been used successfully in bench-scale experimentation. This alternative was rejected from further consideration because it was not technically viable and practicable.

Americium Trivalent Extraction Using Dibutylbutylphosphonate

Dibutylbutylphosphonate is a phosphorus compound that has been proven to be a powerful extractant of Am^{+3} from acid solutions. However, the process development had many difficulties in controlling solution pH during extraction. The diluent employed was carbon tetrachloride, which is highly carcinogenic. This alternative was rejected from further consideration because it was not technically viable and practicable.

Cesium and Strontium Extraction Using Cobalt Dicarbolide

This solvent-extraction process would extract Cs and Sr from nitric acid solutions. Stripping would be accomplished by using strong nitric acid. Russian and Czech processes have been tested using toxic nitrobenzene as the diluent, although essentially no experimental work with dicarbolide extractants has been performed in the United States. This alternative was rejected because it was not technically mature enough for evaluation and was not better than methods addressed in the EIS.

Magnetic Separation and Flotation of Sludge Components

Magnetic separation and flotation of sludge components are physical separation processes that would potentially be applied to sludges to preferentially remove and separate components based on their magnetic characteristics and surface chemistries. The processes are commonly used in the mineral processing industries to separate the components of mined ores. These processes have not been tested for removing selected components from the tank waste sludges. Even in favorable circumstances, a certain percentage of the target material will commonly not be recovered. This alternative was rejected from further consideration because it was not technically viable and practicable.

C.2.3 WASTE TREATMENT FOR ONSITE DISPOSAL OF LOW-ACTIVITY WASTE Boomer et al. 1993)

ectrolytic Denitration of Alkaline Nitrate Solutions

is process, which would use direct current to reduce nitrate in solution, has been the subject of nited investigation. This process was not evaluated because chromium inhibits denitration and toxic ismuth salts must be added to block the inhibiting effect. This alternative was rejected from further onsideration because it was not technically viable and practicable.

irect Calcination of the Low-Activity Waste

n this process the low-activity waste (LAW), without reducing agents such as sugar, would be fed irectly into a calciner that would heat the material sufficiently to decompose carbonates, hydrates, and ther compounds. This process was not selected for detailed evaluation because of the nature of the LAW feed to the calciner. This feed was composed of a major proportion of sodium hydroxide and odium nitrate. The sodium salts would decompose in the calciner and form sodium oxide. Before the dium nitrate and sodium hydroxide could heat sufficiently to calcine, they would melt and the molten lts would create a mush with the other solids in the calciner. This alternative was rejected from rther consideration because it was not technically viable and practicable.

Inorganic Binders Used Directly on Dried Low-Activity Waste
In this process the dried LAW would be mixed with an inorganic binder that would immobilize the dried waste. However, no suitable binder material was identified. Both sulfur and lead had been mentioned as candidate binders. Both of these potential binders presented problems in their application. Sulfur binders may react with sodium nitrate in the waste, which is a powerful oxidizing agent. Lead binders would be expected to be unsatisfactory because the dried salts would float on the lead. In addition, the toxicity of lead would also lead to its rejection as a processing option.
This alternative was rejected from further consideration because it could not be designed to be protective of human health and the environment with reasonable mitigative measures.

Bitumen Binders Used on the Dried Low-Level Waste
For this potential process the LAW would be mixed with a bitumen binder to immobilize the dried waste. This process had the following disadvantages: 1) fire hazard; 2) softening temperature; 3) radiation resistance; and 4) potential reactions of the bitumen with the nitrate in the salts. This alternative was rejected from further consideration because it could not be designed to be protective of human health and the environment with reasonable mitigative measures.

Hot Pressing, Hot Isostatic Pressing, Cold Pressing and Sintering, and Pellitization and Sintering
These compaction processes have been commonly used in industry to agglomerate powders of various kinds such as metals or ceramics. In the hot processing process, the powder first would be compacted with enough force to hold its shape; the compacted shape, then would be heated (in a protective atmosphere if required) until the particles fused at their surfaces and formed a durable shape that would withstand further handling and storage. These processes have only been applied on a laboratory scale. While hot pressing has been used in a demonstration program in Australia, none of these processes have the testing and demonstrated full-scale operation of vitrification. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation.

C.2.4 WASTE TREATMENT FOR OFFSITE DISPOSAL OF HIGH-LEVEL WASTE

Concrete Formed Under Elevated Temperature and Pressure
The ingredients for this process would generally be portland cement, fly ash, sand, clays, and waste (Boomer et al. 1993). This process would use accelerated curing at high temperature and pressure to produce solids that are strong and relatively impermeable. Initial tests on a high sodium nitrate waste produced a waste form that exuded liquid and cracked easily. This process might give more favorable results when the concentration of sodium salts is decreased, but no further test results were available. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation. This waste form would not meet the current waste acceptance systems requirements for the potential geologic repository (DOE 1995q).

Supergrout and Sludge in Concrete
In the supergrout and sludge in concrete processes, additives would be used in the grout to decrease the leachability of radionuclides and improve the properties of the final concrete. Supergrout would be a

:ted from further consideration because the technology was not sufficiently mature to allow detailed luation and is not technically viable and practical.

.ectroosmotic Water Removal from the Saltcake

/uring this process fluids would diffuse through a semipermeable membrane under the influence of an :lectric field. This process has not been analyzed further because of the low mobility of water through salt at low moisture concentrations, and the difficulty in maintaining an effective electric field over large salt volumes. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation.

C.2.6 SPECIALIZED ALTERNATIVES

Seabed Disposal, Space Disposal, Deep Hole Disposal, Ice Sheet Disposal, and Island Disposal

These alternatives would consist of removing the tank waste and capsules from their present locations, packaging them in suitable containers, and transporting them to remote locations for indefinite disposal. These options have been previously investigated for disposal of radioactive waste and have been rejected for further consideration (WHC 1995a). National disposal policy is not within the scope of this EIS.

Geologic Disposal of Tank Contents, Tanks, Equipment, and Contaminated Soil

This alternative would involve removing the tank contents, tanks, ancillary equipment (e.g., pumps, piping), and contaminated soil from their present locations, packaging them in an appropriate manner, and placing them in a suitable potential geologic repository (DOE 1987). Removing the tanks and associated debris is not within the scope of this EIS, but will be evaluated in a future EIS. Therefore, this alternative was rejected from further consideration.

Rock Melting

This alternative would involve pumping HLW into conventionally mined cavities at depths of 1,500 to 1,800 meters (m) (5,000 to 6,000 feet [ft]) (WHC 1995a). The high levels of heat produced by the waste would melt the surrounding rock over time. In time, this melt would resolidify as a low soluble matrix. Using this alternative would require waste that generates extremely high heat. However, the TWRS tank waste (considered as a class) cannot generate the heat required. This alternative was rejected from further consideration because it was not technically viable and practicable, and reevaluating the national HLW disposal policy is not within the scope of this EIS.

Transmutation

This alternative would involve reprocessing the waste by converting it into a form that could be bombarded by radiation, which would convert the long-lived radionuclides into stable or short-lived radioisotopes (WHC 1995a). This alternative had the following potential disadvantages: 1) is anticipated that only 5 to 7 percent of the recycled elements would be transmuted during each reprocessing cycle; 2) it would be expected that it would take up to several decades to develop the advanced technologies that would be required; and 3) it is likely that the fission products would be hazardous and the need for other waste disposal technologies would be necessary. This alternative was grout mixture of waste, special additives, and cement. Sludge in concrete would be HLW directly mixed with grout-forming materials at ambient temperatures and pressures. Waste oxide loadings for these forms have been generally less than those for vitrified products while leaching rates have been greater. These alternatives were rejected from further consideration because they were not technically viable and practicable. This waste form would not meet the current waste acceptance systems requirements for the potential geologic repository (DOE 1995q).

Aqueous Silicates

This waste form would incorporate an alkaline radioactive waste and a clay to form stable aluminosilicate minerals. This process had the following disadvantages: 1) the leaching rate of this waste form exceeded that of other waste forms; 2) immersion in water caused the waste form to crack and swell; and 3) waste loading for these salt forms was less than that for vitrified products. This alternative was rejected from further consideration because it was not technically viable and practicable. This waste form would not meet the current waste acceptance systems requirements for the potential geologic repository (DOE 1995q).

Multiphase High-Level Waste Forms, Including Cement Matrix, Coated Ceramic, Metal Matrix, and Sulfur Matrix

This process would result in a waste form consisting of two parts. The first part would typically be glass or ceramic in the form of marbles or cullet. The second form would be a matrix that covered the glass or ceramic and filled the interstices between the marbles or cullet. No advantage would be gained by using these forms for HLW because the glass or ceramic would be less reactive than the matrix material. The multiphase forms would occupy a higher volume than the glass or ceramic. This waste form would not meet the current waste acceptance systems requirements for the potential geologic repository (DOE 1995q). These alternatives were rejected from further consideration because they could not be designed to be protective of human health and the environment with reasonable mitigative measures.

C.2.5 IN SITU DISPOSAL (Boomer et al. 1991)

Heated Air Drying of Salts

This process would dry the saltcake by inserting a network of piping into the saltcake and forcing large volumes of heated air through the voids in the saltcake. However, excessive pressure would be required to force air through deep layers of the saltcake and could force solution to leak from the tanks. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation.

Resistance Heating and Induction Heating of Salts

During this process resistance heaters or induction coils would be inserted in the saltcake for drying the salts. This process had the following potential disadvantages: 1) poor heat transfer characteristics of the salts would result in excessive heating and possible melting adjacent to the heating elements or induction coils; 2) excessively high power consumptions and current densities would be expected; and 3) induction heating of very large volumes of salts has not been attempted. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation.

C.3.0 ALTERNATIVES IDENTIFIED DURING THE EIS SCOPING PROCESS (DOE 1995b)

The following alternatives were identified by the public during the EIS scoping process as potentially applicable for remediating the tank waste and capsules; however, they did not meet one or more of the criteria identified in Section C.1.0. Section 7.0 identifies issues raised by the public that have been included in this EIS.

C.3.1 WASTE STORAGE AND DISPOSAL

Grout the Retired Canyon Facilities with Hot Grout

This alternative would involve grouting the retired canyon facilities. In this alternative, grout would be the primary tank waste disposal method. Existing grout facilities would be used and the grouted waste would be placed in the retired canyon facilities to harden. This option would leave the HLW onsite in a form that could not be transported to a potential geologic repository. Furthermore, the canyon facilities were designed as chemical processing facilities, not as disposal facilities. Certain areas of the canyons were designed to shield radiation but other areas such as hallways were not. In addition, the canyon facilities were not structurally designed to be filled with grout and the facilities would fail over time. This alternative was rejected from further consideration because it was not technically viable and practicable.

Launch to Sun, Seabed Subduction, and Deep Hole Disposal

The first alternative recommended research to develop technology to launch tank waste to the sun or out of the solar system. The second alternative recommended that canisters of waste be inserted into the sea floor at points of subduction so that the material would eventually be drawn deep into the earth's interior. The third alternative suggested storing the materials several thousand feet down in a stable portion of the continent's thick crust. This could be accomplished by drilling standard oil well holes approximately 3,000 m (10,000 ft) down and then stacking stainless steel canisters on top of each other until they reach a depth of about 1,500 m (5,000 ft). The remaining depth of the holes would be filled with inert material (i.e., cement or clean fill). These alternatives have previously been evaluated for the disposal of commercial nuclear waste and have been rejected (WHC 1995a). Furthermore, national HLW disposal policy is not within the scope of this EIS.

Glass Logs in Grout Vaults, Solids in Tanks to Decay

This alternative would use a furnace to turn the liquid waste from the tanks into glass logs. The logs would be stored in grout vaults so that the Cs-137 could decay to innocuous levels. The tank solids would be left in the tanks to decay. This alternative was composed of two parts, each of which is bounded by the alternatives described in Appendix B. The first part addresses the vitrification of the HLW sludges from the tanks and the storage of the resulting glass product in existing grout vaults. The second portion of the alternative pertains to the decay of radionuclides in the tanks over a period of several hundred years. While this proposed alternative contains elements of the alternatives presented in Appendix B of this EIS, it was not accepted for detailed analysis. The vitrification of the sludge separations from the liquid is addressed in this EIS by the Ex Situ Intermediate Separations and Ex Situ Extensive Separations alternatives. However, storing the resultant HLW glass in the grout vaults would not be acceptable. The HLW glass would receive temporary onsite storage, but would eventually be shipped to the potential geologic repository. The short half-life of Cs-137 would cause it to decay faster than most of the radionuclides in the tank waste. This alternative was rejected from further consideration because it could not be designed to be protective of human health and the environment with reasonable mitigative measures.

Railcar Storage of Tank Waste

This alternative proposed using mobile railcars for transporting and storing tank waste. The alternative would use existing sidings plus new sidings with berms and liners or concrete aprons under the cars. These methods would allow adding early extra storage capacity, storing waste of diverse compositions without mixing, and transporting waste without new pipelines. Railcar storage was not a viable method for consideration in this EIS because 1) storing the tank waste in mobile tank cars would not comply with Federal and State regulations; and 2) using mobile railcars could not conform to the constraints of DOE Order 6430.1a with regard to seismic, safety, and shielding considerations. This alternative was rejected from further consideration because it could not be designed to be protective of human health and the environment with reasonable mitigative measures. Transporting waste by railcar is addressed in the Safe Interim Storage of Hanford Tank Waste EIS (DOE 1995i).

C.3.2 VITRIFICATION

Lead or Stainless-Steel Containers for High-Level Waste

This process would immobilize and dilute the radioactive materials in a glassification process, as appropriate. Following glassification, the treated waste would be encased in lead or stainless-steel containers suitable for long-term storage. Because of its ability to attenuate radiation, lead would seem to be a logical material for consideration in enclosing or surrounding HLW. However, lead is a toxic material with low mechanical strength whose use as a container would be inappropriate if a nontoxic alternate material was available. Stainless steel is such an alternate material and has been used in other countries as a container for HLW glass; it is also the container material proposed for the ex situ alternatives. Lead was rejected for consideration as a container material in the EIS because the technology is not appreciably different or better than those addressed in the EIS in terms of effectiveness, costs, or impacts to human health and the environment.

Unenclosed Furnace in Excavation

This alternative proposed building a 50 ton/day furnace using sodium nitrate from the tank waste liquid phase and making the remainder of the tank waste into a glass. The furnace could be built in an excavation in the ground in the 200 Areas. The commentor suggested that tanks would be necessary but no building would be necessary. This alternative would place the vitrification units belowgrade to alleviate the need for concrete shielding. While placing the treatment facilities belowgrade whenever possible might be considered good design practice, the absence of a roof is not protective of human health and the environment. The roof must be present to shield against radiation leakage and scatter. In addition, the roof serves a vital structural function in protecting against seismic events and preventing outside materials from being blown into the building. This alternative was rejected from further consideration because it could not be designed to be protective of human health and the environment with reasonable mitigative measures.

Placing Marbles or Clinkers Into Casks of Currently Contaminated Steel and Concrete
This alternative would store the vitrified waste product as marbles or clinkers in containers made from materials that have been contaminated in previous operations (i.e., contaminated steel or concrete). While recycling materials is becoming more prevalent in the United States, this particular option has not been accepted for further study in the EIS because the contaminated casks could not be shipped offsite safely without overpacking them, which defeats the purpose of the alternative. The casks made from contaminated material would need to be placed in casks made from noncontaminated material for shipment. This option would also involve constructing an additional shielded processing facility that would become contaminated during use. This alternative was rejected from further consideration because it could not be designed to be protective of human health and the environment with reasonable mitigative measures and was not technically viable or practicable.

Interstitial Space Around Clinkers or Marbles Filled with Lead or Graphite from Material Onsite
This option would use lead or graphite as the matrix material surrounding the clinkers or marbles of the vitrified product. Lead is considered to be a toxic material. In addition, the high density of lead would cause the glass to float, which would reduce its effectiveness in filling the interstices in the glass. At present, no experimental work has been done using graphite as a filler material. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation and was not technically viable or practicable.

C.3.3 WASTE TREATMENT

Burn Waste in a Breeder Reactor or Washington Public Power Supply System Reactor
This alternative suggested burning the waste in a breeder reactor or a Washington Public Power Supply System reactor with a result of 30 years of extra power. Under this concept, selected portions of the TWRS waste would be separated and incorporated into the fuel elements to be used in a breeder or power producing reactor. While certain isotopes in the waste would undergo nuclear decay in such an alternative, the vast majority of the waste would still require some sort of chemical separations and subsequent immobilization. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation and it is not technically viable and practicable.

Separation of Tritium
This option would segregate the tritiated waste from the tank waste and store it until the tritium decayed. As no practicable method has yet been discovered to separate tritium from water, the tritiated waste would not be concentrated. This alternative was rejected from further consideration because it was not technically viable and practicable.

C.3.4 HEALTH RISK, SAFETY, AND MITIGATION

Placing Berms Around Tanks

This alternative proposed placing berms around tanks to avoid the potential for an explosion when waste that contained a mixture of chemicals and nitrogen compounds was vitrified in situ.

Another alternative proposed placing berms around tanks to avoid an explosion in a tank that would cause explosions in other tanks. This alternative would place berms around tanks to avoid explosions in nearby tanks should one of the tanks explode. However, the tanks are situated underground with approximately 6 m (20 ft) of soil fill between them. Should an explosion occur within a tank, the shock wave would have to penetrate the concrete liner of the tank and pass through the soil to affect the other tanks. The presence of a berm on the surface over the tanks would have little effect on this situation. Consequently, this alternative was rejected from further evaluation because it was not technically viable and practicable.

C.3.5 EMISSIONS, EFFLUENTS, AND MONITORING

Use Activated Carbon Filters and Encase Them in Lead or Stainless-Steel Containers

This alternative proposed trapping radioactive gases in activated carbon filters and encasing them in lead and stainless-steel containers that would be suitable for long-term storage. This recommendation was correct in that it anticipates the use of specialized filters to clean the contaminants from the gas streams from the treatment facilities. Activated carbon could be used to remove organic vapors (hydrocarbons) from gas streams. While small concentrations of hydrocarbons could be in the effluent streams from the treatment facilities, a greater concern would be removing radionuclide particles. This is done most efficiently by using high-efficiency particulate air filters as the last element of the gas treatment process. The used high-efficiency particulate air filters would be placed in LAW disposal vaults rather than encasing them in metal, particularly lead, which is a toxic material. Little experimental work has been done using activated carbon on gas streams generated by vitrification. This alternative was rejected from further consideration because the technology was not sufficiently mature to allow detailed evaluation and it was not technically viable and practical.

APPENDIX C
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