

## APPENDIX D WASTE INVENTORIES

This appendix provides additional information about the inventories that drive the proposed alternatives described in Chapter 2 of this *Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS)*. Information provided in this appendix forms the basis for determining the short- and long-term environmental impacts under each alternative, which are described in Chapters 4 and 5, respectively, of this *TC & WM EIS*. Waste inventories supporting the cumulative impacts analysis are not included in Appendix D; they are provided in Appendix S, "Waste Inventories for Cumulative Impact Analyses." Each alternative represents a combination of technologies, processes, and facilities that could accomplish the desired outcome for that alternative. Distribution of the radioactive and chemical constituents of the tank waste among the various waste-form, storage, and management options depends on the technologies and processes used under each alternative. Information on the basis for the chemical and radionuclide composition (1) in the tanks, as well as on equipment and soils and in waste forms; (2) for the decommissioning of the Fast Flux Test Facility; and (3) for the waste management activities at the Hanford Site, including treatment, storage, and disposal of onsite and offsite waste, is provided in Sections D.1, D.2, and D.3, respectively. This information, along with data regarding the technologies and processes that would be used under each alternative, was used as a basis for modeling transport of contaminants in air, water, and soil.

### D.1 TANK CLOSURE ALTERNATIVES

Beginning in 1944, the Federal Government irradiated uranium fuel in nuclear reactors at the Hanford Site (Hanford) to produce plutonium for national defense programs. Uranium and plutonium were recovered from the fuel using a variety of physical and chemical separations processes that generated highly radioactive waste streams. Between 1943 and 1964, the U.S. Department of Energy (DOE) commissioned 12 tank farms containing 149 single-shell tanks (SSTs) to store waste containing the radioactive and chemical constituents. During this time, programs were instituted to recover specific constituents and reduce stored volumes to accommodate production needs. During the 1950s, leakage from the tanks was confirmed. To address this leakage and provide safe storage of the waste, 28 double-shell tanks (DSTs) grouped in 6 additional tank farms were placed in service between 1971 and 1986. Because of the complexity of the production, processing, and waste management operations, the exact radiological and chemical characteristics of each tank are uncertain.

To support the *Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS)* environmental impact analyses, the amounts of radioactive and chemical constituents in the tanks and in leaks, discharges, and waste forms associated with tank operations, retrieval, and closure were estimated. Inventory estimates are presented in the following sections of this appendix:

- D.1.1, Current Tank Inventory of Radioactive and Chemical Constituents
- D.1.2, Tank Ancillary Equipment Waste
- D.1.3, Tank Residual Waste Inventories
- D.1.4, Historical Leaks and Other Releases
- D.1.5, Discharges to Cribs and Trenches (Ditches)
- D.1.6, Tank Waste Retrieval Leaks
- D.1.7, Inventories and Flowsheets
- D.1.8, Distribution of Radioactive Constituents of Potential Concern Under Tank Closure Alternatives

The primary sources of information related to tank inventories and past releases are summarized in the *Inventory and Source Term Data Package* (DOE 2003a), which was developed for this *TC & WM EIS*.

### **D.1.1 Current Tank Inventory of Radioactive and Chemical Constituents**

Constituent concentrations are based on sample data, models, calculations, and engineering assessments. For tanks with no sample data, sample-based templates and engineering templates were used to estimate inventories based on data from tanks containing the same waste type. The estimation methods are summarized in the following paragraphs.

The Best-Basis Inventory (BBI) establishes the inventory of the underground waste storage tanks at Hanford by using sample data, process knowledge, surveillance data, and waste stream composition information from the HDW [Hanford Defined Waste] computer model (Agnew et al. 1997). The BBI is a process that was developed to more fully understand and use the available analytical data for tank samples and use the best available information to estimate tank compositions and inventories. The BBI provides the official estimate of SST and DST contents at Hanford for 24 chemical species and 46 radionuclides. The BBIs are updated on a quarterly basis to incorporate new data and waste transfer information. The BBI used in this environmental impact statement (EIS) reflects the inventory estimates for tank waste as of December 1, 2002. All radionuclides are decayed to January 1, 2001. Sample data that were released and waste transfers that occurred after December 1, 2002, are not included (DOE 2003a).

Sample data that represent the current contents of the tank are the preferred source of waste concentration information for the BBI (DOE 2003a). All of the DSTs and most of the SSTs have been sampled. However, a number of the sampled tanks were analyzed for a limited suite of analytes. In addition, the 23 SSTs listed in Table D-1 either were not sampled or their historical sample data are unusable. Among the 23 unsampled SSTs are 13 of the 18 tanks in the TX tank farm and 6 of the 15 tanks in the SX tank farm. Sampling is not required for retrieval and disposal planning purposes (Simpson, DeFigh-Price, and Banning 1999). Due to these limitations on collected samples, a complete tank inventory cannot be determined based on samples only.

**Table D-1. Unsampled Single-Shell Tanks**

241-B-105	241-BX-102	241-S-108
241-SX-107	241-SX-109	241-SX-110
241-SX-111	241-SX-112	241-SX-114
241-TX-101	241-TX-102	241-TX-103
241-TX-105	241-TX-106	241-TX-108
241-TX-109	241-TX-110	241-TX-111
241-TX-112	241-TX-114	241-TX-115
241-TX-117	241-U-104	

**Source:** DOE 2003a.

Process knowledge concentrations may be derived from information such as historical tank sample data, sample data from other tanks, waste transfers, and chemical additions. Waste-type template concentrations were used when other information was not available. The solid-waste-type templates were based on sample data for a particular waste type, supplemented with process knowledge and waste-type concentrations from the HDW model (Agnew et al. 1997). Liquid-waste-type templates were primarily based on waste-type concentrations from the HDW model, adjusted for process knowledge of mercury and other water-insoluble metals.



Most of the BBI chemical inventories can be traced to sample data or template concentrations based on samples. However, aside from radionuclides such as cesium-137 and its decay product barium-137m; strontium-90 and its decay product yttrium-90; and the isotopes of americium, curium, plutonium, and uranium; the BBI radionuclide inventories are largely based on the HDW model. This is especially true for the SSTs.

However, the BBI does not provide inventory estimates for analytes such as chromium, pertechnetate, polychlorinated biphenyls (PCBs), and volatile and semivolatile organic compounds that may be of concern for retrieval, disposal, and closure purposes. The procedures used to estimate inventories for these constituents are presented in Section D.1.1.3.

Selected tanks are being analyzed for PCBs. To date, 55 tanks have been sampled; 43 showed no PCBs and 12 had positive results for PCBs. These results were used to estimate an inventory across the tank farms. The procedures used to estimate PCB concentrations are presented in Section D.1.1.3.

The BBI includes quantity estimates of 46 radionuclides and 24 chemical constituents. Not all constituents are important in the exposure scenarios used to assess *TC & WM EIS* alternative implementation impacts. Thus, to focus attention on the constituents that control the impacts, DOE performed an initial screening analysis. For radionuclides, groundwater release and direct intrusion scenarios were considered. For the groundwater release screening scenario, only drinking water consumption was considered. Release was assumed to be partition limited, and decay during transport was considered. For the direct intrusion screening scenario, inadvertent soil ingestion and inhalation pathways were considered.

The analysis estimated relative impacts based on distribution of radionuclides in the BBI for all tanks. Radionuclides contributing less than 1 percent of impacts under intruder or well scenarios were eliminated from the detailed analysis. To screen for hazardous chemicals, drinking water ingestion impacts were estimated for the 24 BBI chemical constituents, and those contributing more than 99 percent of impacts were selected for detailed analysis. In addition, reported tank concentrations were reviewed and compared with health-based limits (DOE 2003a); chemical constituents of potential concern (COPCs), when compared with health-based limits (DOE 2003a), were added to the initial list of screened chemicals. The results of this analysis are presented in Table D-2. The screening of the BBI for the groundwater scenarios resulted in reduction of the original set of 46 radionuclides and 24 chemical constituents to a final set of 10 radionuclides and 10 chemical constituents that was used in the analysis of the tank waste. However, a screening of the cumulative impacts analysis data resulted in the addition of other COPCs that are not included in Table D-2. Appendix Q provides details on this screening. As noted in the footnote to Table D-2, americium-241 applies to the inhalation pathway for the intruder analysis scenarios analyzed in Appendix Q. Therefore, americium-241 inventory estimates are not included in Appendix D inventory tables since Appendix D tables apply to the EIS alternatives analysis.

**Table D–2. Constituents Selected for Detailed Analysis**

Radionuclides	Chemicals
Hydrogen-3 (tritium)	Chromium
Carbon-14	Mercury
Strontium-90 <sup>a</sup>	Nitrate
Technetium-99	Lead
Iodine-129	Uranium
Cesium-137 <sup>a</sup>	Acetonitrile
Uranium isotopes	Benzene
Neptunium-237	Butanol (n-butyl alcohol)
Plutonium isotopes	Polychlorinated biphenyls
Americium-241 <sup>b</sup>	2,4,6-Trichlorophenol

<sup>a</sup> Daughter radionuclides for strontium-90 and cesium-137 (yttrium-90 and barium-137m, respectively) are not included; their dose contributions are either incorporated into dose estimates for the parent radionuclide or estimated to be minor.

<sup>b</sup> Applies to intruder analysis scenarios only through inhalation pathway.

#### **D.1.1.1 Current Waste Phase Volume Inventories**

This section summarizes the waste phase volumes in the SSTs and DSTs. There are four main waste phases used in the BBI: retained gas, salt cake, sludge, and supernatant. Salt cake and sludge are often further divided into solid and interstitial liquid phases. Except for retained gas, one or more waste types are associated with each waste phase. Waste types are associated with the waste streams that entered the tank farms from the separations plants or evaporators.

Information such as surveillance data (e.g., waste-surface-level, sludge-level, and liquid-observation-well measurements); in-tank photographs; core-sample extrusion observations; core-sample analyses (to distinguish between the salt cake and sludge waste phases); and waste transfer history were evaluated to determine the waste volumes used in the BBI. Interstitial liquid volumes were calculated using average porosities when no specific information was available for a tank. The volumes of tanks being stabilized by pumping of salt well liquids were estimated prior to the start of pumping and subsequently were adjusted to account for the volume of liquid removed from the tank.

BBI data, as presented in the *Inventory and Source Term Data Package*, provide the waste phase volumes for each tank, as well as a summary of the waste volumes by tank farm and totals for the SSTs and DSTs (DOE 2003a). Note that any retained gas in a tank was assumed to be trapped in the salt cake and sludge waste phases. Supernatant phases have not been found to contain significant quantities of retained gas (Mahoney et al. 1999). As a result, the total salt cake and sludge waste phase tank volume was greater than the values listed in the *Inventory and Source Term Data Package* for tanks containing retained gas (DOE 2003a).

Estimates of current waste volumes and individual tank design (nominal) volumes for each tank farm are presented in Table D–3. Current waste volumes and tank nominal volumes were used in conjunction with estimates of current tank inventories to develop inventory estimates of constituents in past leaks and tank waste retrieval leaks.

**Table D–3. Tank Inventory Volumes**

<b>Tank Farm</b>	<b>Number of Tanks</b>	<b>Location</b>	<b>Nominal Volume of Tank Farms (cubic meters)</b>	<b>Current Volume of Waste<sup>a</sup> (cubic meters)</b>
<b>Single-Shell Tanks</b>				
A	6	200-East Area	22,712	4,338
AX	4	200-East Area	15,142	2,097
B	16	200-East Area	24,908	7,743
BX	12	200-East Area	24,075	5,948
BY	12	200-East Area	34,432	15,789
C	16	200-East Area	24,908	6,653
S	12	200-West Area	34,432	19,777
SX	15	200-West Area	56,781	13,142
T	16	200-West Area	24,908	7,024
TX	18	200-West Area	51,648	24,568
TY	6	200-West Area	17,216	2,398
U	16	200-West Area	24,908	12,153
<b>Total Single-Shell Tanks</b>				<b>121,630</b>
<b>Double-Shell Tanks</b>				
AN	7	200-East Area	30,738	21,181
AP	8	200-East Area	35,129	27,828
AW	6	200-East Area	26,346	16,368
AY	2	200-East Area	7,571	3,257
AZ	2	200-East Area	7,571	7,548
SY	3	200-West Area	13,173	8,979
<b>Total Double-Shell Tanks</b>				<b>85,161</b>
<b>Sum of All Tanks</b>				<b>206,791</b>

<sup>a</sup> Volumes as of December 1, 2002.

**Note:** To convert cubic meters to gallons, multiply by 264.17.

**Source:** Derived from DOE 2003a, 2003b.

### D.1.1.2 Radioactive Best-Basis Inventories

This section summarizes the BBI for each of the screened radioactive constituents in the SSTs and DSTs. The BBI provided the individual tank inventories and concentrations for each waste phase or type. The *Inventory and Source Term Data Package* (DOE 2003a) provided the inventory mass of all of the screened radioactive constituents estimated to be present at each tank farm as of December 1, 2002. Tables D–4 and D–5 summarize the screened radioactive constituent inventories for the SST and DST farms, respectively.

**Table D-4. Single-Shell Tank Radioactive Constituent Inventories by Tank Farm (curies)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Hydrogen-3 (tritium)	3.38×10 <sup>2</sup>	1.30×10 <sup>2</sup>	2.07×10 <sup>1</sup>	1.01×10 <sup>2</sup>	1.33×10 <sup>3</sup>	1.10×10 <sup>2</sup>	1.94×10 <sup>3</sup>	1.32×10 <sup>3</sup>	3.42×10 <sup>1</sup>	2.13×10 <sup>3</sup>	3.50×10 <sup>1</sup>	1.44×10 <sup>3</sup>	8.93×10 <sup>3</sup>
Carbon-14	8.33×10 <sup>1</sup>	6.44×10 <sup>1</sup>	7.88	4.19×10 <sup>1</sup>	5.60×10 <sup>2</sup>	1.58×10 <sup>1</sup>	5.05×10 <sup>2</sup>	2.90×10 <sup>2</sup>	1.48×10 <sup>1</sup>	6.47×10 <sup>2</sup>	7.63	3.56×10 <sup>2</sup>	2.59×10 <sup>3</sup>
Strontium-90	6.52×10 <sup>6</sup>	3.09×10 <sup>6</sup>	1.89×10 <sup>6</sup>	1.30×10 <sup>6</sup>	1.75×10 <sup>6</sup>	9.18×10 <sup>6</sup>	2.52×10 <sup>6</sup>	5.28×10 <sup>6</sup>	3.72×10 <sup>5</sup>	1.17×10 <sup>6</sup>	3.31×10 <sup>5</sup>	9.05×10 <sup>5</sup>	3.43×10 <sup>7</sup>
Technetium-99	6.74×10 <sup>2</sup>	4.13×10 <sup>2</sup>	2.13×10 <sup>2</sup>	3.70×10 <sup>2</sup>	2.54×10 <sup>3</sup>	3.51×10 <sup>2</sup>	2.74×10 <sup>3</sup>	1.76×10 <sup>3</sup>	1.63×10 <sup>2</sup>	3.76×10 <sup>3</sup>	1.02×10 <sup>2</sup>	2.43×10 <sup>3</sup>	1.55×10 <sup>4</sup>
Iodine-129	9.45×10 <sup>-1</sup>	4.81×10 <sup>-1</sup>	8.18×10 <sup>-2</sup>	4.49×10 <sup>-1</sup>	5.55	9.93×10 <sup>-1</sup>	5.93	3.35	1.14×10 <sup>-1</sup>	7.15	1.29×10 <sup>-1</sup>	4.69	2.99×10 <sup>1</sup>
Cesium-137	1.24×10 <sup>6</sup>	6.58×10 <sup>5</sup>	3.58×10 <sup>5</sup>	3.26×10 <sup>5</sup>	2.23×10 <sup>6</sup>	9.93×10 <sup>5</sup>	2.60×10 <sup>6</sup>	2.68×10 <sup>6</sup>	1.65×10 <sup>5</sup>	2.44×10 <sup>6</sup>	5.26×10 <sup>4</sup>	2.32×10 <sup>6</sup>	1.61×10 <sup>7</sup>
Uranium-233, -234, -235, -238	3.29×10 <sup>1</sup>	3.64	2.08×10 <sup>1</sup>	5.09×10 <sup>1</sup>	5.22×10 <sup>1</sup>	4.98×10 <sup>2</sup>	5.18×10 <sup>1</sup>	2.95×10 <sup>1</sup>	2.59×10 <sup>1</sup>	4.79×10 <sup>1</sup>	2.23×10 <sup>1</sup>	3.90×10 <sup>1</sup>	8.75×10 <sup>2</sup>
Neptunium-237	2.20	7.83×10 <sup>-1</sup>	3.38×10 <sup>-1</sup>	7.42×10 <sup>-1</sup>	8.59	5.72	1.12×10 <sup>1</sup>	6.71	2.78×10 <sup>-1</sup>	1.32×10 <sup>1</sup>	2.28×10 <sup>-1</sup>	8.94	5.89×10 <sup>1</sup>
Plutonium-239, -240	3.56×10 <sup>3</sup>	9.83×10 <sup>2</sup>	1.42×10 <sup>3</sup>	2.10×10 <sup>3</sup>	1.32×10 <sup>3</sup>	2.16×10 <sup>4</sup>	5.00×10 <sup>3</sup>	6.99×10 <sup>3</sup>	1.43×10 <sup>3</sup>	1.82×10 <sup>4</sup>	4.04×10 <sup>2</sup>	3.89×10 <sup>3</sup>	6.69×10 <sup>4</sup>

Source: DOE 2003a; SAIC 2011.

**Table D-5. Double-Shell Tank Radioactive Constituent Inventories by Tank Farm (curies)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Hydrogen-3 (tritium)	1.18×10 <sup>2</sup>	1.53×10 <sup>3</sup>	1.70×10 <sup>2</sup>	2.47×10 <sup>1</sup>	1.87×10 <sup>2</sup>	1.09×10 <sup>3</sup>	3.12×10 <sup>3</sup>
Carbon-14	1.93×10 <sup>2</sup>	1.97×10 <sup>2</sup>	8.95×10 <sup>1</sup>	1.65	1.04×10 <sup>1</sup>	3.81×10 <sup>1</sup>	5.29×10 <sup>2</sup>
Strontium-90	1.05×10 <sup>6</sup>	5.20×10 <sup>4</sup>	2.89×10 <sup>5</sup>	6.66×10 <sup>6</sup>	7.95×10 <sup>6</sup>	2.18×10 <sup>5</sup>	1.62×10 <sup>7</sup>
Technetium-99	3.68×10 <sup>3</sup>	4.07×10 <sup>3</sup>	1.86×10 <sup>3</sup>	8.93×10 <sup>1</sup>	2.04×10 <sup>3</sup>	2.46×10 <sup>3</sup>	1.42×10 <sup>4</sup>
Iodine-129	3.81	7.69	2.11	1.42×10 <sup>-1</sup>	1.91	2.66	1.83×10 <sup>1</sup>
Cesium-137	8.46×10 <sup>6</sup>	5.36×10 <sup>6</sup>	3.26×10 <sup>6</sup>	2.89×10 <sup>5</sup>	9.84×10 <sup>6</sup>	2.58×10 <sup>6</sup>	2.98×10 <sup>7</sup>
Uranium-233, -234, -235, -238	7.88	2.85	3.93×10 <sup>1</sup>	3.20	5.67	4.50	6.34×10 <sup>1</sup>
Neptunium-237	8.17	1.43×10 <sup>1</sup>	2.39×10 <sup>1</sup>	5.03	2.70×10 <sup>1</sup>	3.80	8.22×10 <sup>1</sup>
Plutonium-239, -240	4.70×10 <sup>2</sup>	2.48×10 <sup>1</sup>	3.02×10 <sup>3</sup>	2.66×10 <sup>3</sup>	3.52×10 <sup>3</sup>	4.88×10 <sup>3</sup>	1.46×10 <sup>4</sup>

Source: DOE 2003a; SAIC 2011.

### D.1.1.3 Nonradioactive Best-Basis Inventories

The BBI inventory mass for the screened nonradioactive constituents (chromium, mercury, nitrate, lead, and uranium) for each tank farm was provided by the HDW model. Available solid-, liquid-, and gas-phase concentration data were utilized to derive inventory estimates of five additional nonradioactive constituents identified during the screening procedure (DOE 2003a). The calculations and formulas used to estimate inventory masses for those additional constituents are described in the following paragraphs. Estimates of the current tank inventories for the 10 screened chemical constituents at the SST and DST farms are presented in Tables D-6 and D-7, respectively. Due to the reducing environment in the tanks, the nitrite inventory was converted on a molecular-weight basis and added to the nitrate inventory and reported as nitrate.

The volatile constituents acetonitrile, benzene, butanol (n-butyl alcohol), and 2,4,6-trichlorophenol were assumed to be present in the aqueous phase and therefore present in the DST farms. Due to the completion of interim stabilization activities in the SSTs, which removed the remaining secondary quantities of supernatant from the tanks, only small quantities of volatile constituents may be present in the SST farms; for analysis purposes, these small quantities were assumed to be zero. Nonvolatile constituents, such as PCBs, were assumed to be present in both the SST and DST farms.

Estimation of inventory mass for the five screened chemical constituents not included in the BBI used data for waste volume (see Table D-3) and waste phase; concentration of the gas, liquid, or solid phase; density of the phase; and mole fraction. The type of calculation conducted depended upon two factors: the waste phase and the tank farm.

For volatile, nonradioactive constituents with measured liquid-phase concentrations (benzene, 2,4,6-trichlorophenol), the inventory mass is equal to the product of the chemical concentration and the tank farm inventory volume, given in the following equation:

$$M_{BBITF} = [C_{\text{chemical}}] \times V_{BBITF} \times \zeta_m \times \zeta_v$$

where:

$M_{BBITF}$	=	inventory mass for each tank farm, grams
$C_{\text{chemical}}$	=	concentration of the benzene or 2,4,6-trichlorophenol, micrograms per milliliter
$V_{BBITF}$	=	inventory volume for each tank farm, liters
$\zeta_m$	=	conversion constant, grams per microgram
$\zeta_v$	=	conversion constant, milliliters per liter

**Table D-6. Single-Shell Tank Nonradioactive Constituent Inventories by Tank Farm (kilograms)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Chromium	1.62×10 <sup>4</sup>	7.87×10 <sup>3</sup>	1.11×10 <sup>4</sup>	2.20×10 <sup>4</sup>	7.34×10 <sup>4</sup>	5.60×10 <sup>3</sup>	1.20×10 <sup>5</sup>	1.05×10 <sup>5</sup>	1.21×10 <sup>4</sup>	6.13×10 <sup>4</sup>	7.95×10 <sup>3</sup>	5.11×10 <sup>4</sup>	4.95×10 <sup>5</sup>
Mercury	1.59×10 <sup>2</sup>	4.27×10 <sup>1</sup>	1.38×10 <sup>2</sup>	2.27×10 <sup>2</sup>	1.74×10 <sup>2</sup>	3.93×10 <sup>2</sup>	7.15×10 <sup>1</sup>	1.46×10 <sup>2</sup>	1.99×10 <sup>1</sup>	2.83×10 <sup>1</sup>	2.56×10 <sup>2</sup>	2.55×10 <sup>1</sup>	1.68×10 <sup>3</sup>
Nitrate	1.41×10 <sup>6</sup>	7.63×10 <sup>5</sup>	1.90×10 <sup>6</sup>	1.73×10 <sup>6</sup>	6.62×10 <sup>6</sup>	6.56×10 <sup>5</sup>	1.10×10 <sup>7</sup>	6.62×10 <sup>6</sup>	7.47×10 <sup>5</sup>	1.40×10 <sup>7</sup>	8.37×10 <sup>5</sup>	5.46×10 <sup>6</sup>	5.18×10 <sup>7</sup>
Lead	4.02×10 <sup>3</sup>	1.26×10 <sup>3</sup>	6.69×10 <sup>3</sup>	3.66×10 <sup>3</sup>	5.12×10 <sup>3</sup>	2.32×10 <sup>4</sup>	2.23×10 <sup>3</sup>	1.75×10 <sup>3</sup>	4.34×10 <sup>3</sup>	7.12×10 <sup>3</sup>	1.39×10 <sup>3</sup>	1.08×10 <sup>4</sup>	7.16×10 <sup>4</sup>
Uranium	1.10×10 <sup>4</sup>	1.48×10 <sup>3</sup>	2.86×10 <sup>4</sup>	7.35×10 <sup>4</sup>	6.55×10 <sup>4</sup>	1.13×10 <sup>5</sup>	5.19×10 <sup>4</sup>	3.27×10 <sup>4</sup>	3.72×10 <sup>4</sup>	4.56×10 <sup>4</sup>	3.24×10 <sup>4</sup>	4.97×10 <sup>4</sup>	5.42×10 <sup>5</sup>
Acetonitrile	0	0	0	0	0	0	0	0	0	0	0	0	0
Benzene	0	0	0	0	0	0	0	0	0	0	0	0	0
Butanol	0	0	0	0	0	0	0	0	0	0	0	0	0
PCBs	3.05×10 <sup>1</sup>	1.47×10 <sup>1</sup>	5.44×10 <sup>1</sup>	4.18×10 <sup>1</sup>	1.11×10 <sup>2</sup>	4.67×10 <sup>1</sup>	1.39×10 <sup>2</sup>	9.23×10 <sup>1</sup>	4.93×10 <sup>1</sup>	1.73×10 <sup>2</sup>	1.68×10 <sup>1</sup>	8.53×10 <sup>1</sup>	8.54×10 <sup>2</sup>
2,4,6-TCP	0	0	0	0	0	0	0	0	0	0	0	0	0

Note: To convert kilograms to pounds, multiply by 2.2046.

Key: butanol=n-butyl alcohol; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

Source: DOE 2003a; SAIC 2011.

**Table D-7. Double-Shell Tank Nonradioactive Constituent Inventories by Tank Farm (kilograms)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Chromium	1.85×10 <sup>4</sup>	1.03×10 <sup>4</sup>	1.99×10 <sup>4</sup>	2.79×10 <sup>3</sup>	5.09×10 <sup>3</sup>	4.73×10 <sup>4</sup>	1.04×10 <sup>5</sup>
Mercury	4.66	0	2.09×10 <sup>-1</sup>	1.26×10 <sup>2</sup>	4.15	8.95	1.44×10 <sup>2</sup>
Nitrate	6.47×10 <sup>6</sup>	5.65×10 <sup>6</sup>	3.47×10 <sup>6</sup>	1.70×10 <sup>5</sup>	7.74×10 <sup>5</sup>	2.48×10 <sup>6</sup>	1.90×10 <sup>7</sup>
Lead	3.63×10 <sup>3</sup>	9.01×10 <sup>2</sup>	1.51×10 <sup>3</sup>	4.48×10 <sup>3</sup>	4.03×10 <sup>2</sup>	1.57×10 <sup>3</sup>	1.25×10 <sup>4</sup>
Uranium	2.68×10 <sup>3</sup>	1.23×10 <sup>3</sup>	3.95×10 <sup>4</sup>	3.52×10 <sup>3</sup>	5.19×10 <sup>3</sup>	2.38×10 <sup>3</sup>	5.45×10 <sup>4</sup>
Acetonitrile	7.33×10 <sup>3</sup>	9.63×10 <sup>3</sup>	5.67×10 <sup>3</sup>	1.13×10 <sup>3</sup>	2.61×10 <sup>3</sup>	3.11×10 <sup>3</sup>	2.95×10 <sup>4</sup>
Benzene	5.97×10 <sup>-1</sup>	7.85×10 <sup>-1</sup>	4.62×10 <sup>-1</sup>	9.19×10 <sup>-2</sup>	2.13×10 <sup>-1</sup>	2.53×10 <sup>-1</sup>	2.40
Butanol (n-butyl alcohol)	8.59×10 <sup>5</sup>	1.13×10 <sup>6</sup>	6.63×10 <sup>5</sup>	1.32×10 <sup>5</sup>	3.06×10 <sup>5</sup>	3.64×10 <sup>5</sup>	3.45×10 <sup>6</sup>
Polychlorinated biphenyls	2.07×10 <sup>2</sup>	2.71×10 <sup>2</sup>	1.60×10 <sup>2</sup>	3.18×10 <sup>1</sup>	7.36×10 <sup>1</sup>	8.76×10 <sup>1</sup>	8.31×10 <sup>2</sup>
2,4,6-Trichlorophenol	2.75×10 <sup>-1</sup>	3.62×10 <sup>-1</sup>	2.13×10 <sup>-1</sup>	4.23×10 <sup>-2</sup>	9.81×10 <sup>-2</sup>	1.17×10 <sup>-1</sup>	1.11

Note: To convert kilograms to pounds, multiply by 2.2046.

Source: DOE 2003a; SAIC 2011.

For the volatile, nonradioactive constituents with measured vapor-phase concentrations (acetonitrile and butanol [n-butyl alcohol]), four calculations needed to be performed. First, the vaporization pressure was calculated using Antoine's equation. Second, the measured gas-phase concentration was converted to partial pressure using the ideal gas law. Next, Raoult's law was used to determine the molar fraction of the species in the liquid phase. The final calculation for determining the inventory mass was the product of the mole fraction and the ratio of the species' molar mass over the water molar mass times the tank farm inventory volume.

For the first step of the estimation procedure, the equilibrium partial pressure of the constituent was calculated using Antoine's equation:

$$\ln P_{\text{vap}} = A - \frac{B}{T + C}$$

where:

- $P_{\text{vap}}$  = pressure of vaporization for the chemical, millimeters of mercury, at 25 degrees Celsius ( $^{\circ}\text{C}$ ) (77 degrees Fahrenheit [ $^{\circ}\text{F}$ ])
- $A, B, C$  = constants for each chemical (SAIC 2011)
- $T$  = temperature of the chemical after transfer to DST, assumed to be 298 kelvins

In the second step, the partial pressure of the constituent in the vapor phase was calculated using the ideal gas law:

$$P_{\text{partial}} = \lambda_P \times \frac{1}{(\lambda_V + \lambda_M)} \times \frac{[C_{\text{chemical}}] \times R \times T}{MW}$$

where:

- $P_{\text{partial}}$  = partial pressure of the chemical, millimeters of mercury
- $\lambda_P$  = conversion constant for pressure, 760 millimeters of mercury per atmosphere
- $\lambda_V$  = conversion constant for volume, 1,000 liters per cubic meter
- $\lambda_M$  = conversion constant for mass 1,000 milligrams per gram
- $C_{\text{chemical}}$  = concentration of acetonitrile or butanol (n-butyl alcohol), milligrams per cubic meter
- $R$  = gas constant, assumed to be 0.082 liter-atmospheres per mole-kelvin
- $T$  = temperature of the chemical, assumed to be 298 kelvins
- $MW$  = molecular weight of the chemical species, grams per mole

Next, Raoult's law was utilized to calculate the mole fraction of the constituent in the liquid phase through the following equation:

$$x = \frac{P_{\text{partial}}}{P_{\text{vap}}}$$

where:

$x$  = mole fraction

$P_{\text{partial}}$  = partial pressure of the chemical, millimeters of mercury

$P_{\text{vap}}$  = pressure of vaporization for the chemical, millimeters of mercury

Finally, the mole fraction was converted to a mass fraction and used to calculate the inventory mass, using the following equation:

$$M_{BBITF} = x \times \frac{MW_{\text{species}}}{MW_{\text{H}_2\text{O}}} \times \rho_l \times V_{BBITF} \times 10^3$$

where:

$M_{BBITF}$  = inventory mass for each tank farm, grams

$x$  = mole fraction

$MW_{\text{species}}$  = molecular weight of the chemical species, grams per mole

$MW_{\text{H}_2\text{O}}$  = molecular weight of water, grams per mole

$\rho_l$  = density of the tank farm liquid, grams per cubic centimeter (SAIC 2011)

$V_{BBITF}$  = inventory volume for each tank farm, liters

For nonvolatile, nonradioactive constituents (i.e., PCBs), the inventory masses are equivalent to the products of the phase concentration times the density of the phase times the tank farm inventory volume. Even though PCBs were not detected in some tanks in certain tank farms, the inventory data were spread across all of the tank farms to ensure the maximum mass was determined. PCBs have been detected in 12 of 55 tanks sampled. The average concentration of 14 solid-phase samples (DOE 2003a) is 7.80 micrograms per gram.



Using this average concentration and assuming that PCBs are present in all tanks, the inventory of each tank farm was estimated using the following equation:

$$M_{BBITF} = [C_{\text{chemical}}] \times \rho_s \times V_{BBITF} \times \zeta_m \times \zeta_v$$

where:

$M_{BBITF}$	=	inventory mass for each tank farm, grams
$C_{\text{chemical}}$	=	concentration of the PCBs, micrograms per gram
$\rho_s$	=	density of the solid, grams per cubic centimeter (SAIC 2011)
$V_{BBITF}$	=	inventory volume for each tank farm, liters
$\zeta_m$	=	conversion constant, grams per microgram
$\zeta_v$	=	conversion constant, cubic centimeters per liter

#### D.1.1.4 Uncertainty in Best-Basis Inventories

The BBI process follows protocols developed to combine differing types of measurements and estimates to produce the most reliable estimate of inventory. However, the high-level radioactive waste (HLW) tank inventory estimates contain considerable uncertainty regarding the number and quality of the available measurements and the estimation procedures that were used in the absence of measurements. As described in Section D.1.1, the HLW tank inventory estimates were based on waste composition and phase volume measurements, process knowledge calculations, and waste-type templates that were developed based on the sample data and model estimates (Field and Bowen 2003). Six types of waste phases were considered in developing these estimates: supernatant, salt cake solids, salt cake liquids, sludge solids, sludge liquids, and gas. Process knowledge calculations included correlation with a known constituent such as a parent radionuclide with a well-established ratio of parent-to-progeny concentration. The model-derived waste template estimates used in this *TC & WM EIS* were based on Revision 4 of the HDW model (Agnew et al. 1997). Sample analysis data provided the preferred bases for the estimates; calculated and template-based information were assigned lower priority. In each case, the inventory estimates were derived as the products of waste density, volume, and composition.

The uncertainty in the measurement-based estimation is due to the limited number of available samples, the complex nature of the tank contents, and the number of transfers and process activities used to manage the waste. The number of available samples is limited due to safety issues and the cost of obtaining them. Because waste phases are not uniform in nature and may be mixed to some extent, estimates of phase volumes and constituent concentrations are uncertain due to measurement and spatial variability. Processing and transfers designed to increase safety and optimize tank utilization produce additional variation among individual tanks. Estimating inventories using process modeling involves consideration of reactor fuel irradiation and chemical separations operations, as well as transfer and processing of tank contents. Incomplete knowledge of the degree of irradiation, process extraction and separations efficiencies, plant stream flow rates that affect recovery and distribution, and process losses to the environment all contribute to uncertainty in developing process modeling inventory estimates.

Quantitative estimates of inventory variability were expressed using the relative standard deviation (RSD), which is defined as the ratio of the standard deviation to the mean. RSD values were estimated from tank sample or sample-based template data. Sample-based templates are averages of samples taken from tanks whose contents are similar in composition to the contents of the tank for which samples are not available. The RSD values for density were reported as 5.9 and 8.2 percent for SST and DST liquids, respectively, and 7.6 and 6.5 percent for SST and DST solids, respectively (Field and Bowen 2003).

Because of the difficulty in determining the extent of phase volumes and in measuring volume, RSD values for volume were based on quantitative and qualitative information. Five data sources were used to estimate volume: surface-level, conductivity probe, sludge-level, liquid-observation well measurement, and core profiles. For the supernatant, salt cake, and sludge tanks, the average standard deviations of the surface-level readings were 0.64, 29.2, and 10.9 centimeters (0.25, 11.5, and 4.3 inches), corresponding to 2.6, 120, and 45 cubic meters (687; 31,700; and 11,888 gallons), respectively. The RSD for estimating volume was calculated as the standard deviation of the level divided by the total height of the waste in the tank. The mean concentration and its RSD were estimated for the constituents in each waste phase based on sample data. Estimates of concentration RSD based on sample data can range from 0 to 100 percent, while those based on waste-type templates can be much larger (Field and Bowen 2003). Median sample-based RSDs for the inventories in the SSTs and DSTs were calculated at 20 and 29 percent, respectively (DOE 2003a). Median template-based RSDs for inventories of SSTs and DSTs were calculated at 164 and 182 percent, respectively (DOE 2003a).

The above information on RSDs for density, volume, and concentration was combined to develop estimates of inventory RSDs for the individual constituents contained within each tank and at each tank farm (DOE 2003a). For four long-lived radionuclides that are important in determining groundwater impacts—technetium-99, iodine-129, uranium-238, and neptunium-237—RSDs for inventories at the tank farm level ranged from 70 to 231 percent, 44 to 231 percent, 77 to 453 percent, and 46 to 473 percent, respectively. Further quantitative estimates would require assumptions that cannot be fully tested using the current data. For example, regarding the assumption of normal data distribution, the 95th percentile upper confidence limit of the technetium-99 inventories in individual tank farms ranged from 2.2 to 5.6 times the BBI estimate across the 18 tank farms. For the combination of variances, the 95th percentile upper confidence limit of the total technetium-99 inventory was approximately 20 percent greater than the BBI estimate.

The above considerations indicate that greater uncertainty is involved in estimating the inventories of individual tanks and tank farms than in estimating total inventory, and greater uncertainty is associated with using template-based estimates than using sample-based estimates.

#### **D.1.1.5 Best-Basis Inventory Comparison**

As required by the *Technical Guidance Document for Tank Closure Environmental Impact Statement Vadose Zone and Groundwater Revised Analyses (Technical Guidance Document)* (DOE 2005), Table D-8 compares the radioactive and chemical COPCs for the December 2002 BBI estimate (DOE 2003a) with the October 2010 update to the BBI estimate. The October 2010 BBI update was the latest current update to the BBI available for review during the production of this final EIS. For comparison purposes, the table includes the radionuclides decayed to the same date, January 1, 2008.

The differences noted in Table D-8 are primarily due to the BBI Improvement Initiative, which was implemented after December 2002, and included the following:

- Updated the ORIGEN2 [Oak Ridge Isotope Generation and Depletion Code] (Croff 1980) fuel activity estimates
- Updated the HDW model to Revision 5, which accounted for a release of hydrogen-3 (tritium), carbon-14, and iodine-129 to the environment and shipment of offsite technetium-99 with uranium
- Updated the BBI templates with new sample data that added a second type of REDOX [Reduction-Oxidation] Facility waste to the SX tank farm, resulting in an increase in SX tank farm inventory

- Eliminated noncredible sample detection limit values from inventory estimates
- Incorporated new sample data, including iodine-129 analysis of BY tank farm salt cake (CEES 2011)

**Table D-8. Best-Basis Inventory Comparison of Constituents of Potential Concern**

	<b>Column 1 December 2002 BBI, Decay Date: January 1, 2001</b>	<b>Column 2 December 2002 BBI, Decay Date: January 1, 2008</b>	<b>Column 3 October 2010 BBI, Decay Date: January 1, 2008</b>	<b>Percent Change from Decayed December 2002 BBI: (Column 3 – Column 2) /Column 2</b>
<b>Radionuclides</b>	<b>Curies</b>	<b>Curies</b>	<b>Curies</b>	<b>Percent</b>
Americium-241	1.45×10 <sup>5</sup>	1.45×10 <sup>5</sup>	1.56×10 <sup>5</sup>	+7.6
Carbon-14	3.12×10 <sup>3</sup>	3.12×10 <sup>3</sup>	5.59×10 <sup>2</sup>	-82.1
Cesium-137	4.58×10 <sup>7</sup>	3.90×10 <sup>7</sup>	3.88×10 <sup>7</sup>	-0.5
Hydrogen-3 (tritium)	1.21×10 <sup>4</sup>	8.16×10 <sup>3</sup>	2.84×10 <sup>3</sup>	-65.2
Iodine-129	4.82×10 <sup>1</sup>	4.82×10 <sup>1</sup>	2.94×10 <sup>1</sup>	-39.0
Neptunium-237	1.41×10 <sup>2</sup>	1.41×10 <sup>2</sup>	1.19×10 <sup>2</sup>	-15.6
Plutonium-239, -240	8.14×10 <sup>4</sup>	8.14×10 <sup>4</sup>	5.95×10 <sup>4</sup>	-26.9
Strontium-90	5.05×10 <sup>7</sup>	4.27×10 <sup>7</sup>	4.76×10 <sup>7</sup>	+11.5
Technetium-99	2.97×10 <sup>4</sup>	2.97×10 <sup>4</sup>	2.64×10 <sup>4</sup>	-11.1
Uranium-233, -234, -235, -238	9.38×10 <sup>2</sup>	9.38×10 <sup>2</sup>	1.14×10 <sup>3</sup>	+21.7
<b>Chemicals</b>	<b>Kilograms</b>	<b>Kilograms</b>	<b>Kilograms</b>	<b>Percent</b>
Acetonitrile	NR	NR	NR	N/A
Benzene	NR	NR	NR	N/A
Butanol (n-butyl alcohol)	NR	NR	NR	N/A
Chromium	5.98×10 <sup>5</sup>	5.98×10 <sup>5</sup>	5.90×10 <sup>5</sup>	-1.3
Lead	8.41×10 <sup>4</sup>	8.41×10 <sup>4</sup>	8.27×10 <sup>4</sup>	-1.7
Mercury	1.83×10 <sup>3</sup>	1.83×10 <sup>3</sup>	1.99×10 <sup>3</sup>	+8.7
Nitrate <sup>a</sup>	7.09×10 <sup>7</sup>	7.09×10 <sup>7</sup>	7.14×10 <sup>7</sup>	+0.7
Uranium (total)	5.97×10 <sup>5</sup>	5.97×10 <sup>5</sup>	6.45×10 <sup>5</sup>	+8.0
Polychlorinated biphenyls	NR	NR	NR	N/A
2,4,6-Trichlorophenol	NR	NR	NR	N/A

<sup>a</sup> Nitrate values calculated as nitrate plus nitrite (oxidized).

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** BBI=Best-Basis Inventory; N/A=not applicable; NR=not reported.

**Source:** CEES 2011; SAIC 2011.

Additional review of the uncertainty in the BBI estimate found the following:

- The uncertainties in the BBI estimate are important. The relative standard deviations derived for four key radionuclides (Section D.1.1.4) suggest that the inventories at the tank farm level (which form the basis for the alternatives' impacts analyses) support an uncertainty around the estimated inventory of approximately 50 to 400 percent.
- In general, the uncertainties for chemical constituents are lower than those for radionuclides. Inventories of chemical constituents are mostly sample-based, whereas radionuclide inventories rely to a greater degree on model calculations (e.g., the HDW model), engineering assessment, and the use of templates.

- A comparison of the uncertainties estimated for the BBI with another primary source of inventory data used in this EIS, SIM [the Hanford Soil Inventory Model] (Corbin et al. 2005), found similar estimates of uncertainties. Inventories of typical liquid waste streams disposed to the soil column in SIM are estimated to have variation around the mean of approximately 50 to 200 percent. The difference in the upper bounds (200 versus 400) is considered small when compared with the goal of this EIS of estimating impacts to an order of magnitude (10 times).

Additional review of the changes in the BBI estimates over an 8-year period between December 2002 and October 2010 found the following:

- All of the radionuclide COPC estimates, except those for americium-241, strontium-90, and uranium-233, -234, -235, and -238, have decreased since the 2002 BBI estimate was generated. The only radiological risk drivers in this grouping are the uranium isotopes.
- Only two chemical COPC estimates, nitrate and total uranium, have increased, with only total uranium a chemical hazard driver. All the other chemical constituents have decreased.
- Trend analysis of the data found consistency in both the increases and decreases in the estimates and within the uncertainties. Overall, their values have changed very little. For example, the largest decrease in the radiological risk drivers was that of iodine-129, at 65.2 percent. There were no decreases in chemical hazard drivers. The largest increases in the radiological risk drivers were those of uranium-233, -234, -235, and -238 (the dominant increase being in uranium-233), at 21.7 percent. The largest increase in chemical hazard drivers was that of total uranium, at 8.0 percent.
- Technetium-99 and iodine-129 values in the 2002 BBI are near the higher end of the range of values reported between December 2003 and October 2010. The range of values reported between December 2003 and October 2010 is consistent with the uncertainty estimates as well. Continued use of the 2002 BBI estimate for these radionuclides represents a degree of conservatism for key radionuclides, which is considered appropriate.
- Total uranium and uranium isotope values in the 2002 BBI are near the lower end of the ranged of values reported between December 2003 and October 2010. The range of values reported between December 2003 and October 2010 is smaller than the uncertainty itself. In addition, uranium (total) and uranium isotopes are more highly retained in soil and waste forms and less mobile in the environment, and their contribution to the long-term impacts analysis should be smaller than the contribution from the mobile radionuclides (i.e., technetium-99 and iodine-129).

In summary, DOE's decision to continue to use the 2002 BBI for tank waste inventory data in this EIS is based in part on the results of a 2005 review of the 2002 BBI estimates by the Washington State Department of Ecology (Ecology) and several DOE offices, i.e., Office of River Protection; Richland Operations Office; Office of Health, Safety, and Security; Office of Environmental Management; and Office of the General Counsel and in part on a review of the 8-year span of BBI data and uncertainty. The conclusion then, and now, is that the 2002 BBI is appropriate for the analyses in this *TC & WM EIS*. The above review of the BBI estimates from 2002 through 2010 supports this conclusion, as does Section 4.0, Assumptions, in the *Technical Guidance Document* (DOE 2005), dated March 25, 2005, which was approved by DOE and Ecology (SAIC 2011).

### **D.1.2 Tank Ancillary Equipment Waste**

This section presents the estimated waste inventories contained in the ancillary facilities that are currently part of the SST and DST systems. Ancillary equipment includes miscellaneous underground storage tanks (MUSTs) (i.e., vaults); SST system tanks; DST system tanks; and the evaporators, evaporator tanks and vessels, pits, and transfer piping (DOE 2003a, 2003b) associated with the SST and DST farms.

Approximately one-half of the total waste volume estimate for ancillary equipment is credited to the MUSTs. Identification, dimensions, and locations of the MUSTs have been documented (DOE 2003a, 2003b). The reported capacities of the MUSTs range from 10 cubic meters (2,640 gallons) to 190 cubic meters (50,200 gallons) (DOE 2003b).

The pits include heel, pump, salt well, sluice, flush, and valve pits and diversion boxes. The SST farm volumes were derived by assuming a deposition of waste solids with an average thickness of only about 0.01 to 0.02 centimeters (0.004 to 0.008 inches) on the surfaces of the pits and piping (DOE 2003a). Waste volumes for the pits in the DST system were estimated by multiplying the waste volumes in the SST system pits by the ratio of DST system pit surface area to the SST system pit surface area. Waste volumes for the piping in the DST system were estimated in a similar manner. Volumes and surface areas were developed based on extrapolating information from detailed analyses of three SST farms and applying it to the other tank farms. DST void volumes in piping and structures were based on measurements of the six SSTs in the 241-A tank farm, which were then multiplied by a factor of 28/6 to obtain volumes for all 28 DSTs.

For analysis purposes, the volume of waste in the ancillary equipment for a given tank farm was calculated as the product of total waste volume in ancillary equipment for all tank farms times the number of tanks in a given tank farm, divided by the total number of tanks in the entire SST and DST system. Additionally, the concentration of each waste constituent in the ancillary equipment for a given tank farm was assumed to be the same as the average concentration of that constituent in the corresponding tank farm waste.

The inventories of radioactive and nonradioactive waste constituents in the ancillary equipment for a given tank farm were therefore calculated as the volume of waste in the ancillary equipment for that tank farm times the concentrations of each of those constituents in the waste currently stored in that tank farm. For example, the inventory of chromium in the ancillary equipment for the S tank farm was calculated as the volume of waste in the ancillary equipment for the S tank farm times the BBI mass of chromium in the S tank farm waste, divided by the volume of waste currently stored in the S tank farm. Radionuclide inventories were calculated similarly, with inventories and concentrations expressed in terms of curies rather than grams.

Volumes of ancillary equipment waste and quantities of individual constituents were estimated, respectively, as follows:

$$V_{\text{anc}} = \frac{V_{\text{anc\,tot}}}{N_{\text{tot}}} \times N_{\text{tanks}}$$

and

$$M_{\text{anc}} = \frac{M_{\text{BBITF}}}{V_{\text{BBITF}}} \times V_{\text{anc}}$$

where:

- $V_{\text{anc}}$  = ancillary equipment volume for each tank farm, liters
- $V_{\text{anc\,tot}}$  = total ancillary equipment volume, liters
- $N_{\text{tot}}$  = total number of tanks
- $N_{\text{tanks}}$  = number of tanks in tank farm
- $M_{\text{anc}}$  = waste inventory ancillary equipment for each tank farm, grams or curies
- $M_{\text{BBITF}}$  = waste inventory mass for each tank farm, grams or curies
- $V_{\text{BBITF}}$  = inventory volume for each tank farm, liters

Tables D-9 and D-10 represent the ancillary equipment waste radioactive and nonradioactive constituents for the SSTs, respectively. Tables D-11 and D-12 represent the ancillary equipment waste radioactive and nonradioactive constituents for the DSTs, respectively.

**Table D-9. Single-Shell Tank Ancillary Equipment Radioactive Constituent Inventories (curies)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
H-3 (tritium)	2.74	1.46	2.51×10 <sup>-1</sup>	1.20	5.94	1.55	6.92	8.83	4.58×10 <sup>-1</sup>	9.16	5.14×10 <sup>-1</sup>	1.12×10 <sup>1</sup>	5.02×10 <sup>1</sup>
C-14	6.77×10 <sup>-1</sup>	7.21×10 <sup>-1</sup>	9.56×10 <sup>-2</sup>	4.96×10 <sup>-1</sup>	2.50	2.22×10 <sup>-1</sup>	1.80	1.94	1.98×10 <sup>-1</sup>	2.78	1.12×10 <sup>-1</sup>	2.75	1.43×10 <sup>1</sup>
Sr-90	5.29×10 <sup>4</sup>	3.46×10 <sup>4</sup>	2.29×10 <sup>4</sup>	1.54×10 <sup>4</sup>	7.82×10 <sup>3</sup>	1.29×10 <sup>5</sup>	8.98×10 <sup>3</sup>	3.54×10 <sup>4</sup>	4.98×10 <sup>3</sup>	5.02×10 <sup>3</sup>	4.86×10 <sup>3</sup>	6.99×10 <sup>3</sup>	3.29×10 <sup>5</sup>
Tc-99	5.47	4.63	2.59	4.38	1.13×10 <sup>1</sup>	4.95	9.76	1.18×10 <sup>1</sup>	2.18	1.62×10 <sup>1</sup>	1.50	1.88×10 <sup>1</sup>	9.35×10 <sup>1</sup>
I-129	7.67×10 <sup>-3</sup>	5.38×10 <sup>-3</sup>	9.92×10 <sup>-4</sup>	5.31×10 <sup>-3</sup>	2.47×10 <sup>-2</sup>	1.40×10 <sup>-2</sup>	2.11×10 <sup>-2</sup>	2.25×10 <sup>-2</sup>	1.52×10 <sup>-3</sup>	3.07×10 <sup>-2</sup>	1.89×10 <sup>-3</sup>	3.62×10 <sup>-2</sup>	1.72×10 <sup>-1</sup>
Cs-137	1.00×10 <sup>4</sup>	7.37×10 <sup>3</sup>	4.34×10 <sup>3</sup>	3.86×10 <sup>3</sup>	9.96×10 <sup>3</sup>	1.40×10 <sup>4</sup>	9.26×10 <sup>3</sup>	1.79×10 <sup>4</sup>	2.20×10 <sup>3</sup>	1.05×10 <sup>4</sup>	7.73×10 <sup>2</sup>	1.79×10 <sup>4</sup>	1.08×10 <sup>5</sup>
U-233, -234, -235, -238	2.67×10 <sup>-1</sup>	4.08×10 <sup>-2</sup>	2.53×10 <sup>-1</sup>	6.03×10 <sup>-1</sup>	2.33×10 <sup>-1</sup>	7.03	1.85×10 <sup>-1</sup>	1.98×10 <sup>-1</sup>	3.47×10 <sup>-1</sup>	2.06×10 <sup>-1</sup>	3.28×10 <sup>-1</sup>	3.01×10 <sup>-1</sup>	9.99
Np-237	1.78×10 <sup>-2</sup>	8.77×10 <sup>-3</sup>	4.10×10 <sup>-3</sup>	8.79×10 <sup>-3</sup>	3.83×10 <sup>-2</sup>	8.08×10 <sup>-2</sup>	3.99×10 <sup>-2</sup>	4.50×10 <sup>-2</sup>	3.72×10 <sup>-3</sup>	5.66×10 <sup>-2</sup>	3.35×10 <sup>-3</sup>	6.91×10 <sup>-2</sup>	3.76×10 <sup>-1</sup>
Pu-239, -240	2.89×10 <sup>1</sup>	1.10×10 <sup>1</sup>	1.72×10 <sup>1</sup>	2.49×10 <sup>1</sup>	5.87	3.04×10 <sup>2</sup>	1.78×10 <sup>1</sup>	4.68×10 <sup>1</sup>	1.92×10 <sup>1</sup>	7.82×10 <sup>1</sup>	5.94	3.01×10 <sup>1</sup>	5.90×10 <sup>2</sup>

Key: C=carbon; Cs=cesium; H=hydrogen; I=iodine; Np=neptunium; Pu=plutonium; Sr=strontium; Tc=technetium; U=uranium.

Source: SAIC 2011.

**Table D-10. Single-Shell Tank Ancillary Equipment Nonradioactive Constituent Inventories (grams)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Chromium	1.32×10 <sup>5</sup>	8.80×10 <sup>4</sup>	1.34×10 <sup>5</sup>	2.61×10 <sup>5</sup>	3.27×10 <sup>5</sup>	7.90×10 <sup>4</sup>	4.29×10 <sup>5</sup>	7.07×10 <sup>5</sup>	1.62×10 <sup>5</sup>	2.64×10 <sup>5</sup>	1.17×10 <sup>5</sup>	3.95×10 <sup>5</sup>	3.09×10 <sup>6</sup>
Mercury	1.29×10 <sup>3</sup>	4.78×10 <sup>2</sup>	1.68×10 <sup>3</sup>	2.69×10 <sup>3</sup>	7.74×10 <sup>2</sup>	5.55×10 <sup>3</sup>	2.55×10 <sup>2</sup>	9.80×10 <sup>2</sup>	2.66×10 <sup>2</sup>	1.22×10 <sup>2</sup>	3.76×10 <sup>3</sup>	1.97×10 <sup>2</sup>	1.80×10 <sup>4</sup>
Nitrate	1.15×10 <sup>7</sup>	8.55×10 <sup>6</sup>	2.30×10 <sup>7</sup>	2.04×10 <sup>7</sup>	2.95×10 <sup>7</sup>	9.25×10 <sup>6</sup>	3.93×10 <sup>7</sup>	4.44×10 <sup>7</sup>	9.98×10 <sup>6</sup>	6.02×10 <sup>7</sup>	1.23×10 <sup>7</sup>	4.22×10 <sup>7</sup>	3.11×10 <sup>8</sup>
Lead	3.26×10 <sup>4</sup>	1.41×10 <sup>4</sup>	8.12×10 <sup>4</sup>	4.34×10 <sup>4</sup>	2.28×10 <sup>4</sup>	3.27×10 <sup>5</sup>	7.96×10 <sup>3</sup>	1.17×10 <sup>4</sup>	5.80×10 <sup>4</sup>	3.06×10 <sup>4</sup>	2.04×10 <sup>4</sup>	8.37×10 <sup>4</sup>	7.34×10 <sup>5</sup>
Uranium	8.90×10 <sup>4</sup>	1.66×10 <sup>4</sup>	3.47×10 <sup>5</sup>	8.70×10 <sup>5</sup>	2.92×10 <sup>5</sup>	1.59×10 <sup>6</sup>	1.85×10 <sup>5</sup>	2.19×10 <sup>5</sup>	4.97×10 <sup>5</sup>	1.96×10 <sup>5</sup>	4.75×10 <sup>5</sup>	3.84×10 <sup>5</sup>	5.16×10 <sup>6</sup>
Acetonitrile	0	0	0	0	0	0	0	0	0	0	0	0	0
Benzene	0	0	0	0	0	0	0	0	0	0	0	0	0
Butanol	0	0	0	0	0	0	0	0	0	0	0	0	0
PCBs	2.47×10 <sup>2</sup>	1.65×10 <sup>2</sup>	6.59×10 <sup>2</sup>	4.95×10 <sup>2</sup>	4.95×10 <sup>2</sup>	6.59×10 <sup>2</sup>	4.95×10 <sup>2</sup>	6.18×10 <sup>2</sup>	6.59×10 <sup>2</sup>	7.42×10 <sup>2</sup>	2.47×10 <sup>2</sup>	6.59×10 <sup>2</sup>	6.14×10 <sup>3</sup>
2,4,6-TCP	0	0	0	0	0	0	0	0	0	0	0	0	0

Note: To convert grams to ounces, multiply by 0.03527.

Key: butanol=n-butyl alcohol; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

Source: SAIC 2011.

**Table D–11. Double-Shell Tank Ancillary Equipment Radioactive Constituent Inventories  
(curies)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Hydrogen-3 (tritium)	4.16×10 <sup>-1</sup>	4.70	6.64×10 <sup>-1</sup>	1.62×10 <sup>-1</sup>	5.30×10 <sup>-1</sup>	3.90	1.04×10 <sup>1</sup>
Carbon-14	6.80×10 <sup>-1</sup>	6.05×10 <sup>-1</sup>	3.50×10 <sup>-1</sup>	1.08×10 <sup>-2</sup>	2.93×10 <sup>-2</sup>	1.36×10 <sup>-1</sup>	1.81
Strontium-90	3.70×10 <sup>3</sup>	1.60×10 <sup>2</sup>	1.13×10 <sup>3</sup>	4.37×10 <sup>4</sup>	2.25×10 <sup>4</sup>	7.78×10 <sup>2</sup>	7.19×10 <sup>4</sup>
Technetium-99	1.30×10 <sup>1</sup>	1.25×10 <sup>1</sup>	7.28	5.86×10 <sup>-1</sup>	5.77	8.76	4.79×10 <sup>1</sup>
Iodine-129	1.35×10 <sup>-2</sup>	2.36×10 <sup>-2</sup>	8.24×10 <sup>-3</sup>	9.29×10 <sup>-4</sup>	5.40×10 <sup>-3</sup>	9.51×10 <sup>-3</sup>	6.11×10 <sup>-2</sup>
Cesium-137	2.98×10 <sup>4</sup>	1.65×10 <sup>4</sup>	1.28×10 <sup>4</sup>	1.89×10 <sup>3</sup>	2.78×10 <sup>4</sup>	9.20×10 <sup>3</sup>	9.80×10 <sup>4</sup>
Uranium-233, -234, -235, -238	2.78×10 <sup>-2</sup>	8.74×10 <sup>-3</sup>	1.54×10 <sup>-1</sup>	2.10×10 <sup>-2</sup>	1.60×10 <sup>-2</sup>	1.61×10 <sup>-2</sup>	2.43×10 <sup>-1</sup>
Neptunium-237	2.88×10 <sup>-2</sup>	4.40×10 <sup>-2</sup>	9.34×10 <sup>-2</sup>	3.30×10 <sup>-2</sup>	7.64×10 <sup>-2</sup>	1.36×10 <sup>-2</sup>	2.89×10 <sup>-1</sup>
Plutonium-239, -240	1.66	7.61×10 <sup>-2</sup>	1.18×10 <sup>1</sup>	1.75×10 <sup>1</sup>	9.97	1.74×10 <sup>1</sup>	5.84×10 <sup>1</sup>

Source: SAIC 2011.

**Table D–12. Double-Shell Tank Ancillary Equipment Nonradioactive Constituent Inventories  
(grams)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Chromium	6.54×10 <sup>4</sup>	3.18×10 <sup>4</sup>	7.78×10 <sup>4</sup>	1.83×10 <sup>4</sup>	1.44×10 <sup>4</sup>	1.69×10 <sup>5</sup>	3.76×10 <sup>5</sup>
Mercury	1.64×10 <sup>1</sup>	0	8.18×10 <sup>-1</sup>	8.26×10 <sup>2</sup>	1.18×10 <sup>1</sup>	3.19×10 <sup>1</sup>	8.87×10 <sup>2</sup>
Nitrate	2.28×10 <sup>7</sup>	1.73×10 <sup>7</sup>	1.36×10 <sup>7</sup>	1.12×10 <sup>6</sup>	2.19×10 <sup>6</sup>	8.86×10 <sup>6</sup>	6.59×10 <sup>7</sup>
Lead	1.28×10 <sup>4</sup>	2.77×10 <sup>3</sup>	5.93×10 <sup>3</sup>	2.94×10 <sup>4</sup>	1.14×10 <sup>3</sup>	5.59×10 <sup>3</sup>	5.76×10 <sup>4</sup>
Uranium	9.46×10 <sup>3</sup>	3.78×10 <sup>3</sup>	1.55×10 <sup>5</sup>	2.31×10 <sup>4</sup>	1.47×10 <sup>4</sup>	8.51×10 <sup>3</sup>	2.14×10 <sup>5</sup>
Acetonitrile	2.59×10 <sup>4</sup>	2.96×10 <sup>4</sup>	2.22×10 <sup>4</sup>	7.39×10 <sup>3</sup>	7.39×10 <sup>3</sup>	1.11×10 <sup>4</sup>	1.04×10 <sup>5</sup>
Benzene	2.11	2.41	1.81	6.02×10 <sup>-1</sup>	6.02×10 <sup>-1</sup>	9.04×10 <sup>-1</sup>	8.43
Butanol (n-butyl alcohol)	3.03×10 <sup>6</sup>	3.46×10 <sup>6</sup>	2.60×10 <sup>6</sup>	8.66×10 <sup>5</sup>	8.66×10 <sup>5</sup>	1.30×10 <sup>6</sup>	1.21×10 <sup>7</sup>
Polychlorinated biphenyls	7.29×10 <sup>2</sup>	8.33×10 <sup>2</sup>	6.25×10 <sup>2</sup>	2.08×10 <sup>2</sup>	2.08×10 <sup>2</sup>	3.13×10 <sup>2</sup>	2.92×10 <sup>3</sup>
2,4,6-Trichlorophenol	9.72×10 <sup>-1</sup>	1.11	8.33×10 <sup>-1</sup>	2.78×10 <sup>-1</sup>	2.78×10 <sup>-1</sup>	4.17×10 <sup>-1</sup>	3.89

Note: To convert grams to ounces, multiply by 0.03527.

Source: SAIC 2011.

### D.1.3 Tank Residual Waste Inventories

Residual waste is defined as the tank waste remaining in the tank after all waste retrieval actions have been completed. The Hanford Federal Facility Agreement and Consent Order, also known as the Tri-Party Agreement (TPA), allows approximately 10 cubic meters (360 cubic feet) of residual waste for 100-series SSTs and 0.8 cubic meters (30 cubic feet) of residual waste for 200-series SSTs following completion of retrieval operations; it also provides a method for determining the allowed residual waste in each tank on a tank-by-tank basis (Ecology, EPA, and DOE 1989). Three levels of retrieval were considered for the *TC & WM EIS* analysis: 90, 99, and 99.9 percent retrieval of the current inventory of radioactive and nonradioactive constituents. These retrieval percentages were developed to address various aspects related to retrieval levels or activities. Ninety percent retrieval represents a programmatic risk analysis of the tank farms as defined in the TPA Milestone M-45-00, Appendix H, process. Ninety-nine percent retrieval is the goal established in the TPA. The 99.9 percent retrieval rate applies to cases where tank removal was analyzed to limit worker exposure; this rate also reflects multiple uses of retrieval technologies.



This appendix describes three proposed methods for estimating residual waste in the storage tanks following retrieval and presents the results for the method selected (first method) for the *TC & WM EIS* analyses. The three methods are as follows:

1. The first method involves multiplying the existing total tank inventory by a ratio of the final waste volume to the current waste volume of a tank farm. Assume that the inventory is distributed uniformly through the volume of a tank farm. This method represents the case for waste retrieved “as is,” i.e., without washing or leaching. For example, for 99 percent retrieval, the volume of SST residual waste in a tank farm would equal the current waste volume estimate in that tank farm, based on the 2002 BBI, multiplied by 0.01. This result may differ slightly from the TPA Milestone M-45-00, Appendix H, estimate, which used earlier tank volume estimates that were applied across all of the SSTs.
2. A more complex method involves making the following assumptions about which waste phases would remain in the tank following retrieval:
  - All supernatant would be removed, and retained gas would be released from the tank during retrieval.
  - Ten cubic meters (360 cubic feet) of waste would remain in a 100-series SST; 0.8 cubic meters (30 cubic feet) would remain in a 200-series SST.
  - Sludge would be at the bottom of the tank.
  - When a tank contains multiple sludge phases, each phase would remain in the tank in the same proportions that were present prior to retrieval.
  - For tanks where only salt cake exists, the remaining salt cake would be volumetrically proportional to the volumes currently in the tanks.
  - Tanks 241-TX-116 and 241-TX-117 are exceptions: all waste in these tanks would be removed in proportion to current volumes.
  - The fraction of each waste phase remaining (that is, the ratio of current phase volumes for each phase that would remain in the tank following retrieval) is calculated based on the assumptions above.
  - The final remaining constituent inventory (for each phase) is calculated from the initial inventory, current and final volume, and fraction of the phase.
  - The inventories for each phase are then added to get the tank total. This method is representative of dry retrieval by layer without mixing.
3. The third method uses output from the HTWOS [Hanford Tank Waste Operation Simulator] model (for radionuclides only) adjusted to the same final volume as the other two methods. This model applies component- and tank-specific water-wash factors and adds sufficient water to achieve either a 5-molar sodium solution or a 10 weight-percent slurry, whichever is the limiting condition for feed to the Waste Treatment Plant (WTP), to predict the composition of waste retrieved from the tank. Waste remaining in the tank after retrieval is assumed to have the same composition as the tank waste before retrieval (Kirkbride et al. 2002). This method is representative of sluicing-type waste retrieval methods.

Because the last two methods do not provide estimates for both radioactive and chemical constituents for each level of retrieval, the first method (volume retrieval) was applied. The degree of retrieval applicability across the Tank Closure alternatives is summarized in Table D-13. The inventory results of implementing the radioactive constituent procedure for the three retrieval cases are presented in Tables D-14 through D-19. The inventory results of implementing the chemical constituent procedure for the retrieval cases are presented in Tables D-20 through D-25.

**Table D-13. Tank Closure Alternative Retrieval Approaches**

Alternative	Approach
1	Best-Basis Inventory as salt cake for single-shell tanks, supernatant for double-shell tanks
2A	99 percent retrieval estimate; residual as salt cake for single-shell tanks and supernatant for double-shell tanks
2B, 3A, 3B, and 3C	99 percent retrieval estimate; residual stabilized with grout
4	99.9 percent retrieval estimate; residual stabilized with grout
5	90 percent retrieval estimate; residual stabilized with grout
6A and 6B	99.9 percent retrieval; tank and soil removed
6C	99 percent retrieval estimate; residual stabilized with grout

At the time of the preparation of this EIS, retrieval had been completed on seven SSTs, of which three were 100-series tanks and four were 200-series tanks. For the three 100-series tanks (C-103, C-106, and S-112), a review of the estimated residual technetium-99 inventory compared with the expected inventory found inconsistencies between the three tanks and a wide range in the ratio of final curies to expected curies. From this review, DOE concluded that it currently does not have a technical basis for making more-specific assumptions about the expected compositions of the waste “heels” that would remain in the tanks after retrieval, and not much is known about the behavior of, or ability to remove, small volumes of residual waste. It is also noted that the tank closure process, if implemented, would require detailed examination of the tanks and residual waste, as well as preparation of site-specific radiological performance assessments and closure plans. These documents will provide the information and analysis necessary for DOE and the regulators to make decisions on what levels of residual tank waste are acceptable in terms of short- and long-term risks.

**Table D-14. Single-Shell Tank Residual Radioactive Constituent Inventories – 90 Percent Retrieval (curies)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Hydrogen-3 (tritium)	3.38×10 <sup>1</sup>	1.30×10 <sup>1</sup>	2.07	1.01×10 <sup>1</sup>	1.33×10 <sup>2</sup>	1.10×10 <sup>1</sup>	1.94×10 <sup>2</sup>	1.32×10 <sup>2</sup>	3.42	2.13×10 <sup>2</sup>	3.50	1.44×10 <sup>2</sup>	8.93×10 <sup>2</sup>
Carbon-14	8.33	6.44	7.88×10 <sup>-1</sup>	4.19	5.60×10 <sup>1</sup>	1.58	5.05×10 <sup>1</sup>	2.90×10 <sup>1</sup>	1.48	6.47×10 <sup>1</sup>	7.63×10 <sup>-1</sup>	3.56×10 <sup>1</sup>	2.59×10 <sup>2</sup>
Strontium-90	6.52×10 <sup>5</sup>	3.09×10 <sup>5</sup>	1.89×10 <sup>5</sup>	1.30×10 <sup>5</sup>	1.75×10 <sup>5</sup>	9.18×10 <sup>5</sup>	2.52×10 <sup>5</sup>	5.28×10 <sup>5</sup>	3.72×10 <sup>4</sup>	1.17×10 <sup>5</sup>	3.31×10 <sup>4</sup>	9.05×10 <sup>4</sup>	3.43×10 <sup>6</sup>
Technetium-99	6.74×10 <sup>1</sup>	4.13×10 <sup>1</sup>	2.13×10 <sup>1</sup>	3.70×10 <sup>1</sup>	2.54×10 <sup>2</sup>	3.51×10 <sup>1</sup>	2.74×10 <sup>2</sup>	1.76×10 <sup>2</sup>	1.63×10 <sup>1</sup>	3.76×10 <sup>2</sup>	1.02×10 <sup>1</sup>	2.43×10 <sup>2</sup>	1.55×10 <sup>3</sup>
Iodine-129	9.45×10 <sup>-2</sup>	4.81×10 <sup>-2</sup>	8.18×10 <sup>-3</sup>	4.49×10 <sup>-2</sup>	5.55×10 <sup>-1</sup>	9.93×10 <sup>-2</sup>	5.93×10 <sup>-1</sup>	3.35×10 <sup>-1</sup>	1.14×10 <sup>-2</sup>	7.15×10 <sup>-1</sup>	1.29×10 <sup>-2</sup>	4.69×10 <sup>-1</sup>	2.99
Cesium-137	1.24×10 <sup>5</sup>	6.58×10 <sup>4</sup>	3.58×10 <sup>4</sup>	3.26×10 <sup>4</sup>	2.23×10 <sup>5</sup>	9.93×10 <sup>4</sup>	2.60×10 <sup>5</sup>	2.68×10 <sup>5</sup>	1.65×10 <sup>4</sup>	2.44×10 <sup>5</sup>	5.26×10 <sup>3</sup>	2.32×10 <sup>5</sup>	1.61×10 <sup>6</sup>
Uranium-233, -234, -235, -238	3.29	3.64×10 <sup>-1</sup>	2.08	5.09	5.22	4.98×10 <sup>1</sup>	5.18	2.95	2.59	4.79	2.23	3.90	8.75×10 <sup>1</sup>
Neptunium-237	2.20×10 <sup>-1</sup>	7.83×10 <sup>-2</sup>	3.38×10 <sup>-2</sup>	7.42×10 <sup>-2</sup>	8.59×10 <sup>-1</sup>	5.72×10 <sup>-1</sup>	1.12	6.71×10 <sup>-1</sup>	2.78×10 <sup>-2</sup>	1.32	2.28×10 <sup>-2</sup>	8.94×10 <sup>-1</sup>	5.89
Plutonium-239, -240	3.56×10 <sup>2</sup>	9.83×10 <sup>1</sup>	1.42×10 <sup>2</sup>	2.10×10 <sup>2</sup>	1.32×10 <sup>2</sup>	2.16×10 <sup>3</sup>	5.00×10 <sup>2</sup>	6.99×10 <sup>2</sup>	1.43×10 <sup>2</sup>	1.82×10 <sup>3</sup>	4.04×10 <sup>1</sup>	3.89×10 <sup>2</sup>	6.69×10 <sup>3</sup>

Source: DOE 2003a; SAIC 2011.

**Table D-15. Double-Shell Tank Residual Radioactive Constituent Inventories – 90 Percent Retrieval (curies)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Hydrogen-3 (tritium)	1.18×10 <sup>1</sup>	1.53×10 <sup>2</sup>	1.70×10 <sup>1</sup>	2.47	1.87×10 <sup>1</sup>	1.09×10 <sup>2</sup>	3.12×10 <sup>2</sup>
Carbon-14	1.93×10 <sup>1</sup>	1.97×10 <sup>1</sup>	8.95	1.65×10 <sup>-1</sup>	1.04	3.81	5.29×10 <sup>1</sup>
Strontium-90	1.05×10 <sup>5</sup>	5.20×10 <sup>3</sup>	2.89×10 <sup>4</sup>	6.66×10 <sup>5</sup>	7.95×10 <sup>5</sup>	2.18×10 <sup>4</sup>	1.62×10 <sup>6</sup>
Technetium-99	3.68×10 <sup>2</sup>	4.07×10 <sup>2</sup>	1.86×10 <sup>2</sup>	8.93	2.04×10 <sup>2</sup>	2.46×10 <sup>2</sup>	1.42×10 <sup>3</sup>
Iodine-129	3.81×10 <sup>-1</sup>	7.69×10 <sup>-1</sup>	2.11×10 <sup>-1</sup>	1.42×10 <sup>-2</sup>	1.91×10 <sup>-1</sup>	2.66×10 <sup>-1</sup>	1.83
Cesium-137	8.46×10 <sup>5</sup>	5.36×10 <sup>5</sup>	3.26×10 <sup>5</sup>	2.89×10 <sup>4</sup>	9.84×10 <sup>5</sup>	2.58×10 <sup>5</sup>	2.98×10 <sup>6</sup>
Uranium-233, -234, -235, -238	7.88×10 <sup>-1</sup>	2.85×10 <sup>-1</sup>	3.93	3.20×10 <sup>-1</sup>	5.67×10 <sup>-1</sup>	4.50×10 <sup>-1</sup>	6.34
Neptunium-237	8.17×10 <sup>-1</sup>	1.43	2.39	5.03×10 <sup>-1</sup>	2.70	3.80×10 <sup>-1</sup>	8.22
Plutonium-239, -240	4.70×10 <sup>1</sup>	2.48	3.02×10 <sup>2</sup>	2.66×10 <sup>2</sup>	3.52×10 <sup>2</sup>	4.88×10 <sup>2</sup>	1.46×10 <sup>3</sup>

Source: DOE 2003a; SAIC 2011.

**Table D–16. Single-Shell Tank Residual Radioactive Constituent Inventories – 99 Percent Retrieval (curies)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Hydrogen-3 (tritium)	3.38	1.30	2.07×10 <sup>-1</sup>	1.01	1.33×10 <sup>1</sup>	1.10	1.94×10 <sup>1</sup>	1.32×10 <sup>1</sup>	3.42×10 <sup>-1</sup>	2.13×10 <sup>1</sup>	3.50×10 <sup>-1</sup>	1.44×10 <sup>1</sup>	8.93×10 <sup>1</sup>
Carbon-14	8.33×10 <sup>-1</sup>	6.44×10 <sup>-1</sup>	7.88×10 <sup>-2</sup>	4.19×10 <sup>-1</sup>	5.60	1.58×10 <sup>-1</sup>	5.05	2.90	1.48×10 <sup>-1</sup>	6.47	7.63×10 <sup>-2</sup>	3.56	2.59×10 <sup>1</sup>
Strontium-90	6.52×10 <sup>4</sup>	3.09×10 <sup>4</sup>	1.89×10 <sup>4</sup>	1.30×10 <sup>4</sup>	1.75×10 <sup>4</sup>	9.18×10 <sup>4</sup>	2.52×10 <sup>4</sup>	5.28×10 <sup>4</sup>	3.72×10 <sup>3</sup>	1.17×10 <sup>4</sup>	3.31×10 <sup>3</sup>	9.05×10 <sup>3</sup>	3.43×10 <sup>5</sup>
Technetium-99	6.74	4.13	2.13	3.70	2.54×10 <sup>1</sup>	3.51	2.74×10 <sup>1</sup>	1.76×10 <sup>1</sup>	1.63	3.76×10 <sup>1</sup>	1.02	2.43×10 <sup>1</sup>	1.55×10 <sup>2</sup>
Iodine-129	9.45×10 <sup>-3</sup>	4.81×10 <sup>-3</sup>	8.18×10 <sup>-4</sup>	4.49×10 <sup>-3</sup>	5.55×10 <sup>-2</sup>	9.93×10 <sup>-3</sup>	5.93×10 <sup>-2</sup>	3.35×10 <sup>-2</sup>	1.14×10 <sup>-3</sup>	7.15×10 <sup>-2</sup>	1.29×10 <sup>-3</sup>	4.69×10 <sup>-2</sup>	2.99×10 <sup>-1</sup>
Cesium-137	1.24×10 <sup>4</sup>	6.58×10 <sup>3</sup>	3.58×10 <sup>3</sup>	3.26×10 <sup>3</sup>	2.23×10 <sup>4</sup>	9.93×10 <sup>3</sup>	2.60×10 <sup>4</sup>	2.68×10 <sup>4</sup>	1.65×10 <sup>3</sup>	2.44×10 <sup>4</sup>	5.26×10 <sup>2</sup>	2.32×10 <sup>4</sup>	1.61×10 <sup>5</sup>
Uranium-233, -234, -235, -238	3.29×10 <sup>-1</sup>	3.64×10 <sup>-2</sup>	2.08×10 <sup>-1</sup>	5.09×10 <sup>-1</sup>	5.22×10 <sup>-1</sup>	4.98	5.18×10 <sup>-1</sup>	2.95×10 <sup>-1</sup>	2.59×10 <sup>-1</sup>	4.79×10 <sup>-1</sup>	2.23×10 <sup>-1</sup>	3.90×10 <sup>-1</sup>	8.75
Neptunium-237	2.20×10 <sup>-2</sup>	7.83×10 <sup>-3</sup>	3.38×10 <sup>-3</sup>	7.42×10 <sup>-3</sup>	8.59×10 <sup>-2</sup>	5.72×10 <sup>-2</sup>	1.12×10 <sup>-1</sup>	6.71×10 <sup>-2</sup>	2.78×10 <sup>-3</sup>	1.32×10 <sup>-1</sup>	2.28×10 <sup>-3</sup>	8.94×10 <sup>-2</sup>	5.89×10 <sup>-1</sup>
Plutonium-239, -240	3.56×10 <sup>1</sup>	9.83	1.42×10 <sup>1</sup>	2.10×10 <sup>1</sup>	1.32×10 <sup>1</sup>	2.16×10 <sup>2</sup>	5.00×10 <sup>1</sup>	6.99×10 <sup>1</sup>	1.43×10 <sup>1</sup>	1.82×10 <sup>2</sup>	4.04	3.89×10 <sup>1</sup>	6.69×10 <sup>2</sup>

Source: DOE 2003a; SAIC 2011.

**Table D–17. Double-Shell Tank Residual Radioactive Constituent Inventories – 99 Percent Retrieval (curies)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Hydrogen-3 (tritium)	1.18	1.53×10 <sup>1</sup>	1.70	2.47×10 <sup>-1</sup>	1.87	1.09×10 <sup>1</sup>	3.12×10 <sup>1</sup>
Carbon-14	1.93	1.97	8.95×10 <sup>-1</sup>	1.65×10 <sup>-2</sup>	1.04×10 <sup>-1</sup>	3.81×10 <sup>-1</sup>	5.29
Strontium-90	1.05×10 <sup>4</sup>	5.20×10 <sup>2</sup>	2.89×10 <sup>3</sup>	6.66×10 <sup>4</sup>	7.95×10 <sup>4</sup>	2.18×10 <sup>3</sup>	1.62×10 <sup>5</sup>
Technetium-99	3.68×10 <sup>1</sup>	4.07×10 <sup>1</sup>	1.86×10 <sup>1</sup>	8.93×10 <sup>-1</sup>	2.04×10 <sup>1</sup>	2.46×10 <sup>1</sup>	1.42×10 <sup>2</sup>
Iodine-129	3.81×10 <sup>-2</sup>	7.69×10 <sup>-2</sup>	2.11×10 <sup>-2</sup>	1.42×10 <sup>-3</sup>	1.91×10 <sup>-2</sup>	2.66×10 <sup>-2</sup>	1.83×10 <sup>-1</sup>
Cesium-137	8.46×10 <sup>4</sup>	5.36×10 <sup>4</sup>	3.26×10 <sup>4</sup>	2.89×10 <sup>3</sup>	9.84×10 <sup>4</sup>	2.58×10 <sup>4</sup>	2.98×10 <sup>5</sup>
Uranium-233, -234, -235, -238	7.88×10 <sup>-2</sup>	2.85×10 <sup>-2</sup>	3.93×10 <sup>-1</sup>	3.20×10 <sup>-2</sup>	5.67×10 <sup>-2</sup>	4.50×10 <sup>-2</sup>	6.34×10 <sup>-1</sup>
Neptunium-237	8.17×10 <sup>-2</sup>	1.43×10 <sup>-1</sup>	2.39×10 <sup>-1</sup>	5.03×10 <sup>-2</sup>	2.70×10 <sup>-1</sup>	3.80×10 <sup>-2</sup>	8.22×10 <sup>-1</sup>
Plutonium-239, -240	4.70	2.48×10 <sup>-1</sup>	3.02×10 <sup>1</sup>	2.66×10 <sup>1</sup>	3.52×10 <sup>1</sup>	4.88×10 <sup>1</sup>	1.46×10 <sup>2</sup>

Source: DOE 2003a; SAIC 2011.

**Table D-18. Single-Shell Tank Residual Radioactive Constituent Inventories – 99.9 Percent Retrieval (curies)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Hydrogen-3 (tritium)	3.38×10 <sup>-1</sup>	1.30×10 <sup>-1</sup>	2.07×10 <sup>-2</sup>	1.01×10 <sup>-1</sup>	1.33	1.10×10 <sup>-1</sup>	1.94	1.32	3.42×10 <sup>-2</sup>	2.13	3.50×10 <sup>-2</sup>	1.44	8.93
Carbon-14	8.33×10 <sup>-2</sup>	6.44×10 <sup>-2</sup>	7.88×10 <sup>-3</sup>	4.19×10 <sup>-2</sup>	5.60×10 <sup>-1</sup>	1.58×10 <sup>-2</sup>	5.05×10 <sup>-1</sup>	2.90×10 <sup>-1</sup>	1.48×10 <sup>-2</sup>	6.47×10 <sup>-1</sup>	7.63×10 <sup>-3</sup>	3.56×10 <sup>-1</sup>	2.59
Strontium-90	6.52×10 <sup>3</sup>	3.09×10 <sup>3</sup>	1.89×10 <sup>3</sup>	1.30×10 <sup>3</sup>	1.75×10 <sup>3</sup>	9.18×10 <sup>3</sup>	2.52×10 <sup>3</sup>	5.28×10 <sup>3</sup>	3.72×10 <sup>2</sup>	1.17×10 <sup>3</sup>	3.31×10 <sup>2</sup>	9.05×10 <sup>2</sup>	3.43×10 <sup>4</sup>
Technetium-99	6.74×10 <sup>-1</sup>	4.13×10 <sup>-1</sup>	2.13×10 <sup>-1</sup>	3.70×10 <sup>-1</sup>	2.54	3.51×10 <sup>-1</sup>	2.74	1.76	1.63×10 <sup>-1</sup>	3.76	1.02×10 <sup>-1</sup>	2.43	1.55×10 <sup>1</sup>
Iodine-129	9.45×10 <sup>-4</sup>	4.81×10 <sup>-4</sup>	8.18×10 <sup>-5</sup>	4.49×10 <sup>-4</sup>	5.55×10 <sup>-3</sup>	9.93×10 <sup>-4</sup>	5.93×10 <sup>-3</sup>	3.35×10 <sup>-3</sup>	1.14×10 <sup>-4</sup>	7.15×10 <sup>-3</sup>	1.29×10 <sup>-4</sup>	4.69×10 <sup>-3</sup>	2.99×10 <sup>-2</sup>
Cesium-137	1.24×10 <sup>3</sup>	6.58×10 <sup>2</sup>	3.58×10 <sup>2</sup>	3.26×10 <sup>2</sup>	2.23×10 <sup>3</sup>	9.93×10 <sup>2</sup>	2.60×10 <sup>3</sup>	2.68×10 <sup>3</sup>	1.65×10 <sup>2</sup>	2.44×10 <sup>3</sup>	5.26×10 <sup>1</sup>	2.32×10 <sup>3</sup>	1.61×10 <sup>4</sup>
Uranium-233, -234, -235, -238	3.29×10 <sup>-2</sup>	3.64×10 <sup>-3</sup>	2.08×10 <sup>-2</sup>	5.09×10 <sup>-2</sup>	5.22×10 <sup>-2</sup>	4.98×10 <sup>-1</sup>	5.18×10 <sup>-2</sup>	2.95×10 <sup>-2</sup>	2.59×10 <sup>-2</sup>	4.79×10 <sup>-2</sup>	2.23×10 <sup>-2</sup>	3.90×10 <sup>-2</sup>	8.75×10 <sup>-1</sup>
Neptunium-237	2.20×10 <sup>-3</sup>	7.83×10 <sup>-4</sup>	3.38×10 <sup>-4</sup>	7.42×10 <sup>-4</sup>	8.59×10 <sup>-3</sup>	5.72×10 <sup>-3</sup>	1.12×10 <sup>-2</sup>	6.71×10 <sup>-3</sup>	2.78×10 <sup>-4</sup>	1.32×10 <sup>-2</sup>	2.28×10 <sup>-4</sup>	8.94×10 <sup>-3</sup>	5.89×10 <sup>-2</sup>
Plutonium-239, -240	3.56	9.83×10 <sup>-1</sup>	1.42	2.10	1.32	2.16×10 <sup>1</sup>	5.00	6.99	1.43	1.82×10 <sup>1</sup>	4.04×10 <sup>-1</sup>	3.89	6.69×10 <sup>1</sup>

Source: DOE 2003a; SAIC 2011.

**Table D-19. Double-Shell Tank Residual Radioactive Constituent Inventories – 99.9 Percent Retrieval (curies)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Hydrogen-3 (tritium)	1.18×10 <sup>-1</sup>	1.53	1.70×10 <sup>-1</sup>	2.47×10 <sup>-2</sup>	1.87×10 <sup>-1</sup>	1.09	3.12
Carbon-14	1.93×10 <sup>-1</sup>	1.97×10 <sup>-1</sup>	8.95×10 <sup>-2</sup>	1.65×10 <sup>-3</sup>	1.04×10 <sup>-2</sup>	3.81×10 <sup>-2</sup>	5.29×10 <sup>-1</sup>
Strontium-90	1.05×10 <sup>3</sup>	5.20×10 <sup>1</sup>	2.89×10 <sup>2</sup>	6.66×10 <sup>3</sup>	7.95×10 <sup>3</sup>	2.18×10 <sup>2</sup>	1.62×10 <sup>4</sup>
Technetium-99	3.68	4.07	1.86	8.93×10 <sup>-2</sup>	2.04	2.46	1.42×10 <sup>1</sup>
Iodine-129	3.81×10 <sup>-3</sup>	7.69×10 <sup>-3</sup>	2.11×10 <sup>-3</sup>	1.42×10 <sup>-4</sup>	1.91×10 <sup>-3</sup>	2.66×10 <sup>-3</sup>	1.83×10 <sup>-2</sup>
Cesium-137	8.46×10 <sup>3</sup>	5.36×10 <sup>3</sup>	3.26×10 <sup>3</sup>	2.89×10 <sup>2</sup>	9.84×10 <sup>3</sup>	2.58×10 <sup>3</sup>	2.98×10 <sup>4</sup>
Uranium-233, -234, -235, -238	7.88×10 <sup>-3</sup>	2.85×10 <sup>-3</sup>	3.93×10 <sup>-2</sup>	3.20×10 <sup>-3</sup>	5.67×10 <sup>-3</sup>	4.50×10 <sup>-3</sup>	6.34×10 <sup>-2</sup>
Neptunium-237	8.17×10 <sup>-3</sup>	1.43×10 <sup>-2</sup>	2.39×10 <sup>-2</sup>	5.03×10 <sup>-3</sup>	2.70×10 <sup>-2</sup>	3.80×10 <sup>-3</sup>	8.22×10 <sup>-2</sup>
Plutonium-239, -240	4.70×10 <sup>-1</sup>	2.48×10 <sup>-2</sup>	3.02	2.66	3.52	4.88	1.46×10 <sup>1</sup>

Source: DOE 2003a; SAIC 2011.

**Table D–20. Single-Shell Tank Residual Nonradioactive Constituent Inventories – 90 Percent Retrieval (grams)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Chromium	1.62×10 <sup>6</sup>	7.87×10 <sup>5</sup>	1.11×10 <sup>6</sup>	2.20×10 <sup>6</sup>	7.34×10 <sup>6</sup>	5.60×10 <sup>5</sup>	1.20×10 <sup>7</sup>	1.05×10 <sup>7</sup>	1.21×10 <sup>6</sup>	6.13×10 <sup>6</sup>	7.95×10 <sup>5</sup>	5.11×10 <sup>6</sup>	4.95×10 <sup>7</sup>
Mercury	1.59×10 <sup>4</sup>	4.27×10 <sup>3</sup>	1.38×10 <sup>4</sup>	2.27×10 <sup>4</sup>	1.74×10 <sup>4</sup>	3.93×10 <sup>4</sup>	7.15×10 <sup>3</sup>	1.46×10 <sup>4</sup>	1.99×10 <sup>3</sup>	2.83×10 <sup>3</sup>	2.56×10 <sup>4</sup>	2.55×10 <sup>3</sup>	1.68×10 <sup>5</sup>
Nitrate	1.41×10 <sup>8</sup>	7.63×10 <sup>7</sup>	1.90×10 <sup>8</sup>	1.73×10 <sup>8</sup>	6.62×10 <sup>8</sup>	6.56×10 <sup>7</sup>	1.10×10 <sup>9</sup>	6.62×10 <sup>8</sup>	7.47×10 <sup>7</sup>	1.40×10 <sup>9</sup>	8.37×10 <sup>7</sup>	5.46×10 <sup>8</sup>	5.18×10 <sup>9</sup>
Lead	4.02×10 <sup>5</sup>	1.26×10 <sup>5</sup>	6.69×10 <sup>5</sup>	3.66×10 <sup>5</sup>	5.12×10 <sup>5</sup>	2.32×10 <sup>6</sup>	2.23×10 <sup>5</sup>	1.75×10 <sup>5</sup>	4.34×10 <sup>5</sup>	7.12×10 <sup>5</sup>	1.39×10 <sup>5</sup>	1.08×10 <sup>6</sup>	7.16×10 <sup>6</sup>
Uranium	1.10×10 <sup>6</sup>	1.48×10 <sup>5</sup>	2.86×10 <sup>6</sup>	7.35×10 <sup>6</sup>	6.55×10 <sup>6</sup>	1.13×10 <sup>7</sup>	5.19×10 <sup>6</sup>	3.27×10 <sup>6</sup>	3.72×10 <sup>6</sup>	4.56×10 <sup>6</sup>	3.24×10 <sup>6</sup>	4.97×10 <sup>6</sup>	5.42×10 <sup>7</sup>
Acetonitrile	0	0	0	0	0	0	0	0	0	0	0	0	0
Benzene	0	0	0	0	0	0	0	0	0	0	0	0	0
Butanol	0	0	0	0	0	0	0	0	0	0	0	0	0
PCBs	3.05×10 <sup>3</sup>	1.47×10 <sup>3</sup>	5.44×10 <sup>3</sup>	4.18×10 <sup>3</sup>	1.11×10 <sup>4</sup>	4.67×10 <sup>3</sup>	1.39×10 <sup>4</sup>	9.23×10 <sup>3</sup>	4.93×10 <sup>3</sup>	1.73×10 <sup>4</sup>	1.68×10 <sup>3</sup>	8.53×10 <sup>3</sup>	8.54×10 <sup>4</sup>
2,4,6-TCP	0	0	0	0	0	0	0	0	0	0	0	0	0

**Note:** To convert grams to ounces, multiply by 0.03527.

**Key:** butanol=n-butyl alcohol; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

**Source:** DOE 2003a; SAIC 2011.

**Table D–21. Double-Shell Tank Residual Nonradioactive Constituent Inventories – 90 Percent Retrieval (grams)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Chromium	1.85×10 <sup>6</sup>	1.03×10 <sup>6</sup>	1.99×10 <sup>6</sup>	2.79×10 <sup>5</sup>	5.09×10 <sup>5</sup>	4.73×10 <sup>6</sup>	1.04×10 <sup>7</sup>
Mercury	4.66×10 <sup>2</sup>	0	2.09×10 <sup>1</sup>	1.26×10 <sup>4</sup>	4.15×10 <sup>2</sup>	8.95×10 <sup>2</sup>	1.44×10 <sup>4</sup>
Nitrate	6.47×10 <sup>8</sup>	5.65×10 <sup>8</sup>	3.47×10 <sup>8</sup>	1.70×10 <sup>7</sup>	7.74×10 <sup>7</sup>	2.48×10 <sup>8</sup>	1.90×10 <sup>9</sup>
Lead	3.63×10 <sup>5</sup>	9.01×10 <sup>4</sup>	1.51×10 <sup>5</sup>	4.48×10 <sup>5</sup>	4.03×10 <sup>4</sup>	1.57×10 <sup>5</sup>	1.25×10 <sup>6</sup>
Uranium	2.68×10 <sup>5</sup>	1.23×10 <sup>5</sup>	3.95×10 <sup>6</sup>	3.52×10 <sup>5</sup>	5.19×10 <sup>5</sup>	2.38×10 <sup>5</sup>	5.45×10 <sup>6</sup>
Acetonitrile	7.33×10 <sup>5</sup>	9.63×10 <sup>5</sup>	5.67×10 <sup>5</sup>	1.13×10 <sup>5</sup>	2.61×10 <sup>5</sup>	3.11×10 <sup>5</sup>	2.95×10 <sup>6</sup>
Benzene	5.97×10 <sup>1</sup>	7.85×10 <sup>1</sup>	4.62×10 <sup>1</sup>	9.19	2.13×10 <sup>1</sup>	2.53×10 <sup>1</sup>	2.40×10 <sup>2</sup>
Butanol (n-butyl alcohol)	8.59×10 <sup>7</sup>	1.13×10 <sup>8</sup>	6.63×10 <sup>7</sup>	1.32×10 <sup>7</sup>	3.06×10 <sup>7</sup>	3.64×10 <sup>7</sup>	3.45×10 <sup>8</sup>
Polychlorinated biphenyls	2.07×10 <sup>4</sup>	2.71×10 <sup>4</sup>	1.60×10 <sup>4</sup>	3.18×10 <sup>3</sup>	7.36×10 <sup>3</sup>	8.76×10 <sup>3</sup>	8.31×10 <sup>4</sup>
2,4,6-Trichlorophenol	2.75×10 <sup>1</sup>	3.62×10 <sup>1</sup>	2.13×10 <sup>1</sup>	4.23	9.81	1.17×10 <sup>1</sup>	1.11×10 <sup>2</sup>

**Note:** To convert grams to ounces, multiply by 0.03527.

**Source:** DOE 2003a; SAIC 2011.

**Table D–22. Single-Shell Tank Residual Nonradioactive Constituent Inventories – 99 Percent Retrieval (grams)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Chromium	1.62×10 <sup>5</sup>	7.87×10 <sup>4</sup>	1.11×10 <sup>5</sup>	2.20×10 <sup>5</sup>	7.34×10 <sup>5</sup>	5.60×10 <sup>4</sup>	1.20×10 <sup>6</sup>	1.05×10 <sup>6</sup>	1.21×10 <sup>5</sup>	6.13×10 <sup>5</sup>	7.95×10 <sup>4</sup>	5.11×10 <sup>5</sup>	4.95×10 <sup>6</sup>
Mercury	1.59×10 <sup>3</sup>	4.27×10 <sup>2</sup>	1.38×10 <sup>3</sup>	2.27×10 <sup>3</sup>	1.74×10 <sup>3</sup>	3.93×10 <sup>3</sup>	7.15×10 <sup>2</sup>	1.46×10 <sup>3</sup>	1.99×10 <sup>2</sup>	2.83×10 <sup>2</sup>	2.56×10 <sup>3</sup>	2.55×10 <sup>2</sup>	1.68×10 <sup>4</sup>
Nitrate	1.41×10 <sup>7</sup>	7.63×10 <sup>6</sup>	1.90×10 <sup>7</sup>	1.73×10 <sup>7</sup>	6.62×10 <sup>7</sup>	6.56×10 <sup>6</sup>	1.10×10 <sup>8</sup>	6.62×10 <sup>7</sup>	7.47×10 <sup>6</sup>	1.40×10 <sup>8</sup>	8.37×10 <sup>6</sup>	5.46×10 <sup>7</sup>	5.18×10 <sup>8</sup>
Lead	4.02×10 <sup>4</sup>	1.26×10 <sup>4</sup>	6.69×10 <sup>4</sup>	3.66×10 <sup>4</sup>	5.12×10 <sup>4</sup>	2.32×10 <sup>5</sup>	2.23×10 <sup>4</sup>	1.75×10 <sup>4</sup>	4.34×10 <sup>4</sup>	7.12×10 <sup>4</sup>	1.39×10 <sup>4</sup>	1.08×10 <sup>5</sup>	7.16×10 <sup>5</sup>
Uranium	1.10×10 <sup>5</sup>	1.48×10 <sup>4</sup>	2.86×10 <sup>5</sup>	7.35×10 <sup>5</sup>	6.55×10 <sup>5</sup>	1.13×10 <sup>6</sup>	5.19×10 <sup>5</sup>	3.27×10 <sup>5</sup>	3.72×10 <sup>5</sup>	4.56×10 <sup>5</sup>	3.24×10 <sup>5</sup>	4.97×10 <sup>5</sup>	5.42×10 <sup>6</sup>
Acetonitrile	0	0	0	0	0	0	0	0	0	0	0	0	0
Benzene	0	0	0	0	0	0	0	0	0	0	0	0	0
Butanol	0	0	0	0	0	0	0	0	0	0	0	0	0
PCBs	3.05×10 <sup>2</sup>	1.47×10 <sup>2</sup>	5.44×10 <sup>2</sup>	4.18×10 <sup>2</sup>	1.11×10 <sup>3</sup>	4.67×10 <sup>2</sup>	1.39×10 <sup>3</sup>	9.23×10 <sup>2</sup>	4.93×10 <sup>2</sup>	1.73×10 <sup>3</sup>	1.68×10 <sup>2</sup>	8.53×10 <sup>2</sup>	8.54×10 <sup>3</sup>
2,4,6-TCP	0	0	0	0	0	0	0	0	0	0	0	0	0

**Note:** To convert grams to ounces, multiply by 0.03527.

**Key:** butanol=n-butyl alcohol; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

**Source:** DOE 2003a; SAIC 2011.

**Table D–23. Double-Shell Tank Residual Nonradioactive Constituent Inventories – 99 Percent Retrieval (grams)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Chromium	1.85×10 <sup>5</sup>	1.03×10 <sup>5</sup>	1.99×10 <sup>5</sup>	2.79×10 <sup>4</sup>	5.09×10 <sup>4</sup>	4.73×10 <sup>5</sup>	1.04×10 <sup>6</sup>
Mercury	4.66×10 <sup>1</sup>	0	2.09	1.26×10 <sup>3</sup>	4.15×10 <sup>1</sup>	8.95×10 <sup>1</sup>	1.44×10 <sup>3</sup>
Nitrate	6.47×10 <sup>7</sup>	5.65×10 <sup>7</sup>	3.47×10 <sup>7</sup>	1.70×10 <sup>6</sup>	7.74×10 <sup>6</sup>	2.48×10 <sup>7</sup>	1.90×10 <sup>8</sup>
Lead	3.63×10 <sup>4</sup>	9.01×10 <sup>3</sup>	1.51×10 <sup>4</sup>	4.48×10 <sup>4</sup>	4.03×10 <sup>3</sup>	1.57×10 <sup>4</sup>	1.25×10 <sup>5</sup>
Uranium	2.68×10 <sup>4</sup>	1.23×10 <sup>4</sup>	3.95×10 <sup>5</sup>	3.52×10 <sup>4</sup>	5.19×10 <sup>4</sup>	2.38×10 <sup>4</sup>	5.45×10 <sup>5</sup>
Acetonitrile	7.33×10 <sup>4</sup>	9.63×10 <sup>4</sup>	5.67×10 <sup>4</sup>	1.13×10 <sup>4</sup>	2.61×10 <sup>4</sup>	3.11×10 <sup>4</sup>	2.95×10 <sup>5</sup>
Benzene	5.97	7.85	4.62	9.19×10 <sup>-1</sup>	2.13	2.53	2.40×10 <sup>1</sup>
Butanol (n-butyl alcohol)	8.59×10 <sup>6</sup>	1.13×10 <sup>7</sup>	6.63×10 <sup>6</sup>	1.32×10 <sup>6</sup>	3.06×10 <sup>6</sup>	3.64×10 <sup>6</sup>	3.45×10 <sup>7</sup>
Polychlorinated biphenyls	2.07×10 <sup>3</sup>	2.71×10 <sup>3</sup>	1.60×10 <sup>3</sup>	3.18×10 <sup>2</sup>	7.36×10 <sup>2</sup>	8.76×10 <sup>2</sup>	8.31×10 <sup>3</sup>
2,4,6-Trichlorophenol	2.75	3.62	2.13	4.23×10 <sup>-1</sup>	9.81×10 <sup>-1</sup>	1.17	1.11×10 <sup>1</sup>

**Note:** To convert grams to ounces, multiply by 0.03527.

**Source:** DOE 2003a; SAIC 2011.

**Table D–24. Single-Shell Tank Residual Nonradioactive Constituent Inventories – 99.9 Percent Retrieval (grams)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Chromium	1.62×10 <sup>4</sup>	7.87×10 <sup>3</sup>	1.11×10 <sup>4</sup>	2.20×10 <sup>4</sup>	7.34×10 <sup>4</sup>	5.60×10 <sup>3</sup>	1.20×10 <sup>5</sup>	1.05×10 <sup>5</sup>	1.21×10 <sup>4</sup>	6.13×10 <sup>4</sup>	7.95×10 <sup>3</sup>	5.11×10 <sup>4</sup>	4.95×10 <sup>5</sup>
Mercury	1.59×10 <sup>2</sup>	4.27×10 <sup>1</sup>	1.38×10 <sup>2</sup>	2.27×10 <sup>2</sup>	1.74×10 <sup>2</sup>	3.93×10 <sup>2</sup>	7.15×10 <sup>1</sup>	1.46×10 <sup>2</sup>	1.99×10 <sup>1</sup>	2.83×10 <sup>1</sup>	2.56×10 <sup>2</sup>	2.55×10 <sup>1</sup>	1.68×10 <sup>3</sup>
Nitrate	1.41×10 <sup>6</sup>	7.63×10 <sup>5</sup>	1.90×10 <sup>6</sup>	1.73×10 <sup>6</sup>	6.62×10 <sup>6</sup>	6.56×10 <sup>5</sup>	1.10×10 <sup>7</sup>	6.62×10 <sup>6</sup>	7.47×10 <sup>5</sup>	1.40×10 <sup>7</sup>	8.37×10 <sup>5</sup>	5.46×10 <sup>6</sup>	5.18×10 <sup>7</sup>
Lead	4.02×10 <sup>3</sup>	1.26×10 <sup>3</sup>	6.69×10 <sup>3</sup>	3.66×10 <sup>3</sup>	5.12×10 <sup>3</sup>	2.32×10 <sup>4</sup>	2.23×10 <sup>3</sup>	1.75×10 <sup>3</sup>	4.34×10 <sup>3</sup>	7.12×10 <sup>3</sup>	1.39×10 <sup>3</sup>	1.08×10 <sup>4</sup>	7.16×10 <sup>4</sup>
Uranium	1.10×10 <sup>4</sup>	1.48×10 <sup>3</sup>	2.86×10 <sup>4</sup>	7.35×10 <sup>4</sup>	6.55×10 <sup>4</sup>	1.13×10 <sup>5</sup>	5.19×10 <sup>4</sup>	3.27×10 <sup>4</sup>	3.72×10 <sup>4</sup>	4.56×10 <sup>4</sup>	3.24×10 <sup>4</sup>	4.97×10 <sup>4</sup>	5.42×10 <sup>5</sup>
Acetonitrile	0	0	0	0	0	0	0	0	0	0	0	0	0
Benzene	0	0	0	0	0	0	0	0	0	0	0	0	0
Butanol	0	0	0	0	0	0	0	0	0	0	0	0	0
PCBs	3.05×10 <sup>1</sup>	1.47×10 <sup>1</sup>	5.44×10 <sup>1</sup>	4.18×10 <sup>1</sup>	1.11×10 <sup>2</sup>	4.67×10 <sup>1</sup>	1.39×10 <sup>2</sup>	9.23×10 <sup>1</sup>	4.93×10 <sup>1</sup>	1.73×10 <sup>2</sup>	1.68×10 <sup>1</sup>	8.53×10 <sup>1</sup>	8.54×10 <sup>2</sup>
2,4,6-TCP	0	0	0	0	0	0	0	0	0	0	0	0	0

**Note:** To convert grams to ounces, multiply by 0.03527.

**Key:** butanol=n-butyl alcohol; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

**Source:** DOE 2003a; SAIC 2011.

**Table D–25. Double-Shell Tank Residual Nonradioactive Constituent Inventories – 99.9 Percent Retrieval (grams)**

Analyte	Tank Farm						Total
	AN	AP	AW	AY	AZ	SY	
Chromium	1.85×10 <sup>4</sup>	1.03×10 <sup>4</sup>	1.99×10 <sup>4</sup>	2.79×10 <sup>3</sup>	5.09×10 <sup>3</sup>	4.73×10 <sup>4</sup>	1.04×10 <sup>5</sup>
Mercury	4.66	0	2.09×10 <sup>-1</sup>	1.26×10 <sup>2</sup>	4.15	8.95	1.44×10 <sup>2</sup>
Nitrate	6.47×10 <sup>6</sup>	5.65×10 <sup>6</sup>	3.47×10 <sup>6</sup>	1.70×10 <sup>5</sup>	7.74×10 <sup>5</sup>	2.48×10 <sup>6</sup>	1.90×10 <sup>7</sup>
Lead	3.63×10 <sup>3</sup>	9.01×10 <sup>2</sup>	1.51×10 <sup>3</sup>	4.48×10 <sup>3</sup>	4.03×10 <sup>2</sup>	1.57×10 <sup>3</sup>	1.25×10 <sup>4</sup>
Uranium	2.68×10 <sup>3</sup>	1.23×10 <sup>3</sup>	3.95×10 <sup>4</sup>	3.52×10 <sup>3</sup>	5.19×10 <sup>3</sup>	2.38×10 <sup>3</sup>	5.45×10 <sup>4</sup>
Acetonitrile	7.33×10 <sup>3</sup>	9.63×10 <sup>3</sup>	5.67×10 <sup>3</sup>	1.13×10 <sup>3</sup>	2.61×10 <sup>3</sup>	3.11×10 <sup>3</sup>	2.95×10 <sup>4</sup>
Benzene	5.97×10 <sup>-1</sup>	7.85×10 <sup>-1</sup>	4.62×10 <sup>-1</sup>	9.19×10 <sup>-2</sup>	2.13×10 <sup>-1</sup>	2.53×10 <sup>-1</sup>	2.40
Butanol (n-butyl alcohol)	8.59×10 <sup>5</sup>	1.13×10 <sup>6</sup>	6.63×10 <sup>5</sup>	1.32×10 <sup>5</sup>	3.06×10 <sup>5</sup>	3.64×10 <sup>5</sup>	3.45×10 <sup>6</sup>
Polychlorinated biphenyls	2.07×10 <sup>2</sup>	2.71×10 <sup>2</sup>	1.60×10 <sup>2</sup>	3.18×10 <sup>1</sup>	7.36×10 <sup>1</sup>	8.76×10 <sup>1</sup>	8.31×10 <sup>2</sup>
2,4,6-Trichlorophenol	2.75×10 <sup>-1</sup>	3.62×10 <sup>-1</sup>	2.13×10 <sup>-1</sup>	4.23×10 <sup>-2</sup>	9.81×10 <sup>-2</sup>	1.17×10 <sup>-1</sup>	1.11

**Note:** To convert grams to ounces, multiply by 0.03527.

**Source:** DOE 2003a; SAIC 2011.



#### D.1.4 Historical Leaks and Other Releases

Leaks from SSTs have been suspected, investigated, and, in some cases, confirmed. Currently, 67 of Hanford's 149 SSTs are listed as "known or suspected" leakers in the monthly *Waste Tank Summary Report* (Hanlon 2003). This information was compiled in the late 1980s and early 1990s and reflects the state of knowledge at that time. The document contains information of varying quality. For example, leak volumes for tanks 241-SX-113, 241-SX-115, and 241-T-106 are well documented; however, for 19 of the tanks listed in the *Waste Tank Summary Report*, the leak volume estimates provided were based on limited supporting data. The leak volume estimates for the remaining 45 tanks are based on various methods and are further described in the *Waste Tank Summary Report*. Estimates of the total leak losses in the *Waste Tank Summary Report* range from 1.89 million to 3.97 million liters (0.5 million to 1.05 million gallons). Vadose zone field investigations have not been completed for all tank farms, and uncertainties remain regarding the estimated volumes of past leaks; the higher value of 3.97 million liters (1.05 million gallons) reported in the *Waste Tank Summary Report* was used for analysis purposes in this *TC & WM EIS*.

Current efforts to characterize impacts of leaks from the SSTs have focused on developing estimates of the inventories lost to the vadose zone. These efforts include gamma ray contamination detection mapping of the dry wells at the 12 SST farms using a gamma source and ongoing field investigations for four sets of tank farms. Using this information, estimates of inventories lost to the vadose zone have been developed. Analysis results for 20 tanks are documented in the *Inventory and Source Term Data Package* (DOE 2003a); the field investigation reports (CH2M HILL 2002; Connelly 2007, 2008; Jones et al. 2001; Myers 2005); and SIM [the Hanford Soil Inventory Model], Revision 1 (Corbin et al. 2005). This analysis constitutes the best available basis for estimating leak inventories from all SSTs suspected to be leaking. The approach used to extend the available information was to assume that the concentration in a leak from a tank in a given tank farm for which a documented estimate is not available is equal to the average concentration in leaks from tanks in the same tank farm for which documented estimates are available. For losses from the tank farms for which a documented inventory estimate is not available, i.e., the AX tank farm, tank volumes and times of operation were reviewed, and the tank farm was associated with a tank farm for which a documented inventory estimate is available. Thus, average concentrations from the AX tank farm were assumed to be equal to those of documented losses from the A tank farm. The inventory in a leak event was calculated as the product of the concentration in the leak and the leak volume. Results of this analysis are summarized in Tables D-26 through D-29, which present historical leaks (underground releases from the SSTs) and unplanned releases (at or near ground level at the SST farms) of radioactive and nonradioactive constituents from the tank farms.

**Table D–26. Historical Single-Shell Tank Radioactive Constituent Leak Inventories (curies)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
H-3 (tritium)	8.28×10 <sup>-2</sup>	3.20×10 <sup>-3</sup>	1.52×10 <sup>1</sup>	1.09×10 <sup>1</sup>	3.13	1.23	7.12	9.61×10 <sup>1</sup>	5.33×10 <sup>1</sup>	1.08×10 <sup>2</sup>	2.56	6.44	3.04×10 <sup>2</sup>
C-14	1.13×10 <sup>-1</sup>	4.38×10 <sup>-3</sup>	3.10	5.17×10 <sup>-1</sup>	2.20×10 <sup>-1</sup>	1.46×10 <sup>-1</sup>	5.53×10 <sup>-1</sup>	4.79	9.52	1.50×10 <sup>1</sup>	3.40×10 <sup>-1</sup>	1.60×10 <sup>-1</sup>	3.45×10 <sup>1</sup>
Sr-90	2.69×10 <sup>2</sup>	1.04×10 <sup>1</sup>	7.61×10 <sup>3</sup>	4.13×10 <sup>3</sup>	1.49×10 <sup>3</sup>	2.63×10 <sup>2</sup>	4.52×10 <sup>3</sup>	2.29×10 <sup>4</sup>	2.43×10 <sup>4</sup>	5.73×10 <sup>4</sup>	3.17×10 <sup>3</sup>	5.79×10 <sup>2</sup>	1.27×10 <sup>5</sup>
Tc-99	1.24	4.80×10 <sup>-2</sup>	2.18×10 <sup>1</sup>	4.92	2.10	6.61	3.87	3.75×10 <sup>1</sup>	6.74×10 <sup>1</sup>	1.07×10 <sup>2</sup>	2.40	3.57	2.58×10 <sup>2</sup>
I-129	1.46×10 <sup>-3</sup>	5.64×10 <sup>-5</sup>	4.20×10 <sup>-2</sup>	9.35×10 <sup>-3</sup>	3.98×10 <sup>-3</sup>	2.59×10 <sup>-3</sup>	7.44×10 <sup>-3</sup>	7.10×10 <sup>-2</sup>	1.30×10 <sup>-1</sup>	2.06×10 <sup>-1</sup>	4.59×10 <sup>-3</sup>	4.50×10 <sup>-3</sup>	4.83×10 <sup>-1</sup>
Cs-137	4.62×10 <sup>3</sup>	1.78×10 <sup>2</sup>	2.64×10 <sup>4</sup>	4.22×10 <sup>3</sup>	1.54×10 <sup>3</sup>	1.82×10 <sup>4</sup>	1.14×10 <sup>4</sup>	1.26×10 <sup>5</sup>	2.49×10 <sup>4</sup>	1.58×10 <sup>5</sup>	5.64×10 <sup>3</sup>	8.57×10 <sup>3</sup>	3.90×10 <sup>5</sup>
U-233, -234, -235, -238	5.02×10 <sup>-3</sup>	1.94×10 <sup>-4</sup>	2.34×10 <sup>-1</sup>	7.16	3.06	5.41×10 <sup>-3</sup>	8.22×10 <sup>-2</sup>	4.20×10 <sup>-1</sup>	3.49×10 <sup>-1</sup>	3.16	1.33×10 <sup>-1</sup>	1.23×10 <sup>-1</sup>	1.47×10 <sup>1</sup>
Np-237	3.87×10 <sup>-3</sup>	1.50×10 <sup>-4</sup>	6.74×10 <sup>-2</sup>	2.64×10 <sup>-2</sup>	1.12×10 <sup>-2</sup>	2.29×10 <sup>-2</sup>	2.52×10 <sup>-2</sup>	1.65×10 <sup>-1</sup>	2.33×10 <sup>-1</sup>	3.86×10 <sup>-1</sup>	1.15×10 <sup>-2</sup>	2.13×10 <sup>-2</sup>	9.74×10 <sup>-1</sup>
Pu-239, -240	7.30×10 <sup>-1</sup>	2.82×10 <sup>-2</sup>	4.87	3.24	1.38	5.94×10 <sup>-1</sup>	1.64	8.23	1.28×10 <sup>1</sup>	2.87×10 <sup>1</sup>	1.78	1.39	6.54×10 <sup>1</sup>

**Key:** C=carbon; Cs=cesium; H=hydrogen; I=iodine; Np=neptunium; Pu=plutonium; Sr=strontium; Tc=technetium; U=uranium.

**Source:** SAIC 2011.

**Table D–27. Historical Single-Shell Tank Nonradioactive Constituent Leak Inventories (grams)<sup>a</sup>**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Chromium	8.44×10 <sup>3</sup>	3.26×10 <sup>2</sup>	2.35×10 <sup>5</sup>	4.97×10 <sup>4</sup>	2.12×10 <sup>4</sup>	4.15×10 <sup>4</sup>	7.81×10 <sup>5</sup>	3.89×10 <sup>6</sup>	1.10×10 <sup>6</sup>	3.07×10 <sup>6</sup>	8.47×10 <sup>4</sup>	1.61×10 <sup>5</sup>	9.44×10 <sup>6</sup>
Mercury	1.74	6.72×10 <sup>-2</sup>	3.55×10 <sup>2</sup>	3.40×10 <sup>1</sup>	1.45×10 <sup>1</sup>	2.12×10 <sup>1</sup>	6.49×10 <sup>1</sup>	3.57×10 <sup>1</sup>	2.35×10 <sup>2</sup>	1.34×10 <sup>3</sup>	2.71×10 <sup>1</sup>	7.16×10 <sup>1</sup>	2.20×10 <sup>3</sup>
Nitrate	5.19×10 <sup>5</sup>	2.00×10 <sup>4</sup>	3.35×10 <sup>7</sup>	1.65×10 <sup>7</sup>	7.04×10 <sup>6</sup>	4.82×10 <sup>6</sup>	2.63×10 <sup>7</sup>	1.14×10 <sup>8</sup>	6.74×10 <sup>7</sup>	2.44×10 <sup>8</sup>	4.18×10 <sup>7</sup>	1.16×10 <sup>7</sup>	5.68×10 <sup>8</sup>
Lead	5.13×10 <sup>2</sup>	1.98×10 <sup>1</sup>	5.10×10 <sup>4</sup>	5.51×10 <sup>3</sup>	2.35×10 <sup>3</sup>	6.87×10 <sup>3</sup>	1.07×10 <sup>4</sup>	5.75×10 <sup>4</sup>	3.53×10 <sup>4</sup>	1.29×10 <sup>5</sup>	2.49×10 <sup>3</sup>	8.41×10 <sup>2</sup>	3.02×10 <sup>5</sup>
Uranium	4.52×10 <sup>3</sup>	1.74×10 <sup>2</sup>	2.44×10 <sup>5</sup>	1.06×10 <sup>7</sup>	4.52×10 <sup>6</sup>	2.88×10 <sup>3</sup>	1.19×10 <sup>5</sup>	5.52×10 <sup>5</sup>	3.82×10 <sup>5</sup>	1.30×10 <sup>6</sup>	1.04×10 <sup>5</sup>	1.81×10 <sup>5</sup>	1.80×10 <sup>7</sup>
Butanol	5.18×10 <sup>2</sup>	2.00×10 <sup>1</sup>	9.41×10 <sup>4</sup>	6.56×10 <sup>3</sup>	2.79×10 <sup>3</sup>	1.89×10 <sup>4</sup>	3.86×10 <sup>-2</sup>	6.37×10 <sup>1</sup>	3.78×10 <sup>5</sup>	6.13×10 <sup>5</sup>	9.40×10 <sup>3</sup>	1.77×10 <sup>3</sup>	1.13×10 <sup>6</sup>

<sup>a</sup> Acetonitrile, benzene, polychlorinated biphenyls, and 2,4,6-trichlorophenol not reported.

**Note:** To convert grams to ounces, multiply by 0.03527.

**Key:** butanol=n-butyl alcohol.

**Source:** SAIC 2011.

**Table D-28. Single-Shell Tank Farms Unplanned Releases Radioactive Constituent Inventories (curies)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
H-3 (tritium)	N/A	N/A	4.58×10 <sup>-1</sup>	N/A	6.49×10 <sup>-1</sup>	2.32×10 <sup>1</sup>	N/A	N/A	N/A	5.88×10 <sup>-2</sup>	N/A	5.06×10 <sup>-2</sup>	2.44×10 <sup>1</sup>
C-14	N/A	N/A	7.84×10 <sup>-2</sup>	N/A	8.60×10 <sup>-3</sup>	1.89×10 <sup>-1</sup>	N/A	N/A	N/A	7.84×10 <sup>-4</sup>	N/A	8.58×10 <sup>-4</sup>	2.77×10 <sup>-1</sup>
Sr-90	N/A	N/A	5.50×10 <sup>1</sup>	N/A	7.13	1.07×10 <sup>2</sup>	N/A	N/A	N/A	6.88×10 <sup>-1</sup>	N/A	5.03	1.75×10 <sup>2</sup>
Tc-99	N/A	N/A	2.95	N/A	2.15×10 <sup>-2</sup>	1.67	N/A	N/A	N/A	2.01×10 <sup>-3</sup>	N/A	2.27×10 <sup>-2</sup>	4.67
I-129	N/A	N/A	1.77×10 <sup>-3</sup>	N/A	1.88×10 <sup>-4</sup>	2.48×10 <sup>-2</sup>	N/A	N/A	N/A	1.71×10 <sup>-5</sup>	N/A	2.53×10 <sup>-5</sup>	2.68×10 <sup>-2</sup>
Cs-137	N/A	N/A	3.44×10 <sup>2</sup>	N/A	4.87×10 <sup>1</sup>	8.55×10 <sup>2</sup>	N/A	N/A	N/A	4.59	N/A	5.18×10 <sup>1</sup>	1.30×10 <sup>3</sup>
U-233, -234, -235, -238	N/A	N/A	1.57×10 <sup>-3</sup>	N/A	2.22×10 <sup>-3</sup>	1.49×10 <sup>-2</sup>	N/A	N/A	N/A	2.02×10 <sup>-4</sup>	N/A	1.30×10 <sup>-3</sup>	2.02×10 <sup>-2</sup>
Np-237	N/A	N/A	9.69×10 <sup>-3</sup>	N/A	4.95×10 <sup>-4</sup>	5.57×10 <sup>-3</sup>	N/A	N/A	N/A	4.52×10 <sup>-5</sup>	N/A	1.31×10 <sup>-4</sup>	1.59×10 <sup>-2</sup>
Pu-239, -240	N/A	N/A	1.35×10 <sup>-1</sup>	N/A	5.98×10 <sup>-3</sup>	9.33×10 <sup>-1</sup>	N/A	N/A	N/A	6.53×10 <sup>-4</sup>	N/A	1.17×10 <sup>-2</sup>	1.09

**Key:** C=carbon; Cs=cesium; H=hydrogen; I=iodine; N/A=not applicable; Np=neptunium; Pu=plutonium; Sr=strontium; Tc=technetium; U=uranium.

**Source:** SAIC 2011.

**Table D-29. Single-Shell Tank Farms Unplanned Releases Nonradioactive Constituent Inventories (grams)<sup>a</sup>**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Chromium	N/A	N/A	3.53×10 <sup>4</sup>	N/A	3.81×10 <sup>4</sup>	3.93×10 <sup>4</sup>	N/A	N/A	N/A	3.47×10 <sup>3</sup>	N/A	3.76×10 <sup>2</sup>	1.17×10 <sup>5</sup>
Mercury	N/A	N/A	8.40	N/A	3.45×10 <sup>1</sup>	3.92	N/A	N/A	N/A	3.13	N/A	6.14×10 <sup>-1</sup>	5.05×10 <sup>1</sup>
Nitrate	N/A	N/A	3.24×10 <sup>6</sup>	N/A	1.17×10 <sup>7</sup>	9.68×10 <sup>6</sup>	N/A	N/A	N/A	1.07×10 <sup>6</sup>	N/A	2.59×10 <sup>4</sup>	2.57×10 <sup>7</sup>
Lead	N/A	N/A	2.07×10 <sup>3</sup>	N/A	0.00	2.16×10 <sup>4</sup>	N/A	N/A	N/A	0.00	N/A	0.00	2.37×10 <sup>4</sup>
Uranium	N/A	N/A	1.81×10 <sup>3</sup>	N/A	3.30×10 <sup>3</sup>	4.64×10 <sup>3</sup>	N/A	N/A	N/A	2.99×10 <sup>2</sup>	N/A	1.93×10 <sup>3</sup>	1.20×10 <sup>4</sup>
Butanol	N/A	N/A	3.86×10 <sup>-1</sup>	N/A	0.00	6.47×10 <sup>2</sup>	N/A	N/A	N/A	1.15×10 <sup>-2</sup>	N/A	0.00	6.47×10 <sup>2</sup>

<sup>a</sup> Acetonitrile, benzene, polychlorinated biphenyls, and 2,4,6-trichlorophenol not reported.

**Note:** To convert grams to ounces, multiply by 0.03527.

**Key:** butanol=n-butyl alcohol; N/A=not applicable.

**Source:** SAIC 2011.

### **D.1.5 Discharges to Cribs and Trenches (Ditches)**

During the early years of Hanford operations, three classes of liquid waste were produced during fuel reprocessing operations. Uncontaminated aqueous waste, such as cooling water, was discharged to surface ponds. High-volume waste streams with modest radioactive and chemical contamination were discharged to cribs and trenches (ditches). Waste streams that contained isotopes with long half-lives and fission products with high radiation/short half-lives were transferred to underground SSTs. Because many of the cribs and trenches (ditches) are in close proximity to the SST farms, in some cases it is very difficult to clearly identify contamination sources in the vadose zone or groundwater.

In parallel with the development of tank leak inventory estimates, inventory estimates were developed for intentional discharges of tank waste to cribs and trenches (ditches) near the B/BX/BY and T/TX/TY waste management areas (Simpson, Corbin, and Agnew 2001). The proximity of the cribs and trenches (ditches) to the tank farms warrants inclusion of these inventory estimates because they may be appropriate in tank farm vadose zone analyses. All volume and inventory estimates for discharges to cribs and trenches (ditches) were derived from SIM (Corbin et al. 2005).

SIM is an extension and enhancement of previous efforts to quantify contaminant inventories in the Hanford waste storage tanks. SIM provides more details of what went into specific waste sites other than the tanks than previously estimated and provides a more complete picture of these discharges. It is based on historical records and data from various Hanford process facilities that extracted plutonium and uranium from spent nuclear fuel (SNF). SIM generates inventory and uncertainty estimates for liquid waste disposal sites, unplanned releases, and tank leaks over the operating lifetimes in intervals of 1 year from 1944 to 2001 (Corbin et al. 2005).

Information on the vertical distribution of chemicals and radionuclides that were intentionally discharged to the soil column is available. A number of field investigations have examined the contaminant profile in a number of cribs. In general, the levels of contamination have varied, with the highest contaminant concentrations being associated with less-mobile radionuclides like cesium-137 and strontium-90 near the release points. Most mobile contaminants, such as tritium, technetium-99, and nitrate, are generally found in finer-grained materials at minor concentrations. Because of the high volumes of fluids discharged to the cribs, any contaminants that were not strongly sorbed by the soil were rapidly transferred to groundwater. Recent field investigations conducted by CH2M HILL Hanford Group at the 216-B-38 Trench (ditch) provide strong evidence that the trenches (ditches) functioned as designed. Soil analyses as a function of depth show the location of mobile constituents, such as nitrate, and sorbed species, such as cesium-137 and strontium-90 (DOE 2003a).

Estimates of volumes and inventories of radioactive and chemical constituents discharged to six sets of cribs and trenches (ditches) are presented in Tables D-30 and D-31. (Note: The T Trenches and TX Trenches are considered one set.) The grouping of the 33 cribs and trenches (ditches) provided in Tables D-30 and D-31 is as follows:

- T Cribs: 216-T-5, 216-T-7 (2)
- T Trenches: 216-T-14 through 216-T-19 (6)
- TX Trenches: 216-T-21 through 216-T-25 (5)
- TY Cribs: 216-T-26, 216-T-28, 216-T-32 (3)
- B Cribs: 216-B-7 A&B, 216-B-8 (2)
- BX Trenches: 216-B-35 through 216-B-42 (8)
- BY Cribs: 216-B-43 through 216-B-49 (7)

**Table D–30. Radioactive Constituent Discharges to Cribs and Trenches (Ditches)**

	T Cribs	T Trenches	TX Trenches	TY Cribs	B Cribs	BX Trenches	BY Cribs	Total
Volume discharged (liters)	1.10×10 <sup>8</sup>	4.60×10 <sup>8</sup>	8.02×10 <sup>6</sup>	8.24×10 <sup>7</sup>	7.99×10 <sup>7</sup>	1.49×10 <sup>7</sup>	3.38×10 <sup>7</sup>	7.89×10 <sup>8</sup>
<b>Analyte (curies)</b>								
Hydrogen-3 (tritium)	1.00×10 <sup>-1</sup>	5.15×10 <sup>3</sup>	4.89×10 <sup>1</sup>	2.95	2.10×10 <sup>-2</sup>	9.09×10 <sup>1</sup>	2.11×10 <sup>2</sup>	5.50×10 <sup>3</sup>
Carbon-14	3.98×10 <sup>-1</sup>	5.94×10 <sup>-1</sup>	6.48×10 <sup>-1</sup>	3.80	1.71×10 <sup>-1</sup>	1.44	8.17	1.52×10 <sup>1</sup>
Strontium-90	3.96×10 <sup>2</sup>	3.41×10 <sup>2</sup>	5.77×10 <sup>2</sup>	5.80×10 <sup>2</sup>	1.78×10 <sup>3</sup>	1.16×10 <sup>3</sup>	4.74×10 <sup>3</sup>	9.57×10 <sup>3</sup>
Technetium-99	2.05×10 <sup>-1</sup>	9.41×10 <sup>-1</sup>	1.62	1.80	1.75×10 <sup>-1</sup>	8.40	1.28×10 <sup>2</sup>	1.42×10 <sup>2</sup>
Iodine-129	1.49×10 <sup>-5</sup>	8.28×10 <sup>-3</sup>	1.41×10 <sup>-2</sup>	1.70×10 <sup>-2</sup>	6.94×10 <sup>-4</sup>	3.09×10 <sup>-2</sup>	1.65×10 <sup>-1</sup>	2.36×10 <sup>-1</sup>
Cesium-137	4.60×10 <sup>2</sup>	1.82×10 <sup>3</sup>	3.67×10 <sup>3</sup>	6.30×10 <sup>2</sup>	5.42×10 <sup>2</sup>	6.17×10 <sup>3</sup>	1.62×10 <sup>3</sup>	1.49×10 <sup>4</sup>
Uranium-233, -234, -235, -238	2.45×10 <sup>-1</sup>	1.35×10 <sup>-1</sup>	1.85×10 <sup>-1</sup>	3.00	1.58	3.40×10 <sup>-1</sup>	7.17×10 <sup>-1</sup>	6.21
Neptunium-237	1.10×10 <sup>-1</sup>	2.60×10 <sup>-2</sup>	3.73×10 <sup>-2</sup>	8.01×10 <sup>-2</sup>	5.12×10 <sup>-2</sup>	1.07×10 <sup>-1</sup>	1.02	1.43
Plutonium-239, -240	2.81×10 <sup>2</sup>	1.47×10 <sup>1</sup>	3.71	9.45×10 <sup>1</sup>	1.64×10 <sup>2</sup>	6.96	2.82×10 <sup>1</sup>	5.94×10 <sup>2</sup>

Note: To convert liters to gallons, multiply by 0.26417.

Source: SAIC 2011.

**Table D–31. Nonradioactive Constituent Discharges to Cribs and Trenches (Ditches)<sup>a</sup>**

	T Cribs	T Trenches	TX Trenches	TY Cribs	B Cribs	BX Trenches	BY Cribs	Total
Volume discharged (liters)	1.10×10 <sup>8</sup>	4.60×10 <sup>8</sup>	8.02×10 <sup>6</sup>	8.24×10 <sup>7</sup>	7.99×10 <sup>7</sup>	1.49×10 <sup>7</sup>	3.38×10 <sup>7</sup>	7.89×10 <sup>8</sup>
<b>Analyte (kilograms)</b>								
Chromium	2.93×10 <sup>4</sup>	2.61×10 <sup>3</sup>	2.87×10 <sup>3</sup>	1.75×10 <sup>4</sup>	1.79×10 <sup>4</sup>	5.05×10 <sup>3</sup>	5.82×10 <sup>3</sup>	8.09×10 <sup>4</sup>
Mercury	0	6.13	2.86	8.19	1.23×10 <sup>-2</sup>	5.26	1.09×10 <sup>1</sup>	3.33×10 <sup>1</sup>
Nitrate	6.79×10 <sup>6</sup>	8.13×10 <sup>5</sup>	1.04×10 <sup>6</sup>	3.17×10 <sup>6</sup>	4.65×10 <sup>6</sup>	1.77×10 <sup>6</sup>	6.72×10 <sup>6</sup>	2.50×10 <sup>7</sup>
Lead	0	5.50	0	1.46×10 <sup>1</sup>	7.69	0	0	2.78×10 <sup>1</sup>
Uranium	3.63×10 <sup>2</sup>	2.00×10 <sup>2</sup>	2.74×10 <sup>2</sup>	1.11×10 <sup>3</sup>	3.88×10 <sup>2</sup>	5.04×10 <sup>2</sup>	1.06×10 <sup>3</sup>	3.90×10 <sup>3</sup>

<sup>a</sup> Acetonitrile, benzene, polychlorinated biphenyls, and 2,4,6-trichlorophenol not reported.

Note: To convert kilograms to pounds, multiply by 2.2046; liters to gallons, by 0.26417.

Source: SAIC 2011.

### D.1.6 Tank Waste Retrieval Leaks

The amount of leakage that may occur during retrieval of waste from SSTs varies with the details of the individual tank condition and retrieval methods and is largely uncertain. During actual retrieval operations, leak detection and monitoring would be used to minimize leakage to the extent practicable.

The SSTs were constructed as early as 1943. Currently, 67 of Hanford's 149 SSTs are listed as "known or suspected" leakers. The SSTs were formally removed from service in 1980, but still contain approximately 120 million liters (32 million gallons) of waste. Although the River Protection Project plans to minimize the introduction of liquids into suspected leakers (utilizing vacuum-based retrieval), for analysis purposes, all SSTs were assumed to leak during retrieval. The *Tank Waste Remediation System, Hanford Site, Richland, Washington, Final Environmental Impact Statement* (DOE and Ecology 1996) assumed an average of 15,140 liters (4,000 gallons) would leak during SST retrieval. Due to limitations on currently employed leak detection equipment, this assumption was carried forward in this EIS. The leak detection monitoring and mitigation strategy developed for the tank 241-S-112 retrieval demonstration (Hanson 2003) estimated that the best of the three available leak detection methods utilized gamma ray contamination detection mapping of the dry wells and neutron contamination detection mapping of the soil. The 95th percentile upper confidence limit with this method estimated leak detection within a leak volume ranging from 1,140 to 68,000 liters (300 to 18,000 gallons), depending on where the

leak originated in relation to the dry wells. In-tank liquid-balance leak detection methods were even less sensitive, ranging from 68,000 to 310,000 liters (18,000 to 82,000 gallons) at the 95th percentile upper confidence limit. Technologies to assist in mitigation and improve detection of leakage are currently being evaluated and tested by the River Protection Project. Testing conducted on resistivity-based technologies over a 110-day period in 2003 at the Hanford 105A mock-tank test site provided encouraging data for the potential future use of much more sensitive leak detection capabilities (Barnett et al. 2003). High-resolution resistivity has been used on a number of SSTs starting in 2004. See Appendix E, Section E.1.2.2.5.1, for further details.

In addition to leak volume specification, estimating the inventory of each constituent released to the vadose zone requires knowledge of the retrieval method used and the tank inventories addressed during retrieval operations. Retrieval operations that may result in leakage are those that use liquid to sluice salt cake and sludge from the SSTs. Current analysis projects that three volumes of sluicing liquid would remove one volume of SST solids (DOE 2003b). A conservative estimate of the inventory present in the tank during retrieval is provided by the estimate of current tank inventories. These estimates are summarized in Section D.1.1. Given these considerations, the concentration of a constituent in leak liquid would be one-quarter of the volumetric concentration of the constituent in the tank prior to retrieval. For a single tank, the loss of a constituent in leakage during retrieval is estimated using the following equation:

$$M_{\text{retrieval}} = \frac{M_{BBIT}}{V_{BBIT}} \times \frac{1}{4} \times V_{rb}$$

where:

$M_{\text{retrieval}}$  = amount of the radioactive or chemical constituent in tank waste retrieval leaks for a tank, curies or grams

$M_{BBIT}$  = inventory of constituent in the tank, curies or grams

$V_{BBIT}$  = inventory volume in the tank, liters

$V_{rb}$  = volume lost during retrieval, 15,140 liters (4,000 gallons)

The constituent loss estimates for all tank farms were calculated as the sum of the losses from the individual tanks in the tank farm. Estimated retrieval losses of radioactive and chemical constituents on a tank-farm basis are presented in Tables D-32 and D-33.

**Table D-32. Single-Shell Tank Radioactive Constituent Tank Waste Retrieval Leak Inventories (curies)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
H-3 (tritium)	2.10	1.75	3.22×10 <sup>-1</sup>	5.30×10 <sup>-1</sup>	3.78	5.56×10 <sup>-1</sup>	4.39	4.33	4.66×10 <sup>-1</sup>	5.86	3.59×10 <sup>-1</sup>	4.20	2.86×10 <sup>1</sup>
C-14	4.85×10 <sup>-1</sup>	6.15×10 <sup>-1</sup>	8.93×10 <sup>-2</sup>	2.23×10 <sup>-1</sup>	1.59	1.02×10 <sup>-1</sup>	1.15	7.71×10 <sup>-1</sup>	2.97×10 <sup>-1</sup>	1.71	8.03×10 <sup>-2</sup>	1.02	8.13
Sr-90	1.83×10 <sup>5</sup>	3.11×10 <sup>5</sup>	1.47×10 <sup>4</sup>	1.71×10 <sup>4</sup>	5.13×10 <sup>3</sup>	6.95×10 <sup>4</sup>	6.33×10 <sup>3</sup>	1.35×10 <sup>5</sup>	7.15×10 <sup>3</sup>	6.78×10 <sup>3</sup>	2.56×10 <sup>3</sup>	8.58×10 <sup>3</sup>	7.68×10 <sup>5</sup>
Tc-99	5.06	1.48×10 <sup>1</sup>	1.26	2.31	7.46	2.72	6.36	5.55	1.71	1.01×10 <sup>1</sup>	1.36	6.97	6.57×10 <sup>1</sup>
I-129	7.38×10 <sup>-3</sup>	6.54×10 <sup>-3</sup>	9.81×10 <sup>-4</sup>	2.48×10 <sup>-3</sup>	1.57×10 <sup>-2</sup>	4.85×10 <sup>-3</sup>	1.35×10 <sup>-2</sup>	9.27×10 <sup>-3</sup>	1.59×10 <sup>-3</sup>	1.92×10 <sup>-2</sup>	9.23×10 <sup>-4</sup>	1.33×10 <sup>-2</sup>	9.58×10 <sup>-2</sup>
Cs-137	9.37×10 <sup>3</sup>	1.07×10 <sup>4</sup>	2.05×10 <sup>3</sup>	2.60×10 <sup>3</sup>	6.24×10 <sup>3</sup>	1.01×10 <sup>4</sup>	6.14×10 <sup>3</sup>	1.03×10 <sup>4</sup>	1.75×10 <sup>3</sup>	7.61×10 <sup>3</sup>	6.27×10 <sup>2</sup>	8.29×10 <sup>3</sup>	7.57×10 <sup>4</sup>
U-233, -234, -235, -238	5.43×10 <sup>-1</sup>	3.15×10 <sup>-2</sup>	1.58×10 <sup>-1</sup>	5.00×10 <sup>-1</sup>	1.55×10 <sup>-1</sup>	2.18	1.31×10 <sup>-1</sup>	1.83×10 <sup>-1</sup>	1.95×10 <sup>-1</sup>	1.56×10 <sup>-1</sup>	2.21×10 <sup>-1</sup>	1.84×10 <sup>-1</sup>	4.64
Np-237	1.24×10 <sup>-2</sup>	8.43×10 <sup>-3</sup>	2.62×10 <sup>-3</sup>	3.94×10 <sup>-3</sup>	2.52×10 <sup>-2</sup>	3.42×10 <sup>-2</sup>	2.55×10 <sup>-2</sup>	1.96×10 <sup>-2</sup>	4.92×10 <sup>-3</sup>	3.56×10 <sup>-2</sup>	2.33×10 <sup>-3</sup>	2.59×10 <sup>-2</sup>	2.01×10 <sup>-1</sup>
Pu-239, -240	7.47×10 <sup>1</sup>	5.70×10 <sup>1</sup>	1.52×10 <sup>1</sup>	2.53×10 <sup>1</sup>	3.67	2.21×10 <sup>2</sup>	1.29×10 <sup>1</sup>	1.74×10 <sup>2</sup>	2.01×10 <sup>1</sup>	7.47×10 <sup>1</sup>	3.83	1.45×10 <sup>1</sup>	6.97×10 <sup>2</sup>

**Key:** C=carbon; Cs=cesium; H=hydrogen; I=iodine; Np=neptunium; Pu=plutonium; Sr=strontium; Tc=technetium; U=uranium.

**Source:** SAIC 2011.

**Table D-33. Single-Shell Tank Nonradioactive Constituent Tank Waste Retrieval Leak Inventories (kilograms)**

Analyte	Tank Farm												Total
	A	AX	B	BX	BY	C	S	SX	T	TX	TY	U	
Chromium	1.19×10 <sup>2</sup>	4.39×10 <sup>1</sup>	1.20×10 <sup>2</sup>	2.31×10 <sup>2</sup>	2.29×10 <sup>2</sup>	4.15×10 <sup>1</sup>	2.73×10 <sup>2</sup>	4.21×10 <sup>2</sup>	1.38×10 <sup>2</sup>	2.05×10 <sup>2</sup>	7.17×10 <sup>1</sup>	1.92×10 <sup>2</sup>	2.08×10 <sup>3</sup>
Mercury	4.02	2.54	1.18	2.71	4.93×10 <sup>-1</sup>	2.44	1.73×10 <sup>-1</sup>	1.20	1.37×10 <sup>-1</sup>	2.53×10 <sup>-1</sup>	1.57	3.32×10 <sup>-1</sup>	1.70×10 <sup>1</sup>
Nitrate	5.53×10 <sup>3</sup>	1.02×10 <sup>5</sup>	7.29×10 <sup>4</sup>	2.89×10 <sup>4</sup>	1.55×10 <sup>4</sup>	9.53×10 <sup>4</sup>	2.40×10 <sup>4</sup>	5.35×10 <sup>4</sup>	6.19×10 <sup>4</sup>	3.74×10 <sup>4</sup>	7.84×10 <sup>3</sup>	9.72×10 <sup>4</sup>	6.02×10 <sup>5</sup>
Lead	8.21×10 <sup>1</sup>	6.98×10 <sup>1</sup>	4.42×10 <sup>1</sup>	4.26×10 <sup>1</sup>	1.47×10 <sup>1</sup>	3.33×10 <sup>3</sup>	5.05	7.89	4.96×10 <sup>1</sup>	1.90×10 <sup>1</sup>	1.28×10 <sup>1</sup>	9.86×10 <sup>1</sup>	3.77×10 <sup>3</sup>
Uranium	1.79×10 <sup>2</sup>	2.93×10 <sup>1</sup>	2.15×10 <sup>2</sup>	7.25×10 <sup>2</sup>	1.95×10 <sup>2</sup>	8.04×10 <sup>2</sup>	1.35×10 <sup>2</sup>	2.40×10 <sup>2</sup>	2.79×10 <sup>2</sup>	1.59×10 <sup>2</sup>	3.23×10 <sup>2</sup>	2.50×10 <sup>2</sup>	3.53×10 <sup>3</sup>
Acetonitrile	0	0	0	0	0	0	0	0	0	0	0	0	0
Benzene	0	0	0	0	0	0	0	0	0	0	0	0	0
Butanol	0	0	0	0	0	0	0	0	0	0	0	0	0
Polychlorinated biphenyls	1.59×10 <sup>-1</sup>	1.06×10 <sup>-1</sup>	4.25×10 <sup>-1</sup>	3.19×10 <sup>-1</sup>	3.19×10 <sup>-1</sup>	4.25×10 <sup>-1</sup>	3.19×10 <sup>-1</sup>	3.99×10 <sup>-1</sup>	4.25×10 <sup>-1</sup>	4.78×10 <sup>-1</sup>	1.59×10 <sup>-1</sup>	4.25×10 <sup>-1</sup>	3.96
2,4,6-Trichlorophenol	0	0	0	0	0	0	0	0	0	0	0	0	0

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** butanol=n-butyl alcohol.

**Source:** SAIC 2011.

### **D.1.7 Inventories and Flowsheets**

Retrieval of tank waste, processing and stabilization of the waste streams, and closure of the tank farms would generate a number of waste forms for onsite disposal. Volume and constituent inventory estimates for these waste forms are based on the mass balances that are applicable for the set of process operations proposed for each alternative (CEES 2007, 2010; DOE 2003a, 2003b), as well as on additional assumptions related to the generation and recovery efficiencies of the volatile constituents tritium, carbon dioxide, nitrate, mercury, and iodine during thermal processing.

Assumptions applied to these constituents for thermal processes under all alternatives include the following:

- Iodine-129: 80 percent goes to offgas (CEES 2010; Whyatt, Shade, and Stegen 1996)
- Carbon-14: 100 percent to offgas (Zamecnik and Crawford 2003)
- Tritium: 100 percent to offgas (BNI 2002)
- Mercury: 100 percent to offgas (CEES 2010)
- Nitrate: 100 percent to offgas (BNI 2002)
- All hazardous chemicals (organics): 100 percent to offgas

The 11 Tank Closure alternatives developed for this *TC & WM EIS* are differentiated based on waste retrieval, waste treatment, and waste-form characteristics, as described in Table D-34. The retrieval efficiencies considered vary from 90 to 99.9 percent of waste volume. Treatment options considered include the following:

- Retrieval and treatment of transuranic (TRU) waste constituents from selected tanks
- Solid-liquid separations designed to direct long-lived radionuclides to the immobilized high-level radioactive waste (IHLW) stream
- Ion exchange removal of technetium-99 to remove a mobile constituent from the low-activity waste (LAW) stream
- Recovery of iodine-129 from melter offgas to control releases to the atmosphere
- Distribution of recovered activity among waste forms, including IHLW glass and LAW forms, immobilized low-activity waste (ILAW) glass, bulk vitrification glass, cast stone waste, steam reforming waste, sulfate grout, and secondary (iodine) grout
- Treatment of the cesium-137 and strontium-90 capsules



**Table D–34. Tank Closure Alternatives – Summary of Conditions**

Tank Closure Alternative	Retrieval Efficiency (percent)	Supplemental Treatment 200-East Area			Primary LAW Form		TRU Waste Treatment
		Cesium Removal	Technetium-99 Removal	Sulfate Removal	200-East Area	200-West Area	
1	0	(a)	(a)	(a)	(a)	(a)	(a)
2A	99	Yes	No	No	ILAW glass	(b)	No
2B	99	Yes	Yes	No	ILAW glass	(b)	No
3A	99	Yes	No	No	ILAW glass; BV glass	BV glass	Yes
3B	99	Yes	Yes	No	ILAW glass; cast stone waste	Cast stone waste	Yes
3C	99	Yes	No	No	ILAW glass; steam reforming waste	Steam reforming waste	Yes
4	99.9	Yes	No	No	ILAW glass; cast stone waste	BV glass	Yes
5	90	Yes	No	Yes	ILAW glass; cast stone waste	BV glass	Yes
6A	99.9	Yes	No	No	Not applicable	(c)	No
6B	99.9	Yes	No	No	ILAW glass <sup>d</sup>	(c)	No
6C	99	Yes	No	No	ILAW glass <sup>d</sup>	(c)	No

<sup>a</sup> Not applicable; no retrieval or processing under Alternative 1.

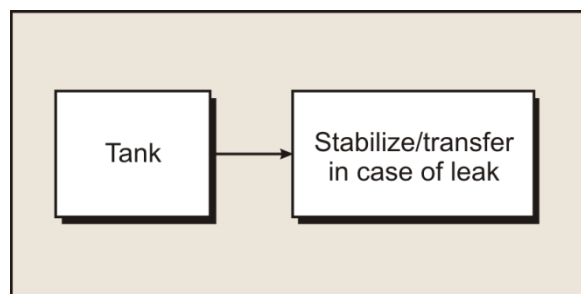
<sup>b</sup> Not applicable; no treatment in the 200-West Area under Alternative 2A or 2B.

<sup>c</sup> Not applicable; no treatment in the 200-West Area under Alternative 6A, 6B, or 6C.

<sup>d</sup> ILAW glass would be managed and disposed of as immobilized high-level radioactive waste glass under Alternatives 6B and 6C.

**Key:** BV=bulk vitrification; ILAW=immobilized low-activity waste; LAW=low-activity waste; TRU=transuranic.

An additional differentiating characteristic is location of the waste processing facilities. Under Tank Closure Alternative 1: No Action, retrieval and processing would not occur, and the waste would be managed in place as required for safety and protection of the environment. This arrangement is represented in Figure D–1. Material balances under Tank Closure Alternative 1 are presented in Tables D–35 and D–36. The BBI estimate, summarized in Tables D–4 through D–7, constitutes the inventories under this alternative.



**Figure D–1. Tank Closure Alternative 1 Flowsheet**

Under Tank Closure Alternatives 2A, 2B, 6A, 6B, and 6C, all processing would occur in the 200-East Area. Under the remaining Tank Closure alternatives, processing would occur in both the 200-East and 200-West Areas. Under all alternatives other than Tank Closure Alternative 1, the initial processing step in both the 200-East and 200-West Areas would be solid-liquid separations, with recovered solids vitrified as IHLW glass. Subsequent processing steps and related mass balances under each Tank Closure alternative are described in the following paragraphs.

**Table D-35. Tank Closure Alternative 1 Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste	4.82×10 <sup>1</sup>	100.0	4.58×10 <sup>7</sup>	100.0	3.12×10 <sup>3</sup>	100.0	1.21×10 <sup>4</sup>	100.0	9.38×10 <sup>2</sup>	100.0	1.41×10 <sup>2</sup>	100.0	8.14×10 <sup>4</sup>	100.0	5.05×10 <sup>7</sup>	100.0	2.97×10 <sup>4</sup>	100.0
<b>Total</b>	<b>4.82×10<sup>1</sup></b>	<b>100.0</b>	<b>4.58×10<sup>7</sup></b>	<b>100.0</b>	<b>3.12×10<sup>3</sup></b>	<b>100.0</b>	<b>1.21×10<sup>4</sup></b>	<b>100.0</b>	<b>9.38×10<sup>2</sup></b>	<b>100.0</b>	<b>1.41×10<sup>2</sup></b>	<b>100.0</b>	<b>8.14×10<sup>4</sup></b>	<b>100.0</b>	<b>5.05×10<sup>7</sup></b>	<b>100.0</b>	<b>2.97×10<sup>4</sup></b>	<b>100.0</b>
<b>Other Inventory</b>																		
Cesium and strontium capsules <sup>b</sup>	0	N/A	4.63×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	2.04×10 <sup>7</sup>	N/A	0	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a). BBI percentages are rounded to the nearest tenth.

<sup>b</sup> Cesium and strontium capsules would remain in storage at the Waste Encapsulation and Storage Facility.

**Key:** %=percent; BBI=Best-Basis Inventory; N/A=not applicable.

**Source:** SAIC 2011.

**Table D-36. Tank Closure Alternative 1 Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste	5.98×10 <sup>5</sup>	100.0	1.83×10 <sup>3</sup>	100.0	7.08×10 <sup>7</sup>	100.0	8.41×10 <sup>4</sup>	100.0	5.97×10 <sup>5</sup>	100.0	2.95×10 <sup>4</sup>	100.0	2.40	100.0	3.45×10 <sup>6</sup>	100.0	1.68×10 <sup>3</sup>	100.0	1.11	100.0
<b>Total</b>	<b>5.98×10<sup>5</sup></b>	<b>100.0</b>	<b>1.83×10<sup>3</sup></b>	<b>100.0</b>	<b>7.08×10<sup>7</sup></b>	<b>100.0</b>	<b>8.41×10<sup>4</sup></b>	<b>100.0</b>	<b>5.97×10<sup>5</sup></b>	<b>100.0</b>	<b>2.95×10<sup>4</sup></b>	<b>100.0</b>	<b>2.40</b>	<b>100.0</b>	<b>3.45×10<sup>6</sup></b>	<b>100.0</b>	<b>1.68×10<sup>3</sup></b>	<b>100.0</b>	<b>1.11</b>	<b>100.0</b>

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a). BBI percentages are rounded to the nearest tenth.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; Kg=kilograms; N/A=not applicable; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

**Source:** SAIC 2011.

The waste forms of the long-lived, mobile radionuclides, technetium-99 and iodine-129, are of interest in regard to long-term performance assessment. Both elements exist as water-soluble species and move through process operations in the liquid phase. To facilitate evaluation of the relative efficiency of retention of these two radionuclides in the LAW forms, separation of technetium-99 from the 200-East Area liquid stream and immobilization into IHLW glass was considered under Tank Closure Alternatives 2B and 3B. Under Tank Closure Alternative 2B, with technetium-99 removal in the WTP, approximately 98 percent of the BBI estimate for technetium-99 would be solidified in IHLW glass; under Tank Closure Alternative 3B, approximately 66 percent of the BBI estimate for technetium-99 would be solidified in IHLW glass. Under this latter alternative, approximately 32 percent of the BBI estimate for technetium-99 would be contained in the 200-East and 200-West Area cast stone waste. Without technetium-99 removal, under Tank Closure Alternatives 3A and 3C, approximately 28 percent of the BBI estimate for technetium-99 would be solidified in ILAW glass, and approximately 70 percent of the BBI estimate for technetium-99 would be solidified in the bulk vitrification glass or steam reforming waste. The remaining 2 percent would be encapsulated in a (secondary waste) grout.

The WTP Pretreatment Facility was originally designed to remove technetium and blend the technetium removed from the LAW vitrification feed with HLW solids for feed to HLW vitrification. However, based on reviews of technetium-99 in ILAW glass, DOE and Ecology agreed to delete technetium removal from the WTP permit (Hedges 2008). With this modification, technetium-99 would not be separated from the pretreated LAW feed and combined with the HLW solids for vitrification processing into IHLW glass. Thus, the technetium-99 content of the resulting IHLW glass would decrease, while the technetium-99 concentration in the ILAW glass would increase.

Various alternatives in this *TC & WM EIS* examine the impacts of removing the technetium-99 in the WTP. Table D-34 indicates whether technetium-99 removal would occur under the various alternatives. If technetium-99 is not removed in WTP pretreatment, most of it would be immobilized in ILAW glass. If technetium-99 removal occurs during WTP pretreatment, most of the technetium-99 would be immobilized in IHLW glass. See Appendix E, Section E.1.2.3.10, for further details.

The distribution of the radionuclides can vary based on how the waste is treated and on the types of waste produced under each Tank Closure alternative. The partitioning of iodine among the waste forms is affected by whether the processing is thermal or nonthermal. In nonthermal processing, iodine would remain in the cast stone waste. Thermal processing in the WTP HLW and LAW melters or in the bulk vitrification and steam reforming processes would leave a portion of the iodine in the feed stream, where it would be volatilized and recovered from the offgas for disposal in a secondary grout. Thus, for thermal processing, it was assumed that approximately 20 percent of the feed iodine would be solidified in ILAW glass, bulk vitrification glass, and steam reforming waste, and approximately 80 percent would be encapsulated in a (secondary waste) grout (CEES 2010). Distribution of technetium-99 and iodine-129 among the waste forms under each alternative is described in detail in the activity balance tables presented in the following text and in Appendix E, Sections E.1.2.3.6 and E.1.2.3.8.

Tank Closure Alternatives 2A and 2B both involve processing waste in the WTP to form IHLW glass and ILAW glass. No supplemental technology would be utilized to treat the LAW portion of the waste. Tank Closure Alternative 2A does not include technetium-99 removal; therefore, the bulk of the technetium-99 would be immobilized in the ILAW glass. Tank Closure Alternative 2B includes technetium-99 removal from the LAW stream, so the majority of the technetium-99 inventory, approximately 97.7 percent of the BBI estimate for technetium-99, would be immobilized in IHLW glass. Under both Tank Closure Alternatives 2A and 2B, it was estimated that approximately 20 percent of the retrieved iodine would be solidified in ILAW glass, while the remaining 80 percent would be encapsulated in grout (secondary waste). Appendix N, Section N.3.8, provides a sensitivity analysis of additional retention of iodine-129 in ILAW glass. Flowsheet schematics for Tank Closure Alternatives 2A and 2B are presented in

Figures D-2 and D-3, respectively. Material balances under Tank Closure Alternatives 2A and 2B are presented in Tables D-37 through D-40.

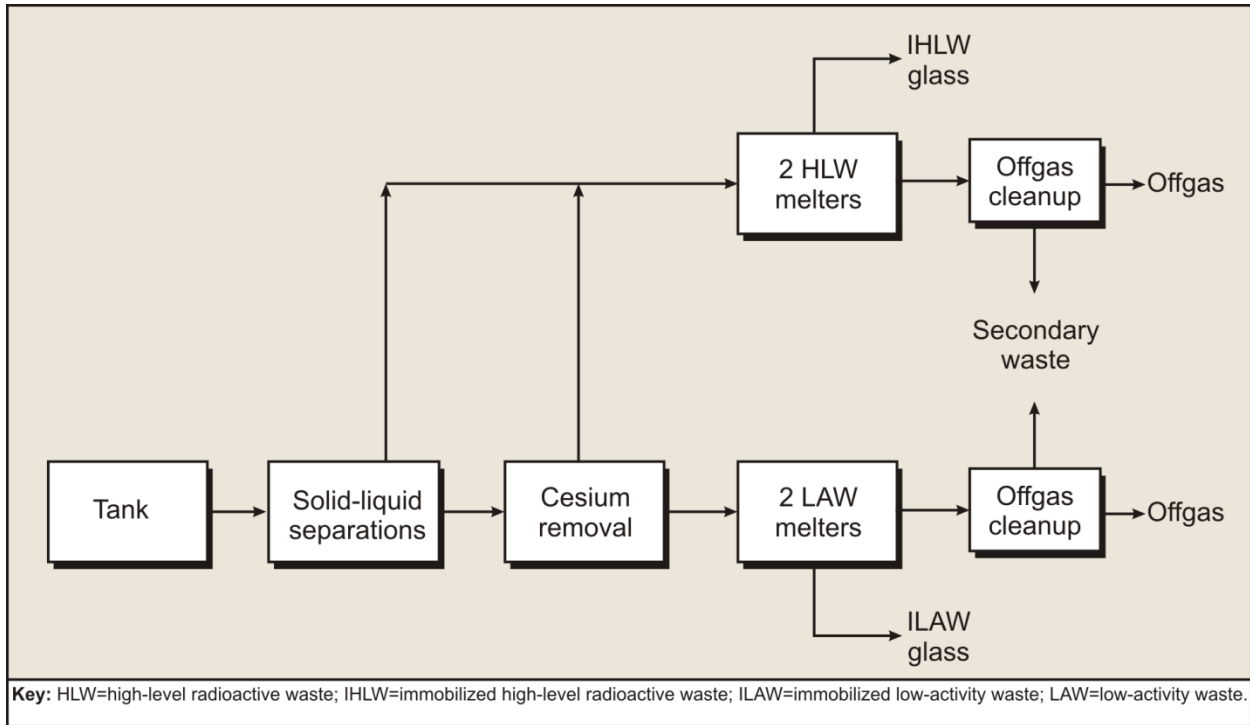


Figure D-2. Tank Closure Alternative 2A Flowsheet

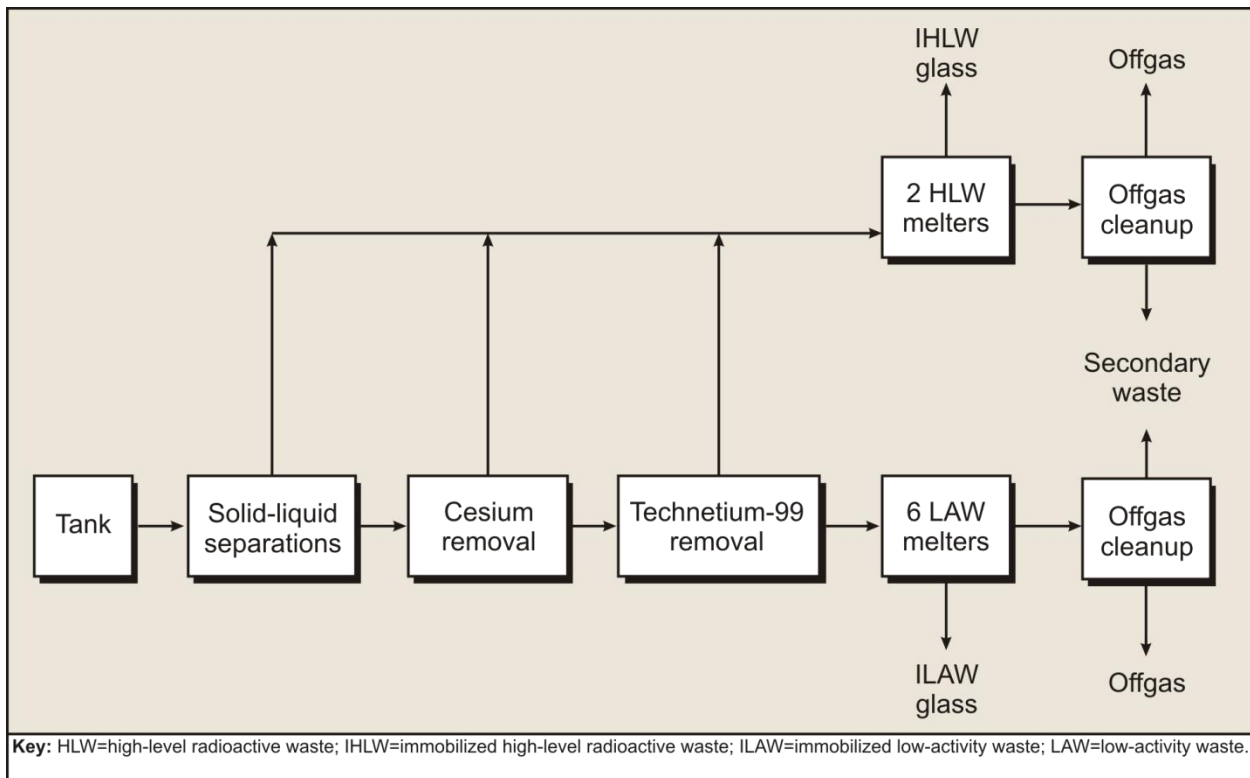


Figure D-3. Tank Closure Alternative 2B Flowsheet

**Table D-37. Tank Closure Alternative 2A Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>1</sup>	1.0	4.58×10 <sup>5</sup>	1.0	3.12×10 <sup>1</sup>	1.0	1.21×10 <sup>2</sup>	1.0	9.38	1.0	1.41	1.0	8.14×10 <sup>2</sup>	1.0	5.05×10 <sup>5</sup>	1.0	2.97×10 <sup>2</sup>	1.0
IHLW glass <sup>c</sup>	6.99×10 <sup>-3</sup>	0.0	4.49×10 <sup>7</sup>	97.9	0	0.0	0	0.0	8.73×10 <sup>2</sup>	93.1	1.40×10 <sup>2</sup>	99.0	8.06×10 <sup>4</sup>	99.0	4.93×10 <sup>7</sup>	97.6	2.47×10 <sup>2</sup>	0.8
ILAW glass and retired LAW melters	9.56	19.8	4.45×10 <sup>5</sup>	1.0	0	0.0	0	0.0	5.47×10 <sup>1</sup>	5.8	8.35×10 <sup>-3</sup>	0.0	1.45	0.0	2.30×10 <sup>3</sup>	0.0	2.88×10 <sup>4</sup>	96.9
ETF-generated solid secondary waste <sup>d</sup>	3.36×10 <sup>1</sup>	69.7	4.59×10 <sup>-1</sup>	0.0	8.51	0.3	0	0.0	4.03×10 <sup>-2</sup>	0.0	5.11×10 <sup>-2</sup>	0.0	6.90×10 <sup>-4</sup>	0.0	6.42	0.0	8.63×10 <sup>1</sup>	0.3
Solid secondary waste <sup>e</sup>	4.65	9.7	1.95×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.64	0.4	2.83×10 <sup>-1</sup>	0.2	1.98×10 <sup>2</sup>	0.2	7.76×10 <sup>5</sup>	1.5	4.31×10 <sup>2</sup>	1.5
<b>Total<sup>f</sup></b>	<b>4.83×10<sup>1</sup></b>	<b>100.2</b>	<b>4.60×10<sup>7</sup></b>	<b>100.2</b>	<b>3.97×10<sup>1</sup></b>	<b>1.3</b>	<b>1.21×10<sup>2</sup></b>	<b>1.0</b>	<b>9.41×10<sup>2</sup></b>	<b>100.3</b>	<b>1.41×10<sup>2</sup></b>	<b>100.2</b>	<b>8.16×10<sup>4</sup></b>	<b>100.2</b>	<b>5.06×10<sup>7</sup></b>	<b>100.1</b>	<b>2.99×10<sup>4</sup></b>	<b>100.5</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>g</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>h</sup>	4.78×10 <sup>1</sup>	N/A	4.69×10 <sup>4</sup>	N/A	3.10×10 <sup>3</sup>	N/A	1.20×10 <sup>4</sup>	N/A	4.65×10 <sup>-1</sup>	N/A	NR	N/A	4.04×10 <sup>1</sup>	N/A	3.55×10 <sup>4</sup>	N/A	1.47×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste.

<sup>f</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent; a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>g</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>h</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported.

**Source:** SAIC 2011.

**Table D-38. Tank Closure Alternative 2A Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>3</sup>	1.0	1.83×10 <sup>1</sup>	1.0	7.08×10 <sup>5</sup>	1.0	8.41×10 <sup>2</sup>	1.0	5.97×10 <sup>3</sup>	1.0	2.95×10 <sup>2</sup>	1.0	2.40×10 <sup>-2</sup>	1.0	3.45×10 <sup>4</sup>	1.0	1.68×10 <sup>1</sup>	1.0	1.11×10 <sup>-2</sup>	1.0
IHLW glass <sup>c</sup>	1.36×10 <sup>5</sup>	22.7	0	0.0	0	0.0	7.45×10 <sup>4</sup>	88.6	5.52×10 <sup>5</sup>	92.5	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters	4.56×10 <sup>5</sup>	76.2	0	0.0	0	0.0	8.88×10 <sup>3</sup>	10.6	3.74×10 <sup>4</sup>	6.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>d</sup>	4.43×10 <sup>1</sup>	0.0	5.55	0.3	9.01×10 <sup>6</sup>	12.7	4.58	0.0	4.00×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>e</sup>	1.94×10 <sup>3</sup>	0.3	1.76×10 <sup>3</sup>	96.4	0	0.0	2.47×10 <sup>2</sup>	0.3	2.32×10 <sup>3</sup>	0.4	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Total<sup>f</sup></b>	<b>6.00×10<sup>5</sup></b>	<b>100.3</b>	<b>1.78×10<sup>3</sup></b>	<b>97.7</b>	<b>9.72×10<sup>6</sup></b>	<b>13.7</b>	<b>8.45×10<sup>4</sup></b>	<b>100.5</b>	<b>5.98×10<sup>5</sup></b>	<b>100.2</b>	<b>2.95×10<sup>2</sup></b>	<b>1.0</b>	<b>2.40×10<sup>-2</sup></b>	<b>1.0</b>	<b>3.45×10<sup>4</sup></b>	<b>1.0</b>	<b>1.68×10<sup>1</sup></b>	<b>1.0</b>	<b>1.11×10<sup>-2</sup></b>	<b>1.0</b>
<b>Other Inventory<sup>g</sup></b>																				
Treatment air emissions <sup>h</sup>	NR	N/A	1.81×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	7.40×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> Includes retired HLW melter inventory. IHLW glass would be stored on site until disposition decisions are made and implemented.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams that would be treated at the ETF.

<sup>f</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during the thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>g</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>h</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

**Source:** SAIC 2011.

**Table D-39. Tank Closure Alternative 2B Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-1</sup>	1.0	4.58×10 <sup>5</sup>	1.0	3.12×10 <sup>1</sup>	1.0	1.21×10 <sup>2</sup>	1.0	9.38	1.0	1.41	1.0	8.14×10 <sup>2</sup>	1.0	5.05×10 <sup>5</sup>	1.0	2.97×10 <sup>2</sup>	1.0
IHLW glass <sup>c</sup>	6.99×10 <sup>-3</sup>	0.0	4.49×10 <sup>7</sup>	97.9	0	0.0	0	0.0	8.73×10 <sup>2</sup>	93.1	1.40×10 <sup>2</sup>	99.0	8.06×10 <sup>4</sup>	99.0	4.93×10 <sup>7</sup>	97.6	2.90×10 <sup>4</sup>	97.7
ILAW glass and retired LAW melters	9.56	19.8	4.45×10 <sup>5</sup>	1.0	0	0.0	0	0.0	5.47×10 <sup>1</sup>	5.8	8.35×10 <sup>-3</sup>	0.0	1.45	0.0	2.30×10 <sup>3</sup>	0.0	2.88×10 <sup>2</sup>	1.0
ETF-generated solid secondary waste <sup>d</sup>	3.36×10 <sup>1</sup>	69.7	4.59×10 <sup>-1</sup>	0.0	8.51	0.3	0	0.0	4.03×10 <sup>-2</sup>	0.0	5.11×10 <sup>-2</sup>	0.0	6.90×10 <sup>-4</sup>	0.0	6.42	0.0	8.63×10 <sup>1</sup>	0.3
Solid secondary waste <sup>e</sup>	4.65	9.7	1.95×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.64	0.4	2.83×10 <sup>-1</sup>	0.2	1.98×10 <sup>2</sup>	0.2	7.76×10 <sup>5</sup>	1.5	4.92×10 <sup>2</sup>	1.7
<b>Total<sup>f</sup></b>	<b>4.83×10<sup>1</sup></b>	<b>100.2</b>	<b>4.60×10<sup>7</sup></b>	<b>100.2</b>	<b>3.97×10<sup>1</sup></b>	<b>1.3</b>	<b>1.21×10<sup>2</sup></b>	<b>1.0</b>	<b>9.41×10<sup>2</sup></b>	<b>100.3</b>	<b>1.41×10<sup>2</sup></b>	<b>100.2</b>	<b>8.16×10<sup>4</sup></b>	<b>100.2</b>	<b>5.06×10<sup>7</sup></b>	<b>100.1</b>	<b>3.02×10<sup>4</sup></b>	<b>101.7</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>g</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
Rubble, soil, and equipment <sup>h</sup>	1.67×10 <sup>-2</sup>	N/A	1.31×10 <sup>4</sup>	N/A	1.47	N/A	6.03	N/A	4.82×10 <sup>-1</sup>	N/A	3.24×10 <sup>-2</sup>	N/A	4.32×10 <sup>1</sup>	N/A	3.05×10 <sup>4</sup>	N/A	9.72	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>i</sup>	4.78×10 <sup>1</sup>	N/A	4.69×10 <sup>4</sup>	N/A	3.10×10 <sup>3</sup>	N/A	1.20×10 <sup>4</sup>	N/A	4.65×10 <sup>-1</sup>	N/A	NR	N/A	4.04×10 <sup>1</sup>	N/A	3.55×10 <sup>4</sup>	N/A	1.47×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. The value for technetium-99 includes 6.04×10<sup>1</sup> curies of technetium-99 that would remain in the spent resin from the technetium-99 removal process.

<sup>f</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent; a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>g</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>h</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. This material would be disposed of in the River Protection Project Disposal Facility.

<sup>i</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported.

**Source:** SAIC 2011.

**Table D-40. Alternative 2B Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>3</sup>	1.0	1.83×10 <sup>1</sup>	1.0	7.08×10 <sup>5</sup>	1.0	8.41×10 <sup>2</sup>	1.0	5.97×10 <sup>3</sup>	1.0	2.95×10 <sup>2</sup>	1.0	2.40×10 <sup>-2</sup>	1.0	3.45×10 <sup>4</sup>	1.0	1.68×10 <sup>1</sup>	1.0	1.11×10 <sup>-2</sup>	1.0
IHLW glass <sup>c</sup>	1.36×10 <sup>5</sup>	22.7	0	0.0	0	0.0	7.45×10 <sup>4</sup>	88.6	5.52×10 <sup>5</sup>	92.5	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters	4.56×10 <sup>5</sup>	76.2	0	0.0	0	0.0	8.88×10 <sup>3</sup>	10.6	3.74×10 <sup>4</sup>	6.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>d</sup>	4.43×10 <sup>1</sup>	0.0	5.55	0.3	9.01×10 <sup>6</sup>	12.7	4.58	0.0	4.00×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>e</sup>	1.94×10 <sup>3</sup>	0.3	1.76×10 <sup>3</sup>	96.4	0	0.0	2.47×10 <sup>2</sup>	0.3	2.32×10 <sup>3</sup>	0.4	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Total<sup>f</sup></b>	<b>6.00×10<sup>5</sup></b>	<b>100.3</b>	<b>1.78×10<sup>3</sup></b>	<b>97.7</b>	<b>9.72×10<sup>6</sup></b>	<b>13.7</b>	<b>8.45×10<sup>4</sup></b>	<b>100.5</b>	<b>5.98×10<sup>5</sup></b>	<b>100.2</b>	<b>2.95×10<sup>2</sup></b>	<b>1.0</b>	<b>2.40×10<sup>-2</sup></b>	<b>1.0</b>	<b>3.45×10<sup>4</sup></b>	<b>1.0</b>	<b>1.68×10<sup>1</sup></b>	<b>1.0</b>	<b>1.11×10<sup>-2</sup></b>	<b>1.0</b>
<b>Other Inventory<sup>g</sup></b>																				
Rubble, soil, and equipment <sup>h</sup>	5.86×10 <sup>2</sup>	N/A	2.22	N/A	3.93×10 <sup>4</sup>	N/A	3.34×10 <sup>1</sup>	N/A	6.60×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>i</sup>	NR	N/A	1.81×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	6.14×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> Includes retired HLW melter inventory. IHLW glass would be stored on site until disposition decisions are made and implemented.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations. These waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams that would be treated at the ETF.

<sup>f</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during the thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>g</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>h</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. This material would be disposed of in the River Protection Project Disposal Facility.

<sup>i</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

**Source:** SAIC 2011.



Tank Closure Alternatives 3A, 3B, and 3C involve processing waste to produce IHLW glass and ILAW glass, but they differ in that Alternative 3A would produce a supplemental bulk vitrification glass, Alternative 3B would produce a supplemental cast stone waste form, and Alternative 3C would produce a supplemental steam reforming waste form from a portion of the LAW stream. Technetium-99 would be immobilized in the ILAW and the bulk vitrification glass or steam reforming waste under Tank Closure Alternatives 3A and 3C, respectively; approximately 66 percent of the estimated BBI for technetium-99 would be immobilized in the IHLW under Alternative 3B using a technetium-99 removal process in the WTP. Under Tank Closure Alternatives 3A, 3B, and 3C, the ILAW glass would contain 5.8 percent of the estimated BBI for iodine-129, and the bulk vitrification glass, cast stone waste, and steam reforming waste would contain 14.0 percent, 70.1 percent, and 14.0 percent, respectively, of the estimated BBI for iodine-129. Flowsheet schematics for Tank Closure Alternatives 3A, 3B, and 3C are presented as Figures D-4, D-5, and D-6, respectively. Material balances under Alternatives 3A, 3B, and 3C are presented in Tables D-41 through D-46.

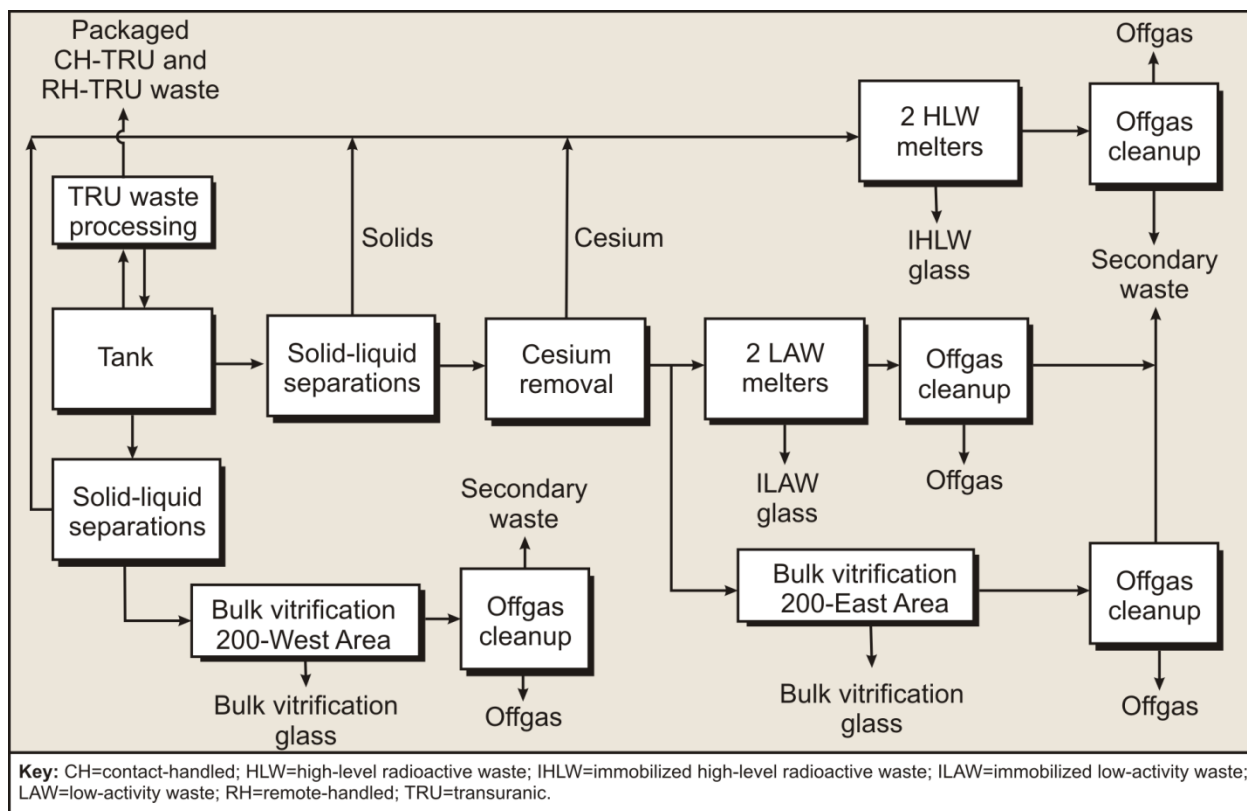


Figure D-4. Tank Closure Alternative 3A Flowsheet

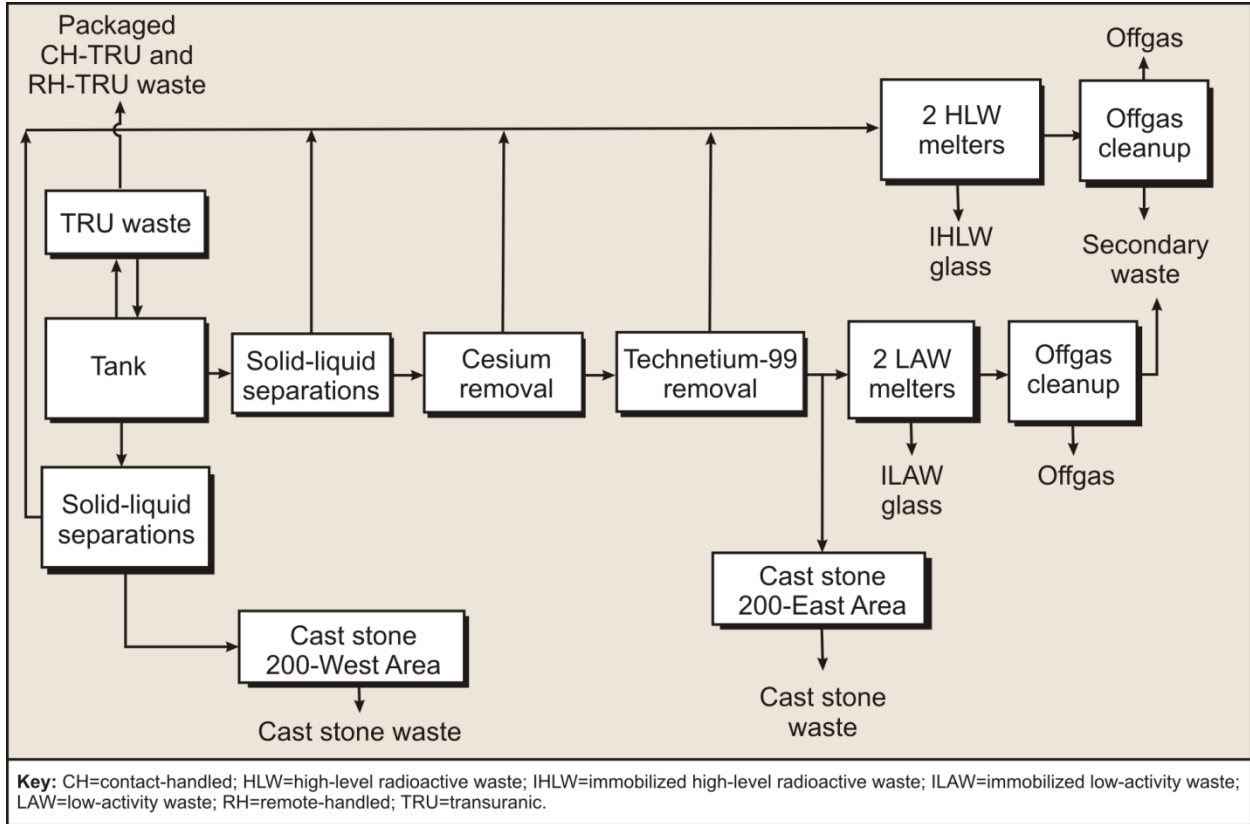


Figure D-5. Tank Closure Alternative 3B Flowsheet

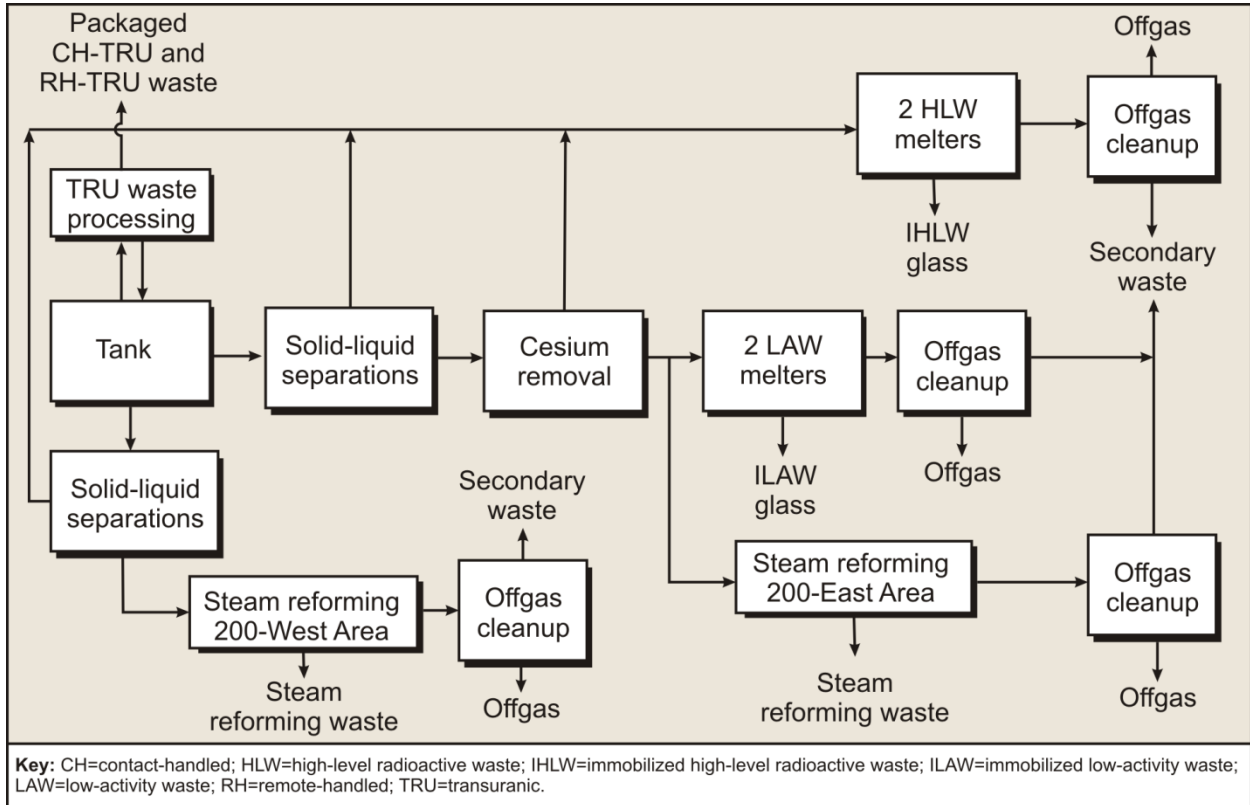


Figure D-6. Tank Closure Alternative 3C Flowsheet

**Table D-41. Tank Closure Alternative 3A Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-1</sup>	1.0	4.58×10 <sup>5</sup>	1.0	3.12×10 <sup>1</sup>	1.0	1.21×10 <sup>2</sup>	1.0	9.38	1.0	1.41	1.0	8.14×10 <sup>2</sup>	1.0	5.05×10 <sup>5</sup>	1.0	2.97×10 <sup>2</sup>	1.0
IHLW glass <sup>c</sup>	1.32×10 <sup>-5</sup>	0.0	4.04×10 <sup>7</sup>	88.1	0	0.0	0	0.0	8.32×10 <sup>2</sup>	88.6	1.38×10 <sup>2</sup>	97.8	7.31×10 <sup>4</sup>	89.8	4.87×10 <sup>7</sup>	96.4	1.49×10 <sup>2</sup>	0.5
ILAW glass and retired LAW melters	2.80	5.8	1.73×10 <sup>5</sup>	0.4	0	0.0	0	0.0	1.63×10 <sup>1</sup>	1.7	2.48×10 <sup>-3</sup>	0.0	4.33×10 <sup>-1</sup>	0.0	6.91×10 <sup>2</sup>	0.0	8.44×10 <sup>3</sup>	28.4
ETF-generated solid secondary waste <sup>d</sup>	3.69×10 <sup>1</sup>	76.5	1.43×10 <sup>1</sup>	0.0	4.74	0.2	0	0.0	8.72×10 <sup>-2</sup>	0.0	5.15×10 <sup>-2</sup>	0.0	9.36×10 <sup>-4</sup>	0.0	5.45×10 <sup>1</sup>	0.0	4.63×10 <sup>1</sup>	0.2
Solid secondary waste <sup>e</sup>	1.36	2.8	1.75×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.33	0.4	2.79×10 <sup>-1</sup>	0.2	1.80×10 <sup>2</sup>	0.2	7.67×10 <sup>5</sup>	1.5	1.28×10 <sup>2</sup>	0.4
200-East Area BV glass <sup>f</sup>	3.67	7.6	2.27×10 <sup>5</sup>	0.5	0	0.0	0	0.0	2.13×10 <sup>1</sup>	2.3	3.25×10 <sup>-3</sup>	0.0	5.68×10 <sup>-1</sup>	0.0	9.10×10 <sup>2</sup>	0.0	1.12×10 <sup>4</sup>	37.8
200-West Area BV glass <sup>f</sup>	3.08	6.4	4.39×10 <sup>6</sup>	9.6	0	0.0	0	0.0	3.01×10 <sup>1</sup>	3.2	9.58×10 <sup>-1</sup>	0.7	1.04×10 <sup>3</sup>	1.3	4.61×10 <sup>5</sup>	0.9	9.42×10 <sup>3</sup>	31.7
Transuranic waste <sup>g</sup>	5.02×10 <sup>-2</sup>	0.1	3.41×10 <sup>5</sup>	0.7	3.85	0.1	3.33	0.0	4.67×10 <sup>1</sup>	5.0	9.79×10 <sup>-1</sup>	0.7	7.39×10 <sup>3</sup>	9.1	7.52×10 <sup>5</sup>	1.5	3.36×10 <sup>2</sup>	1.1
<b>Total<sup>h</sup></b>	<b>4.83×10<sup>1</sup></b>	<b>100.3</b>	<b>4.62×10<sup>7</sup></b>	<b>100.7</b>	<b>3.98×10<sup>1</sup></b>	<b>1.3</b>	<b>1.24×10<sup>2</sup></b>	<b>1.0</b>	<b>9.59×10<sup>2</sup></b>	<b>102.2</b>	<b>1.42×10<sup>2</sup></b>	<b>100.4</b>	<b>8.26×10<sup>4</sup></b>	<b>101.4</b>	<b>5.12×10<sup>7</sup></b>	<b>101.3</b>	<b>3.00×10<sup>4</sup></b>	<b>101.1</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>i</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
Rubble, soil, and equipment <sup>j</sup>	1.67×10 <sup>-2</sup>	N/A	1.31×10 <sup>4</sup>	N/A	1.47	N/A	6.03	N/A	4.82×10 <sup>-1</sup>	N/A	3.24×10 <sup>-2</sup>	N/A	4.32×10 <sup>1</sup>	N/A	3.05×10 <sup>4</sup>	N/A	9.72	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>k</sup>	4.78×10 <sup>1</sup>	N/A	4.69×10 <sup>4</sup>	N/A	3.10×10 <sup>3</sup>	N/A	1.20×10 <sup>4</sup>	N/A	4.52×10 <sup>-1</sup>	N/A	NR	N/A	3.73×10 <sup>1</sup>	N/A	3.55×10 <sup>4</sup>	N/A	1.47×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste.

<sup>f</sup> Includes technetium-99 inventory that resides in the BV waste container insulating material or waste container.

<sup>g</sup> Tank transuranic waste would be disposed of in the Waste Isolation Pilot Plant.

<sup>h</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent; a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>i</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>j</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. This material would be disposed of in the River Protection Project Disposal Facility.

<sup>k</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; BV=bulk vitrification; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported.

**Source:** SAIC 2011.

**Table D-42. Tank Closure Alternative 3A Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>3</sup>	1.0	1.83×10 <sup>1</sup>	1.0	7.08×10 <sup>5</sup>	1.0	8.41×10 <sup>2</sup>	1.0	5.97×10 <sup>3</sup>	1.0	2.95×10 <sup>2</sup>	1.0	2.40×10 <sup>-2</sup>	1.0	3.45×10 <sup>4</sup>	1.0	1.68×10 <sup>1</sup>	1.0	1.11×10 <sup>-2</sup>	1.0
IHLW glass <sup>c</sup>	1.12×10 <sup>5</sup>	18.6	0	0.0	0	0.0	7.07×10 <sup>4</sup>	84.1	5.11×10 <sup>5</sup>	85.5	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters	1.34×10 <sup>5</sup>	22.4	0	0.0	0	0.0	2.67×10 <sup>3</sup>	3.2	1.13×10 <sup>4</sup>	1.9	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>d</sup>	2.76×10 <sup>1</sup>	0.0	5.54	0.3	8.14×10 <sup>6</sup>	11.5	7.56×10 <sup>2</sup>	0.9	7.00×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>e</sup>	8.03×10 <sup>2</sup>	0.1	1.75×10 <sup>3</sup>	96.0	0	0.0	2.17×10 <sup>2</sup>	0.3	2.05×10 <sup>3</sup>	0.3	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
200-East Area BV glass	1.74×10 <sup>5</sup>	29.2	0	0	0	0.0	3.14×10 <sup>3</sup>	3.7	1.47×10 <sup>4</sup>	2.5	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
200-West Area BV glass	1.48×10 <sup>5</sup>	24.7	0	0	0	0.0	3.56×10 <sup>3</sup>	4.2	1.98×10 <sup>4</sup>	3.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
Transuranic waste <sup>f</sup>	2.83×10 <sup>4</sup>	4.7	3.46×10 <sup>1</sup>	1.9	1.07×10 <sup>6</sup>	1.5	6.43×10 <sup>3</sup>	7.7	5.92×10 <sup>4</sup>	9.9	5.93×10 <sup>2</sup>	2.0	4.83×10 <sup>-2</sup>	2.0	6.95×10 <sup>4</sup>	2.0	3.51×10 <sup>1</sup>	2.1	2.23×10 <sup>-2</sup>	2.0
<b>Total<sup>g</sup></b>	<b>6.03×10<sup>5</sup></b>	<b>100.7</b>	<b>1.81×10<sup>3</sup></b>	<b>99.2</b>	<b>9.92×10<sup>6</sup></b>	<b>14.0</b>	<b>8.83×10<sup>4</sup></b>	<b>105.0</b>	<b>6.24×10<sup>5</sup></b>	<b>104.5</b>	<b>8.88×10<sup>2</sup></b>	<b>3.0</b>	<b>7.24×10<sup>-2</sup></b>	<b>3.0</b>	<b>1.04×10<sup>5</sup></b>	<b>3.0</b>	<b>5.19×10<sup>1</sup></b>	<b>3.1</b>	<b>3.34×10<sup>-2</sup></b>	<b>3.0</b>
<b>Other Inventory<sup>h</sup></b>																				
Rubble, soil, and equipment <sup>i</sup>	5.86×10 <sup>2</sup>	N/A	2.22	N/A	3.93×10 <sup>4</sup>	N/A	3.34×10 <sup>1</sup>	N/A	6.60×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>j</sup>	NR	N/A	1.80×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	2.99×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>e</sup> Includes solid LLW and MLLW streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams that would be treated at the ETF.

<sup>f</sup> Tank transuranic waste would be disposed of in the Waste Isolation Pilot Plant.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during the thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>h</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>i</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; BV=bulk vitrification; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; LLW=low-level radioactive waste; MLLW=mixed low-level radioactive waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

**Source:** SAIC 2011.

**Table D-43. Tank Closure Alternative 3B Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-1</sup>	1.0	4.58×10 <sup>5</sup>	1.0	3.12×10 <sup>1</sup>	1.0	1.21×10 <sup>2</sup>	1.0	9.38	1.0	1.41	1.0	8.14×10 <sup>2</sup>	1.0	5.05×10 <sup>5</sup>	1.0	2.97×10 <sup>2</sup>	1.0
IHLW glass <sup>c</sup>	1.32×10 <sup>-5</sup>	0.0	4.04×10 <sup>7</sup>	88.1	0	0.0	0	0.0	8.32×10 <sup>2</sup>	88.6	1.38×10 <sup>2</sup>	97.8	7.31×10 <sup>4</sup>	89.8	4.87×10 <sup>7</sup>	96.4	1.97×10 <sup>4</sup>	66.2
ILAW glass and retired LAW melters	2.80	5.8	1.73×10 <sup>5</sup>	0.4	0	0.0	0	0.0	1.63×10 <sup>1</sup>	1.7	2.48×10 <sup>-3</sup>	0.0	4.33×10 <sup>-1</sup>	0.0	6.91×10 <sup>2</sup>	0.0	8.44×10 <sup>1</sup>	0.3
ETF-generated solid secondary waste <sup>d</sup>	9.85	20.5	4.11×10 <sup>-1</sup>	0.0	2.62	0.1	0	0.0	3.65×10 <sup>-2</sup>	0.0	5.05×10 <sup>-2</sup>	0.0	6.32×10 <sup>-4</sup>	0.0	6.34	0.0	5.82×10 <sup>1</sup>	0.2
Solid secondary waste <sup>e</sup>	1.36	2.8	1.75×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.33	0.4	2.79×10 <sup>-1</sup>	0.2	1.80×10 <sup>2</sup>	0.2	7.67×10 <sup>5</sup>	1.5	3.33×10 <sup>2</sup>	1.1
200-East Area cast stone waste	1.84×10 <sup>1</sup>	38.1	2.28×10 <sup>5</sup>	0.5	1.17×10 <sup>3</sup>	37.3	4.59×10 <sup>3</sup>	38.1	2.13×10 <sup>1</sup>	2.3	3.25×10 <sup>-3</sup>	0.0	5.68×10 <sup>-1</sup>	0.0	9.10×10 <sup>2</sup>	0.0	1.12×10 <sup>2</sup>	0.4
200-West Area cast stone waste	1.54×10 <sup>1</sup>	32.0	4.41×10 <sup>6</sup>	9.6	9.79×10 <sup>2</sup>	31.4	3.85×10 <sup>3</sup>	32.0	3.01×10 <sup>1</sup>	3.2	9.59×10 <sup>-1</sup>	0.7	1.04×10 <sup>3</sup>	1.3	4.61×10 <sup>5</sup>	0.9	9.43×10 <sup>3</sup>	31.7
Transuranic waste <sup>f</sup>	5.02×10 <sup>-2</sup>	0.1	3.41×10 <sup>5</sup>	0.7	3.85	0.1	3.33	0.0	4.67×10 <sup>1</sup>	5.0	9.79×10 <sup>-1</sup>	0.7	7.39×10 <sup>3</sup>	9.1	7.52×10 <sup>5</sup>	1.5	3.36×10 <sup>2</sup>	1.1
<b>Total<sup>g</sup></b>	<b>4.83×10<sup>1</sup></b>	<b>100.3</b>	<b>4.62×10<sup>7</sup></b>	<b>100.7</b>	<b>2.18×10<sup>3</sup></b>	<b>69.9</b>	<b>8.57×10<sup>3</sup></b>	<b>71.1</b>	<b>9.59×10<sup>2</sup></b>	<b>102.2</b>	<b>1.42×10<sup>2</sup></b>	<b>100.4</b>	<b>8.26×10<sup>4</sup></b>	<b>101.4</b>	<b>5.12×10<sup>7</sup></b>	<b>101.3</b>	<b>3.03×10<sup>4</sup></b>	<b>102.0</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>h</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
Rubble, soil, and equipment <sup>i</sup>	1.67×10 <sup>-2</sup>	N/A	1.31×10 <sup>4</sup>	N/A	1.47	N/A	6.03	N/A	4.82×10 <sup>-1</sup>	N/A	3.24×10 <sup>-2</sup>	N/A	4.32×10 <sup>1</sup>	N/A	3.05×10 <sup>4</sup>	N/A	9.72	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>j</sup>	1.40×10 <sup>1</sup>	N/A	4.46×10 <sup>4</sup>	N/A	9.53×10 <sup>2</sup>	N/A	3.50×10 <sup>3</sup>	N/A	4.27×10 <sup>-1</sup>	N/A	NR	N/A	3.67×10 <sup>1</sup>	N/A	3.53×10 <sup>4</sup>	N/A	1.00×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams that would be treated at the ETF, as reported for ETF-generated solid secondary waste in the table above. The value for technetium-99 includes 4.31×10<sup>1</sup> curies of technetium-99 that would remain in the spent resin after the technetium-99 removal process.

<sup>f</sup> Tank transuranic waste would be disposed of in the Waste Isolation Pilot Plant.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent; a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>h</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>i</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported.

**Source:** SAIC 2011.

**Table D-44. Tank Closure Alternative 3B Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>3</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>3</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>3</sup>	1.0	1.83×10 <sup>3</sup>	1.0	7.08×10 <sup>7</sup>	1.0	8.41×10 <sup>4</sup>	1.0	5.97×10 <sup>3</sup>	1.0	2.95×10 <sup>4</sup>	1.0	2.40×10 <sup>-2</sup>	1.0	3.45×10 <sup>4</sup>	1.0	1.68×10 <sup>1</sup>	1.0	1.11×10 <sup>-2</sup>	1.0
IHLW glass <sup>c</sup>	1.12×10 <sup>3</sup>	18.6	0	0.0	0	0.0	7.07×10 <sup>4</sup>	84.1	5.11×10 <sup>3</sup>	85.5	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters	1.34×10 <sup>3</sup>	22.4	0	0.0	0	0.0	2.67×10 <sup>3</sup>	3.2	1.13×10 <sup>4</sup>	1.9	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>d</sup>	1.84×10 <sup>1</sup>	0.0	4.12	0.2	2.63×10 <sup>6</sup>	3.7	5.82	0.0	3.57×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>e</sup>	8.03×10 <sup>2</sup>	0.1	1.31×10 <sup>3</sup>	71.5	0	0.0	2.17×10 <sup>2</sup>	0.3	2.05×10 <sup>3</sup>	0.3	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
200-East Area cast stone waste	1.76×10 <sup>3</sup>	29.4	2.41×10 <sup>2</sup>	13.2	2.69×10 <sup>7</sup>	38.0	3.51×10 <sup>3</sup>	4.2	1.48×10 <sup>4</sup>	2.5	1.11×10 <sup>4</sup>	37.6	9.03×10 <sup>-1</sup>	37.6	1.30×10 <sup>6</sup>	37.6	6.33	0.4	4.16×10 <sup>-1</sup>	37.6
200-West Area cast stone waste	1.49×10 <sup>3</sup>	24.9	2.19×10 <sup>2</sup>	12.0	2.26×10 <sup>7</sup>	31.9	3.98×10 <sup>3</sup>	4.7	1.99×10 <sup>4</sup>	3.3	9.15×10 <sup>3</sup>	31.0	7.45×10 <sup>-1</sup>	31.0	1.07×10 <sup>6</sup>	31.0	5.22×10 <sup>2</sup>	31.0	3.44×10 <sup>-1</sup>	31.0
TRU waste <sup>f</sup>	2.83×10 <sup>4</sup>	4.7	3.46×10 <sup>1</sup>	1.9	1.07×10 <sup>6</sup>	1.5	6.43×10 <sup>3</sup>	7.7	5.92×10 <sup>4</sup>	9.9	5.93×10 <sup>2</sup>	2.0	4.83×10 <sup>-2</sup>	2.0	6.95×10 <sup>4</sup>	2.0	3.51×10 <sup>1</sup>	2.1	2.23×10 <sup>-2</sup>	2.0
<b>Total<sup>g</sup></b>	<b>6.05×10<sup>3</sup></b>	<b>101.1</b>	<b>1.82×10<sup>3</sup></b>	<b>99.8</b>	<b>5.39×10<sup>7</sup></b>	<b>76.1</b>	<b>8.84×10<sup>4</sup></b>	<b>105.1</b>	<b>6.24×10<sup>5</sup></b>	<b>104.5</b>	<b>2.11×10<sup>4</sup></b>	<b>71.6</b>	<b>1.72</b>	<b>71.6</b>	<b>2.47×10<sup>6</sup></b>	<b>71.6</b>	<b>5.81×10<sup>2</sup></b>	<b>34.5</b>	<b>7.93×10<sup>-1</sup></b>	<b>71.6</b>
<b>Other Inventory<sup>h</sup></b>																				
Rubble, soil, and equipment <sup>i</sup>	5.86×10 <sup>3</sup>	N/A	2.22	N/A	3.93×10 <sup>4</sup>	N/A	3.34×10 <sup>1</sup>	N/A	6.60×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>j</sup>	NR	N/A	1.34×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	1.37×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>e</sup> Includes solid LLW and MLLW streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>f</sup> Tank TRU waste would be disposed of in the Waste Isolation Pilot Plant.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during the thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>h</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>i</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; LLW=low-level radioactive waste; MLLW=mixed low-level radioactive waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; TCP=trichlorophenol; TRU=transuranic.

**Source:** SAIC 2011.

**Table D-45. Tank Closure Alternative 3C Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-1</sup>	1.0	4.58×10 <sup>5</sup>	1.0	3.12×10 <sup>1</sup>	1.0	1.21×10 <sup>2</sup>	1.0	9.38	1.0	1.41	1.0	8.14×10 <sup>2</sup>	1.0	5.05×10 <sup>5</sup>	1.0	2.97×10 <sup>2</sup>	1.0
IHLW glass <sup>c</sup>	1.32×10 <sup>-5</sup>	0.0	4.04×10 <sup>7</sup>	88.1	0	0.0	0	0.0	8.32×10 <sup>2</sup>	88.6	1.38×10 <sup>2</sup>	97.8	7.31×10 <sup>4</sup>	89.8	4.87×10 <sup>7</sup>	96.4	1.49×10 <sup>2</sup>	0.5
ILAW glass and retired LAW melters	2.80	5.8	1.73×10 <sup>5</sup>	0.4	0	0.0	0	0.0	1.63×10 <sup>1</sup>	1.7	2.48×10 <sup>-3</sup>	0.0	4.33×10 <sup>-1</sup>	0.0	6.91×10 <sup>2</sup>	0.0	8.44×10 <sup>3</sup>	28.4
ETF-generated solid secondary waste <sup>d</sup>	3.69×10 <sup>1</sup>	76.5	1.44×10 <sup>1</sup>	0.0	4.74	0.2	0	0.0	7.92×10 <sup>-2</sup>	0.0	5.13×10 <sup>-2</sup>	0.0	8.40×10 <sup>-4</sup>	0.0	5.35×10 <sup>1</sup>	0.0	4.63×10 <sup>1</sup>	0.2
Solid secondary waste <sup>e</sup>	1.36	2.8	1.75×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.33	0.4	2.79×10 <sup>-1</sup>	0.2	1.80×10 <sup>2</sup>	0.2	7.67×10 <sup>5</sup>	1.5	1.28×10 <sup>2</sup>	0.4
200-East Area steam reforming waste	3.67	7.6	2.28×10 <sup>5</sup>	0.5	0	0.0	0	0.0	2.13×10 <sup>1</sup>	2.3	3.25×10 <sup>-3</sup>	0.0	5.68×10 <sup>-1</sup>	0.0	9.10×10 <sup>2</sup>	0.0	1.12×10 <sup>4</sup>	37.8
200-West Area steam reforming waste	3.08	6.4	4.41×10 <sup>6</sup>	9.6	0	0.0	0	0.0	3.01×10 <sup>1</sup>	3.2	9.58×10 <sup>-1</sup>	0.7	1.04×10 <sup>3</sup>	1.3	4.61×10 <sup>5</sup>	0.9	9.42×10 <sup>3</sup>	31.7
Transuranic waste <sup>f</sup>	5.02×10 <sup>-2</sup>	0.1	3.41×10 <sup>5</sup>	0.7	3.85	0.1	3.33	0.0	4.67×10 <sup>1</sup>	5.0	9.79×10 <sup>-1</sup>	0.7	7.39×10 <sup>3</sup>	9.1	7.52×10 <sup>5</sup>	1.5	3.36×10 <sup>2</sup>	1.1
<b>Total<sup>g</sup></b>	<b>4.83×10<sup>1</sup></b>	<b>100.3</b>	<b>4.62×10<sup>7</sup></b>	<b>100.7</b>	<b>3.98×10<sup>1</sup></b>	<b>1.3</b>	<b>1.24×10<sup>2</sup></b>	<b>1.0</b>	<b>9.59×10<sup>2</sup></b>	<b>102.2</b>	<b>1.42×10<sup>2</sup></b>	<b>100.4</b>	<b>8.26×10<sup>4</sup></b>	<b>101.4</b>	<b>5.12×10<sup>7</sup></b>	<b>101.3</b>	<b>3.00×10<sup>4</sup></b>	<b>101.1</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>h</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
Rubble, soil, and equipment <sup>i</sup>	1.67×10 <sup>-2</sup>	N/A	1.31×10 <sup>4</sup>	N/A	1.47	N/A	6.03	N/A	4.82×10 <sup>-1</sup>	N/A	3.24×10 <sup>-2</sup>	N/A	4.32×10 <sup>1</sup>	N/A	3.05×10 <sup>4</sup>	N/A	9.72	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>j</sup>	4.78×10 <sup>1</sup>	N/A	4.69×10 <sup>4</sup>	N/A	3.10×10 <sup>3</sup>	N/A	1.19×10 <sup>4</sup>	N/A	4.52×10 <sup>-1</sup>	N/A	NR	N/A	3.72×10 <sup>1</sup>	N/A	3.54×10 <sup>4</sup>	N/A	1.47×10 <sup>1</sup>	N/A

- a Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).
- b Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.
- c To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.
- d Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.
- e Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.
- f Tank transuranic waste would be disposed of in the Waste Isolation Pilot Plant.
- g Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent; a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.
- h To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.
- i Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.
- j Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed for the appropriate treatment processes.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported.

**Source:** SAIC 2011.

**Table D-46. Tank Closure Alternative 3C Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>2</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>5</sup>	1.0	1.83×10 <sup>3</sup>	1.0	7.08×10 <sup>7</sup>	1.0	8.41×10 <sup>2</sup>	1.0	5.97×10 <sup>5</sup>	1.0	2.95×10 <sup>4</sup>	1.0	2.40×10 <sup>-2</sup>	1.0	3.45×10 <sup>4</sup>	1.0	1.68×10 <sup>1</sup>	1.0	1.11×10 <sup>-2</sup>	1.0
IHLW glass <sup>c</sup>	1.12×10 <sup>5</sup>	18.6	0	0.0	0	0.0	7.07×10 <sup>4</sup>	84.1	5.11×10 <sup>5</sup>	85.5	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters	1.34×10 <sup>5</sup>	22.4	0	0.0	0	0.0	2.67×10 <sup>3</sup>	3.2	1.13×10 <sup>4</sup>	1.9	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>d</sup>	2.72×10 <sup>1</sup>	0.0	5.54	0.3	9.18×10 <sup>6</sup>	13.0	7.56×10 <sup>2</sup>	0.9	6.39×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>e</sup>	8.03×10 <sup>2</sup>	0.1	1.75×10 <sup>3</sup>	96.0	0	0.0	2.17×10 <sup>2</sup>	0.3	2.05×10 <sup>3</sup>	0.3	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
200-East Area steam reforming waste	1.76×10 <sup>5</sup>	29.3	0	0	0	0.0	3.16×10 <sup>3</sup>	3.8	1.48×10 <sup>4</sup>	2.5	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
200-West Area steam reforming waste	1.49×10 <sup>5</sup>	24.9	0	0	0	0.0	3.58×10 <sup>3</sup>	4.3	1.99×10 <sup>4</sup>	3.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
TRU waste <sup>f</sup>	2.83×10 <sup>4</sup>	4.7	3.46×10 <sup>1</sup>	1.9	1.07×10 <sup>6</sup>	1.5	6.43×10 <sup>3</sup>	7.7	5.92×10 <sup>4</sup>	9.9	5.93×10 <sup>2</sup>	2.0	4.83×10 <sup>-2</sup>	2.0	6.95×10 <sup>4</sup>	2.0	3.51×10 <sup>1</sup>	2.1	2.23×10 <sup>-2</sup>	2.0
<b>Total<sup>g</sup></b>	<b>6.05×10<sup>5</sup></b>	<b>101.1</b>	<b>1.81×10<sup>3</sup></b>	<b>99.2</b>	<b>1.10×10<sup>7</sup></b>	<b>15.5</b>	<b>8.84×10<sup>4</sup></b>	<b>105.1</b>	<b>6.24×10<sup>5</sup></b>	<b>104.5</b>	<b>8.88×10<sup>2</sup></b>	<b>3.0</b>	<b>7.24×10<sup>-2</sup></b>	<b>3.0</b>	<b>1.04×10<sup>5</sup></b>	<b>3.0</b>	<b>5.19×10<sup>1</sup></b>	<b>3.1</b>	<b>3.34×10<sup>-2</sup></b>	<b>3.0</b>
<b>Other Inventory<sup>h</sup></b>																				
Rubble, soil, and equipment <sup>i</sup>	5.86×10 <sup>3</sup>	N/A	2.22	N/A	3.93×10 <sup>4</sup>	N/A	3.34×10 <sup>1</sup>	N/A	6.60×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>j</sup>	NR	N/A	1.80×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	3.16×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>e</sup> Includes solid LLW and MLLW streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>f</sup> Tank TRU waste would be disposed of in the Waste Isolation Pilot Plant.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during the thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>h</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>i</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes only inventories from facility air emissions, including those from treatment of cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed for the appropriate treatment processes.

**Note:** To convert kilograms to pounds multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; LLW=low-level radioactive waste; MLLW=mixed low-level radioactive waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; TCP=trichlorophenol; TRU=transuranic.

**Source:** SAIC 2011.



Under Tank Closure Alternative 4, the primary-waste forms produced would be IHLW glass, ILAW glass, and a combination of the supplemental-waste forms; i.e., bulk vitrification glass and cast stone waste. The majority of technetium-99 would be immobilized in the ILAW glass and supplemental-waste forms. Under Tank Closure Alternative 4, the ILAW glass, bulk vitrification glass, cast stone waste, and secondary waste would contain 5.8 percent, 6.5 percent, 38.7 percent, and 49.2 percent, respectively, of the BBI estimate for iodine-129. A process flowsheet is presented in Figure D-7, and material balances under Alternative 4 are presented in Tables D-47 and D-48.

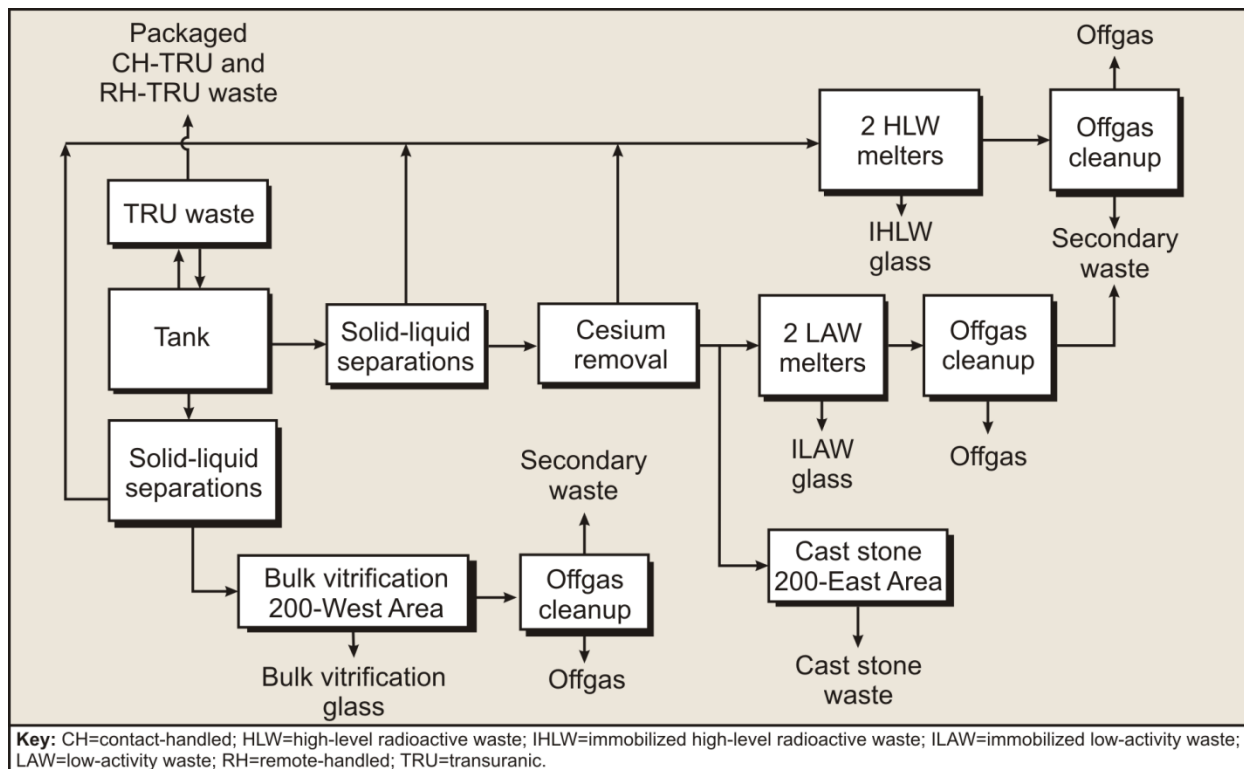


Figure D-7. Tank Closure Alternative 4 Flowsheet

**Table D-47. Tank Closure Alternative 4 Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-2</sup>	0.1	4.58×10 <sup>4</sup>	0.1	3.12	0.1	1.21×10 <sup>1</sup>	0.1	9.38×10 <sup>-1</sup>	0.1	1.41×10 <sup>-1</sup>	0.1	8.14×10 <sup>1</sup>	0.1	5.05×10 <sup>4</sup>	0.1	2.97×10 <sup>1</sup>	0.1
IHLW glass <sup>c, d</sup>	6.55×10 <sup>-3</sup>	0.0	4.08×10 <sup>7</sup>	89.1	0	0.0	0	0.0	8.41×10 <sup>2</sup>	89.6	1.39×10 <sup>2</sup>	98.7	7.40×10 <sup>4</sup>	90.9	4.93×10 <sup>7</sup>	97.6	1.70×10 <sup>2</sup>	0.6
ILAW glass and retired LAW melters <sup>d</sup>	2.81	5.8	2.39×10 <sup>5</sup>	0.5	0	0.0	0	0.0	2.01×10 <sup>1</sup>	2.1	9.80×10 <sup>-2</sup>	0.1	6.17	0.0	1.41×10 <sup>4</sup>	0.0	8.48×10 <sup>3</sup>	28.5
ETF-generated solid secondary waste <sup>e</sup>	2.24×10 <sup>1</sup>	46.4	1.36×10 <sup>1</sup>	0	3.62	0.1	0	0.0	6.71×10 <sup>-2</sup>	0.0	5.20×10 <sup>-2</sup>	0.0	9.47×10 <sup>-4</sup>	0.0	5.49×10 <sup>1</sup>	0.0	3.53×10 <sup>1</sup>	0.1
Solid secondary waste <sup>f</sup>	1.37	2.8	1.77×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.38	0.4	2.82×10 <sup>-1</sup>	0.2	1.82×10 <sup>2</sup>	0.2	7.76×10 <sup>5</sup>	1.5	1.28×10 <sup>2</sup>	0.4
200-East Area cast stone waste	1.86×10 <sup>1</sup>	38.7	2.31×10 <sup>5</sup>	0.5	1.18×10 <sup>3</sup>	37.9	4.66×10 <sup>3</sup>	38.7	2.17×10 <sup>1</sup>	2.3	3.30×10 <sup>-3</sup>	0.0	5.77×10 <sup>-1</sup>	0.0	9.23×10 <sup>2</sup>	0.0	1.14×10 <sup>4</sup>	38.4
200-West Area BV glass <sup>g</sup>	3.11	6.5	4.39×10 <sup>6</sup>	9.6	0	0.0	0	0.0	3.03×10 <sup>1</sup>	3.2	9.66×10 <sup>-1</sup>	0.7	1.05×10 <sup>3</sup>	1.3	4.65×10 <sup>5</sup>	0.9	9.50×10 <sup>3</sup>	32.0
Transuranic waste <sup>h</sup>	5.07×10 <sup>-2</sup>	0.1	3.44×10 <sup>5</sup>	0.8	3.88	0.1	3.36	0.0	4.71×10 <sup>1</sup>	5.0	9.88×10 <sup>-1</sup>	0.7	7.46×10 <sup>3</sup>	9.2	7.59×10 <sup>5</sup>	1.5	3.39×10 <sup>2</sup>	1.1
<b>Total<sup>i</sup></b>	<b>4.84×10<sup>1</sup></b>	<b>100.4</b>	<b>4.63×10<sup>7</sup></b>	<b>100.9</b>	<b>1.19×10<sup>3</sup></b>	<b>38.2</b>	<b>4.68×10<sup>3</sup></b>	<b>38.8</b>	<b>9.64×10<sup>2</sup></b>	<b>102.7</b>	<b>1.42×10<sup>2</sup></b>	<b>100.5</b>	<b>8.28×10<sup>4</sup></b>	<b>101.7</b>	<b>5.14×10<sup>7</sup></b>	<b>101.7</b>	<b>3.01×10<sup>4</sup></b>	<b>101.2</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>j</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
PPF secondary waste and rubble, soil, and equipment <sup>k</sup>	5.84×10 <sup>-2</sup>	N/A	9.01×10 <sup>4</sup>	N/A	1.07×10 <sup>1</sup>	N/A	7.19×10 <sup>1</sup>	N/A	5.16	N/A	1.36×10 <sup>-1</sup>	N/A	5.59×10 <sup>1</sup>	N/A	5.41×10 <sup>4</sup>	N/A	3.14×10 <sup>1</sup>	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>l</sup>	2.96×10 <sup>1</sup>	N/A	4.70×10 <sup>4</sup>	N/A	1.95×10 <sup>3</sup>	N/A	7.46×10 <sup>3</sup>	N/A	4.48×10 <sup>-1</sup>	N/A	NR	N/A	3.77×10 <sup>1</sup>	N/A	3.58×10 <sup>4</sup>	N/A	9.26	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes PPF contribution from clean closure of BX and SX tank farms.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Includes technetium-99 inventory that resides in the BV waste container insulating material or waste container.

<sup>h</sup> Tank transuranic waste would be disposed of in the Waste Isolation Pilot Plant.

<sup>i</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI of percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent; a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>j</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>k</sup> Rubble, soil, and equipment would be generated by clean closure of the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>l</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; BV=bulk vitrification; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported; PPF=Preprocessing Facility.

**Source:** SAIC 2011.

Table D-48. Tank Closure Alternative 4 Chemical Constituents of Potential Concern Balance

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>2</sup>	0.1	1.83	0.1	7.08×10 <sup>4</sup>	0.1	8.41×10 <sup>1</sup>	0.1	5.97×10 <sup>2</sup>	0.1	2.95×10 <sup>1</sup>	0.1	2.40×10 <sup>-3</sup>	0.1	3.45×10 <sup>3</sup>	0.1	1.68	0.1	1.11×10 <sup>-3</sup>	0.1
IHLW glass <sup>c, d</sup>	1.16×10 <sup>5</sup>	19.4	0	0.0	0	0.0	7.14×10 <sup>4</sup>	84.9	5.15×10 <sup>5</sup>	86.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters	1.35×10 <sup>5</sup>	22.6	0	0.0	0	0.0	2.27×10 <sup>3</sup>	3.3	1.31×10 <sup>4</sup>	2.2	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>e</sup>	2.31×10 <sup>1</sup>	0.0	4.86	0.3	5.20×10 <sup>6</sup>	7.3	4.08×10 <sup>2</sup>	0.5	5.60×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>f</sup>	8.22×10 <sup>2</sup>	0.1	1.54×10 <sup>3</sup>	84.2	0	0.0	2.20×10 <sup>2</sup>	0.3	2.08×10 <sup>3</sup>	0.3	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
200-East Area cast stone waste	1.78×10 <sup>5</sup>	29.8	2.44×10 <sup>2</sup>	13.4	2.73×10 <sup>7</sup>	38.6	3.57×10 <sup>3</sup>	4.2	1.50×10 <sup>4</sup>	2.5	1.12×10 <sup>4</sup>	37.9	9.11×10 <sup>-1</sup>	37.9	1.31×10 <sup>6</sup>	37.9	6.39	0.4	4.20×10 <sup>-1</sup>	37.9
200-West Area BV glass	1.49×10 <sup>5</sup>	25.0	0	0.0	0	0.0	3.59×10 <sup>3</sup>	4.3	1.99×10 <sup>4</sup>	3.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
TRU waste <sup>g</sup>	2.86×10 <sup>4</sup>	4.8	3.49×10 <sup>1</sup>	1.9	1.08×10 <sup>6</sup>	1.5	6.49×10 <sup>3</sup>	7.7	5.98×10 <sup>4</sup>	10.0	5.99×10 <sup>2</sup>	2.0	4.88×10 <sup>-2</sup>	2.0	7.01×10 <sup>4</sup>	2.0	3.54×10 <sup>1</sup>	2.1	2.25×10 <sup>-2</sup>	2.0
<b>Total<sup>h</sup></b>	<b>6.09×10<sup>5</sup></b>	<b>101.7</b>	<b>1.82×10<sup>3</sup></b>	<b>99.9</b>	<b>3.37×10<sup>7</sup></b>	<b>47.5</b>	<b>8.86×10<sup>4</sup></b>	<b>105.3</b>	<b>6.26×10<sup>5</sup></b>	<b>104.8</b>	<b>1.18×10<sup>4</sup></b>	<b>40.1</b>	<b>9.62×10<sup>-1</sup></b>	<b>40.1</b>	<b>1.38×10<sup>6</sup></b>	<b>40.1</b>	<b>4.35×10<sup>1</sup></b>	<b>2.6</b>	<b>4.44×10<sup>-1</sup></b>	<b>40.1</b>
<b>Other Inventory<sup>i</sup></b>																				
PPF secondary waste <sup>j</sup> and rubble, soil, and equipment <sup>k</sup>	1.86×10 <sup>3</sup>	N/A	1.28	N/A	7.78×10 <sup>4</sup>	N/A	4.27×10 <sup>1</sup>	N/A	4.85×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	2.68	N/A	1.64×10 <sup>-1</sup>	N/A	NR	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>l</sup>	NR	N/A	1.64×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	8.78×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes PPF contribution from clean closure of the BX and SX tank farms.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Tank TRU waste would be disposed of in the Waste Isolation Pilot Plant.

<sup>h</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during the thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>i</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>j</sup> Includes the solid secondary mixed low-level radioactive waste stream generated by the PPF and the solid waste generated from treating PPF liquid and solid secondary waste, as well as solid secondary waste. Disposal would take place in the River Protection Project Disposal Facility.

<sup>k</sup> Rubble, soil, and equipment would be generated by clean closure of the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>l</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

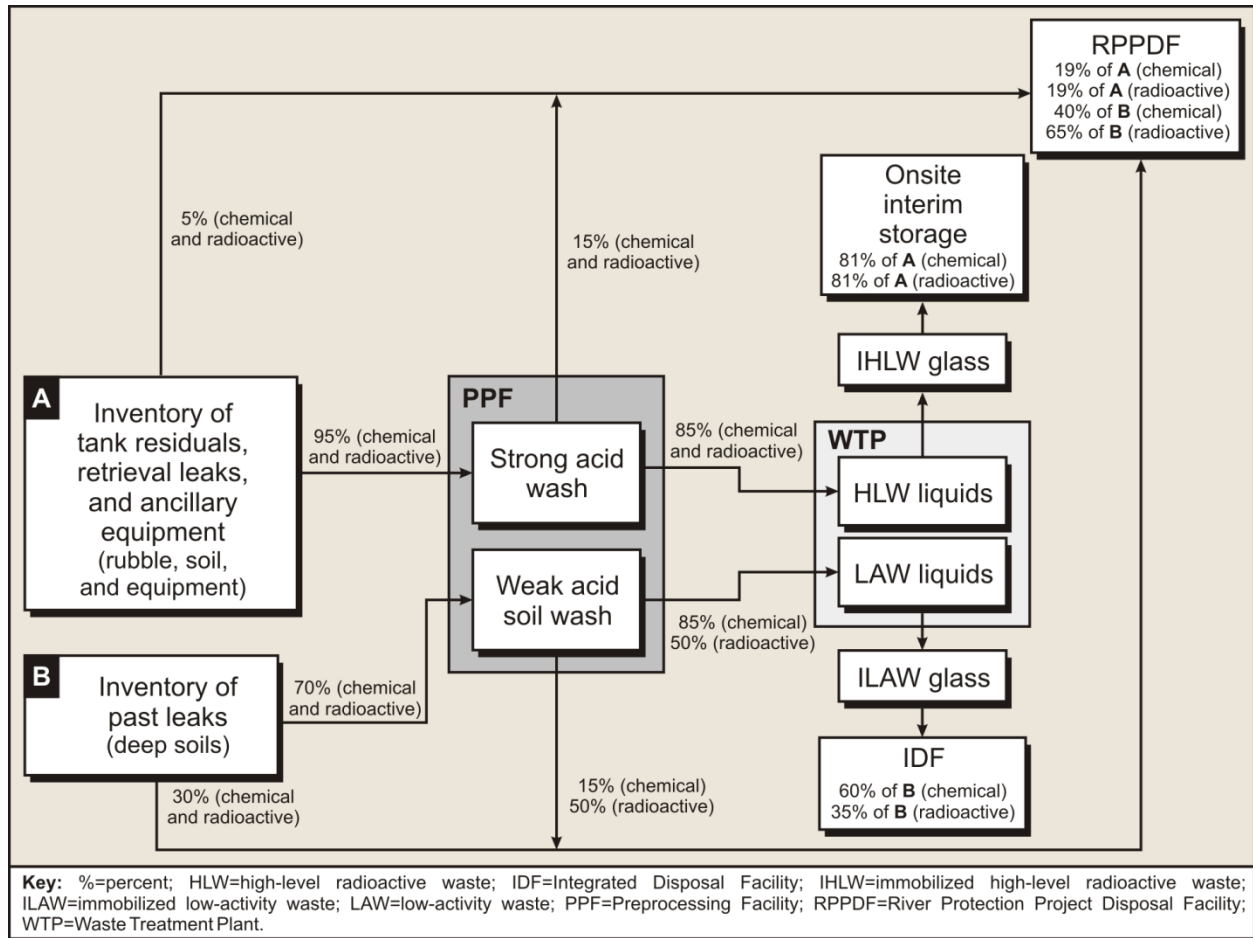
**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; BV=bulk vitrification; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; PPF=Preprocessing Facility; TCP=trichlorophenol; TRU=transuranic.

**Source:** SAIC 2011.

In addition, under Tank Closure Alternative 4 (see Figure D–8 for a simplified flowsheet and Tables D–49 and D–50 for inventories), selected tank farms, represented by the BX and SX tank farms, would undergo clean closure. Under clean closure, the SSTs, soils contaminated with leaks from retrieval activities, and soils contaminated by past tank leaks in these two tank farms would be removed. The more highly contaminated portions of the removed materials would be sent to a proposed Preprocessing Facility (PPF) for decontamination.

Under this alternative, it was assumed that 95 percent of the radioactive and chemical constituent inventory remaining in the tanks and ancillary equipment and from leaks associated with waste retrieval would be sent to the PPF, while 5 percent of the inventory would be packaged and sent directly to the River Protection Project Disposal Facility (RPPDF) as mixed low-level radioactive waste (MLLW). It was further assumed that the PPF processes would be effective at removing 85 percent of the contaminants from the rubble, soil, and equipment contaminated with tank waste retrieval leaks from retrieval activities. This treated material would be sent to the WTP, where it would be processed with the HLW stream. The remaining 15 percent would remain with the contaminated rubble, soil, and equipment and would be disposed of as MLLW in the RPPDF. The resulting value would be 14.25 percent (15 percent of 95 percent), rounded to 14 percent. Thus, 19 percent of the inventory of contaminants from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks would be sent to the RPPDF as MLLW (SAIC 2010a).



**Figure D–8. Tank Closure Alternative 4 Clean Closure of BX and SX Tank Farms Flowsheet**

**Table D–49. Tank Closure Alternative 4 Radioactive Constituents of Potential Concern Inventory from Clean Closure of BX and SX Tank Farms (curies)**

Analyte	MLLW <sup>a</sup>	IHLW Glass <sup>b</sup>	ILAW Glass <sup>c</sup>
Hydrogen-3 (tritium)	7.19×10 <sup>1</sup>	0	0
Carbon-14	1.07×10 <sup>1</sup>	0	0
Strontium-90	5.41×10 <sup>4</sup>	1.69×10 <sup>5</sup>	1.34×10 <sup>4</sup>
Technetium-99	3.14×10 <sup>1</sup>	1.99×10 <sup>1</sup>	2.09×10 <sup>1</sup>
Iodine-129	5.84×10 <sup>-2</sup>	6.55×10 <sup>-3</sup>	8.03×10 <sup>-3</sup>
Cesium-137	9.01×10 <sup>4</sup>	2.86×10 <sup>4</sup>	6.49×10 <sup>4</sup>
Uranium-233, -234, -235, -238	5.16	1.29	3.78
Neptunium-237	1.36×10 <sup>-1</sup>	6.36×10 <sup>-2</sup>	9.54×10 <sup>-2</sup>
Plutonium-239, -240	5.59×10 <sup>1</sup>	2.29×10 <sup>2</sup>	5.73

<sup>a</sup> Represents 19 percent of the contaminant inventory from the residual waste in tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 65 percent of the contaminant inventory in deep soils for the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 85 percent of the highly contaminated rubble, soil, and equipment (tank and ancillary equipment) contaminant inventory resulting from clean closure of the BX and SX tank farms that would reside in the IHLW glass after treatment in the Waste Treatment Plant. IHLW would be disposed of off site; however, it may remain on site until disposition decisions are made and implemented.

<sup>c</sup> Represents the portion of the 50 percent of the highly contaminated rubble and soil (deep soil) contaminant inventory resulting from clean closure of the BX and SX tank farms that would reside in the ILAW glass after treatment in the Waste Treatment Plant. Disposal would take place in an Integrated Disposal Facility.

**Key:** IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; MLLW=mixed low-level radioactive waste.

**Source:** SAIC 2011.

**Table D–50. Tank Closure Alternative 4 Chemical Constituents of Potential Concern Inventory from Clean Closure of BX and SX Tank Farms (kilograms)**

Analyte	MLLW <sup>a</sup>	IHLW Glass <sup>b</sup>	ILAW Glass <sup>c</sup>
Chromium	1.86×10 <sup>3</sup>	3.33×10 <sup>3</sup>	1.36×10 <sup>3</sup>
Mercury	1.28	0	0
Nitrate	7.78×10 <sup>4</sup>	0	0
Lead	4.27×10 <sup>1</sup>	5.34×10 <sup>1</sup>	9.18×10 <sup>1</sup>
Total uranium	4.85×10 <sup>3</sup>	0	1.79×10 <sup>3</sup>
Acetonitrile	NR	NR	NR
Benzene	NR	NR	NR
Butanol (n-butyl alcohol)	2.68	NR	NR
Polychlorinated biphenyls	1.64×10 <sup>-1</sup>	NR	NR
2,4,6-Trichlorophenol	NR	NR	NR

<sup>a</sup> Represents 19 percent of the contaminant inventory from the residual waste in tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 40.5 percent of the contaminant inventory in deep soils for the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 85 percent of the highly contaminated rubble, soil, and equipment (tank and ancillary equipment) contaminant inventory resulting from clean closure of the BX and SX tank farms that would reside in the IHLW glass after treatment in the Waste Treatment Plant. IHLW would be disposed of off site; however, it may remain on site until disposition decisions are made and implemented.

<sup>c</sup> Represents the portion of the 85 percent of the highly contaminated rubble and soil (deep soil) contaminant inventory resulting from clean closure of the BX and SX tank farms that would reside in the ILAW glass after treatment in the Waste Treatment Plant. Disposal would take place in an Integrated Disposal Facility.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; MLLW=mixed low-level radioactive waste; NR=not reported.

**Source:** SAIC 2011.

Additionally, it was assumed that, after the soil contaminated from past tank leaks (deep soil) has been removed, 30 percent of the radioactive and chemical constituent inventory would be included in materials that would be packaged and sent directly to the RPPDF. The remaining 70 percent of the contaminants would be contained in soils that would be routed to the PPF for soil washing. From this 70 percent, it was assumed that the PPF processes would remove 50 percent of the radioactive contaminants and 85 percent of the chemical contaminants. Those radioactive and chemical contaminants removed in the PPF would be sent to the WTP, where they would be processed into ILAW glass. The remaining contaminants (50 percent radioactive and 15 percent chemical) would reside in the decontaminated soil and would be disposed of in the RPPDF. Thus, a total of 65 percent of the radioactive contaminant inventory resulting from past tank leaks would be disposed of in the RPPDF (30 percent disposed of directly and 35 percent [half of 70 percent] disposed of as MLLW after washing in the PPF). Similarly, a total of 41 percent of the chemical contaminant inventory resulting from past tank leaks would be disposed of in the RPPDF (30 percent disposed of directly and 11 percent [15 percent of 70 percent] disposed of as MLLW after washing in the PPF). The following equations were used to calculate the inventory of contaminants due to contaminated tank materials, rubble, soil, and ancillary equipment from clean closure of the BX and SX SST farms (SAIC 2010a):

$$M_{\text{rad}_{\text{soil}}} = 0.19 \times (M_{\text{TR-99.9}} + M_{\text{retrieval}} + M_{\text{anc}}) + 0.65 \times M_{\text{pleak}}$$

and

$$M_{\text{chem}_{\text{soil}}} = 0.19 \times (M_{\text{TR-99.9}} + M_{\text{retrieval}} + M_{\text{anc}}) + 0.41 \times M_{\text{pleak}}$$

where:

- $M_{\text{rad}_{\text{soil}}}$  = inventory of radioactive constituents in contaminated rubble, soil, and equipment disposed of on site
- $M_{\text{TR-99.9}}$  = inventory of radioactive or chemical constituents in tank residual waste following removal of 99.9 percent of the inventory
- $M_{\text{retrieval}}$  = inventory of radioactive or chemical constituents from tank waste retrieval leaks
- $M_{\text{anc}}$  = inventory of radioactive or chemical constituents in ancillary equipment
- $M_{\text{pleak}}$  = inventory of radioactive or chemical constituents in past leaks
- $M_{\text{chem}_{\text{soil}}}$  = inventory of chemical constituents in contaminated rubble, soil, and equipment disposed of on site

Processing steps under Tank Closure Alternative 5 would be similar to those under Tank Closure Alternative 4, except that a step for removal of sulfate from the LAW stream feed to the LAW melter would be added. Most of the technetium-99 would be solidified in LAW forms. Under Tank Closure Alternative 5, the ILAW glass, bulk vitrification glass, cast stone waste, and secondary waste would be 9.6 percent, 5.8 percent, 13.2 percent, and 61.6 percent, respectively, of the BBI estimate for iodine-129. The process flowsheet for Tank Closure Alternative 5 is presented in Figure D-9, and material balance summaries are presented in Tables D-51 and D-52.

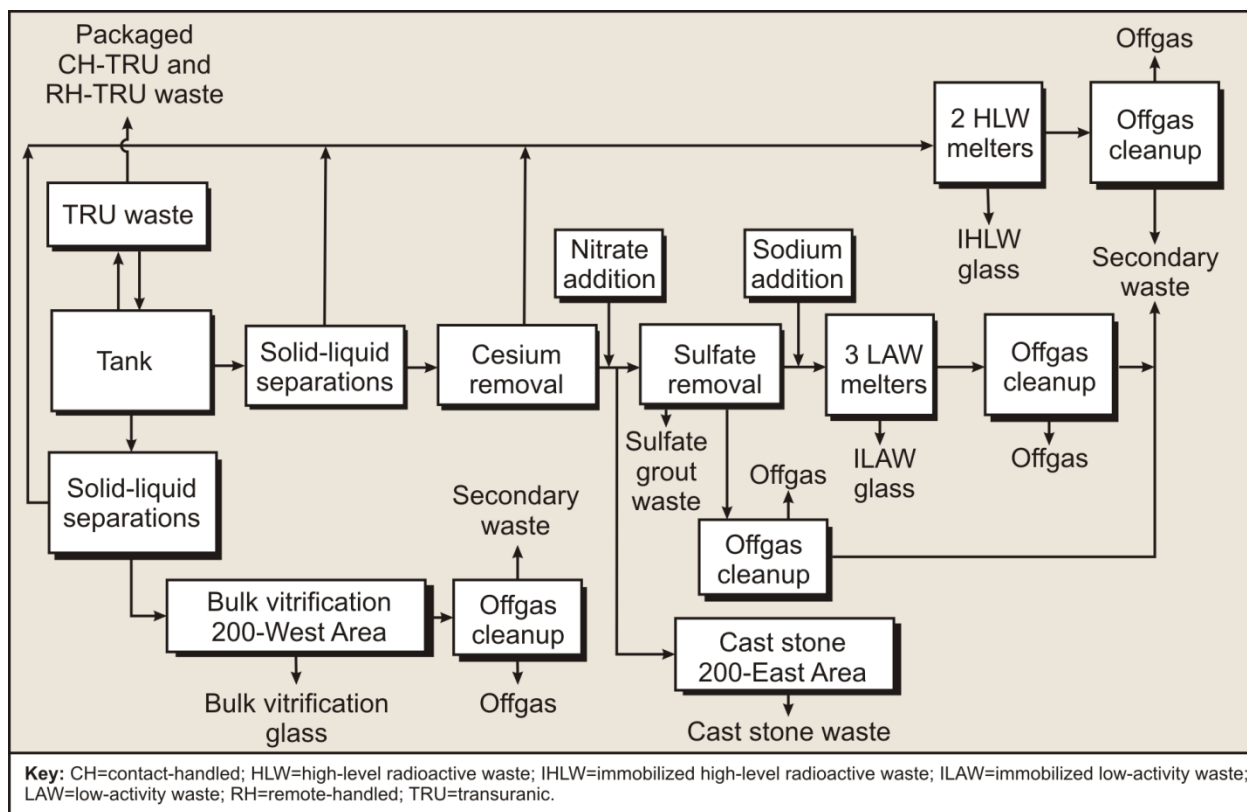


Figure D-9. Tank Closure Alternative 5 Flowsheet

**Table D-51. Tank Closure Alternative 5 Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82	10.0	4.58×10 <sup>6</sup>	10.0	3.12×10 <sup>2</sup>	10.0	1.21×10 <sup>3</sup>	10.0	9.38×10 <sup>1</sup>	10.0	1.41×10 <sup>1</sup>	10.0	8.14×10 <sup>3</sup>	10.0	5.05×10 <sup>6</sup>	10.0	2.97×10 <sup>3</sup>	10.0
IHLW glass <sup>c</sup>	1.32×10 <sup>-5</sup>	0.0	3.63×10 <sup>7</sup>	79.2	0	0.0	0	0.0	7.56×10 <sup>2</sup>	80.6	1.25×10 <sup>2</sup>	88.9	6.65×10 <sup>4</sup>	81.6	4.43×10 <sup>7</sup>	87.6	1.35×10 <sup>2</sup>	0.5
ILAW glass and retired LAW melters	4.61	9.6	2.51×10 <sup>5</sup>	0.5	0	0.0	0	0.0	2.67×10 <sup>1</sup>	2.8	3.92×10 <sup>-3</sup>	0.0	6.13×10 <sup>-1</sup>	0.0	1.14×10 <sup>-1</sup>	0.0	1.39×10 <sup>4</sup>	46.8
Sulfate grout waste <sup>d</sup>	0	0.0%	3.11×10 <sup>4</sup>	0.1	0	0.0	0	0.0	0	0.0	1.63×10 <sup>-4</sup>	0.0	9.98×10 <sup>-2</sup>	0.0	1.14×10 <sup>3</sup>	0.0	0	0.0
ETF-generated solid secondary waste <sup>e</sup>	2.74×10 <sup>1</sup>	56.9	1.36×10 <sup>1</sup>	0.0	1.04	0.0	0	0.0	6.07×10 <sup>-2</sup>	0.0	4.68×10 <sup>-2</sup>	0.0	8.51×10 <sup>-4</sup>	0.0	4.95×10 <sup>1</sup>	0.0	5.03×10 <sup>1</sup>	0.2
Solid secondary waste <sup>f</sup>	2.24	4.7	1.58×10 <sup>5</sup>	0.3	0	0.0	0	0.0	3.07	0.3	2.54×10 <sup>-1</sup>	0.2	1.64×10 <sup>2</sup>	0.2	6.97×10 <sup>5</sup>	1.4	2.08×10 <sup>2</sup>	0.7
200-East Area cast stone waste	6.38	13.2	7.82×10 <sup>4</sup>	0.2	4.05×10 <sup>2</sup>	13.0	1.60×10 <sup>3</sup>	13.2	7.41	0.8	1.13×10 <sup>-3</sup>	0.0	1.97×10 <sup>-1</sup>	0.0	3.16×10 <sup>2</sup>	0.0	3.90×10 <sup>3</sup>	13.1
200-West Area BV glass <sup>g</sup>	2.80	5.8	4.39×10 <sup>6</sup>	9.6	0	0.0	0	0.0	2.73×10 <sup>1</sup>	2.9	8.71×10 <sup>-1</sup>	0.6	9.49×10 <sup>2</sup>	1.2	4.19×10 <sup>5</sup>	0.8	8.56×10 <sup>3</sup>	28.8
TRU waste <sup>h</sup>	4.57×10 <sup>-2</sup>	0.1	3.10×10 <sup>5</sup>	0.7	3.50	0.1	3.03	0.0	4.24×10 <sup>1</sup>	4.5	8.90×10 <sup>-1</sup>	0.6	6.72×10 <sup>3</sup>	8.2	6.84×10 <sup>5</sup>	1.4	3.06×10 <sup>2</sup>	1.0
<b>Total<sup>i</sup></b>	<b>4.83×10<sup>1</sup></b>	<b>100.3</b>	<b>4.61×10<sup>7</sup></b>	<b>100.6</b>	<b>7.22×10<sup>2</sup></b>	<b>23.1</b>	<b>2.80×10<sup>3</sup></b>	<b>23.3</b>	<b>9.57×10<sup>2</sup></b>	<b>102.0</b>	<b>1.42×10<sup>2</sup></b>	<b>100.4</b>	<b>8.25×10<sup>4</sup></b>	<b>101.2</b>	<b>5.11×10<sup>7</sup></b>	<b>101.2</b>	<b>3×10<sup>4</sup></b>	<b>101.1</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>j</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>k</sup>	3.70×10 <sup>1</sup>	N/A	4.47×10 <sup>4</sup>	N/A	2.41×10 <sup>3</sup>	N/A	9.27×10 <sup>3</sup>	N/A	4.07×10 <sup>-1</sup>	N/A	NR	N/A	3.39×10 <sup>1</sup>	N/A	3.32×10 <sup>4</sup>	N/A	1.16×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 90.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Generated by removal of sulfate from the ILAW waste stream. Disposal would take place in an Integrated Disposal Facility.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Includes technetium-99 inventory that resides in the BV waste container insulating material or waste container.

<sup>h</sup> Tank TRU waste disposal would take place in the Waste Isolation Pilot Plant.

<sup>i</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI of percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent; a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>j</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>k</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; BV=bulk vitrification; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported; TRU=transuranic.

**Source:** SAIC 2011.



**Table D-52. Tank Closure Alternative 5 Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>4</sup>	10.0	1.83×10 <sup>2</sup>	10.0	7.08×10 <sup>6</sup>	10.0	8.41×10 <sup>3</sup>	10.0	5.97×10 <sup>4</sup>	10.0	2.95×10 <sup>3</sup>	10.0	2.40×10 <sup>-1</sup>	10.0	3.45×10 <sup>5</sup>	10.0	1.68×10 <sup>2</sup>	10.0	1.11×10 <sup>-1</sup>	10.0
IHLW glass <sup>c</sup>	1.01×10 <sup>5</sup>	16.9	0	0.0	0	0.0	6.43×10 <sup>4</sup>	76.5	4.64×10 <sup>5</sup>	77.8	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters	2.20×10 <sup>1</sup>	0.0	0	0.0	0	0.0	4.40×10 <sup>-1</sup>	0.0	1.86×10 <sup>4</sup>	3.1	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
Sulfate grout waste <sup>d</sup>	2.21×10 <sup>5</sup>	36.9	0	0.0	0	0.0	4.41×10 <sup>3</sup>	5.2	0	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
ETF-generated solid secondary waste <sup>e</sup>	1.15×10 <sup>1</sup>	0.0	4.77	0.3	1.20×10 <sup>7</sup>	16.9	3.67×10 <sup>2</sup>	0.4	5.09×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>f</sup>	3.32×10 <sup>2</sup>	0.1	1.51×10 <sup>3</sup>	82.8	0	0.0	1.90×10 <sup>2</sup>	0.2	1.90×10 <sup>3</sup>	0.3	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
200-East Area cast stone waste	6.11×10 <sup>4</sup>	10.2	8.36×10 <sup>1</sup>	4.6	9.35×10 <sup>6</sup>	13.2	1.22×10 <sup>3</sup>	1.5	5.15×10 <sup>3</sup>	0.9	3.71×10 <sup>3</sup>	12.6	3.02×10 <sup>-1</sup>	12.6	4.35×10 <sup>5</sup>	12.6	2.12	0.1	1.39×10 <sup>-1</sup>	12.6
200-West Area BV glass	1.35×10 <sup>5</sup>	22.5	0	0.0	0	0.0	3.23×10 <sup>3</sup>	3.8	1.80×10 <sup>4</sup>	3.0	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
TRU waste <sup>g</sup>	2.57×10 <sup>4</sup>	4.3	3.15×10 <sup>1</sup>	1.7	9.69×10 <sup>5</sup>	1.4	5.85×10 <sup>3</sup>	7.0	5.38×10 <sup>4</sup>	9.0	5.39×10 <sup>2</sup>	1.8	4.39×10 <sup>-2</sup>	1.8	6.32×10 <sup>4</sup>	1.8	3.19×10 <sup>1</sup>	1.9	2.03×10 <sup>-2</sup>	1.8
<b>Total<sup>h</sup></b>	<b>6.03×10<sup>5</sup></b>	<b>100.8</b>	<b>1.81×10<sup>3</sup></b>	<b>99.4</b>	<b>2.94×10<sup>7</sup></b>	<b>41.5</b>	<b>8.80×10<sup>4</sup></b>	<b>104.6</b>	<b>6.21×10<sup>5</sup></b>	<b>104.1</b>	<b>7.20×10<sup>3</sup></b>	<b>24.4</b>	<b>5.86×10<sup>-1</sup></b>	<b>24.4</b>	<b>8.43×10<sup>5</sup></b>	<b>24.4</b>	<b>2.03×10<sup>2</sup></b>	<b>12.0</b>	<b>2.70×10<sup>-1</sup></b>	<b>24.4</b>
<b>Other Inventory<sup>i</sup></b>																				
Treatment air emissions <sup>j</sup>	NR	N/A	1.56×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	1.57×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 90.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Generated by removal of sulfate from the ILAW feed stream. Disposal would take place in an Integrated Disposal Facility.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Tank TRU waste would be disposed of in the Waste Isolation Pilot Plant.

<sup>h</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>i</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

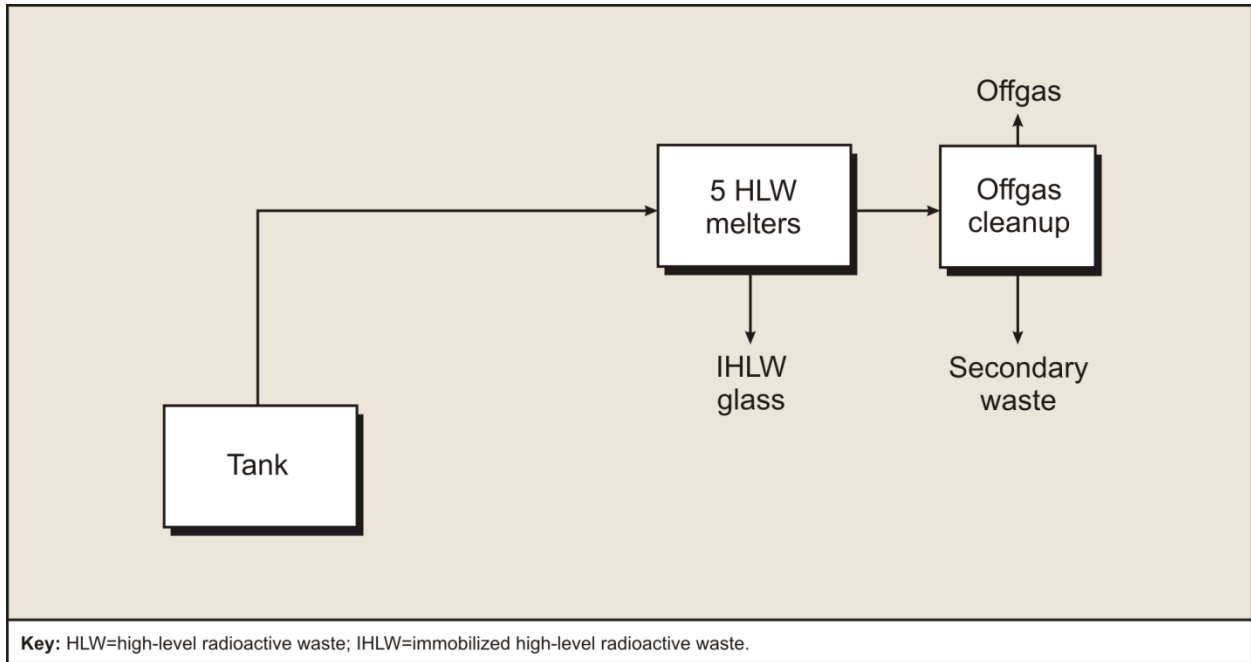
<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; BV=bulk vitrification; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; TCP=trichlorophenol; TRU=transuranic.

**Source:** SAIC 2011.

Under Tank Closure Alternative 6 are three subalternatives (6A, 6B, and 6C); two of these alternatives (6A and 6B) have two options: a Base Case and an Option Case. Under Tank Closure Alternative 6A, Base Case, represented in Figure D-10, all waste streams, including those from clean closure of the SSTs, would be managed as IHLW glass. Under Tank Closure Alternative 6A, Option Case, the six sets of contiguous cribs and trenches (ditches) described in Section D.1.5 (the B Cribs, BX Trenches, BY Cribs, T Cribs, T Trenches, and TX Trenches [considered to be one set], and TY Cribs) would be added to the Alternative 6A, Base Case, inventory. Material balance summaries for these two cases are presented in Tables D-53 through D-56.



**Figure D-10. Tank Closure Alternative 6A Flowsheet**

Table D-53. Tank Closure Alternative 6A, Base Case, Radioactive Constituents of Potential Concern Balance

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-2</sup>	0.1	4.58×10 <sup>4</sup>	0.1	3.12	0.1	1.21×10 <sup>1</sup>	0.1	9.38×10 <sup>-1</sup>	0.1	1.41×10 <sup>-1</sup>	0.1	8.14×10 <sup>1</sup>	0.1	5.05×10 <sup>4</sup>	0.1	2.97×10 <sup>1</sup>	0.1
IHLW glass <sup>c</sup>	9.64	20.0	4.57×10 <sup>7</sup>	99.7	0	0.0	0	0.0	9.36×10 <sup>2</sup>	99.8	1.41×10 <sup>2</sup>	99.9	8.14×10 <sup>4</sup>	99.9	4.98×10 <sup>7</sup>	98.5	2.96×10 <sup>4</sup>	99.6
ETF-generated solid secondary waste <sup>d</sup>	3.41×10 <sup>1</sup>	70.7	4.65×10 <sup>-1</sup>	0.0	8.63	0.3	0	0.0	4.11×10 <sup>-2</sup>	0.0	5.18×10 <sup>-2</sup>	0.0	6.96×10 <sup>-4</sup>	0.0	6.49	0.0	8.74×10 <sup>1</sup>	0.3
Solid secondary waste <sup>e</sup>	4.72	9.8	1.98×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.70	0.4	2.86×10 <sup>-1</sup>	0.2	2.00×10 <sup>2</sup>	0.2	7.84×10 <sup>5</sup>	1.6	4.37×10 <sup>2</sup>	1.5
<b>Total<sup>f</sup></b>	<b>4.85×10<sup>1</sup></b>	<b>100.6</b>	<b>4.60×10<sup>7</sup></b>	<b>100.2</b>	<b>1.18×10<sup>1</sup></b>	<b>0.4</b>	<b>1.21×10<sup>1</sup></b>	<b>0.1</b>	<b>9.41×10<sup>2</sup></b>	<b>100.3</b>	<b>1.41×10<sup>2</sup></b>	<b>100.2</b>	<b>8.16×10<sup>4</sup></b>	<b>100.2</b>	<b>5.06×10<sup>7</sup></b>	<b>100.1</b>	<b>3.02×10<sup>4</sup></b>	<b>101.5</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>g</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
PPF glass and retired PPF melters <sup>h</sup>	5.43×10 <sup>-2</sup>	N/A	2.08×10 <sup>5</sup>	N/A	0	N/A	0	N/A	7.83	N/A	5.26×10 <sup>-1</sup>	N/A	3.53×10 <sup>1</sup>	N/A	6.71×10 <sup>4</sup>	N/A	1.39×10 <sup>2</sup>	N/A
Rubble, soil, and equipment <sup>i</sup>	3.43×10 <sup>-1</sup>	N/A	2.62×10 <sup>5</sup>	N/A	2.36×10 <sup>1</sup>	N/A	2.17×10 <sup>2</sup>	N/A	1.02×10 <sup>1</sup>	N/A	6.68×10 <sup>-1</sup>	N/A	9.92×10 <sup>1</sup>	N/A	1.32×10 <sup>5</sup>	N/A	1.78×10 <sup>2</sup>	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>j</sup>	4.85×10 <sup>1</sup>	N/A	4.72×10 <sup>4</sup>	N/A	3.14×10 <sup>3</sup>	N/A	1.22×10 <sup>4</sup>	N/A	4.74×10 <sup>-1</sup>	N/A	NR	N/A	4.07×10 <sup>1</sup>	N/A	3.58×10 <sup>4</sup>	N/A	1.49×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by waste treatment operations and the PPF. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Disposal would take place in an Integrated Disposal Facility. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>f</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent; a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>g</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>h</sup> Derived from clean closure of all single-shell tank farms. Disposal of PPF glass and retired PPF melters would take place in an Integrated Disposal Facility.

<sup>i</sup> Rubble, soil, and equipment would be generated by clean closure of all single-shell tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; N/A=not applicable; NR=not reported; PPF=Preprocessing Facility.

**Source:** SAIC 2011.

**Table D-54. Tank Closure Alternative 6A, Base Case, Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>5</sup>	0.1	1.83	0.1	7.08×10 <sup>4</sup>	0.1	8.41×10 <sup>4</sup>	0.1	5.97×10 <sup>3</sup>	0.1	2.95×10 <sup>1</sup>	0.1	2.40×10 <sup>-3</sup>	0.1	3.45×10 <sup>3</sup>	0.1	1.68	0.1	1.11×10 <sup>-3</sup>	0.1
IHLW glass <sup>c</sup>	5.96×10 <sup>5</sup>	99.7	0	0.0	0	0.0	8.42×10 <sup>4</sup>	100.1	5.95×10 <sup>5</sup>	99.7	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>d</sup>	4.53×10 <sup>1</sup>	0.0	5.60	0.3	9.16×10 <sup>6</sup>	12.9	4.64	0.0	4.14×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>e</sup>	1.98×10 <sup>3</sup>	0.3	1.78×10 <sup>3</sup>	97.4	0	0.0	2.50×10 <sup>2</sup>	0.3	2.40×10 <sup>3</sup>	0.4	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Total<sup>f</sup></b>	<b>5.99×10<sup>5</sup></b>	<b>100.1</b>	<b>1.78×10<sup>3</sup></b>	<b>97.8</b>	<b>9.23×10<sup>6</sup></b>	<b>13.0</b>	<b>8.45×10<sup>4</sup></b>	<b>100.5</b>	<b>5.98×10<sup>5</sup></b>	<b>100.2</b>	<b>2.95×10<sup>1</sup></b>	<b>0.1</b>	<b>2.40×10<sup>-3</sup></b>	<b>0.1</b>	<b>3.45×10<sup>3</sup></b>	<b>0.1</b>	<b>1.68</b>	<b>0.1</b>	<b>1.11×10<sup>-3</sup></b>	<b>0.1</b>
<b>Other Inventory<sup>g</sup></b>																				
PPF glass and retired PPF melters <sup>h</sup>	8.52×10 <sup>3</sup>	N/A	0	N/A	0	N/A	2.72×10 <sup>2</sup>	N/A	1.62×10 <sup>2</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A
Rubble, soil, and equipment <sup>i</sup>	4.10×10 <sup>3</sup>	N/A	2.40	N/A	2.83×10 <sup>5</sup>	N/A	3.47×10 <sup>2</sup>	N/A	7.66×10 <sup>3</sup>	N/A	1.47	N/A	1.20×10 <sup>-4</sup>	N/A	6.29×10 <sup>2</sup>	N/A	2.82×10 <sup>-1</sup>	N/A	5.54×10 <sup>-5</sup>	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>j</sup>	NR	N/A	1.83×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	3.78×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant and via supplemental treatment processes.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations and the PPF. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Disposal would take place in an Integrated Disposal Facility. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>f</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>g</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>h</sup> Derived from clean closure of all single-shell tank farms. Disposal of PPF glass and retired PPF melters would take place in an Integrated Disposal Facility.

<sup>i</sup> Rubble, soil, and equipment would be generated by clean closure of all single-shell tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; Kg=kilograms; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; PPF=Preprocessing Facility; TCP=trichlorophenol.

**Source:** SAIC 2011.

**Table D-55. Tank Closure Alternative 6A, Option Case, Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-2</sup>	0.1	4.58×10 <sup>4</sup>	0.1	3.12	0.1	1.21×10 <sup>1</sup>	0.1	9.38×10 <sup>-1</sup>	0.1	1.41×10 <sup>-1</sup>	0.1	8.14×10 <sup>1</sup>	0.1	5.05×10 <sup>4</sup>	0.1	2.97×10 <sup>1</sup>	0.1
IHLW glass <sup>c</sup>	9.64	20.0	4.57×10 <sup>7</sup>	99.7	0	0.0	0	0.0	9.36×10 <sup>2</sup>	99.8	1.41×10 <sup>2</sup>	99.9	8.14×10 <sup>4</sup>	99.9	4.98×10 <sup>7</sup>	98.5	2.96×10 <sup>4</sup>	99.6
ETF-generated solid secondary waste <sup>d</sup>	3.43×10 <sup>1</sup>	71.2	4.65×10 <sup>-1</sup>	0.0	8.69	0.3	0	0.0	4.16×10 <sup>-2</sup>	0.0	5.26×10 <sup>-2</sup>	0.0	7.04×10 <sup>-4</sup>	0.0	6.49	0.0	8.80×10 <sup>1</sup>	0.3
Solid secondary waste <sup>e</sup>	4.75	9.9	1.98×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.74	0.4	2.90×10 <sup>-1</sup>	0.2	2.03×10 <sup>2</sup>	0.2	7.84×10 <sup>5</sup>	1.6	4.40×10 <sup>2</sup>	1.5
<b>Total<sup>f</sup></b>	<b>4.88×10<sup>1</sup></b>	<b>101.2</b>	<b>4.60×10<sup>7</sup></b>	<b>100.2</b>	<b>1.18×10<sup>1</sup></b>	<b>0.4</b>	<b>1.21×10<sup>1</sup></b>	<b>0.1</b>	<b>9.41×10<sup>2</sup></b>	<b>100.3</b>	<b>1.41×10<sup>2</sup></b>	<b>100.2</b>	<b>8.16×10<sup>4</sup></b>	<b>100.2</b>	<b>5.06×10<sup>7</sup></b>	<b>100.1</b>	<b>3.02×10<sup>4</sup></b>	<b>101.5</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>g</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
PPF glass and retired PPF melters <sup>h</sup>	1.23×10 <sup>-1</sup>	N/A	2.20×10 <sup>5</sup>	N/A	0	N/A	0	N/A	1.69×10 <sup>1</sup>	N/A	2.67	N/A	9.39×10 <sup>2</sup>	N/A	7.84×10 <sup>4</sup>	N/A	3.44×10 <sup>2</sup>	N/A
Rubble, soil, and equipment <sup>i</sup>	4.96×10 <sup>-1</sup>	N/A	2.72×10 <sup>5</sup>	N/A	3.35×10 <sup>1</sup>	N/A	3.79×10 <sup>3</sup>	N/A	1.42×10 <sup>1</sup>	N/A	1.6	N/A	4.85×10 <sup>2</sup>	N/A	1.39×10 <sup>5</sup>	N/A	2.70×10 <sup>2</sup>	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>j</sup>	4.86×10 <sup>1</sup>	N/A	4.72×10 <sup>4</sup>	N/A	3.15×10 <sup>3</sup>	N/A	1.50×10 <sup>4</sup>	N/A	4.76×10 <sup>-1</sup>	N/A	NR	N/A	4.09×10 <sup>1</sup>	N/A	3.58×10 <sup>4</sup>	N/A	1.50×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by waste treatment operations and the PPF. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Disposal would take place in an Integrated Disposal Facility. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>f</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>g</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>h</sup> Derived from clean closure of all single-shell tank farms and six sets of contiguous cribs and trenches (ditches). Disposal of PPF glass and retired PPF melters would take place in an Integrated Disposal Facility.

<sup>i</sup> Rubble, soil, and equipment would be generated by clean closure of all single-shell tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; N/A=not applicable; NR=not reported; PPF=Preprocessing Facility.

**Source:** SAIC 2011.

**Table D-56. Tank Closure Alternative 6A, Option Case, Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>2</sup>	0.1	1.83	0.1	7.08×10 <sup>4</sup>	0.1	8.41×10 <sup>1</sup>	0.1	5.97×10 <sup>2</sup>	0.1	2.95×10 <sup>1</sup>	0.1	2.40×10 <sup>-3</sup>	0.1	3.45×10 <sup>3</sup>	0.1	1.68	0.1	1.11×10 <sup>-3</sup>	0.1
IHLW glass <sup>c</sup>	5.96×10 <sup>5</sup>	99.7	0	0.0	0	0.0	8.42×10 <sup>4</sup>	100.1	5.95×10 <sup>5</sup>	99.7	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>d</sup>	5.65×10 <sup>1</sup>	0.0	5.79	0.3	1.51×10 <sup>7</sup>	21.3	4.64	0.0	4.19×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>e</sup>	2.47×10 <sup>3</sup>	0.4	1.84×10 <sup>3</sup>	100.6	0	0.0	2.50×10 <sup>2</sup>	0.3	2.43×10 <sup>3</sup>	0.4	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Total<sup>f</sup></b>	<b>5.99×10<sup>5</sup></b>	<b>100.2</b>	<b>1.84×10<sup>3</sup></b>	<b>101.1</b>	<b>1.51×10<sup>7</sup></b>	<b>21.4</b>	<b>8.45×10<sup>4</sup></b>	<b>100.5</b>	<b>5.98×10<sup>5</sup></b>	<b>100.2</b>	<b>2.95×10<sup>4</sup></b>	<b>0.1</b>	<b>2.40×10<sup>-3</sup></b>	<b>0.1</b>	<b>3.45×10<sup>3</sup></b>	<b>0.1</b>	<b>1.68</b>	<b>0.1</b>	<b>1.11×10<sup>-3</sup></b>	<b>0.1</b>
<b>Other Inventory<sup>g</sup></b>																				
PPF glass and retired PPF melters <sup>h</sup>	1.60×10 <sup>5</sup>	N/A	0	N/A	0	N/A	3.11×10 <sup>2</sup>	N/A	2.28×10 <sup>4</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A
Rubble, soil, and equipment <sup>i</sup>	3.69×10 <sup>4</sup>	N/A	1.59×10 <sup>1</sup>	N/A	1.04×10 <sup>7</sup>	N/A	3.58×10 <sup>2</sup>	N/A	9.24×10 <sup>3</sup>	N/A	1.47	N/A	1.20×10 <sup>-4</sup>	N/A	6.29×10 <sup>2</sup>	N/A	2.82×10 <sup>-1</sup>	N/A	5.54×10 <sup>-5</sup>	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>j</sup>	NR	N/A	1.83×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	3.78×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>e</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations and the PPF. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Disposal would take place in an Integrated Disposal Facility. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>f</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>g</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>h</sup> Derived from clean closure of all single-shell tank farms and the six sets of contiguous cribs and trenches (ditches). Disposal of PPF glass and retired PPF melters would take place in an Integrated Disposal Facility.

<sup>i</sup> Rubble, soil, and equipment would be generated by clean closure of all single-shell tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. Disposal would take place in the River Protection Project Disposal Facility. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; Kg=kilograms; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; PPF=Preprocessing Facility; TCP=trichlorophenol.

**Source:** SAIC 2011.

Under Tank Closure Alternative 6B, Base Case, represented in Figure D–11, all waste streams, including those resulting from clean closure of the SSTs, would be managed as IHLW glass. Under Tank Closure Alternative 6B, Option Case, the six sets of contiguous cribs and trenches (ditches) described in Section D.1.5 would be added to the Alternative 6B, Base Case, inventory. Material balance summaries for these two cases are presented in Tables D–57 through D–60. However, under Tank Closure Alternative 6B, the tank waste would be treated in a shorter period of time than under Tank Closure Alternative 6A due to the use of LAW melters. The ILAW glass would be managed as IHLW glass.

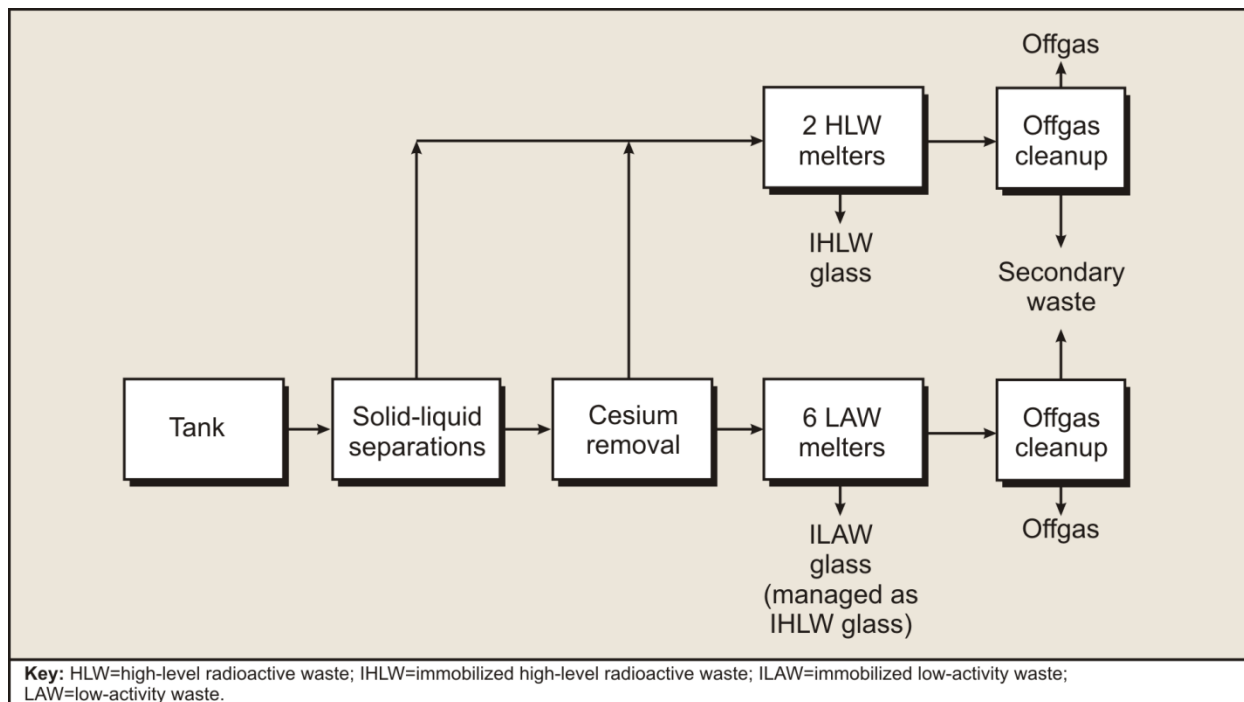


Figure D–11. Tank Closure Alternatives 6B and 6C Flowsheet

**Table D-57. Tank Closure Alternative 6B, Base Case, Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-2</sup>	0.1	4.58×10 <sup>4</sup>	0.1	3.12	0.1	1.21×10 <sup>1</sup>	0.1	9.38×10 <sup>-1</sup>	0.1	1.41×10 <sup>-1</sup>	0.1	8.14×10 <sup>1</sup>	0.1	5.05×10 <sup>4</sup>	0.1	2.97×10 <sup>1</sup>	0.1
IHLW glass <sup>c</sup>	7.05×10 <sup>-3</sup>	0.0	4.53×10 <sup>7</sup>	98.7	0	0.0	0	0.0	8.81×10 <sup>2</sup>	93.9	1.41×10 <sup>2</sup>	99.9	8.14×10 <sup>4</sup>	99.9	4.97×10 <sup>7</sup>	98.5	2.49×10 <sup>2</sup>	0.8
ILAW glass and retired LAW melter <sup>d</sup>	9.65	20.0	4.49×10 <sup>5</sup>	1.0	0	0.0	0	0.0	5.52×10 <sup>1</sup>	5.9	8.43×10 <sup>-3</sup>	0.0	1.47	0.0	2.33×10 <sup>3</sup>	0.0	2.91×10 <sup>4</sup>	97.8
ETF-generated solid secondary waste <sup>e</sup>	3.41×10 <sup>1</sup>	70.7	4.65×10 <sup>-1</sup>	0.0	8.63	0.3	0	0.0	4.11×10 <sup>-2</sup>	0.0	5.18×10 <sup>-2</sup>	0.0	6.96×10 <sup>-4</sup>	0.0	6.49	0.0	8.74×10 <sup>1</sup>	0.3
Solid secondary waste <sup>f</sup>	4.72	9.8	1.98×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.70	0.4	2.86×10 <sup>-1</sup>	0.2	2.00×10 <sup>2</sup>	0.2	7.84×10 <sup>5</sup>	1.6	4.37×10 <sup>2</sup>	1.5
<b>Total<sup>g</sup></b>	<b>4.85×10<sup>1</sup></b>	<b>100.6</b>	<b>4.60×10<sup>7</sup></b>	<b>100.3</b>	<b>1.18×10<sup>1</sup></b>	<b>0.4</b>	<b>1.21×10<sup>1</sup></b>	<b>0.1</b>	<b>9.41×10<sup>2</sup></b>	<b>100.3</b>	<b>1.41×10<sup>2</sup></b>	<b>100.2</b>	<b>8.16×10<sup>4</sup></b>	<b>100.2</b>	<b>5.06×10<sup>7</sup></b>	<b>100.1</b>	<b>2.99×10<sup>4</sup></b>	<b>100.5</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>h</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
PPF glass and retired PPF melter <sup>i</sup>	5.31×10 <sup>-2</sup>	N/A	2.03×10 <sup>5</sup>	N/A	0	N/A	0	N/A	7.66	N/A	5.14×10 <sup>-1</sup>	N/A	3.45×10 <sup>1</sup>	N/A	6.57×10 <sup>4</sup>	N/A	1.35×10 <sup>2</sup>	N/A
Rubble, soil, and equipment <sup>j</sup>	3.43×10 <sup>-1</sup>	N/A	2.62×10 <sup>5</sup>	N/A	2.36×10 <sup>1</sup>	N/A	2.17×10 <sup>2</sup>	N/A	1.02×10 <sup>1</sup>	N/A	6.68×10 <sup>-1</sup>	N/A	9.92×10 <sup>1</sup>	N/A	1.32×10 <sup>5</sup>	N/A	1.78×10 <sup>2</sup>	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>k</sup>	4.81×10 <sup>1</sup>	N/A	4.71×10 <sup>4</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	4.71×10 <sup>-1</sup>	N/A	NR	N/A	4.04×10 <sup>1</sup>	N/A	3.56×10 <sup>4</sup>	N/A	1.48×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melter.

<sup>d</sup> Although processed as ILAW glass, glass and retired melter would be managed and disposed of as IHLW glass.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations and the PPF. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Disposal would take place in an Integrated Disposal Facility. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>h</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>i</sup> Derived from clean closure of all single-shell tank farms. Disposal of PPF glass and retired PPF melter would take place in an Integrated Disposal Facility.

<sup>j</sup> Rubble, soil, and equipment would be generated by clean closure of all single-shell tank farms and six sets of contiguous cribs and trenches (ditches). Disposal would take place in the River Protection Project Disposal Facility.

<sup>k</sup> Includes the air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported; PPF=Preprocessing Facility.

**Source:** SAIC 2011.



**Table D-58. Tank Closure Alternative 6B, Base Case, Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>2</sup>	0.1	1.83	0.1	7.08×10 <sup>4</sup>	0.1	8.41×10 <sup>1</sup>	0.1	5.97×10 <sup>2</sup>	0.1	2.95×10 <sup>1</sup>	0.1	2.40×10 <sup>-3</sup>	0.1	3.45×10 <sup>3</sup>	0.1	1.68	0.1	1.11×10 <sup>-3</sup>	0.1
IHLW glass <sup>c</sup>	1.37×10 <sup>5</sup>	22.9	0	0.0	0	0.0	7.52×10 <sup>4</sup>	89.4	5.57×10 <sup>5</sup>	93.4	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters <sup>d</sup>	4.60×10 <sup>5</sup>	76.9	0	0.0	0	0.0	8.96×10 <sup>3</sup>	10.7	3.78×10 <sup>4</sup>	6.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>e</sup>	4.53×10 <sup>1</sup>	0.0	5.60	0.3	9.16×10 <sup>6</sup>	12.9	4.64	0.0	4.14×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>f</sup>	1.98×10 <sup>3</sup>	0.3	1.78×10 <sup>3</sup>	97.4	0	0.0	2.50×10 <sup>2</sup>	0.3	2.40×10 <sup>3</sup>	0.4	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Total<sup>g</sup></b>	<b>6.00×10<sup>5</sup></b>	<b>100.3</b>	<b>1.78×10<sup>3</sup></b>	<b>97.8</b>	<b>9.23×10<sup>6</sup></b>	<b>13.0</b>	<b>8.45×10<sup>4</sup></b>	<b>100.5</b>	<b>5.98×10<sup>5</sup></b>	<b>100.2</b>	<b>2.95×10<sup>1</sup></b>	<b>0.1</b>	<b>2.40×10<sup>-3</sup></b>	<b>0.1</b>	<b>3.45×10<sup>3</sup></b>	<b>0.1</b>	<b>1.68</b>	<b>0.1</b>	<b>1.11×10<sup>-3</sup></b>	<b>0.1</b>
<b>Other Inventory<sup>h</sup></b>																				
PPF glass and retired PPF melters <sup>i</sup>	8.33×10 <sup>3</sup>	N/A	0	N/A	0	N/A	2.66×10 <sup>2</sup>	N/A	1.59×10 <sup>4</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A
Rubble, soil, and equipment <sup>j</sup>	4.10×10 <sup>3</sup>	N/A	2.40	N/A	2.83×10 <sup>5</sup>	N/A	3.47×10 <sup>2</sup>	N/A	7.66×10 <sup>3</sup>	N/A	1.47	N/A	1.20×10 <sup>-4</sup>	N/A	6.29×10 <sup>2</sup>	N/A	2.82×10 <sup>-1</sup>	N/A	5.54×10 <sup>-5</sup>	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>k</sup>	NR	N/A	1.82×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	6.15×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Although processed as ILAW glass, glass and retired melters would be managed and disposed of as IHLW glass.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations and the PPF. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Disposal would take place in an Integrated Disposal Facility. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>h</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>i</sup> Derived from clean closure of all single-shell tank farms. Disposal of PPF glass and retired PPF melters would take place in an Integrated Disposal Facility.

<sup>j</sup> Rubble, soil, and equipment would be generated by clean closure of all single-shell tank farms and six sets of contiguous cribs and trenches (ditches). Disposal would take place in the River Protection Project Disposal Facility.

<sup>k</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; PPF=Preprocessing Facility; TCP=trichlorophenol.

**Source:** SAIC 2011.

**Table D-59. Tank Closure Alternative 6B, Option Case, Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99	
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI
<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A
<b>Tank Closure Waste Inventory</b>																		
Tank residual waste <sup>b</sup>	4.82×10 <sup>-2</sup>	0.1	4.58×10 <sup>4</sup>	0.1	3.12	0.1	1.21×10 <sup>1</sup>	0.1	9.38×10 <sup>-1</sup>	0.1	1.41×10 <sup>-1</sup>	0.1	8.14×10 <sup>1</sup>	0.1	5.05×10 <sup>4</sup>	0.1	2.97×10 <sup>1</sup>	0.1
IHLW glass <sup>c</sup>	7.05×10 <sup>-3</sup>	0.0	4.53×10 <sup>7</sup>	98.7	0	0.0	0	0.0	8.81×10 <sup>2</sup>	93.9	1.41×10 <sup>2</sup>	99.9	8.14×10 <sup>4</sup>	99.9	4.97×10 <sup>7</sup>	98.5	2.49×10 <sup>2</sup>	0.8
ILAW glass and retired LAW melters <sup>d</sup>	9.65	20.0	4.49×10 <sup>5</sup>	1.0	0	0.0	0	0.0	5.52×10 <sup>1</sup>	5.9	8.43×10 <sup>-3</sup>	0.0	1.47	0.0	2.33×10 <sup>3</sup>	0.0	2.91×10 <sup>4</sup>	97.8
ETF-generated solid secondary waste <sup>e</sup>	3.43×10 <sup>1</sup>	71.2	4.65×10 <sup>-1</sup>	0.0	8.69	0.3	0	0.0	4.16×10 <sup>-2</sup>	0.0	5.26×10 <sup>-2</sup>	0.0	7.04×10 <sup>-4</sup>	0.0	6.49	0.0	8.80×10 <sup>1</sup>	0.3
Solid secondary waste <sup>f</sup>	4.75	9.9	1.98×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.74	0.4	2.90×10 <sup>-1</sup>	0.2	2.03×10 <sup>2</sup>	0.2	7.84×10 <sup>5</sup>	1.6	4.40×10 <sup>2</sup>	1.5
<b>Total<sup>g</sup></b>	<b>4.88×10<sup>1</sup></b>	<b>101.2</b>	<b>4.60×10<sup>7</sup></b>	<b>100.3</b>	<b>1.18×10<sup>1</sup></b>	<b>0.4</b>	<b>1.21×10<sup>1</sup></b>	<b>0.1</b>	<b>9.41×10<sup>2</sup></b>	<b>100.3</b>	<b>1.41×10<sup>2</sup></b>	<b>100.2</b>	<b>8.16×10<sup>4</sup></b>	<b>100.2</b>	<b>5.06×10<sup>7</sup></b>	<b>100.1</b>	<b>2.99×10<sup>4</sup></b>	<b>100.5</b>
<b>Other Inventory</b>																		
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A
Cesium and strontium capsules <sup>h</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A
PPF glass and retired PPF melters <sup>i</sup>	1.23×10 <sup>-1</sup>	N/A	2.19×10 <sup>5</sup>	N/A	0	N/A	0	N/A	1.68×10 <sup>1</sup>	N/A	2.66	N/A	9.34×10 <sup>2</sup>	N/A	7.80×10 <sup>4</sup>	N/A	3.42×10 <sup>2</sup>	N/A
Rubble, soil, and equipment <sup>j</sup>	4.96×10 <sup>-1</sup>	N/A	2.72×10 <sup>5</sup>	N/A	3.35×10 <sup>1</sup>	N/A	3.79×10 <sup>3</sup>	N/A	1.42×10 <sup>1</sup>	N/A	1.60×10 <sup>1</sup>	N/A	4.85×10 <sup>2</sup>	N/A	1.39×10 <sup>5</sup>	N/A	2.70×10 <sup>2</sup>	N/A
<b>Air Emissions</b>																		
Treatment air emissions <sup>k</sup>	4.82×10 <sup>1</sup>	N/A	4.71×10 <sup>4</sup>	N/A	3.13×10 <sup>3</sup>	N/A	1.49×10 <sup>4</sup>	N/A	4.72×10 <sup>-1</sup>	N/A	NR	N/A	4.06×10 <sup>1</sup>	N/A	3.56×10 <sup>4</sup>	N/A	1.48×10 <sup>1</sup>	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Although processed as ILAW glass, glass and retired melters would be managed and disposed of as IHLW glass.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations and the PPF. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Disposal would take place in an Integrated Disposal Facility. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>h</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>i</sup> Derived from clean closure of all single-shell tank farms and the six sets of contiguous cribs and trenches (ditches). Disposal of PPF glass and retired PPF melters would take place in an Integrated Disposal Facility.

<sup>j</sup> Rubble, soil, and equipment would be generated by clean closure of all single-shell tank farms and six sets of contiguous cribs and trenches (ditches). Disposal would take place in the River Protection Project Disposal Facility.

<sup>k</sup> Includes the air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported; PPF=Preprocessing Facility.

**Source:** SAIC 2011.

**Table D-60. Tank Closure Alternative 6B, Option Case, Chemical Constituents of Potential Concern Balance**

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>2</sup>	0.1	1.83	0.1	7.08×10 <sup>4</sup>	0.1	8.41×10 <sup>1</sup>	0.1	5.97×10 <sup>2</sup>	0.1	2.95×10 <sup>1</sup>	0.1	2.40×10 <sup>-3</sup>	0.1	3.45×10 <sup>3</sup>	0.1	1.68	0.1	1.11×10 <sup>-3</sup>	0.1
IHLW glass <sup>c</sup>	1.37×10 <sup>5</sup>	22.9	0	0.0	0	0.0	7.52×10 <sup>4</sup>	89.4	5.57×10 <sup>5</sup>	93.4	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters <sup>d</sup>	4.60×10 <sup>5</sup>	76.9	0	0.0	0	0.0	8.69×10 <sup>3</sup>	10.7	3.78×10 <sup>4</sup>	6.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>e</sup>	5.65×10 <sup>1</sup>	0.0	5.79	0.3	1.51×10 <sup>7</sup>	21.3	4.64	0.0	4.19×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>f</sup>	2.47×10 <sup>3</sup>	0.4	1.84×10 <sup>3</sup>	100.6	0	0.0	2.50×10 <sup>2</sup>	0.3	2.43×10 <sup>3</sup>	0.4	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Total<sup>g</sup></b>	<b>6.01×10<sup>5</sup></b>	<b>100.4</b>	<b>1.84×10<sup>3</sup></b>	<b>101.1</b>	<b>1.51×10<sup>7</sup></b>	<b>21.4</b>	<b>8.45×10<sup>4</sup></b>	<b>100.5</b>	<b>5.98×10<sup>5</sup></b>	<b>100.2</b>	<b>2.95×10<sup>1</sup></b>	<b>0.1</b>	<b>2.40×10<sup>-3</sup></b>	<b>0.1</b>	<b>3.45×10<sup>3</sup></b>	<b>0.1</b>	<b>1.68</b>	<b>0.1</b>	<b>1.11×10<sup>-3</sup></b>	<b>0.1</b>
<b>Other Inventory<sup>h</sup></b>																				
PPF glass and retired PPF melters <sup>i</sup>	1.58×10 <sup>5</sup>	N/A	0	N/A	0	N/A	3.09×10 <sup>2</sup>	N/A	2.26×10 <sup>4</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A
Rubble, soil, and equipment <sup>j</sup>	3.69×10 <sup>4</sup>	N/A	1.59×10 <sup>1</sup>	N/A	1.04×10 <sup>7</sup>	N/A	3.58×10 <sup>2</sup>	N/A	9.24×10 <sup>3</sup>	N/A	1.47	N/A	1.20×10 <sup>-4</sup>	N/A	6.29×10 <sup>2</sup>	N/A	2.82×10 <sup>-1</sup>	N/A	5.54×10 <sup>-5</sup>	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>k</sup>	NR	N/A	1.81×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	6.17×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.9 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Although processed as ILAW glass, glass and retired melters would be managed and disposed of as IHLW glass.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations and the PPF. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Disposal would take place in an Integrated Disposal Facility. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>h</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>i</sup> Derived from clean closure of all single-shell tank farms and the six sets of contiguous cribs and trenches. Disposal of PPF glass and retired PPF melters would take place in an Integrated Disposal Facility.

<sup>j</sup> Rubble, soil, and equipment would be generated by clean closure of all single-shell tank farms and six sets of contiguous cribs and trenches (ditches). Disposal would take place in the River Protection Project Disposal Facility.

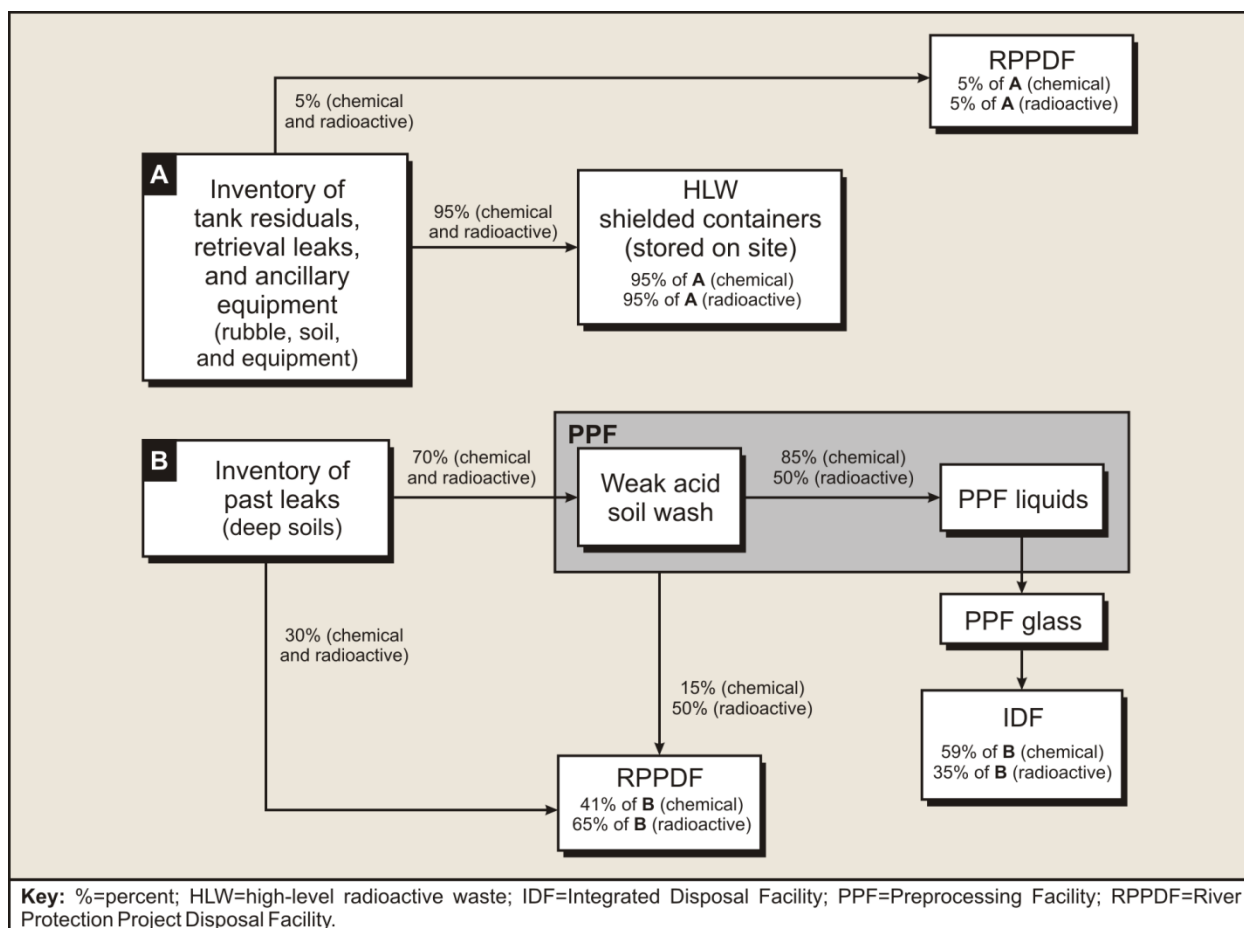
<sup>k</sup> Includes the air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; PPF=Preprocessing Facility; TCP=trichlorophenol.

**Source:** SAIC 2011.

Under Tank Closure Alternatives 6A and 6B, Base Cases (see Figure D–12 for a simplified flowsheet and Tables D–61 through D–64 for inventories), all 12 SST farms would undergo clean closure. Tank residual waste; materials; and highly contaminated rubble, soil, and equipment from tank and ancillary equipment removal activities would be packaged in shielded boxes, stored on site, and managed as IHLW glass. This waste represents 95 percent of the radioactive and chemical constituent inventory remaining in the tanks and the contaminated rubble, soil, and ancillary equipment resulting from leaks associated with waste retrieval. The waste would be managed and stored as IHLW in shielded containers on site. Only 5 percent of the inventory would be packaged and sent directly to the RPPDF.



**Figure D–12. Tank Closure Alternatives 6A and 6B, Base Cases, Clean Closure of Single-Shell Tank Farms Flowsheet**

**Table D–61. Tank Closure Alternative 6A, Base Case, Radioactive Constituents of Potential Concern Inventory from Clean Closure of the SST Farms (curies)**

Analyte	MLLW <sup>a</sup>	PPF Glass and Retired PPF Melters <sup>b</sup>
Hydrogen-3 (tritium)	2.17×10 <sup>2</sup>	0
Carbon-14	2.36×10 <sup>1</sup>	0
Strontium-90	1.32×10 <sup>5</sup>	6.71×10 <sup>4</sup>
Technetium-99	1.78×10 <sup>2</sup>	1.38×10 <sup>2</sup>
Iodine-129	3.43×10 <sup>-1</sup>	5.43×10 <sup>-2</sup>
Cesium-137	2.62×10 <sup>5</sup>	2.08×10 <sup>5</sup>
Uranium-233, -234, -235, -238	1.02×10 <sup>1</sup>	7.83
Neptunium-237	6.68×10 <sup>-1</sup>	5.26×10 <sup>-1</sup>
Plutonium-239, -240	9.92×10 <sup>1</sup>	3.53×10 <sup>1</sup>

<sup>a</sup> Represents 5 percent of the contaminant inventory from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 65 percent of the contaminant inventory in deep soils for the SST farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 50 percent of the highly contaminated deep soil contaminant inventory that would be removed during clean closure of all of the SST farms and treated in the PPF, resulting in PPF glass and retired PPF melters. Disposal would take place in an Integrated Disposal Facility.

**Key:** MLLW=mixed low-level radioactive waste; PPF=Preprocessing Facility; SST=single-shell tank.

**Source:** SAIC 2011.

**Table D–62. Tank Closure Alternative 6A, Base Case, Chemical Constituents of Potential Concern Inventory from Clean Closure of the SST Farms (kilograms)**

Analyte	MLLW <sup>a</sup>	PPF Glass and Retired PPF Melters <sup>b</sup>
Chromium	4.10×10 <sup>3</sup>	8.52×10 <sup>3</sup>
Mercury	2.40	0
Nitrate	2.83×10 <sup>5</sup>	0
Lead	3.47×10 <sup>2</sup>	2.72×10 <sup>2</sup>
Total uranium	7.66×10 <sup>3</sup>	1.62×10 <sup>4</sup>
Acetonitrile	1.47	0
Benzene	1.20×10 <sup>-4</sup>	0
Butanol (n-butyl alcohol)	6.29×10 <sup>2</sup>	0
Polychlorinated biphenyls	2.82×10 <sup>-1</sup>	0
2,4,6-Trichlorophenol	5.54×10 <sup>-5</sup>	0

<sup>a</sup> Represents 5 percent of the contaminant inventory from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 40.5 percent of the contaminant inventory in deep soils for the SST farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 85 percent of the highly contaminated deep soil contaminant inventory that would be removed during clean closure of all of the SST farms and treated in the PPF, resulting in PPF glass and retired PPF melters. Disposal would take place in an Integrated Disposal Facility.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** MLLW=mixed low-level radioactive waste; PPF=Preprocessing Facility; SST=single-shell tank.

**Source:** SAIC 2011.

**Table D–63. Tank Closure Alternative 6B, Base Case, Radioactive Constituents of Potential Concern Inventory from Clean Closure of the SST Farms (curies)**

Analyte	MLLW <sup>a</sup>	PPF Glass and Retired PPF Melters <sup>b</sup>
Hydrogen-3 (tritium)	2.17×10 <sup>2</sup>	0
Carbon-14	2.36×10 <sup>1</sup>	0
Strontium-90	1.32×10 <sup>5</sup>	6.57×10 <sup>4</sup>
Technetium-99	1.78×10 <sup>2</sup>	1.35×10 <sup>2</sup>
Iodine-129	3.43×10 <sup>-1</sup>	5.31×10 <sup>-2</sup>
Cesium-137	2.62×10 <sup>5</sup>	2.03×10 <sup>5</sup>
Uranium-233, -234, -235, -238	1.02×10 <sup>1</sup>	7.66
Neptunium-237	6.68×10 <sup>-1</sup>	5.14×10 <sup>-1</sup>
Plutonium-239, -240	9.92×10 <sup>1</sup>	3.45×10 <sup>1</sup>

<sup>a</sup> Represents 5 percent of the contaminant inventory from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 65 percent of the contaminant inventory in deep soils for the SST farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 50 percent of the highly contaminated deep soil contaminant inventory that would be removed during clean closure of all of the SST farms and treated in the PPF, resulting in PPF glass and retired PPF melters. Disposal would take place in an Integrated Disposal Facility.

**Key:** MLLW=mixed low-level radioactive waste; PPF=Preprocessing Facility; SST=single-shell tank.

**Source:** SAIC 2011.

**Table D–64. Tank Closure Alternative 6B, Base Case, Chemical Constituents of Potential Concern Inventory from Clean Closure of the SST Farms (kilograms)**

Analyte	MLLW <sup>a</sup>	PPF Glass and Retired PPF Melters <sup>b</sup>
Chromium	4.10×10 <sup>3</sup>	8.33×10 <sup>3</sup>
Mercury	2.40	0
Nitrate	2.83×10 <sup>5</sup>	0
Lead	3.47×10 <sup>2</sup>	2.66×10 <sup>2</sup>
Total uranium	7.66×10 <sup>3</sup>	1.59×10 <sup>4</sup>
Acetonitrile	1.47	0
Benzene	1.20×10 <sup>-4</sup>	0
Butanol (n-butyl alcohol)	6.29×10 <sup>2</sup>	0
Polychlorinated biphenyls	2.82×10 <sup>-1</sup>	0
2,4,6-Trichlorophenol	5.54×10 <sup>-5</sup>	0

<sup>a</sup> Represents 5 percent of the contaminant inventory from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 40.5 percent of the contaminant inventory in deep soils for the SST farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 85 percent of the highly contaminated deep soil contaminant inventory that would be removed during clean closure of all of the SST farms and treated in the PPF, resulting in PPF glass and retired PPF melters. Disposal would take place in an Integrated Disposal Facility.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** MLLW=mixed low-level radioactive waste; PPF=Preprocessing Facility; SST=single-shell tank.

**Source:** SAIC 2011.

Soils contaminated from past tank leaks would be removed and managed as described above for Tank Closure Alternative 4. Moderately contaminated soil containing 30 percent of the radionuclide and chemical inventories would be packaged for direct disposal in the RPPDF. Heavily contaminated soil containing 70 percent of the radionuclide and chemical inventories would be processed through soil washing in the PPF. In the PPF, 85 percent of the chemical constituents and 50 percent of the radioactive constituents would be removed from the soil in a liquid waste stream. This liquid waste stream would be further processed in the PPF into PPF glass for onsite storage in an Integrated Disposal Facility (IDF). This liquid waste stream represents 35 percent of the radioactive constituents and 59 percent of the chemical constituents from past leaks. The remaining inventory of contaminants, 35 percent of the initial contaminated soil radionuclide inventory (50 percent of the 70 percent sent to the PPF), and 11 percent of the initial contaminated chemical inventory (15 percent of the 70 percent sent to the PPF) would be managed as MLLW generated by PPF operations. Thus, a total of 65 percent of the inventory of radioactive constituents and 41 percent of the inventory of chemical constituents from past tank leaks would be disposed of in the RPPDF.

The inventories associated with the soil disposed of on site are determined as shown in the following equations:

$$M_{\text{rad}_{\text{soil}}} = 0.65 \times M_{\text{pleak}}$$

and

$$M_{\text{chem}_{\text{soil}}} = 0.41 \times M_{\text{pleak}}$$

where:

$M_{\text{rad}_{\text{soil}}}$  = inventory of radioactive constituents in contaminated soil disposed of on site

$M_{\text{pleak}}$  = inventory of radioactive or chemical constituents from past leaks

$M_{\text{chem}_{\text{soil}}}$  = inventory of chemical constituents in contaminated soil disposed of on site

For the Option Cases under Tank Closure Alternatives 6A and 6B, the soils contaminated from intentional discharges to the six sets of cribs and trenches (ditches) would be added to the inventories from the 12 SST farms. Tables D-65 through D-68 show estimates of radioactive and chemical constituent inventories resulting from clean closure of the SST Farms and the six sets of cribs and trenches (ditches), respectively.

**Table D–65. Tank Closure Alternative 6A, Option Case, Radioactive Constituents of Potential Concern Inventory from Clean Closure of the SST Farms and Six Sets of Cribs and Trenches (Ditches) (curies)**

Analyte	MLLW <sup>a</sup>	PPF Glass and Retired PPF Melters <sup>b</sup>
Hydrogen-3 (tritium)	3.79×10 <sup>3</sup>	0
Carbon-14	3.35×10 <sup>1</sup>	0
Strontium-90	1.39×10 <sup>5</sup>	7.84×10 <sup>4</sup>
Technetium-99	2.70×10 <sup>2</sup>	3.44×10 <sup>2</sup>
Iodine-129	4.96×10 <sup>-1</sup>	1.23×10 <sup>-1</sup>
Cesium-137	2.72×10 <sup>5</sup>	2.20×10 <sup>5</sup>
Uranium-233, -234, -235, -238	1.42×10 <sup>1</sup>	1.69×10 <sup>1</sup>
Neptunium-237	1.60	2.67
Plutonium-239, -240	4.85×10 <sup>2</sup>	9.39×10 <sup>2</sup>

<sup>a</sup> Represents 5 percent of the contaminant inventory from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 65 percent of the contaminant inventory in deep soils for the SST farms and the six sets of contiguous cribs and trenches (ditches). Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 50 percent of the highly contaminated deep soil contaminant inventory that would be removed during clean closure of all of the SST farms and the six sets of contiguous cribs and trenches (ditches) and treated in the PPF, resulting in PPF glass and retired PPF melters. Disposal would take place in an Integrated Disposal Facility.

**Key:** MLLW=mixed low-level radioactive waste; PPF=Preprocessing Facility; SST=single-shell tank.

**Source:** SAIC 2011.

**Table D–66. Tank Closure Alternative 6A, Option Case, Chemical Constituents of Potential Concern Inventory from Clean Closure of the SST Farms and Six Sets of Cribs and Trenches (Ditches) (kilograms)**

Analyte	MLLW <sup>a</sup>	PPF Glass and Retired PPF Melters <sup>b</sup>
Chromium	3.69×10 <sup>4</sup>	1.60×10 <sup>5</sup>
Mercury	1.59×10 <sup>1</sup>	0
Nitrate	1.04×10 <sup>7</sup>	0
Lead	3.58×10 <sup>2</sup>	3.11×10 <sup>2</sup>
Total uranium	9.24×10 <sup>3</sup>	2.28×10 <sup>4</sup>
Acetonitrile	1.47	0
Benzene	1.20×10 <sup>-4</sup>	0
Butanol (n-butyl alcohol)	6.29×10 <sup>2</sup>	0
Polychlorinated biphenyls	2.82×10 <sup>-1</sup>	0
2,4,6-Trichlorophenol	5.54×10 <sup>-5</sup>	0

<sup>a</sup> Represents 5 percent of the contaminant inventory from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 40.5 percent of the contaminant inventory in deep soils for the SST farms and the six sets of contiguous cribs and trenches (ditches). Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 85 percent of the highly contaminated deep soil contaminant inventory that would be removed during clean closure of all of the SST farms and the six sets of contiguous cribs and trenches (ditches) and treated in the PPF, resulting in PPF glass and retired PPF melters. Disposal would take place in an Integrated Disposal Facility.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** MLLW=mixed low-level radioactive waste; PPF=Preprocessing Facility; SST=single-shell tank.

**Source:** SAIC 2011.



**Table D–67. Tank Closure Alternative 6B, Option Case, Radioactive Constituents of Potential Concern Inventory from Clean Closure of the SST Farms and Six Sets of Cribs and Trenches (Ditches) (curies)**

Analyte	MLLW <sup>a</sup>	PPF Glass and Retired PPF Melters <sup>b</sup>
Hydrogen-3 (tritium)	$3.79 \times 10^3$	0
Carbon-14	$3.35 \times 10^1$	0
Strontium-90	$1.39 \times 10^5$	$7.80 \times 10^4$
Technetium-99	$2.70 \times 10^2$	$3.42 \times 10^2$
Iodine-129	$4.96 \times 10^{-1}$	$1.23 \times 10^{-1}$
Cesium-137	$2.72 \times 10^5$	$2.19 \times 10^5$
Uranium-233, -234, -235, -238	$1.42 \times 10^1$	$1.68 \times 10^1$
Neptunium-237	1.60	2.66
Plutonium-239, -240	$4.85 \times 10^2$	$9.34 \times 10^2$

<sup>a</sup> Represents 5 percent of the contaminant inventory from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 65 percent of the contaminant inventory in deep soils for the SST farms and the six sets of contiguous cribs and trenches (ditches). Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 50 percent of the highly contaminated deep soil contaminant inventory that would be removed during clean closure of all of the SST farms and the six sets of contiguous cribs and trenches (ditches) and treated in the PPF, resulting in PPF glass and retired PPF melters. Disposal would take place in an Integrated Disposal Facility.

**Key:** MLLW=mixed low-level radioactive waste; PPF=Preprocessing Facility; SST=single-shell tank.

**Source:** SAIC 2011.

**Table D–68. Tank Closure Alternative 6B, Option Case, Chemical Constituents of Potential Concern Inventory from Clean Closure of the SST Farms and Six Sets of Cribs and Trenches (Ditches) (kilograms)**

Analyte	MLLW <sup>a</sup>	PPF Glass and Retired PPF Melters <sup>b</sup>
Chromium	$3.69 \times 10^4$	$1.58 \times 10^5$
Mercury	$1.59 \times 10^1$	0
Nitrate	$1.04 \times 10^7$	0
Lead	$3.58 \times 10^2$	$3.09 \times 10^2$
Total uranium	$9.24 \times 10^3$	$2.26 \times 10^4$
Acetonitrile	1.47	0
Benzene	$1.20 \times 10^{-4}$	0
Butanol (n-butyl alcohol)	$6.29 \times 10^2$	0
Polychlorinated biphenyls	$2.82 \times 10^{-1}$	0
2,4,6-Trichlorophenol	$5.54 \times 10^{-5}$	0

<sup>a</sup> Represents 5 percent of the contaminant inventory from the residual waste in the tanks, ancillary equipment, and soil contaminated by tank waste retrieval leaks and 40.5 percent of the contaminant inventory in deep soils for the SST farms and the six sets of contiguous cribs and trenches (ditches). Disposal would take place in the River Protection Project Disposal Facility.

<sup>b</sup> Represents the portion of the 85 percent of the highly contaminated deep soil contaminant inventory that would be removed during clean closure of all of the SST farms and the six sets of contiguous cribs and trenches (ditches) and treated in the PPF, resulting in PPF glass and retired PPF melters. Disposal would take place in an Integrated Disposal Facility.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** MLLW=mixed low-level radioactive waste; PPF=Preprocessing Facility; SST=single-shell tank.

**Source:** SAIC 2011.

Waste would be treated the same under Tank Closure Alternatives 6B and 6C; however, the SSTs would be landfill-closed, not clean-closed, under Alternative 6C. The process schematic and material balance summaries under Alternative 6C are presented in Figure D–11 and Tables D–69 and D–70.

**Table D-69. Tank Closure Alternative 6C Radioactive Constituents of Potential Concern Balance**

	Iodine-129		Cesium-137		Carbon-14		Hydrogen-3 (Tritium)		Uranium-233, -234, -235, -238		Neptunium-237		Plutonium -239, -240		Strontium-90		Technetium-99		
	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	Curies	% of BBI	
	<b>Best-Basis Inventory</b>																		
BBI <sup>a</sup>	4.82×10 <sup>1</sup>	N/A	4.58×10 <sup>7</sup>	N/A	3.12×10 <sup>3</sup>	N/A	1.21×10 <sup>4</sup>	N/A	9.38×10 <sup>2</sup>	N/A	1.41×10 <sup>2</sup>	N/A	8.14×10 <sup>4</sup>	N/A	5.05×10 <sup>7</sup>	N/A	2.97×10 <sup>4</sup>	N/A	
<b>Tank Closure Waste Inventory</b>																			
Tank residual waste <sup>b</sup>	4.82×10 <sup>-1</sup>	1.0	4.58×10 <sup>5</sup>	1.0	3.12×10 <sup>1</sup>	1.0	1.21×10 <sup>2</sup>	1.0	9.38	1.0	1.41	1.0	8.14×10 <sup>2</sup>	1.0	5.05×10 <sup>5</sup>	1.0	2.97×10 <sup>2</sup>	1.0	
IHLW glass <sup>c</sup>	6.99×10 <sup>-3</sup>	0.0	4.49×10 <sup>7</sup>	97.9	0	0.0	0	0.0	8.73×10 <sup>2</sup>	93.1	1.40×10 <sup>2</sup>	99.0	8.06×10 <sup>4</sup>	99.0	4.93×10 <sup>7</sup>	97.6	2.73×10 <sup>2</sup>	0.9	
ILAW glass and retired LAW melters <sup>d</sup>	9.56	19.8	4.45×10 <sup>5</sup>	1.0	0	0.0	0	0.0	5.47×10 <sup>1</sup>	5.8	8.35×10 <sup>-3</sup>	0.0	1.45	0.0	2.30×10 <sup>3</sup>	0.0	2.88×10 <sup>4</sup>	96.9	
ETF-generated solid secondary waste <sup>e</sup>	3.36×10 <sup>1</sup>	69.7	4.59×10 <sup>-1</sup>	0.0	8.51	0.3	0	0.0	4.03×10 <sup>-2</sup>	0.0	5.11×10 <sup>-2</sup>	0.0	6.90×10 <sup>-4</sup>	0.0	6.42	0.0	8.63×10 <sup>1</sup>	0.3	
Solid secondary waste <sup>f</sup>	4.65	9.7	1.95×10 <sup>5</sup>	0.4	0	0.0	0	0.0	3.64	0.4	2.83×10 <sup>-1</sup>	0.2	1.98×10 <sup>2</sup>	0.2	7.76×10 <sup>5</sup>	1.5	4.31×10 <sup>2</sup>	1.5	
<b>Total<sup>g</sup></b>	<b>4.83×10<sup>1</sup></b>	<b>100.2</b>	<b>4.60×10<sup>7</sup></b>	<b>100.2</b>	<b>3.97×10<sup>1</sup></b>	<b>1.3</b>	<b>1.21×10<sup>2</sup></b>	<b>1.0</b>	<b>9.41×10<sup>2</sup></b>	<b>100.3</b>	<b>1.41×10<sup>2</sup></b>	<b>100.2</b>	<b>8.16×10<sup>4</sup></b>	<b>100.2</b>	<b>5.06×10<sup>7</sup></b>	<b>100.1</b>	<b>2.99×10<sup>4</sup></b>	<b>100.5</b>	
<b>Other Inventory</b>																			
Solid and liquid secondary waste from cesium and strontium capsules	0	N/A	1.99×10 <sup>5</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	3.16×10 <sup>5</sup>	N/A	0	N/A	
Cesium and strontium capsules <sup>h</sup>	0	N/A	4.59×10 <sup>7</sup>	N/A	0	N/A	0	N/A	0	N/A	0	N/A	0	N/A	1.98×10 <sup>7</sup>	N/A	0	N/A	
Rubble, soil, and equipment <sup>i</sup>	1.67×10 <sup>-2</sup>	N/A	1.31×10 <sup>4</sup>	N/A	1.47	N/A	6.03	N/A	4.82×10 <sup>-1</sup>	N/A	3.24×10 <sup>-2</sup>	N/A	4.32×10 <sup>1</sup>	N/A	3.05×10 <sup>4</sup>	N/A	9.72	N/A	
<b>Air Emissions</b>																			
Treatment air emissions <sup>j</sup>	4.78×10 <sup>1</sup>	N/A	4.69×10 <sup>4</sup>	N/A	3.10×10 <sup>3</sup>	N/A	1.20×10 <sup>4</sup>	N/A	4.66×10 <sup>-1</sup>	N/A	NR	N/A	4.04×10 <sup>1</sup>	N/A	3.55×10 <sup>4</sup>	N/A	1.47×10 <sup>1</sup>	N/A	

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Although processed as ILAW glass, glass and retired melters would be managed and disposed of as IHLW.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility. Excludes liquid secondary waste from the processing of cesium and strontium capsules, which would be reported separately.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because a portion of each would be released to the offgas streams and stack, and a portion (ETF-generated liquid) would be disposed of in the State-Approved Land Disposal Site.

<sup>h</sup> To be treated in the Waste Treatment Plant, resulting in glass waste, which would be stored on site until disposition decisions are made and implemented.

<sup>i</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both iodine-129 capture and air emission releases were assumed.

**Key:** %=percent; BBI=Best-Basis Inventory; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; N/A=not applicable; NR=not reported.

**Source:** SAIC 2011.

Table D-70. Tank Closure Alternative 6C Chemical Constituents of Potential Concern Balance

	Chromium		Mercury		Nitrate		Lead		Total Uranium		Acetonitrile		Benzene		Butanol		PCBs		2,4,6-TCP	
	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI	Kg	% of BBI
<b>Best-Basis Inventory</b>																				
BBI <sup>a</sup>	5.98×10 <sup>5</sup>	N/A	1.83×10 <sup>3</sup>	N/A	7.08×10 <sup>7</sup>	N/A	8.41×10 <sup>4</sup>	N/A	5.97×10 <sup>5</sup>	N/A	2.95×10 <sup>4</sup>	N/A	2.40	N/A	3.45×10 <sup>6</sup>	N/A	1.68×10 <sup>3</sup>	N/A	1.11	N/A
<b>Tank Closure Waste Inventory</b>																				
Tank residual waste <sup>b</sup>	5.98×10 <sup>3</sup>	1.0	1.83×10 <sup>1</sup>	1.0	7.08×10 <sup>5</sup>	1.0	8.41×10 <sup>2</sup>	1.0	5.97×10 <sup>3</sup>	1.0	2.95×10 <sup>2</sup>	1.0	2.40×10 <sup>-2</sup>	1.0	3.45×10 <sup>4</sup>	1.0	1.68×10 <sup>1</sup>	1.0	1.11×10 <sup>-2</sup>	1.0
IHLW glass <sup>c</sup>	1.36×10 <sup>5</sup>	22.7	0	0.0	0	0.0	7.45×10 <sup>4</sup>	88.6	5.52×10 <sup>5</sup>	92.5	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ILAW glass and retired LAW melters <sup>d</sup>	4.56×10 <sup>5</sup>	76.2	0	0.0	0	0.0	8.88×10 <sup>3</sup>	10.6	3.74×10 <sup>4</sup>	6.3	0	0.0	0	0.0	0	0.0	0	0.0	0	0.0
ETF-generated solid secondary waste <sup>e</sup>	4.43×10 <sup>1</sup>	0.0	5.55	0.3	9.01×10 <sup>6</sup>	12.7	4.58	0.0	4.00×10 <sup>1</sup>	0.0	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
Solid secondary waste <sup>f</sup>	1.94×10 <sup>3</sup>	0.3	1.76×10 <sup>3</sup>	96.4	0	0.0	2.47×10 <sup>2</sup>	0.3	2.32×10 <sup>3</sup>	0.4	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Total<sup>g</sup></b>	<b>6.00×10<sup>5</sup></b>	<b>100.3</b>	<b>1.78×10<sup>3</sup></b>	<b>97.7</b>	<b>9.72×10<sup>6</sup></b>	<b>13.7</b>	<b>8.45×10<sup>4</sup></b>	<b>100.5</b>	<b>5.98×10<sup>5</sup></b>	<b>100.2</b>	<b>2.95×10<sup>2</sup></b>	<b>1.0</b>	<b>2.40×10<sup>-2</sup></b>	<b>1.0</b>	<b>3.45×10<sup>4</sup></b>	<b>1.0</b>	<b>1.68×10<sup>1</sup></b>	<b>1.0</b>	<b>1.11×10<sup>-2</sup></b>	<b>1.0</b>
<b>Other Inventory<sup>h</sup></b>																				
Rubble, soil, and equipment <sup>i</sup>	5.86×10 <sup>2</sup>	N/A	2.22	N/A	3.93×10 <sup>4</sup>	N/A	3.34×10 <sup>1</sup>	N/A	6.60×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A
<b>Air Emissions</b>																				
Treatment air emissions <sup>j</sup>	NR	N/A	1.81×10 <sup>3</sup>	N/A	NR	N/A	NR	N/A	NR	N/A	NR	N/A	6.14×10 <sup>2</sup>	N/A	NR	N/A	NR	N/A	NR	N/A

<sup>a</sup> Source of BBI data is *Inventory and Source Term Data Package*, DOE-ORP-2003-02, Rev. 0 (DOE 2003a).

<sup>b</sup> Represents 99.0 percent retrieval. For analysis purposes, waste inventories from tank waste retrieval leaks and ancillary equipment were assumed to be treated in the Waste Treatment Plant.

<sup>c</sup> To be stored on site until disposition decisions are made and implemented. This inventory would include the retired HLW melters.

<sup>d</sup> Although processed as ILAW glass, glass and retired melters would be managed and disposed of as IHLW.

<sup>e</sup> Includes secondary liquids that would be sent to the ETF and treated; solids would be disposed of in an Integrated Disposal Facility.

<sup>f</sup> Includes solid low-level radioactive waste and mixed low-level radioactive waste streams that would be generated by the waste treatment operations. Such waste streams would include debris waste, melter consumables, failed process components, analytical laboratory waste, spent resins, spent carbon adsorbent, high-efficiency particulate air filters, and other process-related waste. Excludes contaminated liquid effluent waste streams, which would be treated at the ETF.

<sup>g</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. BBI percentages were rounded to the nearest tenth. The organic chemicals (acetonitrile, benzene, butanol, PCBs, and 2,4,6-TCP) may not total 100 percent because they would be destroyed during thermal treatment processes. Nitrate may not total 100 percent because it would be volatilized and released through the facility stack.

<sup>h</sup> No chemical constituents of potential concern have been reported in the cesium and strontium capsule secondary-waste streams.

<sup>i</sup> Rubble, soil, and equipment would be generated by removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

<sup>j</sup> Includes the air emissions from all waste treatment processes, including those from treatment of the cesium and strontium capsules. For analysis purposes, both mercury capture and air emission releases were assumed.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** %=percent; BBI=Best-Basis Inventory; butanol=n-butyl alcohol; ETF=Effluent Treatment Facility; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; Kg=kilograms; LAW=low-activity waste; N/A=not applicable; NR=not reported; PCB=polychlorinated biphenyl; TCP=trichlorophenol.

**Source:** SAIC 2011.

For the purpose of long-term impact assessment, constituent inventory estimates are required for three categories of soil: surface, near surface, and deep soil. Surface soil is defined as soil located between 0.15 and 0.3 meters (0.5 and 1 foot) of the surface. For long-term impacts, surface soil constituent inventories are expected to be minor for three reasons. First, surface contamination occurs primarily due to spills, and current operating procedures call for prompt remediation. Second, during the 1990s, a layer of clean soil was placed over the tank farms to reduce the dose to workers. Third, under all Tank Closure alternatives except Alternatives 1 and 2A, all tank farms would be capped or backfilled with clean soil. Near-surface soil is defined as soil located between 0.3 and 4.6 meters (1 and 15 feet) of the surface. Inventories in this category are dominated by the contributions of ancillary equipment, as described in Section D.1.2. Finally, deep soil is defined as soil located at depths greater than 4.6 meters (15 feet). Contamination of deep soil is expected to be due to past leaks, discharges to cribs and trenches (ditches), and tank waste retrieval leaks, for which inventory estimates are presented in Sections D.1.4, D.1.5, and D.1.6, respectively.

During closure of the tank farms, combinations of the three categories of soil would be disposed of on site, either with or without additional cleaning. Inventories of radioactive and chemical constituents projected under each of the Tank Closure alternatives are presented in the following section in conjunction with inventories of the waste forms proposed for waste disposal.

As discussed in Chapter 2, under Tank Closure Alternatives 2B, 3A, 3B, 3C, and 6C, contaminated soil would be removed (upper 4.6 meters [15 feet]) at the BX and SX tank farms only. Under Tank Closure Alternative 4, the BX and SX tank farms would be clean-closed; clean closure would include disposal of ancillary equipment and contaminated soils. Under Tank Closure Alternatives 6A and 6B, Base Cases, all SST farms would be clean-closed; as under Alternative 4, clean closure would include disposal of ancillary equipment and contaminated soils. In addition, Tank Closure Alternatives 6A and 6B include an analysis of expansion of clean closure to include the six sets of contiguous cribs and trenches (ditches) in addition to the SST farms. Therefore, three contaminated-soil onsite disposal analyses were conducted using current tank, ancillary equipment, tank residual, retrieval, and past leak inventory data. The bases for these calculations are the process options described in the scaled data documentation prepared for this *TC & WM EIS* (SAIC 2010a). These options involve the different types of waste, recovery efficiencies, and combinations of processing under each alternative. Material balances reflecting these process options were developed using the analysis described in the following paragraphs.

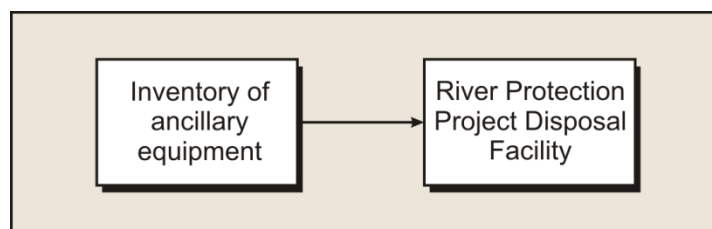
For Tank Closure Alternatives 2B, 3A, 3B, 3C, and 6C (see Figure D–13 for a simplified flowsheet and Tables D–71 and D–72 for inventories), the following calculation was used to determine the radioactive and chemical constituent inventories that would be associated with removal of soil and ancillary equipment within 4.6 meters (15 feet) of the ground surface at the BX and SX tank farms. The calculation is based on the assumptions that the inventory of contaminants excluding ancillary equipment would be minor compared with the inventory of contaminants including ancillary equipment, and that all of the recovered soil, ancillary equipment, and associated inventory of contaminants would be packaged for disposal in the RPPDF. The inventory of contaminants was calculated as follows:

$$M_{\text{soil}} = 1.0 \times M_{\text{anc}}$$

where:

$M_{\text{soil}}$  = inventory of radioactive or chemical constituents in contaminated equipment and soil disposed of on site

$M_{\text{anc}}$  = inventory of radioactive or chemical constituents in ancillary equipment



**Figure D–13. Tank Closure Alternatives 2B, 3A, 3B, 3C, and 6C Contaminated Soil Removal at BX and SX Tank Farms Flowsheet**

**Table D–71. Tank Closure Alternatives 2B, 3A, 3B, 3C, and 6C Radioactive COPC Inventory from Removal of 4.6 Meters (15 Feet) of Soil at the BX and SX Tank Farms (curies)**

Analyte	MLLW <sup>a</sup>
Hydrogen-3 (tritium)	6.03
Carbon-14	1.47
Strontium-90	3.05×10 <sup>4</sup>
Technetium-99	9.72
Iodine-129	1.67×10 <sup>-2</sup>
Cesium-137	1.31×10 <sup>4</sup>
Uranium-233, -234, -235, -238	4.82×10 <sup>-1</sup>
Neptunium-237	3.24×10 <sup>-2</sup>
Plutonium-239, -240	4.32×10 <sup>1</sup>

<sup>a</sup> Represents 100 percent of the ancillary equipment inventory in BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

**Key:** COPC=constituent of potential concern; MLLW=mixed low-level radioactive waste.

**Source:** SAIC 2011.

**Table D–72. Tank Closure Alternatives 2B, 3A, 3B, 3C, and 6C Chemical COPC Inventory from Removal of 4.6 Meters (15 Feet) of Soil at the BX and SX Tank Farms (kilograms)**

Analyte	MLLW <sup>a</sup>
Chromium	5.86×10 <sup>2</sup>
Mercury	2.22
Nitrate	3.93×10 <sup>4</sup>
Lead	3.34×10 <sup>1</sup>
Total uranium	6.60×10 <sup>2</sup>
Acetonitrile	NR
Benzene	NR
Butanol (n-butyl alcohol)	NR
Polychlorinated biphenyls	NR
2,4,6-Trichlorophenol	NR

<sup>a</sup> Represents 100 percent of the ancillary equipment inventory in BX and SX tank farms. Disposal would take place in the River Protection Project Disposal Facility.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** COPC=constituent of potential concern; MLLW=mixed low-level radioactive waste; NR=not reported.

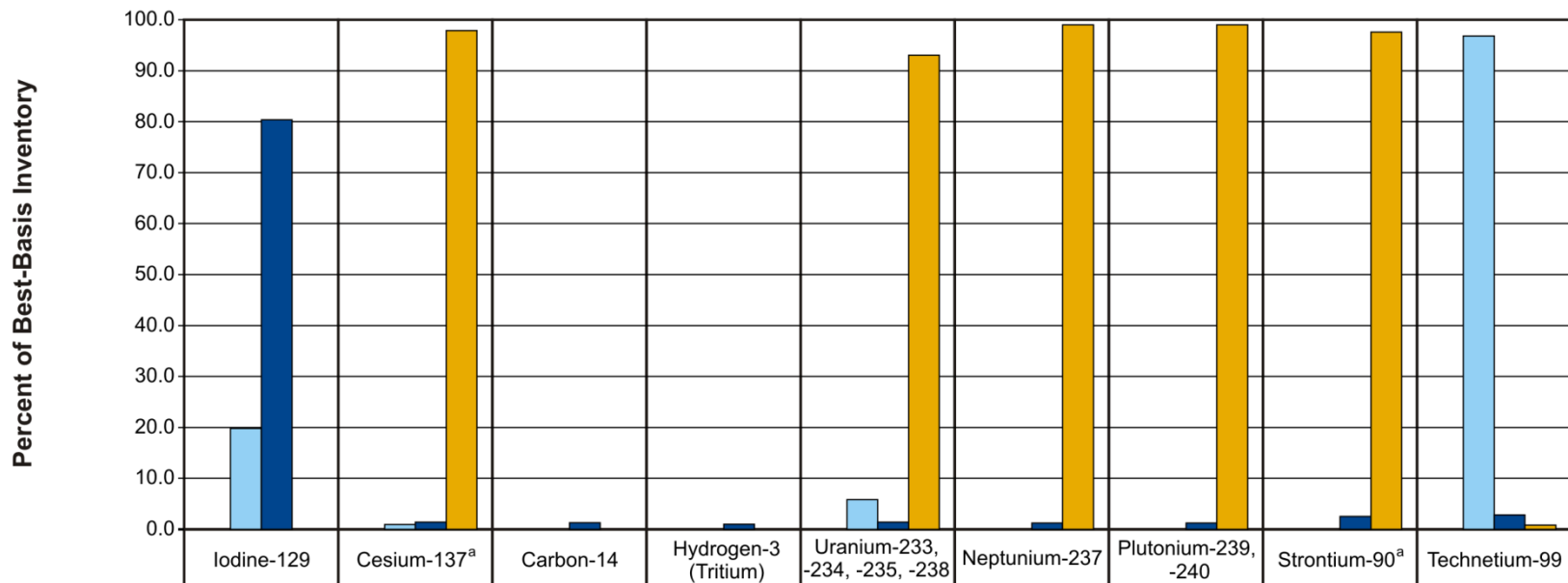
**Source:** SAIC 2011.

### **D.1.8 Distribution of Radioactive Constituents of Potential Concern Under Tank Closure Alternatives**

As discussed in Section D.1.7, the retrieval of tank waste, treatment and stabilization of waste streams, and closure of the tank farms would generate a number of waste forms for both on- and offsite disposal. This section provides both a graphic representation and tabular information on the radioactive COPC inventories for each of the Tank Closure alternatives. Figures D-14 through D-63 (a total of 50 figures) below show the distribution of the nine radioactive COPCs under Tank Closure Alternatives 2A, 2B, 3A, 3B, 3C, 4, 5, the 6A Base and Option Cases, the 6B Base and Option Cases, and 6C (SAIC 2011). These figures include the following for each of the Tank Closure action alternatives:

- A histogram that provides a graphic display of the distribution of the nine radioactive COPCs (iodine-129, cesium-137, carbon-14, tritium, uranium [including uranium-233, -234, -235, and -238], neptunium-237, plutonium [including plutonium-239 and -240], strontium-90, and technetium-99). For each of these COPCs, the histogram provides the total curies in the tank farms (BBI estimate) and the estimated BBI percentage (curie basis) that would be disposed of on site in an IDF as either a glass waste form (ILAW glass, bulk vitrification glass), a grout (cast stone waste, retired LAW melters [grout filled], Effluent Treatment Facility [ETF]-generated solid secondary waste, sulfate grout waste, or tank residual waste), or steam reforming waste. As noted on the histograms, only tank closure waste is included. The histogram excludes waste generated by the FFTF Decommissioning and Waste Management alternatives; offsite waste; onsite non-Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), nontank waste; cesium and strontium capsule waste; waste forms that would be disposed of in the RPPDF; and PPF-generated waste that would contribute to the IHLW glass and ILAW glass.
- Three pie charts that graphically display the disposition of three radioactive COPCs (iodine-129, uranium [including uranium-233, -234, -235, and -238], and technetium-99) under each Tank Closure alternative. As noted on the pie charts, only tank closure waste is included. The pie charts exclude waste generated by the FFTF Decommissioning and Waste Management alternatives; offsite waste; onsite non-CERCLA, nontank waste; cesium and strontium capsule waste; the waste forms that would be disposed of in the RPPDF; and PPF-generated waste that would contribute to the IHLW glass and ILAW glass.
- One summary pie chart that includes all nine radioactive COPCs (iodine-129, cesium-137, carbon-14, tritium, uranium [including uranium-233, -234, -235, and -238], neptunium-237, plutonium [including plutonium-239 and -240], strontium-90, and technetium-99) and displays the disposition of the total of these COPCs under each Tank Closure alternative. As noted on the pie charts, only tank closure waste is included. This pie chart excludes waste generated by the FFTF Decommissioning and Waste Management alternatives; offsite waste; onsite non-CERCLA, nontank waste; cesium and strontium capsule waste; the waste forms that would be disposed of in the RPPDF; and PPF-generated waste that would contribute to the IHLW glass and ILAW glass.

The figures in this section reflect the assumption that IHLW would be disposed of off site (however, this IHLW would be stored on site until disposition decisions are made and implemented). As indicated in its fiscal year 2010 budget request, the Administration terminated the Yucca Mountain program. Notwithstanding the decision to terminate the Yucca Mountain program, DOE remains committed to meeting its obligations to manage and ultimately dispose of HLW and SNF. The Administration convened a Blue Ribbon Commission on America's Nuclear Future in March 2010 to evaluate alternative approaches for meeting these obligations. The commission provides the opportunity for a meaningful dialogue on how best to address this challenging issue and will provide recommendations that will form the basis for working with Congress to revise the statutory framework for managing and disposing of HLW and SNF.



Best-Basis Inventory (curies)	$4.82 \times 10^1$	$4.58 \times 10^7$	$3.12 \times 10^3$	$1.21 \times 10^4$	$9.38 \times 10^2$	$1.41 \times 10^2$	$8.14 \times 10^4$	$5.05 \times 10^7$	$2.97 \times 10^4$
% disposal									
Onsite (glass) <sup>b</sup>	19.8	1.0	0.0	0.0	5.8	0.0	0.0	0.0	96.9
Onsite (grout) <sup>c</sup>	80.4	1.4	1.3	1.0	1.4	1.2	1.2	2.5	2.8
Offsite (glass) <sup>d</sup>	0.0	97.9	0.0	0.0	93.1	99.0	99.0	97.6	0.8
Total <sup>e</sup>	100.2	100.2	1.3	1.0	100.3	100.2	100.2	100.1	100.5

<sup>a</sup> This figure excludes the inventory and secondary waste generated by the treatment of the cesium and strontium capsules.

<sup>b</sup> Immobilized low-activity waste glass.

<sup>c</sup> Tank residuals, retired low-activity waste melters, solid secondary waste, and Effluent Treatment Facility-generated solid secondary waste.

<sup>d</sup> Immobilized high-level radioactive waste glass.

<sup>e</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. Totals may not equal the sum of the contributions due to rounding. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because portions of each would be released to the offgas streams and stack(s) or to the State-Approved Land Disposal Site.

**Note:** Only Tank Closure Alternative 2A waste is included. FFTF Decommissioning alternative waste; Waste Management alternative waste; and offsite- and onsite-generated, non-Comprehensive Environmental Response, Compensation, and Liability Act, nontank waste are excluded.

**Key:** %=percent; FFTF=Fast Flux Test Facility.

**Figure D-14. Tank Closure Alternative 2A Distribution of Radioactive Constituents of Potential Concern**

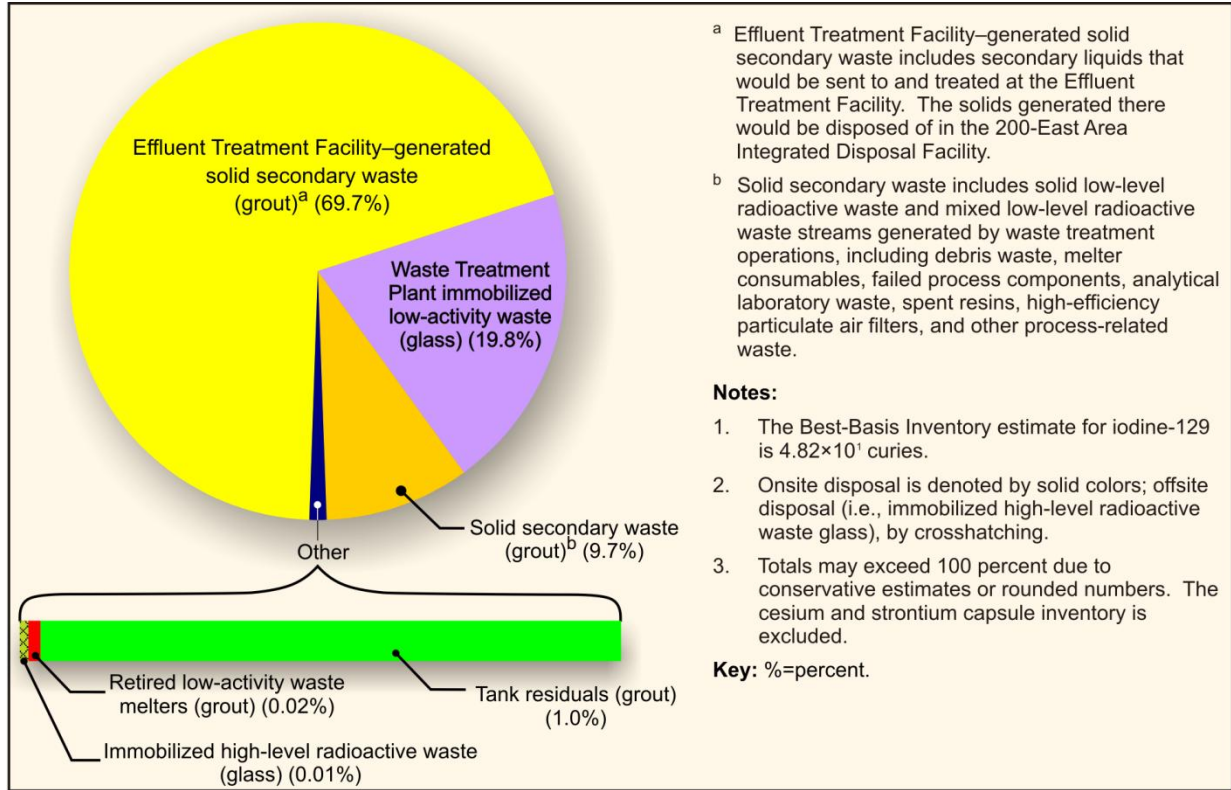


Figure D-15. Tank Closure Alternative 2A Iodine-129 Distribution

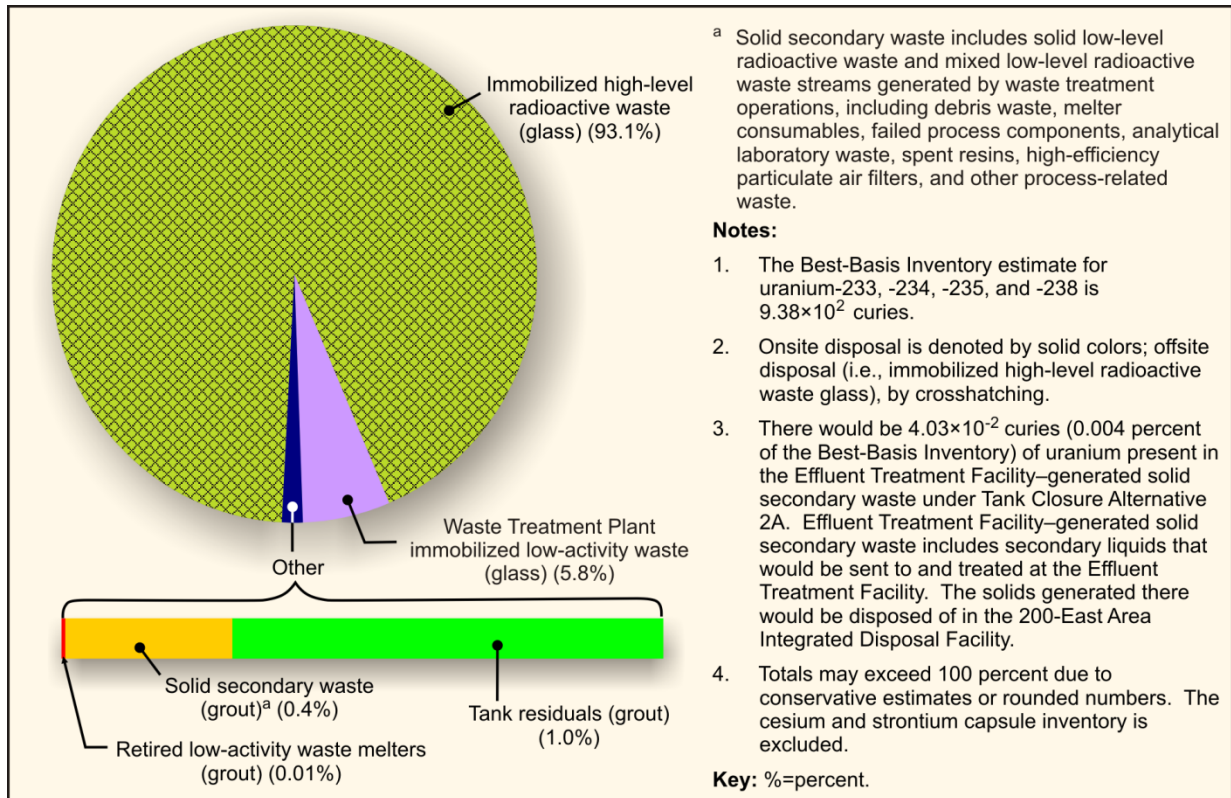
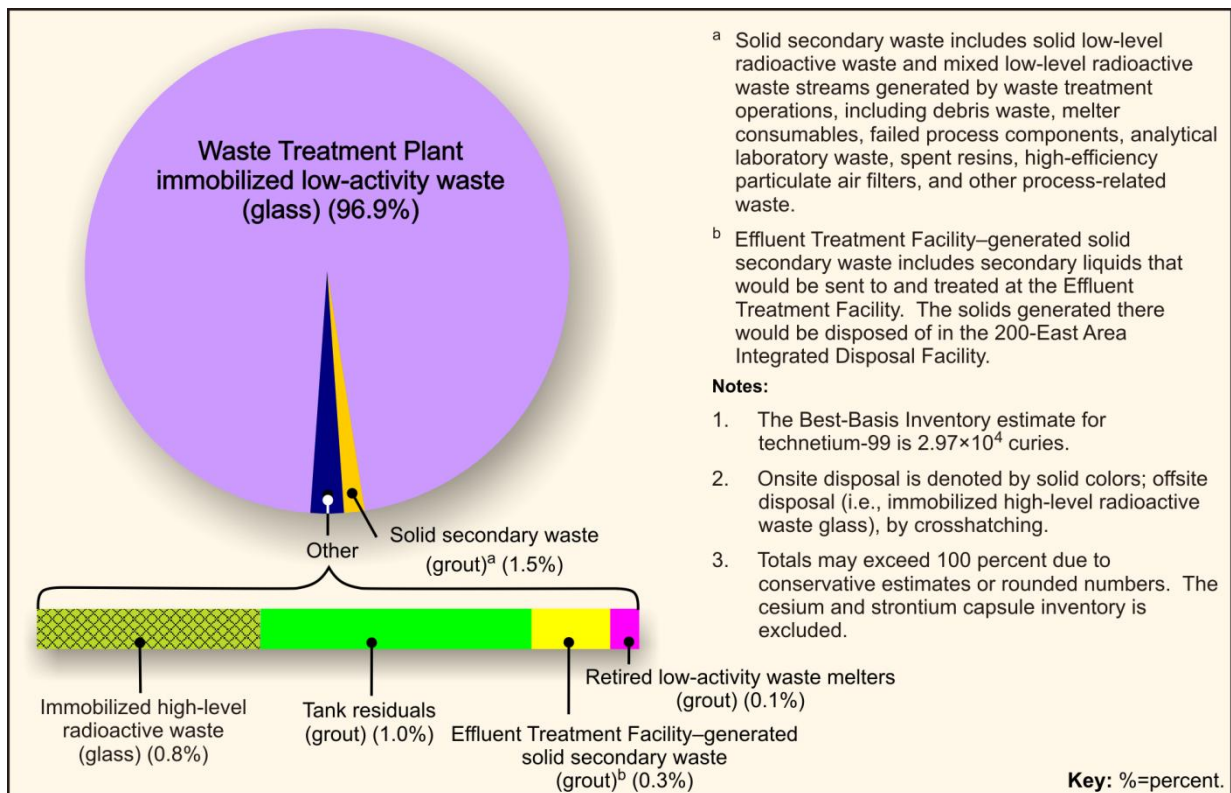
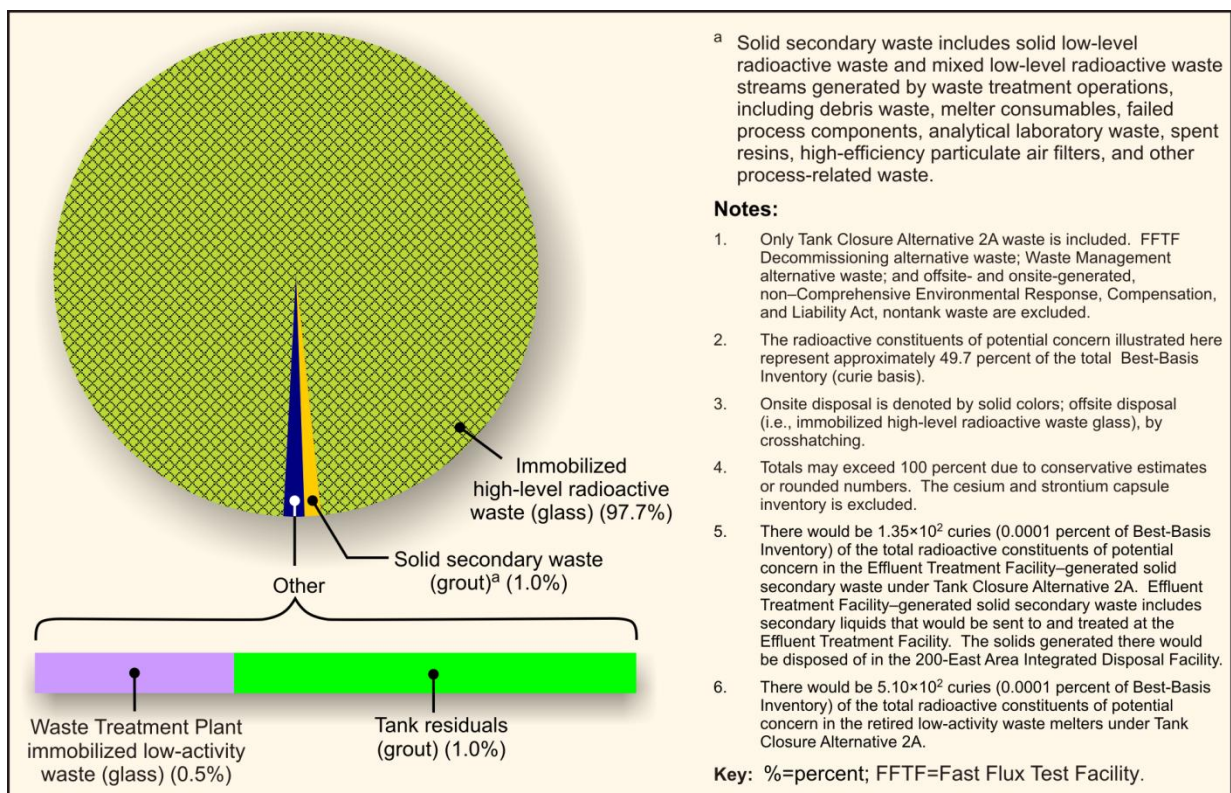


Figure D-16. Tank Closure Alternative 2A Uranium Distribution





**Figure D-17. Tank Closure Alternative 2A Technetium-99 Distribution**



**Figure D-18. Tank Closure Alternative 2A Distribution of Total Radioactive Constituents of Potential Concern**

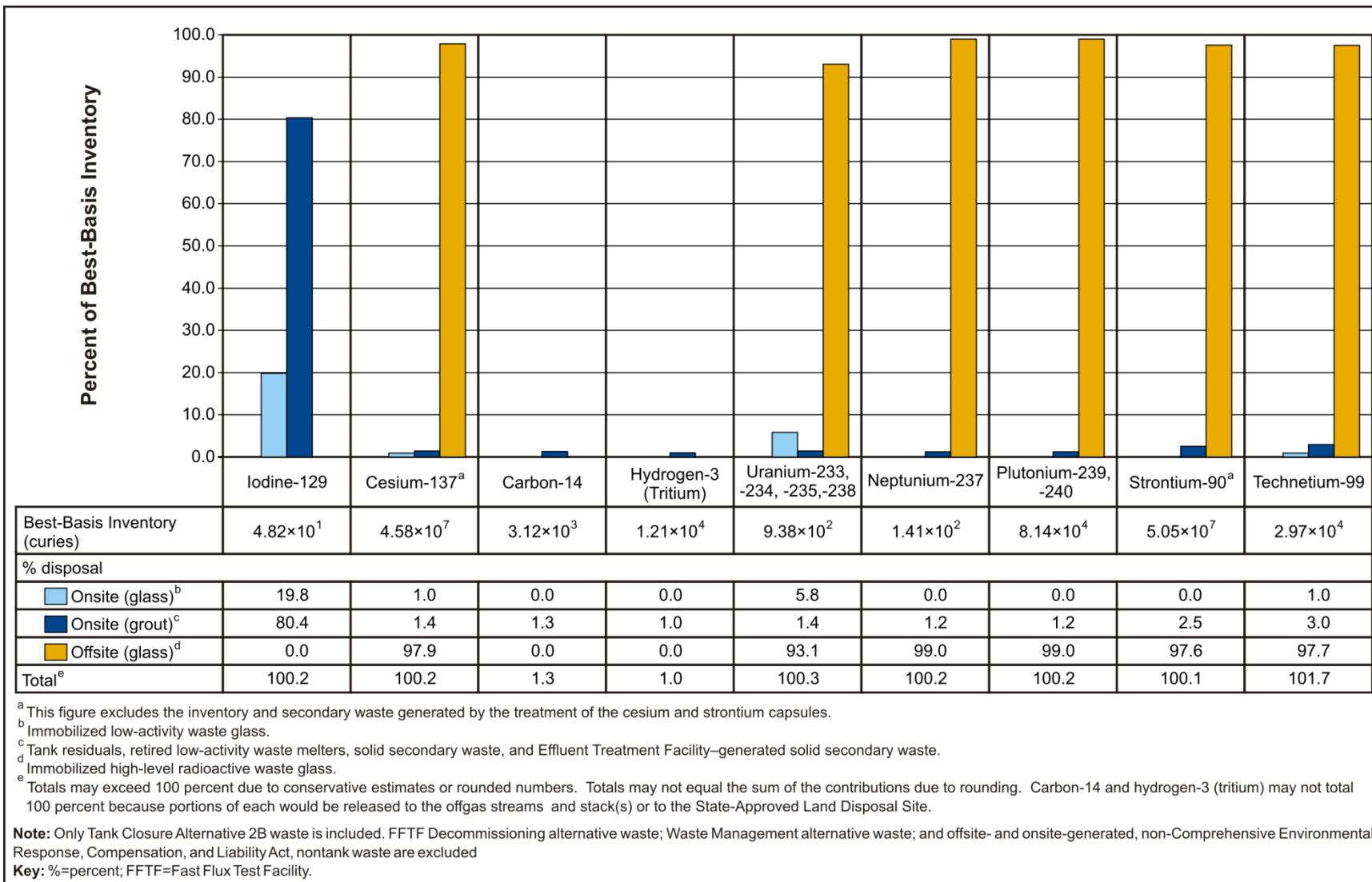


Figure D-19. Tank Closure Alternative 2B Distribution of Radioactive Constituents of Potential Concern



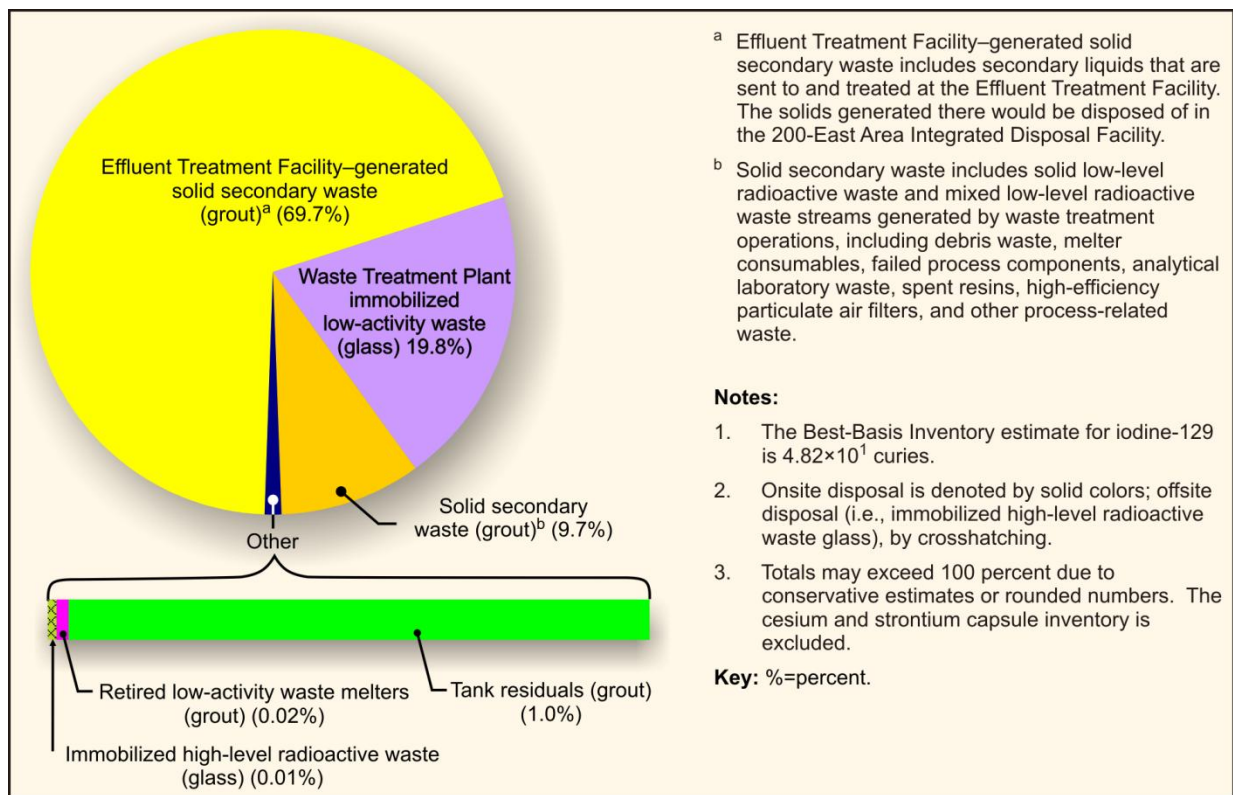


Figure D-20. Tank Closure Alternative 2B Iodine-129 Distribution

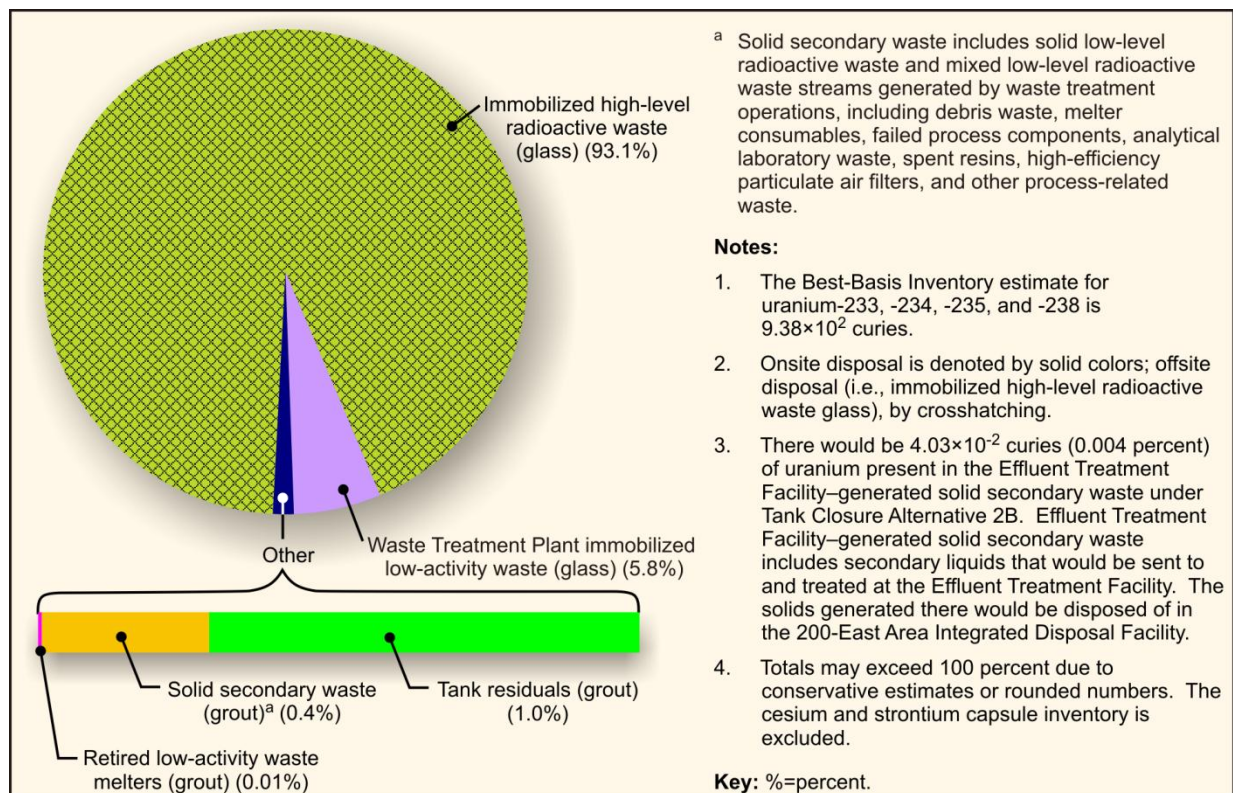


Figure D-21. Tank Closure Alternative 2B Uranium Distribution



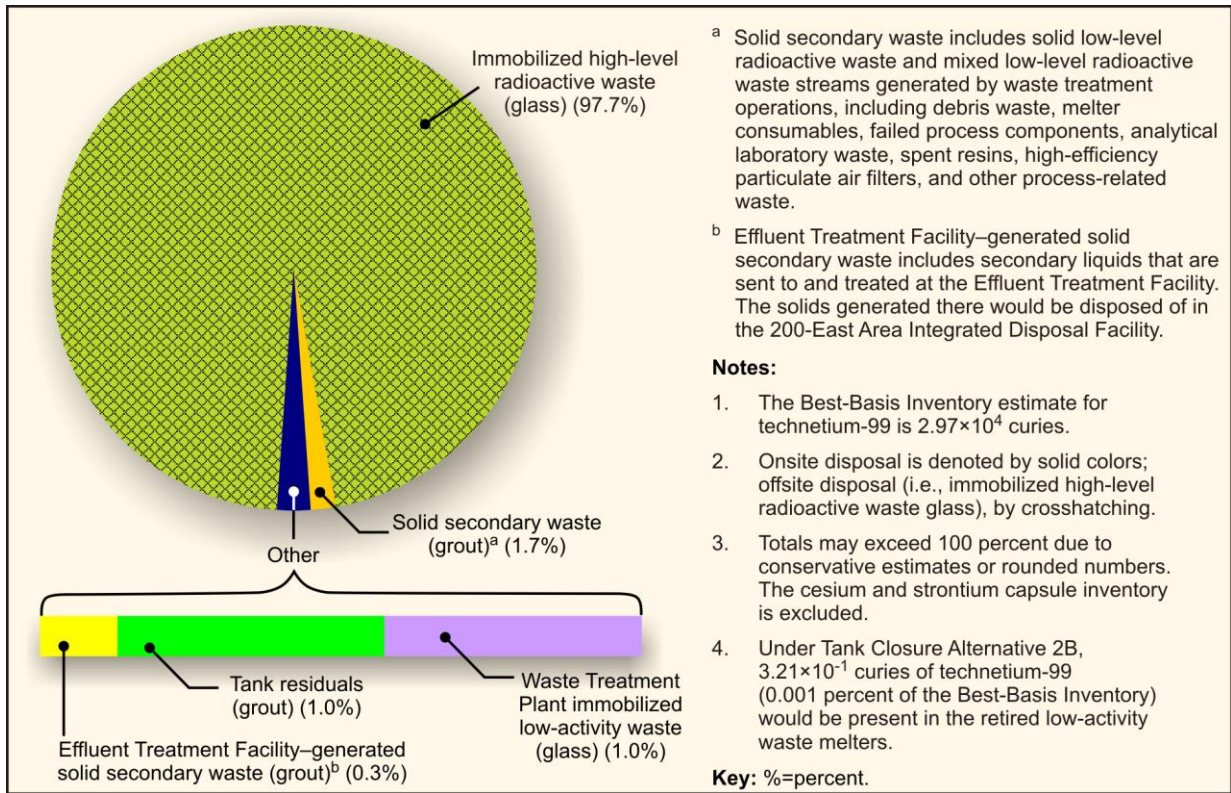


Figure D-22. Tank Closure Alternative 2B Technetium-99 Distribution

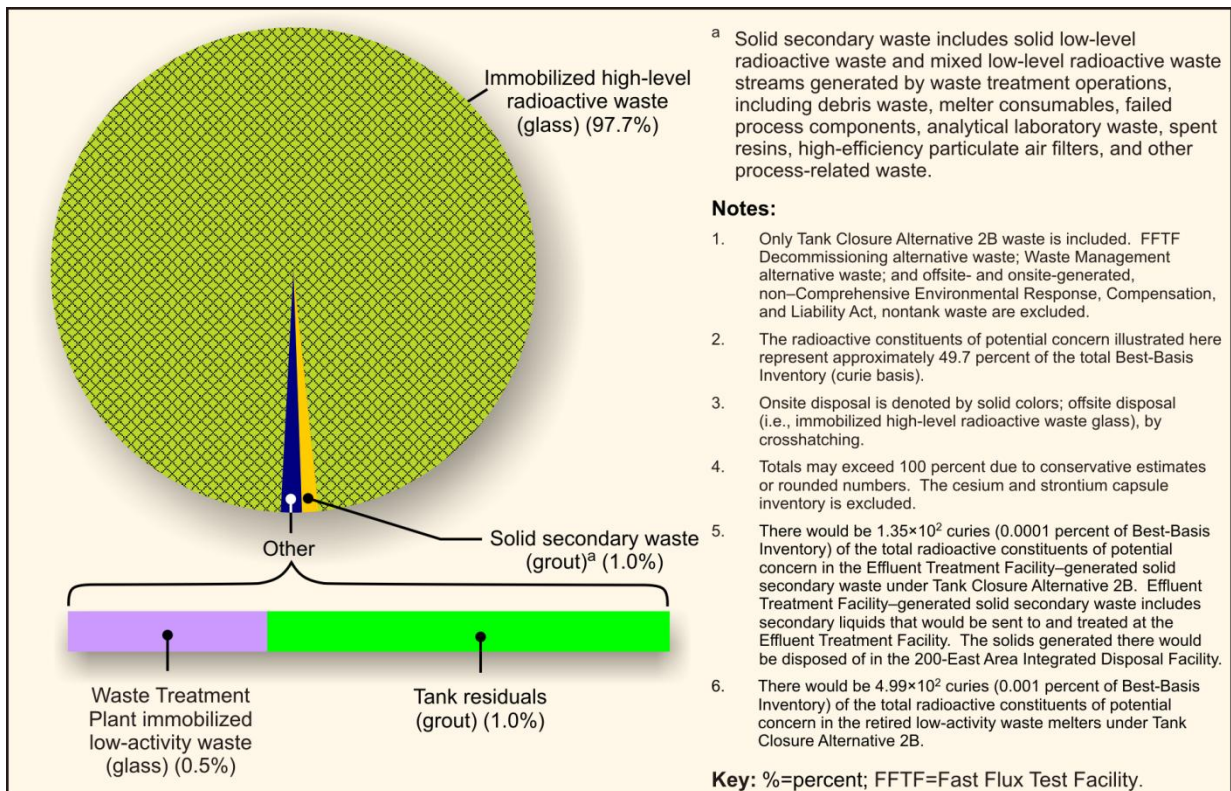
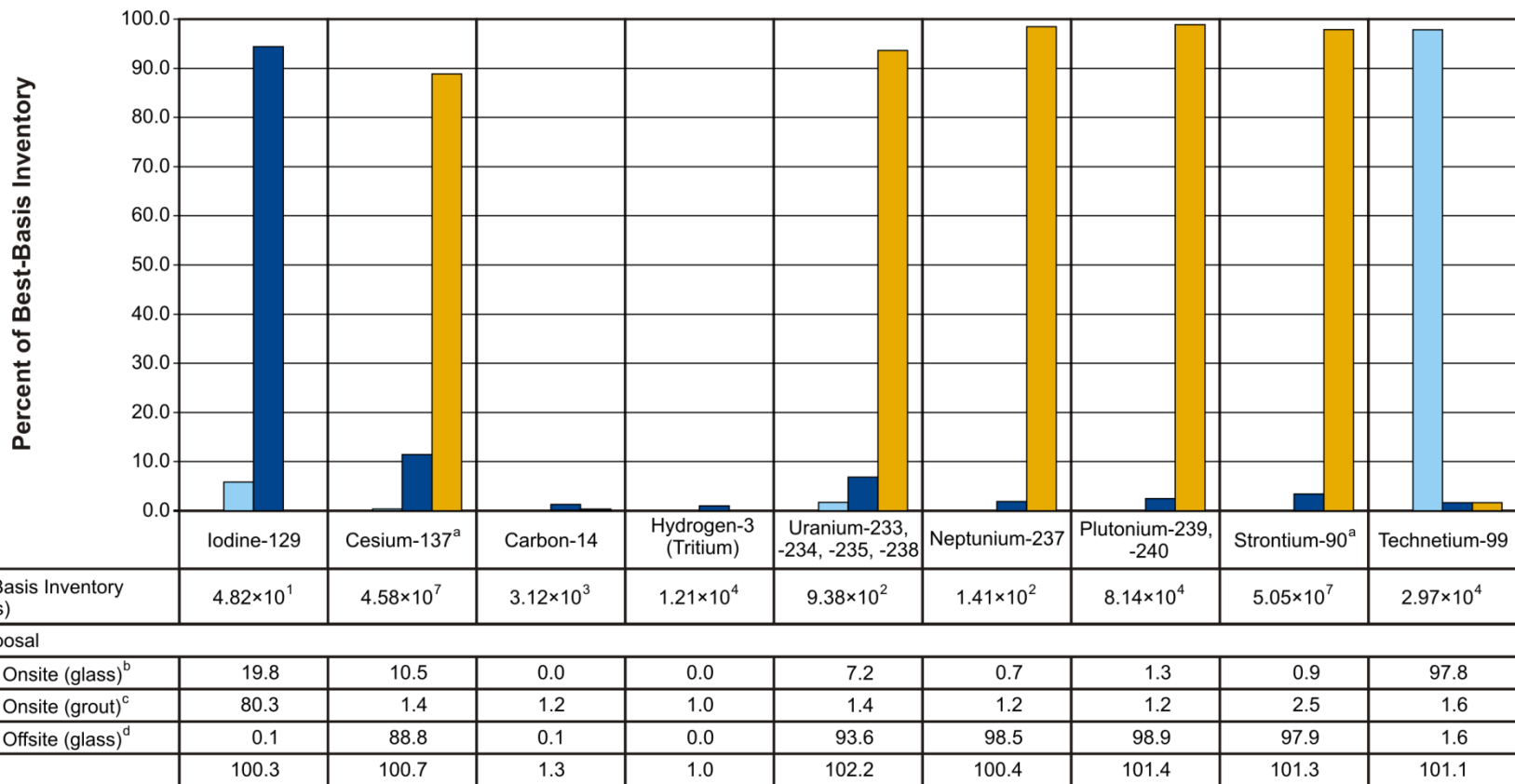


Figure D-23. Tank Closure Alternative 2B Distribution of Total Radioactive Constituents of Potential Concern



<sup>a</sup> This figure excludes the inventory and secondary waste generated by the treatment of the cesium and strontium capsules.

<sup>b</sup> Immobilized low-activity waste glass.

<sup>c</sup> Tank residuals, retired low-activity waste melters, solid secondary waste, and Effluent Treatment Facility-generated solid secondary waste.

<sup>d</sup> Immobilized high-level radioactive waste glass.

<sup>e</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. Totals may not equal the sum of the contributions due to rounding. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because portions of each would be released to the offgas streams and stack(s) or to the State-Approved Land Disposal Site.

**Note:** Only Tank Closure Alternative 3A waste is included. FFTF Decommissioning alternative waste; Waste Management alternative waste; and offsite- and onsite-generated, non-CERCLA=Comprehensive Environmental Response, Compensation, and Liability Act, nontank waste are excluded.

**Key:** %=percent; FFTF=Fast Flux Test Facility.

**Figure D-24. Tank Closure Alternative 3A Distribution of Radioactive Constituents of Potential Concern**



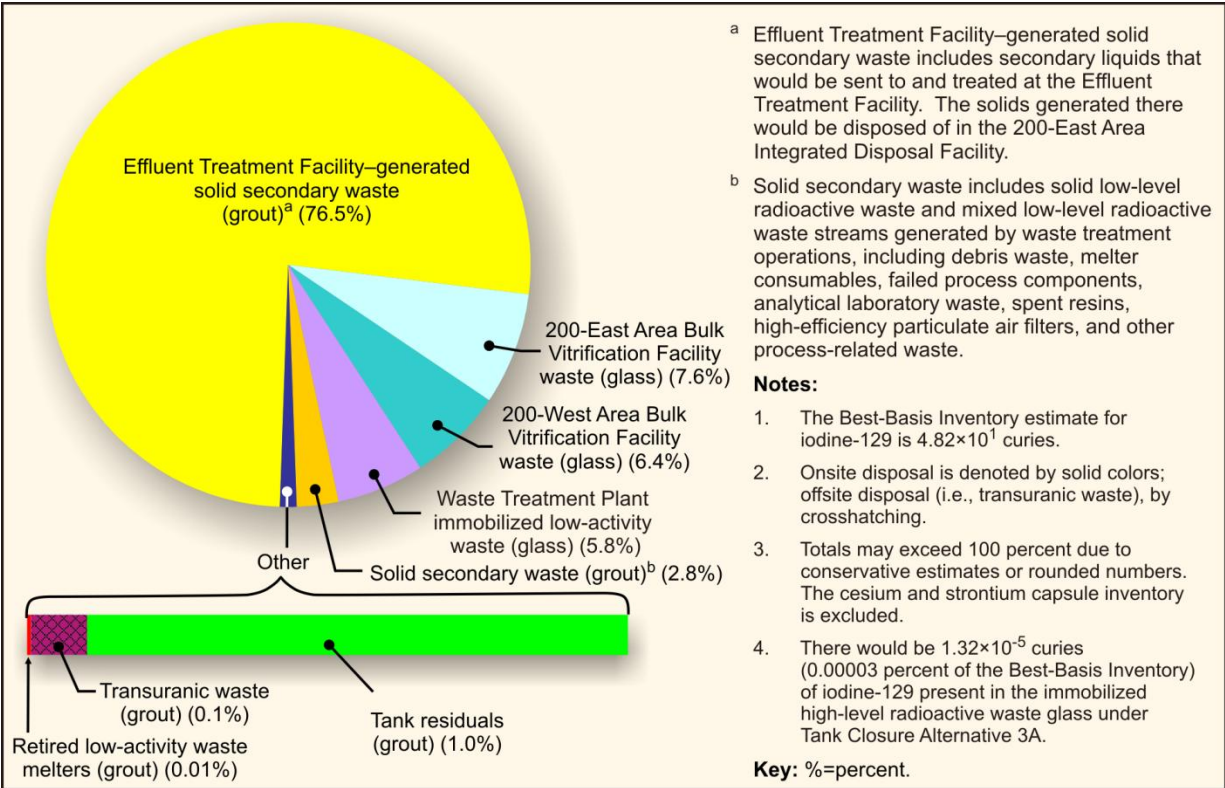


Figure D-25. Tank Closure Alternative 3A Iodine-129 Distribution

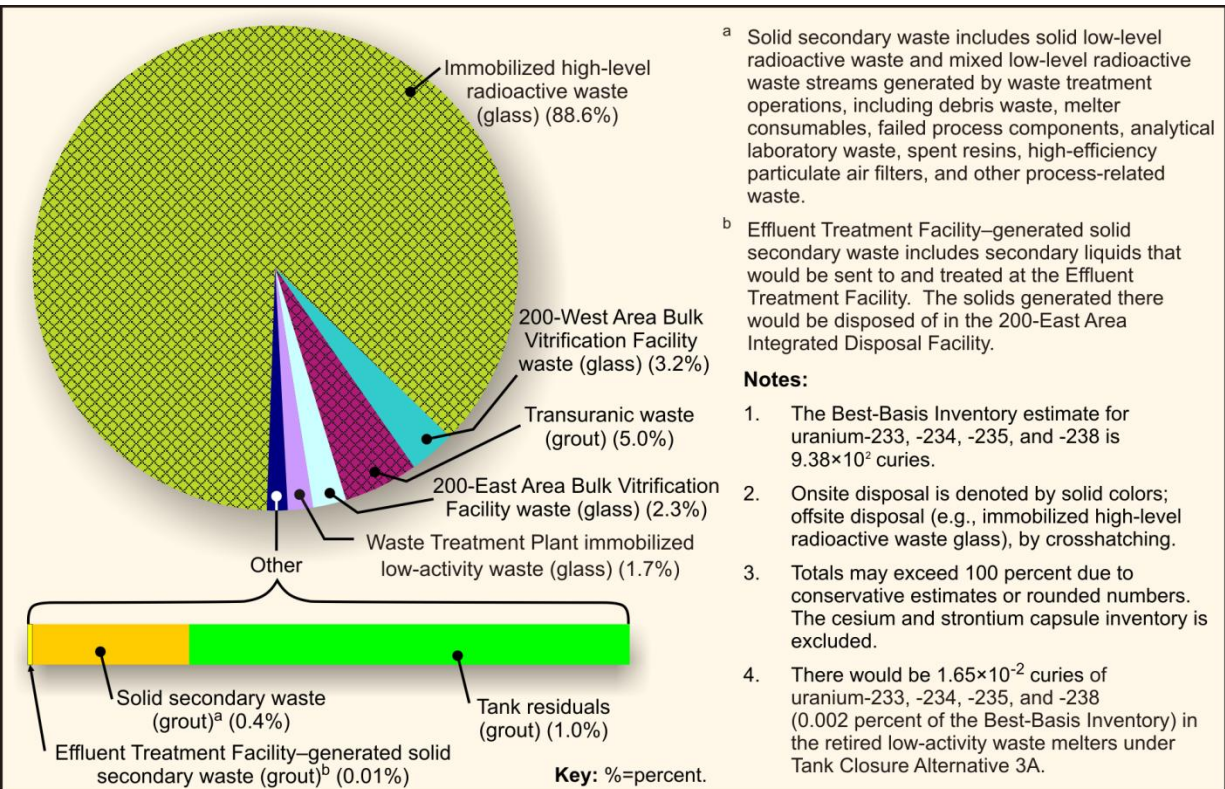
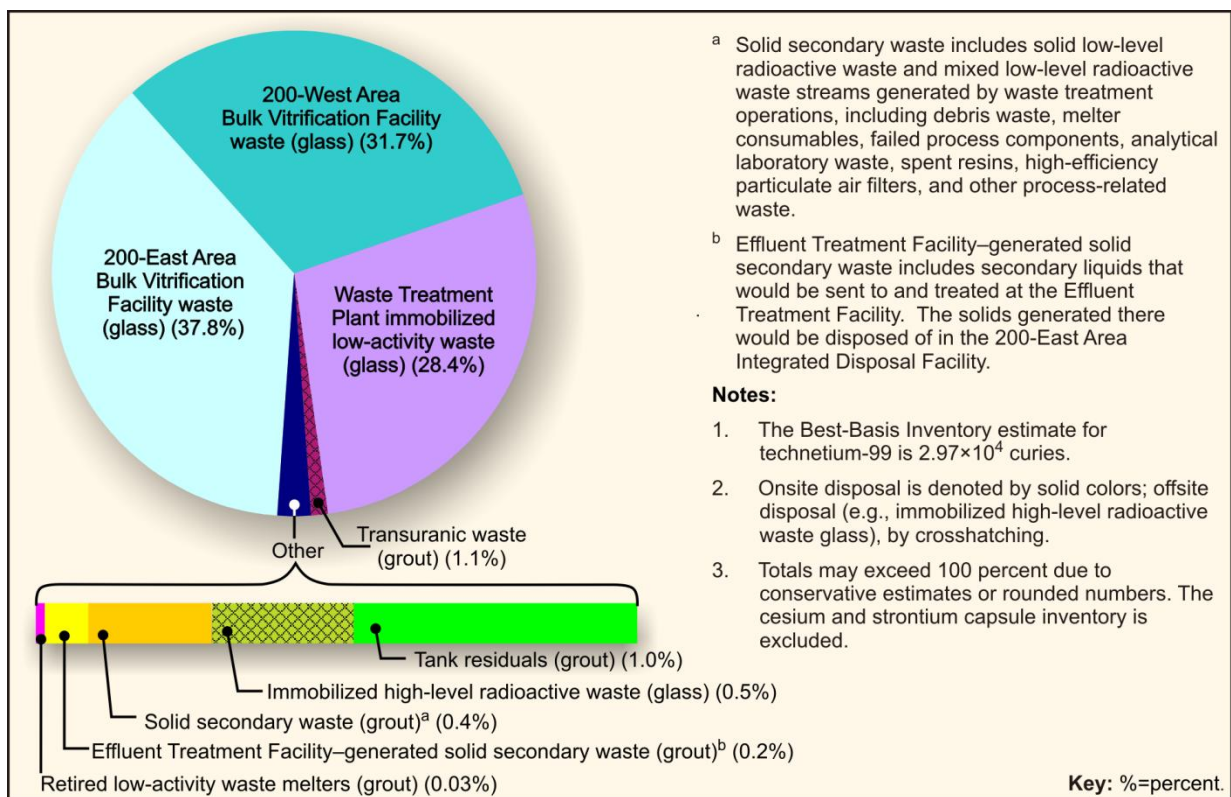
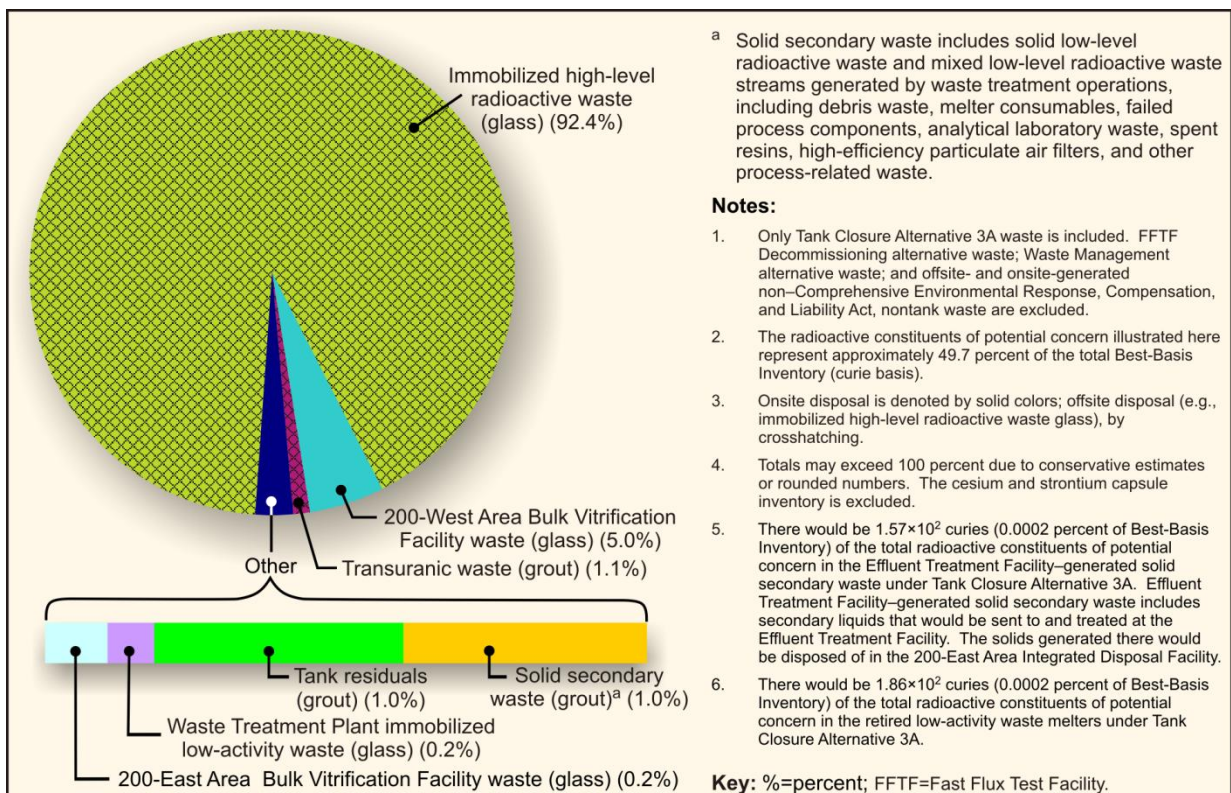


Figure D-26. Tank Closure Alternative 3A Uranium Distribution

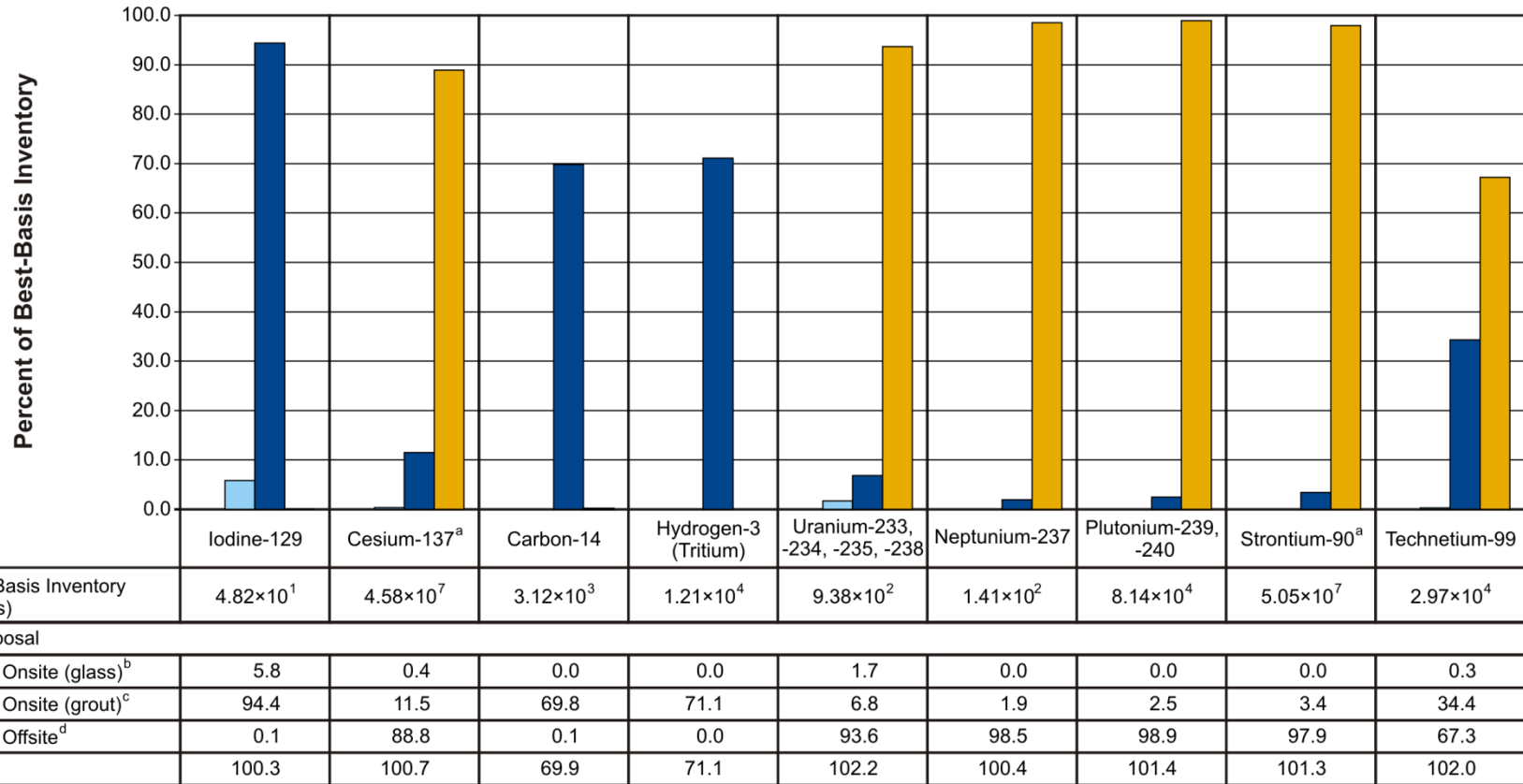




**Figure D-27. Tank Closure Alternative 3A Technetium-99 Distribution**



**Figure D-28. Tank Closure Alternative 3A Distribution of Total Radioactive Constituents of Potential Concern**



Best-Basis Inventory (curies)	Iodine-129	Cesium-137 <sup>a</sup>	Carbon-14	Hydrogen-3 (Tritium)	Uranium-233, -234, -235, -238	Neptunium-237	Plutonium-239, -240	Strontium-90 <sup>a</sup>	Technetium-99
Best-Basis Inventory (curies)	4.82×10 <sup>1</sup>	4.58×10 <sup>7</sup>	3.12×10 <sup>3</sup>	1.21×10 <sup>4</sup>	9.38×10 <sup>2</sup>	1.41×10 <sup>2</sup>	8.14×10 <sup>4</sup>	5.05×10 <sup>7</sup>	2.97×10 <sup>4</sup>
% disposal									
Onsite (glass) <sup>b</sup>	5.8	0.4	0.0	0.0	1.7	0.0	0.0	0.0	0.3
Onsite (grout) <sup>c</sup>	94.4	11.5	69.8	71.1	6.8	1.9	2.5	3.4	34.4
Offsite <sup>d</sup>	0.1	88.8	0.1	0.0	93.6	98.5	98.9	97.9	67.3
Total <sup>e</sup>	100.3	100.7	69.9	71.1	102.2	100.4	101.4	101.3	102.0

<sup>a</sup> This figure excludes the inventory and secondary waste generated by the treatment of the cesium and strontium capsules.

<sup>b</sup> Immobilized low-activity waste glass.

<sup>c</sup> Tank residuals, retired low-activity waste melters, solid secondary waste, and Effluent Treatment Facility-generated solid secondary waste.

<sup>d</sup> Immobilized high-level radioactive waste glass.

<sup>e</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. Totals may not equal the sum of the contributions due to rounding. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because portions of each would be released to the offgas streams and stack(s) or to the State-Approved Land Disposal Site.

**Note:** Only Tank Closure Alternative 3B waste is included. FFTF Decommissioning alternative waste; Waste Management alternative waste; and offsite- and onsite-generated, non-Comprehensive Environmental Response, Compensation, and Liability Act, nontank waste are excluded.

**Key:** %=percent; FFTF=Fast Flux Test Facility.

**Figure D-29. Tank Closure Alternative 3B Distribution of Radioactive Constituents of Potential Concern**



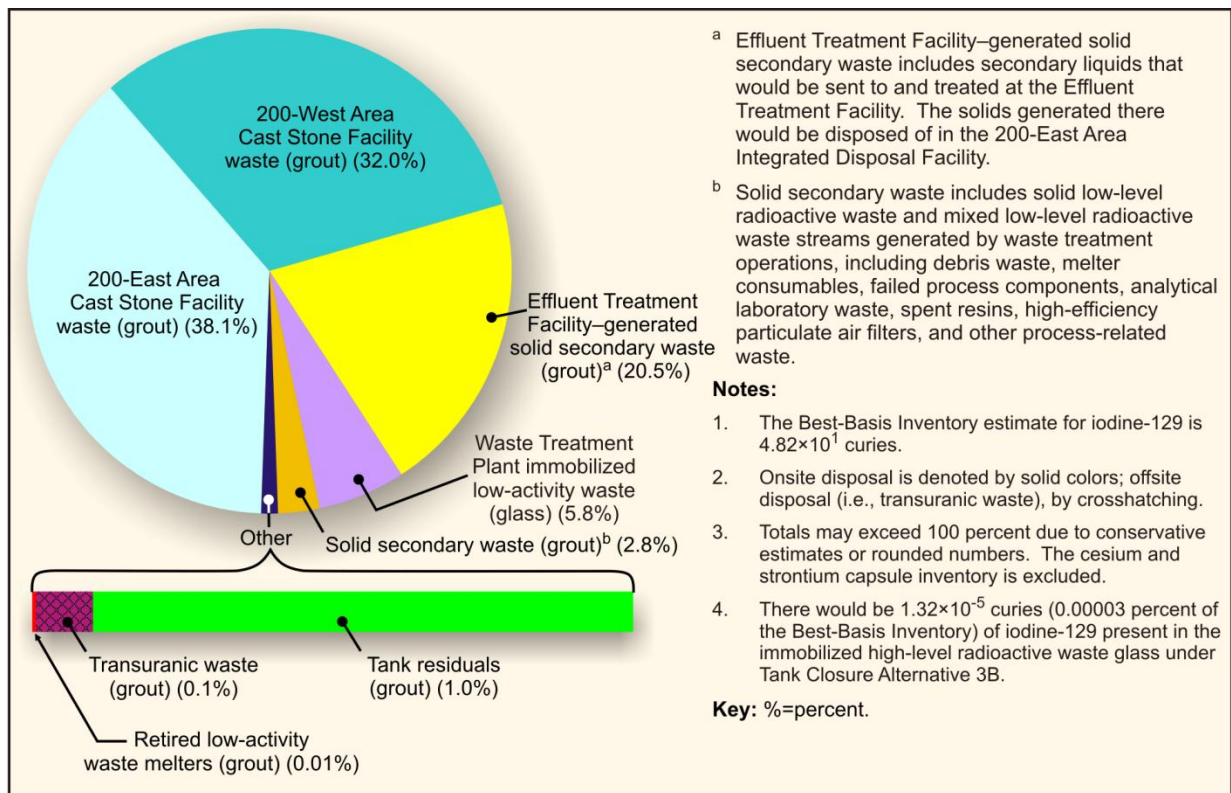


Figure D-30. Tank Closure Alternative 3B Iodine-129 Distribution

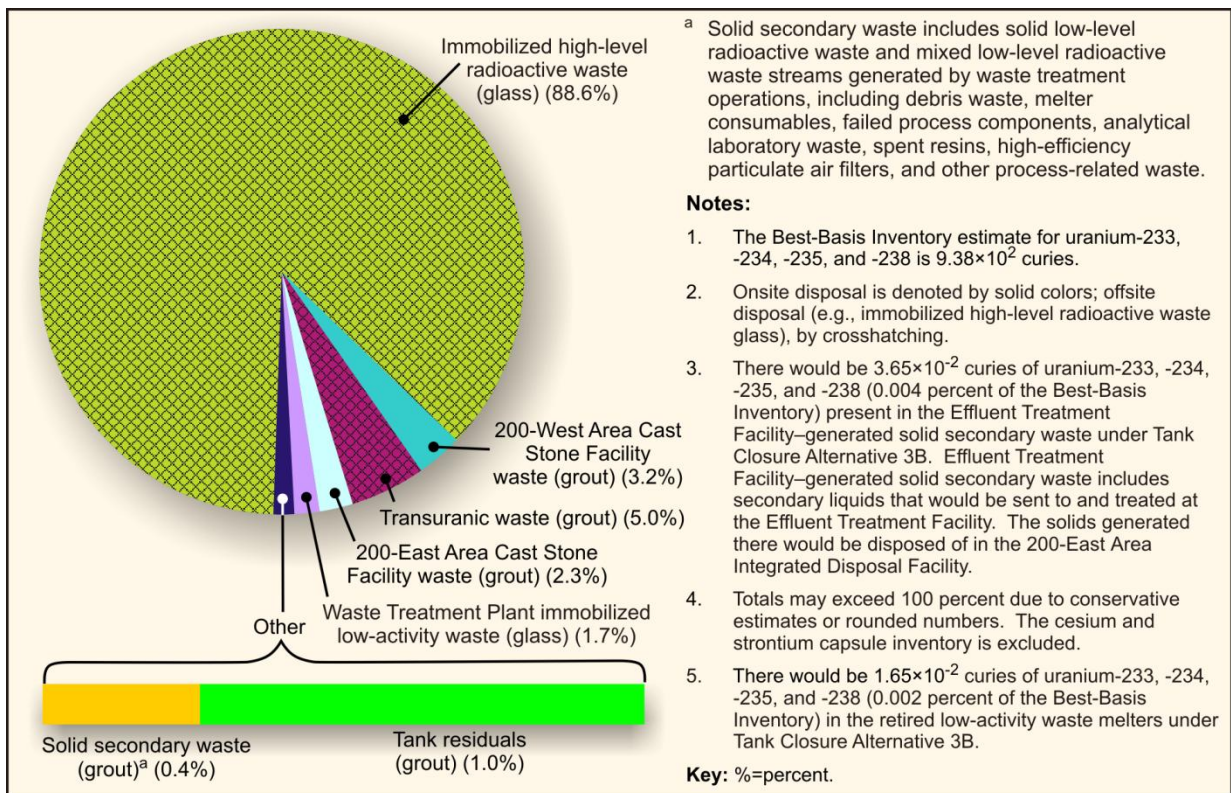


Figure D-31. Tank Closure Alternative 3B Uranium Distribution



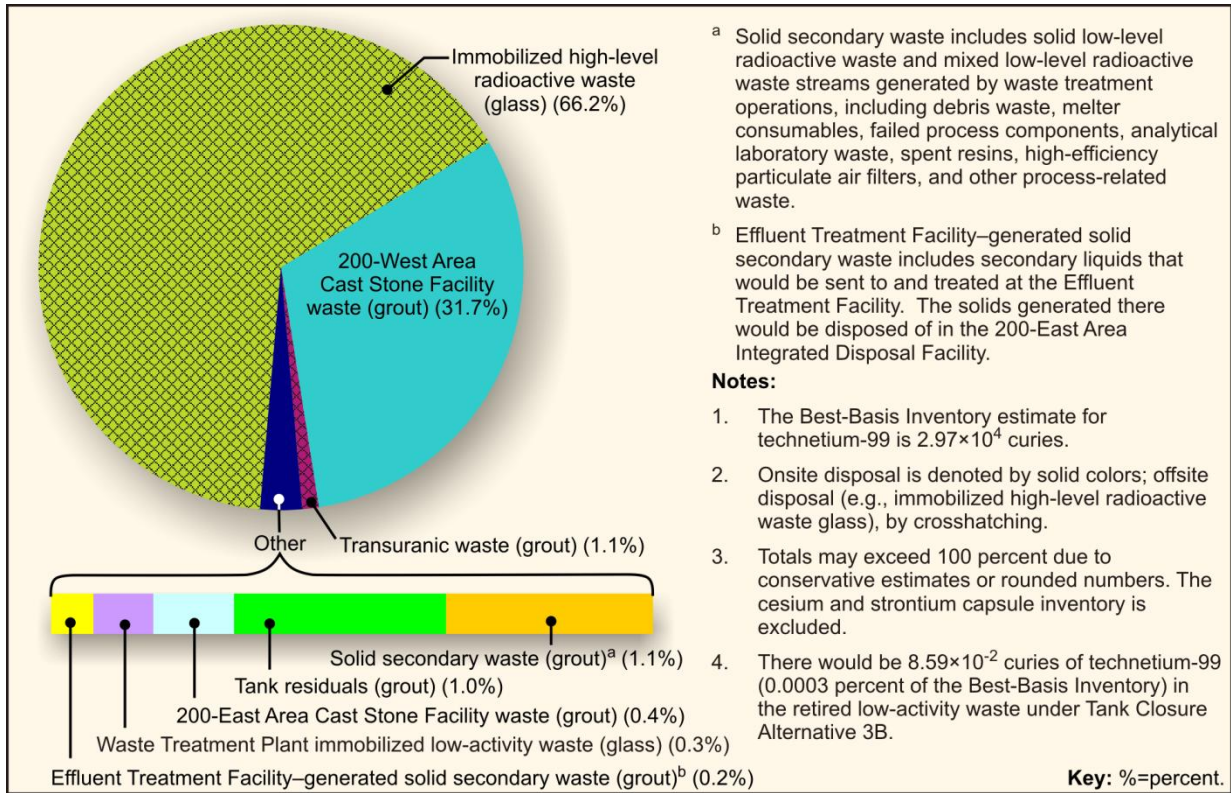


Figure D-32. Tank Closure Alternative 3B Technetium-99 Distribution

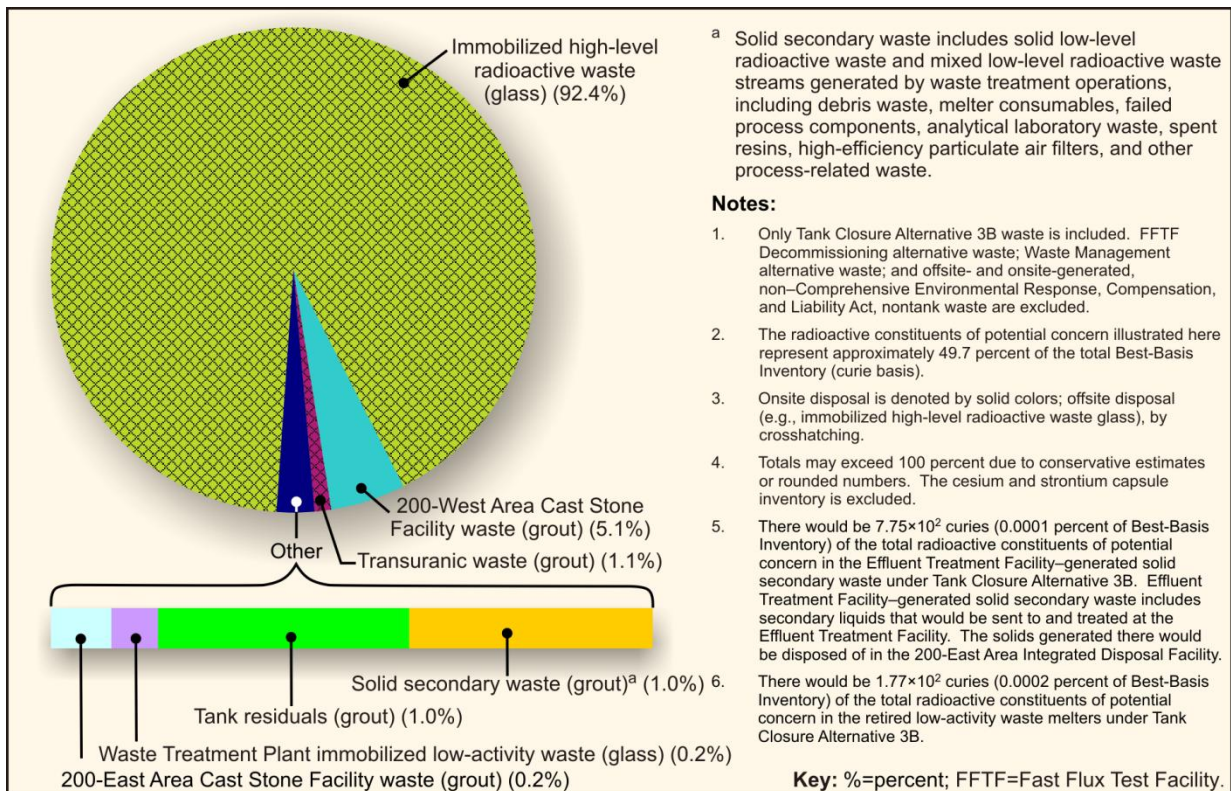
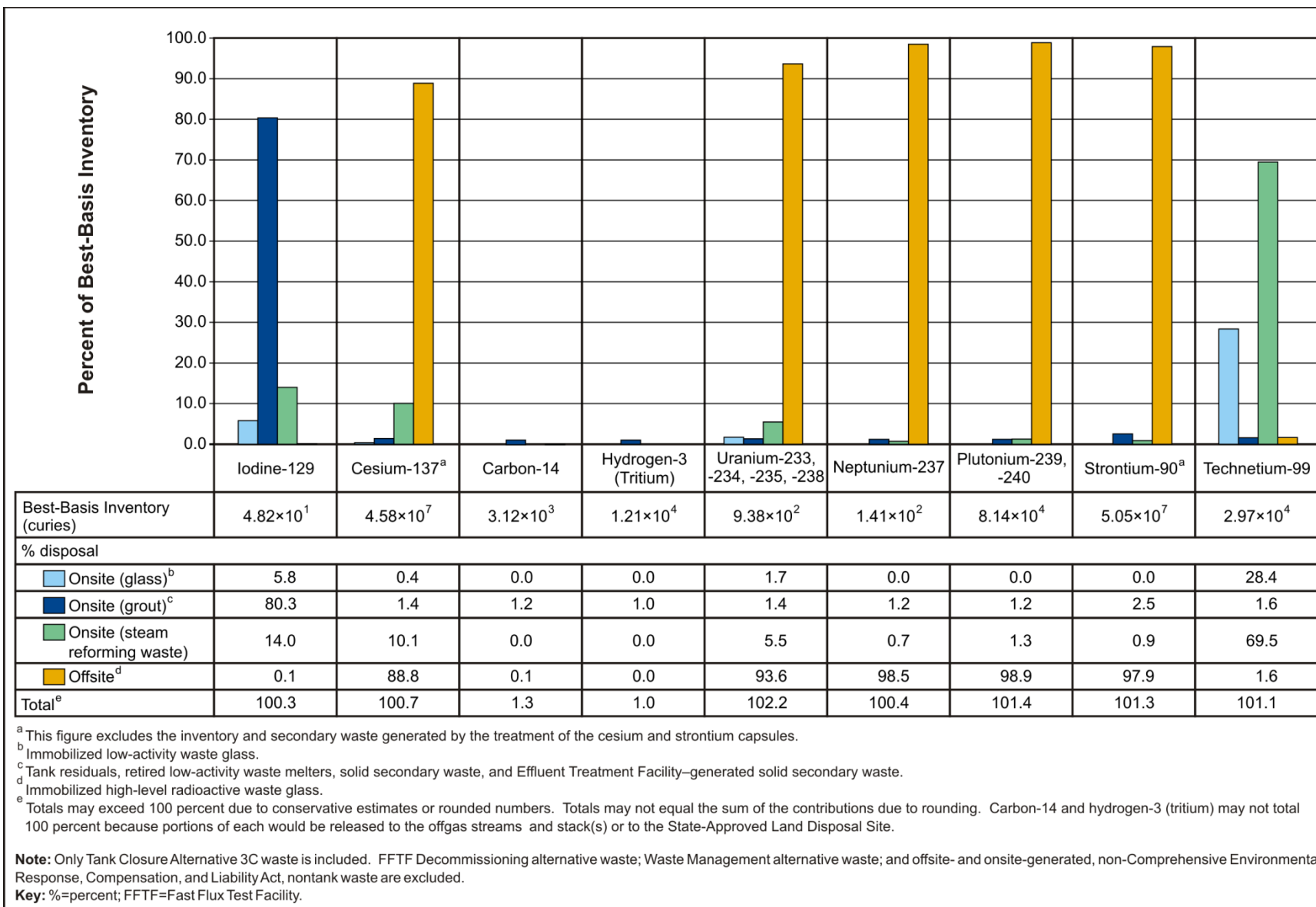


Figure D-33. Tank Closure 3B Distribution of Total Radioactive Constituents of Potential Concern



**Figure D-34. Tank Closure Alternative 3C Distribution of Radioactive Constituents of Potential Concern**



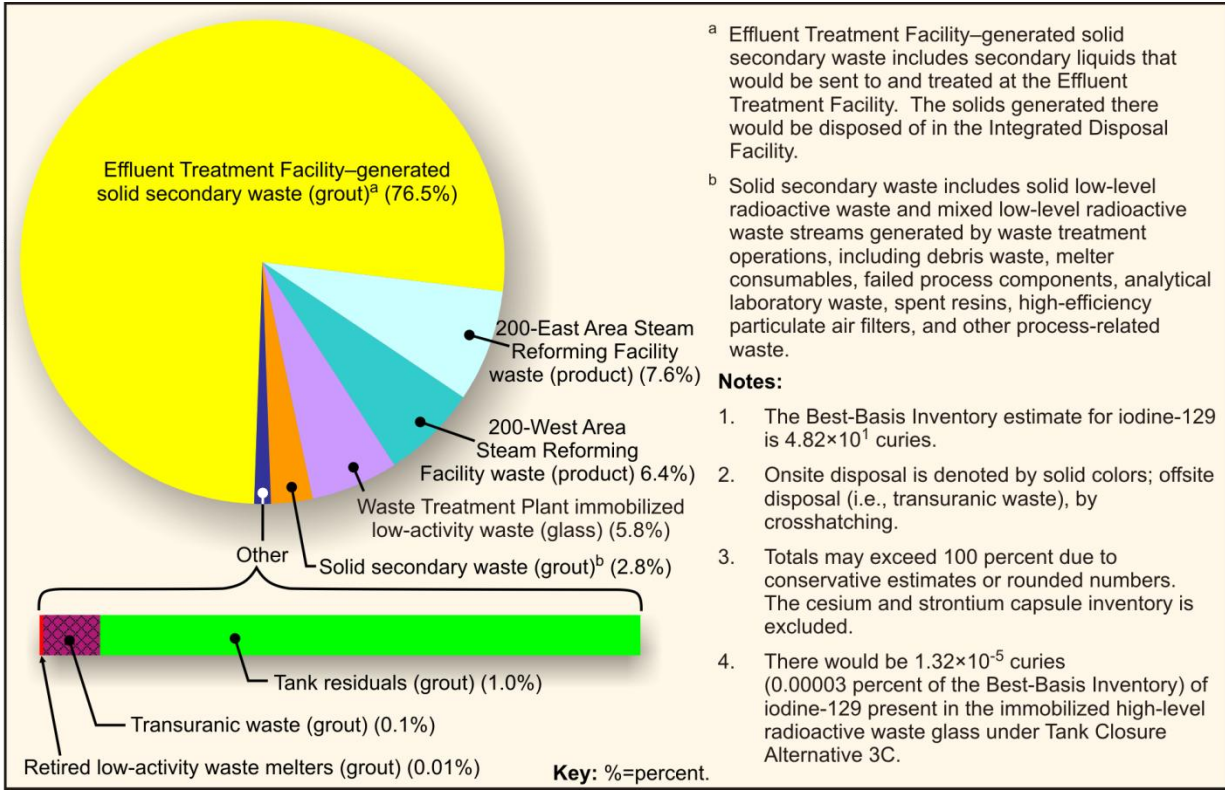


Figure D-35. Tank Closure Alternative 3C Iodine-129 Distribution

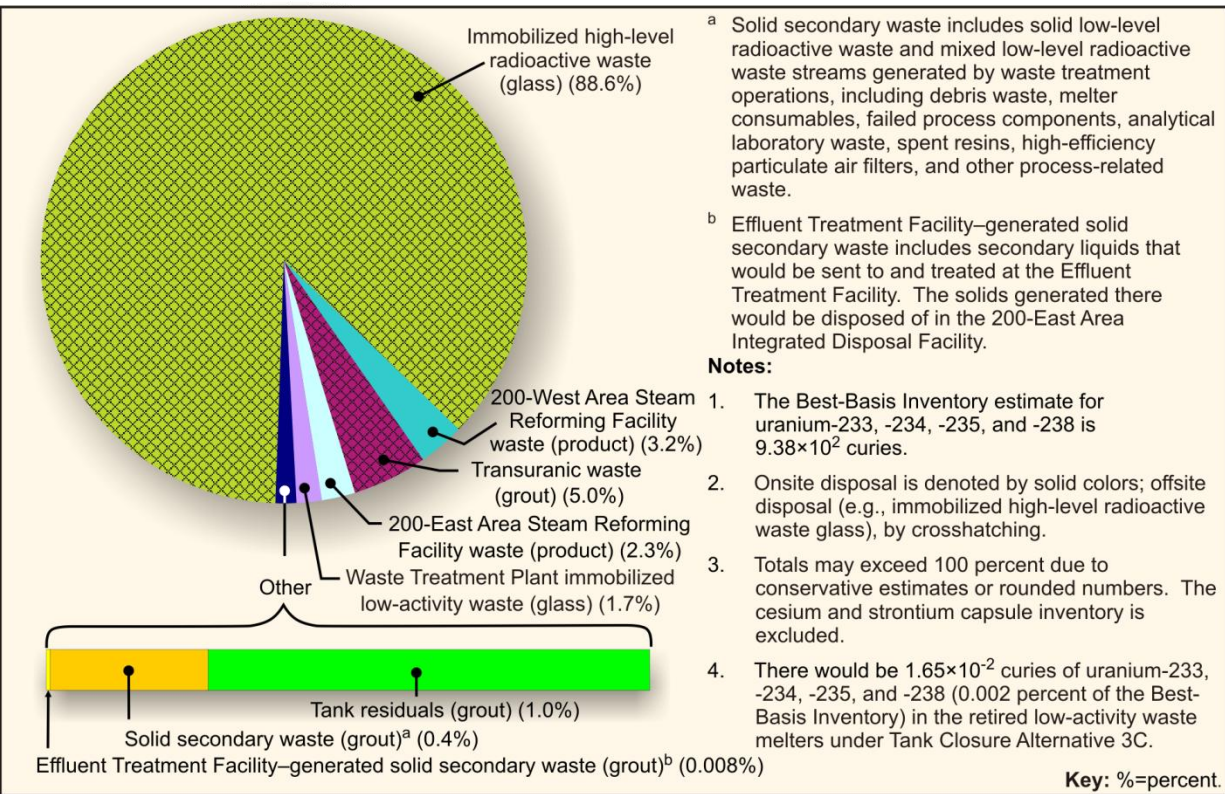
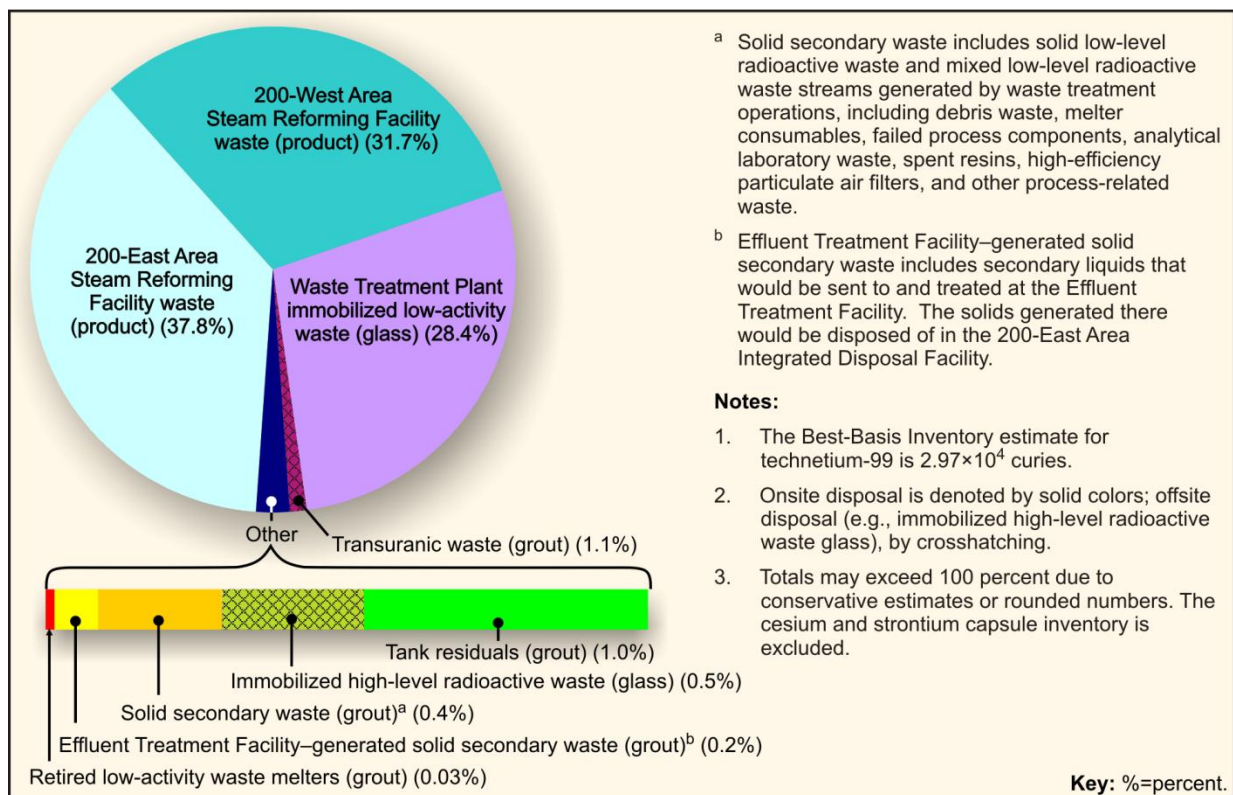
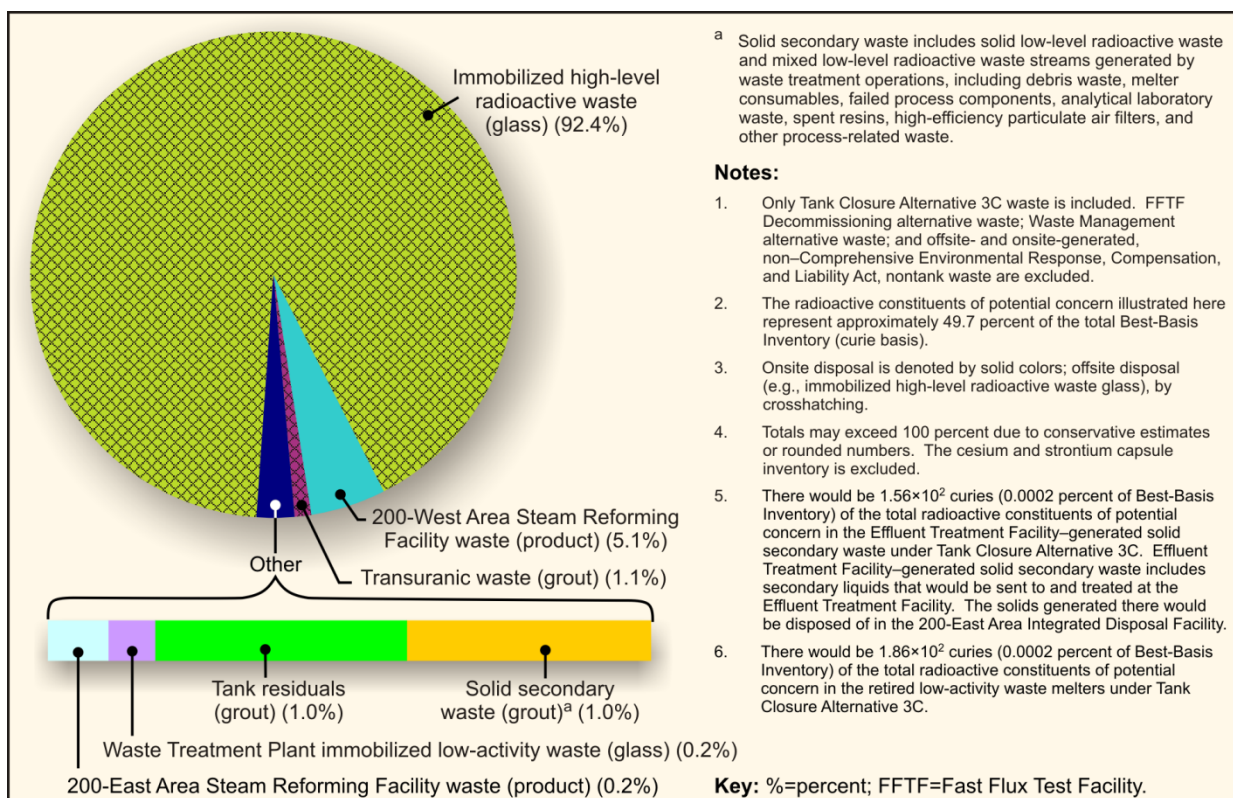


Figure D-36. Tank Closure Alternative 3C Uranium Distribution

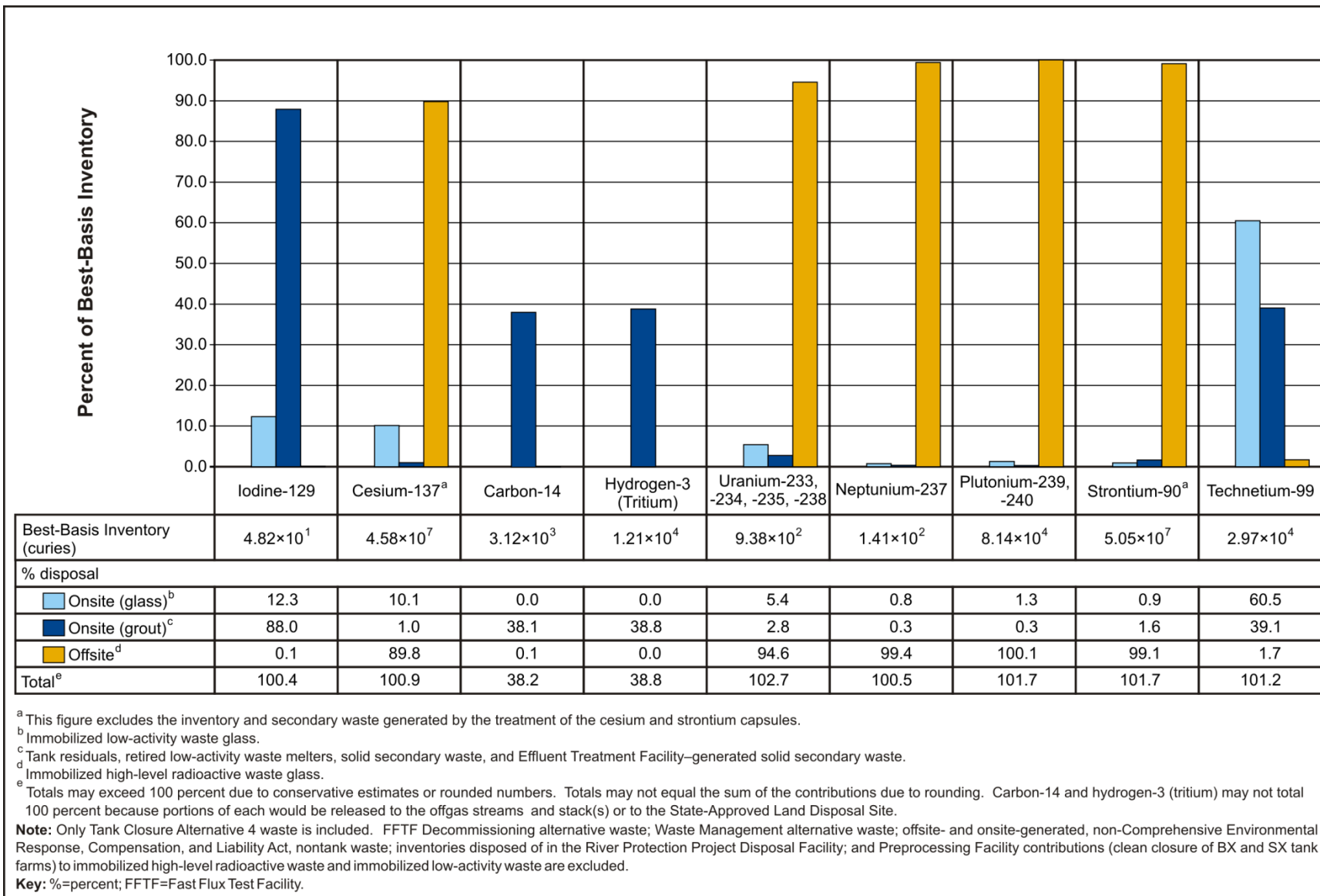


**Figure D-37. Tank Closure Alternative 3C Technetium-99 Distribution**



**Figure D-38. Tank Closure Alternative 3C Distribution of Total Radioactive Constituents of Potential Concern**





**Figure D-39. Tank Closure Alternative 4 Distribution of Radioactive Constituents of Potential Concern**

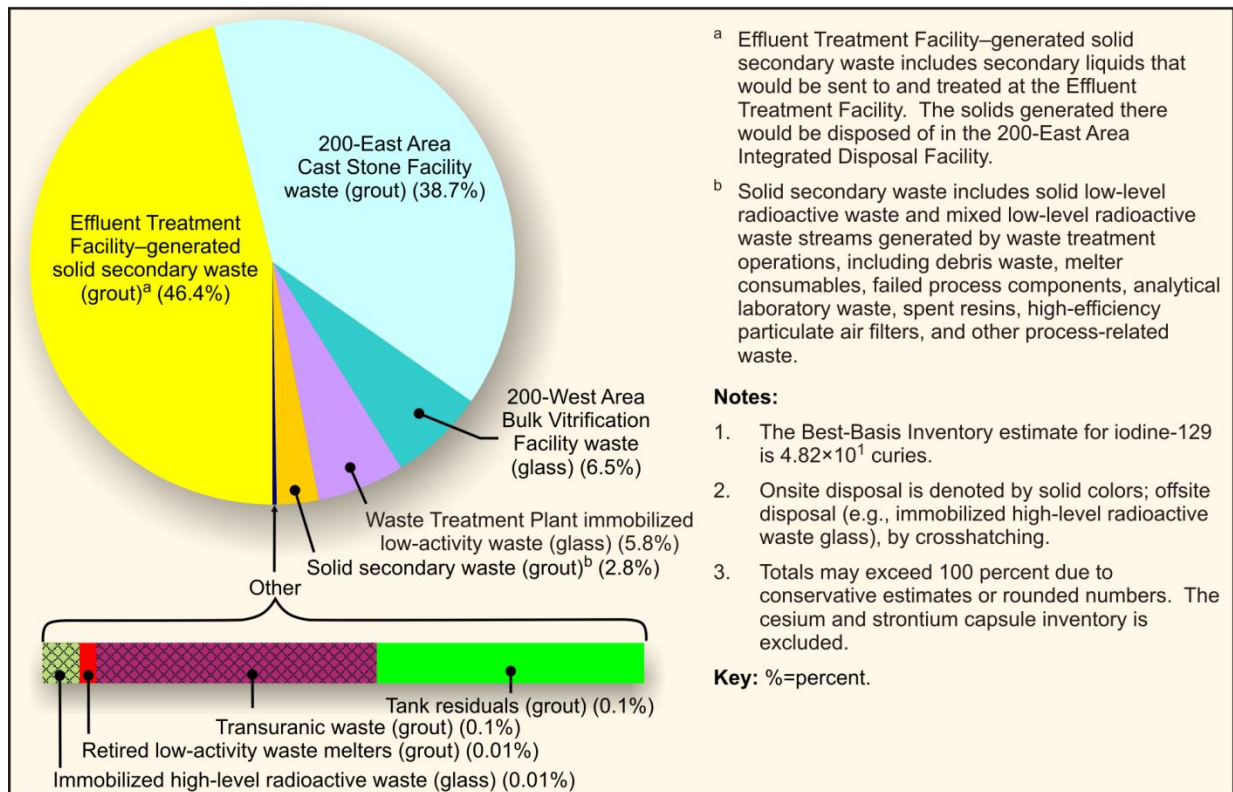


Figure D-40. Tank Closure Alternative 4 Iodine-129 Distribution

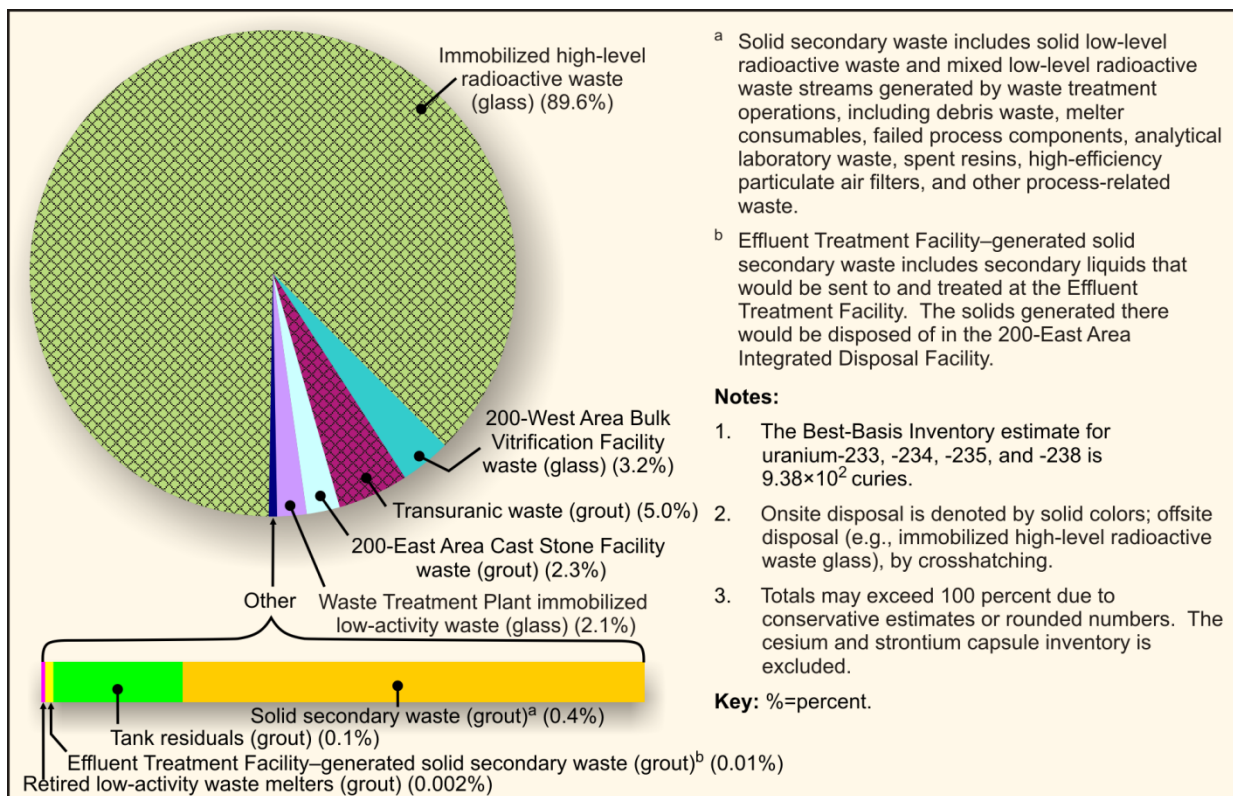


Figure D-41. Tank Closure Alternative 4 Uranium Distribution

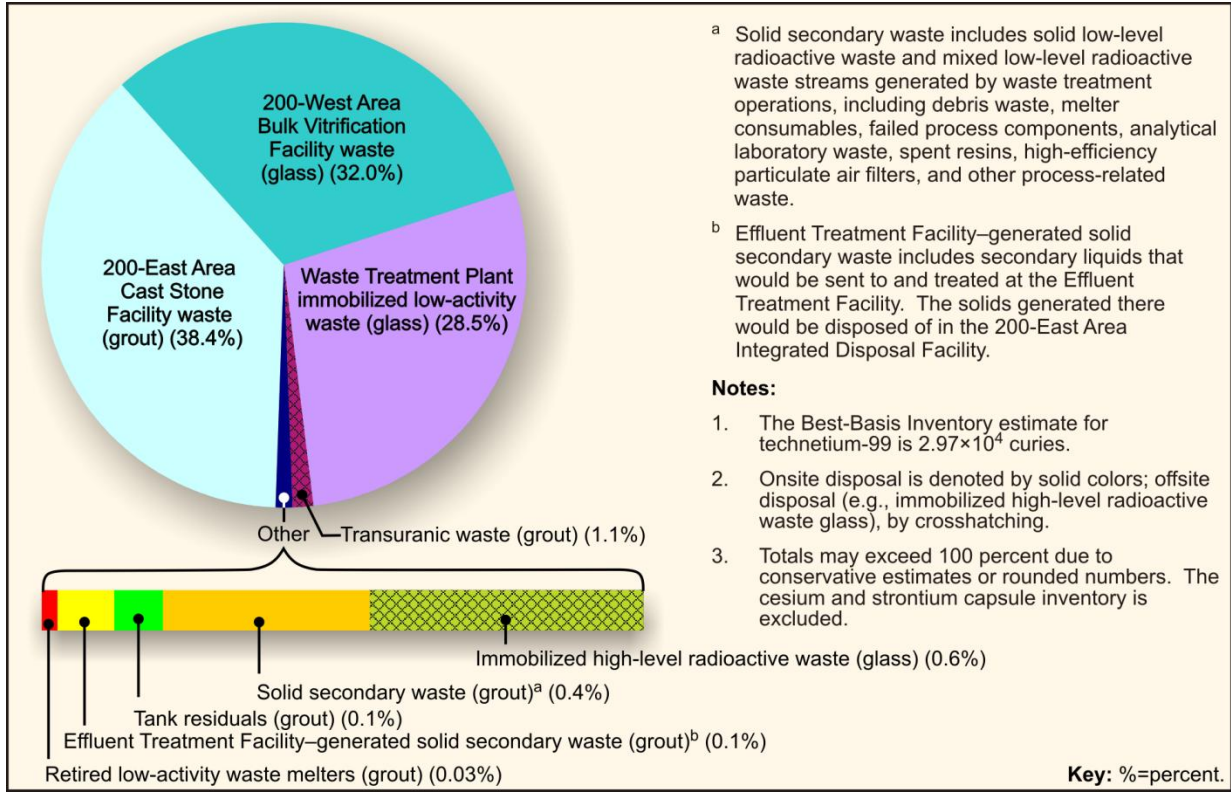


Figure D-42. Tank Closure Alternative 4 Technetium-99 Distribution

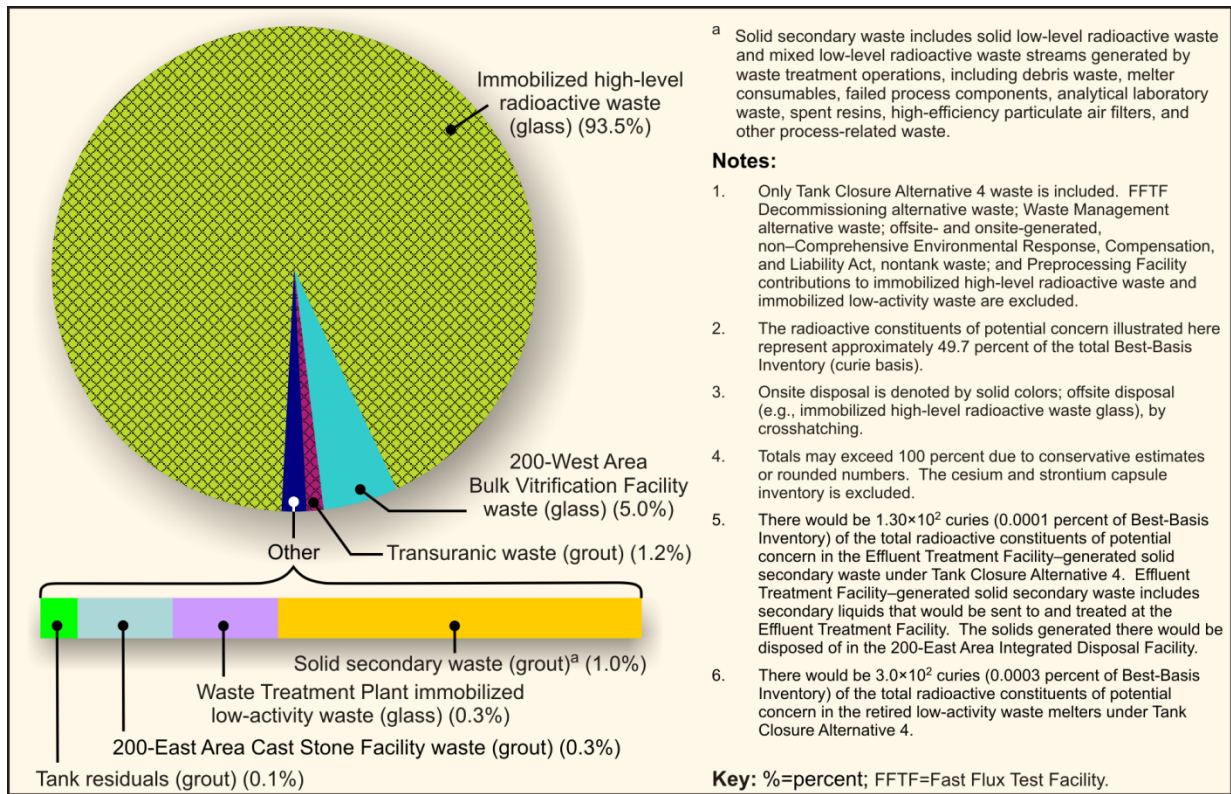
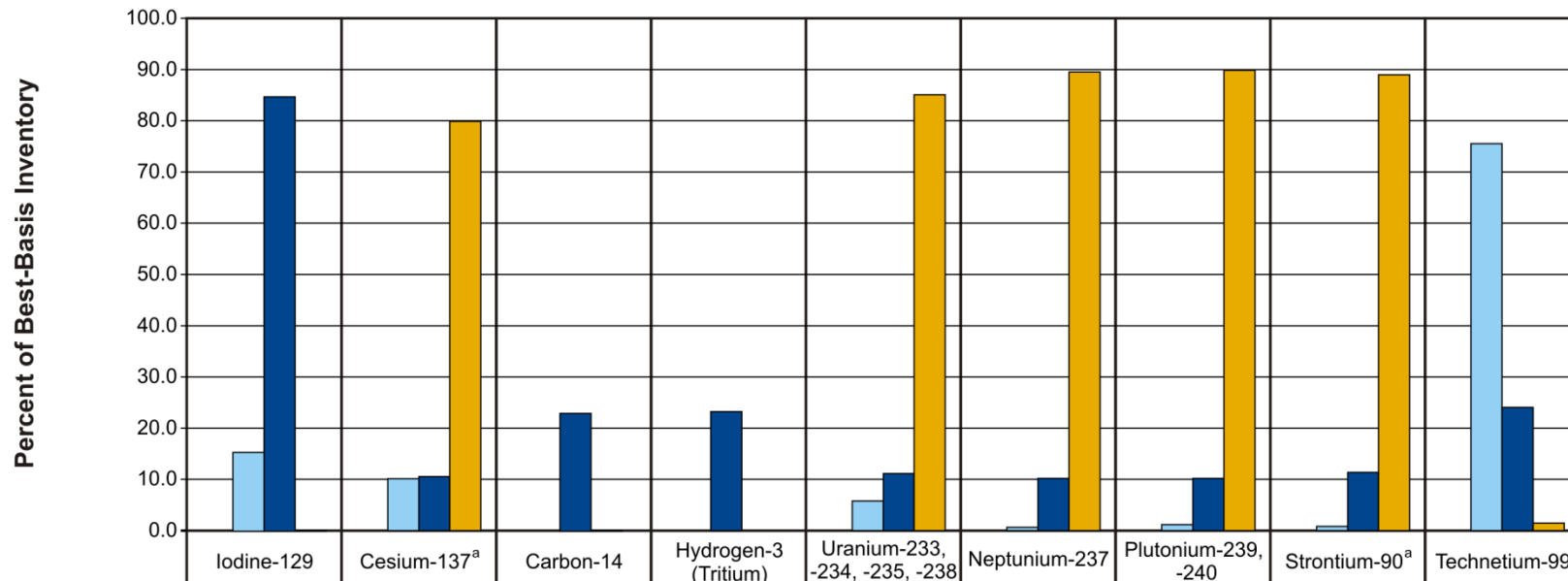


Figure D-43. Tank Closure Alternative 4 Distribution of Total Radioactive Constituents of Potential Concern





Best-Basis Inventory (curies)	Iodine-129	Cesium-137 <sup>a</sup>	Carbon-14	Hydrogen-3 (Tritium)	Uranium-233, -234, -235, -238	Neptunium-237	Plutonium-239, -240	Strontium-90 <sup>a</sup>	Technetium-99
Best-Basis Inventory (curies)	$4.82 \times 10^1$	$4.58 \times 10^7$	$3.12 \times 10^3$	$1.21 \times 10^4$	$9.38 \times 10^2$	$1.41 \times 10^2$	$8.14 \times 10^4$	$5.05 \times 10^7$	$2.97 \times 10^4$
% disposal									
Onsite (glass) <sup>b</sup>	15.4	10.1	0.0	0.0	5.8	0.6	1.2	0.8	75.5
Onsite (grout) <sup>c</sup>	84.8	10.6	23.0	23.2	11.1	10.2	10.2	11.4	24.1
Offsite <sup>d</sup>	0.1	79.9	0.1	0.0	85.1	89.5	89.9	89.0	1.5
Total <sup>e</sup>	100.3	100.6	23.1	23.3	102.0	100.4	101.2	101.2	101.1

<sup>a</sup> This figure excludes the inventory and secondary waste generated by the treatment of the cesium and strontium capsules.

<sup>b</sup> Immobilized low-activity waste glass.

<sup>c</sup> Tank residuals, retired low-activity waste melters, solid secondary waste, and Effluent Treatment Facility-generated solid secondary waste.

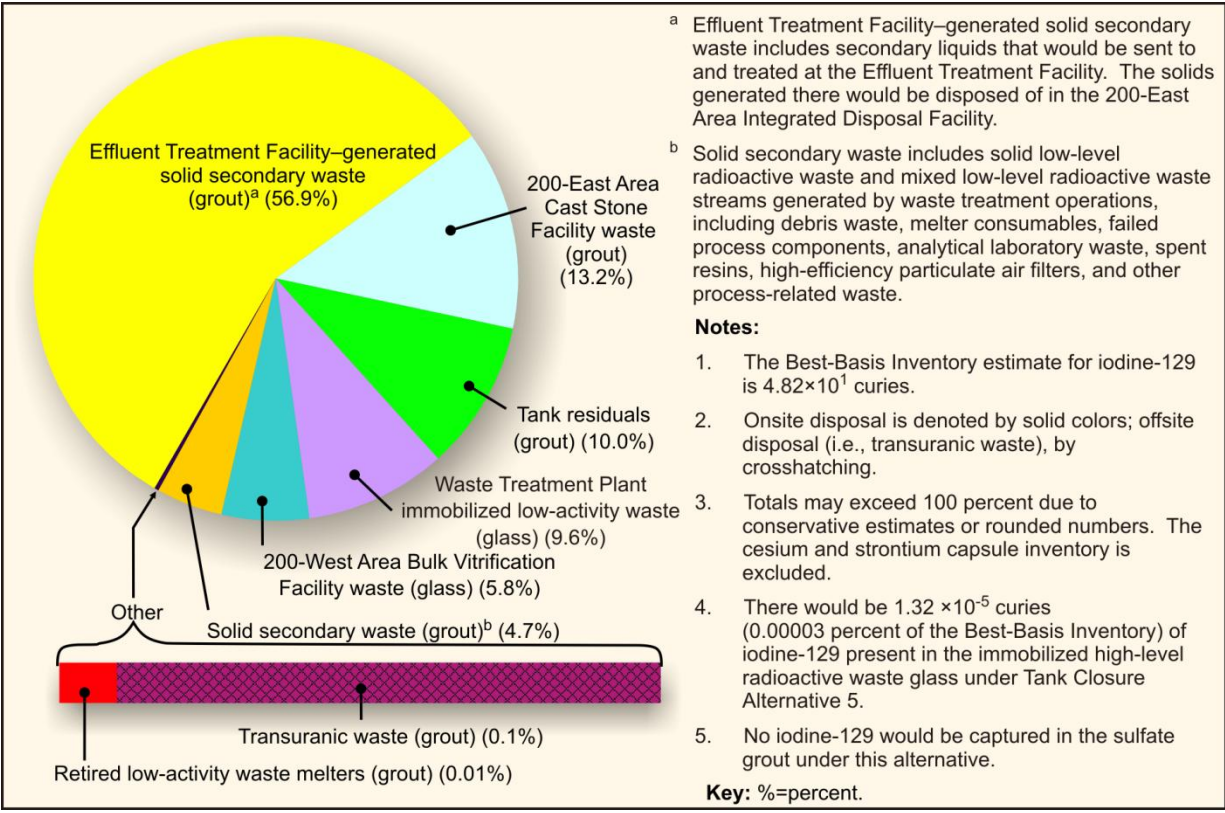
<sup>d</sup> Immobilized high-level radioactive waste glass.

<sup>e</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. Totals may not equal the sum of the contributions due to rounding. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because portions of each would be released to the offgas streams and stack(s) or to the State-Approved Land Disposal Site.

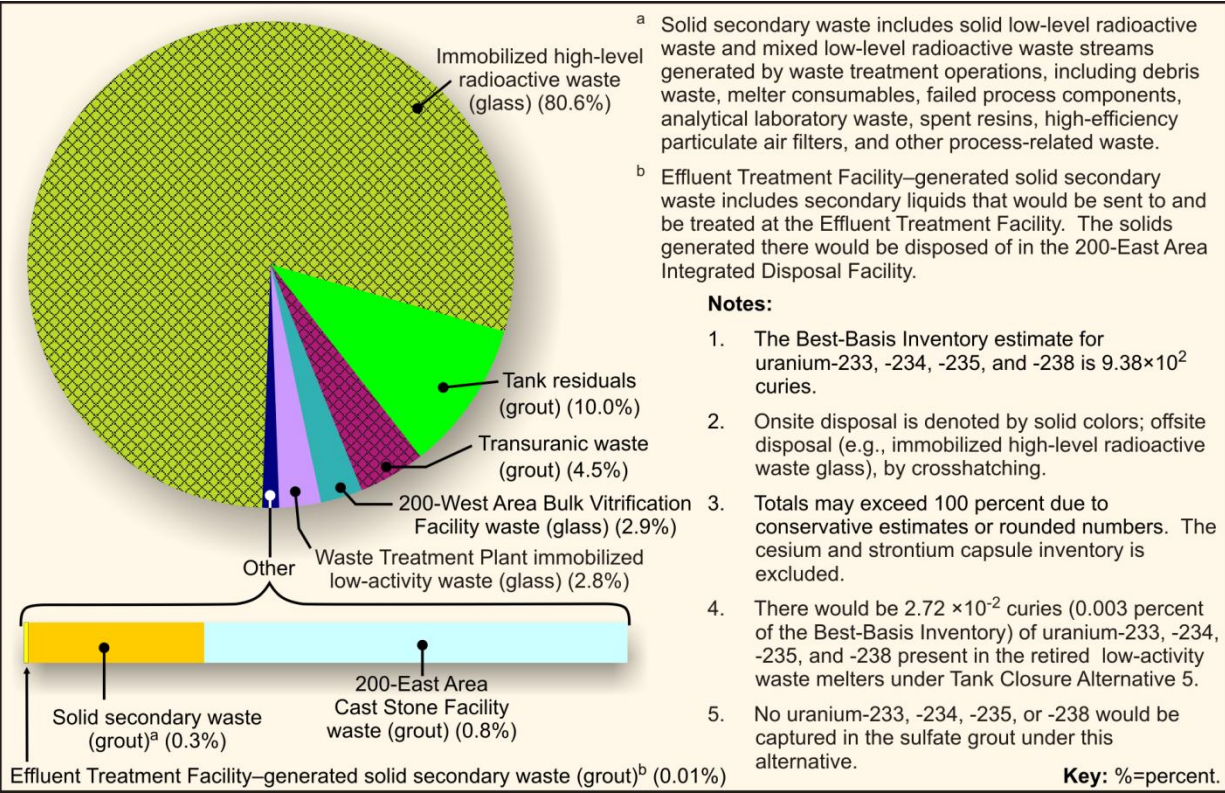
**Note:** Only Tank Closure Alternative 5 waste is included. FFTF Decommissioning alternative waste; Waste Management alternative waste; and offsite- and onsite-generated, non-Comprehensive Environmental Response, Compensation, and Liability Act, nontank waste are excluded.

**Key:** %=percent; FFTF=Fast Flux Test Facility.

**Figure D-44. Tank Closure Alternative 5 Distribution of Radioactive Constituents of Potential Concern**



**Figure D-45. Tank Closure Alternative 5 Iodine-129 Distribution**



**Figure D-46. Tank Closure Alternative 5 Uranium Distribution**



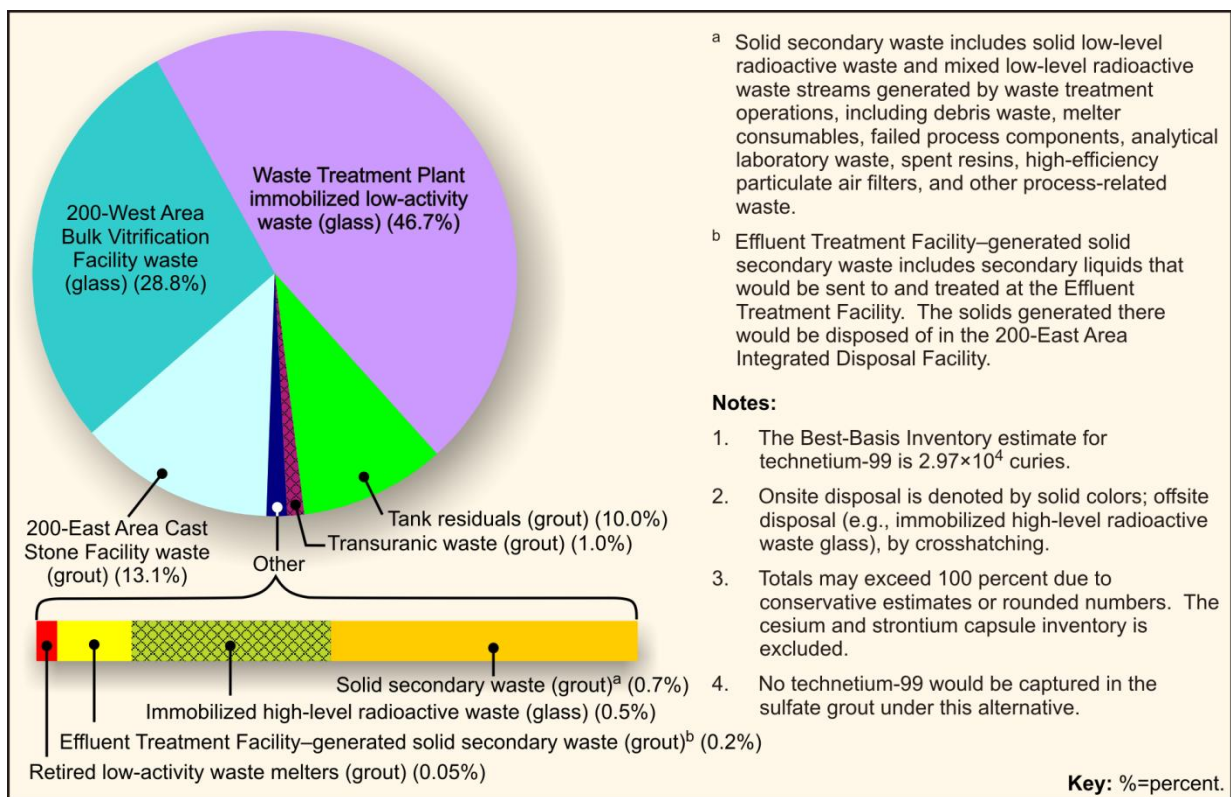


Figure D-47. Tank Closure Alternative 5 Technetium-99 Distribution

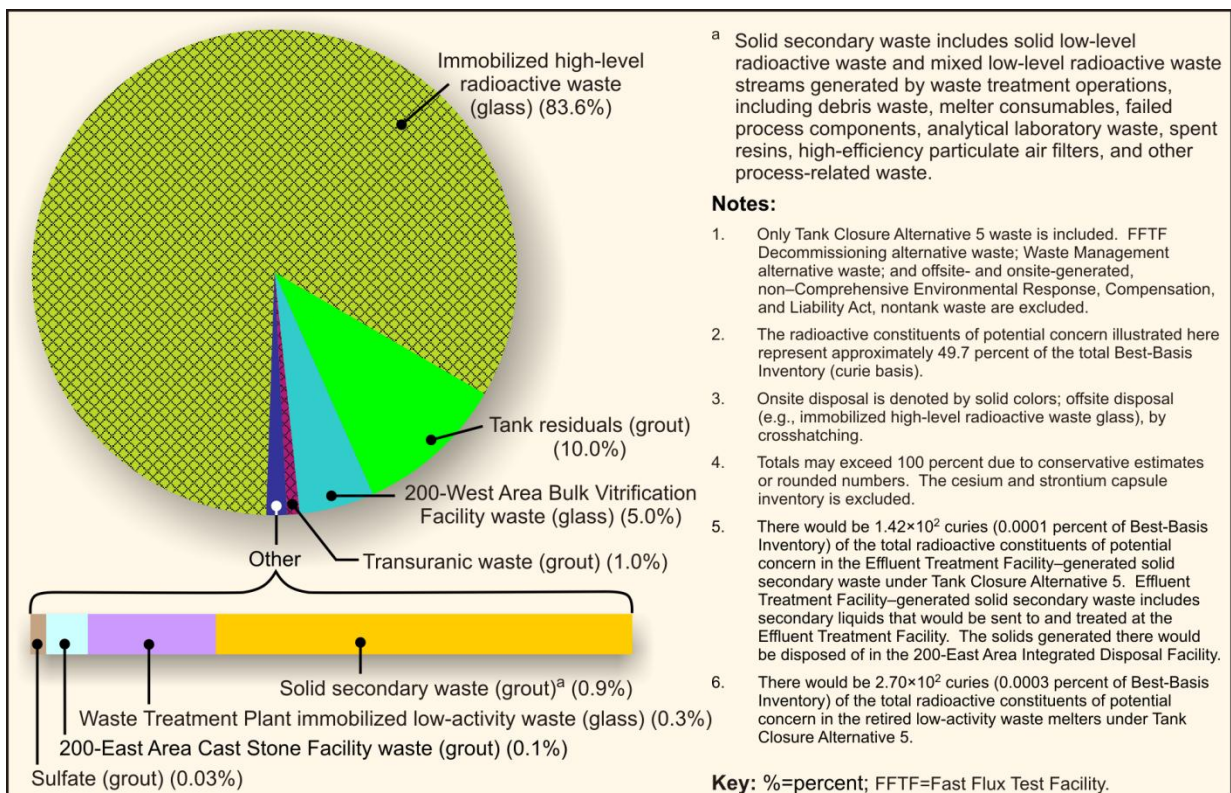
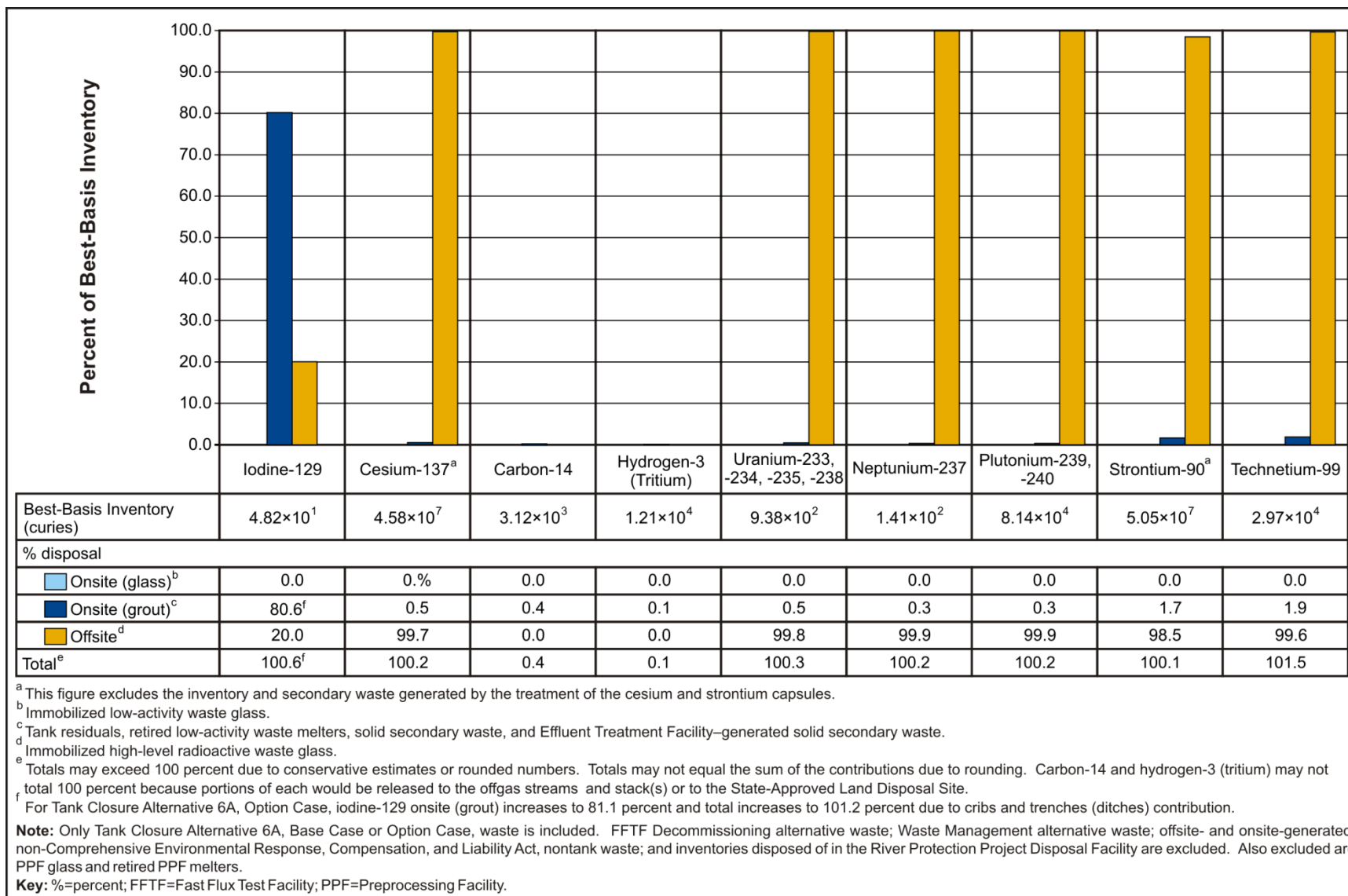
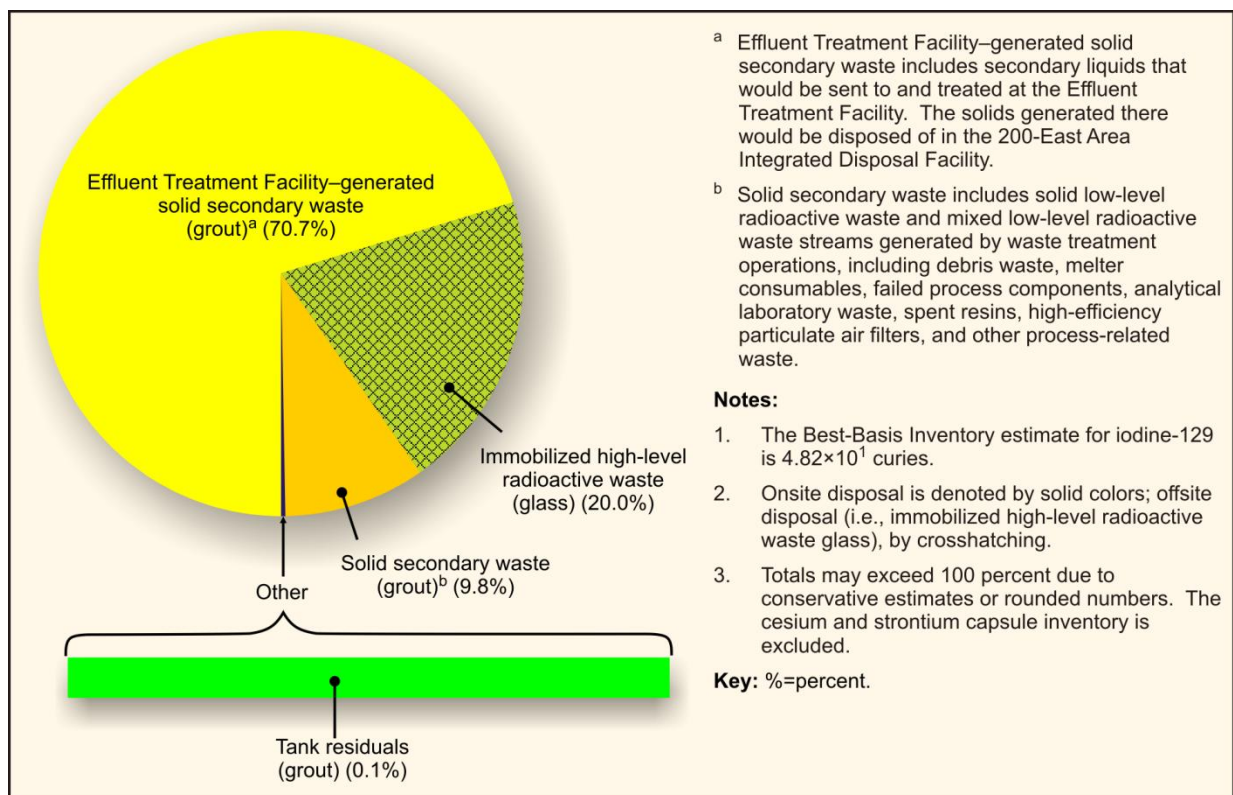


Figure D-48. Tank Closure Alternative 5 Distribution of Total Radioactive Constituents of Potential Concern

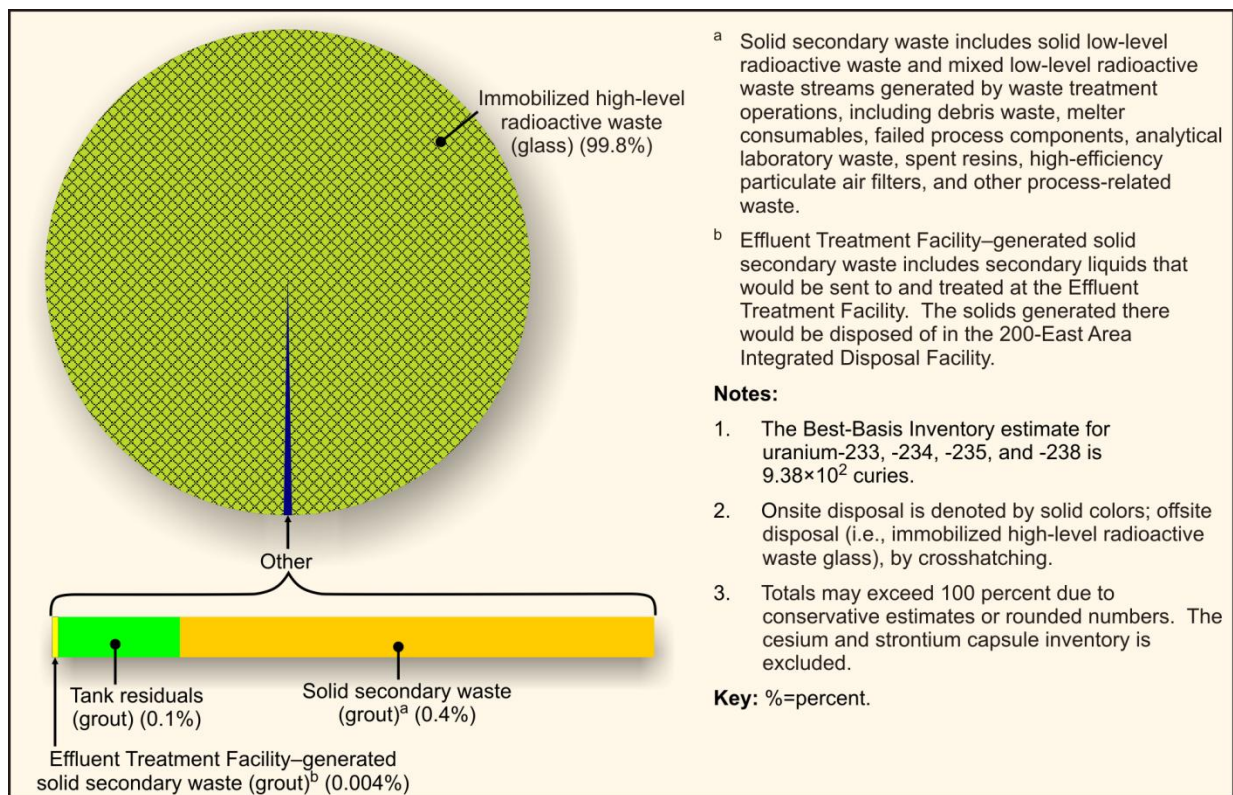


**Figure D-49. Tank Closure Alternative 6A, Base Case or Option Case, Distribution of Radioactive Constituents of Potential Concern**





**Figure D-50. Tank Closure Alternative 6A, Base Case or Option Case, Iodine-129 Distribution**



**Figure D-51. Tank Closure Alternative 6A, Base Case or Option Case, Uranium Distribution**



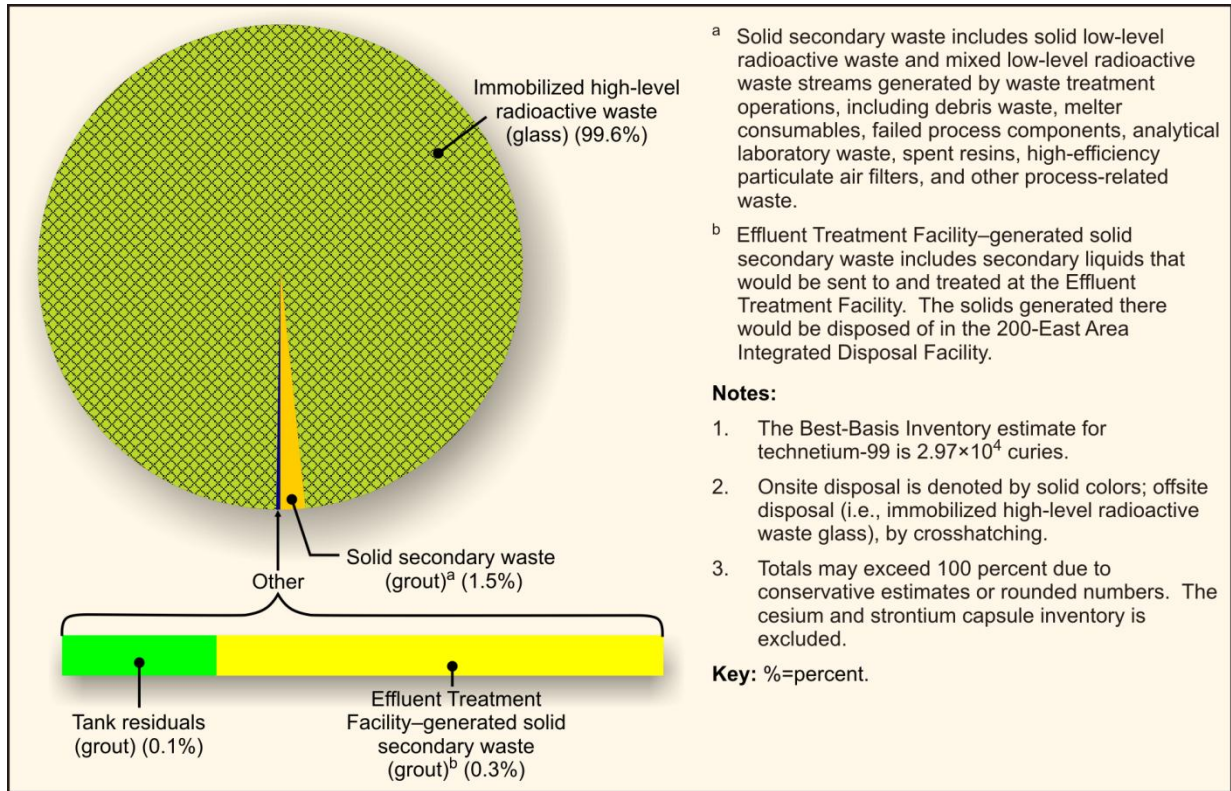


Figure D-52. Tank Closure Alternative 6A, Base Case or Option Case, Technetium-99 Distribution

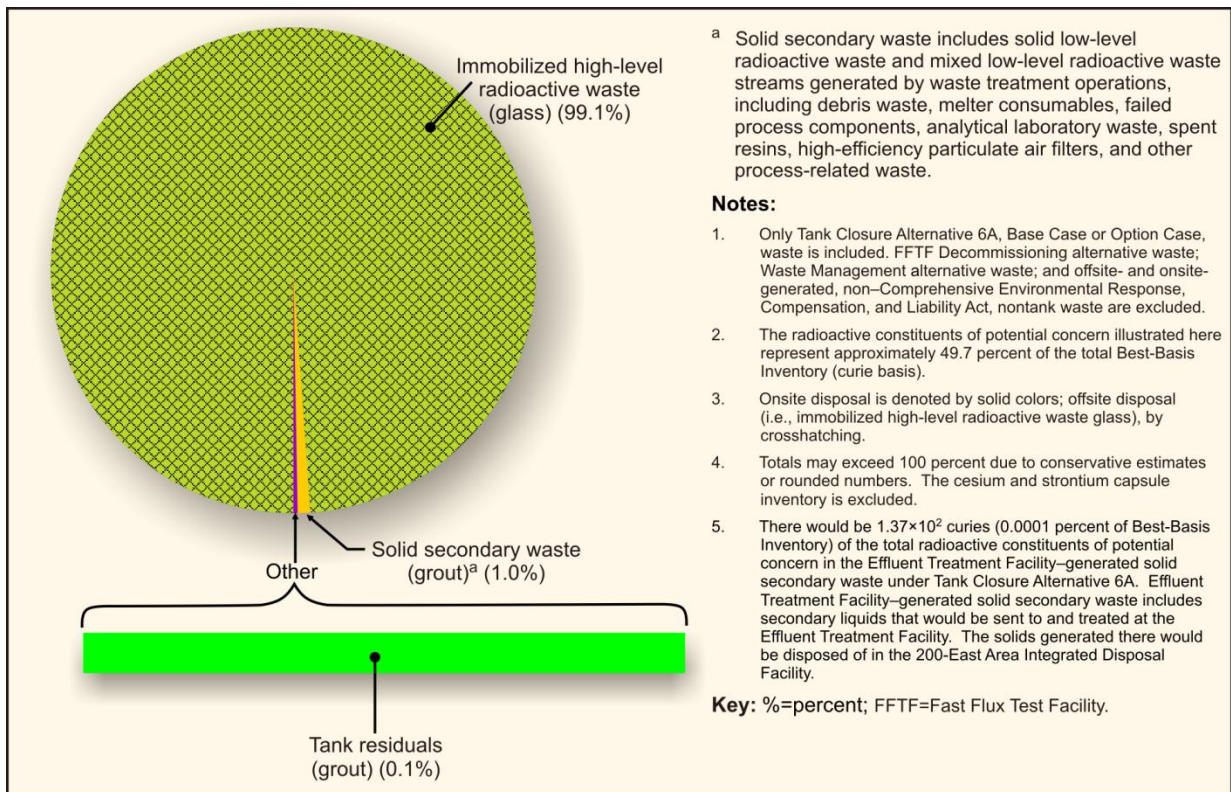
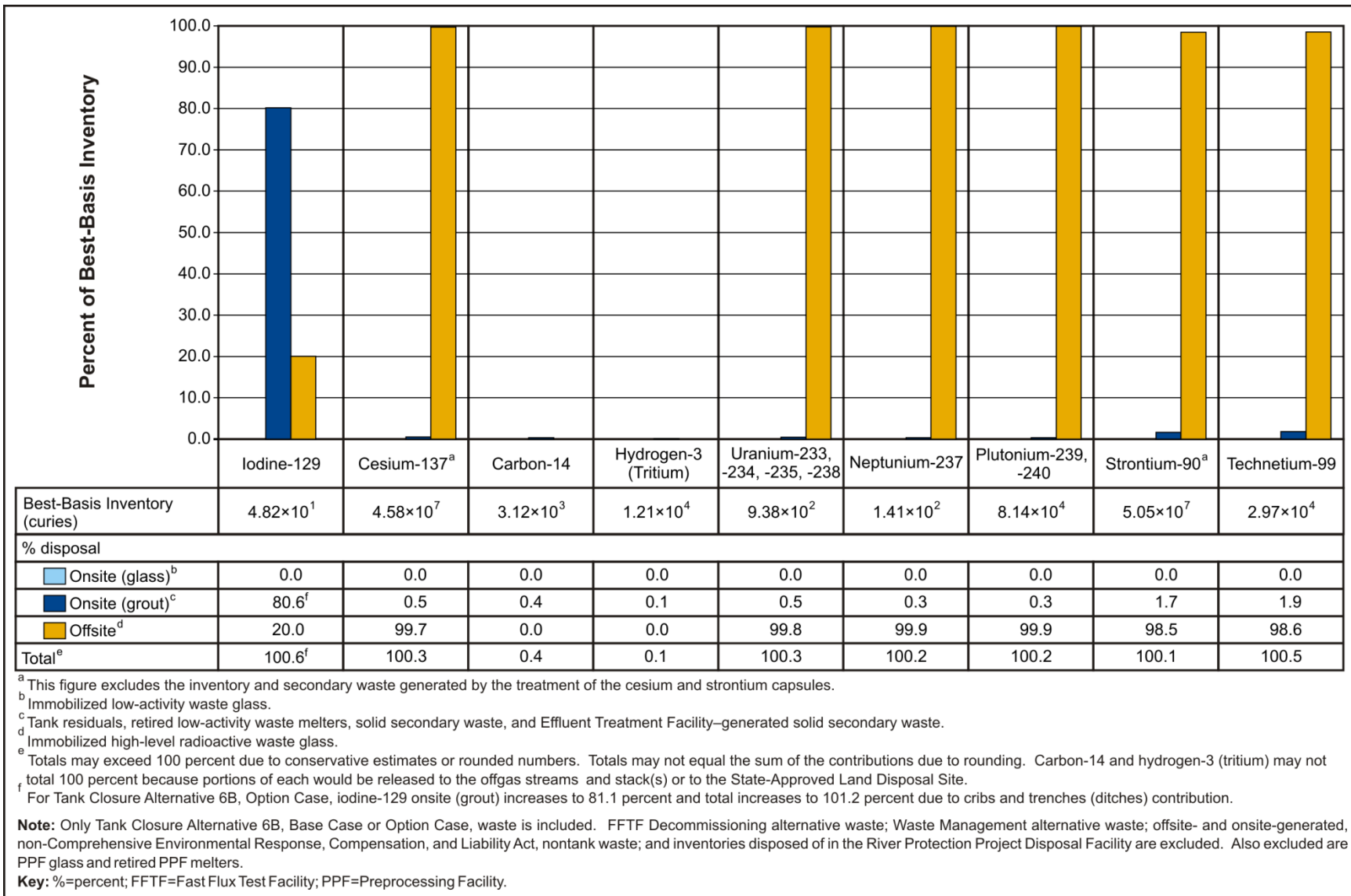


Figure D-53. Tank Closure Alternative 6A, Base Case or Option Case, Distribution of Total Radioactive Constituents of Potential Concern



**Figure D-54. Tank Closure Alternative 6B, Base Case or Option Case, Distribution of Radioactive Constituents of Potential Concern**



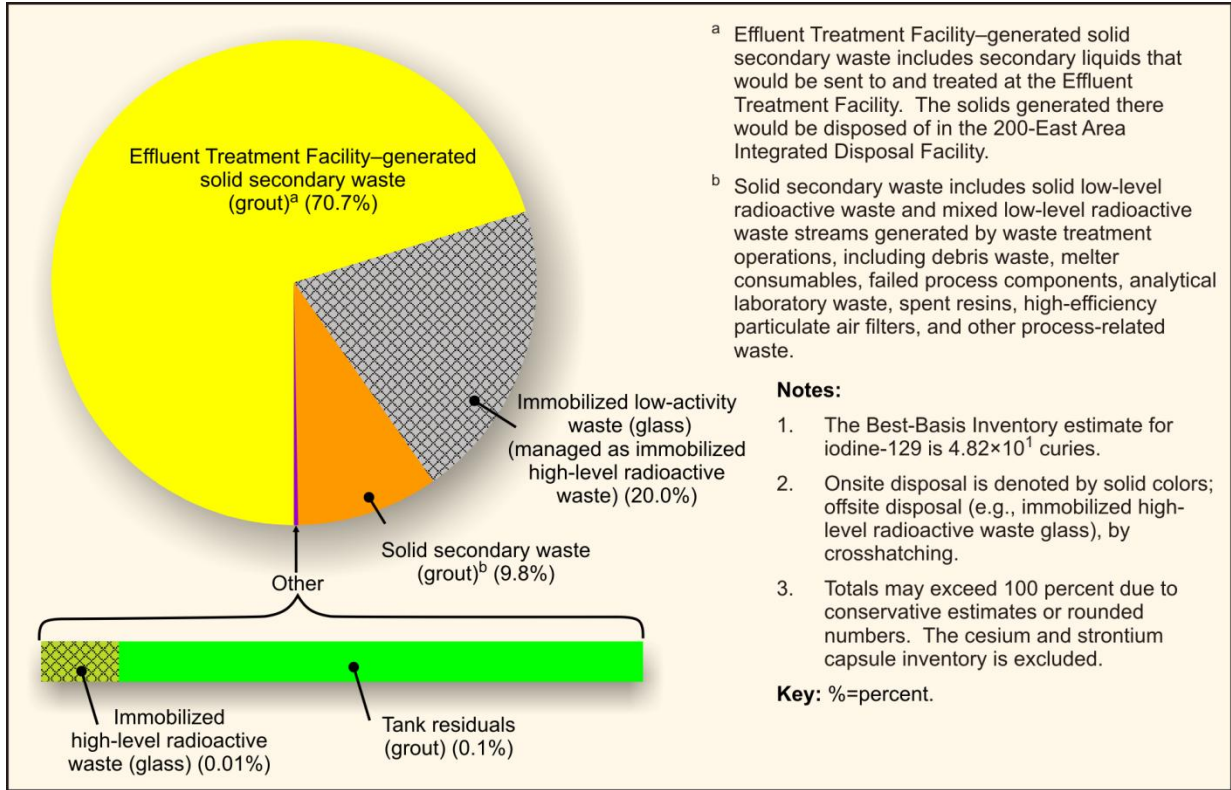


Figure D-55. Tank Closure Alternative 6B, Base Case or Option Case, Iodine-129 Distribution

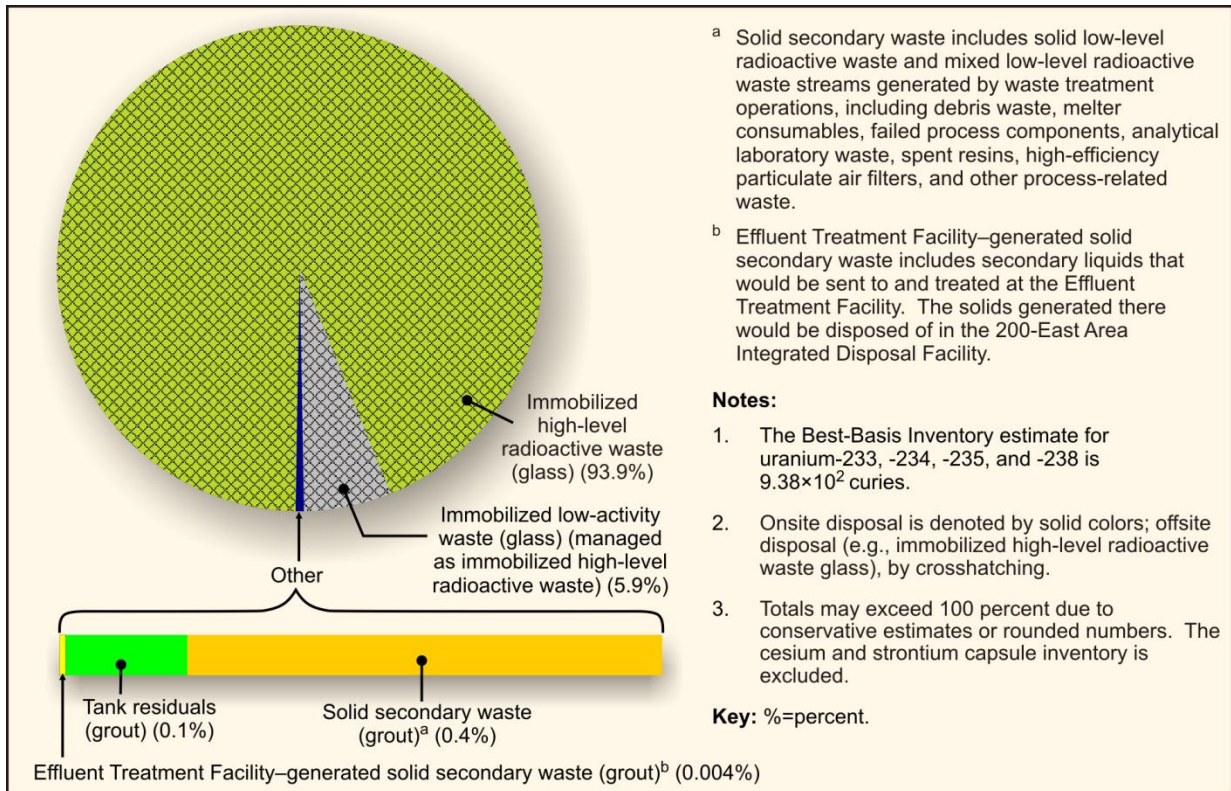


Figure D-56. Tank Closure Alternative 6B, Base Case or Option Case, Uranium Distribution



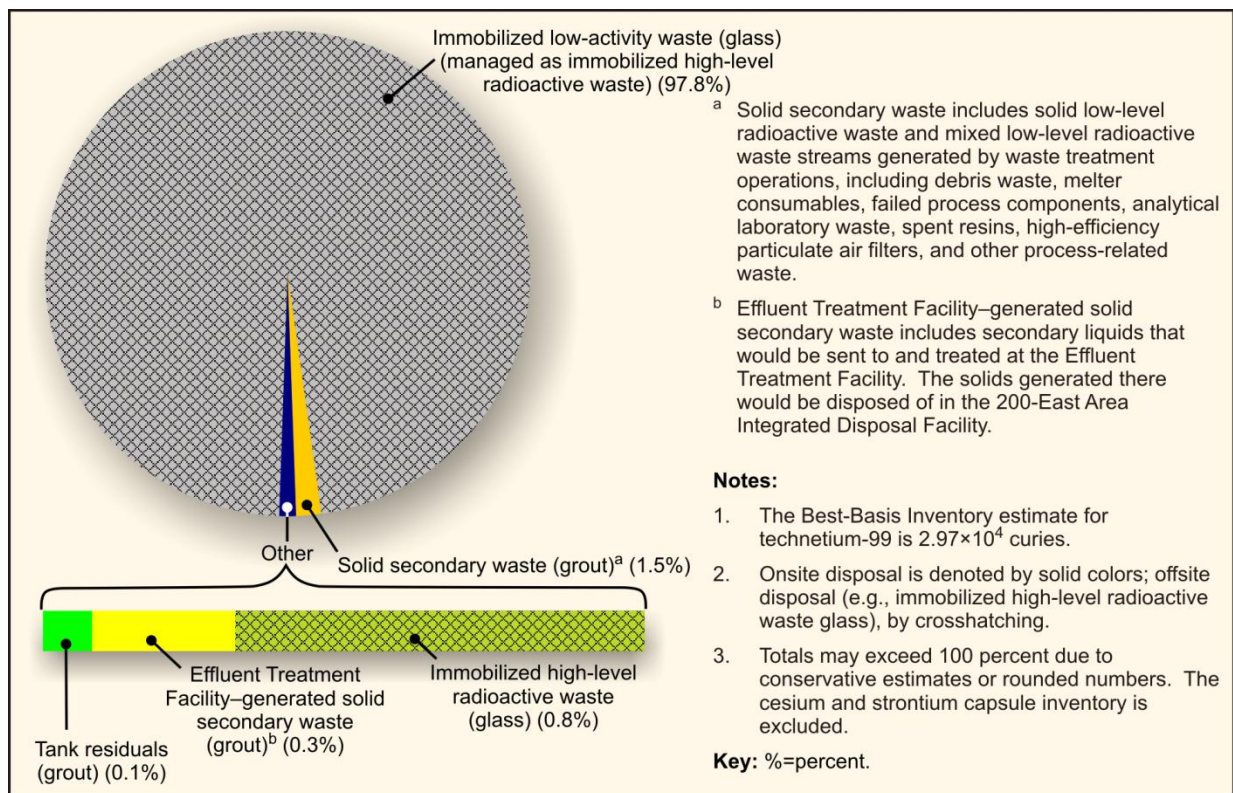


Figure D-57. Tank Closure Alternative 6B, Base Case or Option Case, Technetium-99 Distribution

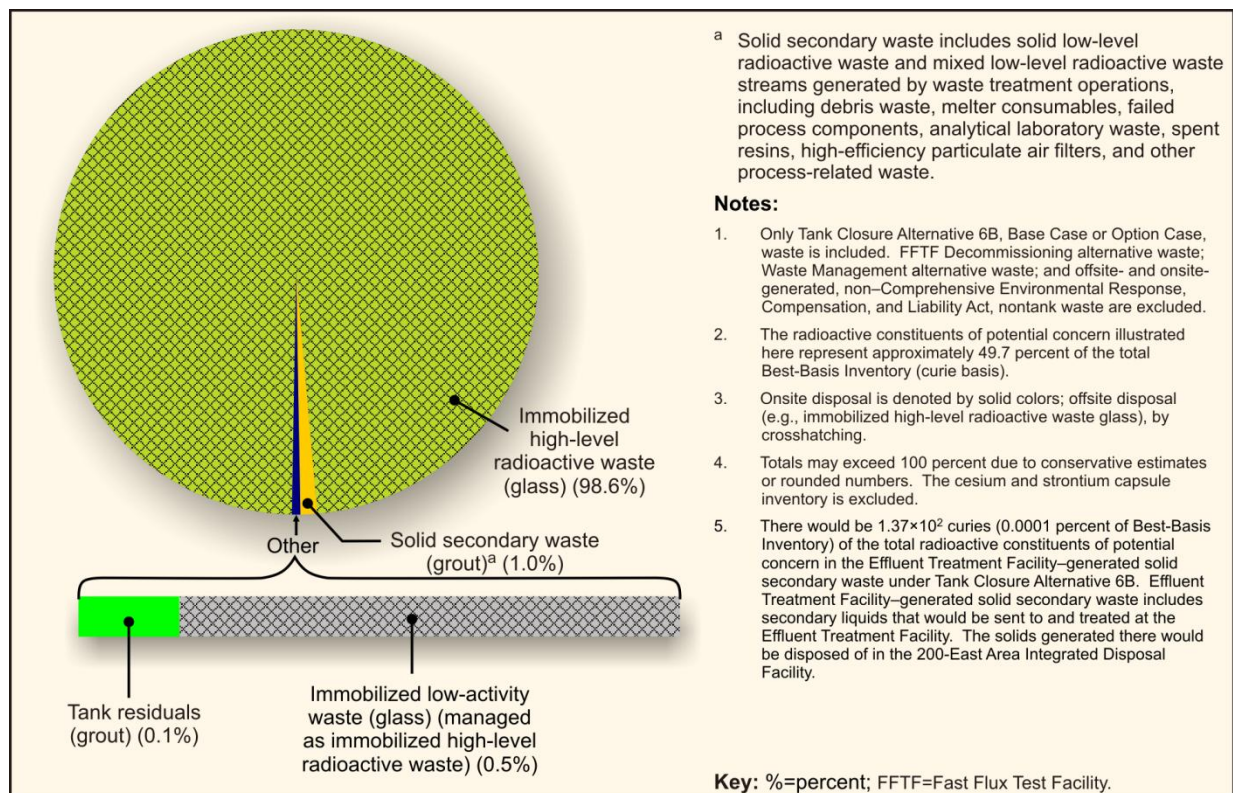
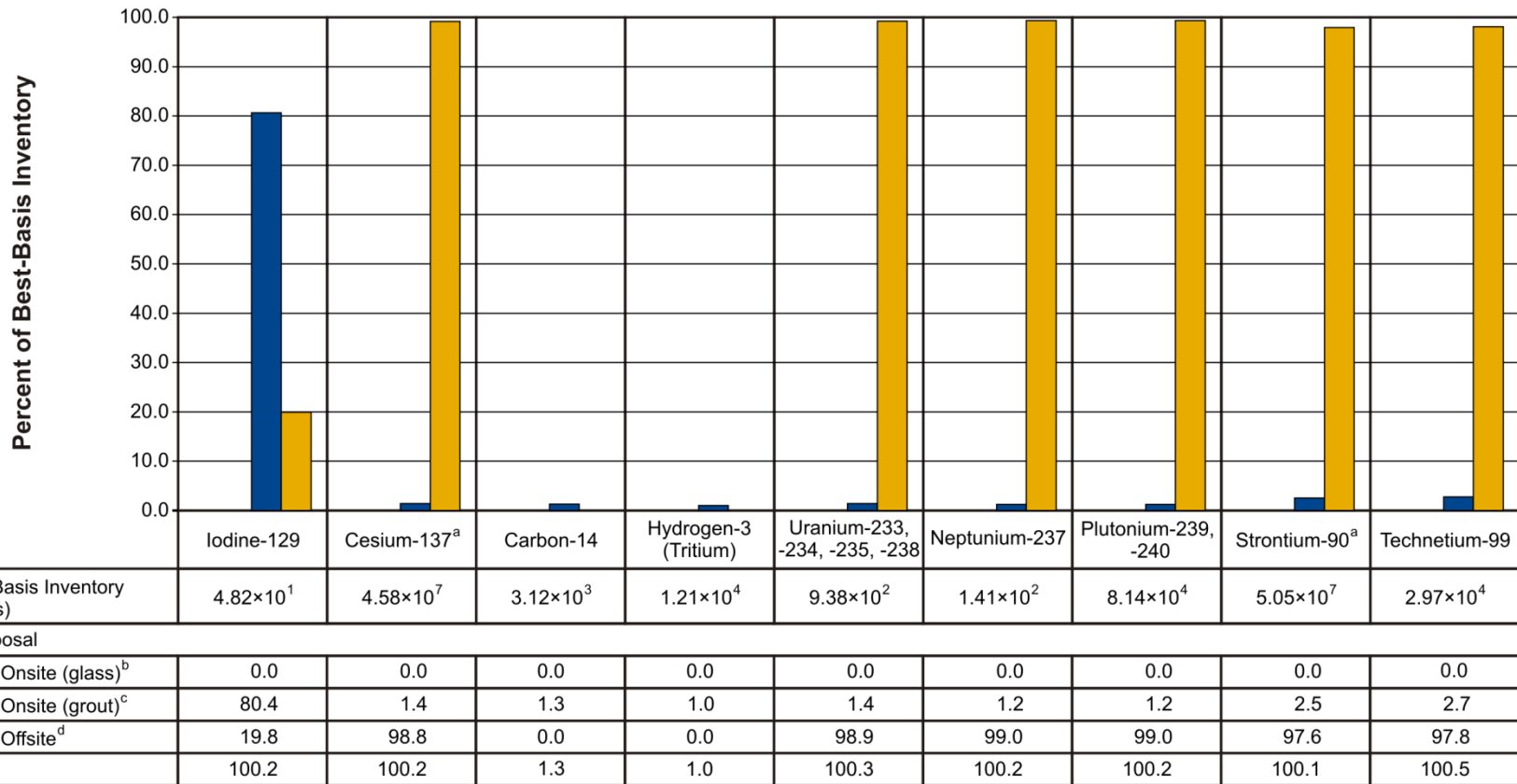


Figure D-58. Tank Closure Alternative 6B, Base Case or Option Case, Distribution of Total Radioactive Constituents of Potential Concern



<sup>a</sup> This figure excludes the inventory and secondary waste generated by the treatment of the cesium and strontium capsules.

<sup>b</sup> Immobilized low-activity waste glass.

<sup>c</sup> Tank residuals, retired low-activity waste melters, solid secondary waste, and Effluent Treatment Facility-generated solid secondary waste.

<sup>d</sup> Immobilized high-level radioactive waste glass.

<sup>e</sup> Totals may exceed 100 percent due to conservative estimates or rounded numbers. Totals may not equal the sum of the contributions due to rounding. Carbon-14 and hydrogen-3 (tritium) may not total 100 percent because portions of each would be released to the offgas streams and stack(s) or to the State-Approved Land Disposal Site.

**Note:** Only Tank Closure Alternative 6C waste is included. FFTF Decommissioning alternative waste; Waste Management alternative waste; and offsite- and onsite-generated, non-Comprehensive Environmental Response, Compensation, and Liability Act, nontank waste are excluded.

**Key:** %=percent; FFTF=Fast Flux Test Facility.

**Figure D-59. Tank Closure Alternative 6C Distribution of Radioactive Constituents of Potential Concern**



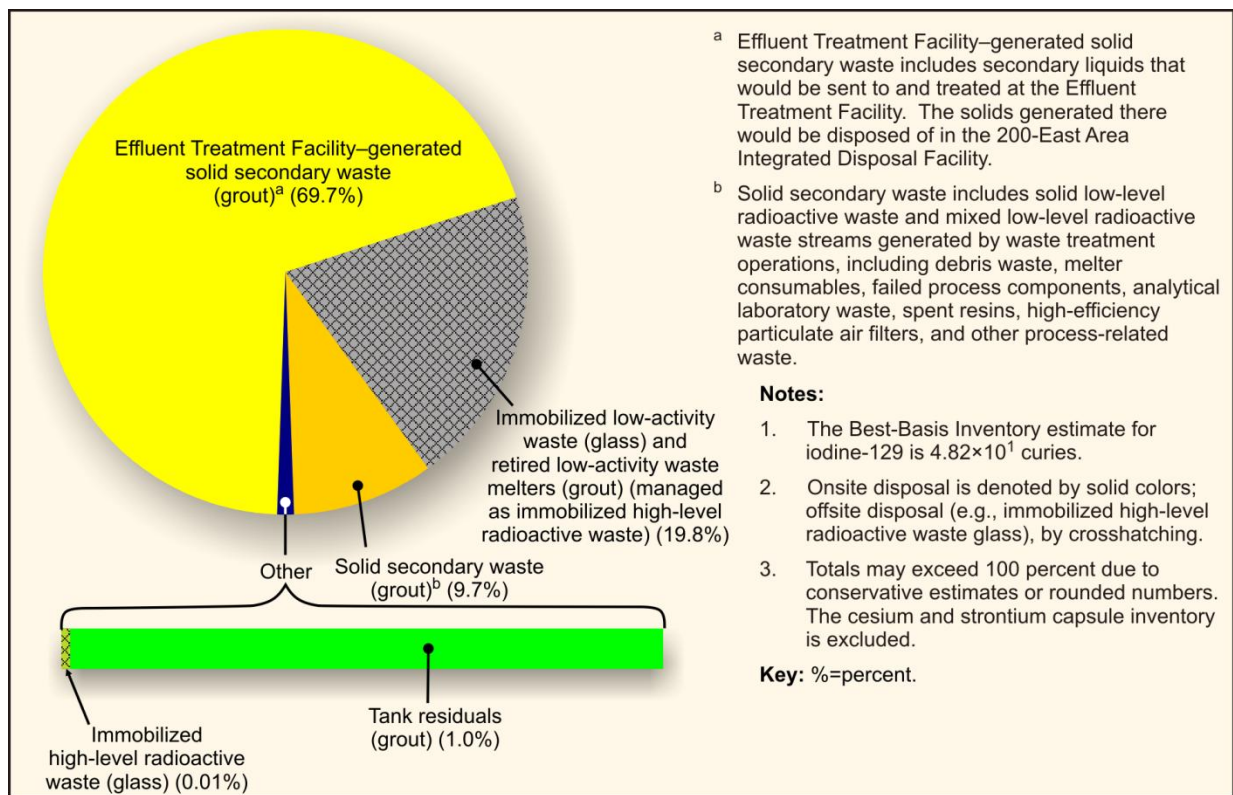


Figure D-60. Tank Closure Alternative 6C Iodine-129 Distribution

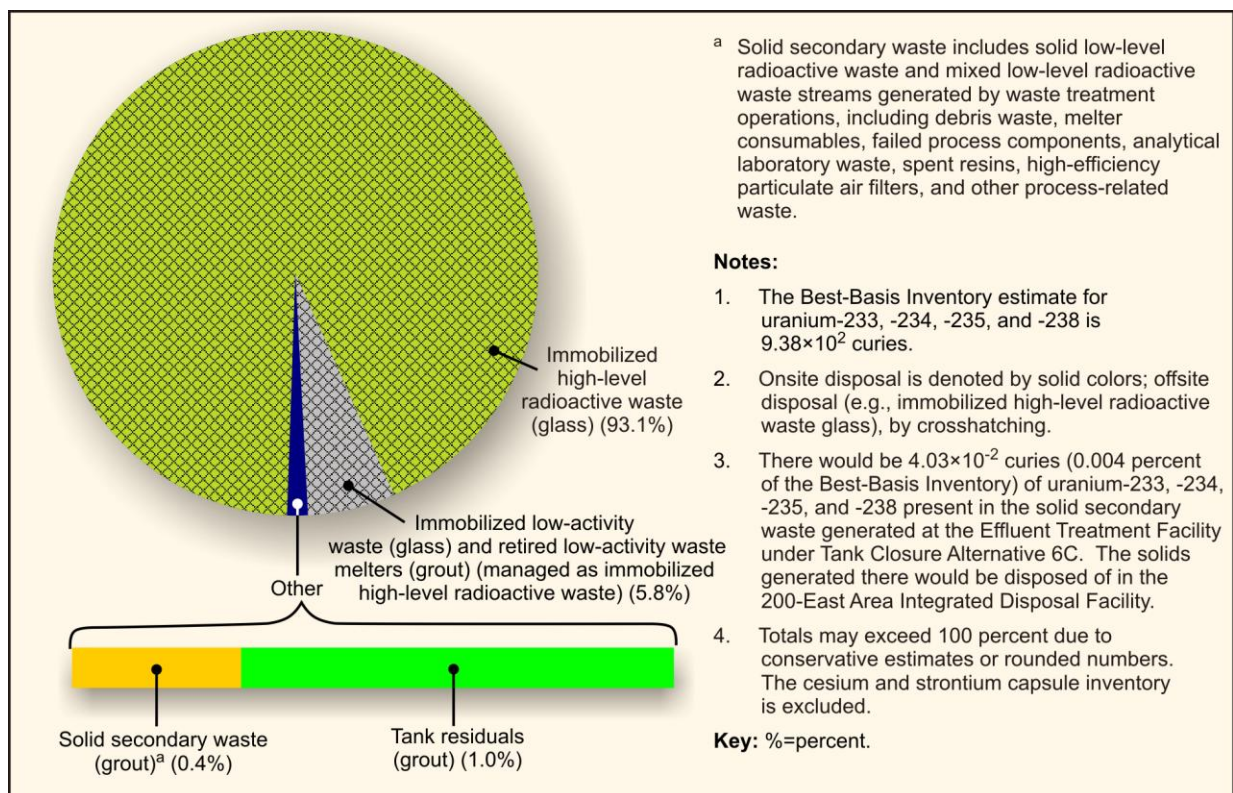


Figure D-61. Tank Closure Alternative 6C Uranium Distribution



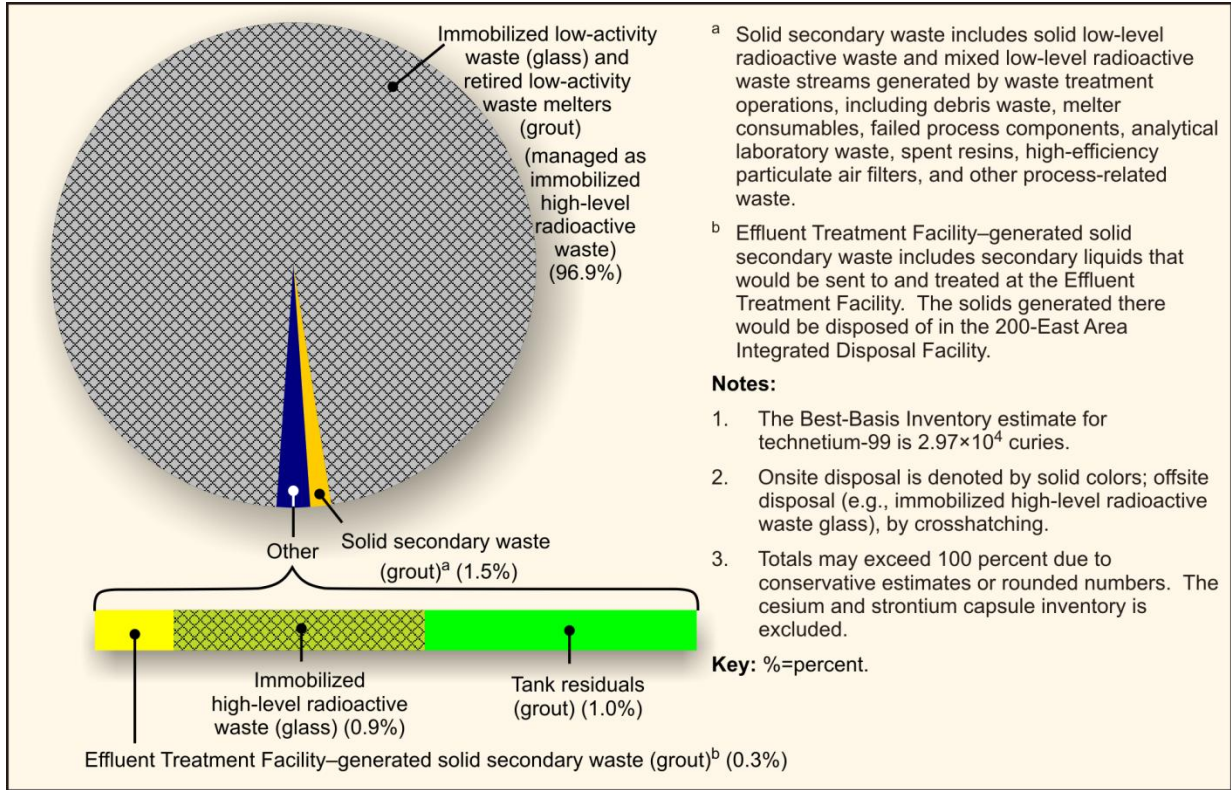


Figure D-62. Tank Closure Alternative 6C Technetium-99 Distribution

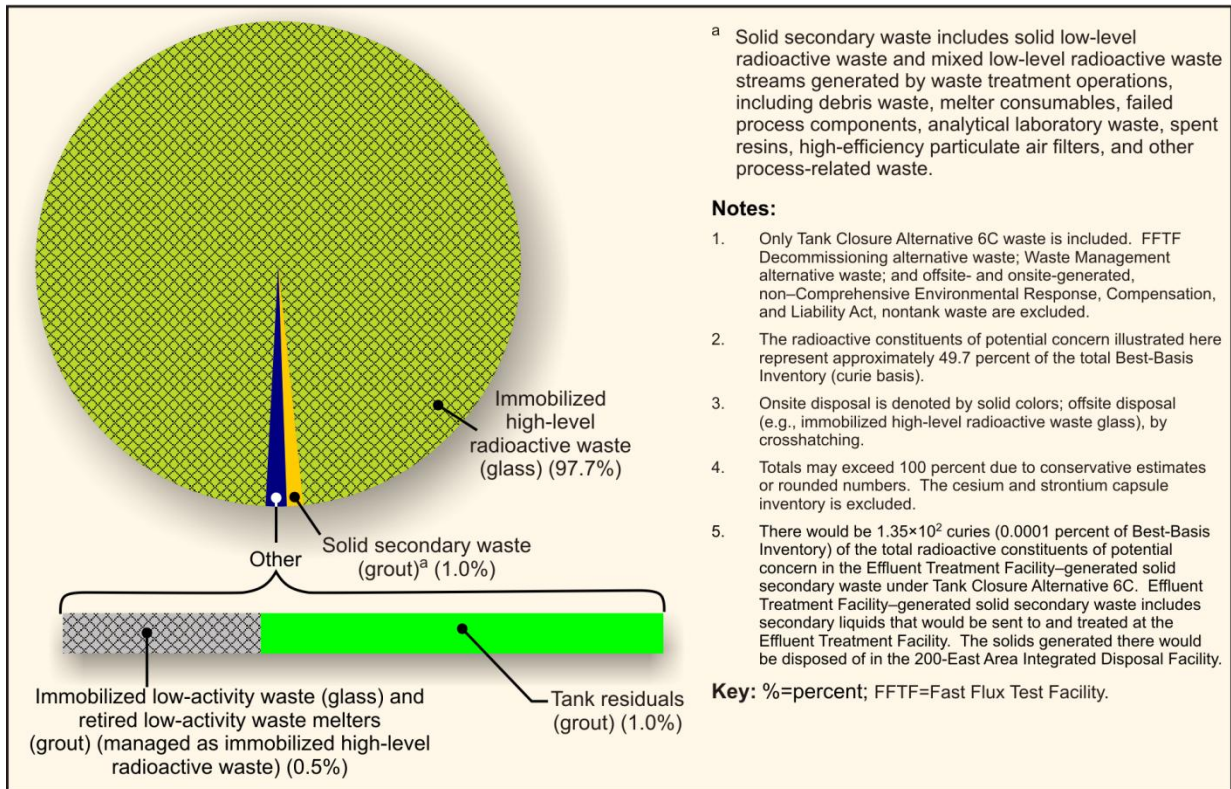


Figure D-63. Tank Closure Alternative 6C Distribution of Total Radioactive Constituents of Potential Concern

## D.2 FFTF DECOMMISSIONING ALTERNATIVES

### D.2.1 Radionuclide and Chemical Inventories

This section summarizes the radionuclide and chemical inventories that were analyzed for each of the three FFTF Decommissioning alternatives. Appendix E, Section E.2.3, provides a summary description of the FFTF Decommissioning alternatives analyzed in this *TC & WM EIS* and is partially reproduced in this section for the reader's convenience. The primary documentation prepared in support of the inventories presented in this section is the *FFTF Radioactive and Hazardous Materials Inventory* (CEES 2006).

The following data supported the development of the radionuclide and chemical inventories for each FFTF Decommissioning alternative.

#### D.2.1.1 Assumptions

Materials that were assumed to be removed during Fast Flux Test Facility (FFTF) deactivation activities include the following:

- Special nuclear materials (associated with flux monitors)
- Nuclear fuel (SNF, nonirradiated fuel)
- Ethylene glycol (approximately 355,830 liters [94,000 gallons])
- Cooling tower chemicals
- Transformer oils containing PCBs (approximately 32,180 liters [8,500 gallons])
- Freon<sup>TM1</sup> R-12 and R-22 (approximately 13,150 kilograms [29,000 pounds])
- Sulfuric acid (approximately 5,700 liters [1,500 gallons])
- Depleted ion exchange resins (approximately 8.5 cubic meters [300 cubic feet])
- Fuel oil (approximately 374,750 liters [99,000 gallons])
- Mobiltherm<sup>TM2</sup> oil (approximately 7,570 liters [2,000 gallons])
- Chemical inventories identified in Attachment 2 of the *Technical Information Document for the Fast Flux Test Facility Closure Project Environmental Impact Statement* (Fluor Hanford 2005a)
- Materials containing asbestos (approximately 76.5 cubic meters [100 cubic yards]) (DOE 1995, 2006a)

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<sup>1</sup> Freon is a registered trademark of E.I. du Pont de Nemours and Company, Wilmington, Delaware.

<sup>2</sup> Mobiltherm is a registered trademark of Socony Mobil Oil Company, Inc., New York, New York.

**D.2.1.2 Fast Flux Test Facility Inventory**

The FFTF radionuclide and hazardous materials inventory is summarized in this section according to the following categories:

1. Sodium inventory
  - a. Volumetric inventory of sodium
  - b. Radionuclide inventory of sodium
2. Radionuclide inventory from activation
  - a. Activated reactor vessel and hardware
  - b. Activated concrete bioshield
3. Radionuclide inventory from contamination
4. Nonradioactive hazardous materials inventory

**D.2.1.3 Fast Flux Test Facility Bulk Sodium Inventory**

The FFTF sodium inventory includes (1) the sodium from the FFTF primary and secondary cooling systems, (2) the sodium in the Fuel Storage Facility and Interim Decay Storage Vessel, and (3) the sodium-potassium alloy from the secondary auxiliary cooling systems (for the primary cold trap filter and Fuel Storage Facility pool) and pressure transducers. The other Hanford bulk sodium, from the Hallam Reactor and Sodium Reactor Experiment, is described in Appendix E, Section E.2.4.1.

The total FFTF sodium inventory has been reported as approximately 984,200 liters (260,000 gallons). The volumes associated with this reported volume for the different FFTF systems are shown in Table D-73. Current estimates have reduced this sodium volume to 958,000 liters (253,000 gallons). As of June 2007, approximately 916,000 liters (242,000 gallons) of radioactively contaminated bulk sodium have been drained from the FFTF reactor vessel, three primary and three secondary heat transport system loops, the Fuel Storage Facility, and the Interim Decay Storage Vessel and associated auxiliary systems; this bulk sodium was transferred to the Sodium Storage Facility. Also, the sodium-potassium alloy, contained in pressure transducers, was removed from FFTF. Additional drainage activities that are planned would result in an estimated volume of approximately 15,140 liters (4,000 gallons) of sodium residuals in the drained systems. Appendix E, Section E.2.4.1, of this EIS contains additional descriptions of the FFTF sodium inventory (Chapin 2007).

**Table D-73. Fast Flux Test Facility Systems Bulk Sodium Volumes**

Fast Flux Test Facility System	Volume (liters)
Primary cooling system	530,000
Secondary cooling system	249,800
Fuel Storage Facility	117,300
Interim Decay Storage Vessel	87,100
<b>Total<sup>a</sup></b>	<b>984,200</b>

<sup>a</sup> The total excludes a nominal 2,271 liters of sodium-potassium alloy that was removed from the Fast Flux Test Facility.

**Note:** To convert liters to gallons, multiply by 0.26417.

**Source:** CEES 2006.

The constituent concentrations and quantities remaining in the FFTF primary and secondary sodium loops are provided in Table D-74. At a minimum, these constituents would be present in the various components in proportion to their sodium residuals volume. Additional quantities of these constituents may be present in the components based on their function (e.g., cold traps); however, the additional quantities in the specific components are unknown.

**Table D-74. Fast Flux Test Facility Sodium Contaminant Constituents**

Constituent	Sodium Analysis			Constituent Inventory		
	Primary Sodium	Secondary Sodium	Units	Quantity in Sodium Residuals (15,140 liters [4,000 gallons])	Quantity in Total FFTF Sodium (984,200 liters [260,000 gallons])	Units
Silver	< 0.02	0.01	ppm by wt	$2.86 \times 10^{-4}$	$1.86 \times 10^{-2}$	kg
Aluminum	0.2	0.5	ppm by wt	$7.17 \times 10^{-3}$	$4.67 \times 10^{-1}$	kg
Boron	< 0.04	0.3	ppm by wt	$4.30 \times 10^{-3}$	$2.79 \times 10^{-1}$	kg
Barium	< 0.02	0.03	ppm by wt	$4.30 \times 10^{-4}$	$2.79 \times 10^{-2}$	kg
Bismuth	< 0.2	0.03	ppm by wt	$2.86 \times 10^{-3}$	$1.86 \times 10^{-1}$	kg
Calcium	0.3	0.5	ppm by wt	$7.17 \times 10^{-3}$	$4.67 \times 10^{-1}$	kg
Cadmium	< 0.01	< 0.1	ppm by wt	$1.43 \times 10^{-3}$	$9.30 \times 10^{-2}$	kg
Chlorine	–	0.5	ppm by wt	$7.17 \times 10^{-3}$	$4.67 \times 10^{-1}$	kg
Cobalt	< 0.02	0.4	ppm by wt	$5.72 \times 10^{-3}$	$3.72 \times 10^{-1}$	kg
Chromium	0.4	0.4	ppm by wt	$5.72 \times 10^{-3}$	$3.72 \times 10^{-1}$	kg
Cesium-137	$< 1 \times 10^{-10}$	–	curies per gram	$1.43 \times 10^{-3}$	$9.30 \times 10^{-2}$	curies
Copper	0.03	0.15	ppm by wt	$2.15 \times 10^{-3}$	$1.40 \times 10^{-1}$	kg
Iron	2.9	39	ppm by wt	$5.58 \times 10^{-1}$	$3.63 \times 10^1$	kg
Hydrogen-3 (tritium)	$1.6 \times 10^{-7}$	–	curies per gram	2.29	$1.49 \times 10^2$	curies
Potassium	2,312	287	ppm by wt	$3.31 \times 10^1$	$2.15 \times 10^3$	kg
Lithium	0.1	< 0.01	ppm by wt	$1.43 \times 10^{-3}$	$9.30 \times 10^{-2}$	kg
Magnesium	0.7	0.05	ppm by wt	$1.00 \times 10^{-2}$	$6.53 \times 10^{-1}$	kg
Manganese	0.4	0.24	ppm by wt	$5.72 \times 10^{-3}$	$3.72 \times 10^{-1}$	kg
Molybdenum	< 0.04	1	ppm by wt	$1.43 \times 10^{-2}$	$9.30 \times 10^{-1}$	kg
Sodium-22	$5.2 \times 10^{-7}$	–	curies per gram	7.44	$4.84 \times 10^2$	curies
Nickel	0.25	22	ppm by wt	$3.15 \times 10^{-1}$	$2.05 \times 10^1$	kg
Lead	0.06	0.3	ppm by wt	$4.30 \times 10^{-3}$	$2.79 \times 10^{-1}$	kg
Silicon	0.1	2	ppm by wt	$2.86 \times 10^{-2}$	1.86	kg
Tin	5	0.02	ppm by wt	$7.17 \times 10^{-2}$	4.67	kg
Strontium	< 0.01	–	ppm by wt	$1.43 \times 10^{-4}$	$9.30 \times 10^{-3}$	kg
Titanium	0.04	–	ppm by wt	$5.72 \times 10^{-4}$	$3.72 \times 10^{-2}$	kg
Total alpha	$1.2 \times 10^{-12}$	–	curies per gram	$1.72 \times 10^{-5}$	$1.12 \times 10^{-3}$	curies
Uranium	< 2	0.001	ppm by wt	$2.86 \times 10^{-2}$	1.86	kg
Vanadium	< 0.02	–	ppm by wt	$2.86 \times 10^{-4}$	$1.86 \times 10^{-2}$	kg

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** FFTF=Fast Flux Test Facility; kg=kilograms; ppm=part(s) per million; wt=weight.

**Source:** CEES 2006.

Gamma energy analyses of the wastewater from cleaning sodium residuals from fuel and fuel-handling components indicated that there are five primary radionuclides present; that the beta-to-alpha ratio is greater than 700; and that cesium-137 and cobalt-60 account for greater than 70 percent of the radionuclides in the waste stream (Fluor Hanford 2005a). The five primary radionuclides in the wash wastewater and their volume percentages are as follows:

- Cesium-137: 94 to 97 percent
- Cesium-134: 2 to 3 percent
- Sodium-22: less than 1 percent
- Cobalt-60: less than 1 percent
- Manganese-54: less than 1 percent

#### **D.2.1.4 Radionuclide Inventory from Activation**

The radionuclide inventory from activation of the reactor vessel and in-vessel components and the concrete bioshield immediately surrounding the reactor vessel is provided in the following sections.

The reactor vessel and in-vessel components have a total of 900,000 curies of activation products, as shown in Table D-75. Table D-76 summarizes the data in Table D-75 and additionally reports the inventory for the Interim Examination and Maintenance (IEM) Cell items and nonfueled hardware that have become activated.

**Table D-75. Activated Reactor Vessel and In-Vessel Component Inventory,  
Decayed to September 2003 (curies)**

	<b>C-14</b>	<b>Co-60</b>	<b>Mo-93</b>	<b>Nb-94</b>	<b>Ni-59</b>	<b>Ni-63</b>	<b>Tc-99</b>	<b>Total</b>
Inner radial shield	3.09×10 <sup>1</sup>	1.48×10 <sup>5</sup>	0	7.91	1.17×10 <sup>2</sup>	1.13×10 <sup>4</sup>	0	1.59×10 <sup>5</sup>
Outer radial shield	1.31×10 <sup>1</sup>	6.49×10 <sup>4</sup>	0	3.78	5.45×10 <sup>1</sup>	5.26×10 <sup>3</sup>	0	7.02×10 <sup>4</sup>
Radial shield support	8.02×10 <sup>-2</sup>	3.84×10 <sup>2</sup>	0	1.74×10 <sup>-2</sup>	3.18×10 <sup>-1</sup>	3.12×10 <sup>1</sup>	0	4.16×10 <sup>2</sup>
Core basket	2.00×10 <sup>-1</sup>	8.49×10 <sup>2</sup>	0	3.20×10 <sup>-2</sup>	7.51×10 <sup>-1</sup>	7.48×10 <sup>1</sup>	0	9.25×10 <sup>2</sup>
Grid plate	4.67×10 <sup>-1</sup>	2.44×10 <sup>3</sup>	0	1.13×10 <sup>-1</sup>	1.86	1.83×10 <sup>2</sup>	0	2.63×10 <sup>3</sup>
Core support structure	8.48×10 <sup>-2</sup>	1.99×10 <sup>2</sup>	0	4.32×10 <sup>-3</sup>	3.00×10 <sup>-1</sup>	2.99×10 <sup>1</sup>	0	2.29×10 <sup>2</sup>
Reactor vessel	6.84×10 <sup>-2</sup>	1.66×10 <sup>2</sup>	0	3.91×10 <sup>-3</sup>	2.42×10 <sup>-1</sup>	2.41×10 <sup>1</sup>	0	1.90×10 <sup>2</sup>
Thermal liner	5.28×10 <sup>-2</sup>	1.28×10 <sup>2</sup>	0	2.94×10 <sup>-3</sup>	1.87×10 <sup>-1</sup>	1.85×10 <sup>1</sup>	0	1.47×10 <sup>2</sup>
Guard vessel	1.63×10 <sup>-2</sup>	2.88×10 <sup>1</sup>	0	8.01×10 <sup>-4</sup>	5.76×10 <sup>-2</sup>	5.71	0	3.46×10 <sup>1</sup>
Core barrel	4.31×10 <sup>-1</sup>	1.95×10 <sup>3</sup>	0	7.68×10 <sup>-2</sup>	1.63	1.64×10 <sup>2</sup>	0	2.12×10 <sup>3</sup>
In-vessel storage modules	9.95×10 <sup>-2</sup>	3.37×10 <sup>2</sup>	0	1.34×10 <sup>-2</sup>	3.45×10 <sup>-1</sup>	3.31×10 <sup>1</sup>	0	3.71×10 <sup>2</sup>
Baffle plate	1.41×10 <sup>-2</sup>	3.03×10 <sup>1</sup>	0	7.16×10 <sup>-4</sup>	5.01×10 <sup>-2</sup>	4.97	0	3.53×10 <sup>1</sup>
Instrument trees	7.28×10 <sup>-2</sup>	3.07×10 <sup>2</sup>	3.86×10 <sup>-9</sup>	1.11×10 <sup>-2</sup>	2.73×10 <sup>-1</sup>	2.69×10 <sup>1</sup>	2.02×10 <sup>-9</sup>	3.34×10 <sup>2</sup>
In-vessel handling machines	4.62×10 <sup>-3</sup>	7.75	6.13×10 <sup>-9</sup>	1.60×10 <sup>-4</sup>	1.63×10 <sup>-2</sup>	1.61	3.21×10 <sup>-9</sup>	9.38
Closure head assembly	1.60×10 <sup>-5</sup>	1.32×10 <sup>-2</sup>	9.58×10 <sup>-8</sup>	4.57×10 <sup>-7</sup>	1.27×10 <sup>-4</sup>	1.21×10 <sup>-2</sup>	5.66×10 <sup>-8</sup>	2.54×10 <sup>-2</sup>
Z ring	6.79×10 <sup>-6</sup>	1.57×10 <sup>-2</sup>	0	5.37×10 <sup>-7</sup>	2.44×10 <sup>-5</sup>	2.45×10 <sup>-3</sup>	0	1.82×10 <sup>-2</sup>
Boron carbide shield	3.26×10 <sup>-8</sup>	0	0	0	0	0	0	3.26×10 <sup>-8</sup>
Steel roof	2.03×10 <sup>-3</sup>	4.19×10 <sup>-7</sup>	0	4.50×10 <sup>-9</sup>	0	0	3.41×10 <sup>-9</sup>	4.27×10 <sup>-7</sup>



**Table D-75. Activated Reactor Vessel and In-Vessel Component Inventory, Decayed to September 2003 (curies) (continued)**

	C-14	Co-60	Mo-93	Nb-94	Ni-59	Ni-63	Tc-99	Total
Row 7 radial reflectors	2.46	1.96×10 <sup>5</sup>	1.06×10 <sup>2</sup>	1.20×10 <sup>1</sup>	5.77×10 <sup>2</sup>	7.86×10 <sup>4</sup>	9.63	2.75×10 <sup>5</sup>
Row 8 and 9 radial reflectors	2.88	2.70×10 <sup>5</sup>	1.17×10 <sup>2</sup>	1.62×10 <sup>1</sup>	7.96×10 <sup>2</sup>	1.02×10 <sup>5</sup>	1.31×10 <sup>1</sup>	3.73×10 <sup>5</sup>
Control and safety rods	9.40×10 <sup>-1</sup>	8.02×10 <sup>3</sup>	3.32×10 <sup>1</sup>	3.53	2.32×10 <sup>1</sup>	1.64×10 <sup>3</sup>	2.70	9.72×10 <sup>3</sup>
In-core shim assemblies	4.40×10 <sup>-1</sup>	3.01×10 <sup>3</sup>	1.83×10 <sup>1</sup>	1.93	1.13×10 <sup>1</sup>	8.10×10 <sup>2</sup>	1.49	3.85×10 <sup>3</sup>
Peripheral shim rod assemblies	8.66×10 <sup>-3</sup>	2.12×10 <sup>1</sup>	2.45×10 <sup>-1</sup>	2.56×10 <sup>-2</sup>	1.60×10 <sup>-1</sup>	1.10×10 <sup>1</sup>	1.90×10 <sup>-2</sup>	3.27×10 <sup>1</sup>
<b>Total</b>	<b>5.23×10<sup>1</sup></b>	<b>6.97×10<sup>5</sup></b>	<b>2.75×10<sup>2</sup></b>	<b>4.56×10<sup>1</sup></b>	<b>1.59×10<sup>3</sup></b>	<b>2.00×10<sup>5</sup></b>	<b>2.69×10<sup>1</sup></b>	<b>8.99×10<sup>5</sup></b>

Key: C=carbon; Co=cobalt; Mo=molybdenum; Nb=niobium; Ni=nickel; Tc=technetium.

Source: CEES 2006.

**Table D-76. Activated Reactor Hardware, Core Components, Nonfueled Hardware, and Interim Examination and Maintenance Cell Items Inventory, Decayed to September 2003 (curies)**

	C-14	Co-60	Mo-93	Nb-94	Ni-59	Ni-63	Tc-99	Total
Reactor hardware	4.56×10 <sup>1</sup>	2.19×10 <sup>5</sup>	1.06×10 <sup>-7</sup>	1.20×10 <sup>1</sup>	1.77×10 <sup>2</sup>	1.72×10 <sup>4</sup>	6.52×10 <sup>-8</sup>	2.37×10 <sup>5</sup>
Core components	6.73	4.77×10 <sup>5</sup>	2.75×10 <sup>2</sup>	3.37×10 <sup>1</sup>	1.41×10 <sup>3</sup>	1.83×10 <sup>5</sup>	2.69×10 <sup>1</sup>	6.62×10 <sup>5</sup>
Nonfueled hardware	4.09×10 <sup>-2</sup>	2.90×10 <sup>3</sup>	1.67	2.05×10 <sup>-1</sup>	6.93	1.11×10 <sup>3</sup>	1.63×10 <sup>-1</sup>	4.02×10 <sup>3</sup>
IEM Cell items	4.01×10 <sup>-2</sup>	2.84×10 <sup>3</sup>	1.63	2.10×10 <sup>-1</sup>	6.79	1.08×10 <sup>3</sup>	1.60×10 <sup>-1</sup>	3.93×10 <sup>3</sup>
<b>Total</b>	<b>5.24×10<sup>1</sup></b>	<b>7.02×10<sup>5</sup></b>	<b>2.78×10<sup>2</sup></b>	<b>4.61×10<sup>1</sup></b>	<b>1.60×10<sup>3</sup></b>	<b>2.02×10<sup>5</sup></b>	<b>2.72×10<sup>1</sup></b>	<b>9.07×10<sup>5</sup></b>

Key: C=carbon; Co=cobalt; IEM=Interim Examination and Maintenance; Mo=molybdenum; Nb=niobium; Ni=nickel; Tc=technetium.

Source: CEES 2006.

The FFTF reactor operated from April 1982 to March 1992 at a time-averaged power level of 206 megawatts. The bioshield surrounding the FFTF reactor vessel is constructed of magnetite concrete with carbon steel rebar and liner. The calculated radionuclide activation products in the bioshield are presented in Table 4.16 of *Activation of the FFTF Biological Shield Wall* (Kidd 2005), which shows them decayed for 13.5 years (September 2006 values) (CEES 2006). These data are reproduced in Table D-77.

**Table D-77. Activation Inventory of Fast Flux Test Facility Bioshield, Decayed to September 2006 (curies)**

	Liner	Rebar	Concrete	Total
Hydrogen-3 (tritium)	2.28×10 <sup>-8</sup>	2.34×10 <sup>-8</sup>	1.73×10 <sup>-5</sup>	1.73×10 <sup>-5</sup>
Argon-39	–	–	1.58×10 <sup>-5</sup>	1.58×10 <sup>-5</sup>
Argon-42	–	–	1.67×10 <sup>-9</sup>	1.67×10 <sup>-9</sup>
Beryllium-10	9.24×10 <sup>-12</sup>	2.81×10 <sup>-11</sup>	–	3.73×10 <sup>-11</sup>
Carbon-14	4.36×10 <sup>-8</sup>	1.81×10 <sup>-7</sup>	7.65×10 <sup>-4</sup>	7.65×10 <sup>-4</sup>
Calcium-41	–	–	1.54×10 <sup>-2</sup>	1.54×10 <sup>-2</sup>
Calcium-48	–	–	4.32×10 <sup>-27</sup>	4.32×10 <sup>-27</sup>
Cobalt-60	2.15×10 <sup>-1</sup>	3.07×10 <sup>-1</sup>	4.66×10 <sup>-4</sup>	5.22×10 <sup>-1</sup>
Cobalt-60m	–	3.50×10 <sup>-16</sup>	1.89×10 <sup>-13</sup>	1.90×10 <sup>-13</sup>
Chromium-50	1.81×10 <sup>-25</sup>	1.68×10 <sup>-25</sup>	5.06×10 <sup>-25</sup>	8.55×10 <sup>-25</sup>
Iron-55	1.99	3.57	1.19×10 <sup>1</sup>	1.74×10 <sup>1</sup>
Iron-60	–	3.50×10 <sup>-16</sup>	1.89×10 <sup>-13</sup>	1.90×10 <sup>-13</sup>

**Table D-77. Activation Inventory of Fast Flux Test Facility Bioshield,  
Decayed to September 2006 (curies) (continued)**

	<b>Liner</b>	<b>Rebar</b>	<b>Concrete</b>	<b>Total</b>
Potassium-40	–	–	$1.24 \times 10^{-8}$	$1.24 \times 10^{-8}$
Potassium-42	–	–	$1.67 \times 10^{-9}$	$1.67 \times 10^{-9}$
Manganese-53	$2.24 \times 10^{-10}$	$1.99 \times 10^{-10}$	$6.13 \times 10^{-10}$	$1.04 \times 10^{-9}$
Molybdenum-100	–	$8.16 \times 10^{-29}$	–	$8.16 \times 10^{-29}$
Molybdenum-93	–	$1.85 \times 10^{-5}$	–	$1.85 \times 10^{-5}$
Niobium-91	–	$9.92 \times 10^{-10}$	–	$9.92 \times 10^{-10}$
Niobium-92	–	$3.92 \times 10^{-13}$	–	$3.92 \times 10^{-13}$
Niobium-93m	–	$1.01 \times 10^{-5}$	–	$1.01 \times 10^{-5}$
Niobium-94	–	$1.34 \times 10^{-11}$	–	$1.34 \times 10^{-11}$
Nickel-59	–	$3.94 \times 10^{-4}$	$2.47 \times 10^{-1}$	$2.47 \times 10^{-1}$
Nickel-63	–	$4.25 \times 10^{-2}$	$2.67 \times 10^1$	$2.68 \times 10^1$
Phosphorus-32	$2.36 \times 10^{-11}$	$5.71 \times 10^{-11}$	–	$8.07 \times 10^{-11}$
Scandium-48	–	–	$6.91 \times 10^{-28}$	$6.91 \times 10^{-28}$
Silicon-32	$2.36 \times 10^{-11}$	$5.71 \times 10^{-11}$	–	$8.07 \times 10^{-11}$
Technetium-99	–	$4.52 \times 10^{-6}$	–	$4.52 \times 10^{-6}$
Vanadium-50	–	$3.00 \times 10^{-21}$	$1.76 \times 10^{-18}$	$1.76 \times 10^{-18}$
Zinc-70	–	–	$2.49 \times 10^{-22}$	$2.49 \times 10^{-22}$
Zirconium-93	–	$2.33 \times 10^{-12}$	–	$2.33 \times 10^{-12}$
Zirconium-96	–	$5.02 \times 10^{-31}$	–	$5.02 \times 10^{-31}$
<b>Total</b>	<b>2.21</b>	<b>3.92</b>	<b><math>3.89 \times 10^1</math></b>	<b><math>4.50 \times 10^1</math></b>

Source: CEES 2006.

#### **D.2.1.5 Radionuclide Inventory from Contamination**

Contamination within FFTF is primarily confined to the reactor containment vessel, internal surfaces of system components that handled primary sodium and radioactive argon, cells within the Reactor Containment Building (RCB), decontamination areas, liquid radioactive waste holding and exporting systems, sodium removal and sampling systems, fuel handling systems, IEM Cell, and Contaminated Equipment Repair Shop. The contaminated areas within the FFTF facilities are listed in Table D-78.

A hot spot of 150 roentgens per hour on contact was identified in the piping downstream of the 5-standard-cubic-foot-per-minute vapor trap (York 2005). This radiation level correlates with an estimated source of 3.5 curies of cesium-137.

The *Technical Information Document for the Fast Flux Test Facility Closure Project Environmental Impact Statement* (Fluor Hanford 2005a) identifies the IEM Cell as the cell with the greatest amount of contamination. The FFTF Decommissioning alternatives scaled data sets (SAIC 2010b) estimate that contamination within the IEM Cell equates to  $9.95 \times 10^{-4}$  curies of cesium-137, decayed to 2005. This indicates that the inventory due to contamination makes up a very small fraction of the inventory associated with activated structures and components.

**Table D-78. Contaminated Areas Within the Fast Flux Test Facility**

Building	Cell Number	Description	Average Contamination Level (dpm/100 cm <sup>2</sup> )
Reactor Containment Building	524	Heat compartment	No data <sup>a</sup>
	528	Fuel transfer port adapter storage	0
	544	CLEM grapple change box pit	No data <sup>a</sup>
	548	IEM Cell	5.16×10 <sup>4</sup> (1998)
	549	Radioactive argon gas pipeline	0
	567	Electromagnetic pump cell	No data <sup>a</sup>
	FTP 1	–	8.72×10 <sup>4</sup> (2002)
	FTP 2	–	3.75×10 <sup>4</sup> (1997)
	FTP 3	–	7.00×10 <sup>3</sup> (1997)
Heat Transport System Service Building South	490	Sodium sampling cell	No data <sup>a</sup>
Reactor Service Building	201	Sodium Removal System	No data <sup>a</sup>
	205	Sodium Removal System	No data <sup>a</sup>
Maintenance and Storage Facility	–	Decontamination I	No data <sup>a</sup>
	–	Decontamination II	No data <sup>a</sup>
	–	Contaminated Equipment Repair Shop	No data <sup>a</sup>
	17	Radioactive Liquid Waste Tank Area	No data <sup>a</sup>

<sup>a</sup> These areas are identified as contaminated or potentially contaminated, but no survey data or contamination estimates are available.

**Key:** CLEM=Closed-Loop Ex-Vessel Machine; dpm/100 cm<sup>2</sup>=disintegrations per minute per 100 square centimeters; FTP=fuel transfer port; IEM=Interim Examination and Maintenance.

**Source:** CEES 2006.

### D.2.1.6 Hazardous Materials Inventory

The following materials are either planned for removal or have been removed from FFTF during the deactivation activities: ethylene glycol, Mobiltherm™ oil, transformer oils containing PCBs, cooling tower chemicals, sulfuric acid, Freon™ R-12 and R-22, depleted ion exchange resins, fuel oil, and asbestos-containing materials. The balance of the chemical inventory is identified in Attachment 2 of the *Technical Information Document for the Fast Flux Test Facility Closure Project Environmental Impact Statement* (Fluor Hanford 2005a).

The remaining hazardous materials include approximately 47,900 kilograms (105,600 pounds) of lead and 37,694 kilograms (83,100 pounds) of depleted uranium. The lead would be removed to the extent practicable during FFTF deactivation activities (DOE 2006a). The depleted uranium would remain in the facility under FFTF Decommissioning Alternatives 1 and 2 and would be removed under FFTF Decommissioning Alternative 3.

## **D.2.2 FFTF Decommissioning Alternative 1: No Action**

Council on Environmental Quality regulations require that National Environmental Policy Act analyses include a No Action Alternative. Under this alternative, deactivation of the FFTF complex and support buildings would be completed, as specified by previous FFTF National Environmental Policy Act decisions (*Environmental Assessment, Sodium Residuals Reaction/Removal and Other Deactivation Work Activities, Fast Flux Test Facility (FFTF) Project, Hanford Site, Richland, Washington* [DOE 2006a]), and maintained in a long-term surveillance and maintenance (S&M) condition for the foreseeable future. The facility would be monitored and periodic S&M would be performed to ensure that the environmental and safety issues are minimized and addressed.

### **D.2.2.1 Facility Disposition**

The FFTF RCB, along with the rest of the buildings within the 400 Area Property Protected Area (PPA), would be maintained in a long-term S&M condition after completion of all deactivation activities. The buildings would be left standing with a maintained exterior that would be capable of protecting them from the elements. They would be unoccupied, with essential safety-related systems left operational. Such systems could include, but would not be limited to, fire protection, emergency lighting, ventilation, air monitoring, and inert gas systems used to isolate piping and equipment containing sodium residuals. Other radioactive or chemical waste and materials would be removed during deactivation.

### **D.2.2.2 Process Components**

The reactor vessel, piping systems, and tanks (contained above and below grade within the RCB and immediately adjacent buildings) would be left in place under an inert gas blanket. Deactivation activities would be complete, including draining of the bulk sodium and removal of SNF, lead shielding, remote-handled special components, small-bore piping, valves, and other components. Some systems would be deactivated and de-energized and isolated (e.g., those not associated with maintaining safety-related functions) per the deactivation plans.

### **D.2.2.3 Sodium Residuals**

Sodium residuals in the RCB vessels and cooling systems' piping would be left in place untreated, but under an inert gas blanket. During deactivation activities, the FFTF bulk sodium would be drained from the reactor systems and stored as a solid in tanks in the Sodium Storage Facility within the 400 Area. The small amount of sodium-potassium alloy would be blended with the content of the bulk sodium storage containers. The Hallam Reactor and Sodium Reactor Experiment sodium would remain in its current storage location (Hanford 200-West Area).

### **D.2.2.4 Demolition and Other Waste**

There would be no demolition under the No Action Alternative; therefore, no demolition waste would be generated. Solid and liquid radioactive and/or hazardous waste generated during deactivation would be managed and disposed of on site. Activities associated with the No Action Alternative would not generate substantial additional quantities of solid waste for disposal. The small amounts of radioactive solid waste generated during S&M activities would be disposed of on site in disposal facilities approved for Hanford's operational waste at Low-Level Radioactive Waste Burial Ground (LLBG) 218-W-5, trenches 31 and 34. Other regulated waste, such as PCBs, asbestos, and hazardous waste, would be handled in a similar manner under all of the alternatives. The volume of this waste is expected to be small, and it would be disposed of in accordance with existing *Hanford Site Solid Waste Acceptance Criteria* (Fluor Hanford 2005b) or offsite treatment contracts.

### D.2.2.5 End State

The facilities and infrastructure within the 400 Area PPA, including the RCB, would be maintained in a 100-year administrative control condition with appropriate monitoring and controls (to ensure that environmental or safety concerns are minimized) (SAIC 2010b).

Matching the list of radionuclides and chemicals identified in the above tables with the COPCs identified in Appendix D, Section D.1.1, resulted in a report of the following radionuclides (in curies): cesium-137, carbon-14, tritium, and technetium-99, as well as the following chemicals (in kilograms): chromium, lead, and uranium. Table D-79 summarizes each of these radioactive and chemical COPCs under FFTF Decommissioning Alternative 1.

**Table D-79. FFTF Decommissioning Alternative 1 Radioactive and Chemical Constituents of Potential Concern Balance**

	Cesium-137	Carbon-14	Hydrogen-3 (Tritium)	Technetium-99	Chromium	Lead	Total Uranium
	Curies				Kilograms		
<b>Inventory Remaining at the FFTF Site</b>							
Sodium residuals <sup>a</sup>	1.43×10 <sup>-3</sup>	0	2.29	0	5.72×10 <sup>-3</sup>	4.30×10 <sup>-3</sup>	2.86×10 <sup>-2</sup>
Hardware <sup>b</sup>	0	5.24×10 <sup>1</sup>	0	2.72×10 <sup>1</sup>	0	0	3.77×10 <sup>4</sup>
Bioshield	0	7.65×10 <sup>-4</sup>	1.73×10 <sup>-5</sup>	4.52×10 <sup>-6</sup>	0	0	0
<b>Total Remaining Inventory</b>	<b>1.43×10<sup>-3</sup></b>	<b>5.24×10<sup>1</sup></b>	<b>2.29</b>	<b>2.72×10<sup>1</sup></b>	<b>5.72×10<sup>-3</sup></b>	<b>4.30×10<sup>-3</sup></b>	<b>3.77×10<sup>4</sup></b>
<b>Inventory Disposed of in LLBG 218-W-5, Mixed Waste Trenches 31 and 34</b>							
Secondary waste <sup>c</sup>	1.43×10 <sup>-3</sup>	5.31×10 <sup>-5</sup>	1.88×10 <sup>-6</sup>	1.19×10 <sup>-3</sup>	1.42×10 <sup>-4</sup>	1.07×10 <sup>-4</sup>	2.14×10 <sup>-2</sup>

<sup>a</sup> The inventory for the approximately 15,142 liters (4,000 gallons) of sodium residuals includes FFTF components, e.g., the reactor and miscellaneous traps.

<sup>b</sup> Hardware includes activated reactor hardware, the depleted uranium shield, core components, nonfueled hardware, and Interim Examination and Maintenance Cell items.

<sup>c</sup> Secondary-waste inventories were estimated from 2006 *Solid Waste Integrated Forecast Technical (SWIFT) Report, FY2006–FY2035* database information (Barcot 2005). For analysis purposes, it was conservatively assumed that 100 percent of the cesium-137 inventory would be captured in the secondary waste.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** FFTF=Fast Flux Test Facility; LLBG=low-level radioactive waste burial ground.

**Source:** SAIC 2011.

### D.2.3 FFTF Decommissioning Alternative 2: Entombment

Under this alternative, the portions of the FFTF RCB (and structures within) that are above grade level (i.e., 168 meters [550 feet] above mean sea level) would be decontaminated as necessary, dismantled, and removed. The RCB structures below grade level, as well as the FFTF reactor vessel and radioactive and contaminated equipment, piping, and other materials and components that have become radioactive or otherwise contaminated, would remain in place. Sodium residuals would be removed from the RCB and treated either in existing 400 Area facilities or in place. In addition, the below-grade RCB structures would be filled with grout or other suitable fill material to immobilize remaining hazardous chemicals and radioactive materials to the maximum extent practicable and to prevent subsidence. The RCB fill material may include other demolition debris containing hazardous or radioactive materials, as allowed by regulations. An engineered, modified Resource Conservation and Recovery Act (RCRA)

Subtitle C barrier that is compliant with regulations would be constructed over the filled area. The entombed area would include the barrier together with the lower RCB and adjacent structures and immobilized internal structures.

The FFTF support buildings would be decontaminated as necessary and demolished. The area previously occupied by the facilities would be backfilled with soil to eliminate void spaces, compacted such that natural settling would not result in depressions (to avoid potential ponding of water), recontoured, and revegetated. An appropriate monitoring program for the PPA would also be established. The following sections provide additional descriptions of the activities that would be conducted under FFTF Decommissioning Alternative 2.

#### **D.2.3.1 Facility Disposition**

Appendix E, Table E-14, summarizes the proposed decommissioning activities for each building under both FFTF Decommissioning Alternative 2: Entombment and FFTF Decommissioning Alternative 3: Removal. Under the Entombment Alternative, all above-grade structures that are part of the main RCB and the two immediately adjacent support facilities (Buildings 491E and 491W) would be dismantled, and the demolition waste would be disposed of in an IDF or consolidated in the below-grade spaces. Below-grade structures would be filled with demolition waste, as practicable, and stabilized with suitable fill material (e.g., grout) to immobilize hazardous chemical and radioactive materials and prevent subsidence in the future.

All other ancillary buildings, including their internal equipment and components, would be demolished, as noted in Appendix E, Table E-14, and the contaminated demolition debris would be disposed of in an IDF or consolidated within available below-grade spaces within the RCB or Buildings 491E and 491W. All radioactive and/or hazardous material would be removed. Wood and large steel components would also be removed. Foundation rubble (e.g., concrete and rebar) could remain. The area previously occupied by these facilities would be backfilled with soil, compacted, contoured, and revegetated. As indicated in Appendix E, Table E-14, some of these buildings would be either completely or partially within the footprint (including the side slope) of the engineered barrier over the RCB.

#### **D.2.3.2 Process Components**

The reactor vessel, piping systems, and tanks located above grade within the RCB and immediately adjacent buildings would be dismantled and placed in below-grade spaces, as practicable, or transported to an IDF for disposal. Deactivation activities would be complete, including draining of the bulk sodium and removal of SNF, lead shielding, remote-handled special components, small-bore piping, valves, and other components. Systems located below grade (including regulated waste) would be grouted in place after treatment of any SNF sodium residuals. The small-diameter (less than 20.3 centimeters [8 inches]) piping would be removed, treated (cleaned of sodium) in the 400 Area, and disposed of on site in an IDF or placed in below-grade spaces within the RCB.

#### **D.2.3.3 Sodium Residuals**

All sodium residuals would be removed from the RCB systems or treated in place. It was assumed that sodium would be drained from plant systems to the extent practicable, followed by moist gas passivation and/or flushing with water to stabilize sodium residuals. Sodium residuals in small-diameter piping would be treated in the 400 Area after removal of the components from the reactor plant.



#### **D.2.3.4 Demolition and Other Waste**

Demolition debris from facility decommissioning (chemically hazardous or radioactive solid waste) would be handled in the same way under both action alternatives, except that the disposition of the volumes of debris would change. Under FFTF Decommissioning Alternative 2, the debris not placed in the RCB or other voids or used as backfill would be transported to an IDF for disposal. Solid waste resulting from any of the processing options (for sodium residual waste, bulk sodium, etc.) would be included with the analyses of those options.

Radioactive liquid waste resulting from treatment of the sodium residuals also would be handled in the same way under both action alternatives. The liquid volume would be reduced at FFTF (through either ion exchange and reuse or evaporation), and the remaining liquids would be transported to the 200 Area ETF for processing and disposal. For the analyses in this *TC & WM EIS*, it was assumed that a 90 percent reduction in volume could be achieved prior to shipment of the liquid to the ETF for processing. Any other sources of radioactive waste (such as decontamination solutions) are expected to result in very small volumes compared with waste produced as a result of treating sodium residuals.

Other regulated waste, such as PCBs, asbestos, and nonradioactive hazardous waste, would be handled in a similar manner under all of the alternatives. The volume of this waste is expected to be small, and it would be disposed of in accordance with existing *Hanford Site Solid Waste Acceptance Criteria* (Fluor Hanford 2005b) or offsite treatment contracts.

#### **D.2.3.5 End State**

For analysis in this *TC & WM EIS*, it was assumed that a modified RCRA Subtitle C barrier would be constructed over the RCB and Buildings 491E and 491W, which contain residual radioactive and/or hazardous waste.

In addition, the barrier would extend over part or all of the immediately adjacent facility footprints. The barrier would be circular with a radius of about 39.2 meters (128.5 feet), not including the side slope used for drainage. The side slope would be about 5.2 meters (17.1 feet) using a 3 horizontal:1 vertical slope. Minimal postclosure care would be required. The remainder of the PPA would be backfilled with soil, compacted, recontoured, and revegetated.

The modified RCRA Subtitle C barrier would be designed to provide containment and hydrologic protection for a performance period of 500 years. This performance period is conservatively based on radionuclide concentration and activity limits for Category 3 low-level radioactive waste (LLW). The modified RCRA Subtitle C barrier would be composed of eight layers of durable material with a combined minimum thickness of about 1.7 meters (5.7 feet), excluding the grading fill layer, which would range from zero at the edge to approximately 0.8 meters (2.6 feet) at the center for a 2 percent drainage slope. The design would also incorporate an asphaltic concrete layer to reduce the likelihood of bio-intrusion or inadvertent human intrusion (SAIC 2010b). Further information on the modified RCRA Subtitle C barrier can be found in Appendix E, Section E.1.2.5.4.1.

Matching the list of radionuclides and chemicals identified in Tables D-74 through D-77 with the COPCs identified in Section D.1.1, resulted in a report of the following radionuclides (in curies): cesium-137, carbon-14, tritium, and technetium-99, as well as the following chemicals (in kilograms): chromium, lead, and uranium. Table D-80 is a summary of each of the radioactive and chemical COPCs under FFTF Decommissioning Alternative 2.

**Table D–80. FFTF Decommissioning Alternative 2 Radioactive and Chemical  
Constituents of Potential Concern Balance**

	Cesium-137	Carbon-14	Hydrogen-3 (Tritium)	Technetium-99	Chromium	Lead	Total Uranium
	Curies				Kilograms		
<b>Inventory Remaining at the FFTF Site</b>							
Hardware <sup>a</sup>	0	5.24×10 <sup>1</sup>	0	2.72×10 <sup>1</sup>	0	0	3.77×10 <sup>4</sup>
Bioshield	0	7.65×10 <sup>-4</sup>	1.73×10 <sup>-5</sup>	4.52×10 <sup>-6</sup>	0	0	0
<b>Total remaining inventory</b>	<b>0</b>	<b>5.24×10<sup>1</sup></b>	<b>1.73×10<sup>-5</sup></b>	<b>2.72×10<sup>1</sup></b>	<b>0</b>	<b>0</b>	<b>3.77×10<sup>4</sup></b>
<b>Inventory Disposed of in an IDF</b>							
Sodium residuals <sup>b</sup>	1.43×10 <sup>-3</sup>	0	2.29	0	5.72×10 <sup>-3</sup>	4.30×10 <sup>-3</sup>	2.86×10 <sup>-2</sup>
Secondary waste <sup>c</sup>	1.43×10 <sup>-3</sup>	6.33×10 <sup>-4</sup>	3.58×10 <sup>-7</sup>	1.48×10 <sup>-2</sup>	1.79×10 <sup>-3</sup>	1.34×10 <sup>-3</sup>	4.09×10 <sup>-3</sup>
<b>Total inventory disposed of in an IDF</b>	<b>2.86×10<sup>-3</sup></b>	<b>6.33×10<sup>-4</sup></b>	<b>2.29</b>	<b>1.48×10<sup>-2</sup></b>	<b>7.50×10<sup>-3</sup></b>	<b>5.64×10<sup>-3</sup></b>	<b>3.27×10<sup>-2</sup></b>

<sup>a</sup> Hardware includes activated reactor hardware, the depleted uranium shield, core components, nonfueled hardware, and Interim Examination and Maintenance Cell items.

<sup>b</sup> The inventory for the approximately 15,142 liters (4,000 gallons) of sodium residuals includes FFTF components, e.g., the reactor and miscellaneous traps.

<sup>c</sup> Secondary-waste inventories were estimated from 2006 *Solid Waste Integrated Forecast Technical (SWIFT) Report, FY2006–FY2035* database information (Barcot 2005). For analysis purposes, it was conservatively assumed that 100 percent of the cesium-137 inventory would be captured in the secondary waste.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** FFTF=Fast Flux Test Facility; IDF=Integrated Disposal Facility.

**Source:** SAIC 2011.

## D.2.4 FFTF Decommissioning Alternative 3: Removal

Under this alternative, the portions of the RCB (and structures within) that are above grade level would be decontaminated as necessary, dismantled, and removed. All sodium residuals would be removed from the RCB or treated in place to neutralize the chemical reactivity of the metallic sodium. Below grade level, the reactor vessel and contaminated reactor vessel internals, other radioactively contaminated equipment, piping, materials, and other components, along with any asbestos, depleted uranium shielding, and lead shielding, would also be removed. Such radioactively contaminated equipment, piping, materials, and components would include the intermediate heat exchangers, primary pumps, primary isolation valves, primary overflow tanks, IEM Cell equipment, 8.5- to 12.2-meter (28- to 40-foot) test assembly hardware, and the Interim Decay Storage Vessel. Additional radioactively contaminated equipment from the RCB and FFTF heat transport system would also be removed. Upon removal, this equipment would be disposed of in an IDF. The below-grade RCB and the FFTF support buildings outside the RCB area would be decontaminated as necessary and demolished. The area previously occupied by the facilities would then be backfilled with soil, compacted, recontoured, and revegetated. An appropriate monitoring program would also be established. The following sections describe the activities to be conducted under the Removal Alternative.

#### **D.2.4.1 Facility Disposition**

All above-grade structures that are part of the RCB and the immediately adjacent support facilities with substructures (basements) would be dismantled, and the contaminated demolition debris would be disposed of in an IDF. The RCB would be demolished down to grade level and the support facilities would be demolished to 0.91 meters (3 feet) below grade. Below-grade radioactively contaminated components and equipment (including the reactor vessel) would be removed. However, the reinforced concrete shell in the RCB would remain and be backfilled with either soil or grout, compacted, recontoured, and revegetated. Small amounts of radioactive activation products in structural concrete and steel would remain. All small-diameter piping would be removed, and sodium residuals would be either treated in place or removed from the RCB for treatment at an onsite facility to neutralize the chemical reactivity of the metallic sodium.

All other ancillary buildings, including their internal equipment and components, would be demolished and removed (down to a depth of 0.91 meters [3 feet] below grade). The contaminated demolition debris would be disposed of in an IDF, and the vacated spaces would be backfilled, compacted, recontoured, and revegetated. All radioactive and/or hazardous material would be removed. Wood and large steel components would also be removed. Foundation rubble, e.g., concrete and rebar, would remain.

#### **D.2.4.2 Process Components**

The above- and below-grade reactor vessel, piping systems, and tanks within the RCB and the immediately adjacent buildings would be dismantled and transported to an IDF for disposal. Deactivation activities would be completed, including draining of the bulk sodium and removal of SNF, lead shielding, remote-handled special components, small-bore piping, valves, and other components. Radioactively contaminated equipment, piping, tanks, hazardous materials (including asbestos and lead shielding), and other components would also be removed for disposal in an IDF. The reactor vessel (along with any attached depleted uranium shielding and/or internal piping and equipment) would be filled with grout, removed, packaged, and transported to an IDF for disposal. Uncontaminated material (i.e., material that is clean of radioactive or hazardous substances) would not be removed and, as previously stated, the reinforced concrete shell would remain. All small-diameter piping would be removed. The small-diameter piping would be treated in the 400 Area to remove sodium residuals and would be disposed of on site in an IDF.

#### **D.2.4.3 Sodium Residuals**

Sodium residuals would be treated the same under both FFTF Decommissioning action alternatives. All sodium residuals would be removed from the RCB systems or treated in place. It was assumed that sodium would be drained from the plant systems to the extent practicable, followed by moist gas passivation and/or flushing with water to stabilize sodium residuals. Sodium residuals in small-diameter piping would be treated in the 400 Area after the piping has been removed from the reactor plant.

#### **D.2.4.4 Demolition and Other Waste**

Demolition debris, radioactive solid waste, radioactive liquid waste, and other regulated hazardous waste would be handled in the same manner under both FFTF Decommissioning action alternatives; only the disposition of the volume of waste would change. The approaches to waste handling also would be the same, and demolition waste would be disposed of in an IDF under both action alternatives.

### D.2.4.5 End State

Below-grade portions of structures would be backfilled with soil, compacted, recontoured, and revegetated. Although there would be no anticipated need for an engineered barrier, it was assumed for analysis purposes that an appropriate postclosure care program would be established (SAIC 2010b).

Matching the list of radionuclides and chemicals identified in the previous tables with the COPCs identified in Section D.1.1 resulted in a report of the following radionuclides (in curies): cesium-137, carbon-14, tritium, and technetium-99, as well as the following chemicals (in kilograms): chromium, lead, and uranium. Table D–81 is a summary of each of the radioactive and chemical COPCs under FFTF Decommissioning Alternative 3.

**Table D–81. FFTF Decommissioning Alternative 3 Radioactive and Chemical Constituents of Potential Concern Balance**

	Cesium-137	Carbon-14	Hydrogen-3 (Tritium)	Technetium-99	Chromium	Lead	Total Uranium
	Curies				Kilograms		
<b>Inventory Remaining at the FFTF Site</b>							
Bioshield	0	$7.65 \times 10^{-4}$	$1.73 \times 10^{-5}$	$4.52 \times 10^{-6}$	0	0	0
<b>Inventory Disposed of in an IDF</b>							
Sodium residuals <sup>a</sup>	$1.43 \times 10^{-3}$	0	2.29	0	$5.72 \times 10^{-3}$	$4.30 \times 10^{-3}$	$2.86 \times 10^{-2}$
Hardware <sup>b</sup>	0	$5.24 \times 10^1$	0	$2.72 \times 10^1$	0	0	$3.77 \times 10^4$
Secondary waste <sup>c</sup>	$1.43 \times 10^{-3}$	$6.41 \times 10^{-4}$	$1.27 \times 10^{-6}$	$1.50 \times 10^{-2}$	$1.81 \times 10^{-3}$	$1.36 \times 10^{-3}$	$1.36 \times 10^{-2}$
<b>Total Inventory Disposed of in an IDF</b>	<b><math>2.86 \times 10^{-3}</math></b>	<b><math>5.24 \times 10^1</math></b>	<b>2.29</b>	<b><math>2.72 \times 10^1</math></b>	<b><math>7.52 \times 10^{-3}</math></b>	<b><math>5.65 \times 10^{-3}</math></b>	<b><math>3.77 \times 10^4</math></b>

<sup>a</sup> The inventory for the approximately 15,142 liters (4,000 gallons) of sodium residuals includes FFTF components, e.g., the reactor and miscellaneous traps.

<sup>b</sup> Hardware includes activated reactor hardware, the depleted uranium shield, core components, nonfueled hardware, and Interim Examination and Maintenance Cell items.

<sup>c</sup> Secondary-waste inventories were estimated from 2006 *Solid Waste Integrated Forecast Technical (SWIFT) Report, FY2006–FY2035* database information (Barcot 2005). For analysis purposes, it was conservatively assumed that 100 percent of the cesium-137 inventory would be captured in the secondary waste.

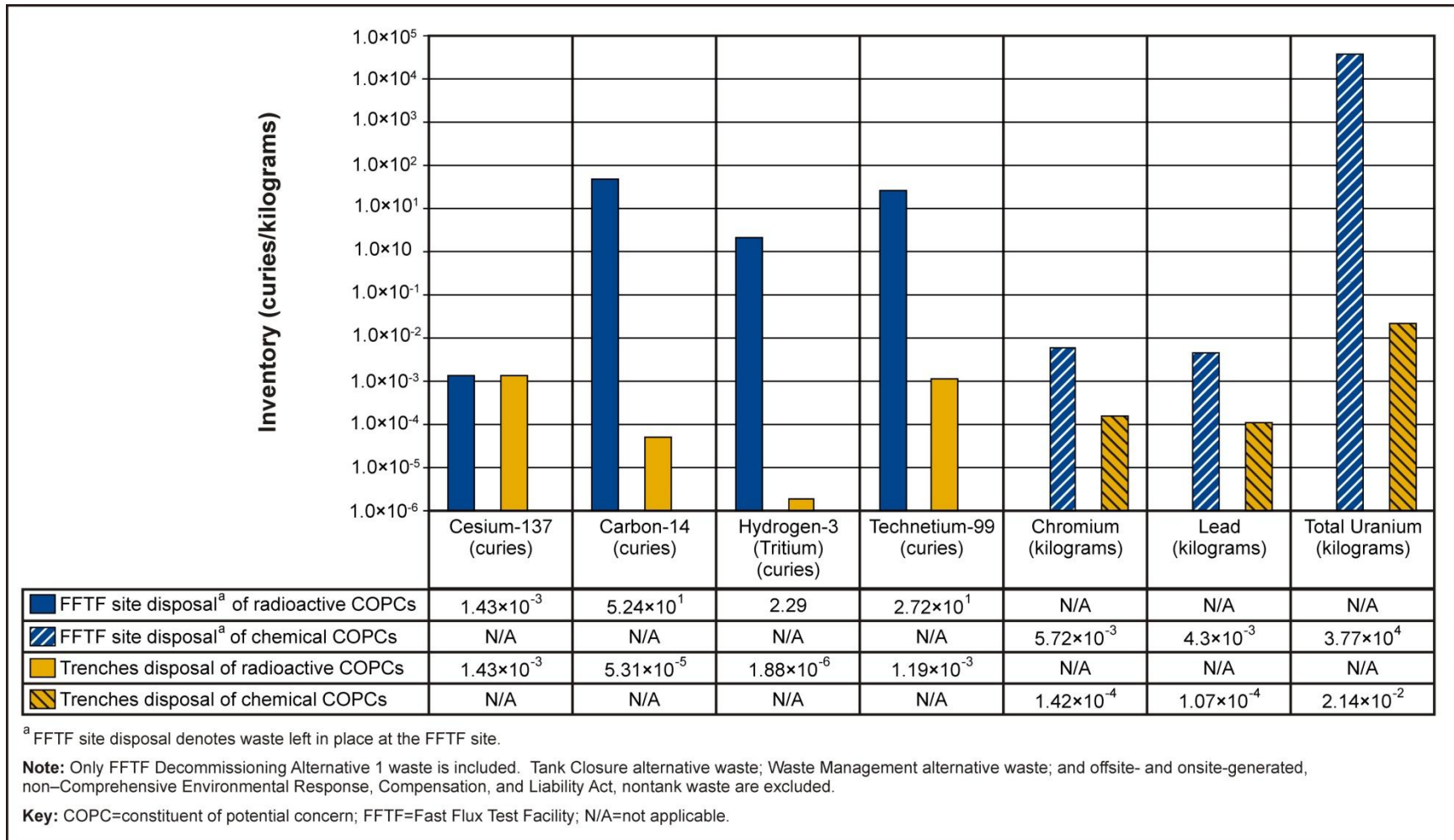
**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** FFTF=Fast Flux Test Facility; IDF=Integrated Disposal Facility.

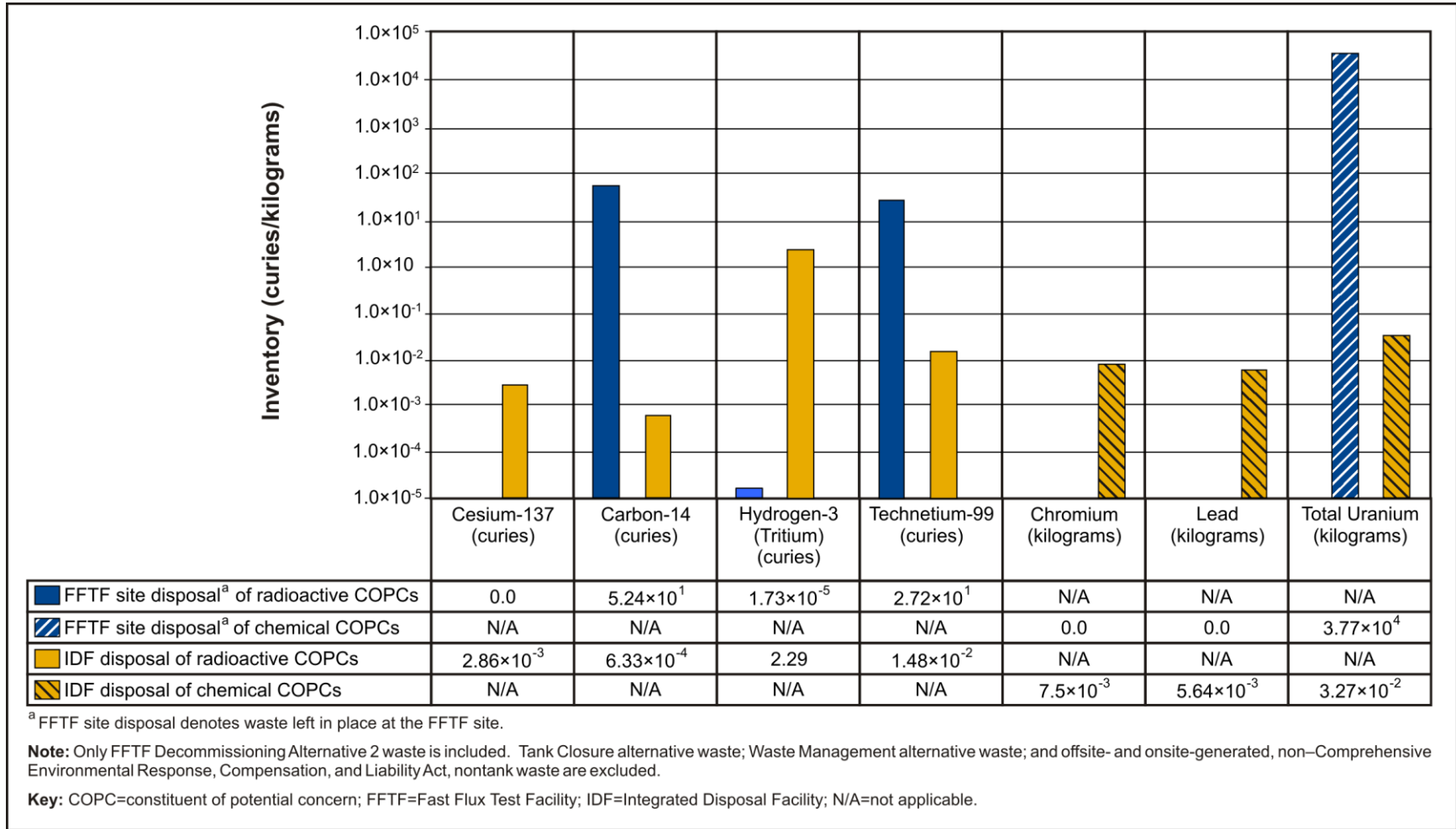
**Source:** SAIC 2011.

### D.2.5 Distribution of Fast Flux Test Facility Waste

As discussed above, under each of the three FFTF Decommissioning alternatives, a number of waste streams would be generated, and this waste would be disposed of differently. This section provides histograms (see Figures D–64, D–65, and D–66) depicting the distribution of the radioactive COPCs between the FFTF site and an IDF under each of the FFTF Decommissioning alternatives (SAIC 2011). The COPCs shown include both radionuclides (cesium-137, carbon-14, tritium, and technetium-99) and chemicals (chromium, lead, and total uranium).



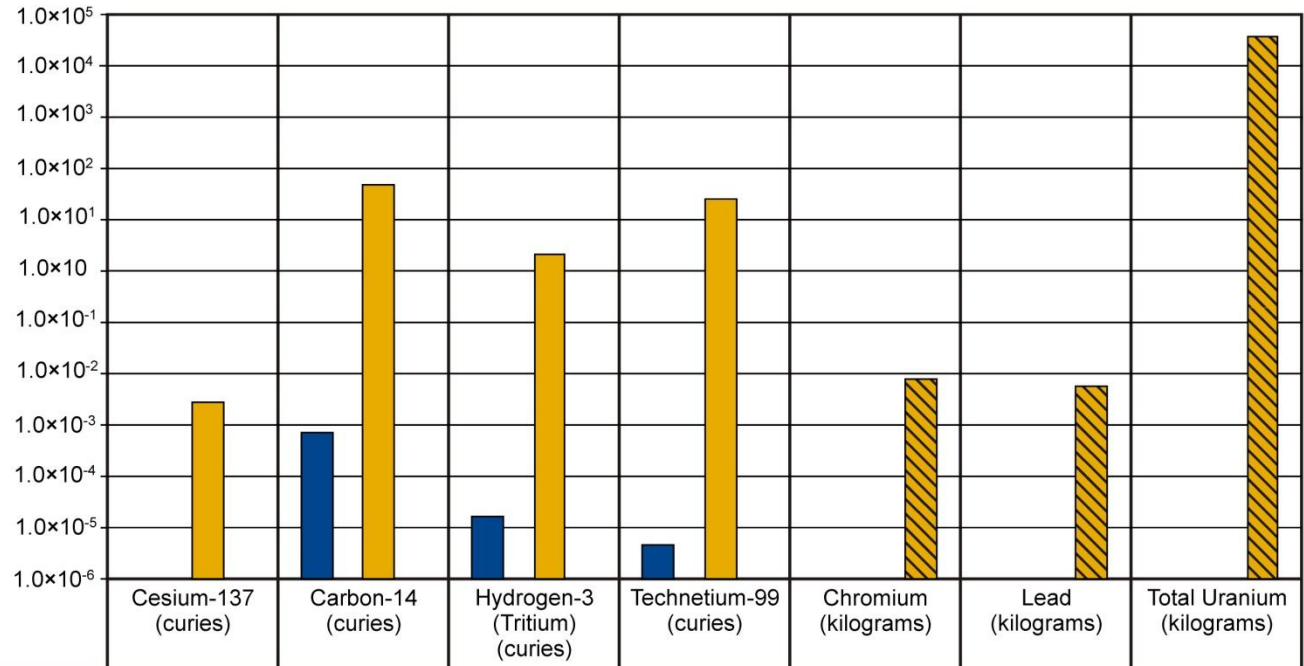
**Figure D-64. FTFF Decommissioning Alternative 1  
Distribution of Radioactive and Chemical Constituents of Potential Concern**



**Figure D-65. FFF Decommissioning Alternative 2  
Distribution of Radioactive and Chemical Constituents of Potential Concern**



Inventory (curies/kilograms)



■ FTFF site disposal <sup>a</sup> of radioactive COPCs	0.0	$7.65 \times 10^{-4}$	$1.73 \times 10^{-5}$	$4.52 \times 10^{-6}$	N/A	N/A	N/A
▨ FTFF site disposal <sup>a</sup> of chemical COPCs	N/A	N/A	N/A	N/A	0.0	0.0	0.0
■ IDF disposal of radioactive COPCs	$2.86 \times 10^{-3}$	$5.24 \times 10^1$	2.29	$2.72 \times 10^1$	N/A	N/A	N/A
▨ IDF disposal of chemical COPCs	N/A	N/A	N/A	N/A	$7.52 \times 10^{-3}$	$5.65 \times 10^{-3}$	$3.77 \times 10^4$

<sup>a</sup> FTFF site disposal denotes waste left in place at the FTFF site.

**Note:** Only FTFF Decommissioning Alternative 3 waste is included. Tank Closure alternative waste; Waste Management alternative waste; and offsite- and onsite-generated, non-Comprehensive Environmental Response, Compensation, and Liability Act, nontank waste are excluded.

**Key:** COPC=constituent of potential concern; FTFF=Fast Flux Test Facility; IDF=Integrated Disposal Facility; N/A=not applicable.

**Figure D-66. FTFF Decommissioning Alternative 3  
Distribution of Radioactive and Chemical Constituents of Potential Concern**

## **D.3 WASTE MANAGEMENT ALTERNATIVES**

### **D.3.1 Radionuclide and Chemical Inventories**

This section summarizes the radionuclide and chemical inventories analyzed for each of the three Waste Management alternatives. Appendix E, Section E.3.1, provides a summary description of the Waste Management alternatives analyzed in this *TC & WM EIS* and is partially reproduced in this section for the reader's convenience. Within the Waste Management alternatives, only three waste generators were identified for inclusion in the *TC & WM EIS* alternatives analyses, as follows:

- Secondary LLW and MLLW from operation of LLBG 218-W-5, trenches 31 and 34
- Secondary LLW and MLLW from operation of the Waste Receiving and Processing Facility (WRAP)
- Secondary LLW and MLLW from operation of the T Plant complex

Operation of an IDF and the RPPDF were estimated to generate insignificant quantities of secondary waste (e.g., workers' personal protective equipment and other contaminated waste materials). Data found for operation of the Central Waste Complex (CWC) concluded that it also generates insignificant quantities of secondary waste (SAIC 2010c). Generators of onsite non-CERCLA, non-tank-activity waste, and offsite waste are identified in Sections D.3.5 and D.3.6, respectively.

#### **D.3.1.1 Assumptions**

Assumptions for the Waste Management alternatives include the following:

- Due to uncertainties regarding the future needs of the waste management facilities at Hanford, the scope of the Waste Management alternatives included the assumption that expanded capabilities of the current treatment operations at the T Plant complex, the CWC, and WRAP would be necessary.
- Tank closure activities would generate the following waste streams, which would be disposed of on site in an IDF: ILAW glass; retired LAW melters; bulk vitrification glass; steam reforming waste; cast stone waste; sulfate grout waste; ETF-generated solid secondary waste; other solid secondary waste from tank farm and treatment processes, including treatment of the cesium and strontium capsules; and PPF glass. Rubble, soil, and equipment generated from clean closure activities would be disposed of in the RPPDF.
- Treatment of offsite LLW and MLLW would be completed off site either at the generator site or at a commercial treatment facility prior to shipment to Hanford. Section D.3.6 provides the offsite waste inventories and the basis for the inventory estimates.
- No additional offsite TRU or mixed TRU waste would be received at Hanford.
- Non-CERCLA, nontank LLW and MLLW would be generated at Hanford through 2035. Section D.3.5 provides this inventory and the basis for the inventory estimates.

- For analysis purposes, continued operation of LLBG 218-W-5, trenches 31 and 34, was analyzed under the Waste Management alternatives; however, it was assumed that IDF operations would commence in 2009 and that all waste generated that is appropriate for IDF disposal would be disposed of in an IDF.
- Activities proposed under the Tank Closure alternatives were assumed to determine the requirements for the ETF, 242-A Evaporator, and Borrow Area C; therefore, operations and replacement of these facilities were analyzed under the Tank Closure alternatives and not the Waste Management alternatives.
- Packaging and shipment of waste currently stored in a glass or ceramic form (commonly referred to as “German Logs”) were not analyzed in this *TC & WM EIS* (SAIC 2010c).

### D.3.2 Waste Management Alternative 1: No Action

Under this alternative, storage and treatment of LLW, MLLW, and TRU waste would continue at the CWC, and disposal would continue at LLBG 218-W-5, trenches 31 and 34, until an estimated operational closure date of 2035. Likewise, storage and treatment of onsite LLW, MLLW, and TRU waste would continue at WRAP and the T Plant complex. No shipments of offsite LLW, MLLW, or TRU waste would be accepted. For analysis purposes, it was assumed that construction of the 200-East Area IDF (IDF-East) would be discontinued in 2008 and no closure barriers would be constructed over the disposal trenches and waste treatment facilities. Administrative controls would be maintained for 100 years following operational closure of the disposal trenches.

Table D–82 shows the radioactive and chemical COPC inventories for Waste Management Alternative 1. These inventories would be disposed of in lined trenches 31 and 34 at LLBG 218-W-5.

**Table D–82. Waste Management Alternative 1 Inventory<sup>a</sup>**

	I-129	Cs-137	C-14	H-3 (Tritium)	U-233, -234, -235, -238	Np-237	Pu-239, -240	Sr-90	Tc-99	Cr	Hg	Pb
	Curies									Kilograms		
Secondary waste <sup>b</sup>	NR	2.39×10 <sup>-3</sup>	NR	NR	NR	NR	NR	1.63×10 <sup>-3</sup>	2.13×10 <sup>-3</sup>	NR	NR	NR

<sup>a</sup> Only three chemicals were reported (nitrate, total uranium, acetonitrile, benzene, butanol [n-butyl alcohol], 2,4,6-trichlorophenol, and polychlorinated biphenyls were not reported).

<sup>b</sup> Secondary waste includes workers’ personal protective equipment and other contaminated materials. Disposal would be in Low-Level Radioactive Waste Burial Ground 218-W-5, trenches 31 and 34.

**Key:** C=carbon; Cr=chromium; Cs=cesium; H=hydrogen; Hg=mercury; I=iodine; Np=neptunium; NR=not reported; Pb=lead; Pu=plutonium; Sr=strontium; Tc=technetium; U=uranium.

**Source:** SAIC 2011.

### D.3.3 Waste Management Alternative 2: Disposal in IDF, 200-East Area Only

Under this alternative, storage and treatment of LLW, MLLW, and TRU waste would continue using both the existing and the proposed expanded capabilities at the CWC, the T Plant complex, and WRAP. Appendix E, Sections E.3.2, E.3.3, and E.3.4, describe in detail the expanded CWC, T Plant complex, and WRAP facilities, respectively. Offsite waste would be limited to 62,000 cubic meters (81,000 cubic yards) of LLW and 20,000 cubic meters (26,000 cubic yards) of MLLW, with reception of shipments estimated to occur from 2010 through 2046. Onsite (Hanford), non-CERCLA, nontank waste would be generated through 2035. For analysis purposes, operation of LLBG 218-W-5, trenches 31 and 34, would continue through 2050. IDF-East would begin operations in 2009. Under this alternative, IDF-East would accept the following waste: tank closure activity waste; FFTF decommissioning waste; waste

management facility-generated (secondary) waste; onsite non-CERCLA, non-tank-activity waste; and offsite waste. A new disposal facility, the RPPDF, would be constructed for disposal of lightly contaminated rubble, soil, and equipment resulting from clean closure of tank farm facilities.

To reduce the combinations of IDF and RPPDF configurations that would require analysis in this *TC & WM EIS*, three disposal groups were developed and analyzed, as follows:

- *Disposal Group 1:* This group supports Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, 5, and 6C; FFTF Decommissioning Alternatives 2 and 3; and Waste Management Alternative 2 for onsite non-CERCLA, non-tank-activity waste and offsite waste. Both IDF-East and the RPPDF would operate through 2050, with capacities of 1.2 million cubic meters (1.57 million cubic yards) and 1.08 million cubic meters (1.41 million cubic yards), respectively.
- *Disposal Group 2:* This group supports Tank Closure Alternatives 2A and 6B, both Base and Option Cases; FFTF Decommissioning Alternatives 2 and 3; and Waste Management Alternative 2 for onsite non-CERCLA, non-tank-activity waste and offsite waste. Both IDF-East and the RPPDF would operate through 2100, with capacities of 425,000 cubic meters (556,000 cubic yards) and 8.37 million cubic meters (10.9 million cubic yards), respectively.
- *Disposal Group 3:* This group supports Tank Closure Alternative 6A, Base and Option Cases; FFTF Decommissioning Alternatives 2 and 3; and Waste Management Alternative 2 for onsite non-CERCLA, non-tank-activity waste, and offsite waste. Both IDF-East and the RPPDF would operate through 2165, with capacities of 425,000 cubic meters (556,000 cubic yards) and 8.37 million cubic meters (10.9 million cubic yards), respectively.

Table D–83 shows the radioactive and chemical COPC inventories for Waste Management Alternatives 2 and 3 (discussed below). Under Waste Management Alternative 2, disposal of these inventories would occur in IDF-East.

**Table D–83. Waste Management Alternatives 2 and 3 Radioactive and Chemical  
Constituents of Potential Concern Balance<sup>a</sup>**

	I-129	Cs-137	C-14	H-3 (Tritium)	U-233, -234, -235, -238	Np-237	Pu-239, -240	Sr-90	Tc-99	Cr	Hg	Pb
	Curies									Kilograms		
Secondary waste <sup>b</sup>	1.43×10 <sup>-5</sup>	1.04	4.04×10 <sup>-5</sup>	3.03×10 <sup>1</sup>	1.05×10 <sup>-3</sup>	6.17×10 <sup>-6</sup>	1.86	6.28	9.95×10 <sup>-2</sup>	1.39×10 <sup>1</sup>	2.29	2.32×10 <sup>2</sup>

<sup>a</sup> Only three chemicals were reported (nitrate, total uranium, acetonitrile, benzene, butanol [n-butyl alcohol], 2,4,6-trichlorophenol, and polychlorinated biphenyls were not reported).

<sup>b</sup> Secondary waste includes workers' personal protective equipment and other contaminated materials, as well as Waste Receiving and Processing Facility and T Plant complex low-level radioactive waste and mixed low-level radioactive waste, which would be disposed of in an Integrated Disposal Facility in the 200-East Area under Waste Management Alternative 2 and in the 200-West Area under Waste Management Alternative 3.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** C=carbon; Cr=chromium; Cs=cesium; H=hydrogen; Hg=mercury; I=iodine; Np=neptunium; Pb=lead; Pu=plutonium; Sr=strontium; Tc=technetium; U=uranium.

**Source:** SAIC 2011.

### **D.3.4 Waste Management Alternative 3: Disposal in IDF, 200-East and 200-West Areas**

Under this alternative, activities would be the same as those under Waste Management Alternative 2, except disposal of the waste would be split between IDF-East and a new IDF site in the 200-West Area (IDF-West). Except for the waste from tank closure activities, all of the waste streams discussed under

Waste Management Alternative 2 would be disposed of in IDF-West. Only the tank closure waste would be disposed of in IDF-East.

The three disposal groups under Waste Management Alternative 3 are as follows:

- *Disposal Group 1:* This group supports Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, 5, and 6C; FFTF Decommissioning Alternatives 2 and 3; and Waste Management Alternative 3 for onsite non-CERCLA, non-tank-activity waste and offsite waste. IDF-East would have a capacity of 1.1 million cubic meters (1.43 million cubic yards), IDF-West would have a capacity of 90,000 cubic meters (118,000 cubic yards), and the RPPDF would have a capacity of 1.08 million cubic meters (1.41 million cubic yards). All three facilities would operate through 2050.
- *Disposal Group 2:* This group supports Tank Closure Alternatives 2A and 6B, both Base and Option Cases; FFTF Decommissioning Alternatives 2 and 3; and Waste Management Alternative 3 for onsite non-CERCLA, non-tank-activity waste and offsite waste. IDF-East would have a capacity of 340,000 cubic meters (445,000 cubic yards), IDF-West would have a capacity of 90,000 cubic meters (118,000 cubic yards), and the RPPDF would have a capacity of 8.37 million cubic meters (10.9 million cubic yards). IDF-East and the RPPDF would operate through 2100. IDF-West would operate through 2050.
- *Disposal Group 3:* This group supports Tank Closure Alternative 6A, Base and Option Cases; FFTF Decommissioning Alternatives 2 and 3; and Waste Management Alternative 3 for onsite non-CERCLA, non-tank-activity waste and offsite waste. IDF-East would have a capacity of 340,000 cubic meters (445,000 cubic yards), IDF-West would have a capacity of 90,000 cubic meters (118,000 cubic yards), and the RPPDF would have a capacity of 8.37 million cubic meters (10.9 million cubic yards). IDF-East and the RPPDF would operate through 2165. IDF-West would operate through 2050.

Table D–83 shows the radioactive and chemical COPC inventories for Waste Management Alternatives 2 (discussed above) and 3. Under Waste Management Alternative 3, disposal of these inventories would occur in IDF-West.

### **D.3.5 Radionuclide and Chemical Inventory Estimates for Onsite Non-CERCLA, Non-Tank-Activity Waste**

This section summarizes the non-CERCLA, non-tank-waste-related radioactive and chemical waste inventories that would be generated at Hanford. Examples of facilities and operations that are expected to generate such waste include the Plutonium Finishing Plant; the T Plant complex; the Waste Encapsulation and Storage Facility; WRAP; the Waste Sampling and Characterization Facility; groundwater sampling activities; Pacific Northwest National Laboratory; the Cold Vacuum Drying Facility; the Canister Storage Building; and the Liquid Waste Processing Facilities, which include the Liquid Effluent Retention Facility, the ETF, the State-Approved Land Disposal Site, and the Treated Effluent Disposal Facility.

Estimates of the radionuclide and chemical inventories for the above sources were developed from the Hanford *Solid Waste Integrated Forecast Technical (SWIFT) Report, FY2006–FY2035* database (Barcot 2005). From this source, the volume of LLW and MLLW for the period 2006 through 2035 was estimated to be approximately 5,300 cubic meters (6,930 cubic yards) (SAIC 2011).

Table D–84 is a summary of the radioactive COPC inventory for the onsite non-CERCLA, non-tank-activity waste. Table D–85 is a summary of the chemical COPC inventory for the onsite non-CERCLA, non-tank-activity waste.

**Table D-84. Onsite Non-CERCLA, Non-Tank-Activity Waste Radioactive Constituents of Potential Concern Inventory<sup>a</sup>**

	Iodine-129	Cesium-137	Carbon-14	Hydrogen-3 (Tritium)	Uranium-233, -234,-235,-238	Neptunium-237	Plutonium-239, -240	Strontium-90	Technetium-99
Curies									
CH- and RH-LLW	9.98×10 <sup>-5</sup>	1.36×10 <sup>3</sup>	8.17×10 <sup>-1</sup>	2.68×10 <sup>3</sup>	2.24×10 <sup>-1</sup>	4.38×10 <sup>-5</sup>	4.22	1.75×10 <sup>3</sup>	7.95×10 <sup>-1</sup>
CH- and RH-MLLW	1.22×10 <sup>-3</sup>	1.35×10 <sup>3</sup>	6.88×10 <sup>-3</sup>	8.28×10 <sup>2</sup>	5.12×10 <sup>-1</sup>	7.33×10 <sup>-3</sup>	3.81	1.73×10 <sup>3</sup>	4.17×10 <sup>-1</sup>
<b>Total</b>	<b>1.32×10<sup>-3</sup></b>	<b>2.71×10<sup>3</sup></b>	<b>8.24×10<sup>-1</sup></b>	<b>3.51×10<sup>3</sup></b>	<b>7.36×10<sup>-1</sup></b>	<b>7.37×10<sup>-3</sup></b>	<b>8.03</b>	<b>3.48×10<sup>3</sup></b>	<b>1.21</b>

<sup>a</sup> Onsite generators only, including the Plutonium Finishing Plant, T Plant complex, Waste Encapsulation and Storage Facility, Waste Receiving and Processing Facility, Waste Sampling and Characterization Facility, groundwater sampling activities, Pacific Northwest National Laboratory, Cold Vacuum Drying Facility, Canister Storage Building, and Liquid Waste Processing Facilities (Liquid Effluent Retention Facility, Effluent Treatment Facility, State-Approved Land Disposal Site, and Treated Effluent Disposal Facility).

**Key:** CERCLA=Comprehensive Environmental Response, Compensation, and Liability Act; CH=contact-handled; LLW=low-level radioactive waste; MLLW=mixed low-level radioactive waste; RH=remote-handled.

**Source:** SAIC 2011.

**Table D-85. Onsite Non-CERCLA, Non-Tank-Activity Waste Chemical Constituents of Potential Concern Inventory<sup>a</sup>**

	Acetonitrile	Arsenic (As)	Benzene	Boron (B)	Butanol (N-butyl Alcohol)	Cadmium (Cd)	Chromium (Cr)	Fluorine (F)	Lead (Pb)	Manganese (Mn)	Mercury (Hg)	Molybdenum (Mo)	Nickel (Ni)	Nitrate (NO <sub>3</sub> )	PCBs	Silver (Ag)	Strontium (Sr)	Total Uranium (U)	2,4,6- Trichlorophenol
Kilograms																			
CH- and RH- MLLW	3.91	6.70	1.02	3.66	1.39×10 <sup>-3</sup>	4.95×10 <sup>1</sup>	1.80×10 <sup>2</sup>	2.74×10 <sup>2</sup>	2.58×10 <sup>4</sup>	4.76×10 <sup>1</sup>	8.99×10 <sup>1</sup>	9.39×10 <sup>-5</sup>	1.97	2.97×10 <sup>3</sup>	2.50×10 <sup>1</sup>	7.80×10 <sup>1</sup>	3.13	9.48×10 <sup>-1</sup>	NR

<sup>a</sup> Onsite generators only, including the Plutonium Finishing Plant, T Plant complex, Waste Encapsulation and Storage Facility, Waste Receiving and Processing Facility, Waste Sampling and Characterization Facility, groundwater sampling activities, Pacific Northwest National Laboratory, Cold Vacuum Drying Facility, Canister Storage Building, and Liquid Waste Processing Facilities (Liquid Effluent Retention Facility, Effluent Treatment Facility, State-Approved Land Disposal Site, and Treated Effluent Disposal Facility).

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** CERCLA=Comprehensive Environmental Response, Compensation, and Liability Act; CH=contact-handled; MLLW=mixed low-level radioactive waste; NR=not reported; PCB=polychlorinated biphenyl; RH=remote-handled.

**Source:** SAIC 2011.

### D.3.6 Projected Volumes, Radionuclide and Chemical Inventories for Offsite Waste

As part of DOE's January 6, 2006, Settlement Agreement with the State of Washington (as amended on June 5, 2008) regarding *State of Washington v. Bodman* (Civil No. 2:03-cv-05018-AAM), signed by DOE, Ecology, the Washington State Attorney General's Office, and the U.S. Department of Justice, this *TC & WM EIS* evaluated the transportation of LLW and MLLW from other DOE sites to Hanford for disposal. The volume of this offsite waste was established in the "Record of Decision for the Solid Waste Program, Hanford Site, Richland, WA: Storage and Treatment of Low-Level Waste and Mixed Low-Level Waste; Disposal of Low-Level Waste and Mixed Low-Level Waste, and Storage, Processing, and Certification of Transuranic Waste for Shipment to the Waste Isolation Pilot Plant" (69 FR 39449). The volumes are limited to 62,000 cubic meters (81,100 cubic yards) of LLW and 20,000 cubic meters (26,200 cubic yards) of MLLW. This volume was determined to be a reasonable starting point and followed the 2006 Settlement Agreement and its associated Memorandum of Understanding between DOE and Ecology, and was reflected in the 2006 Notice of Intent (71 FR 5655). The Preferred Alternative for waste management in the *Draft* and this *Final TC & WM EIS* also included limitations on, and exemptions for, offsite waste importation at Hanford, at least until the WTP is operational.

The DOE Office of River Protection and the *TC & WM EIS* team, in coordination with the DOE Office of Environmental Management, developed a report, *Analysis of Offsite-Generated Waste Projections, "Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site,"* dated July 13, 2006 (DOE 2006b), which documents the methodology and analysis related to offsite LLW and MLLW potentially requiring disposal at Hanford. The following is an excerpt from this report (DOE 2006b), followed by a summary of the projected waste characteristics, volumes, and radionuclide and chemical inventories. Offsite waste is analyzed under Waste Management Alternatives 2 and 3 only. It was assumed that no offsite waste would be accepted under Waste Management Alternative 1: No Action.

#### BACKGROUND

The Hanford Site hosts one of two regional disposal facilities for the Department of Energy's (DOE) low-level radioactive waste (LLW) and mixed LLW (MLLW) resulting from a February 2000 Record of Decision on the *Waste Management Programmatic Environmental Impact Statement (WM PEIS)*. The *Hanford Solid Waste Environmental Impact Statement* (2004) is the site-specific National Environmental Policy Act document that analyzed specific impacts resulting from disposal of onsite- and offsite-generated LLW and MLLW at Hanford. In January 2006, as a result of a settlement agreement with the State of Washington, the DOE agreed to prepare a new, expanded, comprehensive environmental impact statement (EIS) that combines the scope of the 2004 *Solid Waste EIS* and the developing *TC & WM EIS*. The Environmental Management (EM) Office of Disposal Operations, formerly the Office of Commercial Disposition Options, was asked to compile offsite-generated waste data as input to this new EIS. Waste data, e.g., projected waste volumes, radionuclide inventories, and hazardous chemical constituents are needed for analysis of impacts to humans and the environment within the EIS.

The information needed for the EIS was not readily available, so efforts were undertaken to use existing corporate information, supplemented by information from DOE waste managers. The EM program has corporate performance metrics that capture the actual and projected volume of LLW and MLLW for disposal from "baselined" projects. The information was not sufficiently detailed for modeling purposes, e.g. LLW and MLLW are combined, and data on radionuclide or hazardous chemical constituents is not collected and maintained corporately.



Waste volume projections and “disposition maps” were developed for the EM program in 1999 and 2000 as part of the EM Integration Project. At that time the EM Corporate Information System (Integrated Planning, Accountability, and Budgeting System or IPABS) was developed, including a “stream disposition data” module that provided detail on where individual waste streams were treated and disposed. Largely because of the resource requirements to supply and maintain the stream disposition data, EM management decided to forgo collection of waste volume information at the stream level as a corporate performance metric, in favor of waste volume disposed at the site or project level. Disposition maps, which schematically showed waste streams both within a site and between sites, were not produced between 2001 and 2005.

Due to various program planning needs associated with waste disposition, the Office of Commercial Disposition Options developed a new complex-wide LLW and MLLW data set and new, simplified disposition maps. The data requirements were significantly streamlined with the assistance of EM and other DOE waste managers. A new data collection module was constructed in September 2005, and data was compiled in late 2005 and early 2006. This data was readily available for analysis. Since the new data reflects only currently planned activities within EM, additional information was required to forecast LLW and MLLW that might be sent to Hanford from all offsite sources, e.g., unplanned EM projected waste volumes and waste from other DOE programs.

LLW and MLLW is generated at numerous DOE sites across the complex. Most of the volume of LLW and MLLW is generated from cleanup projects, versus ongoing operations. Over the past several years waste inventories that had been historically stored waiting for treatment and disposal, often called “legacy waste,” have nearly all been disposed due to contract incentives aimed at reducing life-cycle waste management infrastructure and costs. Estimates of potential, future offsite generated LLW and MLLW volumes requiring disposal in DOE regional disposal facilities are comprised primarily of waste generated in cleanup and decommissioning projects, rather than legacy waste. Much of this work is yet to be planned. Therefore, there are significant uncertainties in waste volume projections because waste is yet to be generated, and little characteristic information is available as previously discussed. This is a change from the situation during the early years of the EM program when most MLLW was in storage awaiting treatment and disposition.

In addition to uncertainties in waste volume, the newly collected LLW and MLLW waste data did not include radionuclide or hazardous chemical data needed for EIS modeling. EM has not collected radionuclide and hazardous constituent information since the 1990’s, when data was collected to support the Federal Facilities Task Force and the *WM PEIS* development. Documented information on radionuclides is found in the *Low-Level Waste Capacity Report*, Revision 2, produced in 2000. This document continues to serve as a source for waste characteristics.

It is difficult to predict the radionuclide and hazardous chemical composition of waste projected in the future, particularly from cleanup programs, because the waste does not exist until the cleanup work progresses. Forecasts are based on best available characterization of the site or facility, the technology selected for cleanup, and the work plans. For this reason, the forecast waste characteristics data in most instances relies on representative information from similar waste streams recently sent to disposal. Actual LLW and MLLW disposal profiles were requested from waste managers and several were judged to have the necessary data for modeling and be suitable for projected waste streams. The Rocky Flats Environmental Technology Site was a source of recent waste

profiles for MLLW, one of which covered debris including metals, solvents, and waste requiring macro-encapsulation. The characteristics of this stream were judged to be a reasonable representation for radiological and hazardous chemical constituents of MLLW from future cleanup projects.

## **DISCUSSION**

For the purposes of the new consolidated EIS, the volumes of offsite-generated LLW and MLLW in the existing *Hanford Solid Waste EIS* Record of Decision, namely 62,000 cubic meters for LLW and 20,000 cubic meters for MLLW, should continue to be used in the new EIS. These values sufficiently accommodate current projections and include anticipated new projections for sites where significant cleanup activities and operations are not yet fully scoped. Due to the timing of the EIS and the implementation of resulting record of decision, offsite waste forecasts are largely assumed to begin in 2010, so examination of post 2010 waste volume data collected by EM was the starting point of the analysis. The makeup of the waste volume forecast is discussed below and the attached table summarizes the information.

### **Environmental Management**

A high degree of uncertainty exists in how much LLW could be shipped from EM sites to Hanford after 2010. Based on current practices, waste from EM sites without onsite disposal capacity can be expected to utilize both DOE regional and commercial disposal facilities. Only EM sites completing cleanup beyond 2010 are considered in this forecast. Sites that are major EM contributors to EM LLW disposal projections in 2011 to 2035 (over 1,000 cubic meters) are: Idaho National Laboratory (INL), Paducah, and Oak Ridge. Future waste projections from expected decommissioning at Portsmouth and West Valley, and additional work at Paducah have not yet been developed and reported to EM, but must also be considered.

The recently collected planning data includes no EM offsite shipments of LLW and MLLW projected for the Hanford regional disposal facility. It is not surprising that current baselines do not include shipments to Hanford because, due to the current suspension of off-site shipments, EM projects were replanned to utilize alternate sites. About 112,000 cubic meters of LLW are projected to go to the regional disposal site at Nevada Test Site (NTS) between 2011 and 2035. No MLLW is currently proposed to be disposed at NTS after closure of the current facility at the end of 2010. About 11,700 cubic meters of LLW and 900 cubic meters of MLLW were identified as needing a disposal facility to be determined (TBD) after 2010, some of which may be disposed in a commercial facility. DOE policy, economic factors, and waste acceptance criteria are key to waste management decisions. Coincidentally, the 62,000 cubic meters in the *Hanford Solid Waste EIS* Record of Decision equates to about half of the life-cycle LLW projection for offsite disposal for NTS and TBD combined.

West Valley Demonstration Project is at the site of a former commercial reprocessing plant where DOE and the State of New York are responsible for cleanup. West Valley has a site-wide *Decommissioning and Long-Term Stewardship* EIS in preparation, but agreement on the end state has not occurred. Thus, there is no “baselined” scope of work beyond 2010 and no baseline estimate of future waste from West Valley, although a draft EIS is available with a range of waste projections. LLW from West Valley is expected to contain a variety of radionuclides, including transuranics and fission products, and be in a variety of forms. West Valley is expected to produce significant volumes of waste for

offsite disposal between 2010 and 2030 through facility decommissioning activities. Based on discussions with site waste managers and information in the draft EIS, waste volumes associated with Alternative 4, a “delayed in-place” decommissioning were assumed for this forecast. A LLW volume of 12,000 cubic meters was judged to be a reasonable forecast. Although Alternative 4 in the draft EIS does not have an estimate of MLLW volumes, other alternatives indicated that MLLW debris might be generated during decommissioning at West Valley. Due to the distinct possibility of MLLW generation at West Valley, 500 cubic meters of MLLW was judged to be a reasonable forecast. No radiological or hazardous chemical information was available for modeling, so representative information was selected. For LLW, the complex-wide radiological profile in the DOE Capacity Report was selected as representative; for MLLW a representative Rocky Flats debris stream profile with radiological and hazardous chemical data was selected which included metals, solvents, and waste requiring macro-encapsulation.

Idaho National Laboratory (INL) is managed by the Office of Nuclear Energy (NE); however, EM has a large cleanup project that generates waste at that site during the first several years of the period of concern. EM currently operates the low-level waste disposal area for operational LLW and the Idaho CERCLA Disposal Facility for Comprehensive Environmental Response, Cleanup [*sic*; Compensation], and Liability Act (CERCLA) waste at INL. The INL is examining future alternatives for closure of their onsite disposal facility for LLW from operations. Closure may be required to implement the terms of their final remedy decision currently being developed. Closure of this INL disposal facility would require another disposal option for the LLW currently disposed there which is generated by NE, EM, and Naval Reactors; therefore, modeling of a Hanford alternative is appropriate.

The existing NE programs at INL estimated approximately 1,100 cubic meters of remote-handled LLW and approximately 10 cubic meters of MLLW shipped to Hanford after 2010. Because of the proximity of Hanford versus NTS, Hanford disposal would be a logical place for this and other future waste not capable of being disposed of commercially due to higher activity levels (e.g., equivalent of Class B and C commercial LLW). The annual waste quantities are consistent with those reported between 2010 and 2035 to EM’s planning data base. After discussions with waste managers at DOE-Idaho Operations, a representative radiological profile for modeling LLW consisting of Test Reactor Area depleted demineralizer resins was used for the radiological characteristics. This is an existing and ongoing post-2010 remote-handled LLW stream disposed of at INL. The same discussions suggested use of an INL MLLW debris waste stream from the INTEC facility for radiological characteristics and tank farm-related waste information for the chemical characteristics for the small MLLW stream.

The INL plans to play a prominent role in development of the Generation IV prototype nuclear reactor, piloting of an Advanced Fuel Cycle Facility, and expansion for the Center of Advanced Energy Studies generating waste far into the future. In addition, some EM MLLW was historically managed as transuranic waste, but when surveyed has a radionuclide concentration of 10 to 100 nanocuries per gram. The forecast includes future new LLW and MLLW streams from INL. No characteristics information is available, but the waste projected between 2010 and 2020 is assumed to be similar to other waste at INL. The existing profile for Test Reactor Area depleted demineralizer resins is appropriate for the LLW stream of 6,500 cubic meters, while the Rocky Flats

radiological and chemical characteristics are representative for the MLLW stream of 6,330 cubic meters.

Portsmouth and Paducah sites are home to large enrichment plants that will be decommissioned after 2010 by the EM program. Significant volumes of waste are expected to be generated and disposed then at DOE and/or commercial disposal facilities. However, no data is available from these projects, because they are in the early design stage and work scope is not yet planned. The forecast includes 6,500 cubic meters of LLW from each site. Portsmouth waste is forecast between 2010 and 2020, while Paducah waste is forecast between 2015 and 2035. No MLLW was assumed from these sites, since the waste is largely debris from large enrichment plants contaminated primarily with uranium. Representative waste characteristics were selected from existing cleanup waste profiles from the Oak Ridge Gaseous Diffusion Plant (East Tennessee Technology Park) where decommissioning a similar site is progressing and is scheduled to be complete by 2010. Four representative profiles were judged to be appropriate and applied proportionally to the projected waste volumes at Portsmouth and Paducah.

Los Alamos National Laboratory (LANL) is operated by the National Nuclear Security Administration and has onsite disposal facilities for its LLW. The EM program at LANL is currently characterizing waste historically managed as TRU waste. A portion of this waste when characterized does not meet the definition of TRU waste and cannot be disposed on site at LANL because MLLW disposal is not permitted. Projected MLLW that falls between 10 and 100 nanocuries per gram is a candidate for the Hanford forecast after closure of the NTS MLLW facility. The forecast volume of LANL MLLW between 2010 and 2020, when all TRU characterization work is expected to be complete, is 400 cubic meters. As a result of discussions with waste managers at LANL, radiological profiles were obtained for inorganic cemented sludge from an on-site water treatment plant. No chemical profile was available for the LANL sludge, so comparable INL chemical characterization data for two batches of MLLW sludge was obtained and judged as representative.

Savannah River Site (SRS) utilizes both onsite and commercial disposal facilities for its LLW. No LLW is forecast to be shipped to Hanford. In discussions with waste managers at SRS, a waste stream with 100 cubic meters of MLLW was identified as a candidate for disposal at Hanford in 2010-2012 following the NTS MLLW facility closure. SRS waste managers provided a radiological profile for the MLLW which contains some Pu-238 and Pu-239 constituents. No chemical characteristics were available, so the chemical profile for Rocky Flats debris MLLW waste was judged as representative. To accommodate future, as yet unplanned MLLW generation at SRS, another MLLW stream is included in the forecast with 6,330 cubic meters between 2010 and 2035. The same Rocky Flats debris waste profile was judged as representative for the radiological and hazardous chemical constituents.

### **Office of Science Waste**

The Office of Science (SC) is responsible for ongoing operations at eight DOE laboratory sites. Historically, the SC laboratories shipped LLW to Hanford for disposal, but were prevented from doing so recently due to legal impediments. SC waste managers indicated most waste generated from operations is now planned for NTS or commercial disposal.

Three SC-operated laboratories: Argonne National Laboratory (ANL), Brookhaven National Laboratory (BNL), and Oak Ridge National Laboratory (ORNL) forecasted future waste that could be disposed at Hanford. ANL forecasted 100 cubic meters of LLW from decommissioning of facilities between 2011 and 2018. Radiological characteristics of this future LLW volume was not available from waste managers, so the Capacity Report complex wide profile was judged to be appropriate due to the variety of nuclear applications at ANL. BNL waste managers identified two LLW streams totaling 70 cubic meters with corresponding radioactive waste profiles. The streams include sealed sources (disposed between 2010 and 2015) and decommissioning waste from the Brookhaven Linear Isotope Production facility between 2030 and 2035.

SC waste managers estimated 730 cubic meters of LLW between 2010 and 2035 timeframe from ongoing operations in Oak Ridge. Radiological characteristics of this future LLW volume was not available from waste managers, so the Capacity Report complex wide profile was judged to be appropriate due to the variety of nuclear applications at ORNL. In addition to operations waste, there are a number of facilities at the Oak Ridge Reservation that have not yet been scheduled for decommissioning by SC, EM, or NNSA. The scope of the work and resulting waste is uncertain, but additional waste is likely after 2010. Some of this waste will be disposed off site at DOE regional disposal facilities and commercial facilities, consistent with the Oak Ridge experience to date. The forecast includes a LLW stream of 6,500 cubic meters and a MLLW stream of 6,330 cubic meters for future waste from Oak Ridge. For LLW, the Capacity Report complex-wide profile was judged appropriate due to the variety of waste from cleanup. For MLLW, the Rocky Flats debris stream also applied at West Valley, INL, and SRS forecasts was judged appropriate for the variety of waste expected from cleanup.

### **Naval Reactors**

Naval Reactors (a part of NNSA) produces LLW as a result of operations of various shipyards and laboratories across the nation. In addition to Naval Reactors LLW already disposed at Hanford, a new Naval Reactors waste stream is included in the forecast for analytical purposes. As mentioned previously, LLW generated at the Naval Reactors Facility (NRF) at INL is currently disposed onsite at INL, but the LLW disposal facility used by Naval Reactors at INL may close in the near future as a result of the site cleanup agreement. Discussions with Naval Reactors waste managers resulted in a projected volume of 22,000 cubic meters of routine LLW from the NRF at INL that is included in the Hanford forecast between 2008 and 2046. A radiological profile has been provided by Naval Reactors for this LLW.

### **CONCLUSIONS**

DOE used available waste volume projections to frame the forecast for the *Hanford Tank Closure and Waste Management EIS*. The analysis focused on ongoing operations and post-2010 cleanup activities that will generate wastes requiring or utilizing DOE regional disposal facilities. After contacting waste managers, expert judgment was applied to waste projection and characteristics data to develop a waste forecast for the new Hanford EIS. Considerable uncertainty remains in the waste projections, due to limited planning data and the uncertainties in the cleanup program scope from where most waste volumes arise. However, conservative assumptions were employed to support EIS analyses. This analysis confirms the need to maintain the waste volumes included in the record of decision from the 2004 *Hanford Solid Waste EIS* (62,000 cubic meters for LLW and

20,000 cubic meters of MLLW) and provides assumptions for modeling purposes, including offsite sources, timing, and sources of radiological/chemical characteristics.

The process described above resulted in estimated waste volumes, waste characteristics, final waste forms, and shipment dates for the waste generated by other DOE sites that would be shipped to Hanford for disposal. Radioactivity estimates (measured in curies) for over 110 isotopes and chemical estimates (measured in kilograms) for 41 chemical compounds also were developed.

As stated above, the *Analysis of Offsite-Generated Waste Projections, "Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site"* (DOE 2006b) includes estimates for wastes generated at the West Valley Demonstration Project that may require disposal at Hanford. The estimates were 12,000 cubic meters (15,700 cubic yards) of LLW and 500 cubic meters (650 cubic yards) of MLLW. Since then, DOE has prepared the *Final Environmental Impact Statement for Decommissioning and/or Long-Term Stewardship at the West Valley Demonstration Project and Western New York Nuclear Service Center* (DOE and NYSERDA 2010), which slightly revised these estimates. Under the Sitewide Removal Alternative, which is the alternative that would result in the largest waste volume requiring offsite disposal, the revised estimated volumes are approximately 13,710 cubic meters (17,930 cubic yards) of LLW and 510 cubic meters (670 cubic yards) of MLLW (Burandt 2008). Due to the high degree of uncertainty involved in estimating waste shipments to Hanford after 2010, the current estimates of 12,000 cubic meters (15,700 cubic yards) of LLW and 500 cubic meters (650 cubic yards) of MLLW are considered reasonable estimates and appropriate for analysis purposes in this *TC & WM EIS*. Additionally, since the above analysis was performed, DOE has initiated planning for a new MLLW disposal facility at the Nevada National Security Site, formerly the Nevada Test Site, to continue to provide two DOE residual disposal facilities consistent with the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (DOE 1997) and the LLW and MLLW Record of Decision (65 FR 10061).

Table D-86 summarizes the DOE sites, waste form characteristics, and volumes and years of shipment projected for waste shipments from other DOE sites to Hanford. Table D-87 summarizes the 9 radioactive COPC inventories associated with the potential wastes from each DOE site. Table D-88 summarizes the 15 chemical COPC inventories associated with the potential wastes from each DOE site.

**Table D–86. Offsite Waste Projection Characteristics by U.S. Department of Energy Site**

DOE Site and Waste Category	Waste Category <sup>a</sup>	Final Waste Form	Waste-Form Volume (cubic meters)	Year of Shipment	
				Start	End
<b>West Valley Demonstration Project</b>					
WV-Class A	LLW-Class A	Grouted	11,000	2022	2042
WV-Class B	LLW-Class B	Grouted	200	2022	2042
WV-Class C	LLW-Class C	Grouted	800	2022	2042
<b>Idaho National Laboratory</b>					
RH-LLW	RH-LLW	Resins	30	2022	2022
RH-LLW	RH-LLW	Resins	200	2023	2027
RH-LLW	RH-LLW	Resins	200	2030	2032
RH-LLW	RH-LLW	Resins	200	2033	2037
RH-LLW	RH-LLW	Resins	200	2038	2042
RH-LLW	RH-LLW	Resins	270	2042	2047
<b>Brookhaven National Laboratory</b>					
BNL sealed sources	LLW	Sealed sources	5	2022	2027
BNL-2 – Brookhaven Linear Isotope Production Facility	LLW	Encapsulated activated metals, concrete debris, lead (solid)	65	2042	2047
<b>Oak Ridge National Laboratory – LLW</b>	LLW	Grout	730	2022	2047
<b>Argonne National Laboratory – LLW</b>	LLW	Grout	100	2023	2030
<b>Naval Reactors</b>					
LLW – Bettis, Idaho	LLW	Solid	22,000	2022	2046
<b>Paducah</b>					
LLW No. 1	LLW	Solids (metal)	845	2027	2047
LLW No. 2	LLW	Solids (metal)	195	2027	2047
LLW No. 3	LLW	Solids (metal)	1,690	2027	2047
LLW No. 4	LLW	Solids (metal)	3,770	2027	2047
<b>Portsmouth</b>					
LLW No. 1	LLW	Solids (metal)	845	2022	2032
LLW No. 2	LLW	Solids (metal)	195	2022	2032
LLW No. 3	LLW	Solids (metal)	1,690	2022	2032
LLW No. 4	LLW	Solids (metal)	3,770	2022	2032
<b>Idaho National Laboratory – RH-LLW<sup>b</sup></b>	RH-LLW	Resins	0	N/A	N/A
<b>Oak Ridge Reservation – LLW</b>	LLW	Grout	6,500	2022	2047
<b>Total LLW</b>			<b>55,500</b>		



**Table D–86. Offsite Waste Projection Characteristics by U.S. Department of Energy Site  
(continued)**

DOE Site and Waste Category	Waste Category <sup>a</sup>	Final Waste Form	Waste-Form Volume (cubic meters)	Year of Shipment	
				Start	End
West Valley Demonstration Project – MLLW	MLLW	Debris	500	2022	2042
Los Alamos National Laboratory – MLLW	MLLW	Cemented sludges	400	2022	2032
Savannah River Site – MLLW	RH-MLLW	Grout	100	2022	2024
<b>Idaho National Laboratory</b>					
CH-MLLW	CH-MLLW	Debris	1	2022	2022
CH-MLLW	CH-MLLW	Debris	2	2023	2028
CH-MLLW	CH-MLLW	Debris	2	2028	2033
CH-MLLW	CH-MLLW	Debris	2	2033	2037
CH-MLLW	CH-MLLW	Debris	2	2038	2042
CH-MLLW	CH-MLLW	Debris	1	2042	2047
Idaho National Laboratory – MLLW	MLLW-D&D	Debris	6,330	2022	2047
Savannah River Site – MLLW	MLLW-D&D	Debris	6,330	2022	2047
Oak Ridge Reservation – MLLW	MLLW-D&D	Debris	6,330	2022	2047
<b>Total MLLW</b>			<b>20,000</b>		

<sup>a</sup> Per *Hanford Site Solid Waste Acceptance Criteria*, Rev. 12 (Fluor Hanford 2005b).

<sup>b</sup> As part of a reevaluation of the inventories within the Waste Management alternatives, the Idaho National Laboratory RH-LLW resins waste, with a volume of 6,500 cubic meters, was not modeled in the groundwater analysis in this environmental impact statement. Such an action by DOE could be the result of a number of changes at the Hanford site, such as revisions to the Integrated Disposal Facility waste acceptance criteria.

**Note:** To convert cubic meters to cubic yards, multiply by 1.308.

**Key:** BNL=Brookhaven National Laboratory; CH=contact-handled; D&D=decontamination and decommissioning; DOE=U.S. Department of Energy; LLW=low-level radioactive waste; MLLW=mixed low-level radioactive waste; N/A=not applicable; RH=remote-handled; WV=West Valley.

**Source:** SAIC 2011.

**Table D–87. Summary of Offsite Radioactive Constituents of Potential Concern Inventories by U.S. Department of Energy Site**

DOE Site and Waste Category	Radionuclide Inventory (curies)								
	Iodine-129	Cesium-137	Carbon-14	Hydrogen-3 (Tritium)	Uranium-233, -234, -235, -238	Neptunium-237	Plutonium-239, -240	Strontium-90	Technetium-99
West Valley Demonstration Project – LLW	3.04×10 <sup>-2</sup>	3.71×10 <sup>5</sup>	2.90×10 <sup>1</sup>	3.42×10 <sup>4</sup>	3.39×10 <sup>1</sup>	1.22×10 <sup>-2</sup>	1.09×10 <sup>1</sup>	4.28×10 <sup>5</sup>	2.99
INL – RH-LLW	2.20	2.20×10 <sup>3</sup>	8.80×10 <sup>2</sup>	5.50×10 <sup>2</sup>	NR	NR	7.40	8.25×10 <sup>2</sup>	5.72×10 <sup>1</sup>
Brookhaven National Laboratory – LLW	NR	9.20×10 <sup>3</sup>	NR	2.44×10 <sup>-4</sup>	NR	NR	2.44×10 <sup>-4</sup>	4.04	NR
Oak Ridge National Laboratory – LLW	1.85×10 <sup>-3</sup>	2.26×10 <sup>4</sup>	1.77	2.08×10 <sup>3</sup>	2.06	7.45×10 <sup>-4</sup>	6.64×10 <sup>-1</sup>	2.61×10 <sup>4</sup>	1.82×10 <sup>-1</sup>
Argonne National Laboratory – LLW	2.53×10 <sup>-4</sup>	3.09×10 <sup>3</sup>	2.42×10 <sup>-1</sup>	2.85×10 <sup>2</sup>	2.83×10 <sup>-1</sup>	1.02×10 <sup>-4</sup>	9.10×10 <sup>-2</sup>	3.57×10 <sup>3</sup>	2.49×10 <sup>-2</sup>
Naval Reactors – LLW	NR	3.85×10 <sup>1</sup>	1.46	3.26	4.73×10 <sup>-4</sup>	NR	5.06×10 <sup>-2</sup>	2.06×10 <sup>1</sup>	1.16
Paducah – LLW	NR	NR	NR	NR	4.63	1.79×10 <sup>-2</sup>	2.31×10 <sup>-2</sup>	NR	6.95×10 <sup>2</sup>
Portsmouth – LLW	NR	NR	NR	NR	4.63	1.79×10 <sup>-2</sup>	2.31×10 <sup>-2</sup>	NR	6.95×10 <sup>2</sup>
INL – RH-LLW <sup>a</sup>	0	0	0	0	NR	NR	0	0	0
Oak Ridge Reservation – LLW	1.64×10 <sup>-2</sup>	2.01×10 <sup>5</sup>	1.57×10 <sup>1</sup>	1.85×10 <sup>4</sup>	1.84×10 <sup>1</sup>	6.63×10 <sup>-3</sup>	5.92	2.32×10 <sup>5</sup>	1.62
<b>Total LLW</b>	<b>2.25</b>	<b>6.09×10<sup>5</sup></b>	<b>9.28×10<sup>2</sup></b>	<b>5.56×10<sup>4</sup></b>	<b>6.39×10<sup>1</sup></b>	<b>5.55×10<sup>-2</sup></b>	<b>2.51×10<sup>1</sup></b>	<b>6.91×10<sup>5</sup></b>	<b>1.45×10<sup>3</sup></b>
West Valley Demonstration Project – MLLW	NR	NR	NR	NR	8.00	NR	1.14×10 <sup>1</sup>	NR	NR
Los Alamos National Laboratory – MLLW	NR	1.28×10 <sup>-2</sup>	NR	NR	1.66×10 <sup>-1</sup>	1.28×10 <sup>-2</sup>	1.28	1.28×10 <sup>-2</sup>	NR
Savannah River Site – RH-MLLW	NR	NR	NR	NR	NR	2.05×10 <sup>-5</sup>	1.16×10 <sup>1</sup>	NR	NR
INL – RH-MLLW	1.60×10 <sup>-2</sup>	1.60×10 <sup>4</sup>	4.80×10 <sup>-2</sup>	8.00×10 <sup>1</sup>	1.12	8.64×10 <sup>-1</sup>	1.90×10 <sup>1</sup>	1.60×10 <sup>4</sup>	3.04
INL – MLLW	NR	NR	NR	NR	1.01×10 <sup>2</sup>	NR	1.44×10 <sup>2</sup>	NR	NR
Savannah River Site – MLLW	NR	NR	NR	NR	1.01×10 <sup>2</sup>	NR	1.44×10 <sup>2</sup>	NR	NR
Oak Ridge Reservation – MLLW	NR	NR	NR	NR	1.01×10 <sup>2</sup>	NR	1.44×10 <sup>2</sup>	NR	NR
<b>Total MLLW</b>	<b>1.60×10<sup>-2</sup></b>	<b>1.60×10<sup>4</sup></b>	<b>4.80×10<sup>-2</sup></b>	<b>8.00×10<sup>1</sup></b>	<b>3.13×10<sup>2</sup></b>	<b>8.77×10<sup>-1</sup></b>	<b>4.76×10<sup>2</sup></b>	<b>1.60×10<sup>4</sup></b>	<b>3.04</b>
<b>Total LLW and MLLW</b>	<b>2.26</b>	<b>6.25×10<sup>5</sup></b>	<b>9.28×10<sup>2</sup></b>	<b>5.57×10<sup>4</sup></b>	<b>3.77×10<sup>2</sup></b>	<b>9.32×10<sup>-1</sup></b>	<b>5.01×10<sup>2</sup></b>	<b>7.07×10<sup>5</sup></b>	<b>1.46×10<sup>3</sup></b>

<sup>a</sup> As part of a reevaluation of the inventories within the Waste Management alternatives, the INL RH-LLW resins waste, with the following inventory, was not modeled in the groundwater analysis in this environmental impact statement: iodine-129=1.30×10<sup>1</sup> curies; cesium-137=1.30×10<sup>1</sup> curies; carbon-14=5.20×10<sup>3</sup> curies; hydrogen-3 (tritium)=3.25×10<sup>3</sup> curies; plutonium-239, -240=4.37×10<sup>1</sup> curies; strontium-90=4.88×10<sup>3</sup> curies; technetium-99=3.38×10<sup>2</sup> curies. Such an action by DOE could be the result of a number of changes at the Hanford site, such as revisions to the Integrated Disposal Facility waste acceptance criteria.

**Key:** DOE=U.S. Department of Energy; INL=Idaho National Laboratory; LLW=low-level radioactive waste; MLLW=mixed low-level radioactive waste; NR=not reported; RH=remote-handled.

**Source:** SAIC 2011.

**Table D-88. Summary of Offsite Chemical Constituents of Potential Concern Inventories by U.S. Department of Energy Site**

DOE Site and Waste Category	Chemical Inventory (kilograms)														
	Arsenic (As)	Boron (B)	Cadmium (Cd)	Chromium (Cr)	Fluorine (F)	Lead (Pb)	Manganese (Mn)	Mercury (Hg)	Molybdenum (Mo)	Nickel (Ni)	Nitrate (NO <sub>3</sub> )	PCBs	Silver (Ag)	Strontium (Sr)	Total Uranium (U)
WVDP – LLW	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR
Idaho National Laboratory – RH-LLW	5.06×10 <sup>-1</sup>	NR	3.30×10 <sup>-3</sup>	2.24	NR	NR	NR	NR	NR	NR	NR	NR	6.93×10 <sup>-3</sup>	NR	NR
Brookhaven National Laboratory – LLW	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR
Oak Ridge National Laboratory – LLW	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR
Argonne National Laboratory – LLW	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR
Naval Reactors – LLW	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR
Paducah – LLW	3.77×10 <sup>-3</sup>	NR	5.95×10 <sup>-2</sup>	2.15	NR	5.26×10 <sup>-1</sup>	NR	NR	NR	NR	NR	NR	3.77×10 <sup>-3</sup>	NR	NR
Portsmouth – LLW	3.77×10 <sup>-3</sup>	NR	5.95×10 <sup>-2</sup>	2.15	NR	5.26×10 <sup>-1</sup>	NR	NR	NR	NR	NR	NR	3.77×10 <sup>-3</sup>	NR	NR
Idaho National Laboratory – RH-LLW <sup>a</sup>	0	NR	0	0	NR	NR	NR	NR	NR	NR	NR	NR	0	NR	NR
Oak Ridge Reservation – LLW	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR	NR
<b>Total LLW</b>	<b>5.14×10<sup>-1</sup></b>	<b>NR</b>	<b>1.22×10<sup>-1</sup></b>	<b>6.55</b>	<b>NR</b>	<b>1.05</b>	<b>NR</b>	<b>NR</b>	<b>NR</b>	<b>NR</b>	<b>NR</b>	<b>NR</b>	<b>1.45×10<sup>-2</sup></b>	<b>NR</b>	<b>NR</b>

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Appendix D • Waste Inventories

**Table D-88. Summary of Offsite Chemical Constituents of Potential Concern Inventories by U.S. Department of Energy Site (continued)**

DOE Site and Waste Category	Chemical Inventory (kilograms)														
	Arsenic (As)	Boron (B)	Cadmium (Cd)	Chromium (Cr)	Fluorine (F)	Lead (Pb)	Manganese (Mn)	Mercury (Hg)	Molybdenum (Mo)	Nickel (Ni)	Nitrate (NO <sub>3</sub> )	PCBs	Silver (Ag)	Strontium (Sr)	Total Uranium (U)
WVDP – MLLW	4.67×10 <sup>-2</sup>	NR	4.34×10 <sup>-2</sup>	1.68×10 <sup>-2</sup>	NR	1.49×10 <sup>-2</sup>	NR	8.70×10 <sup>-3</sup>	NR	7.75×10 <sup>-1</sup>	NR	NR	5.15×10 <sup>-2</sup>	NR	NR
Los Alamos National Laboratory – MLLW	2.19	NR	3.13	6.43×10 <sup>1</sup>	NR	1.34×10 <sup>2</sup>	NR	8.04	NR	NR	NR	NR	1.83×10 <sup>1</sup>	NR	NR
Savannah River Site – RH-MLLW	9.34×10 <sup>-3</sup>	NR	8.68×10 <sup>-3</sup>	3.35×10 <sup>-3</sup>	NR	2.97×10 <sup>-3</sup>	NR	1.74×10 <sup>-3</sup>	NR	1.55×10 <sup>-1</sup>	NR	NR	1.03×10 <sup>-2</sup>	NR	NR
Idaho National Laboratory – RH-MLLW	5.50×10 <sup>-1</sup>	NR	1.26×10 <sup>2</sup>	9.00	NR	4.41	NR	2.08×10 <sup>1</sup>	NR	NR	NR	NR	2.70	NR	NR
Idaho National Laboratory – MLLW	5.91×10 <sup>-1</sup>	NR	5.49×10 <sup>-1</sup>	2.12×10 <sup>-1</sup>	NR	1.88×10 <sup>-1</sup>	NR	1.10×10 <sup>-1</sup>	NR	9.81	NR	NR	6.52×10 <sup>-1</sup>	NR	NR
Savannah River Site – MLLW	5.91×10 <sup>-1</sup>	NR	5.49×10 <sup>-1</sup>	2.12×10 <sup>-1</sup>	NR	1.88×10 <sup>-1</sup>	NR	1.10×10 <sup>-1</sup>	NR	9.81	NR	NR	6.52×10 <sup>-1</sup>	NR	NR
Oak Ridge Reservation – MLLW	5.91×10 <sup>-1</sup>	NR	5.49×10 <sup>-1</sup>	2.12×10 <sup>-1</sup>	NR	1.88×10 <sup>-1</sup>	NR	1.10×10 <sup>-1</sup>	NR	9.81	NR	NR	6.52×10 <sup>-1</sup>	NR	NR
<b>Total MLLW</b>	<b>4.57</b>	<b>NR</b>	<b>1.31×10<sup>2</sup></b>	<b>7.39×10<sup>1</sup></b>	<b>NR</b>	<b>1.39×10<sup>2</sup></b>	<b>NR</b>	<b>2.91×10<sup>1</sup></b>	<b>NR</b>	<b>3.04×10<sup>1</sup></b>	<b>NR</b>	<b>NR</b>	<b>2.30×10<sup>1</sup></b>	<b>NR</b>	<b>NR</b>
<b>Total LLW and MLLW</b>	<b>5.09</b>	<b>NR</b>	<b>1.31×10<sup>2</sup></b>	<b>8.05×10<sup>1</sup></b>	<b>NR</b>	<b>1.40×10<sup>2</sup></b>	<b>NR</b>	<b>2.91×10<sup>1</sup></b>	<b>NR</b>	<b>3.04×10<sup>1</sup></b>	<b>NR</b>	<b>NR</b>	<b>2.31×10<sup>1</sup></b>	<b>NR</b>	<b>NR</b>

<sup>a</sup> As part of a reevaluation of the inventories within the Waste Management alternatives, the Idaho National Laboratory RH-LLW resins waste, with the following inventory, was not modeled in the groundwater analysis in this environmental impact statement: arsenic=2.99 kilograms; cadmium=1.95×10<sup>-2</sup> kilograms; chromium=1.33×10<sup>1</sup> kilograms; silver=4.10×10<sup>-2</sup> kilograms. Such an action by DOE could be the result of a number of changes at the Hanford site, such as revisions to the IDF waste acceptance criteria.

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Key:** DOE=U.S. Department of Energy; LLW=low-level radioactive waste; MLLW=mixed low-level radioactive waste; NR=not reported; PCB=polychlorinated biphenyl; RH=remote-handled; WVDP=West Valley Demonstration Project.

**Source:** SAIC 2011.

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## APPENDIX E

### DESCRIPTIONS OF FACILITIES, OPERATIONS, AND TECHNOLOGIES

Appendix E provides additional information about the technologies, processes, and facilities for the three key activities of this *Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington*: tank closure, Fast Flux Test Facility decommissioning, and waste management. Section E.1 includes this information for tank closure; Section E.2, for Fast Flux Test Facility decommissioning; and Section E.3, for waste management.

#### E.1 TANK CLOSURE

This section provides additional information about the technologies, processes, and facilities that would be used under the proposed alternatives for closure of the Hanford Site (Hanford) single-shell tank (SST) system, as described in Chapter 2 of this *Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington (TC & WM EIS)*. Information provided in this appendix forms the basis for determining the environmental impacts of each alternative, as described in Chapter 4.

Each alternative relies on a combination of technologies, processes, and facilities that could accomplish the desired outcome for that alternative. Distribution of the radioactive and chemical constituents of the tank waste among the various waste form storage and management options depends on which technologies and processes the alternative includes. Appendix D provides information on the basis for the chemical and radionuclide composition in the tanks, as well as equipment, soils, and waste forms. These data, along with information concerning which technologies and processes would be used under a particular alternative, formed the basis for modeling transport of contaminants in air, water, and soil media.

Sections E.1.1 and E.1.2 provide detailed descriptions of the technologies, processes, and facilities utilized in one or more of the alternatives described in Chapter 2. A matrix showing the technologies and processes that were assumed to be implemented for each alternative is provided. In many cases, those technologies were selected to provide bounding environmental consequences and do not necessarily represent the exact technologies or processes that could be implemented to achieve the desired outcome. Furthermore, this *TC & WM EIS* does not attempt to analyze all possible permutations of the alternatives using available technologies and processes, but instead attempts to group activities logically into reasonable alternatives for analysis. The technologies, processes, and facilities analyzed in detail in this environmental impact statement (EIS) have sufficient performance data to make conservative assumptions regarding construction, operations, and decommissioning impacts. However, comprehensive and specific engineering designs may still need to be developed once a series of technologies is selected for implementation.

Section E.1.3 discusses technology options that were initially considered, but were not analyzed in detail, as well as the rationale for selecting the technologies that were analyzed. The former are technologies that, due to their lack of maturity, cannot be analyzed in detail at this time using reasonable and conservative engineering estimates regarding construction, operations, and decommissioning impacts. Should continued research and development (R&D) indicate additional benefits over the technologies analyzed in detail, these maturing technologies can then be analyzed in further detail and incorporated into the tank closure program.

##### E.1.1 Current River Protection Project

This section describes the current River Protection Project's (RPP's) activities for storage, retrieval, treatment, and disposal of waste in the Hanford tanks. The current program is based primarily on implementing Phase I of the Preferred Alternative as identified in the *Tank Waste Remediation System*,

*Hanford Site, Richland, Washington, Final Environmental Impact Statement (TWRS EIS)* (DOE and Ecology 1996). Phase I was initiated to treat approximately 10 percent by volume of the tank waste, which contains 25 percent of the total radioactivity in the waste, by the year 2018. Without providing additional waste treatment capacity, it would be possible to continue waste treatment at the same rate as in Phase I until the major facilities reach the end of their design lives. However, such an action would not achieve the objective of treating all of the waste; thus, it was not considered a reasonable alternative for this *TC & WM EIS*. The alternatives evaluated in this *TC & WM EIS* (except the No Action Alternative) involve various activities in addition to the current program to complete treatment of the tank waste and provide for final disposition of the SSTs. Those activities are described in Section E.1.2.

The current RPP activities can be divided into three main areas: (1) routine tank farm operations (operations and maintenance of the tank farm system), (2) tank system upgrades, and (3) planned Waste Treatment Plant (WTP) operations (WTP construction and operations).

### **E.1.1.1 Routine Tank Farm Operations**

Routine tank farm operations include waste retrieval and transfer operations, evaporation, SST system closure activities, double-shell tank (DST) integrity assessments, and tank farm system maintenance and life extension activities. Also included in routine tank system operations are the ongoing monitoring activities necessary to ensure compliance with nuclear safety Authorization Basis (AB)<sup>1</sup> requirements and environmental, occupational safety and health, and other applicable regulatory requirements. Administrative and technical support required to accomplish this work is also included.

#### **E.1.1.1.1 Tank Farm Facilities**

The primary components of the tank farm system are the 177 SSTs and DSTs located in the 200-East and 200-West Areas of Hanford. The tanks are grouped in tank farms with common support equipment such as primary tank ventilation systems; shielded concrete pits for pumps, valves, and jumpers (a short length of pipe or electrical cable used to make a connection, usually temporary, between equipment, tanks, or valves); buildings for monitoring and control instrumentation; and change houses for operators, maintenance personnel, and radiological control technicians.

##### **E.1.1.1.1.1 Single-Shell Tanks**

The SSTs were built from 1943 to 1964 to hold the liquid radioactive waste created by the production and separation of plutonium. The number and capacities of the SSTs are as follows:

- 16 of 208,000-liter (55,000-gallon) capacity
- 60 of 2.0-million-liter (530,000-gallon) capacity
- 48 of 2.9-million-liter (758,000-gallon) capacity
- 25 of 3.8-million-liter (1-million-gallon) capacity

The total nominal holding capacity of the SSTs is approximately 356 million liters (94 million gallons) (DOE 2003a). The tanks currently contain approximately 122 million liters (32.1 million gallons) of radioactive and hazardous waste (DOE 2003b). These tanks contain salt cake and sludge; most of their free liquids were evaporated or transferred to the newer DSTs to reduce the chance of leaks. Table E-1 shows the distribution of the SSTs among the tank farms.

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<sup>1</sup> The nuclear safety AB, often referred to as the “Authorization Basis,” or “AB,” consists of a set of operating limits that define the envelope of safe operations for U.S. Department of Energy nuclear facilities. The AB is established through rigorous analysis of possible accident scenarios and impacts on the public, workers, and facilities. Changes to the design or operation of the facility must be evaluated against the AB to ensure that safe conditions are maintained as a result of the change.

**Table E–1. Distribution of Single-Shell Tanks Among Tank Farms**

Tank Farm	Number of Tanks	Tank Capacity (liters)	Location	Years Constructed
A	6	$3.8 \times 10^6$	200-East Area	1954–55
AX	4	$3.8 \times 10^6$		1963–64
B	12	$2.0 \times 10^6$		1943–44
	4	$2.1 \times 10^5$		1943–44
BX	12	$2.0 \times 10^6$		1946–47
BY	12	$2.9 \times 10^6$		1948–49
C	12	$2.0 \times 10^6$		1943–44
	4	$2.1 \times 10^5$		1943–44
S	12	$2.9 \times 10^6$	200-West Area	1950–51
SX	15	$3.8 \times 10^6$		1953–54
T	12	$2.0 \times 10^6$		1943–44
	4	$2.1 \times 10^5$		1943–44
TX	18	$2.9 \times 10^6$		1947–48
TY	6	$2.9 \times 10^6$		1951–52
U	12	$2.0 \times 10^6$		1943–44
	4	$2.1 \times 10^5$		1943–44

**Note:** To convert liters to gallons, multiply by 0.26417.

#### E.1.1.1.1.2 Double-Shell Tanks

The DSTs were built from 1968 to 1986 to support fuel reprocessing, isotope recovery, and tank waste management operations. The numbers and nominal capacities of the DSTs are as follows:

- 4 of approximately 3.8-million-liter (1-million-gallon) capacity
- 24 of approximately 4.4-million-liter (1.16-million-gallon) capacity

The DSTs have a total nominal holding capacity of approximately 121 million liters (32 million gallons) (DOE 2003a). They contain approximately 85 million liters (22.5 million gallons) of radioactive and hazardous waste (DOE 2003b). Generally, the tanks contain liquids and settled salts. Some tanks also contain a bottom layer of sludge. Distribution of the DSTs among the tank farms is shown in Table E–2.

**Table E–2. Distribution of Double-Shell Tanks Among Tank Farms**

Tank Farm	Number of Tanks	Tank Capacity (liters)	Location	Years Constructed
AY	2	$3.8 \times 10^6$	200-East Area	1968–70
AZ	2	$3.8 \times 10^6$		1971–77
AW	6	$4.4 \times 10^6$		1978–80
AN	7	$4.4 \times 10^6$		1980–82
AP	8	$4.4 \times 10^6$		1983–86
SY	3	$4.4 \times 10^6$	200-West Area	1973–76

**Note:** To convert liters to gallons, multiply by 0.26417.

#### **E.1.1.1.2 Operations and Maintenance**

Waste retrieval and transfer operations are conducted to support staging and preparation of waste for eventual delivery to the WTP. Activities include planning the sequence for transferring waste currently stored in the DSTs to the WTP and retrieving and transferring waste from the SSTs to the DST system for eventual treatment. Transfer operations involve determining the compatibility of the wastes to be transferred, establishing pipeline transfer routes, monitoring the waste volumes in the sending and the receiving tanks, flushing the lines when the transfers are complete, and documenting the entire process.

DST integrity assessments involve nondestructive examination (NDE) of the DST system to assess fitness for storing waste. Chemical adjustments of the tank contents are conducted as required to meet corrosion control specifications. Periodic assessment reports are required under the Hanford Federal Facility Agreement and Consent Order (also known as the Tri-Party Agreement [TPA]) (Ecology, EPA, and DOE 1989) Milestone Series M-48.

Tank farm maintenance and life extension activities involve any physical maintenance required to ensure continued operation of the tanks and associated systems as designed. Upgrades to extend the useful life of various tank farm systems and facilities have been identified. One important facility is the 242-A Evaporator, which would be needed for volume reduction of the waste to be stored in the DSTs. The 242-A Evaporator is described in Section E.1.2.3.2. DST space is limited, and volume reduction by evaporating excess water has been used extensively at Hanford to avoid the need for new storage tanks. Upgrades to the 222-S Analytical Laboratory also are needed to characterize waste for safe storage and treatment but are not analyzed in this EIS.

##### **E.1.1.1.2.1 Waste Retrieval and Transfer Operations**

The primary objectives of waste retrieval and transfer operations are to retrieve waste from the SSTs and deliver waste feed to the WTP for treatment. Current tank space within the DSTs is limited, and some space must be held in reserve in the event that one of the tanks leaks, which would require immediate transfer of its contents to other tanks that are sound. The following activities support waste transfer and tank space operations:

- Develop and maintain operating procedures for waste retrieval and transfer operations.
- Perform waste compatibility analyses to ensure that waste meets applicable requirements.
- Establish routes (pipelines) for receipt of waste from the SSTs and transfer of waste within the DST system, including transfers to double-contained receiver tanks, cross-site transfers, transfers to support evaporator operations, and receipt of waste from other Hanford facilities. Typical activities include changing jumpers in pits and setting valves in preparation to transfer or receive waste.
- Fabricate and maintain piping jumpers necessary to route waste as needed.
- Perform operations as necessary to transfer waste, monitor systems during the transfers, and perform post-transfer line flushes.

At the end of each transfer, a material balance calculation based on the liquid-level reduction in the sending tank and liquid-level rise in the receiving tank is performed to verify that all of the waste has been transferred. This material balance, combined with analytical measurements of the transferred waste, allows a determination of radionuclide/chemical-specific retrieval efficiencies. The results can be evaluated to determine any deviations from the uniform (i.e., nonselective) retrieval efficiencies assumed for analysis purposes in this *TC & WM EIS*.

#### **E.1.1.1.2.2 Maintenance**

Maintenance consists primarily of preventive and corrective maintenance activities that are needed to provide operable and functional equipment to support system operations. This includes maintenance of all SST and DST system components, as well as maintenance of the waste feed delivery system being constructed to supply waste feed to the WTP. Maintenance also includes the administrative and technical support required to perform maintenance work.

A safe and effective maintenance program is necessary to support tank farm operations designed for safe storage of waste and retrieval of waste to be processed by the WTP. Preventive and corrective maintenance measures are necessary to ensure operable equipment is calibrated, functionally tested, and thus compliant with AB and regulatory requirements. Preventive and corrective maintenance activities include the following:

- Calibration of all process instrumentation, such as liquid-level detectors, leak detection probes, flowmeters, and temperature indicators
- Maintenance of tank-related equipment, such as pumps, valves, and jumpers
- Maintenance of tank farm ventilation equipment, such as high-efficiency particulate air (HEPA) filters, pressure gauges, fans, and portable exhausters

Maintenance involves repair or removal and replacement of parts and equipment. As with any industrial process, this entails technical and administrative support for design, procurement, installation, and operability testing. Because the process involves managing both hazardous and radioactive waste, these functions have additional safety and regulatory considerations that must be taken into account.

#### **E.1.1.1.2.3 Contaminated Equipment Removal**

As a result of routine tank farm operations and maintenance activities, equipment becomes worn out, broken, or otherwise unfit for its intended use. The various tank farms contain equipment that is no longer required and has been contaminated with hazardous waste, radioactive materials, or both. Contaminated equipment is disposed of according to appropriate Hanford procedures. Contaminated equipment characterized as dangerous or extremely hazardous is removed and may require treatment and disposal (WAC 173-303). Radioactive waste must meet the U.S. Department of Energy's (DOE's) Atomic Energy Act (42 U.S.C. 2011 et seq.) radioactive waste management criteria via the *Hanford Site Solid Waste Acceptance Criteria* (HSSWAC) (Fluor Hanford 2005a) prior to onsite disposal. The structural equipment is characterized, inventoried, and disposed of as either reusable equipment or waste. Aboveground, out-of-service contaminated equipment that is not reusable and is inaccessible or difficult to remove (based on the risk and cost involved) is sealed to prevent contamination spread.

#### **E.1.1.1.2.4 Contamination Zone Reduction**

This activity provides a systematic method for implementing a graded approach to characterizing, cleaning up, and reposting tank farm outdoor areas that are currently radioactive contamination or buffer areas. The graded approach prioritizes areas in which to eliminate habitat, reduce access to contaminated components, perform necessary surveys, and clean up areas to release. Localized high-contamination areas are immediately stabilized with a fixative.

Zone reduction of a tank farm involves all areas within the fence perimeter, extending to approximately 15 meters (50 feet) outside the fence perimeter. Decontamination of structures is not planned. Above-grade components, buildings, and structures are surveyed only on the exterior surfaces and

verified as sealed or isolated to prevent spread of radioactive contamination. Examples of items typically removed include the following:

- Miscellaneous equipment, jumpers, and tools
- Low-level radioactive waste (LLW), lead shielding plates, and shielded sample holders
- Exhausters, exhauster parts, and diesel generators
- Old change trailers, industrial vacuums, and telescopic lighting towers

#### **E.1.1.1.2.5 Routine Surveillance and Monitoring**

Routine surveillance and monitoring activities are conducted to ensure compliance with AB requirements and environmental, occupational safety and health, and other applicable regulatory requirements. Surveillance and monitoring conducted for all tank farms and facilities include daily, weekly, monthly, quarterly, and semiannual surveillances.

Routine radiological and chemical control surveys, including vapor monitoring, are conducted to characterize workplace conditions, identify areas requiring changes in radiological posting, control sources of radiological exposure, and verify control of radioactive contamination.

Surveillance and monitoring are conducted to verify compliance with Resource Conservation and Recovery Act (RCRA) requirements; leak detection, monitoring, and mitigation requirements; Clean Air Act requirements for air emissions control and monitoring; and DOE's Atomic Energy Act requirements.

The tank monitoring and control system remotely monitors field equipment and automatically collects surveillance data to ensure AB controls and environmental requirements are met. Actions required to operate and maintain the tank monitoring and control system include performing routine system maintenance, resolving hardware and software issues, performing hardware and software upgrades, performing data backups and data management, and maintaining system operating procedures and specifications.

The Surveillance Analysis Computer System is a data management system for processing and reviewing data collected from the waste tanks regarding interstitial liquid levels, surface levels, and temperatures. Operation and maintenance of the Surveillance Analysis Computer System involves the following activities: performing routine system maintenance, resolving hardware and software issues, performing hardware and software upgrades, performing data backups and data management, and maintaining system operating procedures and specifications. In addition, Surveillance Analysis Computer System operation involves data quality verification and validation, as well as operation and maintenance of a personal computing Surveillance Analysis Computer System, which enables Hanford users to access the database and generate graphical reports.

An important element of tank farm surveillance and monitoring is the system engineer's function of data surveillance and evaluation. It is within this function that data on tank liquid levels (surface and interstitial) are analyzed to establish a baseline for each tank, as well as alarm limits that indicate a significant change in liquid-level readings. Anomalies in tank waste levels and interstitial liquid levels are investigated and reported to operations, and potential tank leaks are identified.

In addition to the activities mentioned above, staff coverage is required to respond to plant conditions that can affect compliance with the AB safety envelope and environmental requirements. This element covers continuity of operations, unusual or off-normal event monitoring and mitigation, responses to alarms, and emergency responses and participation in emergency drills (DOE 2003c).



#### E.1.1.1.2.6 Safety

Applicable requirements from the AB and controls derived from analysis of facility-specific work scope, operating conditions, hazards, and impacts are implemented to ensure the facility operates within the established safety envelope. Implementation of the safety basis for safe storage and management of legacy and new waste, operation and maintenance of tank farm facilities, waste retrieval and disposal, and closure of the tank farm facilities is required. The Safety Basis Implementation Improvement Project is a discrete set of activities designed to raise the level of rigor associated with safety basis implementation.

Activities for AB compliance verification include the following:

- Performing AB compliance assessments and documenting results
- Addressing questions on current AB implementation
- Tracking, trending, and monitoring technical safety requirement compliance and corrective action management support
- Developing and maintaining procedures for AB implementation
- Providing an interface for AB implementation with the tank farm contractor and outside organizations

Activities for ensuring implementation of changes to the AB include the following:

- Identifying items affected by an AB change (e.g., technical safety requirements, procedures, equipment, training, operations, surveillance and maintenance [S&M], and administrative controls)
- Verifying adequate implementation of the AB changes

#### E.1.1.2 Tank System Upgrades

Tank farm upgrade and construction projects are under way to provide systems necessary for the retrieval and transfer of waste to the WTP and for the storage or disposal of waste produced by the treatment process. Included in the construction projects is the administrative and technical support required to conduct these activities. The following list provides a description of ongoing or recently completed tank farm upgrade and construction projects (DOE 2003c):

- ***Project W-211, Initial Tank Retrieval System.*** Provides new pumps (mixer, decant, slurry, and supernatant pumps), electrical system upgrades, instrumentation and control system upgrades, chemical addition system, and valving upgrades.
- ***Project W-464, Immobilized High-Level Waste Interim Storage Facility.*** Provides storage capacity for the first 880 canisters of immobilized high-level radioactive waste (IHLW) through a retrofit of the Canister Storage Building (CSB).
- ***Project W-520, Integrated Disposal Facility.*** Provides a disposal facility for the permanent, environmentally safe disposal of mixed low-level radioactive waste (MLLW) and immobilized low-activity waste (ILAW).
- ***Project W-521, Waste Feed Delivery Systems.*** Provides retrieval, transfer, and feed delivery systems to connect tank farms to the WTP.

- **Project W-522, Seven DST Retrieval Systems.** Provides retrieval systems for seven DSTs to enable retrieval of DST waste and delivery of such waste to the WTP.
- **Project E-525, DST Transfer System Modifications.** Designs and constructs modifications to DST farm structures, systems, and components not addressed by other projects to ensure that they comply with regulatory, safety, and contractual requirements.

### **E.1.1.3 Planned Waste Treatment Plant Operations**

A major aspect of the RPP's current program is treatment of waste in the WTP. The WTP, currently under construction, would be used to pretreat and immobilize (by vitrification to borosilicate glass) waste currently stored in the SSTs and DSTs. The waste that would be treated in the WTP is radioactive and dangerous waste, as defined by WAC 173-303, "Dangerous Waste Regulations."

The WTP, as designed, consists of waste treatment facilities, an Analytical Laboratory, and support facilities commonly referred to as the "balance of facilities" (BOF). The WTP is currently under construction in the 200-East Area of Hanford. The three waste treatment facilities are the Pretreatment Facility, the Low-Activity Waste (LAW) Vitrification Facility, and the High-Level Radioactive Waste (HLW) Vitrification Facility.

Waste would be transferred to the WTP by pipeline for pretreatment and vitrification. The pretreatment process is designed to separate the solid and liquid fractions of the waste into HLW and LAW streams. The solids would be washed and/or leached to remove chemical constituents and thereby reduce the required volume of IHLW glass. The liquid fraction forms the LAW stream once selected radionuclides have been removed. The separated radionuclides would be combined with the pretreated solids to form the HLW stream.

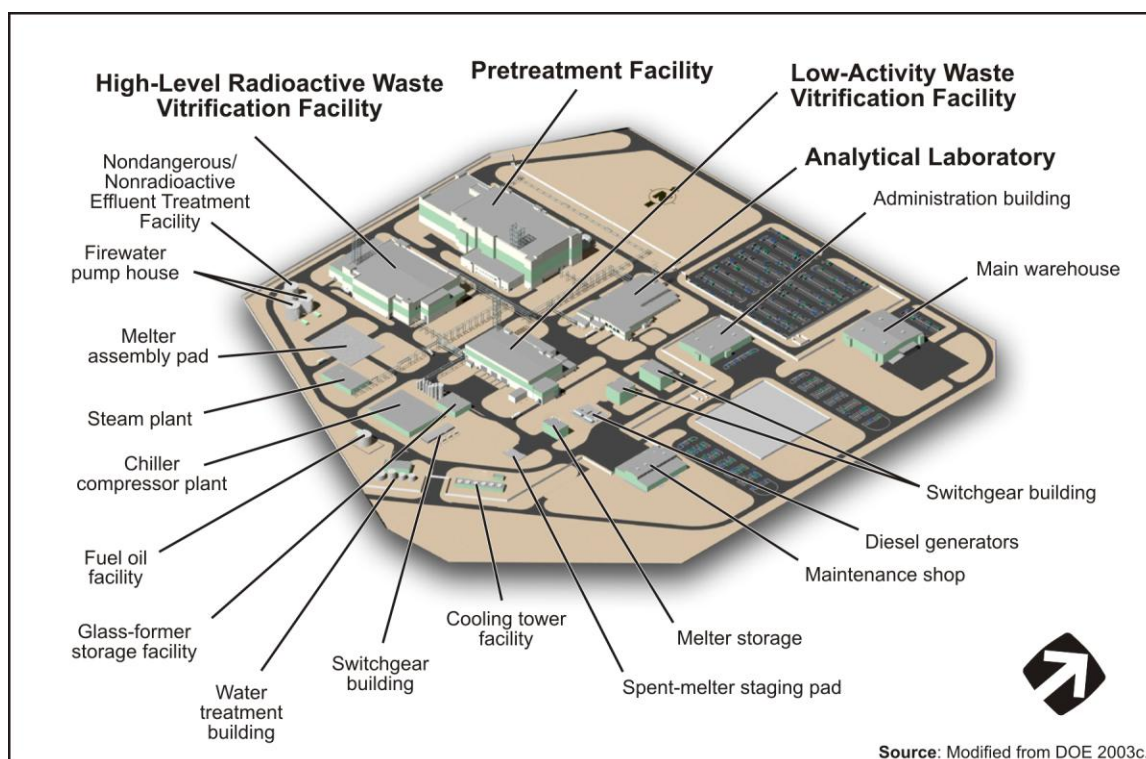
In the HLW and LAW vitrification processes, the HLW and LAW streams would be combined with glass-forming materials and melted to form a liquid about the consistency of molasses that can be poured into stainless steel containers. After the glass cools and solidifies, the containers would be sealed and decontaminated in preparation for storage or permanent disposal. The dangerous waste constituents would be removed or destroyed during the melting process or, along with the radioactive constituents, they would be immobilized in the durable glass matrix. The WTP is designed to accommodate a theoretical maximum capacity (TMC) of 6 metric tons of glass IHLW per day and 30 metric tons of glass ILAW per day.

The WTP processes would generate secondary waste, including offgas, liquid effluents, and miscellaneous solid wastes. The offgas streams would be treated to a level that protects human health and the environment in accordance with air emission regulations and permits. Liquid effluents would be directed to the Liquid Waste Processing Facilities (LWPFs) for treatment and/or onsite disposal in permitted facilities. The LWPFs are discussed in Section E.1.2.3.3. Miscellaneous solid wastes such as used equipment, laboratory waste, and other chemically and radioactively contaminated materials would be transferred to permitted disposal facilities on site. Certain wastes would be designated transuranic (TRU) waste and packaged for disposal at the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico. Nonradioactive dangerous waste would be transferred to a permitted offsite disposal facility.

### E.1.1.3.1 Waste Treatment Plant Facilities

The WTP, currently under construction, would be a newly constructed facility on Hanford property owned by DOE. The site lies at the eastern end of the 200-East Area of Hanford. Figure E–1 shows the WTP site plot plan with all of the major facilities identified. The following summarizes characteristics of the main WTP facilities, with details provided later in Section E.1.2.3.1.

**Pretreatment.** The Pretreatment Facility would be a five-story concrete and steel building that houses the necessary tanks and process equipment for separating the waste into high- and low-activity fractions. Most of the equipment would be inside process cells with thick concrete walls for radiation shielding. The facility would include a set of four large tanks (1,438,300 liters [380,000 gallons] each) for receipt of waste feed from the DSTs.



**Figure E–1. Waste Treatment Plant Facilities**

**LAW Vitrification.** The LAW Vitrification Facility would house feed preparation tanks, two large melters, and equipment for sealing and decontaminating containers filled with ILAW glass. The feed preparation tanks would be housed in a shielded cell, while the melters would be equipped with 2.54-centimeter-thick (1-inch-thick) steel panels for radiation shielding.

**HLW Vitrification.** The HLW Vitrification Facility would house feed preparation tanks and two separate HLW melter cells. Shared cells would be included for canister welding, decontamination, and lag storage of filled canisters. All process areas would be within shielded cells for radiation protection. Process and maintenance operations would be carried out with remote equipment.

**Analytical Laboratory.** The Analytical Laboratory would support the three treatment facilities described above. It is equipped with hot cells for handling highly radioactive samples and is capable of characterizing samples of tank waste (DOE 2003c). The WTP BOF would include systems, utilities, and infrastructure required to support plant operations.

## **E.1.2 Descriptions of Tank Closure Alternative Facilities and Operations**

Each of the alternatives evaluated in this *TC & WM EIS* comprises a set of technologies or activities that would enable specific outcomes under that alternative. Some technologies and activities are common to more than one alternative, while others have commonality but differ in the extent of implementation. A brief description of each Tank Closure alternative is included in the Reader's Guide. Also included in the Reader's Guide is a simplified process flow diagram of the major facilities and operations analyzed in this *TC & WM EIS*. Table E-3 provides a comparison of key parameters for each of the Tank Closure alternatives.

Each of the technologies and activities is described in greater detail in the following sections. The descriptions are intended to delineate the technical basis for the alternatives in this *TC & WM EIS* and to provide supporting information regarding the basis for analysis of impacts. Much of the information presented is drawn from more-detailed data packages that are available in the administrative record for this *TC & WM EIS*.

### **E.1.2.1 Facility Upgrades**

Storage upgrades required to support planned tank farm and WTP operations include tank farm infrastructure upgrades, upgrades to the CSB for interim storage of IHLW glass, and construction of additional IHLW Interim Storage Modules.

#### **E.1.2.1.1 Tank Space Upgrades and Replacement Double-Shell Tanks**

This section discusses options for storage of waste retrieved from SSTs and miscellaneous underground storage tanks (MUSTs). Additional waste storage is included in the *TC & WM EIS* alternatives where sufficient DST space would not be available within the existing tank farms or where treatment extends beyond the useful life of the existing DSTs and replacements are required.

##### **E.1.2.1.1.1 Waste Volumes and Available Capacity**

This section presents an analysis of current and future DST capacities and waste volumes. The quantities of waste and available DST space were taken from current sources and are accurate to the limitations cited in the referenced documentation.

Active waste storage currently takes place in the existing DST system. Storage is provided by 28 DSTs with an approximate capacity of 4.3 million liters (1.14 million gallons) each. Of these tanks, 25 are located in the 200-East Area and 3 in the 200-West Area. None of the DSTs has leaked to date.

Waste is also stored in 149 older, inactive SSTs. Of these 149 SSTs, 67 are known or suspected to have leaked. There are also 66 smaller underground MUSTs associated with the SST and DST systems, 61 of which are under the jurisdiction of the RPP. They vary in size from approximately 3,000 to 190,000 liters (800 to 50,000 gallons), contain sludge and process materials similar to the waste in the SSTs and DSTs, and have a total waste volume that is less than 1 percent of the SST waste. Of these 61 MUSTs, 13 are part of the DST system and therefore are not included within the scope of the closure action analyzed in this EIS (DOE 2003d; Hebdon 2001).

**Table E-3. Comparison of the Tank Closure Alternatives**

	Alternative 1:	Alternative 2A:	Alternative 2B:	Alternative 3A:	Alternative 3B:	Alternative 3C:	Alternative 4:	Alternative 5:	Alternative 6A:	Alternative 6B:	Alternative 6C:
	No Action	Existing WTP Vitrification; No Closure	Expanded WTP Vitrification; Landfill Closure	Existing WTP Vitrification with Thermal Supplemental Treatment (Bulk Vitrification); Landfill Closure	Existing WTP Vitrification with Nonthermal Supplemental Treatment (Cast Stone); Landfill Closure	Existing WTP Vitrification with Thermal Supplemental Treatment (Steam Reforming); Landfill Closure	Existing WTP Vitrification with Supplemental Treatment Technologies; Selective Clean Closure/Landfill Closure	Expanded WTP Vitrification with Supplemental Treatment Technologies; Landfill Closure	All Vitrification/No Separations; Clean Closure	All Vitrification with Separations; Clean Closure	All Vitrification with Separations; Landfill Closure
<b>Storage</b>											
Existing	✓										
New WRFs			✓	✓	✓	✓	✓	✓		✓	✓
New DSTs		✓						✓	✓		
<b>Retrieval</b>											
90 percent								✓			
99 percent		✓	✓	✓	✓	✓					✓
99.9 percent							✓		✓	✓	
<b>Treatment</b>											
<b>WTP</b>											
Existing vitrification only		✓		✓	✓	✓	✓				✓
Expanded LAW vitrification			✓					✓		✓	✓
Expanded HLW vitrification									✓		
Replacement of WTP		✓							✓		
Technetium-99 removal			✓		✓						
Sulfate removal								✓			
Cesium and strontium capsules		✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
<b>Non-WTP</b>											
Tank mixed TRU waste supplemental treatment				✓	✓	✓	✓	✓			
Thermal supplemental treatment				✓		✓	✓	✓			
Nonthermal supplemental treatment					✓		✓	✓			
<b>Disposal (including post-treatment storage)</b>											
<b>On Site</b>											
ILAW		✓	✓	✓	✓	✓	✓	✓		(a)	(a)
IHLW <sup>b</sup>		✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
Sulfate grout								✓			
Contaminated soil			✓	✓	✓	✓	✓		✓	✓	✓
SSTs							(c)		(d)	(d)	
<b>Off Site</b>											
Tank mixed TRU waste to WIPP				✓	✓	✓	✓	✓			
<b>Closure</b>											
Clean closure									✓	✓	
Selective clean closure/landfill closure							✓				
Landfill closure			✓	✓	✓	✓	✓	✓			✓
Modified RCRA Subtitle C barrier			✓	✓	✓	✓	✓		(e)	(e)	✓
Hanford barrier								✓			

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<sup>a</sup> Under Alternatives 6B and 6C, ILAW glass would be interim-stored on site and managed as IHLW glass.

<sup>b</sup> Although disposition decisions have not been made and implemented, these alternatives do not assume the inventory in the IHLW canisters remains on site. However, the number of storage facilities needed to store all the IHLW is one more than the number of canister storage facilities analyzed under Tank Closure Alternative 2B.

<sup>c</sup> Under Alternative 4, SSTs at the BX and SX tank farms would be removed and treated in the Preprocessing Facility.

<sup>d</sup> Under Alternatives 6A and 6B, all SSTs would be removed and packaged in shielded boxes for onsite storage pending disposition.

<sup>e</sup> Base Case: Modified RCRA Subtitle C barrier over six sets of cribs and trenches (ditches) in the B and T Areas. Option Case: Remove six sets of cribs and trenches (ditches) in the B and T Areas and remediate their deep-soil plumes.

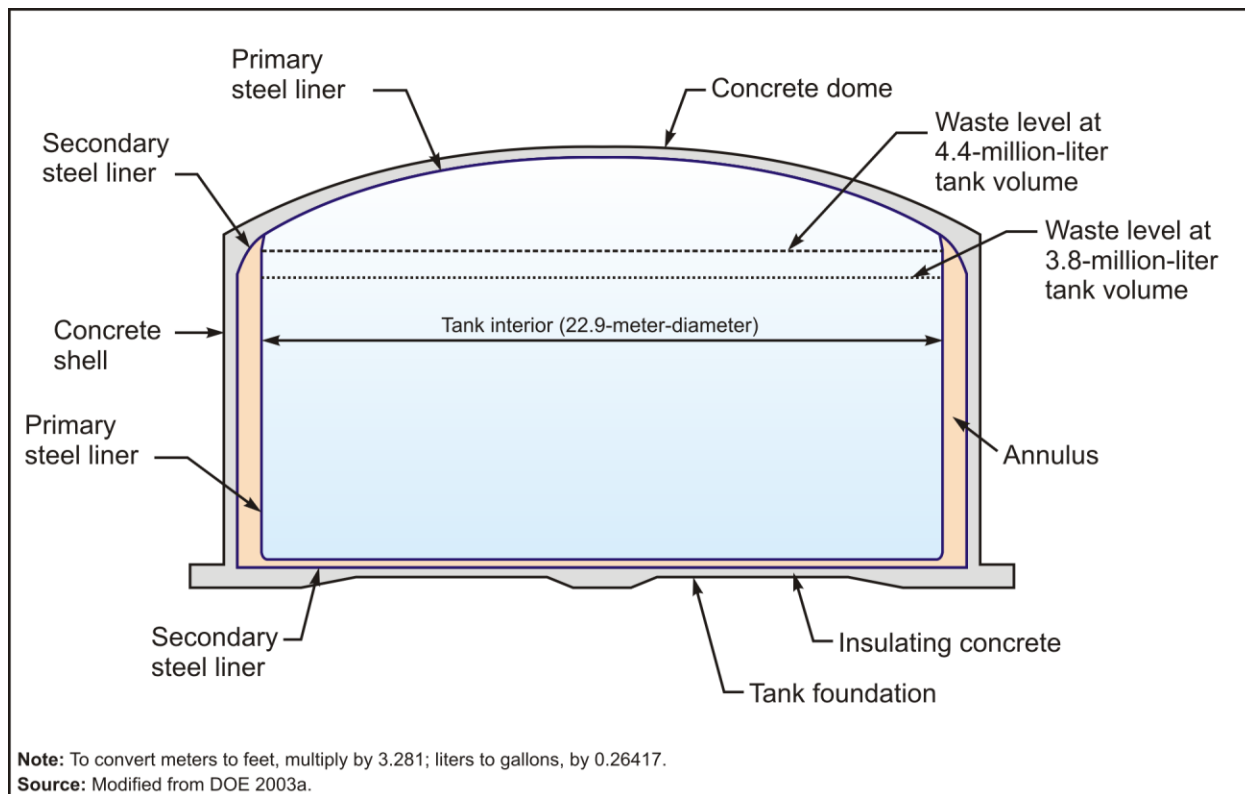
**Key:** DST=double-shell tank; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; RCRA=Resource Conservation and Recovery Act; SST=single-shell tank; TRU=transuranic; WIPP=Waste Isolation Pilot Plant; WRF=waste receiver facility; WTP=Waste Treatment Plant.

Currently, the availability of DST tank space limits retrieval from SSTs and MUSTs. SSTs and MUSTs cannot be used for long-term storage of retrieved waste because of their age, suspect integrity, and lack of secondary-containment capability. The total available tank volume in the DSTs is 117 million liters (31 million gallons). Deducting the tank volume set aside for administrative controls (emergency reserve, WTP feed staging, and restricted tank space), the effective capacity of the DST system is approximately 102 million liters (27 million gallons). The DSTs currently contain approximately 85 million liters (22.5 million gallons) of waste, leaving an available capacity of about 17 million liters (4.5 million gallons). Additional capacity of 15 million liters (4 million gallons) could be achieved by evaporating waste to reduce volume and raising tank capacity administrative control limits. Even with this additional available capacity, all of the alternatives, except Tank Closure Alternative 1, would require additional storage capacity to support retrieval and WTP operations. Waste receiver facilities (WRFs), as discussed in Section E.1.2.2.8, would be provided under Tank Closure Alternatives 2B through 5 and 6B and 6C. Storage facility upgrades entailing new or replacement DSTs would be provided under Alternatives 2A, 5, and 6A.

#### **E.1.2.1.1.2 Storage Requirements**

Under Tank Closure Alternative 5, the construction of four new DSTs would be required to provide sufficient DST storage capacity to match the rate at which waste would need to be removed for treatment to complete all treatment of tank waste by 2033.

An analysis of the additional DST storage capacity necessary under Tank Closure Alternative 5 was performed. However, that analysis did not account for operational upsets such as temporary decreases in treatment throughput or DST failures. Therefore, 15 million liters (4 million gallons) of new storage capacity was selected for analysis of Alternative 5 (Fredenburg 2003). New, below-grade DSTs, similar to existing DSTs (see Figure E-2), would be constructed.



**Figure E-2. Cross-Sectional View of Representative Hanford Site Double-Shell Tank**

Under Tank Closure Alternatives 2A and 6A, replacement of existing DSTs would be required due to useful-life constraints and the potential for integrity failure of the DSTs. Tank Closure Alternative 2A assumes the 28 DSTs would have to be replaced once to extend storage and treatment of tank waste to 2093 because WTP expansion and supplemental technologies would not be used to expedite treatment. Tank Closure Alternative 6A, which treats all tank waste as HLW, requires replacement of the 28 DSTs three times (i.e., construction of 84 new DSTs) to extend storage and treatment of tank waste to 2163.

All other alternatives analyzed in this *TC & WM EIS* provide sufficient treatment capacity by expanding the WTP or supplemental technologies to allow the use of existing DST storage capacities, thereby eliminating the need to construct new or replacement DSTs.

#### **E.1.2.1.2 Interfacing Facilities**

The following facilities would interface with storage, retrieval, and treatment of tank waste:

- **242-A Evaporator.** Services include evaporating the liquid fraction of DST waste. Operation of the existing evaporator is planned to continue until fiscal year 2018. Replacement evaporators would need to be constructed and operated to support the Tank Closure alternatives. The 242-A Evaporator is described in Section E.1.2.3.2.
- **222-S Analytical Laboratory.** Services include providing a dedicated laboratory facility to deliver analytical chemistry services in support of characterization. The laboratory is expected to operate as long as required to support tank waste characterization, tank waste retrieval, and waste feed delivery to the WTP. Upgrades or replacements to the 222-S Analytical Laboratory are not analyzed in this EIS because its use is expected to be limited following the start of operations of the WTP Analytical Laboratory.
- **Effluent Treatment Facility (ETF)/Liquid Effluent Retention Facility (LERF).** Services include processing of liquid effluents designated as radioactive and dangerous wastes. The LERF would need a life extension upgrade in 2015. Operation of the ETF is planned to continue until fiscal year 2025. After the life extension upgrade, the LERF would operate through the end of the WTP service life. Replacement ETFs would need to be constructed and operated to support the Tank Closure alternatives. The LWPFs, including the LERF and ETF, are described in Section E.1.2.3.3.
- **Treated Effluent Disposal Facility (TEDF).** The 200 Area TEDF is permitted for disposal of nonradioactive, nondangerous liquid effluents. The TEDF is described in Section E.1.2.3.3.

#### **E.1.2.1.3 Interim Storage of Immobilized High-Level Radioactive Waste**

Safe interim onsite storage of IHLW glass would be necessary until disposition decisions are made and implemented. The *TWRS EIS* anticipated that the canisters would be stored in the CSB, along with multiccanister overpacks of spent nuclear fuel (SNF) from Hanford's K Basins (DOE and Ecology 1996).

Interim storage capacity, including that of the CSB, would be expanded through construction of a new IHLW Shipping/Transfer Facility and IHLW Interim Storage Modules. These facilities would include load-out capabilities for transferring IHLW glass canisters to overpacks for potential future shipment and new IHLW Interim Storage Modules similar to the CSB that could be individually constructed as additional capacity is required (DOE 2003e).



The IHLW Shipping/Transfer Facility would be constructed concurrently to support IHLW glass canister shipments. Construction of additional storage modules is included under each of the *TC & WM EIS* alternatives to provide storage capacity for IHLW glass produced in the WTP. In the case of Tank Closure Alternatives 6A, 6B, and 6C, all of the waste would be managed as IHLW glass, and appropriate storage facilities are considered for IHLW glass, ILAW glass, and waste from closure of the tank farms.

#### **E.1.2.1.3.1 Assumptions and Uncertainties**

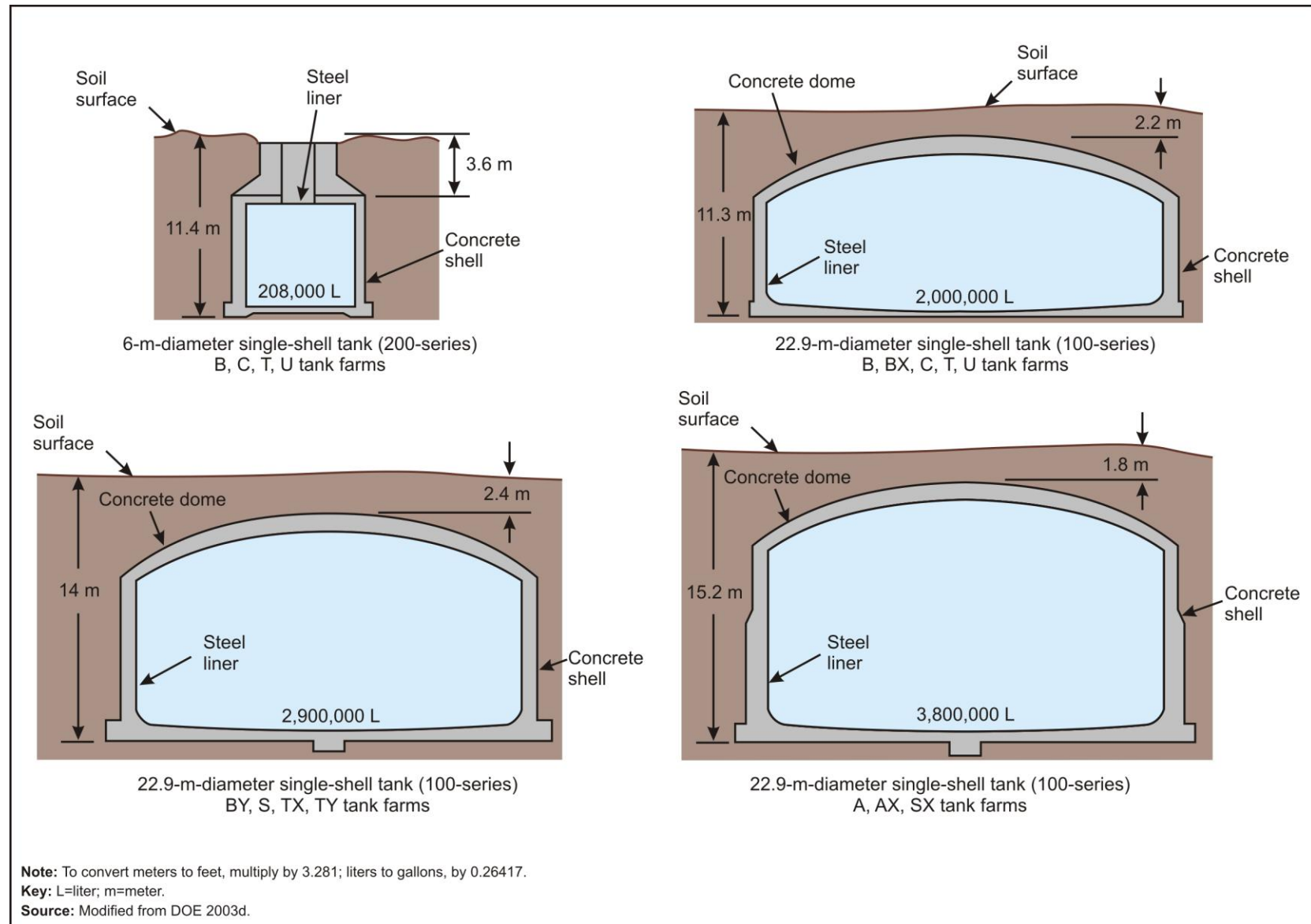
Due to uncertainties regarding the timing for shipment of IHLW glass canisters off site and the capacity for receiving all waste managed as HLW (Tank Closure Alternatives 6A, 6B, and 6C), it was assumed that onsite storage facilities would be required for all IHLW glass.

#### **E.1.2.2 Waste Retrieval and Storage**

The following sections describe the technologies, facilities, assumptions, and uncertainties associated with options for retrieval of waste from SSTs and transfer to DSTs and WRFs for storage and eventual treatment. Depending on the alternative, retrieved waste may be transferred directly to a supplemental treatment facility.

| The SST design details are as follows (see Figure E-3):

- | • B, C, T, and U tank farm 200-series tanks (6-meter-diameter [20-foot-diameter]) with 208,000-liter (55,000-gallon) nominal tank capacity
- | • B, BX, C, T, and U tank farm 100-series tanks (22.9-meter-diameter [75-foot-diameter]) with 2.0-million-liter (530,000-gallon) nominal tank capacity
- | • BY, S, TX, and TY tank farm 100-series tanks (22.9-meter-diameter [75-foot-diameter]) with 2.9-million-liter (758,000-gallon) nominal tank capacity
- | • A, AX, and SX tank farm 100-series tanks (22.9-meter-diameter [75-foot-diameter]) with 3.8-million-liter (1-million-gallon) nominal tank capacity



**Figure E-3. Cross-Sectional Views of Hanford Site Single-Shell Tanks**

Retrieval technologies are evaluated with respect to their ability to achieve certain waste retrieval benchmarks established to allow assessment of the environmental impacts associated with the degree of waste retrieval proposed under the various Tank Closure alternatives. The four waste retrieval benchmarks considered in this *TC & WM EIS* are as follows:

- A **0 percent retrieval** involves no removal of tank waste. The 0 percent retrieval was analyzed for the No Action Alternative.
- A **90 percent retrieval** involves removing tank waste to achieve a residual waste volume equal to 102 cubic meters (3,600 cubic feet) for the 100-series SSTs and 8.5 cubic meters (300 cubic feet) for the 200-series SSTs.
- A **99 percent retrieval** involves removing tank waste to achieve a residual waste volume equal to 10.2 cubic meters (360 cubic feet) for the 100-series SSTs and 0.85 cubic meters (30 cubic feet) for the 200-series SSTs.
- A **99.9 percent retrieval** involves removing tank waste to achieve a residual waste volume equal to 1 cubic meter (36 cubic feet) for the 100-series SSTs and 0.08 cubic meters (3 cubic feet) for the 200-series SSTs. The 99.9 percent retrieval rate was used in cases where tank removal was analyzed to limit worker exposure. It also reflects multiple uses of retrieval technologies.

Zero percent retrieval was analyzed under Tank Closure Alternative 1; 90 percent, Tank Closure Alternative 5; 99 percent, Tank Closure Alternatives 2A, 2B, 3A, 3B, 3C, and 6C; and 99.9 percent, Tank Closure Alternatives 4, 6A, and 6B.

#### **E.1.2.2.1 Modified Sluicing**

Modified sluicing has been identified as an option for retrieval of 99 percent of the waste from DSTs and from SSTs not designated as known or suspected leakers. Modified sluicing is similar to past-practice sluicing, in which nozzles were used to spray liquid into the tanks to dissolve and dislodge the waste, then pump it out for processing. Although modified sluicing uses less water, it would not be used to retrieve waste from SSTs that are known or suspected to have leaked in the past.

##### **E.1.2.2.1.1 Technology Description**

A modified sluicing retrieval system has been used as described below. Actual equipment types, locations, and quantities may vary on a tank-by-tank basis.

##### **E.1.2.2.1.2 Process**

Modified sluicing is the introduction of liquid into the waste at low-to-moderate pressures and volumes. Pressures of 345,000 to 1.38 million pascals (50 to 200 pounds per square inch) and flow rates from 300 to 910 liters (80 to 240 gallons) per minute are typical (DOE 2003a). At lower pressures and flow rates, the retrieval action is primarily dissolution and retrieval of soluble species. At higher pressures and flow rates, the retrieval action is due to both dissolution of soluble species and the breaking apart of solid materials and suspension as a waste slurry. A transfer pump inside the tank pumps the waste to a receiver tank.

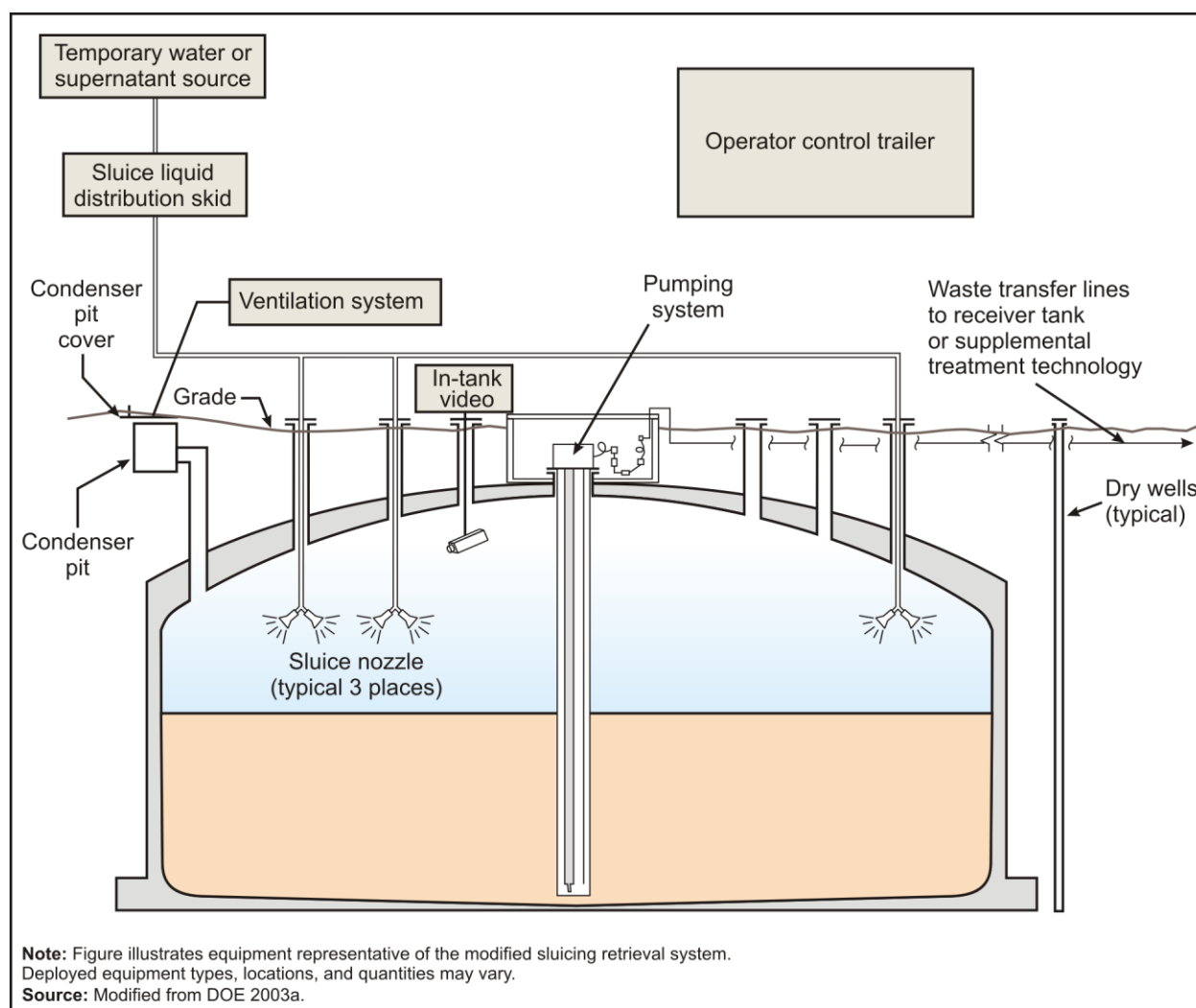
Modified sluicing differs from past-practice sluicing in the following ways:

- Past-practice sluicing introduces sluicing liquid from a single sluice nozzle in bulk fashion by a flooding-type action. Modified sluicing introduces sluice liquid in a controlled fashion using two to three sluicing nozzles and then pumps out the resultant waste slurry at approximately the same rate that the sluice liquid is introduced. This operating strategy maintains a minimal liquid inventory within the tank at all times.
- The use of two to three sluice nozzles in modified sluicing allows a more thorough distribution of sluicing liquid over the tank contents.

The modified sluicing waste retrieval system could be used to retrieve 99 percent of the waste from the DSTs and nonleaking 100-series SSTs. The DSTs would have pumps installed to help dislodge waste solids and thereby aid retrieval.

### E.1.2.2.1.3 Facilities

The equipment used for modified sluicing is shown in Figure E-4. The major equipment items and their functions are discussed below.



**Figure E-4. Modified Sluicing Equipment Schematic**

The sluice nozzles (typically three) discharge sluice liquid in a concentrated stream onto the waste surface. The sluice nozzles are placed through risers around the perimeter of the tank and are remotely manipulated. The nozzles can typically be rotated 360 degrees horizontally and adjusted 90 degrees vertically (between pointing straight down and horizontally). Depending on the waste type, sluicing systems may use water or supernatant from a DST as the retrieval liquid.

The sluice liquid distribution skid provides the structural support, confinement, and leak detection (as required) for the sluice liquid distribution manifold (piping, valves, flowmeters, filters, and various fittings); the instrumentation/power enclosures; and a pump. Sluice liquid is piped to the sluice liquid distribution skid, where it is pumped to the various sluice nozzles.

The ventilation system maintains the tank pressure within specified limits during the retrieval operation. The ventilation system consists of an inlet filter assembly and a portable exhauster skid that includes HEPA filters, air emission monitors/samplers, and associated ductwork. The inlet filter connects to an existing riser on the tank. Collectively, the tank ventilation primary assembly performs the following functions:

- Maintains a negative pressure within the tank to prevent the release of airborne radioactive particulates to the environment
- Controls emissions of the air/gas stream before the effluents are discharged to the environment
- Minimizes the amount of condensation in the tank headspace

The field instrument electrical skid includes receptacles that provide power to the ventilation system and the water distribution skid, as well as a power center that provides single-phase power for transfer line heat tracing and various instrumentation.

In-tank video systems are installed to allow operators to monitor and assess the tank retrieval operations and performance.

The pumping system consists of a pump located in the tank. The system pumps waste slurry from the tank.

The operator control trailer contains the instrumentation, video, and power capability necessary for monitoring and controlling the waste retrieval operation.

Before modified sluicing is initiated, the proper configuration of all equipment is established. All of the necessary portable equipment and instrumentation are delivered to the tank and set up. Jumpers are set and hose-in-hose transfer lines (HIHTLs) are installed (if required) to establish a waste transfer route. A waste transfer pump is installed, sluice nozzles are positioned for delivering the sluice water, and the receiving tank is made ready to receive the waste.

Sluicing is initiated by starting the sluice liquid delivery pump. This introduces the sluice liquid into the tank through the in-tank sluice nozzles. The sluice liquid affects the surface of the waste, dislodging and suspending waste particles, dissolving soluble species, and creating a waste slurry. Liquid is delivered to the SST until the waste slurry level rises and covers the inlet to the waste transfer pump, which is near the floor of the tank. The waste transfer pump is then started. The waste transfer pump delivers the waste slurry to the receiver tank. To mitigate the potential for leakage during waste retrieval, the liquid volumes are minimized.

Sluice liquid is delivered to the SST simultaneously with the removal of waste slurry from the SST until preestablished conditions for the extent of waste retrieval are met. The in-tank video system is used to observe retrieval progress and make adjustments as necessary. The transfer line between the SST and the receiver tank is periodically flushed to prevent potential line blockages.

This process is continued until either the waste retrieval goals are met or the equipment has retrieved waste to the extent of its design and functional capability.

#### **E.1.2.2.1.4 Assumptions and Uncertainties**

Key assumptions as to modified sluicing include the following:

- For analysis purposes, it was assumed that the construction, operations, and deactivation impacts assessed for tank 241-S-112 waste retrieval were representative of the proposed modified sluicing system.
- Temporary facilities associated with modified sluicing would be moved and reused for four tank retrievals.

Key uncertainties with modified sluicing include the following:

- Ability to retrieve hard heels (i.e., dense, highly compacted waste on the tank bottom)
- Ability to remove waste from tank walls and in-tank equipment
- Effectiveness of sluicing regarding both retrieval rates and the ability to meet retrieval goals (e.g., 99 percent)
- Ability to retrieve all types of sludge

Modified sluicing concepts have been used at the Hanford Cold Test Facility (*S-112 Project Proof-of-Principle Test Plan/Procedure for Cold Test Facility* [Enderlin et al. 2002]). Modified sluicing is also similar to the process used to retrieve waste from tank 241-C-106, which was based in large part on past-practice sluicing at Hanford (Carpenter 2002). Based on the results of testing at the Hanford Cold Test Facility and the similarity of modified sluicing to past-practice sluicing, no significant technical implementation issues are foreseen for construction and use of modified sluicing systems.

#### **E.1.2.2.2 Mobile Retrieval System**

The mobile retrieval system (MRS) has been identified as an option for retrieving 99 percent of the waste from the 100-series SSTs designated as known or suspected leakers. The MRS with a chemical wash has been identified as an option for retrieving 99.9 percent of the waste from all DSTs and all 100-series SSTs. The MRS employs vacuum-assisted retrieval to retrieve waste with a minimum amount of water added. This minimizes the potential for leaks during retrieval.

##### **E.1.2.2.2.1 Technology Description**

The following is a description of a typical MRS that would be used. Actual equipment types, locations, and quantities may vary on a tank-by-tank basis.

##### **E.1.2.2.2.2 Process**

The MRS is a vacuum-driven system that uses mostly air and a small amount of water to retrieve waste. The MRS uses two major pieces of equipment: an articulated-mast system (AMS) and an

in-tank vehicle (ITV). The AMS is typically located in the central region of the tank because it requires a large-diameter access riser (30.5 centimeters [12 inches]) that typically does not exist in other locations of the tank. The mast contains a waste vacuum system on an articulated arm that can be extended horizontally and rotated up to 360 degrees to access a circular area in the center of the tank, which measures approximately 9.1 meters (30 feet) in diameter. The arm carries a vacuum-operated unit that sucks up the waste and lifts it from the tank to a skid-mounted vessel on the tank farm surface. The ITV can be moved around the entire tank to physically push the waste, carry a sluice nozzle, and carry a vacuum hose-and-nozzle assembly.

A typical retrieval campaign involves first using the AMS to retrieve waste from the central area of the tank. The ITV can then be used to plow, push, or jet waste from the outer areas of the tank toward the center of the tank. Sluice water can be added to aid this process. The ITV can also be used to remove waste from the tank using its vacuum hose-and-nozzle assembly. At the end of the retrieval campaign, the ITV can be used for rinsing the tank walls and in-tank equipment.

The AMS and ITV systems transfer waste from the tank into a skid-mounted vessel in approximately 3,000-liter (800-gallon) batches. The waste is then pumped from this vessel to the waste receiver tank. Water is added to the waste as it is pumped from the vessel to form a slurry suitable for pipeline transfer. An ultrasonic de-agglomeration system may be used to prevent particulates from clustering and plugging the transfer lines.

For retrieval operations from SSTs known or suspected to have leaks, leak detection and monitoring systems and procedures are used to help ensure that little or no waste is leaked to the tank farm soil.

An MRS was designed, built, and tested in the Cold Test Facility at Hanford (*241-C-104 Waste Retrieval System Preliminary Design Report (Project W-523)* [Carpenter 2002]). The testing showed that the MRS can be used to retrieve 99 percent of waste from leaking 100-series SSTs, and the MRS with a chemical wash can be used to retrieve 99.9 percent of waste from all DSTs and all 100-series SSTs.

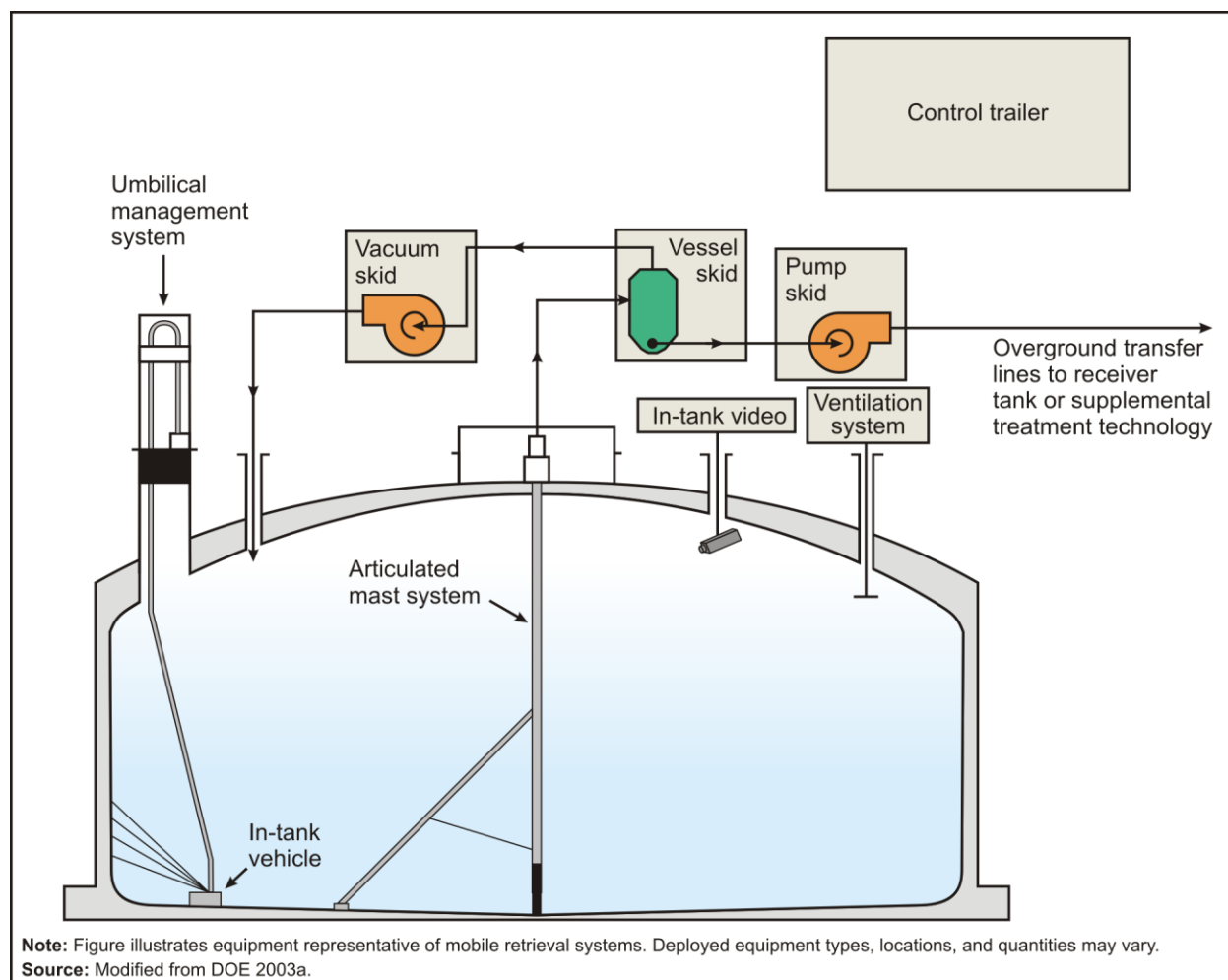
Installation and utilization of an example of an MRS in an SST (241-C-107), the Mobile Arm Retrieval System, took another step in December 2010 when the tank farms contractor cut a 140-centimeter-diameter (55-inch-diameter) hole in the top of the tank. The hole was cut through 38.1 centimeters (15 inches) of concrete and steel rebar to place a new riser for the MRS unit that will be used to remove approximately 935,000 liters (247,000 gallons) of radioactive and chemical waste. The hole is the largest cut into an active DOE radioactive waste storage tank and provides access for the largest robotic arm (the AMS) developed to date for removing wastes from the tanks. Retrieval of waste from tank 241-C-107 is scheduled for the summer of 2011 (Olds 2010).

#### **E.1.2.2.2.3 Facilities**

The MRS consists of two primary pieces of equipment, the AMS and the ITV. Additional above-tank support equipment is required for each. Supporting the AMS are vessel, vacuum, and pump skids and a decontamination unit. Supporting the ITV are an umbilical management system and associated decontamination units. Each component is described in greater detail below (DOE 2003a). Major equipment required for the MRS is shown in Figure E-5.

The AMS is a mast-deployed system used to retrieve and convey waste out of the tank. An articulated arm can be extended from the mast for an effective cleaning radius of nominally 4.6 meters (15 feet). At the end of the arm is a fluid-kinetics-driven unit that dislodges waste and provides the kinetic assist to vacuum-induced forces sufficient to draw waste out of the tank.





**Figure E-5. Mobile Retrieval System Schematic**

The vessel skid holds a waste batch vessel connected to the AMS. The vessel is nominally 3,500 liters (925 gallons) in size and can accumulate nominally about 3,000 liters (800 gallons) of waste. The waste batch vessel is placed under a vacuum created by the vacuum skid. This vacuum draws waste into the vessel, where the waste is separated from the gas stream. The gas stream continues to the vacuum skid and back to the tank. When the waste batch vessel is full, the waste is pumped to a receiver tank. Water is mixed with the waste as it is being pumped to achieve a target specific gravity.

The vacuum skid holds two liquid-ring vacuum pumps, along with a seal tank. Only one pump is needed to obtain the vacuum to retrieve waste from the tank; the second pump provides redundancy. The seal tank collects entrained liquid separated from the gas stream at the pump discharge and provides sealing liquid for vacuum pump operation.

A pump skid holds booster pumps and associated equipment for transferring retrieved waste from the vessel skid to the waste receiver tank through double-encased pipelines. The booster pump draws the conditioned waste slurry from the vessel skid through an ultrasonic de-agglomeration unit. The ultrasonic unit is about 1.8 meters (6 feet) long and is fitted with transducer banks on the outside of the unit. Ultrasonic energy (nominally 30- to 40-kilohertz frequency) is transmitted through the pipe wall to the waste slurry to break the binding force of agglomerated waste particles. Two booster pumps are provided on the skid for redundancy. The nominal operating conditions for a single pump are a pressure of 1.17 million pascals (170 pounds per square inch) with a flow rate of 320 liters (85 gallons) per minute.

Each pump is nominally capable of producing a pressure of 1.72 million pascals (250 pounds per square inch) at the pump outlet.

The ITV is an adaptation of a commercially available crawler-type vehicle. A platform on the unit contains a pump and a jetting system for moving waste toward the AMS. The ITV is deployed after the waste within the effective cleaning radius of the AMS has been removed. Waste is either jetted or pushed by the ITV toward the AMS for retrieval. At the end of the retrieval campaign, clean water is supplied to the ITV for rinsing the tank walls and in-tank equipment.

The umbilical management system, located within the upper module of the ITV, stores, deploys, and withdraws hoses connected to the ITV. It consists of a reel used to manage the ITV hose and hydraulic lines. Water supply and hydraulic lines are connected to the umbilical management system in the upper module of the ITV. Water and hydraulic fluid are then conveyed to the ITV through the umbilical line tethered to the ITV.

Decontamination systems use heated water to decontaminate the ITV as it is removed from the tank. Decontamination of the ITV is performed in three stages. Stage 1 involves low-pressure spraying through a ring located in the riser where the ITV is installed. Stage 2 involves high-pressure spraying through rings located in a lower module. Stage 3 is final decontamination performed by an ultrasonic unit inside an upper module. This decontamination unit consists of a tank nominally 3,400 liters (900 gallons) in capacity. The decontamination water from the system drains back into the SST and may be removed using the AMS.

In-tank video systems are installed to allow tank farm operators to monitor the retrieval process and assess MRS performance for meeting retrieval objectives.

A ventilation system maintains a negative tank pressure (below atmospheric) within the specified limits during the retrieval operation. This ensures that airflow is pulled into the tank at all times and airborne contamination is not released from the tank. The ventilation system consists of an inlet filter assembly and a skid-mounted portable exhaustor with HEPA filters, air emission monitors/samplers, and associated ductwork. The tank inlet filter and portable exhaustor inlet ductwork are connected to existing risers on the tank. Collectively, the tank ventilation system performs the following functions:

- Prevents release of airborne radioactive particulates to the environment
- Controls emissions of the air/gas stream before the effluents are discharged to the environment
- Minimizes the amount of condensation in the tank headspace

A control trailer is near the tank from which waste is retrieved. Operation and monitoring of the system are performed from the control trailer. The closed-circuit television monitors and camera manipulation controls are also located within the control trailer.

Prior to initiating retrieval, proper configuration of all equipment is established. All of the necessary portable equipment and instrumentation are delivered to the tank and set up. Jumpers are set and HIHTLs are installed (if required) to establish a waste transfer route. The AMS and ITV systems are installed, and the receiving tank is made ready to receive the waste.

#### **E.1.2.2.2.4 Assumptions and Uncertainties**

For analysis purposes, it was assumed that the construction, operations, and deactivation impacts assessed for tank 241-C-104 retrieval were representative of the proposed MRS.

Temporary facilities associated with the MRS would be moved and reused for four tank retrievals.

The ITV is an adaptation of a commercially available system. This technology has a proven track record in the commercial petroleum and chemical manufacturing industries. The AMS and support equipment have been tested at the Hanford Cold Test Facility with simulated waste. However, it is not certain how well they would function when used for retrieval of actual SST waste. For retrieval analysis, this *TC & WM EIS* assumed that this retrieval technology would function as designed and tested.

The MRS was designed to be placed through existing SST risers. It is uncertain whether all SSTs have enough risers to use the MRS. New risers may be required for some tanks.

Key uncertainties with the MRS include the following:

- Ability to remove hard heels (i.e., dense, highly compacted waste on the tank bottom)
- Ability to remove waste from tank walls, airlift circulators, and other in-tank equipment
- Effectiveness in meeting retrieval goals (i.e., 99 percent or 99.9 percent with a chemical wash)
- Ability to maneuver around in-tank structures and debris

#### **E.1.2.2.3 Vacuum-Based Retrieval**

For analysis purposes, vacuum-based retrieval (VBR) was assumed for removal of 99 percent of the waste from 200-series SSTs, MUSTs, and WRF tanks. VBR with a chemical wash has been identified for removal of 99.9 percent of the waste from all 200-series SSTs, MUSTs, and WRF tanks.

##### **E.1.2.2.3.1 Technology Description**

The following is a description of a typical VBR system that has been used. Actual equipment types, locations, and quantities may vary on a tank-by-tank basis.

##### **E.1.2.2.3.2 Process**

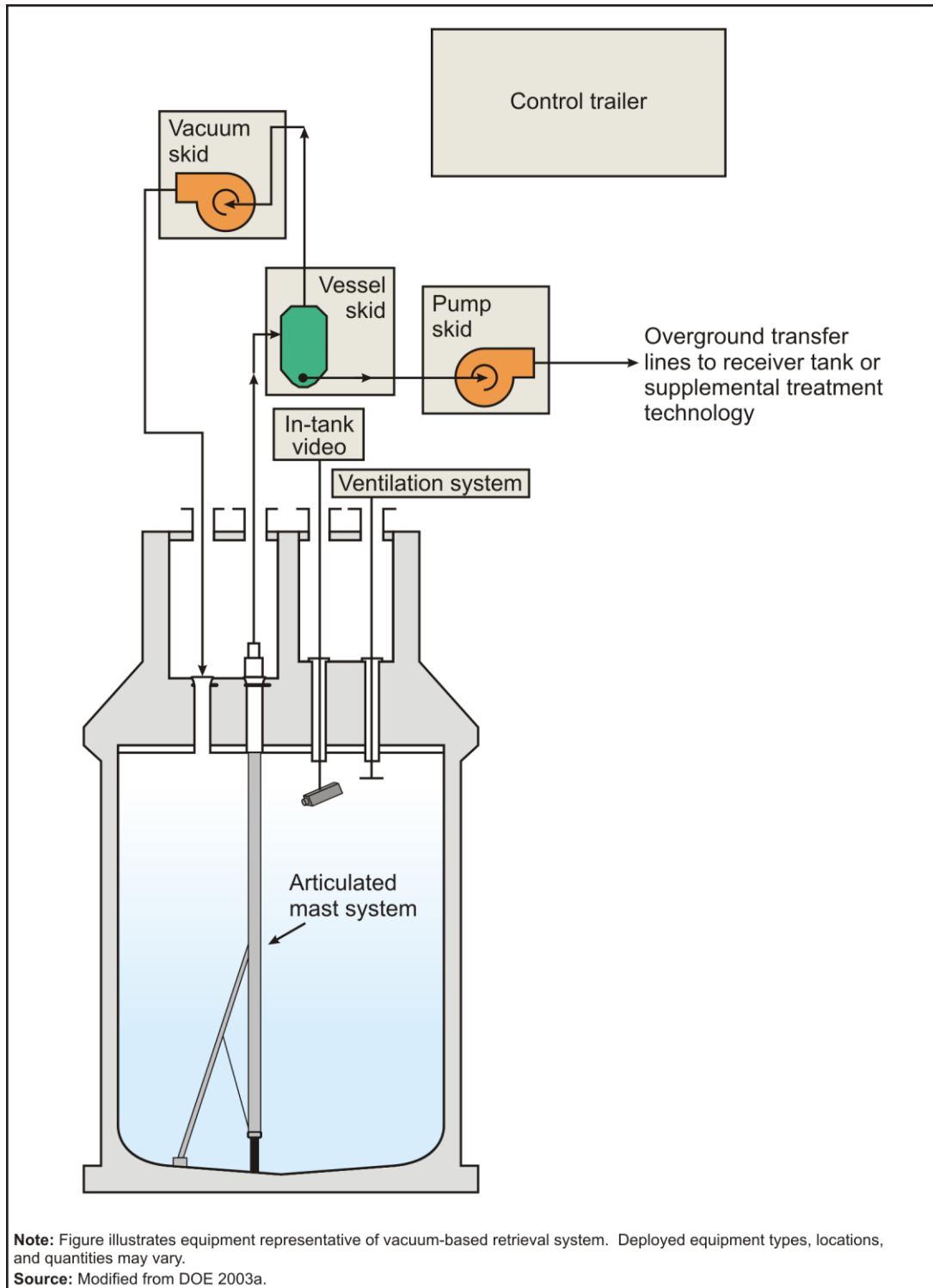
VBR uses little liquid and is accomplished using a vacuum system, with air as the conveyance medium. The vacuum system would be deployed on an AMS located in the central region of the tank. The VBR system is essentially an MRS without the ITV.

The AMS has a 4.6-meter (15-foot) reach from the stationary vertical mast. It is thus capable of reaching the entire tank base of 200-series SSTs (6-meter [20-foot] diameter), but only a portion of the tank base of the 100-series SSTs, which have a 22.9-meter (75-foot) internal diameter.

VBR can potentially be used to retrieve 99 percent of the waste from 200-series SSTs, MUSTs, and WRF tanks, and VBR with a chemical wash can potentially be used to retrieve 99.9 percent of the waste from all 200-series SSTs, MUSTs, and WRF tanks.

##### **E.1.2.2.3.3 Facilities**

The VBR system was identified for retrieving waste from the 200-series SSTs, MUSTs, and WRFs. The VBR system consists of one primary piece of equipment, the AMS. Additional above-tank support equipment is required for the AMS, including vessel, vacuum, and pump skids and a decontamination unit. The facilities are described in greater detail below. The descriptions for the AMS facilities were derived from the *Waste Retrieval and Storage Data Package* (DOE 2003a). Major equipment required for the VBR system is shown in Figure E-6.



**Figure E-6. Vacuum-Based Retrieval System Schematic**

The AMS is a mast-deployed system used to retrieve waste and convey it out of the tank. An extendable articulated arm provides a nominal effective cleaning radius of 4.6 meters (15 feet). A fluid-kinetics-driven unit at the end of the arm dislodges waste and provides sufficient kinetic assist to vacuum-induced forces to draw waste out of the tank.

A waste batch vessel on the vessel skid is connected to the AMS. The vessel is nominally 3,500 liters (925 gallons) in size and can accumulate approximately 3,000 liters (800 gallons) of waste. The waste batch vessel is placed under a vacuum created by the vacuum skid. This vacuum draws waste into the vessel, where the waste is separated from the gas stream. The gas stream continues to the vacuum skid. When the waste batch vessel is full, the waste is conveyed to a receiver tank. Water is mixed with the waste as it is being pumped to achieve a specific target gravity.

Two liquid-ring vacuum pumps are located on the vacuum skid, along with a seal tank. Only one pump is needed to obtain the vacuum needed to retrieve waste from the tank; the second pump provides redundancy. The seal tank collects entrained liquid separated from the gas stream at the pump discharge and supplies sealing liquid (water) for vacuum pump operation.

A booster pump draws the conditioned waste slurry from the vessel skid through an ultrasonic de-agglomeration unit. The ultrasonic unit is about 1.8 meters (6 feet) long and is fitted with transducer banks on the outside of the unit. Ultrasonic energy (nominally 30- to 40-kilohertz frequency) is transmitted through the pipe wall to the waste slurry to break the binding force of agglomerated waste particles. Two booster pumps are provided on the skid for redundancy. The nominal operating conditions for a single pump are a pressure of 1.17 million pascals (170 pounds per square inch) with a flow rate of 320 liters (85 gallons) per minute. Each pump is nominally capable of producing pressure of 1.72 million pascals (250 pounds per square inch) at the pump outlet.

In-tank video systems are installed to allow tank farm operators to monitor and assess tank retrieval operations and performance.

The ventilation system maintains the tank at a slightly negative (less than atmospheric) pressure within specified limits during operation. The ventilation system consists of an inlet filter assembly and a portable exhaustor skid that includes HEPA filters, air monitoring/sampling equipment, and associated ductwork. The inlet filter connects to an existing riser on the tank. Collectively, the tank ventilation system performs the following functions:

- Prevents release of airborne radioactive particulates to the environment
- Controls emissions of the air/gas stream before the effluents are discharged to the environment
- Minimizes the amount of condensation in the tank headspace

A control trailer is near the tank from which waste is retrieved. Operation and monitoring of the system are performed from the control trailer. The closed-circuit television monitors and camera manipulation controls are also located within the control trailer.

Before VBR is initiated, proper configuration of all equipment is established. All of the necessary portable equipment and instrumentation are delivered to the tank and set up. Jumpers are set and HIHTLs are installed (if required) to establish a waste transfer route. The AMS is installed, and the receiving tank is made ready to receive the waste.

#### **E.1.2.2.3.4 Assumptions and Uncertainties**

For analysis purposes, it was assumed that the construction, operations, and deactivation impacts assessed for tank 241-C-104 retrieval were representative of the proposed VBR system.

Temporary facilities associated with the VBR system would be moved and reused for four tank retrievals.

VBR has been used on the 241-C-200 series tanks, which are the smaller 208,000-liter-capacity (55,000-gallon-capacity) tanks. VBR has not been used on one of the larger tanks yet. There is uncertainty regarding how VBR would work on a large scale.

The AMS was designed to be inserted into the Hanford tanks through existing penetrations (risers) through the roof of the tank. The only uncertainty associated with implementation of the vacuum system is whether the tanks identified for retrieval using the VBR system would contain available penetrations of an appropriate size to accommodate the system or if new risers would need to be installed. It may be necessary to remove existing equipment from, or to install new risers on some tanks to allow installation of the AMS.

The VBR system, as described in this appendix, may have to be adapted to suit various configurations of the MUSTs. The VBR system has been proof-of-principle tested at the Hanford Cold Test Facility (Enderlin et al. 2002).

#### **E.1.2.2.4 Chemical Wash System**

A chemical wash system has been identified to supplement the MRS and the VBR system to retrieve 99.9 percent of the waste from Hanford tanks. The chemical wash would be conducted after as much waste as technically practical has been removed using those technologies.

##### **E.1.2.2.4.1 Technology Description**

The following is a description of a typical chemical wash system. Actual equipment types, locations, and quantities may vary on a tank-by-tank basis.

##### **E.1.2.2.4.2 Process**

Chemical washing of internal tank surfaces and equipment is identified for retrieval of residual waste after MRS and VBR system retrieval to meet closure criteria performance objectives. This additional measure may be needed because the other retrieval methods may not directly meet the performance objective, would require significant operational time, or would affect other tank farm operations. For example, the MRS option may not be able to meet performance objectives because in-tank equipment may not allow direct access to some regions within the tank. Specifically, chemical washing is identified for use in conjunction with MRS and VBR system retrieval of 99.9 percent of the waste.

An example of chemical washing is the use of oxalic acid as a solvent for the waste (Sams 2004). Much of the radioactive tank waste was generated from acidic processes; therefore, the use of acids or other chemicals could dissolve the waste into a solution that is more readily removed from the tank. The same methods used to deliver water or waste supernatant into a tank can be used to introduce other chemicals, provided the delivery system's construction materials would not be degraded by the chemical being used. In the same manner, the same equipment used to remove waste by the MRS or VBR can be used to remove the chemical washing solutions if the removal system construction materials are not incompatible with the chemical being used.

The specific chemicals to be used for retrieving residual waste from Hanford tanks would be selected using the following considerations:

- Minimize potential safety impacts (e.g., worker health and nuclear safety).
- Optimize effectiveness in retrieving residual waste.
- Optimize compatibility with construction materials for the tank, retrieval system, and waste transfer lines.
- Optimize compatibility with the waste materials.

- Minimize impacts on downstream waste processing (i.e., vitrification or supplemental treatment processes [STPs]).
- Minimize potential for contaminants to be mobilized into the soil column.

The chemical wash system can be used to supplement the MRS and the VBR system to retrieve 99.9 percent of the waste from tanks.

#### **E.1.2.2.4.3 Facilities**

A chemical addition skid provides structural support, confinement, and leak detection (as required) for the chemical addition manifold (piping, valves, flowmeters, pressure indicators, and various fittings), instrumentation/power enclosures, and pumps. The chemicals would be trucked to Hanford, and the tank on the truck would be used as the primary holding mechanism for the chemical. The chemical (acid or base) would be piped to the chemical addition skid, where it would be pumped to the equipment used to disperse it within the tank.

A chemical manifold would be connected to the chemical truck and piping that runs to the chemical addition skid. Included on the manifold are valves and pressure indicators. The chemical manifold would be equipped with a water flush port to flush the chemical wash system after completion of the chemical addition campaign.

Flexible hoses would be used to provide pathways for the chemical between the chemical truck/tank and chemical addition skid and between the chemical addition skid and retrieval equipment located at the tank.

#### **E.1.2.2.4.4 Assumptions and Uncertainties**

For analysis purposes, it was assumed that chemicals that can effectively clean tanks to the 99.9 percent retrieval goal would also be compatible with safety requirements (e.g., worker health and safety, nuclear safety), construction materials used (e.g., tank liner and retrieval system), tank waste to be treated, and waste feed composition requirements for the WTP or supplemental technologies.

Key uncertainties with chemical washing include chemical compatibility with the DST system, worker safety issues, and waste feed acceptability to the WTP or other solidification technologies.

Chemical (acid) washing in combination with modified sluicing was demonstrated on Hanford SST 241-C-106 from August to December 2003. The objective was to demonstrate 99 percent retrieval of tank waste within acceptable confidence intervals. The results of the demonstration, as reported in *Stage 1 Retrieval Data Report for Single-Shell Tank 241-C-106*, published in May 2004 (Sams 2004), suggest that the combined technologies reached their limit of capability but did not achieve the 99 percent criteria of retrieval, which corresponds to 10.2 cubic meters (360 cubic feet) of remaining tank waste. Based on mass balances, the tank waste remaining was approximately 10.5 cubic meters (370 cubic feet) at the termination of the demonstration. The chemical (oxalic acid) wash was delivered in six stages. After the sixth stage, it was determined that subsequent acid additions would not yield beneficial retrieval of additional tank waste. Chemical (acid) washing has not been demonstrated with MRS or VBR systems at Hanford.



#### **E.1.2.2.5 Leak Detection and Monitoring**

Safe retrieval of tank waste would involve use of procedures, technologies, and systems for detecting environmental releases.

The DSTs are designed with multiple systems for detecting leaks from the primary tank. These systems are installed in the annulus of the DST, the space between the primary and secondary tank liners. In addition, the DSTs have leak detection runners and a leak collection sump located in the base of the outer tank to provide additional protection from potential leakage of waste to the soil.

The SSTs were designed with limited capabilities for detecting and monitoring leaks. Because of the physical limitations of the existing tank systems, detection, monitoring, and mitigation of liquid releases from the SSTs during waste retrieval operations would be more difficult. However, equipment and procedures developed for the SST interim stabilization program would be applicable for leak detection and monitoring during SST retrieval operations.

##### **E.1.2.2.5.1 Technology Description**

Leak detection and monitoring technologies currently available to support waste retrieval from SSTs include the following:

- **High-resolution resistivity.** The high-resolution resistivity leak detection and monitoring system measures changes in the soil resistivity that may occur when conductive tank liquid enters the surrounding soil. Such conductive liquid can alter the current or voltage measured by this leak detection equipment. A resistivity that either decreases or increases with time could result from a number of factors, one of which is a leaking tank. The system has been used on a number of SSTs starting in 2004 (Schofield 2008).
- **Dry-well monitoring.** All 100-series SSTs are equipped with dry wells, which are vertical pipes placed in the soil around the perimeter of the tank extending downward from grade level to depths of about 23 to 46 meters (75 to 150 feet). These pipes are used for deploying gamma radioactivity monitors and neutron moisture probes to detect any liquid waste that has moved from the tank into the soil surrounding the tank. Some of the earliest leaks from SSTs were detected through routine monitoring of tank farm dry wells.
- **Chemical process mass balance.** During the retrieval operation, the amount of waste removed from a tank and any liquid added to aid retrieval would be compared with the amount of material transferred into the receiver tank. If the amount of waste appearing in the receiver tank were less than the amount of waste retrieved, plus any liquid added, a leak would be suspected. These parameters would be tracked continually during the retrieval operation to warn operations personnel of potential leaks.
- **Static-liquid-level observation.** This can also be used to detect a suspected leak. If all transfers out of, and all additions to, a tank are halted, the liquid level of that tank is expected to level off and remain unchanged after the liquid level stabilizes. A measurable drop in the liquid level would indicate the possibility of a leak. For example, a liquid-level change of 1.3 centimeters (0.5 inches) in a 100-series tank (22.9-meter [75-foot] diameter) is equivalent to a change in volume of 5,200 liters (1,400 gallons).

Performance limitations are associated with each of these technologies, and current plans for near-term waste retrieval include the combined use of all three technologies.

#### **E.1.2.2.5.2 Physical Condition of the Single-Shell Tanks**

The SSTs were initially constructed at Hanford as early as 1943 to store radioactive waste. The most significant degradation of the physical integrity of the SSTs occurred while the SSTs were in service and receiving waste from processing facilities (1944 to 1980). Some of the failures were attributed to buildup of high-heat sludge, resulting in elevated temperatures at the bottom of the tank. Tank waste temperatures have cooled over the years, and high-heat sludge has been removed from SSTs, making this failure mechanism unlikely.

Stress-corrosion cracking and pitting/crevice corrosion are the failure mechanisms most applicable to the SSTs that have leaked in the past. The rate at which these modes of corrosion may have progressed in nonleaking SSTs is unknown. However, the general condition and age of the SSTs suggest that new SST leaks could occur during retrieval actions that involve additions of liquid to the tanks (DOE 2003a).

The first SSTs known to leak were tanks 241-TY-109 and 241-U-101 in 1959. As analyzed, 67 of Hanford's 149 SSTs are listed as known or suspected leakers. The SSTs were formally removed from service in 1980, but still contain approximately 120 million liters (32 million gallons) of waste. Estimates of total leak volumes from individual SSTs range from approximately 1,140 liters to 1.05 million liters (300 to 277,000 gallons) per tank (Hanlon 2003). Although the RPP plans to minimize introduction of liquids into suspected leakers (utilizing VBR), for the analysis, all tanks were assumed to leak during retrieval. The *TWRS EIS* (DOE and Ecology 1996) assumed each tank would leak an average of 15,140 liters (4,000 gallons) during SST retrieval. Due to limitations on currently employed leak detection equipment, this assumption was carried forward in this EIS. The leak detection monitoring and mitigation strategy developed for the tank 241-S-112 retrieval demonstration (Hanson 2003) estimated that the best of the three available leak detection methods utilized dry-well gamma logging and neutron soil logging. The 95th percentile upper confidence limit with this method estimated leak detection within a leak volume range from 1,140 to 68,000 liters (300 to 18,000 gallons), depending on where the leak originated in relation to the dry wells. In-tank liquid-balance leak detection methods were less sensitive, ranging from 68,000 to 310,000 liters (18,000 to 82,000 gallons) at the 95th percentile upper confidence limit.

Potential for leakage during retrieval operations would vary based on the height of liquid above potential leak sites (hydrostatic-pressure-driving force) and duration of the retrieval operation.

Technologies to assist in mitigation and improve detection of leakage are currently being evaluated and tested by the RPP. Testing conducted on resistivity-based technologies over a 110-day period in 2003 at the Hanford 105A mock tank test site provided encouraging data for potential future use of much more sensitive leak detection capabilities (Barnett et al. 2003).

#### **E.1.2.2.5.3 Assumptions and Uncertainties**

This *TC & WM EIS* adopts the assumption that each of the 149 SSTs would leak an average of 15,140 liters (4,000 gallons) to soils surrounding the tank during retrieval operations (see Appendix D, Section D.1.6). It is expected that many of the SSTs would not leak during waste retrieval. However, given the limited sensitivity of some SST leak detection systems, larger leak volumes could occur.

No leakage was assumed to occur from the DSTs during retrieval operations because the DSTs have provisions for leak containment and collection.

Similarly, no leakage was assumed to occur from MUST retrieval operations, which would use the VBR system that employs small volumes of water.

#### **E.1.2.2.6 New Risers**

Risers are vertical pipes that extend from aboveground through the dome of the waste tank. Risers of various diameters are used to install pumps, instrumentation, and other equipment in the tanks. New risers may need to be installed on some tanks to provide access for retrieval system equipment. Early SST designs included fewer risers with smaller diameters than newer SSTs and DSTs.

##### **E.1.2.2.6.1 Technology Description**

The following is a description of a typical riser installation in a Hanford waste tank. The need for and location of new actual equipment risers on a tank may vary on a tank-by-tank basis.

##### **E.1.2.2.6.2 Process**

The process of installing a new riser on a Hanford waste tank involves (1) excavating soil from grade down to the tank dome, (2) core drilling through the dome to install the new riser, and (3) backfilling with clean soil.

##### **E.1.2.2.6.3 Facilities**

For each tank from which waste is to be retrieved, existing risers would be checked for size, location, and total number to determine whether they would accommodate required retrieval equipment. As-found equipment in the risers would be evaluated for removal as required. Based on this evaluation, the need for one or more new risers would be determined.

##### **E.1.2.2.6.4 Assumptions and Uncertainties**

Uncertainties in the analysis pertaining to new risers include the actual number of new risers that would be required and the specific tanks on which they would be installed.

Uncertainties with the schedule and cost of installing new risers include the degree of soil contamination present at the new riser location and the number and configuration of underground obstructions (e.g., electrical conduits, transfer lines).

#### **E.1.2.2.7 Waste Transfer**

This section summarizes approaches for transferring waste between tanks (i.e., from SSTs to DSTs and from DSTs to other DSTs) and from tanks to treatment facilities (i.e., from SSTs, DSTs, and WRFs to treatment facilities).

Existing and future waste transfer approaches are discussed below. Future waste transfer system components include underground piping, HIHTLs, container transport, and WRFs.

##### **E.1.2.2.7.1 Existing Single-Shell Tank Waste Transfer Systems**

None of the existing SST transfer piping would be used because (1) the pipelines are single-wall construction and noncompliant with current regulations, (2) some of the pipelines are plugged, (3) many of the pipelines leak, and (4) the pipelines are up to 60 years old.

#### **E.1.2.2.7.2 Existing Double-Shell Tank Transfer Lines**

An existing, extensive system of underground piping connecting the DSTs is operated routinely. This piping would be used for final retrieval of DST waste. In particular, waste removed from DSTs in the 200-West Area would be transferred to selected DSTs in the 200-East Area through the existing underground cross-site transfer system that connects the SY tank farm in the 200-West Area with the DSTs in the 200-East Area.

Waste from various DSTs in the 200-East Area would be transferred through existing underground pipelines to DSTs in the AP tank farm, then through a new underground pipeline to the WTP. The DST transfer system would continue to service the DSTs through the end of the mission.

#### **E.1.2.2.7.3 Use of Hose-in-Hose Transfer Lines**

Hanford uses an HIHTL configuration placed on or near the surface with appropriate shielding. Interim stabilization project efforts currently use this type of transfer line in addition to double-contained underground transfer lines (DOE 2003a).

#### **E.1.2.2.7.4 Future Waste Transfer System**

Existing transfer lines would be used for retrieval of waste from DSTs and WRFs to the extent practicable. Because the DST transfer system would continue to service the DSTs through the end of the mission and the SST transfer system offers limited utility, future transfer system upgrades would be primarily oriented to service the SSTs, through the use of HIHTLs.

The two primary methods for transferring tank waste are via pipeline or container transport. The *Tank Waste Technical Options Report* (Boomer et al. 1993) evaluated these two tank waste transfer alternatives. Public health and safety parameters estimated in the report for container transport and pipeline transfer were roughly equivalent. Occupational health and safety impacts and capital operating costs were estimated to be somewhat higher for container transport than for pipeline transport. For purposes of estimating impacts, this *TC & WM EIS* assumed the tank waste would be transferred predominantly by pipeline. Container transport of tank waste may be used selectively.

The VBR system, MRS, and modified sluicing would make extensive use of HIHTLs. The MRS and the VBR system previously engineered for tank 241-C-104 include approximately 460 meters (1,500 feet) of HIHTLs; the modified sluicing system for tank 241-S-112 includes approximately 230 meters (750 feet) of HIHTLs. These distances of HIHTLs were included in all applications of the VBR system, MRS, and modified sluicing. This length of transfer line would be sufficient to transfer waste beyond the tank farm boundary or to nearby supplemental treatment facilities, but would be insufficient to deliver waste to more-distant locations (DOE 2003a).

#### **E.1.2.2.7.5 New Underground Transfer Lines**

New underground transfer lines would be used to transfer waste beyond the distances that can be reached by HIHTLs. The general configuration of the SST farms suggests that the maximum distances for underground transfer lines are from the B tank farm complex to the 200-East Area DSTs and from the T tank farm complex to the 200-West Area DSTs. For analysis purposes, the 200-East Area destination is designated as the “AY/AZ DST farm” because of its location relative to the B tank farm complex. The 200-West Area destination is designated as the “SY DST farm” because it is the only DST farm in the 200-West Area.

#### **E.1.2.2.7.6 Technology Description**

New underground transfer lines and associated components (valve pits, diversion boxes, booster pumps, and instrumentation) would be constructed. Transfer lines would be double-walled, RCRA-compliant lines.

#### **E.1.2.2.7.7 Process**

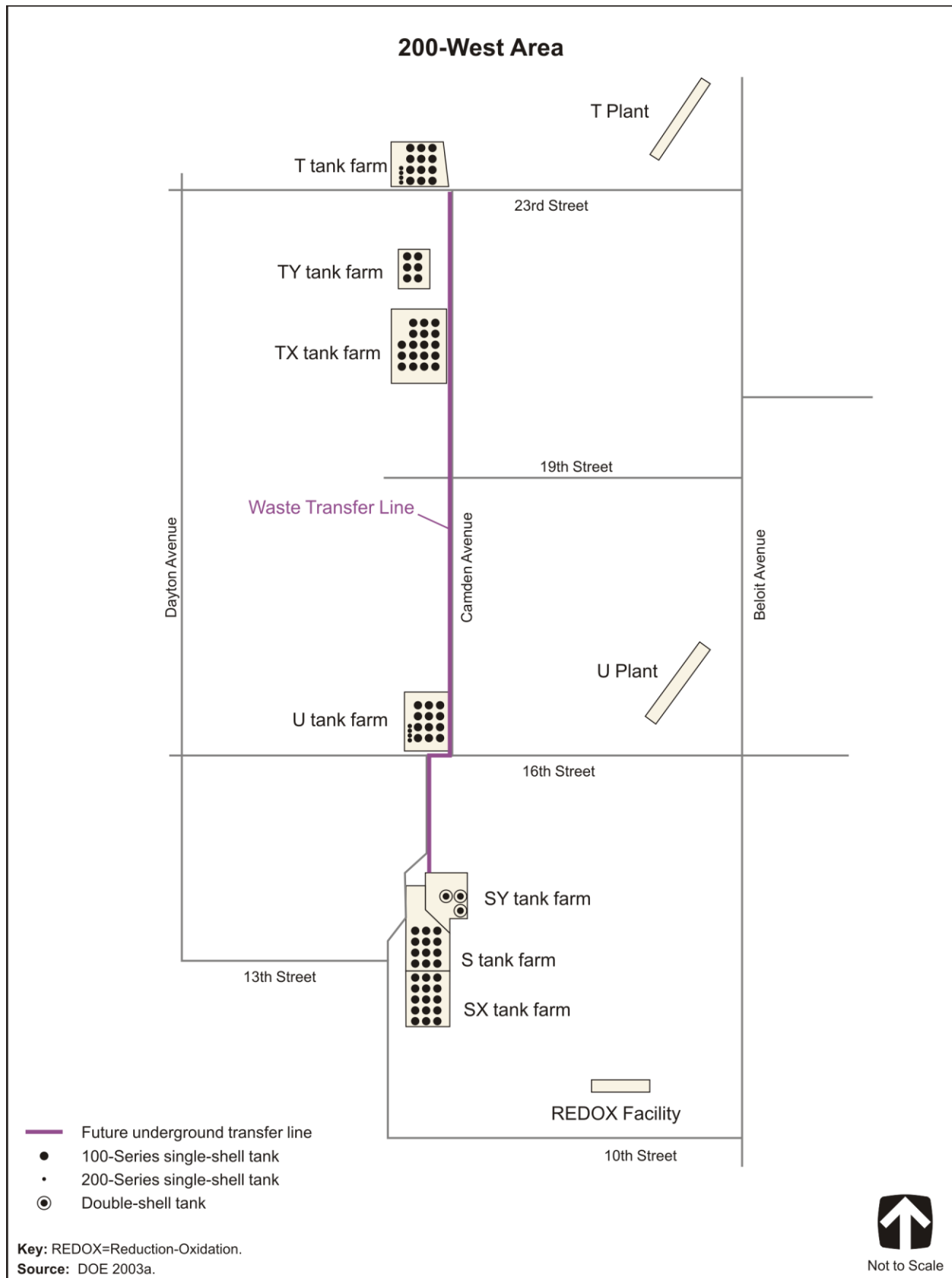
Processes involved in operating underground waste transfer lines include establishing the desired route (setting valves and jumpers in pits and diversion boxes), pumping waste from the source tank to the receiver tank, recycling supernatant back to the source tank if required, flushing the lines after the waste has been transferred, and verifying the volume transferred by material balance. In addition, monitoring and periodic leak testing of transfer lines would be conducted.

#### **E.1.2.2.7.8 Facilities**

Two new transfer line routes are currently planned to connect SST farms to DST farms. One underground transfer line route would originate at the boundary of the T SST farm, be routed past the TX, TY, and U SST farms, and terminate at the SY DST farm (see Figure E-7). The second transfer line route would originate at the boundary of the B SST farm, run beside the C SST farm, and terminate at the 200-East Area DST complex (see Figure E-8). The AY/AZ DST farm is selected for this purpose.

Three transfer lines (one primary transfer, one recycle, and one spare) would be constructed for each route described above. The transfer lines would be double-walled pipe with heat-tracing and leak-detection features. Booster pump stations, access points to leak detection instrumentation, and other equipment would be provided along the transfer route as required. Transfer lines could be constructed on grade and shielded with concrete enclosures or placed in trenches below grade. To provide a conservative estimate of environmental impacts of excavation, it was assumed that transfer lines would be constructed in trenches below grade.

Following completion of waste retrievals, the underground transfer lines would be left in place. These transfer lines would have been essentially decontaminated during routine waste retrieval and transfer operations through use of routine transfer line flushes. Leaving the lines in place would result in 2,100 meters (7,000 feet) each of primary, recycle, and spare transfer lines from the T tank farm to the SY tank farm and 2,300 meters (7,500 feet) each of primary, recycle, and spare transfer lines between the B tank farm and the AY/AZ tank farms (DOE 2003a).



**Figure E-7. Future 200-West Area Underground Transfer Line**

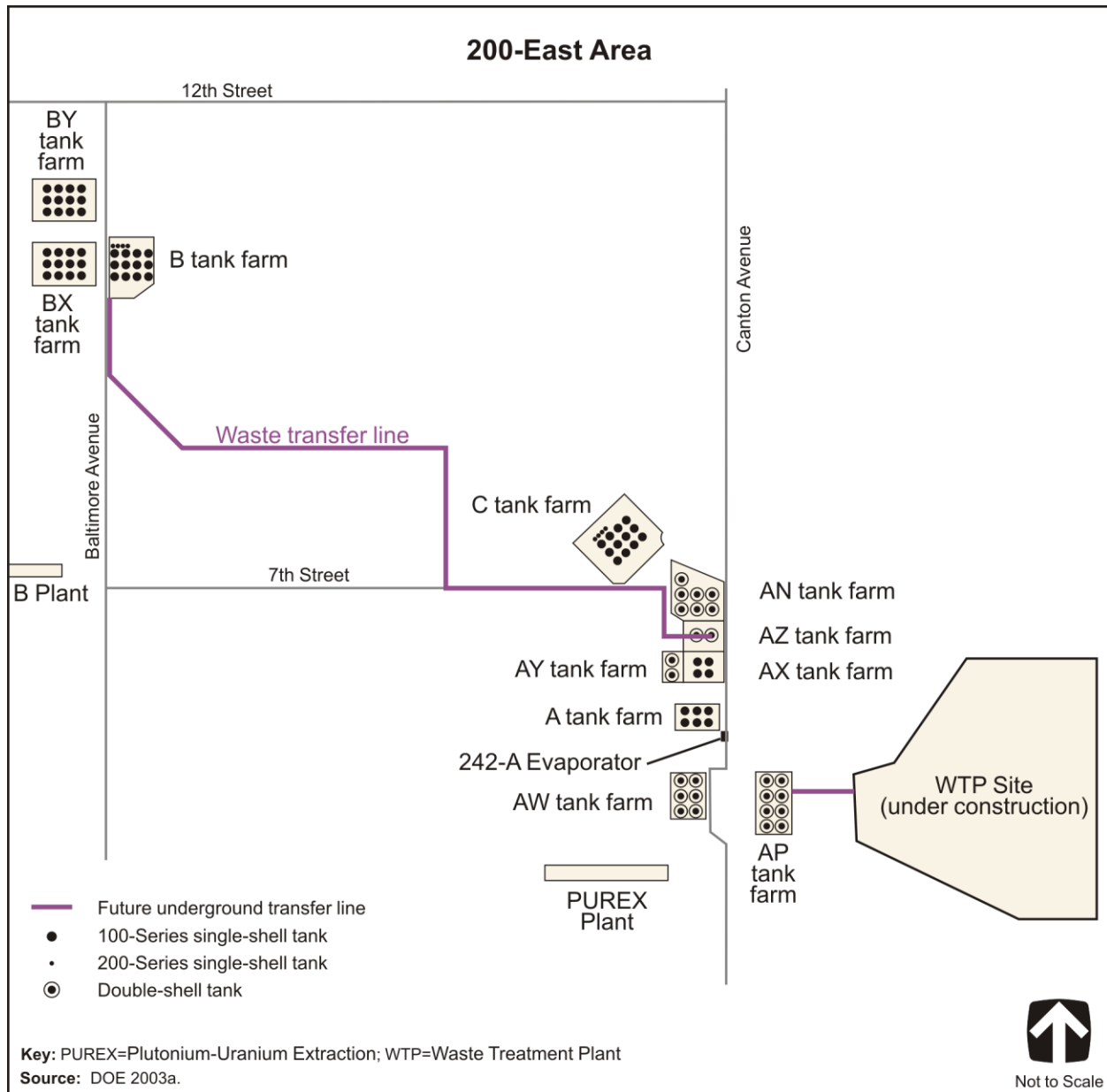


Figure E-8. Future 200-East Area Underground Transfer Line

E.1.2.2.7.9 Assumptions and Uncertainties

For planning purposes, it was assumed that the following constraints apply:

- HIHTL lengths would not exceed those identified for tank 241-C-104 or 241-S-112 retrieval.
- Below-grade waste transfer lines would remain in place at completion of the retrieval activities.
- Existing transfer lines in the northeast (B, BX, and BY tank farms) and northwest (T, TX, and TY tank farms) quadrants of the tank farm system can support a maximum of six simultaneous retrievals or transfers in each tank farm and a total of six simultaneous retrievals or transfers in each quadrant.

- For the southwest (S, SX, SY, and U tank farms) quadrant, operations would be limited to two simultaneous retrievals or transfers per tank farm and six simultaneous retrievals or transfers per quadrant.
- For the southeast (A, AX, C, and DST tank farms) quadrant, the limits would be two simultaneous retrievals or transfers, both of which would be allowable within the same tank farm. A maximum of seven simultaneous transfer operations can be performed, and the waste from up to two SSTs may be transferred into one DST at a time (Hanlon 2003).

The additional underground transfer lines and WRFs identified in this *TC & WM EIS* may not increase the allowable number of simultaneous transfer operations. Process modeling would be used to evaluate applicable constraints on the identified waste transfer system.

#### **E.1.2.2.8 Waste Receiver Facilities**

WRFs would facilitate retrieval of waste from SSTs and MUSTs for transfer to the DST system. WRFs may also be used to condition waste from SSTs and DSTs for transfer to waste treatment facilities. WRF construction and operations are analyzed under all Tank Closure alternatives, except 1, 2A, and 6A.

##### **E.1.2.2.8.1 Technology Description**

WRFs accumulate waste during retrieval; condition waste by dissolution, dilution, or size reduction of particles; and provide batches of waste for subsequent transfer. The WRFs could also be used to recirculate sluicing liquids back to the SSTs. Not all SST retrievals would necessarily require WRFs.

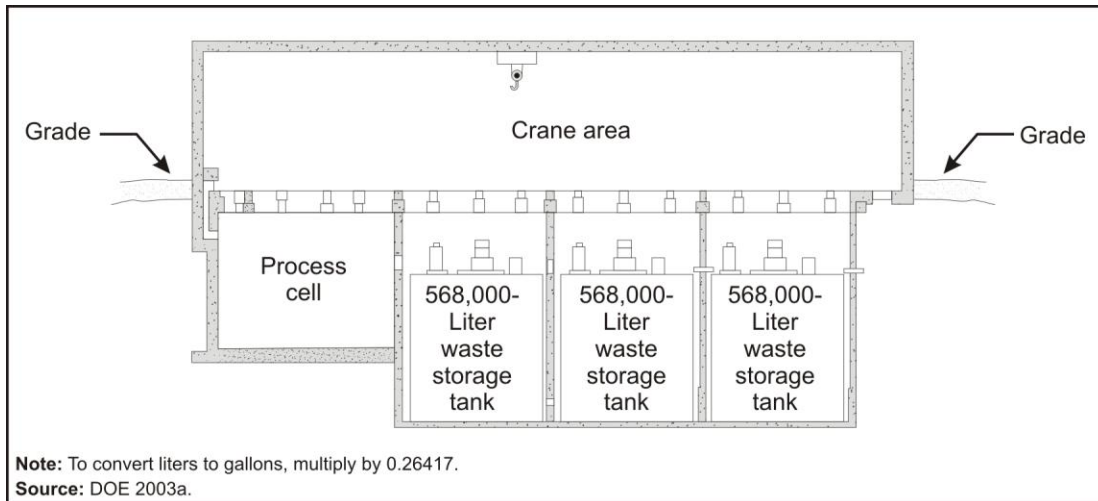
##### **E.1.2.2.8.2 Process**

In the WRFs, waste can be conditioned through dissolution, dilution, and size reduction (of particles suspended in the waste slurry), if necessary. The actual size of the WRFs would vary based on many design considerations. For this EIS, the WRFs analyzed would contain four process cells serviced by an overhead crane. Three of the cells would each contain a 568,000-liter (150,000-gallon) tank and agitator, while the remaining cell would contain transfer pumps and miscellaneous equipment used for waste conditioning and decontamination. Each WRF would also contain an equipment load-out bay and crane maintenance area. Three pipe runs would service each WRF. The two runs that would service the tank farms would include three lines each (one transfer, one recycle, and one spare). The run that would transfer slurry from the WRF to the DSTs or waste treatment facilities would contain two lines (one transfer and one spare) (DOE 2003a).

##### **E.1.2.2.8.3 Facilities**

A WRF would have a facility footprint approximately 61 meters long by 15 meters wide (200 feet long by 50 feet wide), with four separate process cells. Operations in each process cell would be conducted remotely (no personnel present during processing). The cell would be serviced by an overhead crane. Each WRF would also have a separate control room for operations personnel, an equipment load-out bay, and a dedicated maintenance area. The tanks, cells, and crane area would be appropriately ventilated. See Figure E-9 for a schematic diagram of a WRF.





**Figure E-9. Waste Receiver Facility Schematic**

A waste retrieval and transfer logistical analysis conducted using the HTWOS [Hanford Tank Waste Operation Simulator] model was used as the basis for the number and locations of WRFs required. Four WRFs would be constructed near clusters of SSTs, as follows:

- Adjacent to the B, BX, and BY tank farms in the 200-East Area (WRF-B)
- Between the T and TY tank farms in the 200-West Area (WRF-T)
- Between the U and S tank farms in the 200-West Area (WRF-S)
- Adjacent to the A tank farm in the 200-East Area (WRF-A)

Under alternatives that require WRF construction, this would be integrated with construction of new transfer lines, as discussed in the previous section (DOE 2003a).

#### **E.1.2.2.8.4 Assumptions and Uncertainties**

- Uncertainties associated with WRFs are related primarily to the total number of WRFs required, the number of tanks within each WRF, and the size of the tanks within each WRF.
- Four WRFs were assumed to be constructed, with a maximum of three 568,000-liter (150,000-gallon) tanks per WRF.

#### **E.1.2.2.9 Waste Staging for Treatment**

Several alternatives considered in this *TC & WM EIS* require construction of additional storage tanks. Alternative 5 requires the construction of four new DSTs to provide additional storage capacity, while Alternatives 2A and 6A would require replacement of existing DSTs when they reach the end of their useful lives.

##### **E.1.2.2.9.1 Technology Description**

All of the alternatives evaluated in this *TC & WM EIS* require the use of existing storage facilities (with or without modification) to store retrieved tank waste. Alternative 5 requires additional storage capacity, while several other alternatives require replacement of DSTs as they reach the end of their useful lives.

Alternative 5 proposes the use of existing storage facilities that would be modified or supplemented with new waste storage facilities to support near-term retrieval and treatment of tank waste. The storage option selected to support Alternative 5 is the construction of new underground DSTs.

Modification of existing facilities to increase storage space using administrative controls is discussed in Section E.1.2.1.1. Impacts of that option would be less than those of constructing new DSTs.

#### **E.1.2.2.9.2 Process**

Retrieval systems for DSTs have been designed and installed in selected DSTs to support waste feed delivery for the WTP. The retrieval systems consist of a combination of mixer and retrieval pumps designed to slurry the contents of the tank and to pump the waste out of the tank into the waste transfer system. Construction, operations, and deactivation impacts of DST retrieval are included for analysis in this EIS.

#### **E.1.2.2.9.3 Facilities**

Construction of new DSTs to provide safe storage space for retrieved SST/DST waste was evaluated in the early 1990s as a multifunction waste tank facility project (DOE 2003a). A facility similar to the multifunction facility would be constructed on approximately 10 hectares (24 acres) near the WTP site in the 200-East Area. Electric power, raw water, steam, sanitary water, and process lines would be routed to the site. Discussions in the following sections are based on the multifunction facility design.

New DSTs would be constructed in modules, with four DSTs per module, similar to the multifunction waste tank facility design. Each tank would have a 23-meter (75-foot) internal diameter and a capacity of up to 4.5 million liters (1.2 million gallons). The outer tank would be steel-lined reinforced concrete that provides the secondary containment required by DOE directives. The inner tank would be enclosed within the outer tank and separated from the outer tank liner by an annular space where leak detection equipment would be installed (see Figure E-2). The outer tank would serve as secondary containment in the event of a leak from the primary tank liner. In addition, the base of the outer tank would be equipped with leak detection runners leading to a sump for detecting and mitigating possible leakage from the outer tank of a DST.

Each primary tank would be equipped with a transfer pump, and transfer lines would be installed to connect the tank system with the AP tank farm, other tank farms, or the WTP, as required. All transfer lines, process piping, sample lines, and drain lines for tank waste would be encased in secondary piping equipped with leak detection systems (DOE 2003a).

#### **E.1.2.2.9.4 Assumptions and Uncertainties**

Construction of new DSTs is technically feasible based on previous design, construction, and operations experience. The approach established here provides a solid technical basis for cost, schedule, and resource estimates.

New DSTs would be constructed in sets of four tanks. Each set of four tanks would require 10 hectares (24 acres) of land.

New DSTs would be constructed of carbon steel. A weather enclosure would not be required.

### **E.1.2.3 Waste Treatment**

This section presents detailed information on the waste treatment technologies identified with the various alternatives considered in this *TC & WM EIS*. The subsections begin with WTP treatment as currently planned and go on to describe technologies that could be used to expand WTP treatment capacity, particularly for ILAW glass. Following the discussion on WTP activities, several subsections are dedicated to upgrading or replacing facilities that currently support the RPP program. The remaining subsections describe supplemental technologies that could be used either in conjunction with, or separate from, the WTP to accelerate treatment of Hanford tank waste.

The WTP would pretreat and immobilize (by vitrification to borosilicate glass) waste now stored in underground tanks at Hanford. The WTP consists of three individual waste treatment facilities (Pretreatment, HLW Vitrification, and LAW Vitrification), a standalone Analytical Laboratory, and the BOF. Figure E-1 shows the WTP site plot plan with all of the major facilities identified. Figure E-10 shows the interrelationships between the WTP and the supplemental treatment technologies for Hanford tank waste that are evaluated in this *TC & WM EIS*.

WTP pretreatment is discussed in Section E.1.2.3.1.1; WTP HLW vitrification, in Section E.1.2.3.1.2; and WTP LAW vitrification, in Section E.1.2.3.1.3.

The following subsections discuss existing treatment facilities that would require upgrading or replacement: the 242-A Evaporator (see Section E.1.2.3.2) and LWPFs (see Section E.1.2.3.3).

The following subsections also discuss new supplemental treatment facilities such as the Cesium and Strontium Capsule Processing Facility (see Section E.1.2.3.4); thermal supplemental treatment facilities (see Sections E.1.2.3.6 and E.1.2.3.8); nonthermal supplemental treatment facilities (see Section E.1.2.3.7); the Sulfate Removal Facility (see Section E.1.2.3.9); technetium removal facilities (see Section E.1.2.3.10); and mixed TRU waste processing facilities (see Section E.1.2.3.11). Bulk vitrification and steam reforming are the two representative thermal supplemental treatment technologies analyzed in this EIS; cast stone, the nonthermal.

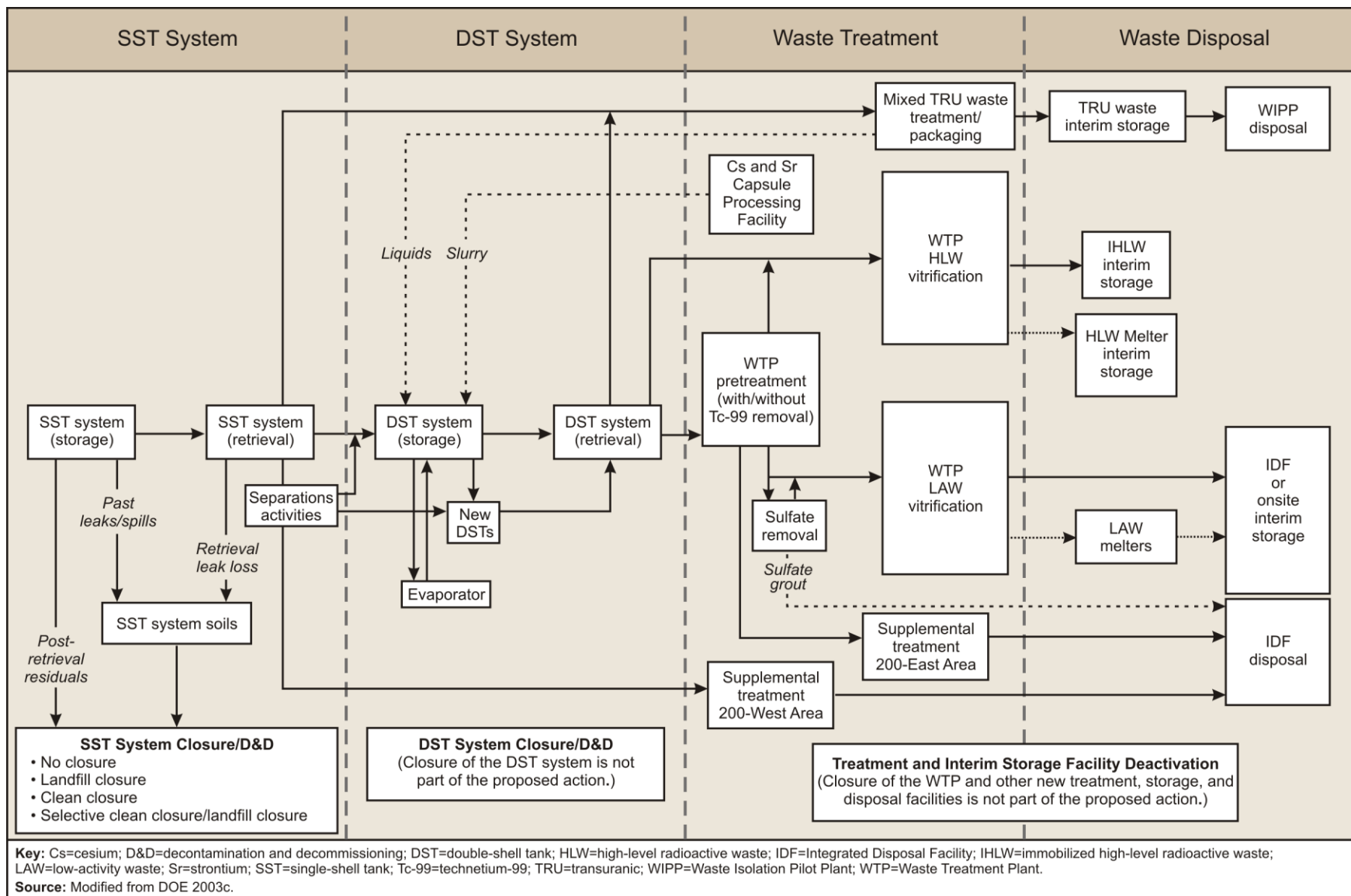


Figure E-10. Relationships Between Waste Treatment Plant Tank Systems and Supplemental Treatment Technologies

The primary-waste forms that would be generated by each of the supplemental treatment technologies are also discussed. Further treatment and disposal that may be needed (e.g., liquid effluents) would be accomplished by utilizing existing Hanford treatment and/or disposal facilities (e.g., the LWPFs). Table E-4 provides the technical basis of key features for supplemental technologies.

**Table E-4. Basis for Supplemental Treatment Estimates**

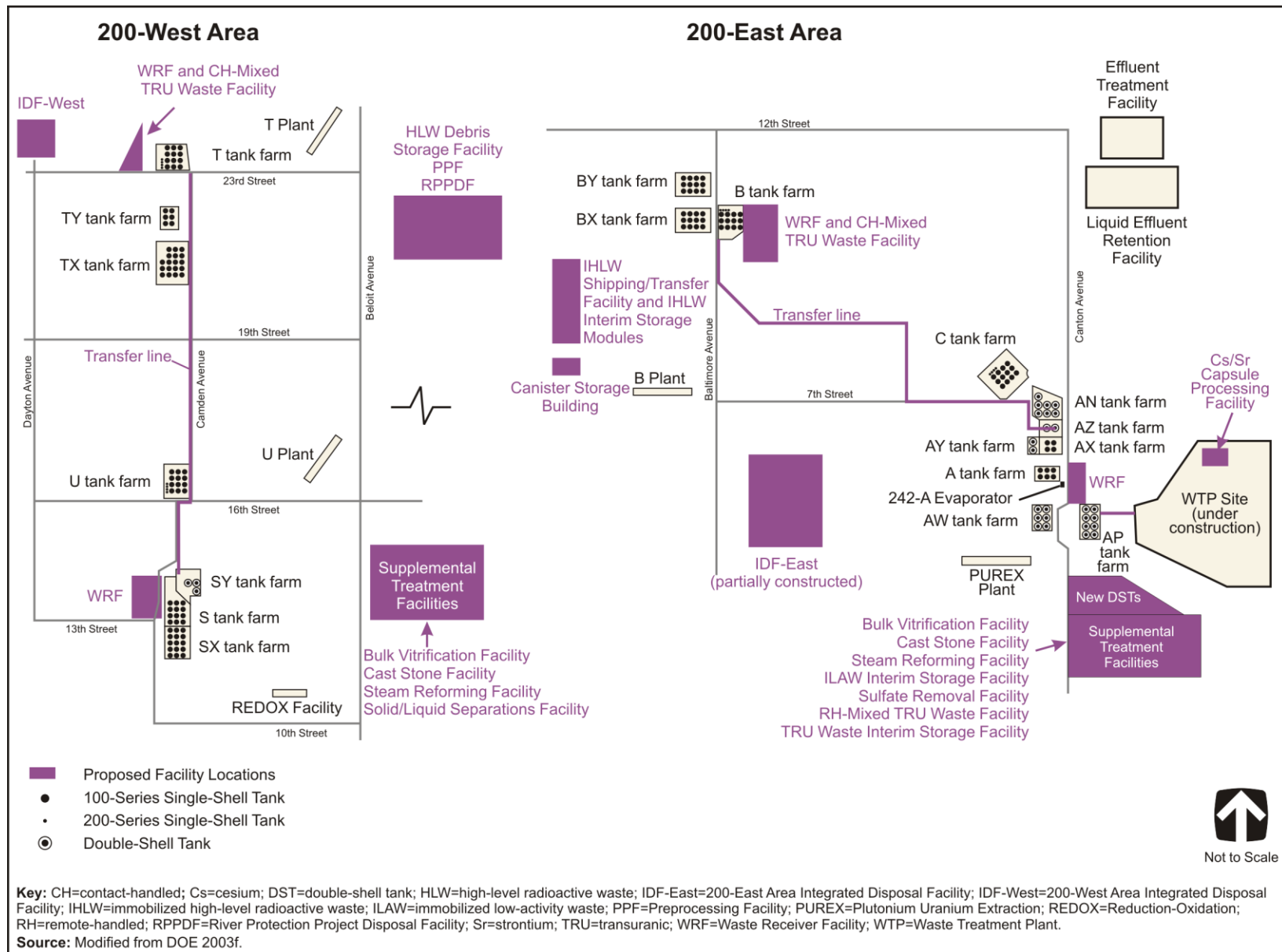
<b>Analysis Features</b>	<b>Mixed TRU Waste</b>	<b>Bulk Vitrification</b>	<b>Cast Stone</b>	<b>Steam Reforming</b>	<b>Sulfate Removal</b>
Facility size and layout	CH-TRU waste procurement specification	Vendor proposal	Existing grout facility modified for containers	Scaled from vendor information on existing commercial facilities	Modified WTP pretreatment process
Construction resources	Scaled by ratio of supplemental treatment facility to WTP	Scaled by ratio of supplemental treatment facility to WTP	Scaled by ratio of supplemental treatment facility to WTP	Scaled by ratio of supplemental treatment facility to WTP	Scaled by ratio of supplemental treatment facility to WTP
Operations supplies	TRU waste packaging procurement specification (57 liters absorbent per drum)	Target sodium oxide loading in glass, 20 percent by mass	Based on SRS-tested formulation	Based on bench-scale simulant testing at Hazen Research Facility and Idaho National Laboratory	Mass balance calculations
Operations staffing	Facility-specific staffing estimate	Facility-specific staffing estimate	Based on grout facility restart estimate	Facility-specific staffing requirement	Facility-specific staffing estimate
Waste form characteristics <sup>a</sup>	Dewatered sludge in 0.21-cubic-meter drums	Sodium aluminosilicate glass in 54.3-cubic-meter steel containers	Cast stone in 11.5-cubic-meter steel containers	Mineralized granulated material in 2.25-cubic-meter steel containers	Solidified grout in 3.6-cubic-meter steel containers

<sup>a</sup> Basis for all is material balance calculations.

**Note:** To convert cubic meters to cubic yards, multiply by 1.308; liters to gallons, by 0.26417.

**Key:** CH=contact-handled; SRS=Savannah River Site; TRU=transuranic; WTP=Waste Treatment Plant.

Final locations for the supplemental treatment facilities have not been selected. For analysis purposes, representative locations of these facilities in the 200-East and 200-West Areas are shown in Figure E-11. Supplemental treatment facilities would be south of the WTP in the 200-East Area and in the southeastern portion of the 200-West Area. Figure E-11 also shows locations of the mobile Contact-Handled (CH)-Mixed TRU Waste Facilities. The two mobile packaging units may require up to three locations in the 200-West Area and one location in the 200-East Area. The Remote-Handled (RH)-Mixed TRU Waste Facility would be located in the southeastern portion of the 200-East Area, near the AP tank farm.



**Figure E-11. Proposed Locations of Core Zone**

The current approach to waste immobilization at Hanford uses borosilicate glass produced in joule-heated melters. Possible improvements in glass formulation and enhancements to the melter technology could be used in conjunction with the currently planned approach. These enhancements could increase waste loadings, increase throughput, or otherwise lead to cost reduction and schedule improvement. DOE conducted a technical review of alternatives for vitrification of Hanford tank waste that could achieve cost reductions within reasonable long-term program risks. Enhancements to the waste forms that may allow troublesome chemical constituents to be incorporated include adjustments to the formulations and establishment of acceptable limits of crystalline phases in the waste glass (DOE 2003f).

Melter technology improvements to increase waste loading, melting rate, and predictability of processing are being sought. Enhancements to the existing joule-heated ceramic melter technology include forced convection, higher operating temperatures, and configuration changes to increase tolerance for accumulation of crystals in the melter.

#### **E.1.2.3.1 Waste Treatment Plant**

The WTP, currently under construction, would be a new facility on Hanford property owned by DOE. The site lies at the eastern end of the 200-East Area of Hanford. The WTP is designed to accommodate a TMC of 6 metric tons of glass IHLW per day and 30 metric tons of glass ILAW per day by the employment of two HLW melters and two LAW melters.

The WTP would be constructed, permitted, and operated to treat radioactively contaminated dangerous waste stored at Hanford and managed by the Office of River Protection (ORP).

The WTP would include three primary processing facilities: Pretreatment, LAW Vitrification, and HLW Vitrification. These facilities would be supported by the Analytical Laboratory and the BOF. The WTP would have a 40-year design life. Under Tank Closure Alternatives 2A and 6A (Base and Option Cases), replacement of the WTP would be required because the estimated operational life of the facility would be exceeded. For analysis purposes, it was assumed that replacement of the WTP would be required after 60 years of operation. This time period assumes a 20-year period of life extension activities that would extend the 40-year design life. Therefore, Tank Closure Alternative 2A would require one WTP replacement, and Tank Closure Alternative 6A would require two WTP replacements. For analysis purposes, it was assumed that these replacement facilities would be adjacent to the WTP, currently under construction.

#### **PROCESS OVERVIEW**

The following discussion presents an overview of the facilities, laboratory, and BOF systems at the WTP. Figure E-1 is a graphic representation of the WTP site showing the facilities' locations.

The WTP would receive waste from the DSTs in batches. The waste feed solutions and slurries would be transferred by pipeline to the WTP for pretreatment and immobilization by vitrification. The pretreatment process is designed to separate the solid and liquid fractions of the waste, wash and/or leach the solids to remove chemical constituents that drive the total IHLW glass volume, and remove selected radionuclides from the liquid portion of the waste. The pretreated liquid waste would be routed to the LAW Vitrification Facility; the pretreated solids and separated radionuclides would be routed to the HLW Vitrification Facility.

In the vitrification facilities, pretreated tank waste would be combined with glass-forming materials and melted at high temperatures to form a liquid that would be poured into stainless steel containers. After the glass cools and hardens, the containers would be sealed and decontaminated in preparation for storage and permanent disposal. Dangerous waste and radioactive constituents in the waste would be either removed or immobilized in the durable glass matrix as a result of the vitrification process. The offgas from the processes would be treated to a level that protects human health and the environment.

During routine operation of the WTP, miscellaneous wastes such as used equipment, laboratory waste, and other chemically and radioactively contaminated materials would be generated. This newly generated waste is referred to as “secondary waste.” Secondary waste would be transferred to accumulation or storage facilities at the WTP and then either transferred to Hanford facilities or transferred off Hanford, as appropriate. Nonradioactive dangerous waste may also be generated by operations, laboratory, and maintenance activities. This waste would be managed at the WTP until it can be transferred to a permitted disposal facility.

A simplified block flow diagram of the WTP process appears in Figure E-12.

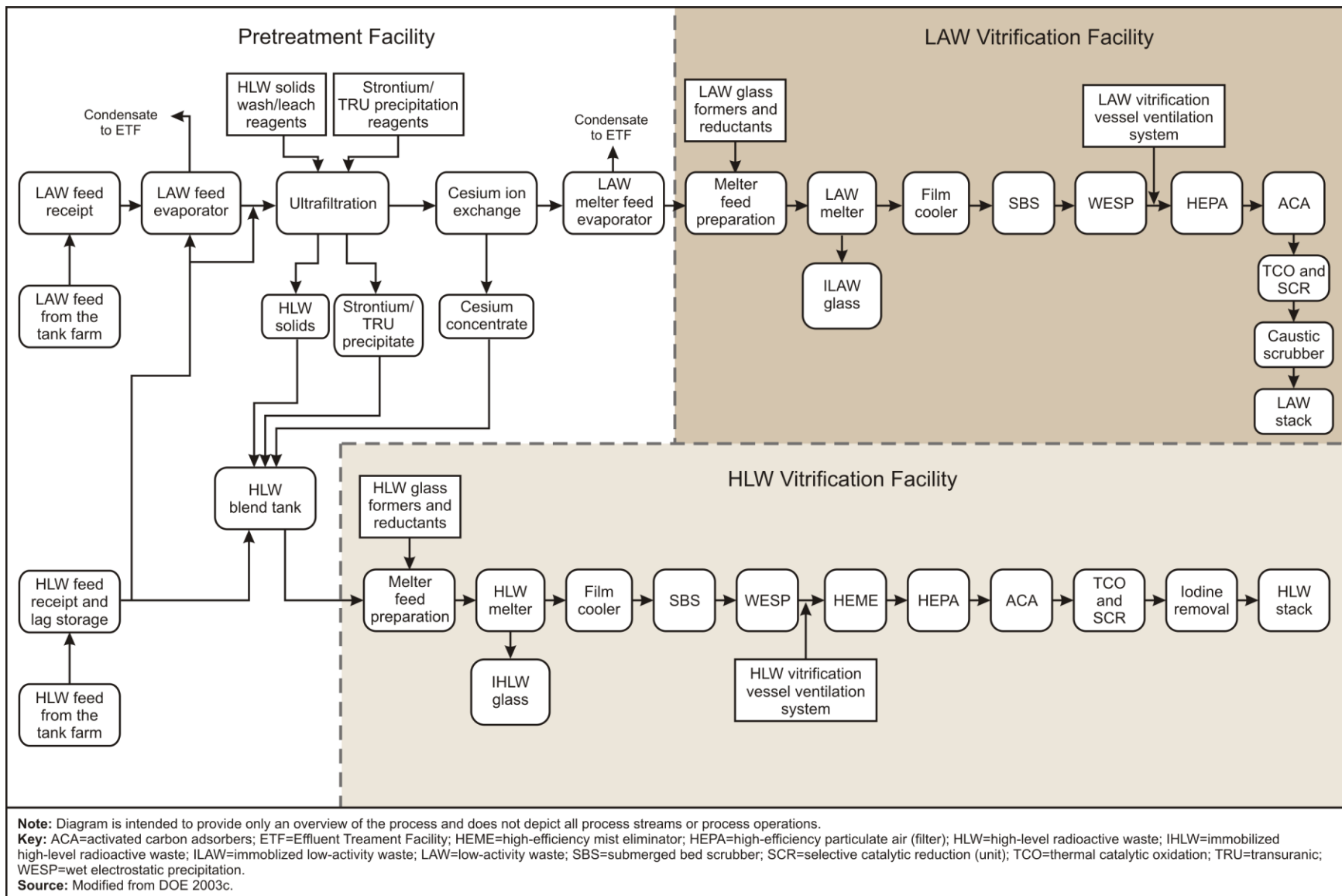
The pretreatment process divides waste feeds into LAW and HLW streams. The LAW stream would consist primarily of salt solutions with much of the radioactivity removed. The HLW stream would consist primarily of solids separated from the tank waste and selected radioactive components that would be removed from the aqueous LAW stream. The LAW vitrification process would immobilize the pretreated LAW component, producing a vitrified product known as ILAW glass. The HLW vitrification process would immobilize the pretreated HLW component, producing a vitrified product known as IHLW glass. Offgas generated by the individual processes would be treated by the offgas treatment systems.

Waste would be transferred from the DSTs in batches through double-walled pipelines to the WTP waste receipt tanks. HLW solids, strontium, transuranics, and cesium would be separated from the waste feed by ultrafiltration, precipitation, and ion exchange processes. The pretreated HLW solids and separated radionuclides would be blended and transferred to the HLW Vitrification Facility as pretreated HLW feed. The pretreated supernatant and permeate (the liquid-waste stream, with inherent solids and gases) resulting from the separations processes would be transferred to the LAW Vitrification Facility as pretreated LAW feed.

The LAW vitrification process would immobilize the pretreated LAW component by first adding glass-forming materials to the pretreated LAW feed, then vitrifying the mixture in a joule-heated, ceramic-lined melter. Similarly, the HLW vitrification process would immobilize the pretreated HLW component by adding glass-forming materials to the pretreated HLW feed, then vitrifying the mixture in a joule-heated, ceramic-lined melter. The molten glass products (ILAW and IHLW) would be poured from the melters into stainless steel containers and allowed to cool and solidify. The containers would then be sealed, decontaminated, and transferred to either an Integrated Disposal Facility (IDF) (ILAW glass only) or the IHLW Interim Storage Modules.

Detailed process descriptions provided in the following section are based on process descriptions presented in the *No Action Baseline Tank Farm Operations and Waste Treatment Data Package* (DOE 2003c) and the *Flowsheet Bases, Assumptions, and Requirements* document (BNI 2005).





**Figure E-12. Simplified Block Flow Diagram of the Current Waste Treatment Process**

### E.1.2.3.1.1 Waste Treatment Plant Pretreatment

The pretreatment process involves operations for pretreating tank waste before immobilization. The major types of tank waste are differentiated into waste envelopes based on chemical characteristics and are designated as “Envelopes A, B, C, and D.” The following are descriptions of these waste envelopes:

- **Envelope A.** This waste envelope constitutes the largest volume of the LAW to be treated, has a nominal concentration of 6- to 8-molar sodium, and contains radioactive cesium at concentrations high enough to warrant its removal prior to vitrification.
- **Envelope B.** This waste envelope is a small-volume feed stream similar to Envelope A, except that it contains higher concentrations of compounds that limit waste loading in the glass (such as sulfates) and has a nominal concentration of 2- to 4-molar sodium. This waste envelope also contains higher concentrations of radioactive cesium than Envelope A. Radioactive cesium is present at concentrations sufficient to warrant its removal prior to vitrification.
- **Envelope C.** This waste envelope has a nominal concentration of 6- to 8-molar sodium, but contains organic compounds that can hold strontium and TRU waste components in solution. This waste would require treatment (e.g., precipitation) to remove both the strontium and transuranics. As with Envelopes A and B, radioactive cesium is present at concentrations sufficient to warrant its removal prior to vitrification.
- **Envelope D.** This waste envelope represents the undissolved solids from the tank waste and contains approximately 10 to 200 grams (0.35 to 7 ounces) of unwashed solids per liter.

In general, Envelopes A, B, and C consist of the supernatant portion of the tank waste and make up the LAW feed to pretreatment. Envelope D consists of the undissolved solids portion of the tank waste and makes up the HLW feed to pretreatment. LAW feeds (Envelopes A, B, and C) would be transferred to the WTP as concentrated solutions that contain up to 2 weight-percent entrained solids.

The pretreatment process has a TMC of 2,950 metric tons of sodium per year to support LAW vitrification and sufficient solids processing capacity to support 6 metric tons of glass per day for HLW vitrification.

The initial step in the pretreatment of LAW would be evaporation of dilute feeds, or dilution of concentrated feeds, to approximately 5-molar sodium concentration. This would dissolve soluble salts and prepare the feed for subsequent processing. From there, the waste would go through the following processes:

- Envelope A and B feeds would be blended with HLW feeds (Envelope D) in an ultrafilter preparation tank and then filtered through an ultrafiltration system to remove any entrained solids. The LAW permeate (liquid stream) would be processed through the cesium ion exchange process, and the concentrated solids slurry would be caustic-leached (if warranted), washed, oxidative-leached (if warranted), washed, and blended with cesium concentrate from the ion exchange and strontium and transuranic waste solids from the strontium/transuranic waste precipitation process. The entrained solids would be stored before HLW vitrification and the LAW permeate would be stored before LAW vitrification (BNI 2005).
- Envelope C feeds contain organic complexants that cause the strontium and some TRU waste to remain in solution. This waste would undergo a strontium/transuranic waste precipitation process before filtration. The filtration step would separate the strontium/transuranic waste solids, manganese oxide and entrained solids from the LAW permeate. The strontium/transuranic waste

precipitate would be washed and stored for blending with HLW feed before HLW vitrification. The strontium/transuranic waste precipitate would not be caustic-leached. Envelope C permeates would be processed through the cesium ion exchange process (BNI 2005).

- After filtration, the LAW permeate undergoes cesium ion exchange processing and the cesium eluate is concentrated by evaporation and then blended with pretreated HLW solids before transfer to the HLW vitrification process. The last step in the Pretreatment Facility process would be to concentrate the treated LAW liquid by evaporation before transferring the waste to the LAW vitrification process (BNI 2005).

Radioactive or dangerous waste liquid effluents would be either recycled in pretreatment or sent to the ETF.

### **PRETREATMENT FACILITY**

The Pretreatment Facility would contain the equipment necessary for preparing the LAW and HLW feeds for vitrification. The Pretreatment Facility would also contain the equipment to receive waste transfers from the Hanford tank farms, including receipt tanks that can hold a total of 5.8 million liters (1.5 million gallons). Equipment that mixes, handles, and transports highly radioactive waste would be remotely operated and located within black cells, cells which workers do not enter for the life of the facility.

The building ventilation system would be designed to cascade the supply air from areas with low potential for contamination into areas with high potential for contamination. The ventilation discharge from the high-contamination areas would then be filtered before release. A completely separate system would vent the process tanks and vessels.

The structure of the Pretreatment Facility would be supported by a reinforced-concrete foundation. The superstructure would be constructed of structural steelwork with external metal siding and a metal roof. The process cells within the Pretreatment Facility would be constructed of reinforced concrete to protect facility operators from radiation and to ensure facility containment in a post-seismic or design-basis event. The cell floors and a portion of the cell walls would be lined with stainless steel to provide a surface capable of lifelong decontamination and to provide a tank leaks barrier for the process tanks.

The Pretreatment Facility was originally designed to remove technetium. Based on reviews of technetium-99 in ILAW glass, DOE and the Washington State Department of Ecology (Ecology) agreed to delete technetium removal from the WTP permit (Hedges 2008). Construction of the Pretreatment Facility to date has eliminated the capability to remove technetium from the LAW stream. This *TC & WM EIS* assumes that technetium-99 removal could be completed in the existing Pretreatment Facility and analyzes it under Tank Closure Alternatives 2B and 3B. Design and construction modifications would need to be made to add the technetium-99 removal capability for these alternatives. Section E.1.2.3.10 has additional information on technetium removal within the WTP.

#### **E.1.2.3.1.2 Waste Treatment Plant High-Level Radioactive Waste Vitrification**

The baseline HLW vitrification process consists of two melter systems operated in parallel. Each melter system would be equipped with feed preparation vessels; a joule-heated, ceramic-lined melter; a canister-filling system; and primary offgas treatment. These systems would be supported by HLW melter feed vessel, canister welding, and decontamination systems; a condensate collection and recycle system; and secondary offgas treatment.

HLW feed concentrate would be transferred from the Pretreatment Facility to the HLW melter feed vessel. The feed would be blended with glass-forming chemicals and other additives and mixed to maintain the solids in suspension. The melter-feed slurry would then be transferred to the melter-feed vessels that provide continuous feed to the HLW melters.

The melter feed would be introduced at the top of the melter and would form a “cold cap” above the surface of the melt pool. The melt pool would be maintained at a temperature of approximately 1,050 to 1,150 degrees Celsius (°C) (1,900 to 2,100 degrees Fahrenheit [°F]). Water and volatile components in the feed would evaporate or decompose. The volatile components would be drawn off through the melter offgas system. Nonvolatile components would undergo chemical reactions to form oxides, which would then dissolve in the glass melt. The glass pool would be agitated to improve heat transfer and thereby increase the glass production rate.

HLW glass would be poured into stainless steel canisters (0.6 meters [2 feet] in diameter by 4.6 meters [15 feet] high) by means of an airlift system inside the melter. The glass would be poured in a series of “lifts” (layers) over a period of approximately 2 days for each canister. An infrared-level detector would monitor the fill height of glass in the canister. Each canister would hold approximately 3.2 metric tons of glass IHLW (assuming the use of the “thin-wall” canister). After being filled, the canister would be removed from the melter pour station and allowed to cool sufficiently to allow subsequent handling.

The filled canister would be transferred to a station within the facility, where it would be inspected and prepared for sealing. A lid would be welded to the top of the canister using a remotely operated, automated welding process. The IHLW glass canister is decontaminated using a nitric acid/cerium (IV) chemical milling process to remove a thin layer of the canister outer wall material and any contaminants adhering to it. The decontaminated canister would be swabbed remotely to confirm that any remaining removable contamination is below specification limits. Canisters that meet waste acceptance technical requirements for DOE’s Civilian Radioactive Waste Management System (DOE 2007) would be transferred to the onsite IHLW Interim Storage Modules.

Each HLW melter would have a dedicated primary offgas treatment system, including a film cooler, submerged-bed scrubber, wet electrostatic precipitator, high-efficiency mist eliminator, and HEPA filters to remove particulates and condensables, including entrained or volatilized radionuclides, from the melter offgas. Condensate from the scrubbers would be collected in the liquid effluent system and recycled to the Pretreatment Facility along with the neutralized canister decontamination solution.

After processing by the primary offgas treatment system, the offgas would be routed to a secondary offgas system. The secondary offgas system would destroy organics, reduce nitrogen oxides, and remove halides and mercury. The major system components would include a thermal catalytic oxidizer (oxidizes volatile organic compounds [VOCs]), selective catalytic reducer (converts nitrogen oxides to nitrogen and water), silver mordenite column (removes iodine-129 and other halogens), and sulfur-impregnated activated-carbon bed (removes mercury) (BNI 2005).

Each HLW melter would have a TMC of 3 metric tons of glass IHLW per day, or a total of 6 metric tons of glass IHLW per day for the two melters combined. The actual glass production rates would be somewhat lower based on the operating efficiency of each facility. The operating efficiency of the Pretreatment Facility is included in this analysis because the availability of pretreated waste feed is an important factor in determining overall plant operating efficiency.

The analyses assumed there would be sufficient overall availability to support a throughput per HLW melter of 2.5 metric tons of glass IHLW per day, or a combined total of 5 metric tons of glass IHLW per day for two melters, for a total operating efficiency (TOE) of 83 percent (DOE 2003f). The total amount

of IHLW glass produced and the duration of the processing campaigns to treat all retrieved waste would vary based on the alternatives.

### **HIGH-LEVEL RADIOACTIVE WASTE VITRIFICATION FACILITY**

The HLW Vitrification Facility would house all of the processing systems and equipment for producing IHLW glass from pretreated HLW feed. The HLW Vitrification Facility would be designed for remote maintenance of contaminated equipment.

The building ventilation system would be designed to cascade the supply air from areas with low potential for contamination into areas with a higher potential for contamination. The ventilation discharge for the contaminated areas would be filtered before release.

The HLW Vitrification Facility is constructed of reinforced concrete and structural steel. The below-grade portions, structural walls, process areas, and radiation areas are constructed of reinforced concrete. The superstructure is made of structural steel with external metal siding and a metal roof. Facility structural appendages are supported by reinforced-concrete foundations.

The process cells within the HLW Vitrification Facility are being constructed of reinforced concrete to protect facility workers from radiation. The cell floors and a portion of the walls are being lined with stainless steel capable of lifelong decontamination. The cell liners also act as secondary containment, providing a barrier for potential leakage from process tanks and equipment. The HLW Vitrification Facility is designed to allow replacement of the melters, which have an expected service life of 5 years each.

#### **E.1.2.3.1.3 Waste Treatment Plant Low-Activity Waste Vitrification**

The baseline WTP LAW vitrification process would consist of two large-scale melter systems operated in parallel. Each melter system would have a set of feed-preparation vessels; a large-capacity, joule-heated, locally shielded, ceramic melter; an ILAW product container-filling system; and a primary offgas treatment system. The following description applies to each of the LAW melter systems.

Pretreated LAW feeds would be received in one of two common LAW concentrate-receipt vessels inside the LAW vitrification building. Batches of concentrated LAW feed would be transferred from these vessels to melter feed preparation vessels, where glass formers would be added and blended to form a uniform batch for the LAW melter. The melter feed would be a slurry and would be continuously agitated. The slurry feed would then be transferred to the melter feed vessels, where it would be fed continuously to the LAW melters.

The melter feed would enter the melter from the top and form a cold cap above the melt pool. The melt pool would be maintained at a temperature of approximately 1,050 to 1,150 °C (1,900 to 2,100 °F). Water and volatile components present in the feed would evaporate or decompose. The volatile components would be drawn off through the melter offgas treatment system. The nonvolatile components would undergo chemical reactions to form oxides or other compounds that dissolve in the glass melt. Bubblers would be used to agitate the glass pool to improve heat transfer and thereby increase the glass production rate. The glass would be poured from the melter by an airlift system into stainless steel containers 1.22 meters in diameter by 2.3 meters high (4 feet in diameter by 7.5 feet high). Each container would hold approximately 6 metric tons of glass ILAW.

After being filled, each ILAW glass container would be allowed to cool before being sealed and decontaminated. Samples of the ILAW glass would be taken as required and analyzed to confirm that the ILAW glass meets specifications. Inert fill would be added, if necessary. A lid would be attached to the top of the container using equipment remotely operated from outside the cell. External contamination

would be removed using a carbon dioxide pellet decontamination process. A remote-controlled container swabbing system would be used to confirm that residual surface contamination levels are below limits for transport to the disposal site. Sealed and decontaminated containers of ILAW glass would then be loaded on transporters for transfer to an IDF.

Each LAW melter system would be equipped with its own primary offgas equipment, including a film cooler, submerged-bed scrubber, and wet electrostatic precipitator. Particulates and condensables, including radionuclides volatilized or entrained in the melter offgas stream, would be captured in the submerged-bed scrubber and wet electrostatic precipitator. Condensate from the scrubber and the precipitator would be collected in the liquid effluent system and recycled to the Pretreatment Facility.

The offgas from the primary offgas treatment systems would be combined with offgas from the vessel ventilation system and passed through a secondary offgas treatment system. The secondary offgas treatment system would provide final filtration, destroy organics, reduce nitrogen oxides, and remove halides and mercury. This would be accomplished by using HEPA filters, activated carbon absorbers (mercury; halides; and acid gases, including iodine-129), a catalyst skid (to remove nitrogen oxides, carbon monoxide, and VOCs), thermal catalytic oxidizers (to oxidize VOCs and carbon monoxide to carbon dioxide and water), a selective catalytic reducer (to reduce nitrogen oxides in offgas to nitrogen and water), and a caustic scrubber (to remove acid gases such as sulfur dioxides) (BNI 2005).

Each LAW melter would have a TMC of 15 metric tons of glass ILAW per day, or a total of 30 metric tons of glass ILAW per day for the two melters. The actual glass production rates would be somewhat lower based on the operating efficiency of each facility. The operating efficiency of the Pretreatment Facility is included in this analysis because the availability of pretreated waste feed is an important factor in determining overall plant operating efficiency.

The analyses assumed there would be sufficient overall availability to support a throughput per LAW melter of 10.7 metric tons of glass ILAW per day, or a total of 21.4 metric tons of glass ILAW per day for two melters, 32.1 metric tons of glass ILAW per day for three melters, and 64.2 metric tons of glass ILAW per day for six melters, for a TOE of 71 percent (DOE 2003f). The total amount of ILAW glass produced and the duration of the processing campaigns to treat all retrieved waste would vary based on the alternatives for which these options are used. Total ILAW glass production in the WTP is addressed in Section E.1.2.4.

#### **LOW-ACTIVITY WASTE VITRIFICATION FACILITY**

The WTP LAW Vitrification Facility would house all of the processing systems and equipment for production of ILAW glass from pretreated LAW feed. The LAW feed receipt and melter feed preparation vessels would be located within a shielded concrete cell lined with stainless steel for containment and ease of decontamination. The vitrification portion of the facility would be designed to facilitate local melter maintenance and reduce reliance on remote maintenance operations. Each melter would be equipped with a steel shielding box. The locally shielded melters would be located in a large melter gallery area. This arrangement allows operators to perform routine maintenance (such as bubbler-tube replacement) locally, using specially designed maintenance equipment to control contamination spread and worker exposure. Along with this melter confinement design, the glass-pour cell arrangement would be designed to provide a container turntable and a container elevator system to place containers in position for glass filling. This approach would provide the needed residence time for container cooling and improve container contamination control.

The building ventilation system is designed to cascade the supply air from areas with a low potential for contamination into areas with a higher potential for contamination. The ventilation discharge for the contaminated areas would then be filtered before release.

The LAW Vitrification Facility building is constructed of reinforced concrete and structural steel. The below-grade portion of the building structure is reinforced concrete, and the superstructure is made of reinforced concrete and structural steel with exterior metal siding and a metal roof. The facility appendage structure is supported by reinforced-concrete foundations.

The process cells within the LAW Vitrification Facility are constructed of reinforced concrete to protect facility operators from radiation. A protective coating would be applied to the concrete floor and walls of the LAW melter gallery to aid in decontamination activities, if necessary. The cell and cave floors and a portion of the walls are being lined with stainless steel capable of lifelong decontamination to provide a leakage barrier for the process tanks. The LAW Vitrification Facility is designed to allow periodic replacement of the melters, which is expected every 5 years.

#### **E.1.2.3.1.4 Expanded Low-Activity Waste Vitrification**

The LAW Vitrification Facility was originally designed to produce 30 metric tons of glass ILAW using three melters. Improvements in melter technology have demonstrated a vitrification capacity of 30 metric tons of glass per day can be achieved with two melters. Construction of the LAW Vitrification Facility is proceeding with two melters with a TMC of 30 metric tons of glass ILAW per day.

Additional LAW vitrification capacity needed to accelerate treatment of the tank waste is addressed using two approaches in this *TC & WM EIS*. The first approach is the installation of additional melter capacity in the LAW Vitrification Facility currently under construction as part of the WTP, bringing the total design capacity from a TMC of 30 metric tons of glass ILAW per day to a TMC of 45 metric tons of glass ILAW per day (Tank Closure Alternative 5). The installation of additional melter capacity in the existing LAW Vitrification Facility, though technically possible, would require design modifications for additional infrastructure tie-ins and enhanced cooling capacity. The second approach includes the installation of this additional melter capacity in the existing LAW Vitrification Facility, as well as construction of a second LAW Vitrification Facility, for a TMC of 90 metric tons of glass ILAW per day (Tank Closure Alternatives 2B, 6B, and 6C).

Note that these alternatives are described in terms of their TMC. For analysis purposes, this *TC & WM EIS* assumes a nominal throughput rate of 10.7 metric tons of glass ILAW per day for one LAW melter and 21.4 metric tons of glass ILAW per day for two melters combined.

The process for expanded WTP LAW vitrification would essentially be the same as that for the currently planned LAW vitrification. However, LAW pretreatment would be required to enable the LAW Vitrification Facility to operate at or near its design capacity, which, in turn, means the facility would have a high TOE. This could be accomplished by fully utilizing available processing capacity, reducing sampling and analysis requirements, and minimizing downtime for maintenance and repairs.

The waste treatment processing timeframes are based on a 14 weight-percent sodium oxide loading in the ILAW glass, assuming the same nonwaste sodium additions as indicated in the *Waste Treatment and Supplemental Technology Data Package* (DOE 2003f), i.e., a maximum of 11,500 metric tons of nonwaste sodium additions. It is important to note that the quantity of nonwaste sodium added during pretreatment (e.g., caustic leaching of HLW solids) would affect the total quantity of containers of ILAW glass produced. In addition, the quantity of sulfate present in the LAW feed relative to sodium could reduce the allowable sodium oxide loading in the glass. Thus, combining expanded LAW vitrification with supplemental treatment technology such as sulfate removal could reduce both the amount of ILAW glass produced and the processing time required to treat all of the retrieved waste. Section E.1.2.3.1.7 discusses WTP assumptions and uncertainties.

## EXPANDED LOW-ACTIVITY WASTE VITRIFICATION FACILITIES

A second LAW Vitrification Facility of the same basic design as the currently planned LAW Vitrification Facility would be constructed on unoccupied land within the WTP complex. A location adjacent to the Pretreatment Facility and critical support facilities is preferred. Each LAW Vitrification Facility would receive pretreated LAW from the WTP Pretreatment Facility. Each facility would be supported by the Analytical Laboratory, as well as systems and utilities from an expanded BOF support infrastructure. Each facility would have a 40-year design life.

### E.1.2.3.1.5 Waste Forms/Disposal Packages

The vitrified IHLW form would consist of borosilicate glass in a sealed and decontaminated stainless steel canister. The glass would be poured from the melter by means of an airlift system into stainless steel canisters 0.6 meters in diameter by 4.6 meters high (2 feet in diameter by 15 feet high). Each canister would hold approximately 3.2 metric tons of IHLW glass.

The vitrified ILAW form would consist of borosilicate glass in a sealed and decontaminated stainless steel container. The glass would be poured from the melter by means of an airlift system into stainless steel containers 1.2 meters in diameter by 2.3 meters high (4 feet in diameter by 7.5 feet high). Each container would hold approximately 6 metric tons of ILAW glass.

### PRIMARY WASTE

The primary-waste streams would follow two distinct pathways. The IHLW glass (as vitrified by the WTP) would be stored on site until disposition decisions are made and implemented. Section E.1.2.4 discusses the post-treatment storage and disposal and product performance characteristics for the IHLW glass.

ILAW glass (as vitrified by the WTP) would be disposed of on site except under Tank Closure Alternatives 6B and 6C, wherein the ILAW glass would be managed as HLW. Section E.1.2.4 discusses the disposal options and product performance characteristics for the ILAW glass, as well as the quantities of primary waste generated by the WTP under the various alternatives analyzed.

### WASTE FORM PERFORMANCE

The DOE Office of Civilian Radioactive Waste Management has established the performance requirement for IHLW. The chemical durability (i.e., leach resistance) of the IHLW glass must be equal to or better than that for the reference glass. The reference glass is borosilicate glass prepared in support of the *Defense Waste Processing Facility Environmental Assessment* and commonly referred to as “Savannah River EA [environmental assessment] glass” (DOE 2007).

The long-term impacts of ILAW glass disposal were analyzed in the *Hanford Immobilized Low-Activity Waste Performance Assessment: 2001 Version* (Mann et al. 2001). This document was reviewed by Ecology and approved by DOE Headquarters. A wide variety of cases was analyzed.

The rate of release of hazardous constituents depends strongly on the nature of the waste form used to immobilize the constituents. The nature of those waste forms, an analysis of their long-term performance, and methods used to estimate and characterize release rates from the ILAW glass are presented in Appendix M.



## **SECONDARY SOLID WASTE**

Secondary solid waste would be generated by the WTP in every area where waste processing takes place. Tank farm waste would be pumped to the Pretreatment Facility, where it would be subjected to a number of processing steps, including ultrafiltration, ion exchange, and mixing. As a result of these activities, fouled or worn ultrafilter membranes and spent ion exchange resins would be produced as process expendables. Additionally, spent HEPA cartridges would be produced by the process vent systems, and a number of pretreatment maintenance wastes would be produced, such as worn pipe jumpers, failed mechanical or instrumentation equipment, and degreasing or decontamination materials. The resultant solid waste would be CH or RH, as applicable, and would be placed into either drums (HEPA cartridges, failed equipment, decontamination materials) or boxes (ultrafilter modules, pipe jumpers) or sluiced into high-integrity containers and dewatered (used ion exchange resins). After the filtration and maintenance wastes were securely packaged, they would be transferred to the Central Waste Complex (CWC) for stabilization, if necessary, and repackaging. The stabilized waste would then be transported to an IDF for disposal (BNI 2005).

HLW melter feed streams would be sent to the HLW Vitrification Facility for processing in the HLW melters. Vitrifying these HLW streams would produce contaminated secondary waste. Solid secondary waste produced by the melters would include expended-melter bubblers and thermowells. In addition, from each melter line one melter would be taken out of service every 5 years. Operation of the melters would produce offgases that would pass through a number of offgas treatment system components and result in the production of contaminated waste byproducts. The HLW melter offgas treatment system would produce solid waste such as spent HEPA filters, spent sulfur-carbon adsorbent, spent silver mordenite adsorbent, and fouled thermal catalyst and selective reduction catalyst modules. HLW Vitrification Facility maintenance operations would produce decontamination material, worn or fouled pipe jumpers, and failed equipment. Additionally, spent HEPA filters would be discharged from the vessel pulse jet vent system. Solid waste is RH and would be placed into drums or boxes. After the filtration, sorption, and maintenance wastes were securely packaged in accordance with the HSSWAC, they would be transferred to the CWC for stabilization, if necessary, and repackaging. The stabilized waste would then be transported to an IDF for disposal. Packaging and transport of the HLW melters taken out of service are described in Section E.1.2.4.4 (BNI 2005).

As a result of processing waste through the LAW melter system, solid and liquid secondary waste would be produced. Melter operations would produce a number of waste products. Offgases from operation of the melters would pass through a number of the offgas treatment system components identified in Section E.1.2.3.1.3, resulting in the production of contaminated solid and liquid waste byproducts. Maintenance of the LAW Vitrification Facility would result in additional contaminated solid waste. The LAW vitrification process would produce expended-melter bubbler/thermowell waste and, periodically, melters taken out of service. This *TC & WM EIS* assumes the melters would require replacement every 5 years. Operation of the melter offgas and vent system would produce loaded or occasionally failed HEPA filters, expended sulfur-carbon adsorbent modules, fouled or damaged scrubber packing, fouled thermal catalytic modules, and fouled selective catalytic reduction modules. LAW Vitrification Facility maintenance operations would produce decontamination materials, worn or fouled pipe jumpers, and failed process equipment. Spent HEPA filters would also be produced during operation of the process vessel vent system. The solid waste would be RH and would be placed into drums or boxes. After the filtration, sorption, and maintenance wastes were securely packaged in accordance with the HSSWAC, they would be transferred to the CWC for stabilization, if necessary, and repackaging. The stabilized waste would then be transported to an IDF for disposal. Packaging and transport of the LAW melters taken out of service are described in Section E.1.2.4.4 (BNI 2005).

## SECONDARY LIQUID WASTE

The WTP liquid effluent collection system would consist of two subsystems: the radioactive liquid disposal system and the nonradioactive liquid disposal system.

The radioactive liquid disposal system would receive effluent waste from various sources within the LAW Vitrification Facility, including the melter primary offgas treatment system, LAW vitrification secondary offgas/vessel vent treatment system, processes, vessel washes, floor drains and sumps, and vessel vent header drains, all of which are expected to have low levels of radioactive contamination. Waste from the HLW Vitrification Facility areas also would be received, including the primary and secondary offgas treatment and HLW canister decontamination handling system, vessel washes, offgas drains, and the floor drains and sump system. In addition, the radioactive liquid disposal system would receive waste from Pretreatment Facility sources such as the radioactive liquid disposal vessels and tanks that are primarily used to collect solutions that may be discharged to the LERF/ETF, including process condensates from evaporators, caustic waste from the LAW caustic scrubber, and spent reagents from the resin addition process (BNI 2005).

The nonradioactive liquid disposal system for all three WTP facilities would receive effluent from the floor drains and sump systems that is not expected to be radioactively contaminated. The purpose of the system is to receive effluents for interim storage, to sample them for radioactivity, and, after the contents have been determined to be free of radioactive contamination, to transfer the effluents to the BOF for ultimate disposal at the TEDF. If radioactive contamination is detected, the contents would be transferred to the Pretreatment Facility for processing (BNI 2005).

For *TC & WM EIS* analysis purposes, it was assumed that solid waste would be treated (e.g., by grouting or encapsulation) as needed to meet the IDF waste acceptance criteria. Liquid effluents would be routed to the LERF/ETF or the TEDF, as appropriate. Solid waste generated at the ETF would be treated as needed to meet the IDF waste acceptance criteria. Additional discussion of secondary-waste streams and their disposal is included in Section E.1.2.4.5.

### E.1.2.3.1.6 Other Waste Treatment Plant Facilities

#### ANALYTICAL LABORATORY

The Analytical Laboratory would be a standalone facility east of the Pretreatment Facility and north of the LAW Vitrification Facility. The laboratory would analyze chemical and radiochemical samples to support WTP operations. The laboratory would be capable of processing and analyzing nonradioactive samples to support the WTP glass-forming chemical supply operation. The reserve capacity of the laboratory would be used to conduct limited and infrequent technology testing. Environmental samples (including effluent) would be sent to other laboratories for analysis.

The laboratory would contain hot cells with shield walls and remote manipulators to prepare the highly radioactive samples. The laboratory would also have chemical and radiochemical analysis modules, administrative areas, and equipment rooms. A cascading ventilation system would be used to ensure air movement from low-contamination areas to high-contamination areas. The exhaust from the radioactive areas would be filtered. Samples would be transported to the laboratory via a remote pneumatic system or manual transport inside shielded containers. Radioactive liquid waste from laboratory operations would be transferred to the Pretreatment Facility for processing.

The laboratory would be a steel frame building built on a reinforced-concrete foundation with exterior metal siding and a metal roof.

**BALANCE OF FACILITIES**

The BOF would include the facilities, support systems, and utilities that make up the WTP site infrastructure. The BOF support systems and utilities would include electric power, heating and cooling, process steam, process water, chilled water, compressed air, fire protection water, potable water, sanitary sewer, and storm drains.

Some of the larger support BOF would include the glass-former storage silos and glass-former handling and transport facility, the melter assembly building, the expended-melter storage areas, wet-chemical storage, and a storage facility for secondary waste from WTP processing.

Currently, WTP is planning to use diesel fuel during operations for facility heating and cooling, emergency power, and process steam. As an example of the potential reduction in air emission releases caused by the switch from diesel fuel to natural gas, a comparison was prepared using Tank Closure Alternative 2A. This comparison is for a 2 × 2 WTP configuration that is estimated to use approximately 5.3 billion liters (1.4 billion gallons) of diesel fuel over the estimated operating period of 75 years. Replacing this volume of diesel fuel with an estimated volume of 5.44 million cubic meters (192 million cubic feet) of natural gas would reduce facility air emissions. This reduction is shown in Table E-5.

**Table E-5. Comparison of Annual Criteria Pollutant and  
Toxic Pollutant Emissions for the WTP (2 × 2 Configuration)**

Constituent	Unit	Diesel Fuel	Natural Gas	Reduction
Carbon monoxide	tons/year	1.11×10 <sup>2</sup>	6.62×10 <sup>1</sup>	4.51×10 <sup>1</sup>
Nitrogen dioxide	tons/year	2.41×10 <sup>2</sup>	3.14×10 <sup>1</sup>	2.09×10 <sup>2</sup>
PM <sub>10</sub>	tons/year	3.50	3.35	1.47×10 <sup>-1</sup>
Sulfur dioxide	tons/year	1.08×10 <sup>1</sup>	9.59	1.18
Carbon dioxide	tons/year	1.59×10 <sup>5</sup>	3.35×10 <sup>3</sup>	1.56×10 <sup>5</sup>
VOCs	tons/year	4.05×10 <sup>2</sup>	6.30×10 <sup>1</sup>	3.42×10 <sup>2</sup>
Ammonia	tons/year	1.67	2.52×10 <sup>-2</sup>	1.65
Benzene	tons/year	1.05×10 <sup>-2</sup>	1.68×10 <sup>-3</sup>	8.86×10 <sup>-3</sup>
1,3-Butadiene	tons/year	3.81×10 <sup>-6</sup>	7.78×10 <sup>-7</sup>	3.03×10 <sup>-6</sup>
Formaldehyde	tons/year	1.18×10 <sup>-4</sup>	2.63×10 <sup>-5</sup>	9.14×10 <sup>-5</sup>
Toluene	tons/year	2.97×10 <sup>-4</sup>	2.58×10 <sup>-4</sup>	3.88×10 <sup>-5</sup>
Xylene	tons/year	9.99×10 <sup>-4</sup>	7.28×10 <sup>-4</sup>	2.71×10 <sup>-4</sup>

**Key:** PM<sub>10</sub>=particulate matter with an aerodynamic diameter less than or equal to 10 micrometers; VOC=volatile organic compound; WTP=Waste Treatment Plant.

**Source:** CEES 2010a.

DOE is currently considering whether to pursue an alternate fuel source for the WTP and 242-A Evaporator. A meeting was held in April 2010 between DOE and Cascade Natural Gas, an intrastate gas supplier, to discuss feasibility of a natural gas pipeline to the 200-East Area that would provide natural gas to the WTP and the 242-A Evaporator. The tentative routing of the pipeline would include boring under the Columbia River just south of the 300 Area and connecting with the existing northwest pipeline just north of the Tri-City Airport. One line would be provided to the WTP and another would be provided to the 242-A Evaporator. DOE would conduct the appropriate National Environmental Policy Act (NEPA) analysis if a decision is made to pursue this action.

**E.1.2.3.1.7 Waste Treatment Plant Assumptions and Uncertainties**

The capture of several select radionuclides in the ILAW glass is an important consideration when evaluating the performance of the ILAW process. The high temperatures associated with the ILAW process would cause volatilization of some of the select radionuclides, emitting offgases that would ultimately be captured in secondary-waste streams. The estimated fractions of select radionuclides between final waste form product and secondary-waste streams as a percent of the feed stream to the ILAW process are provided in Table E-6.

**Table E-6. Partitioning Factors of Radioactive Constituents of Potential Concern in Immobilized Low-Activity Waste Glass<sup>a</sup>**

Radioactive Constituent of Potential Concern	Percent of ILAW Feed in ILAW Glass	Percent of ILAW Feed in Liquid Secondary Waste
Iodine-129 <sup>b</sup>	20.0	80.0
Cesium-137	99.8	0.4
Carbon-14	0.0	0.3
Uranium-233	99.7	0.4
Uranium-234	99.7	0.4
Uranium-235	99.8	0.4
Neptunium-237	99.8	0.2
Uranium-238	99.8	0.4
Plutonium-239	99.8	0.2
Plutonium-240	99.8	0.2
Hydrogen-3 (tritium)	0.0	99.0
Strontium-90	99.5	1.5
Technetium-99	98.5	1.8

<sup>a</sup> Totals may not equal 100 percent due to rounding.

<sup>b</sup> Partitioning factor was directed by the Office of River Protection (CEES 2007:Attachment 4).

**Key:** ILAW=immobilized low-activity waste.

**Source:** Data extracted from CEES 2007.

The behavior of iodine-129 in thermal processes and the fraction that would be captured in the final waste form are difficult to predict. Therefore, this fraction was assumed, based on ORP guidance (CEES 2007:Attachment 4), in contrast to the use of the HTWOS simulations partitioning factors for the other radioactive constituents of potential concern (COPCs). Further demonstration and testing of this technology should provide the necessary performance data to confirm these assumptions. If necessary, design changes may have to be implemented should the actual fractions in secondary-waste streams be demonstrated to be higher than anticipated. Carbon-14 is highly volatile and would produce very high fractions in the offgas. Incorporation of treatment technologies in the final facility design may be required to reduce these emissions. Select radionuclides that exist in particulate form would be recycled into the waste feed. Appendix N, Section N.5.8, provides the results of a sensitivity analysis of additional retention of iodine-129 in ILAW glass.

The following basis and assumptions related to the WTP baseline were also used for analysis purposes in this *TC & WM EIS*:

- A 99 percent retrieval of the tank waste could be achieved using currently available retrieval technologies. A retrieval rate of 99.9 percent could be achieved using advanced retrieval technologies (e.g., chemical washing) with the currently available retrieval technologies.
- The Pretreatment Facility would be capable of providing sufficient feed to support the full operation of both the HLW Vitrification Facility and two LAW Vitrification Facilities simultaneously.
- This *TC & WM EIS* assumes the WTP would process approximately 60,000 metric tons of sodium combined from tank waste (approximately 48,000 metric tons) and additions made during pretreatment (12,000 metric tons). The precise quantity of sodium to be processed is uncertain because of the developing process flowsheets for the pretreatment and treatment technologies. The amount of sodium processed influences the time required for waste treatment through the WTP and the amount of IHLW and ILAW produced.
- ORP has been reviewing the WTP design. One of the areas assessed in detail is related to the amount of sodium hydroxide (caustic) that may need to be added as a process reagent. Sodium hydroxide is added in the pretreatment process to solubilize aluminum. This process is referred to as “caustic leaching.” The net result of the evaluation of the flowsheet model indicates that increases in the overall sodium quantity utilized in the caustic leaching process are necessary, and these would cause corresponding increases in the sodium disposed of as LAW. Early indications are that the total sodium requiring disposal as LAW may increase from 60,000 to 90,000 metric tons. To address this issue, DOE is evaluating various R&D tasks to minimize this impact. Potential solutions include recycling sodium hydroxide, maintaining the treatment process from ultrafiltration through LAW immobilization at elevated temperature, and treating or separating the aluminum within the waste tank prior to sending it to the WTP.
- Due to uncertainties with how well caustic leaching works for different types of waste, more sodium may be required to limit the number of IHLW glass canisters produced. If the total sodium value becomes as high as 90,000 metric tons, there could be an increase in the number of ILAW glass canisters by 46,000 based on the current flowsheet assumptions used for this EIS. Additionally, the waste treatment processing timeframes for the WTP could be extended by as much as 50 percent, or 13 years. However, the higher sodium volume would reduce the contaminant concentrations in the WTP annual air emissions (Burandt 2007).
- The WTP vitrification facilities would begin vitrifying waste in 2018.
- The effects of potential improvements in IHLW glass loading were incorporated by relaxing three glass-property model constraints. Glass loading to these limits may not be achievable without additional research and testing. However, these limit changes indicate that a reduction in the number of IHLW glass canisters is achievable with a positive effect on the RPP program. The maximum allowable viscosity of the IHLW glass was increased from 5.5 pascal-seconds to 10 pascal-seconds to reduce excessive convection currents during melting operations, thereby reducing corrosion/erosion of the melter materials and achieving better overall control of the HLW melter. The maximum allowable chromium oxide loading was increased from 0.5 percent to 1.0 percent. The maximum allowable spinel liquidus temperature was increased from 1,050 °C to 1,100 °C (1,922 °F to 2,012 °F) (DOE 2003g).

- The IHLW glass canisters would be “thin-wall” canisters, with a capacity of approximately 3.2 metric tons of glass IHLW per canister. Although this canister has not been approved, it was assumed it would be approved in the future. The currently approved Hanford canister would have a capacity of approximately 3.06 metric tons of glass IHLW per canister, or approximately 5 percent less (DOE 2003g). Under Tank Closure Alternative 2B, a change from the thin-wall canister would result in approximately 600 additional IHLW glass canisters, i.e., from 12,000 to 12,600.
- RPP supporting facilities (e.g., the LWPFs and the 242-A Evaporator) would be available as required for the duration of the waste treatment mission.

The following considerations contribute uncertainty to the analysis:

- Assumed nominal throughputs have been demonstrated only on pilot-scale facilities.
- Construction of a second LAW Vitrification Facility would require substantial resources. Skilled workers, equipment, and production capability might not be available, resulting in schedule delays and added costs.
- The ability of the RPP interfacing facilities to support expanded LAW vitrification capacity is uncertain.
- Upgrades to the Analytical Laboratory and BOF may be needed to support operation of expanded LAW Vitrification Facilities.

#### **E.1.2.3.2 242-A Evaporator**

The continued operation of the 242-A Evaporator is required to support treatment of tank waste. The current and future mission of the evaporator is to support environmental restoration and remediation of Hanford by optimizing the 200 Area DST waste volumes in support of the tank farm management and WTP operations. To accomplish this mission, the 242-A Evaporator would require multiple replacements under some Tank Closure alternatives analyzed in this EIS. The 242-A Evaporator’s estimated useful life is 25 years. The 242-A Evaporator also depends on the continued operation of the LWPFs to accept and treat both contact (process condensate) and noncontact (steam condensate and cooling water) effluent waste streams. The LWPFs are discussed in detail in Section E.1.2.3.3.

Under Tank Closure Alternative 1, the 242-A Evaporator would be utilized to support the tank farm management program until failure of the facility, with no planned replacement. Under Tank Closure Alternative 6A, the operational lifespan of the evaporator was assumed to extend from the present until approximately 2163. This would require the construction of six replacement evaporators (one each in 2018, 2043, 2068, 2093, 2118, and 2143). Under Tank Closure Alternative 2A, two replacement evaporators are required, one in 2040 and one in 2065. Under all other Tank Closure alternatives, only one replacement evaporator was assumed to be required.

The 242-A Evaporator is not operated continuously. Based on historical operating data, it was assumed the 242-A Evaporator would process approximately 3.78 million liters (1 million gallons) of liquid tank waste per campaign and would complete two campaigns per year, each lasting only about 21 days. The following background, process, and facility descriptions are based on the *Documented Safety Analysis for the 242-A Evaporator* (Campbell 2004).

#### **E.1.2.3.2.1 Background**

The existing 242-A Evaporator was constructed from 1974 to 1977 and is located in the 200-East Area of Hanford. The 242-A Evaporator began operations in 1977. Between 1977 and the late 1980s, the 242-A Evaporator's mission was to support ongoing defense-related production of nuclear weapons material, to concentrate and transfer SST waste into the DST system, and to manage DST waste by liquid volume reduction.

The 242-A Evaporator was shut down in April 1989, pending a determination regarding whether the process condensate required management as a mixed waste. Ecology subsequently concluded that the process condensate stream is a mixed waste stream, and thus is a discharge regulated by Ecology. As a result, direct discharge to the 216-A-37-1 Crib was eliminated. The determination led to a 5-year shutdown of the 242-A Evaporator until the LERF basins were constructed for storing process condensate.

The LERF basins store process condensate prior to treatment in support of 242-A Evaporator operations. The ETF was constructed to reduce the concentrations of ammonia, residual organics, and dissolved radionuclides in the process condensate to levels that permit direct disposal of the treated liquid effluent to the State-Approved Land Disposal Site (SALDS). The LERF, ETF, and SALDS are all part of the LWPFs described in Section E.1.2.3.3.

During the shutdown, the 242-A Evaporator was extensively upgraded. The upgrades were completed in fiscal year 1993, and operations restarted in April 1994. The current and future mission of the 242-A Evaporator is to support environmental restoration and remediation of Hanford by optimizing the 200-East and 200-West Areas' DST waste volumes. To support this mission, a life extension study was prepared to identify the work scope needed to extend the facility life through 2016. This study was revisited in January 2001 due to a need for the facility through 2019. The upgrades completed in 1993 and the implementation of the work scope outlined in the life extension study are expected to extend the 242-A Evaporator's useful life through 2018, when it has been assumed the first replacement would be required.

#### **E.1.2.3.2.2 Process**

The 242-A Evaporator is designed to reduce waste volume and the number of DSTs required to store liquid waste generated at Hanford. The process uses a conventional, forced-circulation, vacuum evaporation system operating at low pressure (approximately 8,000 pascals [1.2 pounds per square inch]) and low temperature (approximately 50 °C [122 °F]) to concentrate radioactive waste solutions.

The liquid tank waste feed is pumped from a DST feed tank (241-AW-102) through an underground encased feed line to the 242-A Evaporator. Using a thermal-assisted evaporative process (steam), the waste feed is concentrated in vessel C-A-1 to achieve a specified volume reduction. The concentrated tank waste slurry is returned to the DST farm. The slurry can be directed to a specific DST via the tank farm transfer piping system. The tank farm transfer system is physically connected to the 242-A Evaporator in-facility process piping at the exterior walls of the 242-A Evaporator and consists of single and encased piping, pumps, valve pits, diversion boxes, clean-out boxes, and support systems such as leak detectors and cathodic protection. Process offgases and water vapor are passed through one primary and two secondary condensers, creating a process condensate and air emissions. Air emissions are filtered and released to the environment from the vessel ventilation exhaust system. Process condensate is collected in a collection tank prior to transfer to external treatment facilities.

The 242-A Evaporator process condensate, steam condensate, and cooling water effluent streams are transferred to other waste handling facilities. Process condensate (contact effluent) is transferred to LERF via an underground transfer line approximately 1.6 kilometers (1 mile) away. Effluent waste streams received in the LERF are eventually treated in the ETF and discharged in the SALDS. Monitoring for leak detection is provided at regular intervals along the transfer line. Steam condensate and cooling water (noncontact effluents) are transferred via separate underground transfer lines to the TEDF.

The average evaporator campaign processes 3.68 million liters (973,000 gallons) of waste to remove 2.17 million liters (573,000 gallons) of water and return 1.51 million liters (400,000 gallons) of slurry to the DSTs (CEES 2006a).

## **SECONDARY-WASTE STREAMS**

There are four secondary-waste streams from the evaporator. The process condensate is discharged to the LERF for blending and future treatment at the ETF at a maximum flow rate of 189 liters (50 gallons) per minute. The noncontact discharge effluents—steam condensate, cooling water, and facility wastewater—are discharged to the TEDF at a maximum flow rate of 680 liters (180 gallons) per minute, 14,000 liters (3,700 gallons) per minute, and 1,900 liters (500 gallons) per minute, respectively.

### **E.1.2.3.2.3 Facilities**

The 242-A Evaporator is located in the 200-East Area. Replacement evaporators were assumed to be similar in size and capacity to the existing 242-A Evaporator. Construction resources required for the replacement evaporators were scaled from WTP construction data. It was assumed that deactivation of the evaporators would require 6 months and that 1 percent of the waste processed through the evaporator in 1 year would remain after operations. The following four principal structures make up the 242-A Evaporator:

- 242-A Building: main process building
- 242-AB Building: control room
- 242-A-81 Building: water services building
- 207-A Retention Basins

The principal process components of the 242-A Evaporator system are located in the 242-A Building, which is composed of two adjoining but independent structures. The first structure contains processing and service areas and is a reinforced-concrete shear wall and slab structure. The second structure is separated from the first by a seismic joint and contains operating and personnel support areas.

The 242-A Building is 23 meters wide by 33 meters long (75 feet wide by 108 feet long). The 242-AB Building is an addition to the main structure of the 242-A Building, is of similar design and construction, and is 14 meters long by 12 meters wide (45 feet long by 40 feet wide). The 242-A Evaporator Facility is 33 meters long by 23 meters wide by 22 meters high (108 feet long by 75 feet wide by 72 feet high) (main building) and 14 meters long by 12 meters wide by 4 meters high (45 feet long by 40 feet wide by 12 feet high) (support building) for a total of 14,400 cubic meters (605,000 cubic feet). The facility boundary is assumed to encompass approximately 0.4 hectares (1 acre).

The principal process components of the 242-A Evaporator system include the reboiler, evaporator vessel C-A-1, recirculation pump, recirculation pipe loop, slurry product pump, condensers, jet vacuum system, and condensate collection tank.

A computer-based monitoring and control system is located in the 242-AB Building control room and is used to operate and monitor the 242-A Evaporator. The emergency diesel generator and the package boiler units are the only equipment that normally use diesel fuel within the facility boundaries. The



emergency diesel generator is run for 15 minutes a month as a safety check to ensure its operational capability. The package boilers provide normal building heating and steam to the reboiler units during a campaign. The package boilers use approximately 182,000 liters (48,000 gallons) of diesel fuel to heat the building during a normal calendar year. Additionally, they use approximately 295,300 liters (78,000 gallons) of diesel fuel during a campaign. Diesel fuel requirements are estimated by multiplying the projected number of campaigns per year (two) by 295,260 liters (78,000 gallons) and then adding 182,000 liters (48,000 gallons) for normal building heating.

Water is provided from the 200-East Area Power House to the 242-A-81 Water Services Building. Steam used in the 242-A Evaporator process and for building heat is provided by package boilers located in the 242A-AB Building.

The 207-A Retention Basins have been disconnected from the 242-A Evaporator and are no longer used. Prior to the temporary shutdown in 1989 and the subsequent construction of the ETF to treat process condensate, the 207-A Retention Basins were used for temporary storage of 242-A Evaporator process condensate and steam condensate prior to disposal. The basins contain low levels of residual radioactive contamination. This EIS does not analyze the decontamination and decommissioning (D&D) of the 207-A Retention Basins (CEES 2006a).

As discussed in Section E.1.2.3.1.6, DOE is considering switching from diesel fuel to natural gas at the 242-A Evaporator for facility operations such as heating the building and providing steam for waste campaigns. This EIS estimates that, under Tank Closure Alternative 2A, the evaporator would use approximately 69.3 million liters (18.3 million gallons) of diesel fuel over the estimated operating period of 88 years. Replacing this volume of diesel fuel with an estimated volume of 96,300 cubic meters (3.4 million cubic feet) of natural gas would reduce facility air emissions. This reduction is shown Table E-7.

**Table E-7. Comparison of Annual Criteria Pollutant and Toxic Pollutant Emissions  
for the 242-A Evaporator**

Constituent	Unit	Diesel Fuel	Natural Gas	Reduction
Carbon monoxide	tons/year	1.35×10 <sup>1</sup>	8.05	5.49
Nitrogen dioxide	tons/year	6.29×10 <sup>1</sup>	8.19	5.47×10 <sup>1</sup>
PM <sub>10</sub>	tons/year	4.42	4.23	1.86×10 <sup>-1</sup>
Sulfur dioxide	tons/year	2.14×10 <sup>-2</sup>	1.90×10 <sup>-2</sup>	2.35×10 <sup>-3</sup>
Carbon dioxide	tons/year	2.34×10 <sup>3</sup>	4.93×10 <sup>1</sup>	2.29×10 <sup>3</sup>
VOCs	tons/year	5.13	7.99×10 <sup>-1</sup>	4.33
Ammonia	tons/year	9.88×10 <sup>-2</sup>	1.49×10 <sup>-3</sup>	9.74×10 <sup>-2</sup>
Benzene	tons/year	1.33×10 <sup>-2</sup>	2.12×10 <sup>-3</sup>	1.12×10 <sup>-2</sup>
1,3-Butadiene	tons/year	1.68×10 <sup>-2</sup>	3.44×10 <sup>-3</sup>	1.34×10 <sup>-2</sup>
Formaldehyde	tons/year	5.57×10 <sup>-4</sup>	1.25×10 <sup>-4</sup>	4.33×10 <sup>-4</sup>
Toluene	tons/year	5.83×10 <sup>-3</sup>	5.07×10 <sup>-3</sup>	7.63×10 <sup>-4</sup>
Xylene	tons/year	4.06×10 <sup>-3</sup>	2.96×10 <sup>-3</sup>	1.10×10 <sup>-3</sup>

**Key:** PM<sub>10</sub>=particulate matter with an aerodynamic diameter less than or equal to 10 micrometers; VOC=volatile organic compound.

**Source:** CEES 2010a.

#### **E.1.2.3.2.4 Assumptions and Uncertainties**

The following is a list of the assumptions for construction, operation, and deactivation of evaporators as needed for replacement of the existing 242-A Evaporator (CEES 2006a). The 242-A Evaporator has been an operational facility since 1977; therefore, very little uncertainty is associated with the size of the facility, required process equipment, and production capacity.

- Evaporator facility life is estimated to be 25 years. Construction of a replacement evaporator and supporting facilities would take 3 years to complete.
- The capacity of the replacement evaporators would be similar to the production rates achieved in the 242-A Evaporator, averaging about 3.78 million liters (1 million gallons) per campaign and two campaigns per year, with each campaign lasting approximately 21 days. In practice, the 242-A Evaporator and its replacements could process more than two campaigns per year to support Hanford's needs; however, the air emissions assumed for that year would increase and the facility's useful life would likely decrease.
- The operating resources and emissions of the replacement evaporator are the same as the operating resources and emissions of the existing 242-A Evaporator.
- The TEDF and LERF are operating and available to receive the evaporator secondary-waste streams.
- Evaporator deactivation, including the removal of radioactive/hazardous materials to ensure adequate protection of workers, public health and safety, and the environment, would place the facility in a stable and known condition. Decommissioning and dismantling of the facility are not included in the scope of this *TC & WM EIS*. Deactivation would take place between campaigns and would take 6 months to complete.
- Evaporator deactivation assumes that 1 percent of 1 year's throughput would remain in the facility and it would take 6 months to clean up the facility with no campaigns during that period. Drain waste and flush waste would be returned to tank 241-AW-102 during deactivation activities.

#### **E.1.2.3.3 Liquid Waste Processing Facilities**

The 200 Area LWPFs include the LERF, ETF, SALDS, and TEDF. These facilities receive, store, treat, and dispose of liquid effluents from Hanford cleanup activities.

The LERF and ETF were constructed and commissioned circa 1995 to support 242-A Evaporator operations. The initial mission of the LERF was to store all 242-A Evaporator process condensate until it could be treated in the ETF. The LERF and ETF were constructed to support the determination that 242-A Evaporator process condensate was a mixed waste and could no longer be discharged directly into the ground without treatment.

The LERF, a series of three liquid storage basins, was initially constructed to provide permitted interim storage of the 242-A Evaporator process condensate. The LERF is capable of storing up to 4.9 million liters (13 million gallons) of liquid effluent waste awaiting treatment at the ETF. The effective storage capacity may be slightly lower due to recirculation and contingency space needs. All process condensate is received in the LERF prior to treatment in the ETF.

The ETF process reduces the concentration of contaminants, including ammonia, residual organics, and dissolved radionuclides, to levels that allow direct disposal of the treated liquid effluent in the SALDS. A byproduct of the ETF process, a concentrated, powdery solid-waste material, is placed in approved containers for transfer to storage and/or final disposal.

The LERF, ETF, and SALDS are designed to accept and treat regulated effluent discharges prior to disposal. These effluents are process condensates that have been in contact with radioactive tank waste.

Separate from the LERF and ETF is the TEDF, a collection and disposal facility for unregulated liquid effluents. These effluents are primarily steam condensates, cooling water, boiler blowdown, and rainwater that have not contacted tank waste and do not require treatment prior to disposal. The TEDF consists of approximately 17.7 kilometers (11 miles) of network piping, three pump stations, one sample station, and two 2-hectare (5-acre) disposal ponds (Fluor Hanford 2004a).

The alternatives in this *TC & WM EIS* depend on the continued operation of the 200 Area LWPFs, both for contact and noncontact effluent discharges. In fact, the ETF's role is expected to expand beyond treating liquid effluent generated from the operation of the 242-A Evaporator. Under this EIS, the LWPFs would also be required to support the WTP, bulk vitrification, steam reforming, and other processes, should they be selected for implementation.

For analysis purposes in this EIS, it was assumed that the ETF's main building 2025 and support building 2025-EA would require replacement to provide continued support for proposed treatment technologies selected under this EIS. However, the LERF (retention basin), TEDF (disposal pond), and SALDS (crib) are well suited to life extension projects; therefore, it was assumed replacement of these systems would not be required.

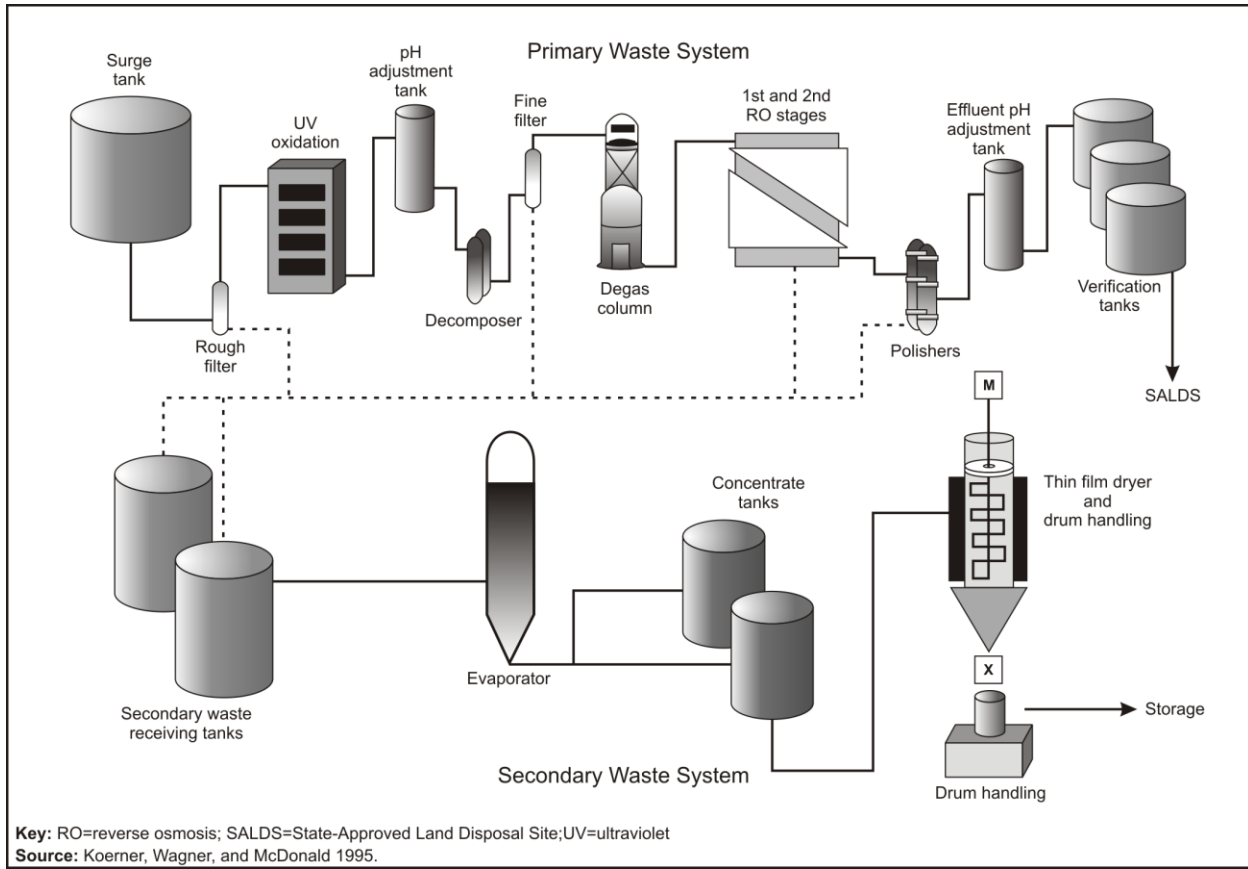
The design life of the ETF is 30 years, with the current ETF's useful life ending in 2025. Replacement of the ETF would occur every 30 years thereafter. Under Tank Closure Alternative 1, the ETF would be utilized to support the 242-A Evaporator until failure of the facility, with no planned replacements. Under Tank Closure Alternative 2A, the operational lifespan of the ETF was assumed to be from the present until approximately 2093. This would require the complete construction of two replacement facilities. Under Tank Closure Alternative 6A, the operational lifespan of the ETF was assumed to be from the present until approximately 2163. This would require the construction of five replacement facilities. All other Tank Closure alternatives would require only one replacement of the ETF.

#### **E.1.2.3.3.1 Process**

The ETF operation collects, treats, and stores MLLW. This includes the load-in station and transfer piping, as well as the tanks and process equipment within the ETF. The following is a brief discussion of the treatment processes available at the ETF. The ETF is an RCRA-permitted facility; as such, it can only receive liquid effluent meeting certain waste acceptance criteria and must successfully treat the liquid effluents prior to discharge. All effluent treated in the ETF is first interim-stored in the LERF basins. Maximum processing rates for the ETF are estimated to be 210 million liters (55.5 million gallons) per year based on a TOE of 70 percent (CEES 2006a). The following process steps are associated with the ETF treatment train as illustrated in Figure E-13 (Koerner, Wagner, and McDonald 1995):

- **Feed Receipt:** ETF feed is received in the 378,540-liter (100,000-gallon) surge tank. Recirculation mixes the surge tank contents continuously before they are introduced into the treatment process.

- **Suspended Solids Removal:** Removal of suspended solids is accomplished via a two-step process. Gross removal of suspended solids (greater than 2 micrometers) is performed before the ultraviolet oxidation system step. The second step, fine filtration (greater than 0.5 micrometers) is accomplished prior to introduction of the slurry into the degasification system. The removal of suspended solids greater than 0.5 micrometers is necessary for the protection of downstream equipment.
- **Organic Destruction:** Ultraviolet light oxidation destroys organics. The destruction process is enhanced with the use of hydrogen peroxide.
- **pH Adjustment:** The process effluent's pH (relative acidity or alkalinity) is controlled through the metered addition of sodium hydroxide or sulfuric acid. The pH is maintained at a value of approximately 4.0. Adjustment of pH is performed after ultraviolet oxidation and again after the reverse osmosis treatment process.
- **Hydrogen Peroxide Decomposition:** Hydrogen peroxide added during the destruction of organics in the ultraviolet oxidation system must be removed prior to the reverse osmosis process step. Hydrogen peroxide is decomposed using a granulated activated-carbon column that breaks down the hydrogen peroxide.
- **Degasification:** Removal of dissolved carbon dioxide is accomplished via degasification in a packed column. The reduction in carbonate loading is necessary for downstream process steps such as reverse osmosis and polishing.
- **Dissolved Solids Removal:** Removal of dissolved solids is provided by reverse osmosis. The effluent continues through to the polishing step. The concentrated liquid waste is redirected to additional secondary evaporation steps.
- **Effluent Quality Verification:** All treated effluent exits the polishers and is temporarily stored in one of three verification tanks. Each verification tank has a storage capacity of 2,536,200 liters (670,000 gallons) and is designed to hold 72 hours of effluent discharge. Prior to discharge to the SALDS, effluent may be retained for verification of quality. During storage, laboratory analyses may be performed to confirm compliance with the appropriate requirements before discharge. Effluent that does not meet discharge requirements is recirculated back to the surge tank for additional treatment.
- **Effluent Disposal:** Effluents suitable for disposal may be transferred through the liquid effluent disposal system for release to the SALDS.
- **Secondary-Waste Treatment:** Liquid waste from the primary treatment train may be concentrated in an evaporator and/or dewatered in the thin film dryer. The resultant powdery solid waste is packaged in drums for disposal (Koerner, Wagner, and McDonald 1995).



**Figure E-13. Effluent Treatment Facility Process Flowsheet**

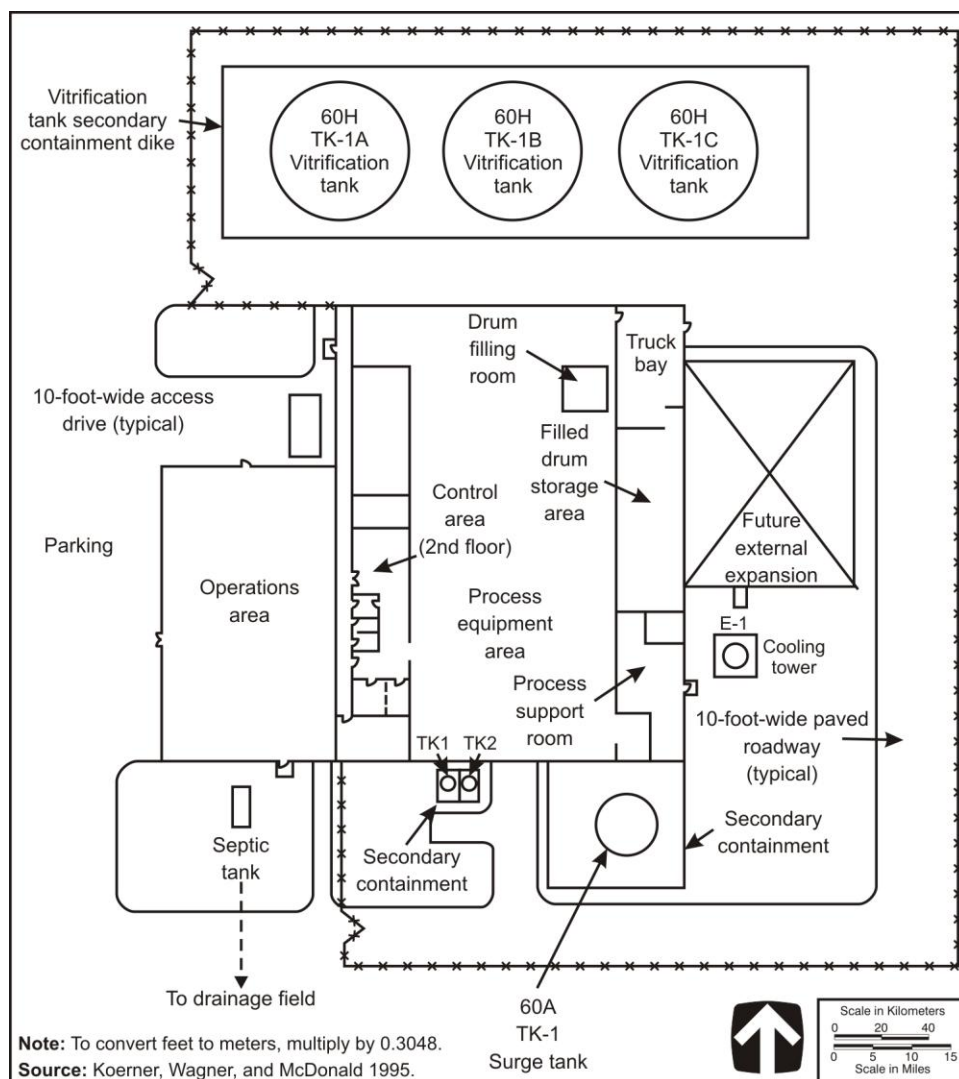
**PRIMARY- AND SECONDARY-WASTE STREAMS**

There are two significant waste streams associated with the ETF. The primary-liquid-waste stream is verified in the verification tanks and sent to the SALDS for final disposition. Three of the 2004 quarterly monitoring reports cited a discharge volume of 80.5 million liters (21.3 million gallons) for a 9-month period (Fluor Hanford 2004a, 2004b, 2005b). Based on these 2004 data and operations supported by the ETF, the annualized discharge rate in 2004 was approximately 107.1 million liters (28.3 million gallons). As analyzed in this EIS, the operation of the WTP, IDF(s), groundwater leachate collection systems, and supplemental treatment facilities could increase the waste loading on the ETF. The secondary-waste stream is a solid waste generated from the thin-film dryer. The powder and/or sludge solid-waste stream is packaged in 208-liter (55-gallon) drums and is directed to final disposition, depending on the source of the effluent that was processed. Waste from effluent that is a result of a Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) cleanup activity is sent to the Environmental Restoration Disposal Facility for disposal; LLW and MLLW from ongoing site activities would be sent to the currently operational lined trenches 31 and 34 in Low-Level Radioactive Waste Burial Ground (LLBG) 218-W-5 or to an IDF for disposal (CEES 2006b).

**E.1.2.3.3.2 Facilities**

The ETF is a permanent facility with an estimated operating design life of 30 years and has been classified as a radiological facility. The ETF is a single building containing a control room that provides the capability to control the treatment process and the collection, recycling, and disposal systems. External storage is provided for the treated waste during analysis and/or verification. The building also contains a chemical makeup and storage area, secondary-waste treatment and storage systems, and offgas

and ventilation systems. In addition, there are various support systems, including fire protection, communications, sanitary and raw water, and electrical. The ETF also includes a load-in station that provides the capability to unload, store, and transfer liquid waste to the ETF or LERF from tank trucks (Koerner, Wagner, and McDonald 1995). The ETF layout is presented in Figure E-14.



**Figure E-14. Effluent Treatment Facility Layout**

The ETF is in the northeast corner of the 200-East Area of Hanford, adjacent to the LERF and approximately 1 mile north of the 242-A Evaporator facility. The ETF's main building has a footprint of 4,100 square meters (44,000 square feet) and is 12 meters (40 feet) tall, while the support building's footprint is 950 square meters (10,300 square feet) and it is 3.7 meters (12 feet) tall; thus, the total footprint is 5,000 square meters (54,300 square feet). Therefore, each ETF replacement is based on constructing a 5,000-square-meter (54,300-square-foot) building. The facility boundary was assumed to occupy 29 hectares (72 acres) (Koerner, Wagner, and McDonald 1995).

### **E.1.2.3.3.3 Assumptions and Uncertainties**

- The ETF main building 2025 and the ETF support building 2025-EA would require replacement. The LERF (retention basin), TEDF (disposal pond), and SALDS (crib) are well suited to life extension projects. The LERF, TEDF, and SALDS would continue their useful life within the ETF system with the replacement ETF.
- The ETF's useful life is estimated to be 30 years. Construction of a replacement ETF and supporting facilities would take 3 years to complete.
- The current design capacity of the ETF would be sufficient to support all current Hanford activities, as well as the tank closure activities implemented under this EIS.
- The operating resources and emissions of the replacement ETF would be the same as the operating resources and emissions of the existing ETF.
- The LERF and SALDS would remain in operation and available to support ETF operations.
- All solid waste would be disposed of in either the Environmental Restoration Disposal Facility (for CERCLA waste), the currently operational mixed waste lined trenches, or an IDF. This solid waste would be grouted, containerized waste.
- ETF deactivation, including the removal of radioactive/hazardous materials to ensure adequate protection of workers, public health and safety, and the environment, would place the facility in a stable and known condition. Decommissioning and dismantling of the facility are not included in the scope of this *TC & WM EIS*. Deactivation would take place between campaigns and would take 6 months to complete.
- ETF deactivation assumes that 1 percent of 1 year's throughput would remain in the facility and that it would take 6 months to clean up the facility with no campaigns during that period. Drain waste and flush waste would be returned to the LERF during deactivation activities (CEES 2006b).
- ETF-generated solid waste nitrate concentrations were estimated from waste volumes and are considered conservative.

### **E.1.2.3.3.4 ETF Enhancements**

DOE recently began investigating solutions to potential issues regarding treatment and disposal of future WTP secondary waste. In June 2010, a presentation was made at the Waste Treatment Forum that provided some insight into potential solutions for treating secondary-waste streams generated at the WTP that would allow disposal in an IDF. One solution presented was to solidify these wastes prior to disposal by adding a solidification capability to the ETF. This is just one of many technologies being studied and, as of the time of publication of this EIS, a technology had not been down-selected for implementation. Only system-level functional, performance, design, interface, and test requirements were available; mass balance calculations and/or preliminary design information were not. Therefore, specific changes to the existing ETF were not evaluated in this EIS. Except Tank Closure Alternative 1, a minimum of one complete ETF replacement was evaluated under each of the Tank Closure alternatives; if implemented, these replacements are expected to bound the environmental impacts of a solidification capability at the ETF. Long-term performance and the potential long-term human health impacts related to disposal of secondary waste at an IDF are discussed in Chapter 7 of this EIS (CEES 2010b).

#### **E.1.2.3.4 Cesium and Strontium Capsule Processing Facility**

The construction of a capsule processing facility would be necessary for extracting cesium and strontium waste from the storage capsules prior to treatment through the WTP HLW melters. The capsule processing facility would also prepare the cesium and strontium waste into a slurry waste stream acceptable for treatment in the WTP. The final waste form would be canistered waste that is assumed to be stored on site as IHLW until disposition decisions are made and implemented.

Under Tank Closure Alternative 1, the cesium and strontium capsules would be stored indefinitely in the Waste Encapsulation and Storage Facility (WESF) in a manner similar to the present; therefore, construction of a capsule processing facility would not be required. Under all other Tank Closure alternatives analyzed in this EIS, the cesium and strontium waste would be vitrified in the WTP. The immobilization of cesium and strontium capsule waste would take place during a separate campaign, after treatment of all tank HLW is completed in the WTP. The cesium and strontium WTP campaign is expected to add 1 year of processing time to the WTP HLW melters. The capsule processing facility would be built so that processing of cesium canisters could begin approximately 14 months prior to the completion of WTP processing of tank HLW.

Based on estimated production rates, the capsule processing facility would require 26 months to de-encapsulate all the cesium and strontium capsules and prepare the cesium and strontium slurry feed. The WTP requires an estimated 12 months to vitrify the slurry feed. Thus, to maintain a continuous WTP feed, the capsule processing facility must begin operations 14 months in advance of the cesium and strontium campaign and store this WTP feed in the DSTs. It is estimated that an additional 340 canisters would be produced during the cesium and strontium treatment campaign (CEES 2006c).

##### **E.1.2.3.4.1 Background**

The cesium and strontium capsules were generated at Hanford during the 1970s and 1980s, when cesium and strontium isotopes were separated from other tank waste, converted to cesium chloride and strontium fluoride, and then encapsulated for long-term storage.

Currently, there are 1,335 cesium capsules and 601 strontium capsules stored in the WESF pool cells. Most of the capsules are composed of an inner and outer capsule. However, 23 of the cesium capsules are Type W capsules containing material from the 324 Building B cell. The Type W containers are triple-containment capsules versus the normal double-containment configuration (SAIC 2010a).

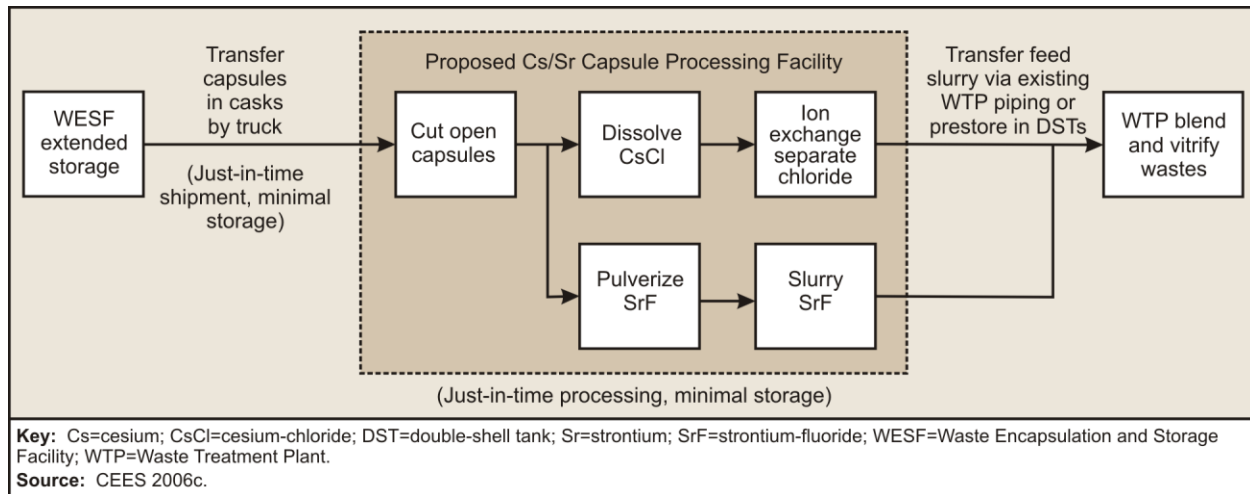
Originally, 640 strontium capsules were produced at the WESF; 601 are presently stored in the WESF pool cells. Of the 39 strontium capsules remaining and not stored at Hanford, only 4 capsules stored at the Nevada National Security Site (NNSS), formerly the Nevada Test Site, have a commitment for disposal by other DOE programs. Though there are no formal plans, the other 35 strontium capsules could be returned to the WESF; therefore, they are conservatively included in this EIS analysis for potential processing through the WTP (Claghorn 1996).

The cesium capsules are 6.67 centimeters (2.6 inches) in diameter and 51.1 centimeters (20.1 inches) long, and the strontium capsules are 6.67 centimeters (2.6 inches) in diameter and 52.8 centimeters (20.8 inches) long (Jeppson 1973). The capsules contain approximately 68 million curies of radioactivity, decayed to January 2000 (CEES 2006c).



### E.1.2.3.4.2 Process

The Cesium and Strontium Capsule Processing Facility would receive the capsules on a “just-in-time” basis for de-encapsulation and preparation of the waste into a suitable WTP slurry feed. The waste slurry would then be stored in a DST prior to treatment through the WTP. This EIS analyzes the immobilization of the cesium and strontium slurry feed as a separate, 1-year-long WTP campaign; however, the cesium and strontium slurry feed could be mixed with the late-stage tank waste feed for consistency. A simplified cesium and strontium capsule processing flowsheet is provided in Figure E-15.



**Figure E-15. Simplified Cesium and Strontium Capsule Processing Flowsheet**

#### TRANSPORTATION OF CAPSULES FROM THE WESF TO CAPSULE PROCESSING FACILITY

Transportation from the WESF load-out area to the capsule processing facility would require the use of two cask designs. The transportation of cesium and strontium capsules has been successfully accomplished using both government-owned and commercially available shipping casks. Many casks are potentially available to transport cesium and strontium capsules. However, the size limitations of the WESF loading area (G cell), the need to transport a large amount of capsules in each shipment, and the lead time required to certify other casks for capsule transport limit the choices to two shipping casks. One cask, the Beneficial Uses Shipping System (BUSS) cask, is a high-payload 4,000-watt (13,650 British thermal units [BTUs] per hour) cask certified by both DOE and the U.S. Nuclear Regulatory Commission (NRC) to transport cesium and strontium capsules. The cask was fabricated specifically for WESF capsules. The BUSS cask is limited to transporting those cesium and strontium capsules certified as “special form” radioactive material.

The other cask, the General Electric Model 2000 (GE-2000), is currently certified by NRC for only 600 watts (2,047 BTUs per hour) and its allowable payload did not specifically include cesium or strontium. However, an amended safety analysis has been submitted to NRC, and approval for the transport of both cesium and strontium capsules with a payload of 2,000 watts (6,824 BTUs per hour) is expected.

To meet the just-in-time processing strategy and accomplish the 26-month delivery schedule assumed for the capsule processing facility, both the BUSS and GE-2000 casks would be needed (Claghorn 1996).

#### DE-ENCAPSULATION OF CAPSULES

Once the casks are received at the capsule processing facility, they would be disassembled, and the double-lined (not the triple-lined Type W) capsules would be removed from the inner basket. The ends of

the metal capsules would then be cut off and the contents slid out of the outer liner. For the cesium capsules, the ends of the inner liners also would be cut off.

Although the same method used to dismantle cesium capsules can be used to cut the strontium outer capsules, the same method cannot be used to cut the strontium inner capsules. Based on experience at Oak Ridge National Laboratory (ORNL) (ORNL 1988), strontium fluoride is usually stuck to the inner Hastelloy capsule wall. The chop saw used to open the strontium inner capsules is modified to make two longitudinal cuts 180 degrees apart on the Hastelloy capsule using a saw with a 0.13-meter (0.43-foot) disc-type blade. Using this procedure, ORNL was able to process about six capsules in three 8-hour shifts.

The solid waste (inner and outer capsule metal lining) generated must be properly packaged and disposed of. It is estimated that the 8,500 kilograms (18,700 pounds) of shredded metal would result in 100 drums of RH solid MLLW for disposal on site.

### **CESIUM SLURRY FEED PREPARATION**

Once exposed by cutting off the ends of the inner capsule liner, the cesium chloride would be dissolved in water, and the chloride removed using an ion exchange. The volume of water needed to process the cesium capsules is conservatively estimated to be approximately 4,540 liters (1,200 gallons), resulting in approximately 5,680 liters (1,500 gallons) of processed solution.

Removal of the chloride from the cesium chloride solution would be required as an additional step prior to introducing the slurry feed into the WTP HLW melters. The process would use an anion exchanger to exchange the chloride ion with hydroxide. Hydroxide is the anion of choice because of the high selectivity of hydroxide to chloride. The chlorine-loaded ion exchange resin would be disposed of as MLLW. The WTP Pretreatment Facility would have similar resin waste, and a similar disposal path could be utilized. There are a number of ion exchange resins that could be used to remove chlorides.

### **STRONTIUM SLURRY FEED PREPARATION**

Strontium fluoride is not water soluble, so it must be pulverized to prepare it for blending with an HLW feed. Once the strontium salt is separated from the capsule into several pieces, the pieces would be transferred into a jaw crusher to crush the salt into particles less than 300 micrometers (0.01 inches) in diameter (number 50 sieve). The strontium powder then would be transferred to a holding tank as a slurry with suspended solids.

#### **E.1.2.3.4.3 Facilities**

The WESF is located in the 200-East Area and consists of the 225-B Encapsulation Building and several other support structures. The WESF was originally designed to process, encapsulate, and store the extracted radionuclides, cesium, and strontium that are generated from other chemical processing operations. Construction of the WESF was started in 1971 and completed in 1974. Processing of cesium continued until 1983, and strontium processing lasted until 1985. The cesium and strontium capsules at Hanford are currently stored and monitored in pool cells at the WESF. The operations, maintenance, deactivation, and decommissioning of the WESF are not analyzed in this EIS. Data presented in this EIS regarding the WESF are provided for information purposes only.

The capsule processing facility would be adjacent to the WTP in the 200-East Area, the building would have a footprint of 3,800 square meters (40,500 square feet), and the facility boundaries would encompass 1.6 hectares (3.9 acres). The Cesium and Strontium Capsule Processing Facility would be constructed as a heavily shielded facility. The construction requirements for the capsule processing facility are scaled

from the WTP facility data. That scaling factor was increased by 20 percent to account for the increased shielding required in the Cesium and Strontium Capsule Processing Facility.

Based on an operating efficiency of 80 percent, the capsule processing facility's processing rate must be 1,140 capsules per year to de-encapsulate the inventory of cesium and strontium capsules within the 26-month operating time. Transportation of casks from the WESF would be scheduled to provide just-in-time delivery for processing. The capsule processing facility is not currently planned to store casks.

Once the capsule contents are prepared into an acceptable slurry feed, the slurry would be pumped to the DST system to await feed into the WTP HLW melter. Storage in the DST system would be required because the capsule processing facility would require 26 months of operation to de-encapsulate and prepare the cesium and strontium slurry feed; however, the WTP HLW melter would require only 12 months to vitrify the waste. It would be possible to bypass the DST system and feed the cesium and strontium slurry directly into the WTP; however, this would require additional design modifications to the feed system for the WTP. This EIS assumes prestorage in the DST system. The capsule slurry feed can either be blended with tank waste or fed to the melter as a unique campaign. The vitrification of a distinctly different feed would potentially require a new glass formulation or chemical additives. For determining peak cesium and strontium processing impacts, this EIS assumes immobilization of the cesium and strontium capsule slurry feed would occur as a separate campaign after all tank waste treatment has been completed (CEES 2006c).

#### **E.1.2.3.4.4 Assumptions and Uncertainties**

The following lists the assumptions for the construction, operation, and deactivation of a capsule processing facility that are the basis for this EIS analysis. Discussion of the level of uncertainty associated with each assumption is included.

- The treatment of capsule waste was assumed to be a separate vitrification campaign from the processing of tank HLW. For analysis purposes, the cesium and strontium slurry feed would be treated as a separate waste stream through the WTP after completing the treatment of all tank waste. It was estimated that the vitrification of cesium and strontium capsule waste into canistered waste would result in the production of an additional 340 canisters of glass (1.2 cubic meters [42.4 cubic feet] each) that would be stored on site as IHLW. The estimated number of canisters produced (340) is uncertain and may not be consistent with the current glass properties model. Without blending, additional glass-forming or other chemical additives would be required to maintain an acceptable final waste form. Blending the cesium and strontium slurry feed with the tank waste would require the authorization, funding, and construction of the Cesium and Strontium Capsule Processing Facility to be completed, with sufficient WTP operations remaining to support this alternative. Other analysis suggests that sufficient blending may not lead to an increase in the number of canisters from the inclusion of cesium and strontium capsules in the HLW melter feed stream (i.e., the volume of glass would be constrained by something other than the cesium and strontium).
- Chloride would be removed before treatment at the WTP. The ion exchange resin was conservatively assumed to be a nitrate-based resin as discussed in trade study WHC-SD-WM-ES-382 (Claghorn 1996). The more efficient hydroxide resin, if utilized, could reduce secondary waste considerably.
- Processing rates in the capsule processing facility would be coordinated with WESF packing, load-out, and transportation rates. Facility modifications or additional storage facilities would not be required at the WESF to support capsule load-out.

- The Cesium and Strontium Capsule Processing Facility would be 98.4 meters (30 feet) tall, on average. The facility would have a volume of 34.4 million cubic meters (1,215 million cubic feet), based on the 3,760-square-meter (40,500-square-foot) footprint (Claghorn 1996).
- The transportation of the cesium and strontium capsules would require the use of both the BUSS and the GE-2000 casks to accomplish the transfer of capsules within the 26 months analyzed for operation of the capsule processing facility. Transportation of capsules from the WESF to the capsule processing facility, using both BUSS and GE-2000 cask designs, would rely on an aggressive schedule that assumes the availability of both casks. The transportation casks and the cesium and strontium capsules may have to be recertified. The transportation of capsules was assumed to be direct from the WESF to the capsule processing facility, with no interim storage.
- The resources required to maintain the WESF through the disposition of the capsules were assumed to be the same as these used for normal routine operations and maintenance of the facility. Routine operations and maintenance of the WESF were not analyzed in this EIS (CEES 2006c).

#### **E.1.2.3.4.5 Cesium and Strontium Capsule Dry Storage**

As an option to processing the cesium and strontium capsules in a new process facility and treating the waste as HLW in the WTP as described above, dry storage of the capsules was analyzed. Presently, the capsules are stored in water pool cells at the WESF. This option would retrieve and store the capsules in a new dry storage facility. Specifically, under this option, a new dry storage facility would be constructed in the 200-East Area of Hanford, operated, and then deactivated. Construction activities would require all of the resources necessary to build an approximately 6,500-square-meter (70,000-square-foot) dry storage facility. Operations activities would include maintaining dry storage of the containerized cesium and strontium capsules, including surveillance and maintenance, for 145 years. The operational period of 145 years (Tank Closure Alternative 6A) was chosen because it represents the maximum storage time under all of the Tank Closure alternatives. Operations would also include retrieving the capsules at the WESF, containerizing the capsules, and transferring the containers to the new dry storage facility. Deactivation activities would include placing the dry storage facility in a stable and known condition following removal of the capsules and all radioactive and/or hazardous materials.

The estimated material resources and labor requirements, air emissions, and secondary waste generated during construction, operation, and deactivation of this dry storage facility would be exceeded or bounded by those associated with processing the capsules. Therefore, the analysis concluded that the environmental impacts of dry storage of the capsules were bounded by the processing of the capsules (CEES 2010c).

#### **E.1.2.3.5 Supplemental Waste Treatment Options**

This *TC & WM EIS* analyzes three representative technologies out of the several viable options for accomplishing the supplemental treatment and immobilization of LAW. These three waste treatment options, bulk vitrification, cast stone, and steam reforming, were among the technologies selected for further testing and evaluation when DOE sought to identify candidate technologies for accelerating the tank waste cleanup schedule.

##### **E.1.2.3.5.1 Supplemental Technology Selection**

Technologies for treating Hanford tank waste have been researched and evaluated for a number of years. A systematic review of possible technologies was conducted in the early 1990s and resulted in the issuance of the *TWRS EIS* (DOE and Ecology 1996) and a subsequent Record of Decision (ROD)

(62 FR 8693). These documents were reviewed again in early 2002. Vendors, national laboratories, and universities were consulted regarding additional technologies for the purpose of establishing a list of possible LAW treatment technologies. Only technologies that could meet the criterion of closing the LAW treatment gap by accelerating cleanup and reducing risk while maintaining cleanup quality were retained for further consideration.

In March 2002, the Cleanup Challenge and Constraints Team (C3T) was established through a Memorandum of Understanding between DOE and Ecology. As part of the C3T, a Mission Acceleration Initiative (MAI) working group was formed to select candidate supplemental technologies to augment the WTP.

The technologies were grouped into families that employed the same basic principles, but differed in their implementation (i.e., all calcination technologies were grouped together and all polymer-based microencapsulation technologies were grouped together). Tank 241-S-112 was selected as a good representative for the targeted LAW source SSTs.

Technology experts prepared a short briefing on each technology and its application to treatment of tank 241-S-112 waste. Additionally, separations technologies were combined with immobilization technologies to constitute complete treatment options.

In April 2002, DOE evaluated over 50 options for potential supplemental technologies. DOE's results were documented in the *Mission Acceleration Initiative Demonstration Information Package* (CH2M HILL 2003a). The results were reviewed by ORP representatives, who selected the following seven representative technology options, two with two suboptions, for more-detailed evaluation:

- Bulk vitrification
- Active-metal reduction
- Steam reforming
- Clean salt
  - Without cesium ion exchange
  - With cesium ion exchange
- Clean salt and sulfate removal
  - Without cesium ion exchange
  - With cesium ion exchange
- Cast stone
- Sulfate removal

Technical data for the seven options were developed, and the Hanford C3T MAI subgroup performed the final evaluation to select appropriate technologies for further development. Details of the selection process can be found in the *Evaluation of Low-Activity Waste Feed Supplemental Treatment Options by the C3T Mission Acceleration Initiative Team for the Office of River Protection* (Choho and Gasper 2002).

The evaluation criteria used for the final selection of candidate technologies were based on compliance and safety, project utility, operability, technical risk, and programmatic risk. A workshop was held to define the supplemental treatment goals, criteria, and measures by which performance could be judged versus the criteria. The team established six goals, 10 selection criteria, and 14 measures to guide the technology evaluation selection process. The six goals were to (1) ensure worker and public safety, (2) provide environmental protection comparable to the current vitrified waste disposal plan, (3) maximize schedule acceleration, (4) maximize cost-effectiveness, (5) maximize operability, and (6) minimize overall system interface impacts.

Using both qualitative and quantitative measures to score the technologies with respect to the goals and criteria, the MAI subgroup selected the following three LAW immobilization approaches for further development:

- Bulk vitrification
- Cast stone
- Steam reforming

Additionally, the subgroup selected one pretreatment option, sulfate removal, for further development.

The alternatives analyzed in this *TC & WM EIS* use the cast stone process to represent supplemental LAW treatment technologies that do not require heat for solidification of the waste (i.e., a nonthermal treatment process). Bulk vitrification and steam reforming are technologies that require heat input for waste solidification. In the *TC & WM EIS* alternatives, bulk vitrification and steam reforming are the STPs used to represent the thermal treatment technologies. Sulfate removal is also included in one of the alternatives as a pretreatment process outside the WTP. The following sections describe the supplemental LAW treatment technologies included in the alternatives analyzed in this *TC & WM EIS*.

Complete descriptions for supplemental treatment options are presented in the following sections: Section E.1.2.3.6, thermal treatment – bulk vitrification; Section E.1.2.3.7, nonthermal treatment – cast stone; Section E.1.2.3.8, thermal treatment – steam reforming; Section E.1.2.3.9, sulfate removal; Section E.1.2.3.10, technetium-99 removal; and Section E.1.2.3.11, mixed TRU waste processing. The level of maturity for each technology is also discussed in these sections.

Technologies that were not analyzed in detail in this EIS (see Section E.1.3) are not precluded from selection as supplemental treatment technologies to treat tank waste. As information matures so the candidate technologies can be evaluated at relative parity by the decisionmakers, technologies other than those analyzed in detail by this EIS may be chosen for use. The known impacts of any candidate treatment technology can be evaluated against the impacts of the technologies analyzed in detail by this EIS. Impacts of the technology would be evaluated relative to the impacts in this EIS, and additional NEPA analysis would be required before selection of that treatment technology.

#### **E.1.2.3.5.2 Separations Activities**

Each of the *TC & WM EIS* alternatives that includes use of supplemental treatment technologies in the 200-East Area of Hanford would include use of the pretreatment capability provided by the WTP. In contrast, waste feeds for supplemental treatment technologies used in the 200-West Area would not undergo WTP pretreatment, but would instead be subject to solid-liquid separations activities. These activities would primarily entail the application of a solid-liquid separations process that would be conducted in a new 200-West Area Solid-Liquid Separations Facility using waste feed from 35 SSTs that have tentatively been identified to contain cesium-137 concentrations of less than 0.05 curies per liter (0.19 curies per gallon) (see Table E-8). Waste contained in many of the 35 tanks was received from processing facilities that removed radionuclides, such as cesium, strontium, and transuranics. The extent of separations activities would depend on the waste feed being processed and the immobilization operation being used.

**Table E–8. Designation of Low-Activity Waste Tanks**

<b>Tank</b>	<b>Cesium-137 (curies per liter for 7-molar sodium solution)</b>	<b>Sodium (kilograms)</b>	<b>Sodium/Cesium (molar ratio)</b>	<b>Cesium-137 (curies)</b>
241-TX-106	0.050	4.39×10 <sup>5</sup>	3.00×10 <sup>5</sup>	189,000
241-BY-110	0.049	4.67×10 <sup>5</sup>	3.03×10 <sup>5</sup>	199,000
241-TX-103	0.049	1.83×10 <sup>5</sup>	3.04×10 <sup>5</sup>	77,700
241-TX-108	0.046	1.62×10 <sup>5</sup>	3.20×10 <sup>5</sup>	65,300
241-TX-105	0.046	7.40×10 <sup>5</sup>	3.25×10 <sup>5</sup>	294,000
241-TX-115	0.045	7.12×10 <sup>5</sup>	3.29×10 <sup>5</sup>	279,000
241-TX-112	0.043	8.19×10 <sup>5</sup>	3.42×10 <sup>5</sup>	309,000
241-TX-110	0.043	5.80×10 <sup>5</sup>	3.45×10 <sup>5</sup>	217,000
241-S-110	0.042	4.82×10 <sup>5</sup>	3.49×10 <sup>5</sup>	178,000
241-TX-111	0.042	4.44×10 <sup>5</sup>	3.51×10 <sup>5</sup>	163,000
241-B-104	0.042	2.24×10 <sup>5</sup>	3.55×10 <sup>5</sup>	81,300
241-BY-103	0.041	5.68×10 <sup>5</sup>	3.59×10 <sup>5</sup>	204,000
241-TX-114	0.040	6.81×10 <sup>5</sup>	3.75×10 <sup>5</sup>	234,000
241-BY-109	0.036	3.25×10 <sup>5</sup>	4.07×10 <sup>5</sup>	103,000
241-BX-111	0.036	2.47×10 <sup>5</sup>	4.15×10 <sup>5</sup>	76,800
241-BY-111	0.033	4.58×10 <sup>5</sup>	4.51×10 <sup>5</sup>	131,000
241-BY-102	0.033	4.29×10 <sup>5</sup>	4.53×10 <sup>5</sup>	122,000
241-BX-110	0.031	2.74×10 <sup>5</sup>	4.83×10 <sup>5</sup>	73,200
241-BY-105	0.030	6.90×10 <sup>5</sup>	4.92×10 <sup>5</sup>	181,000
241-TX-117	0.027	5.80×10 <sup>5</sup>	5.42×10 <sup>5</sup>	138,000
241-BY-112	0.027	5.94×10 <sup>5</sup>	5.51×10 <sup>5</sup>	139,000
241-TX-118	0.027	3.01×10 <sup>5</sup>	5.58×10 <sup>5</sup>	69,600
241-S-112	0.026	9.83×10 <sup>5</sup>	5.61×10 <sup>5</sup>	226,000
241-BY-108	0.025	2.33×10 <sup>5</sup>	5.82×10 <sup>5</sup>	51,600
241-B-107	0.018	1.49×10 <sup>5</sup>	8.39×10 <sup>5</sup>	22,900
241-B-106	0.017	7.07×10 <sup>4</sup>	8.77×10 <sup>5</sup>	10,400
241-TX-116	0.017	7.08×10 <sup>5</sup>	8.78×10 <sup>5</sup>	104,000
241-TY-102	0.015	8.74×10 <sup>4</sup>	9.97×10 <sup>5</sup>	11,300
241-B-109	0.011	1.57×10 <sup>5</sup>	1.32×10 <sup>6</sup>	15,300
241-B-105	0.011	3.26×10 <sup>5</sup>	1.36×10 <sup>6</sup>	30,900
241-B-103	0.011	5.79×10 <sup>4</sup>	1.37×10 <sup>6</sup>	5,440
241-B-101	0.011	1.11×10 <sup>5</sup>	1.39×10 <sup>6</sup>	10,300
241-S-109	0.007	7.71×10 <sup>5</sup>	2.09×10 <sup>6</sup>	47,600
241-TX-113	0.004	9.18×10 <sup>5</sup>	3.51×10 <sup>6</sup>	33,700
241-T-109	0.003	5.74×10 <sup>4</sup>	5.65×10 <sup>6</sup>	1,310
<b>Total</b>				<b>4.09×10<sup>6</sup></b>

**Note:** To convert kilograms to pounds, multiply by 2.2046.

**Source:** DOE 2003b.

The designation of the contents of the 35 tanks listed in Table E–8 as LAW is based on the analysis found in the *Technical Basis for Classification of Low-Activity Waste Fraction from Hanford Site Tanks*, which stated that waste containing less than 0.05 curies per liter of cesium-137 was not economically practical

for recovery (Petersen 1996). At this concentration, not separating additional cesium-137 from the waste in the 35 tanks would result in the addition of no more than 5 million curies of cesium-137 in the ILAW glass. In accepting the DOE information, NRC concurred with this analysis.

The *TC & WM EIS* mass balances (CEES 2010d) model all 35 LAW tanks identified in Table E-8 through the 200-West Area Solid-Liquid Separations Facility before treatment using the 200-West Area STPs (bulk vitrification, cast stone, or steam reforming). While it is recognized that some of the waste tanks are located in the 200-East Area, for analysis purposes, the waste in all 35 tanks was assumed to require 200-West Area Solid-Liquid Separations Facility pretreatment. Similarly, for analysis purposes, the option for cesium ion exchange in the 200-West Area Solid-Liquid Separations Facility was not exercised; therefore, the mass balances provide conservative estimates of the cesium-137 contribution to the 200-West Area STPs and their waste products (bulk vitrification glass, cast stone waste, steam reforming waste). Waste from the remaining 114 SSTs and the DSTs would be pretreated in the WTP.

This solid-liquid separations (settling and decanting) was assumed to return 50 percent of the solids to the WTP for processing. Strontium and transuranics would be precipitated with a chemical addition during this settling process, resulting in a portion of the strontium-90 and transuranic radionuclides being forwarded to the WTP and the balance to the STP in the 200-West Area.

Some precipitation, settling, and decanting could be conducted in the existing underground storage tanks. However, this *TC & WM EIS* assumes that all separations activities would occur in the 200-West Area Solid-Liquid Separations Facility (DOE 2003f).

#### **E.1.2.3.6 Thermal Supplemental Treatment Technology – Bulk Vitrification**

The bulk vitrification process is one of two representative thermal supplemental treatment technologies (the other is steam reforming) analyzed in this EIS. Bulk vitrification is a technology that is currently being used domestically and internationally for radioactive, hazardous, and mixed waste treatment. Bulk vitrification can be accomplished in situ or in large containers. The application of bulk vitrification to tank waste can be performed only by use of the In-Container Vitrification (ICV)<sup>TM</sup> process. Waste vitrified in large containers offers several advantages. The content of the incoming waste stream is carefully controlled and homogenized, resulting in a more reliable final waste form. The final waste form is also characterized by a small surface-to-volume ratio, which minimizes the potential for waste form leaching. In-container bulk vitrification would be used to supplement the treatment of LAW.

Bulk Vitrification Facilities may be placed in one or both of the 200-East and 200-West Areas as necessary to supplement the current WTP configuration and capacity. The construction and operation of a Bulk Vitrification Facility in the 200-East Area is analyzed under Tank Closure Alternative 3A; in the 200-West Area, under Tank Closure Alternatives 3A, 4, and 5.

The 200-East Area Bulk Vitrification Facility would be near the WTP and would accept a fraction of the LAW generated from the WTP Pretreatment Facility. The 200-East Area Bulk Vitrification Facility would provide supplemental treatment to the WTP by immobilizing up to 37 percent (approximately 18,000 metric tons) of the tank waste sodium currently existing in the Hanford tank system. However, the 200-East Area Bulk Vitrification Facility would actually be required to process more than the 18,000 metric tons of tank waste sodium due to the addition of nonwaste sodium during the WTP pretreatment step. Approximately 5,500 metric tons of sodium would be added to the supplemental treatment waste stream during the WTP pretreatment step. The Pretreatment Facility is designed to accept high-curie waste feed from the DST system and to separate the tank waste into HLW and LAW fractions.



The 200-West Area Bulk Vitrification Facility would immobilize up to 32 percent of the tank waste sodium (approximately 15,000 metric tons) consisting of pretreated waste from the 35 SSTs with low cesium-137 concentrations (listed in Table E-8 and discussed in Section E.1.2.3.5.2). These 35 SSTs are located in both the 200-East and 200-West Areas and contain approximately 15,000 metric tons of sodium. Waste feeds for supplemental treatment technologies used in the 200-West Area would be subject to solid-liquid separations activities consisting of a settling and decanting process to reduce the solids content of the waste received from the LAW tanks. The settling and decanting steps were assumed to return 50 percent of the solids to the WTP for processing (CH2M HILL 2004).

#### **E.1.2.3.6.1 Technology Description**

Bulk vitrification would use an ICV<sup>TM</sup> process. Pretreated tank waste, or retrieved waste with low radioactivity, would be used as feed for the ICV<sup>TM</sup> process. The waste would be mixed with soil or with other glass formers, such as those planned for use in the WTP LAW melters. The mixture would be dried to optimize the process, and the excess water would be treated in the ETF. After mixture with additional soil additives, the dried mixture would then be vitrified in a steel box. Once the vitrification process is complete, soil would be used to fill the void spaces and the steel box would be sealed. The steel box would also serve as the final disposal container. During the melt process, air emissions would be collected by a sealed container hood and would be directed to an offgas treatment system. The secondary waste generated by the offgas treatment system (e.g., filters, liquid effluents) would be stabilized, packaged for disposal, or directed to another treatment facility. A flow diagram of the production-scale bulk vitrification process appears in Figure E-16.

#### **E.1.2.3.6.2 Process**

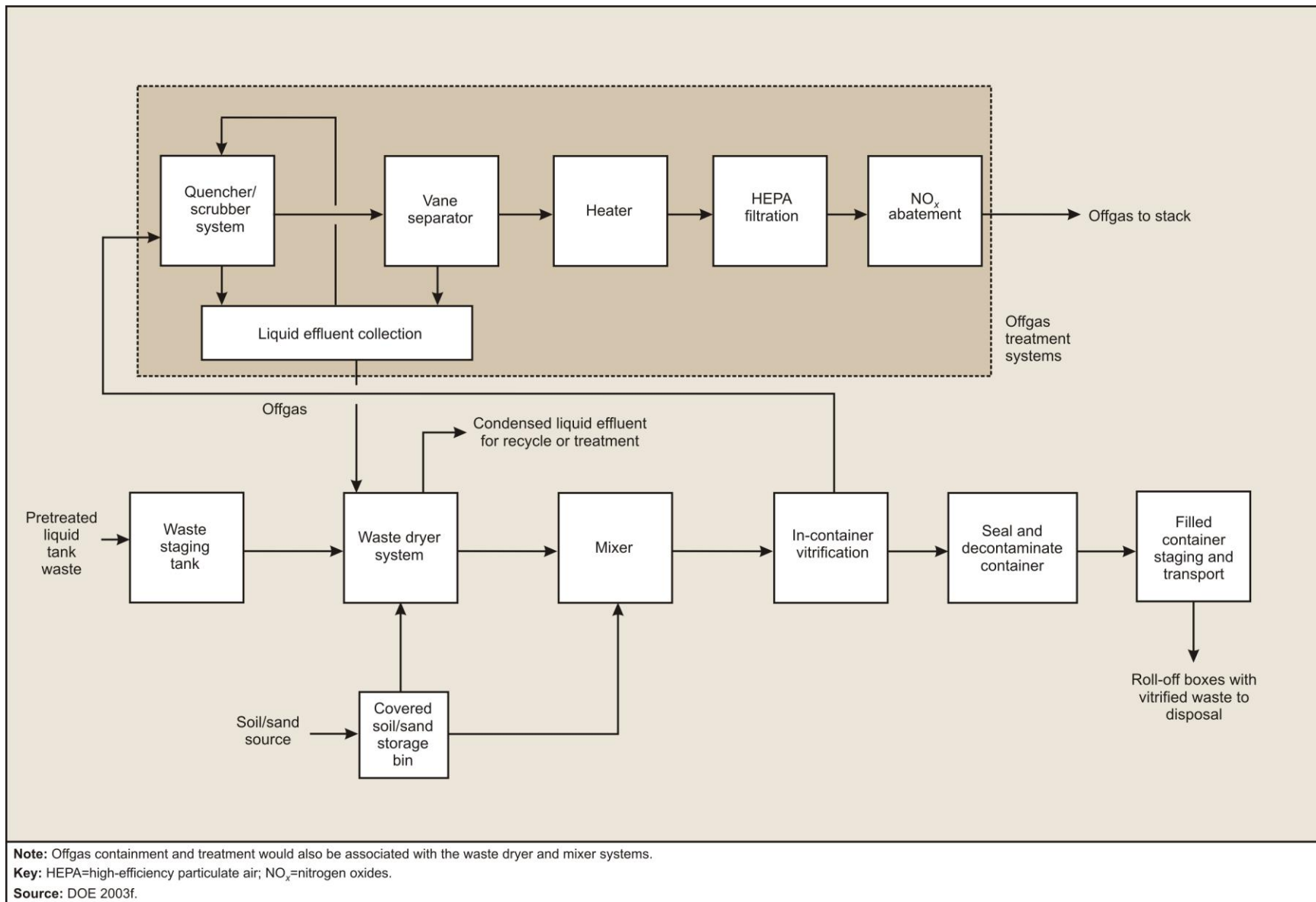
The ICV<sup>TM</sup> process would be the final treatment for the tank waste prior to final disposition on site. The description that follows is based on use of soil and/or sand as glass-forming materials. However, if necessary to meet performance requirements, other glass formulations, including those used in the WTP LAW melters, could be used in the bulk vitrification system. The bulk vitrification process would only receive LAW either from the WTP pretreatment system or the 200-West Area solid-liquid separations pretreatment process.

In the bulk vitrification process, the LAW feed would be mixed with soil and/or sand prior to vitrification. In the initial processing step, the incoming waste stream would be mixed with approximately 20 percent of the total amount of soil and/or sand required to complete the process. A vacuum drying process would be used to remove free water from the mixture. The dryer would utilize an internal waste blending and agitation process to convert the mixture into a granular form. The remaining 80 percent of soil and/or sand would be added in a mixer prior to loading the waste feed into the vitrification container.

The amount of soil and/or sand added would be predetermined so as to achieve approximately 20 weight-percent sodium oxide loading in the final glass waste form. Graphite would be added to the mixture for electrical conductivity to start the vitrification process. The dried waste and soil/sand mixture would then be conveyed to the vitrification container.

Graphite electrodes would be used to melt the waste and sand/soil mixture in the container. An electrical current would be applied to heat and melt the mixture to produce glass. The operating temperature of the melt would be approximately 1,600 °C (2,900 °F).

The Bulk Vitrification Facility is currently configured with parallel processing lines that could process more than one vitrification container at a time. The standard vitrification container would be a steel roll-off box modified for this specific purpose. After completion of the melt, the container would be topped off with clean soil to fill the void spaces, allowed to cool, and sealed.



**Figure E-16. Bulk Vitrification Supplemental Treatment Process Flow Diagram**

The container would be a rectangular, steel box approximately 2.4 meters wide by 3.0 meters high by 7.3 meters long (8 feet wide by 10 feet high by 24 feet long). Each container would hold an estimated 42.6 metric tons of glass. The final dimensions of the containers may change during detailed design of the facilities.

The process of vitrifying the waste would result in volume reduction as a result of the loss of volatile components and reduction of void space during the melting process. The waste volume could be reduced by one-third to one-half by the melting process. As a result, additional waste feed may be added to the container during the melt step. At completion of the melt, the amount of vitrified waste in each container would be roughly 50 to 60 percent of the container volume. The remaining space in the container would be taken up by a perimeter insulating barrier consisting of sand and/or refractory material and soil and/or sand added on top of the vitrified waste to fill void space, prior to sealing the container. The perimeter insulating barrier is expected to be 31 centimeters (12 inches) thick to separate the vitrified waste from the container walls. The purpose of the insulating barrier is to retain heat, ensuring a thorough melt, and to maintain the integrity of the container, ensuring worker safety. The contents of the filled and sealed container would be cooled for approximately 3 days and then transferred to an IDF for near-surface burial (DOE 2003f).

The high temperatures associated with the vitrification process would generate an offgas stream that would require treatment. Nonradioactive offgas emissions would include nitrogen oxides, sulfur oxides, chlorides, fluorides, organics, water, and mercury. Radioactive emissions would potentially include carbon-14, iodine-129, hydrogen-3 (tritium), and smaller fractions of volatile radionuclides such as technetium-99.

The offgas treatment system would consist of a series of unit operations to control emissions of selected offgas constituents. Filters, scrubbers, condensers, and selective catalytic reduction would be used to treat the offgas before discharge into the atmosphere through a stack. The offgas would be directed to a quenching and scrubber system to cool the offgas; remove particulates; and neutralize the sulfur oxides, chlorides, and fluorides. The scrubber solution would be recycled if feasible and/or may be treated and disposed of separately. Sintered metal filtration may be employed to remove particulate emissions prior to introduction into the quenching and scrubber system. Particulates removed by sintered metal filtration would be recycled back into the dryer for mixing with waste feed. Tritium and water would be condensed from the offgas. The scrubbed offgas would then enter a series of two HEPA filters for final particulate removal. After passing through the HEPA filters, a high-efficiency gas absorber could be used to remove radioactive iodine and organic carbon prior to entering the final offgas treatment unit. The final step in offgas treatment would be removal of nitrous oxides through selective catalytic reduction. Removal of mercury in the offgas stream, if necessary, could be accomplished by adding an activated-carbon system. For analysis purposes, this EIS assumes a mercury abatement technology would be applied, resulting in a solid waste stream with the mercury that would be disposed of in an IDF. In addition, in estimating air emissions impacts, this EIS assumes no mercury abatement.

### **E.1.2.3.6.3 Facilities**

The exact locations of the Bulk Vitrification Facilities, if constructed, have not been determined. For purposes of evaluation, it was assumed that the Bulk Vitrification Facilities would be northeast of the 202-S REDOX [Reduction-Oxidation] Facility in the 200-West Area and/or in the 200-East Area near the WTP. In practice, the Bulk Vitrification Facilities could be built in other locations in the 200-East or 200-West Area. See Figure E-11 for proposed locations of the supplemental treatment facilities in the 200-East and 200-West Areas.

The Bulk Vitrification Facilities would contain separate areas for processing and supporting equipment and personnel. The processing equipment areas that handle tank waste would be inside shielded cells or

shielded rooms. Supporting process systems, such as the process offgas system, would also be in shielded areas. The building ventilation systems would be within the reinforced area of the main structure and within the secondary-containment system.

Support systems such as the generator and compressor systems and water chiller would be outside the reinforced structure. Other systems outside of the reinforced structure would include the power supply skid for the vitrification system and a covered soil storage bin. Figure E–17 illustrates a typical Bulk Vitrification Facility process layout.

### **PROCESS AREAS**

The Bulk Vitrification Facilities would consist of multiple structures containing processing and supporting equipment and personnel. The Bulk Vitrification Facilities would include the following process systems:

- Feed receipt and mixing with soil and/or sand
- Waste drying
- Vitrification
- Container sealing and decontamination
- Filled-container staging and transport
- Offgas treatment

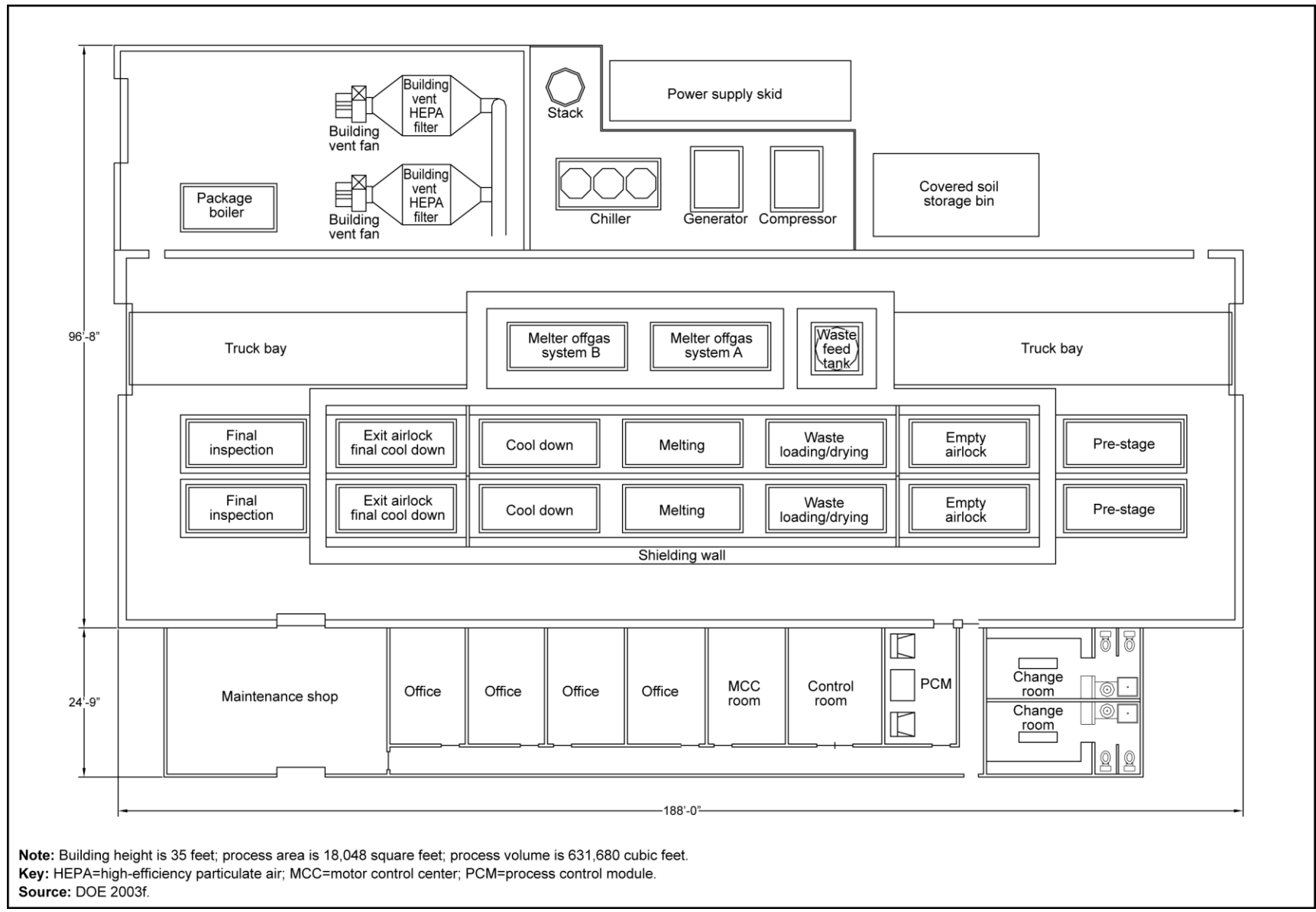
Each Bulk Vitrification Facility would contain parallel and independent vitrification process trains. The containers used for vitrification would be loaded one at a time into the shielded area through an air lock. A rail system would be used to move the containers into and through the process area. The process offgas system would be connected to the bulk vitrification container lid. The main process area would have enough space for three containers in each of the parallel processing lines, one container in the filling position, one in the vitrification position, and one in the initial cooldown position. The vitrification step is estimated to take approximately 2 days to complete. After initial cooldown, each container would be moved into an air lock for final cooling. After final cooling, the container would be moved to a final inspection area outside of the shielded process area. Truck bays on each end of the vitrification process line would be provided for delivery and removal of the vitrification containers.

The remaining shielded process area would be used for receipt and mixing of the tank waste. Waste retrieved from a tank or pretreated waste from the WTP or the solid-liquid separations process would be transferred to a holding tank and slowly added to the mixer/dryer, which already contains a soil and sand mixture, until the maximum waste loading is achieved per the glass formulation analysis. The drying and mixing equipment would be above the vitrification process lines and would be connected to the offgas containment and treatment system.

### **PROCESS SUPPORT AREAS**

The process support areas would contain the building offgas system, including parallel systems for air movement and filtration. These systems would be within the main process facility, but outside of the shielded process areas. Other process support equipment would be outside the main process facility structure.

A maintenance shop, offices, a control room, and change rooms would be used for facility operations and by maintenance personnel. These facilities would be outside the main process structure. The construction contractor may erect temporary facilities during construction of the Bulk Vitrification Facilities. Typically, these facilities would be limited to trailers used as construction site offices and change and restroom facilities for the construction workers (DOE 2003f).



**Note:** Building height is 35 feet; process area is 18,048 square feet; process volume is 631,680 cubic feet.  
**Key:** HEPA=high-efficiency particulate air; MCC=motor control center; PCM=process control module.  
**Source:** DOE 2003f.

**Figure E-17. Supplemental Treatment Bulk Vitrification Facility Layout**

#### **E.1.2.3.6.4 Waste Form/Disposal Package**

##### **PRIMARY WASTE**

The disposal container for vitrified tank waste would be a roll-off box, a large rectangular steel container designed to be moved after filling. For analysis purposes, the container's dimensions were assumed to be 2.4 meters wide by 3.0 meters high by 7.3 meters long (8 feet wide by 10 feet high by 24 feet long), resulting in a volume of 54.3 cubic meters (71.1 cubic yards). The container would be lined on the bottom and sides with approximately 31 centimeters (12 inches) of insulating material before the waste mixture is added, reducing the effective volume for storing immobilized waste glass. The insulating material would be specifically selected soil and/or sand or other refractory material characterized by a high melting temperature and low conductance of heat. The surface temperature of the insulating material next to the waste would be monitored during the vitrification process. After vitrification of the waste, the containers would be staged to allow cooling before transfer to the disposal site. Each container, upon completion of the melt, would contain approximately 42.6 metric tons of glass, and the total container payload, including container and insulating layers, is estimated to weigh approximately 110 metric tons (Ecology 2005; May et al. 2004). The density of glass formed by the bulk vitrification process was assumed to be 2,500 kilograms per cubic meter (156 pounds per cubic foot). Approximately 3 days of cooling would be required to reach the desired external surface temperature prior to transport of the containers for final disposition.

The primary components of the bulk vitrification glass are soluble sodium and aluminum compounds. The soil mixed with the waste would be primarily aluminum silicate, and the sand would provide a source of additional silica. The final form after treatment would be a vitrified mass of sodium aluminum silicate (borosilicate glass), with minor components and contaminants chemically bonded and encapsulated within the glass matrix.

In general, bulk vitrification glass can be formulated at a higher waste loading than ILAW glass, due in part to the higher melting temperatures that can be achieved. The higher temperature can overcome viscosity constraints that normally affect WTP melters. Moreover, because bulk vitrification melters are essentially one-time-use equipment, they are not subject to other glass formulation constraints such as melter refractory corrosion and buildup of crystalline phases over time. It is estimated that an approximate 20 weight-percent sodium oxide loading in the final glass waste form can be achieved (CEES 2010d).

Quantities of primary waste generated by the bulk vitrification process for each alternative are provided in Section E.1.2.4.

##### **Waste Form Performance**

Bulk vitrification containers are planned to be disposed of at Hanford in an IDF. The final waste acceptance criteria have not been established; however, it was assumed that the criteria would be consistent with the HSSWAC (Bagaasen, Westsik, and Brouns 2005; Fluor Hanford 2005a).

The bulk vitrification glass container in its final packaged waste form would be required to meet certain physical limitations prior to acceptance for disposal in an IDF. Some of these physical requirements include a mass of less than 85 metric tons, less than 10 percent void space, and an external surface temperature of the package below 50 °C (122 °F) (Bagaasen, Westsik, and Brouns 2005). Current Demonstration Bulk Vitrification System (DBVS) design criteria estimate the mass of the waste container at nearly 110 metric tons. Accordingly, either the IDF criteria would need to be relaxed to permit the heavier bulk vitrification containers, or the size and mass of the final container would need to be reduced.

Bulk vitrification glass can be formulated to achieve performance that meets requirements established for ILAW glass. The performance of bulk vitrification glass would continue to be evaluated during the scheduled operation of a full-scale testing facility using actual Hanford tank waste from tank 241-S-109 (see Section E.1.2.3.6.6). The rate of release of hazardous constituents depends strongly on the nature of the waste form used to immobilize the constituents. The nature of the waste forms, an analysis of their long-term performance, the methods used to estimate release rates from bulk vitrification glass, and the values of parameters that characterize such release rates are presented in Appendix M.

## **SECONDARY WASTE**

Secondary waste expected from bulk vitrification operations would include solid waste from routine operations and liquid effluents. An example of solid waste would be filtering media from the offgas treatment systems and debris waste. Liquid effluents generated from the offgas treatment system would be collected and sent to the ETF for treatment, with the resultant grouted waste form disposed of in an IDF. Condensed liquid effluent from the waste dryer system would consist primarily of water generated at a rate of 870 kilograms (1,900 pounds) per hour. This condensed liquid effluent would be sent either to the DSTs or to the ETF.

### **E.1.2.3.6.5 Assumptions and Uncertainties**

The waste acceptance criteria for bulk vitrification glass have not been established; however, it was assumed that the criteria would be consistent with the HSSWAC (Bagaasen, Westsik, and Brouns 2005; Fluor Hanford 2005a). The preliminary design of the vitrification container, as detailed in the DBVS (see Section E.1.2.3.6.6), has an estimated mass of 110 metric tons; however, the IDF acceptance criteria limits disposal of packaged units to 85 metric tons.

The capture of several select radionuclides in the final waste form product is an important consideration when evaluating the performance of the bulk vitrification process as a potential supplemental thermal LAW treatment option. The high temperatures associated with the thermal bulk vitrification process would cause some of the select radionuclides to offgas and be captured in secondary-waste streams. The assumed fractions of select radionuclides between final waste form product and secondary-waste streams as a percentage of the feed stream to the bulk vitrification process are provided in Table E-9.

The behavior of technetium-99 and iodine-129 in thermal processes and the fractions that would be captured in the final waste form are difficult to predict. Therefore, fractions of these COPCs were assumed based on ORP guidance (CEES 2007:Attachments 3 and 4). This is in contrast to the use of the HTWOS simulations partitioning factors for the other radioactive COPCs. Further demonstration and testing of this technology should provide the necessary performance data to confirm these assumptions. If necessary, design changes may have to be implemented should the actual fractions in secondary-waste streams be demonstrated to be higher than anticipated. Carbon-14 is highly volatile and would produce very high fractions in the offgas. Incorporation of treatment technologies may be required in the final facility design to reduce these emissions. The select radionuclides that exist in particulate form would be recycled back into the waste feed.

Engineering-scale testing of the bulk vitrification process suggests that some modifications to the final production facility design may be required to eliminate some unfavorable final waste form characteristics. During engineering-scale and large-scale testing, results suggested that technetium-99 might present itself in a more soluble form deposited as a vesicular glass layer on top of the bulk vitrification melt (Pierce et al. 2005). This would affect the release rates from the final waste form in an IDF. The very high temperatures associated with bulk vitrification volatilize and drive off technetium-99 from the waste feed prior to its incorporation into the vitrified glass matrix. The volatilized technetium-99 then condenses on the surface of the melt prior to being carried away in the offgas (Pierce et al. 2005).

**Table E–9. Partitioning Factors of Radioactive Constituents of Potential Concern in Bulk Vitrification Glass<sup>a</sup>**

Radioactive Constituent of Potential Concern	Percentage of BV Feed in BV Glass	Percentage of BV Feed in Liquid Secondary Waste
Iodine-129 <sup>b</sup>	20.0	80.0
Cesium-137	99.5	0.0
Carbon-14	0.0	0.1
Uranium-233	99.9	0.1
Uranium-234	99.9	0.1
Uranium-235	99.9	0.1
Neptunium-237	99.9	0.1
Uranium-238	99.0	0.1
Plutonium-239	100.0	0.0
Plutonium-240	100.0	0.0
Hydrogen-3 (tritium)	0.0	99.8
Strontium-90	100.0	0.0
Technetium-99 <sup>c</sup>	93.4	0.1

<sup>a</sup> Totals may not equal 100 percent due to rounding.

<sup>b</sup> Partitioning factors were directed by ORP (CEES 2007:Attachment 4).

<sup>c</sup> Partitioning factors were directed by ORP (CEES 2007:Attachment 3). This direction included the assumption that 6.5 percent of technetium-99 resides in the waste container insulating material or in the waste container in a leachable form. This 6.5 percent is not included in the BV glass percentage.

**Key:** BV=bulk vitrification; ORP=Office of River Protection.

**Source:** Data extracted from CEES 2007.

Another engineering-scale study of the bulk vitrification process suggested that metal inclusions in the final waste form may corrode and increase release rates of technetium-99 from the final waste form (Bacon et al. 2006). It is expected that these final waste form challenges can be overcome through careful process control and modifications to the final process design.

#### **E.1.2.3.6.6 Demonstration Bulk Vitrification System Project**

The DBVS is a full-scale test facility that would receive waste from tank 241-S-109, mix the waste with soil, dry the soil/waste mixture, and blend in glass former additives to produce a dried waste/additive mixture. The DBVS would then use the ICV<sup>TM</sup> process to convert the dried mixture into boxes of vitrified waste. The ICV<sup>TM</sup> waste would be cooled and interim-stored until the containers are transferred to an IDF for disposal. The DBVS would treat the process offgas to a level that protects human health and the environment and meets applicable requirements. Secondary liquid waste would be filtered and sent to the ETF for treatment and disposal. The process systems would include the following (CH2M HILL 2006a):

- Clean Soil System
- Waste Receipt System
- Waste Mixer/Dryer and Condensate Recovery Systems
- Dried-Waste Handling System
- ICV<sup>TM</sup> System
- Offgas Treatment System
- Secondary-Waste Storage System



The DBVS final design was completed in 2006, and initiation of construction is currently not scheduled. The work is being conducted under a research, development, and demonstration permit issued by Ecology. Fifty bulk vitrification containers are scheduled for production during the DBVS project (Bagaasen, Westsik, and Brouns 2005).

The current design for the DBVS process is consistent with the preliminary conceptual design for the production-scale Bulk Vitrification Facilities, with one notable exception. The DBVS design utilizes a one-step soil addition process, whereas the production-scale process would add soil amendments in two steps. The two-step soil addition process should result in a more uniform waste feed and a more favorable final waste form. Final design of the Bulk Vitrification Facilities as a supplemental technology to the WTP would be largely determined by the results of the DBVS project.

The overall mission drivers for the DBVS project are included in the Justification of Mission Need for the DBVS project, which was approved by DOE in July 2006. The mission-need document requires that the DBVS project complete the following actions:

- Process approximately 720,000 liters (190,000 gallons) of tank 241-S-109 waste into fifty 100-metric-ton boxes of vitrified product.
- Store and dispose of these boxes at the Hanford 200-East Area IDF (IDF-East).
- Evaluate the waste form characteristics.
- Gather pilot plant operability data.
- Determine the overall life-cycle system performance of bulk vitrification and produce a comparison between the bulk vitrification process and construction of a second LAW Vitrification Facility or other supplemental treatment alternatives as provided in TPA Milestone M-62-08 (CH2M HILL 2006a; Ecology, EPA, and DOE 1989).

#### **E.1.2.3.6.7 Demonstration Bulk Vitrification System Technical Review**

In May 2006, CH2M HILL Hanford Group, Inc., chartered an Expert Review Panel to review the current status of the DBVS. It was the consensus of the Expert Review Panel that bulk vitrification is a technology that requires further development and evaluation to determine its potential for meeting the Hanford waste stabilization mission (CH2M HILL 2006a). No fatal flaws (issues that would jeopardize the overall DBVS mission that cannot be mitigated) were found. However, a number of technical issues were found that could significantly affect the ability of the process to meet its overall mission, as stated in the project's Justification of Mission Need document, if not satisfactorily resolved.

The following is a summary of the key issues identified by the Expert Review Panel (CH2M HILL 2006a) that need technical and management actions:

- Additional cold testing is needed to underpin process design and operations (e.g., flow of dried-waste feed, prevention of secondary phases, and balancing of the offgas systems) before radioactive feed is introduced.
- The mixer-dryer and offgas systems need special attention in the next project phase, as most of the development work to date has been focused on the ICV<sup>TM</sup> process.
- System complexity should be reduced to enhance system operability and availability.

- Process sampling and monitoring plans should be improved to ensure that essential operational and needed R&D data from DBVS test runs are captured.
- A better understanding of the DBVS process flowsheet from a chemical point of view is critical to success, both in building a high-reliability production plant and in troubleshooting and recovering from any problems that may occur during operation.
- The feed compositions to be tested in the DBVS project should reflect the spectrum of wastes expected to be processed by bulk vitrification so that a comparison with other supplemental treatment alternatives can be made.
- Potential nuclear safety issues, including confinement strategy, implementation of Integrated Safety Management, and responses to off-normal events, need to be resolved before startup of radioactive waste processing.
- The project needs to ensure that its design and specifications meet the required codes and standards.
- The risk identification and management process has been developed, needs to be improved, and must be effectively utilized in future stages of the project.

The Expert Review Panel identified the following:

- Nineteen technical issues that could result in a failure of the DBVS to meet established DBVS performance requirements unless addressed prior to startup of hot operations
- Twenty-six areas of concern that may result in changes to design or require additional testing to determine if the design is adequate (now or later)
- Thirteen suggested improvements that the project should consider to enhance safety, cost, schedule, or efficiency during test operations and potential transition to a production system downstream

The Expert Review Panel charter focused this review on the technical basis for the existing DBVS design. This review team did not review overall project cost and schedule estimates, nor did it specifically evaluate the efficacy of bulk vitrification technology implementation versus other alternative treatment pathways that DOE may choose in the future (Hamilton 2006a).

#### **E.1.2.3.7 Nonthermal Supplemental Treatment Technology – Cast Stone**

Cast stone is the representative nonthermal supplemental treatment technology analyzed in this *TC & WM EIS*. Cast stone is a waste form in which liquid waste slurries may be immobilized in a process similar to grouting; however, in this application, the grouted waste would be placed in large containers to enhance the durability of the final waste form and facilitate retrieval, if needed. Substantial performance data are available to support the utilization of cast stone as a supplemental treatment technology. Treatment of hazardous waste with Portland cement formulations is common, and the grouting of radioactive waste has been extensively demonstrated worldwide. This nonthermal treatment process does not require specialized equipment and uses readily available materials. Cast stone is an advanced formulation designed to bind waste constituents more tightly in a cementitious matrix. Cast stone would be used to supplement the treatment of LAW.

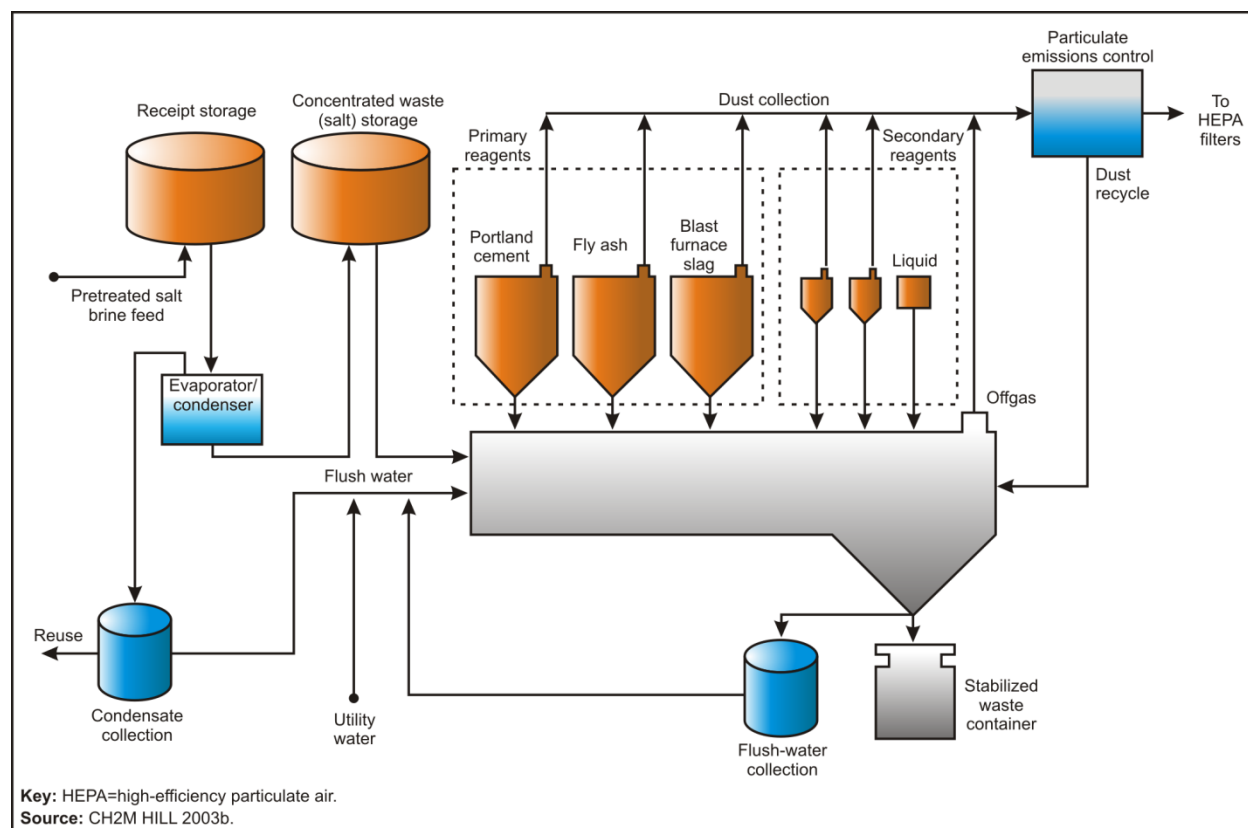
Cast Stone Facilities may be placed in one or both of the 200-East and 200-West Areas as necessary to supplement the current WTP configuration and capacity. The construction and operation of a Cast Stone Facility in the 200-East Area is analyzed under Tank Closure Alternatives 3B, 4, and 5. The construction and operation of a Cast Stone Facility in the 200-West Area is analyzed under Tank Closure Alternative 3B.

The 200-East Area Cast Stone Facility would be near the WTP and would accept a fraction of the LAW generated from the WTP Pretreatment Facility. The 200-East Area Cast Stone Facility would provide supplemental treatment to the WTP by immobilizing up to 37 percent (approximately 18,000 metric tons of sodium) of the total LAW currently existing in the Hanford tank system. However, the 200-East Area Cast Stone Facility would actually be required to process more than the 18,000 metric tons of tank waste sodium due to the addition of nonwaste sodium during the WTP pretreatment step. Approximately 5,500 metric tons of sodium would be added to the supplemental treatment waste stream during the WTP pretreatment step.

The 200-West Area Cast Stone Facility would immobilize up to 32 percent of the tank waste sodium (approximately 15,000 metric tons) consisting of pretreated waste from the 35 SSTs with low cesium-137 concentrations (listed in Table E-8 and discussed in Section E.1.2.3.5.2). These 35 SSTs are located in both the 200-East and 200-West Areas and contain approximately 15,000 metric tons of sodium. Waste feeds for supplemental treatment technologies used in the 200-West Area would be subject to solid-liquid separations activities. This 200-West Area solid-liquid separations process would consist of a settling and decanting process to reduce the solids content of the waste received from the LAW tanks. The settling and decanting steps were assumed to return 50 percent of the solids to the WTP for processing (CH2M HILL 2004).

#### **E.1.2.3.7.1 Technology Description**

The cast stone process can be performed at ambient temperatures and pressures and involves mixing the waste with readily available dry materials (i.e., Portland cement, fly ash, and slag) and casting the wet mix in large containers to produce a waste form exhibiting satisfactory physical and chemical characteristics. The LAW liquid waste stream, either directly from the solid-liquid separations process (200-West Area), or from the WTP Pretreatment Facility (200-East Area), would be stored in dissolved salt cake storage tanks. The LAW feed would be sampled to verify sodium molarity (nominal 5 molar with a range of 4 molar to 10 molar) and to optimize the grout formulations (CH2M HILL 2003b). The LAW feed then would be mixed with dry cementitious binding materials and other additives to produce a wet slurry. The slurry would be cast into large containers for solidification to bind the hazardous and radioactive constituents in the cement matrix. When the cast stone waste has cured sufficiently for transport and storage, the containers would be transported to onsite disposal in an IDF. A flow diagram of the cast stone process is illustrated in Figure E-18.



**Figure E-18. Diagram of Proposed Cast Stone Process**

#### E.1.2.3.7.2 Process

The cast stone process would be the final treatment for the tank LAW prior to final disposition on site in an IDF. LAW liquid waste feed would be received from either direct DST retrieval operations, the solid-liquid separations process (200-West Area), or the WTP Pretreatment Facility (200-East Area). The LAW liquid waste feed would be received and temporarily stored in dissolved salt storage tanks for mixing, blending, homogenization, and sampling. The LAW is sampled to verify sodium molarity and optimize grout formulations (CH2M HILL 2003b). Based on past experience, the lead time necessary to stage, sample, and assay the waste in preliminary dissolved salt storage tanks and to verify an acceptable cast stone formulation would be 5 months. Temporary storage of a 5-molar sodium cast stone feed at maximum throughput rates for 5 months would require the use of two DSTs.

After acceptance and assaying of the LAW feed, the liquid waste would then be pumped to an evaporator to concentrate the waste feed into a 10-molar sodium solution. The evaporator condensate would be recycled in the Cast Stone Facility or directed to the ETF for treatment. The 10-molar sodium liquid waste feed would be stored in holding tanks to cool the liquid waste stream prior to the stabilization process. The contents of each waste tank would be treated as an individual feed batch.

Dry materials (i.e., Portland cement, fly ash, and slag) would be procured directly from vendors, delivered to the Cast Stone Facilities by truck, and stored in individual silos. The materials would be blended before being mixed with the liquid waste using “off-the-shelf” technology for conveying, mixing, and dust suppression. The nominal grout formulation is expected to be approximately 8.2 percent Portland cement, 44.9 percent fly ash, and 46.9 percent slag. The target waste loading is 18.8 weight-percent waste solids in the cast stone. In addition to the dry reagents, a secondary reagent, ferrous sulfate monohydrate, would be added to the slurry mixture to reduce the leachability of hexavalent chromium from the final waste form. The addition of surfactants, plasticizers, and/or antifoaming agents also may

be necessary to improve slurry mixture flow properties and reduce porosity of the final waste form (CH2M HILL 2003b).

The mixer would receive waste slurry at a rate of up to 380 liters (100 gallons) per minute into the stabilization process. The waste slurry and the cement-forming materials would be metered into twin-screw paddle mixers to mix for 10 to 15 minutes. The mixer would discharge into a surge tank that feeds a positive displacement pump. The pump would discharge directly into the cast stone waste containers. The cast stone waste containers would be 2.7 meters long by 2.7 meters wide by 1.5 meters high (9 feet long by 9 feet wide by 5 feet high), with a net volume of approximately 11.5 cubic meters (15 cubic yards). Each container would be filled to 10 cubic meters (13.1 cubic yards) and would contain 20 metric tons of grout. Each 10-cubic-meter (13.1-cubic-yard) container of cast stone waste would represent approximately 5,010 liters (1,324 gallons) of concentrated 10-molar sodium LAW (CH2M HILL 2003b).

After being filled, the containers would be closed and sealed. The exterior surface of the containers would be checked for contamination and decontaminated as necessary. The cast stone waste containers would then be stored in the Staging Area for further curing. The initial curing stage, prior to final disposition on site, is estimated to require 5 days.

Offgas generated by the process would be passed through a HEPA filtration system and then discharged through a stack to the atmosphere. General plant housekeeping, among other activities, would generate liquid effluents (e.g., the cast stone mixer must be routinely washed to prevent buildup of solidified cast stone on internal surfaces). All liquid effluents produced would be returned to the cast stone feed tank and disposed of in subsequent cast stone batches.

### **E.1.2.3.7.3 Facilities**

The exact location of the Cast Stone Facilities, if constructed, has not been determined. For purposes of evaluation, it was assumed that the Cast Stone Facilities would be northeast of the 202-S REDOX Facility in the 200-West Area and/or in the 200-East Area near the WTP. Each Cast Stone Facility would cover approximately 3,810 square meters (41,000 square feet) in area. In practice, the Cast Stone Facilities could be built in other locations in the 200-East or 200-West Area. The larger container sizes discussed in the *Containerized Cast Stone Facility Pre-Conceptual Engineering Report* (CH2M HILL 2003b) may lead to some consideration that only one Cast Stone Facility would be needed to supplement the WTP and still meet LAW immobilization milestones. However, for the purposes of this EIS, a conservative approach would be to consider the construction and operation of two Cast Stone Facilities, one in the 200-East Area and one in the 200-West Area. See Figure E-11 for proposed locations of the supplemental treatment facilities in the 200-East and 200-West Areas.

Cast Stone Facilities would consist of multiple connected structures. The core structure would consist of shielded and reinforced process containment cells. The following process systems would be housed in each of the Cast Stone Facilities:

- LAW receipt, conditioning, and storage
- Dry-material storage and blending
- LAW stabilization
- Cast stone container filling
- Cast stone curing and container staging
- Container decontamination and venting
- Cast stone container storage and transport
- Process support systems

#### **E.1.2.3.7.4 Process Areas**

General descriptions of the cast stone process systems and facilities are provided in the following subsections.

##### **LOW-ACTIVITY WASTE RECEIPT, CONDITIONING, AND STORAGE SYSTEMS**

The LAW liquid waste feed would be received in two 113,600-liter (30,000-gallon) dissolved salt cake storage tanks. The LAW feed would be received intermittently, but at an average of 38 liters (10 gallons) per minute. Each storage tank would temporarily store LAW for up to 5 months while assay and grout formulations were optimized. The LAW liquid feed would be conditioned to a concentrated 10-molar sodium solution through an evaporator and transferred to two smaller 68,100-liter (18,000-gallon) holding tanks. The concentrated LAW would then be introduced into the stabilization process. The holding tanks would be sized to store up to 2 days of LAW feed (CH2M HILL 2003b).

##### **DRY-MATERIAL STORAGE AND BLENDING SYSTEM**

The dry-material storage system would receive bulk materials by either rail or truck, store the materials in dedicated silos, and deliver the dry materials to the dry-material blending system via a pneumatic conveyer system and mechanical conveyance. Dry materials used in the cast stone process would consist of Portland cement, blast furnace slag, and fly ash. The dry-material storage system would be a nonradiological operation. If preblended dry materials were procured from a vendor, they would be transferred directly to the staging vessel of the dry-material blending system. A centralized receiving, storage, and blending system may be used to provide dry-material feed to both the 200-East and 200-West Area Cast Stone Facilities.

The dry-material blending system would proportionately blend the cement, slag, and fly ash into a homogeneous mixture, stage the blended material, and deliver the dry-material mixture to the cast stone mixer. Dry materials would be blended with a horizontal-shaft, ribbon-blade mixer or similar equipment.

A second dry reagent, ferrous sulfate monohydrate, would be introduced into the grout formulation. Due to the smaller quantities required, this material would be received and stored in large sacks or containers and would not require silo storage.

##### **LOW-ACTIVITY WASTE STABILIZATION SYSTEM**

The liquid LAW and dry-material mixing system would be designed to consolidate radioactive service functions within a single process cell. This cell would contain the liquid-handling tanks, pumps, valves, and instrumentation. The concentrated liquid LAW feed would be transferred into the mixers at a rate of up to 380 liters (100 gallons) per minute. The mixing process for each 10-cubic-meter (13.1-cubic-yard) batch would take approximately 15 minutes to complete. Within the mixing cell, the waste slurry and the cement-forming materials would be metered into a typical twin-screw paddle mixer. The mixing operations would be shielded for radiation protection and remotely operated.

##### **CAST STONE CONTAINER-FILLING SYSTEM**

The mixed slurry would be transferred into cast stone waste containers with a net volume of 11.5 cubic meters (15 cubic yards). The inner containment flooring and sides would be made of a continuously welded steel plate to provide leak-tight construction. Cast stone slurry would be transferred to steel containers through either a gravity-feed steel chute or a positive-displacement pump and steel piping. The cast stone waste containers would be filled to capacity through a port in the top of the container. The containers would be filled to minimize void space. The cast stone container-filling operation would be shielded for radiation protection and remotely operated.

## **DECONTAMINATION AND VENTING SYSTEM**

The decontamination system would be a utility module that supports the container decontamination station, as well as the flushing and decontamination of the cast stone process system. The decontamination system would store, feed, and meter decontamination solutions to the container decontamination station and the cast stone process system. This system would house process support equipment, but would not contain radioactive service components. It was assumed that decontamination tanks and associated pumps would be required and that a caustic agent would be used as the primary decontamination solution at a rate of about 3,800 liters (1,000 gallons) per year.

The air filtration system would maintain a negative air pressure within process vessels and process areas. Air would flow from those areas with the least potential for contamination to areas with the highest potential for contamination. Air collected by the air filtration system would be passed through HEPA filters before being discharged to a stack. The air filtration system would include heaters to prevent condensation in the HEPA filters, redundant banks of HEPA filters, exhaust fans, and a stack with a continuous air radiation monitor and a gas sampler train.

## **CAST STONE CONTAINER STORAGE AND TRANSPORT SYSTEM**

The empty-container staging area would provide sufficient storage capacity for a 2-week supply of empty cast stone containers. The containers would be stacked on a concrete pad. A forklift would be used to transfer empty containers to the container transport system for delivery to the cast stone container-filling system. This area of the facility would be contact-operated and -maintained.

The cast stone container transport system would move the containers from the empty-container staging area through an air lock into the container filling area. After the cast stone container is filled, the cast stone container transport system would move the container to a capping and decontamination station and then to the filled-container staging area.

The filled-container staging and transport system would provide indoor interim storage for curing of the cast stone, an inspection area, and a load-out area. The filled-container storage area would accommodate 5 days of waste production. It is estimated that the filled containers would require up to 5 days to reach surface temperatures low enough for final disposition in onsite burial facilities. Containers would be inspected and sampled periodically to ensure proper compressive strength and the absence of free water.

The load-out area would be sized to accommodate the trucks that would be used to haul the containers to the disposal site. A bridge crane would be used to remove the containers from the cast stone container transport system, stack them in the filled-container staging area, and load them onto trucks for shipment to the disposal site (an IDF). The cast stone waste would continue to cure to higher compressive strength after transfer. The staging facility would be a concrete-walled building with a lined concrete floor. The load-out area would be sized to store up to 2 days of facility production to provide for flexibility in scheduling transportation.

## **PROCESS SUPPORT SYSTEMS**

Process support systems that provide auxiliary support to the Cast Stone Facility's primary systems include, but are not limited to, flush water systems, instrumentation and control systems, offgas monitoring and control, chilled water systems, backup emergency electrical generation, and fire protection systems. Flush water systems would be used to rinse the grout mixers between shifts. Chilled water systems would assist in regulating and dissipating heat generation during the curing process.

The construction contractor may erect temporary facilities during construction of the Cast Stone Facilities. Typically, these facilities are limited to trailers used as construction site offices and change and restroom facilities for the construction workers.

For analysis purposes in this EIS, the Cast Stone Facilities would be deactivated at the end of their process mission. Closure of the Cast Stone Facilities is not analyzed in this EIS.

#### **E.1.2.3.7.5 Waste Form/Disposal Package**

##### **PRIMARY WASTE**

A grout formulation similar to formulations demonstrated at Hanford, the Savannah River Site (SRS), and other large commercial grout-based projects was used as a basis for the evaluation in this *TC & WM EIS*. An SRS formulation tested with a similar radioactive tank waste composition was used to calculate the required grout formation and process chemicals. The SRS grout formulation used to make chemical calculations consisted of 4 weight-percent Portland cement, 25 weight-percent fly ash, 25 weight-percent slag, and 46 weight-percent waste salt solution.

Cast stone waste formulation trials have been completed with a limited number of actual tank waste samples and simulants. These formulations have used the baseline cast stone feed stream composition of 5-molar sodium and 3-molar total nitrate. Addition of the grout-forming materials (i.e., Portland cement, fly ash, and slag) increases the slurry volume to 1.4 times the feed volume. Based on the assumptions used in the *TC & WM EIS* analysis, the estimated waste loading of the cast stone waste would be 7.8 weight-percent sodium oxide (CH2M HILL 2003b). It is possible that actual cast stone formulations may be tailored to adjust for batch-to-batch variations as waste is retrieved from different tanks.

A disposal package consisting of an 11.5-cubic-meter (15-cubic-yard) steel container nominally 2.7 meters long by 2.7 meters wide by 1.5 meters high (9 feet long by 9 feet wide by 5 feet high) would be utilized. Each container, for mass balance estimates, was assumed to be filled with 10 cubic meters (13.1 cubic yards) of grout (approximately 20 metric tons of grout). The density of grout formed by the cast stone process was assumed to be 2,000 kilograms per cubic meter (125 pounds per cubic foot). The gross weight of a full cast stone waste container would be less than 25 metric tons, and the container was assumed to be equipped with a bolted lid and to be made of carbon steel, though the actual package to be used has not been determined.

Quantities of primary waste generated by the cast stone process are provided in Section E.1.2.4, Waste Post-Treatment Storage and Disposal.

##### **WASTE FORM PERFORMANCE**

Cast stone waste containers were assumed to be disposed of at Hanford in an IDF. The final waste acceptance criteria have not been established; however, it was assumed that the criteria would be consistent with the HSSWAC (CH2M HILL 2003b; Fluor Hanford 2005a).

The cast stone waste container in its final packaged waste form would be required to meet certain physical limitations prior to acceptance for disposal in an IDF. Some of these physical requirements include mass, void space, and external surface temperature. The mass cannot exceed 25 metric tons. The waste container must be at least 90 percent full. The external surface temperature of the package must be below 50 °C (122 °F) (CH2M HILL 2003b).

Data are available on the performance of grouted waste forms produced previously for Hanford tank waste, as well as similar waste at the SRS. Retention of waste constituents within the cast stone waste is enhanced through addition of fly ash and slag to the grout formulation. The rate of release of hazardous



constituents depends strongly on the nature of the waste form used to immobilize the constituents. The nature of the waste forms, analysis of their long-term performance, the methods used to estimate release rates from the cast stone waste, and values of parameters characterizing such release rates are presented in Appendix M.

In addition to the primary dry reagents, a secondary dry reagent, ferrous sulfate monohydrate, may be used to reduce the leachability of hexavalent chromium in the final waste form.

Waste form performance may be enhanced by introducing reagents such as surfactants, plasticizers, and antifoaming agents into the grout slurry mixture during the mixing stage.

## **SECONDARY WASTE**

Secondary-waste generation for cast stone waste would be minimal because cast stone is a nonthermal process. Excess process liquids generated from cast stone operations include liquids from mixer washing and general housekeeping activities. All liquid effluents produced would be returned to the cast stone waste feed tank and disposed of in subsequent cast stone waste batches. The cast stone process may also include evaporation of waste feeds. It was assumed that any condensate produced would be treated in an existing facility such as the ETF.

The offgas and venting systems would generate a limited amount of HEPA filtering media that would require handling and disposal.

### **E.1.2.3.7.6 Assumptions and Uncertainties**

An extensive worldwide body of experience exists on the grouting of LLW. Grout disposal is the technology used for the majority of LLW in the industry. The Cast Stone Facilities would not require the development of any unique process equipment because the cast stone process utilizes readily available materials.

The capture and immobilization of several select radionuclides in the final waste form product are important considerations when evaluating the performance of the cast stone process as a potential supplemental nonthermal LAW treatment option. The final waste form for the nonthermal cast stone process was assumed to retain 100 percent of select radionuclides due to the low temperatures associated with this process (CEES 2007). The performance measures of most concern are the ability for long-term confinement of technetium-99, iodine-129, and nitrate. Particularly, the leachability of technetium-99 from the final waste form is suspect. A consideration would be to pretreat the LAW by removing technetium prior to immobilizing the LAW in the Cast Stone Facilities. This EIS evaluates the incorporation of technetium removal technologies as a WTP pretreatment step prior to introduction of the grout waste form into the selected supplemental pretreatment process to mitigate questionable waste performance factors associated with technetium-99. This technetium-99 removal option is evaluated under Tank Closure Alternative 3B.

As reported in *Hanford Containerized Cast Stone Facility Task 1 – Process Testing and Development Final Test Report* (Lockrem 2005), a series of laboratory-scale demonstrations were conducted on actual samples of LAW from Hanford tanks. The purpose of these demonstrations was to optimize dry-reagent formulation assumptions and to validate the final waste form. The results of these demonstrations suggest that specific dry-reagent formulations, coupled with the use of a secondary dry reagent (ferrous sulfate monohydrate) to control hexavalent chromium, could yield favorable final waste form performance. Additional tests on final waste form performance assessments are planned (Lockrem 2005).

### **E.1.2.3.8 Thermal Supplemental Treatment Technology – Steam Reforming**

The steam reforming process is one of the two representative thermal supplemental treatment technologies (the other is bulk vitrification) analyzed in this EIS. Steam reforming is used extensively in nonradioactive processing in the petroleum industry and has recently been applied to treating radioactive waste. The application and testing of steam reforming on radioactive waste feeds is limited. A commercial facility in Erwin, Tennessee, uses the fluidized-bed steam reforming (FBSR) process to treat LLW (chiefly ion exchange resins from commercial nuclear power reactors). This facility has been in operation since 1999 and uses a 1.14-meter-diameter (45-inch-diameter) FBSR system (Gasper et al. 2002). The facility size and system are such as to constitute a pilot-scale demonstration of the operational features of a full-scale Hanford plant, which is based on 1.83-meter-diameter (72-inch-diameter) FBSR systems. This treatment technology uses a high-temperature fluidized bed to destroy nitrates and to incorporate in a granular, mineralized waste form radionuclides, sodium, sulfate, chlorine, and fluorine from the waste. DOE has selected steam reforming to treat approximately 3.8 million liters (1 million gallons) of sodium-bearing waste at Idaho National Laboratory (INL) (formerly Idaho National Engineering and Environmental Laboratory) and is currently performing pilot-scale demonstrations using nonradioactive surrogate waste at the Hazen Research Facility (70 FR 75165). The Hazen Research Facility concluded a pilot-scale demonstration in February 2006 that processed 29,900 liters (7,900 gallons) of INL simulant waste during two continuous operational runs totaling 498 hours (THOR and WGI 2006).

Steam Reforming Facilities may be placed in both the 200-East and 200-West Areas as necessary to supplement the current WTP configuration and capacity.

The 200-East Area Steam Reforming Facility would be near the WTP and would accept a fraction of the LAW generated from the WTP Pretreatment Facility. The 200-East Area Steam Reforming Facility would provide supplemental treatment to the WTP by treating up to 37 percent (approximately 18,000 metric tons of sodium) of the tank waste sodium currently existing in the Hanford tank system. However, the 200-East Area Steam Reforming Facility would actually be required to process more than the 18,000 metric tons of tank waste sodium due to the addition of nonwaste sodium during the WTP pretreatment step. Approximately 5,500 metric tons of sodium would be added to the supplemental treatment waste stream during the WTP pretreatment step.

The 200-West Area Steam Reforming Facility would immobilize up to 32 percent (approximately 15,000 metric tons of sodium) of the tank waste sodium, consisting of pretreated waste from the 35 SSTs with low cesium-137 concentrations (listed in Table E-8 and discussed in Section E.1.2.3.5.2). These 35 SSTs are located in both the 200-East and 200-West Areas and contain approximately 15,000 metric tons of sodium. Waste feeds for supplemental treatment technologies used in the 200-West Area would be subject to solid-liquid separations activities. This 200-West Area solid-liquid separations process would consist of a settling and decanting process to reduce the solids content of the waste received from the LAW tanks. The settling and decanting steps were assumed to return 50 percent of the solids to the WTP for processing (CH2M HILL 2004).

#### **E.1.2.3.8.1 Technology Description**

Steam reforming is not a batch process; thus, it depends on a continuous waste feed. The Steam Reforming Facility receives pretreated tank waste (retrieved waste with low radioactivity) and dilutes the waste stream with water to an optimal sodium molarity of 2.9 molar, or approximately 50 percent water content. The dilution of tank LAW is required to transform the steam reforming waste feed into a pumpable liquid for introduction into the fluidized-bed vessel. The 200-East Area Steam Reforming Facility is expected to receive pretreated feed at a nominal molarity of 7.0 molar, whereas the 200-West Area Steam Reforming Facility is expected to receive pretreated feed at 5.0 molar. Within the

fluidized-bed vessel, the steam reforming process converts LAW solutions (tank waste) to granular minerals; volatilizes water; and decomposes organic compounds, nitrate, and nitrite present in the tank waste to carbon dioxide, water, and nitrogen. The steam reforming process uses several chemical consumables, including sucrose, clay, iron oxide, oxygen, and nitrogen. These chemical additives are added at various stages of the process to facilitate reduction and/or conversion. The offgas from the steam reformer process would be treated to remove radionuclides, acid gases, and other pollutants before discharge. The mineralized product (steam reforming waste) was assumed to be suitable for packaging for disposal as a free-form granulated material. However, this waste form may require additional immobilization, either through the use of additives during the steam reforming process or the formulation of monolithic waste forms using cement binders (e.g., grout) (DOE 2003f). For analysis purposes, this EIS assumes the mineralized product (steam reforming waste) would be suitable for packaging and disposal. A flow diagram of a production-scale steam reforming process appears in Figure E-19.

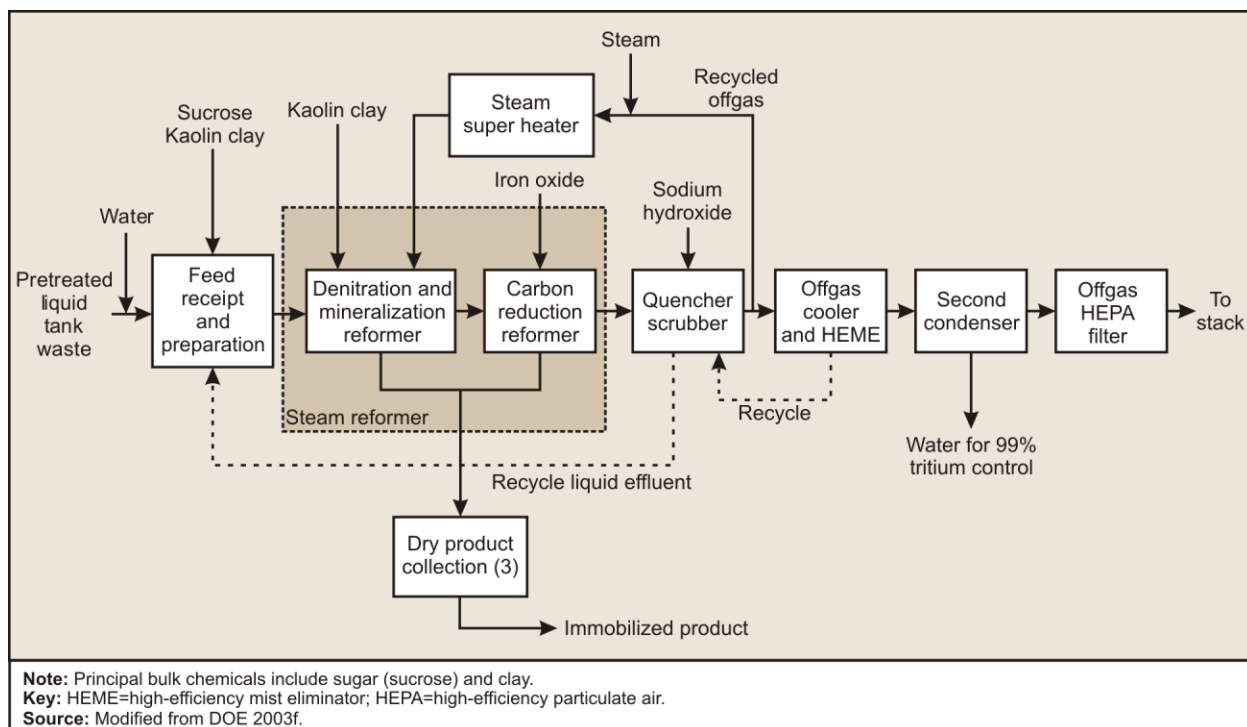


Figure E-19. Steam Reforming Supplemental Treatment Process Flow Diagram

#### E.1.2.3.8.2 Process

The tank waste would first be diluted with water to an acceptable sodium molarity (approximately 2.9 molar) and water content (approximately 50 percent). Then, the diluted liquid waste feed would be mixed in a batch/feed tank with select coreactants, including a soluble, carbon-containing reducing agent (sucrose), referred to in this EIS as a “carbon reductant,” as well as a mineralization additive (clay). The waste feed mixture would be injected by the waste feed pumps into the bottom of the de-nitrator reformer fluid bed just above the fluidizing-gas distributors. Other coreactants (e.g., iron oxide) would be introduced directly into the fluidized-bed vessels during both startup and operation.

In the first fluidized-bed vessel (the de-nitrator reformer), the waste would be converted to processed mineralized waste, organics would be destroyed, and partial destruction of nitrogen compounds would occur. Steam reforming would be operated at 700 to 800 °C (1,290 to 1,470 °F) and under a slight vacuum (–20- to –40-inch water column). The second fluidized-bed vessel (the carbon reduction reformer) would be used to complete the destruction of nitrogen compounds. The upper zone of the

carbon reduction fluid bed would be operated under oxidizing conditions by injection of oxygen. This oxidizing zone would convert residual carbon reductants and organics into carbon dioxide and water vapor. In the steam reformer vessels, the granular/particle bed material would be fluidized with low-pressure, super-heated steam and recycled offgas.

The following chemical reactions and physical changes would take place in the steam reformer:

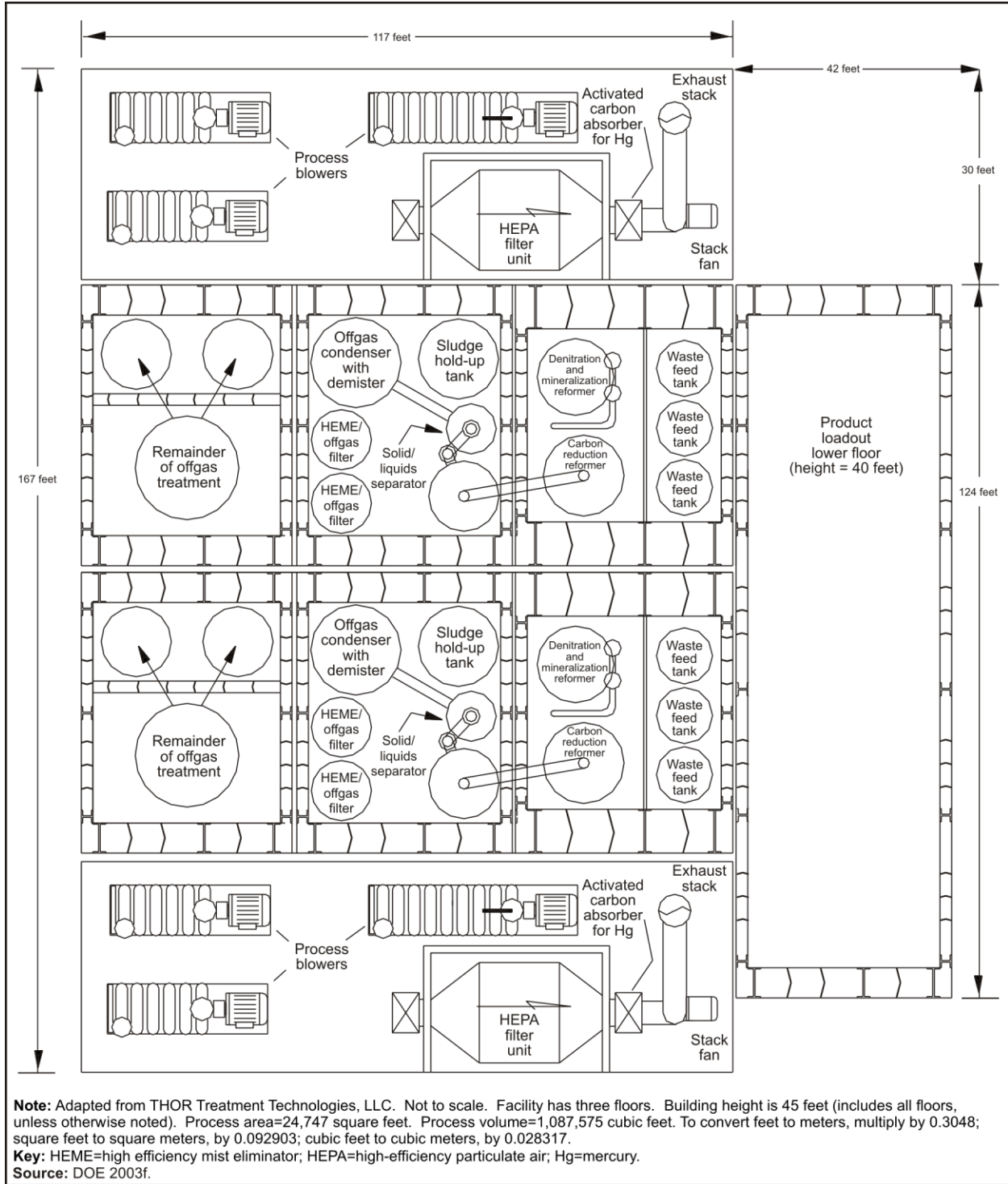
- All liquids would be evaporated.
- A majority of the radioactive cesium, technetium, and other radionuclides would be incorporated into the crystalline structure of the mineralized processed waste in the lower reactor zone. Tritium and carbon-14 would be volatilized to the offgas treatment system. Approximately 20 percent of the iodine-129 would be incorporated into the final waste product, with the balance incorporated into the secondary-waste streams. Section E.1.2.3.8.5 provides additional information on assumptions and uncertainties regarding radioactive constituent partitioning.
- Sodium, potassium, and aluminum in the waste feed would be converted into sodium-alumina-silicate, a stable, mineralized, processed-waste product that contains essentially all the radionuclides and inorganic elements in the waste feed stream.
- Nitrates and nitrites in the waste feed would be reduced to nitrogen gas in the presence of carbon, carbon monoxide, hydrogen, iron, and iron oxide reductants in the bed.
- Organics would be initially converted into light volatile hydrocarbons such as methane, carbon monoxide, hydrogen, carbon dioxide, and water in the de-nitrator reformer bed. In the upper zone of the carbon reduction reformer bed, oxygen would be injected to oxidize the gaseous constituents more fully.
- The steam reforming process would destroy organics in the waste. The process is a nonincineration thermal treatment system. Studies have confirmed that the formation of dioxin and furan do not take place in the steam reforming process.

Certain hazardous metals would be reduced to a nonhazardous valence state (e.g., chromium [VI] would be reduced to chromium [III]) and become chemically bound in the mineralized waste product. Other hazardous metals, such as lead, would also be chemically bound in the mineralized waste sodium-alumina-silicate product. Mercury would volatilize into the offgas and would be captured with activated carbon.

The steam reforming process would produce a mineralized and granular waste form (steam reforming waste) consisting primarily of nepheline ( $\text{NaAlSiO}_4$ ), nosean ( $\text{Na}_8[\text{AlSiO}_4]_6\text{SO}_4$ ), and other aluminosilicate derivatives (DOE 2003f).

### **E.1.2.3.8.3 Facilities**

The 200-West Area Steam Reforming Facility would be required to treat, on average, 26.9 liters (7.1 gallons) per minute of diluted LAW solution that has a sodium concentration of 2.9 molar. Based on information provided by THOR Treatment Technologies, LLC, a single 1.83-meter-diameter (72-inch-diameter) steam reformer unit would have a processing rate of 19.7 liters (5.2 gallons) per minute for 2.9-molar sodium feed solution, assuming 70 percent TOE. Therefore, two 1.83-meter-diameter (72-inch-diameter) steam reformer units, along with ancillary process and service equipment, would be required to achieve the desired processing rate. The typical configuration for a dual steam reformer unit facility is illustrated in Figure E-20; such a facility would occupy an area 49 meters wide by 52 meters long (160 feet wide by 170 feet long).



**Figure E-20. Dual Steam Reforming Facility Layout**

To meet the supplemental treatment needs for the 200-West Area, a dual Steam Reforming Facility would need to be constructed, occupying an area 49 meters wide by 52 meters long (160 feet wide by 170 feet long), or a footprint of approximately 2,530 square meters (27,200 square feet). An additional area would provide space for the control room, change room, standby generator, electrical equipment building, cooling water system, nitrogen supply, instrument air, cooling water, boiler/steam supply, and other

equipment. The total area for the dual Steam Reforming Facility and the ancillary support buildings and equipment would be 4,900 square meters (53,000 square feet).

The 200-East Area Steam Reforming Facility would be near the WTP and be required to treat on average 56.8 liters (15 gallons) per minute of diluted LAW solution with a sodium concentration of 2.9 molar, assuming 70 percent TOE. Therefore, three 1.83-meter-diameter (72-inch-diameter) steam reformer units, along with ancillary process and service equipment, would be required to meet the treatment needs for the 200-East Area. For conservative evaluation, two dual Steam Reforming Facilities (total of four 1.83-meter [72-inch] reformer units) are proposed to be constructed in the 200-East Area.

Two dual Steam Reforming Facilities constructed side by side in the 200-East Area would occupy an area of 98 meters wide by 52 meters long (320 feet wide by 170 feet long) or have a footprint of approximately 5,050 square meters (54,400 square feet). An additional area would be needed to provide the same ancillary equipment and support systems as for the 200-West Area Facility. The total area for both the dual Steam Reforming Facilities and the ancillary process and service equipment would be 8,800 square meters (95,000 square feet).

Each Steam Reforming Facility would be composed of a dual Steam Reforming Facility consisting of two 1.83-meter (72-inch) reformer units with the following major process systems (DOE 2003f):

- Waste batch and feed
- Steam reformers (de-nitration and mineralization reformer vessel and carbon reduction vessel)
- Product packaging
- Offgas treatment systems (quencher/scrubber, HEPA filtrations, and other air pollution control equipment)

## **PROCESS AREAS**

### **Waste Batch and Feed System**

The waste batch and feed system includes the following main hardware components:

- Waste batch tanks
- Waste feed pumps
- Associated instrumentation and controls, piping, valves, etc.

The three waste batch tanks would accept waste from the tank farms directly or as pretreated tank waste from the WTP. The waste would be sampled and assayed for alkali metals, calcium, aluminum, nitrate, and nitrite content. In addition to diluting the tank waste to approximately 50 percent water content, corresponding to a 2.9-molar sodium solution, the requisite coreactants and reductants would be added to the waste in the tank. The tank contents would then be thoroughly mixed until the liquid waste mixture is ready for feed injection into the first fluidized-bed vessel (the de-nitrator reformer). The steam reforming process relies on continuous waste feed, so it is not a batch process. Therefore, feed tank volumes would need to be sized large enough to allow enough time for incoming tank waste to be characterized and prepared by dilution and chemical additives.

### **Steam Reformer**

In the first fluidized-bed vessel (the de-nitrator reformer), the waste would be converted to processed mineralized waste, organics would be destroyed, and partial destruction of nitrogen compounds would

occur. Steam reforming would be operated at 700 to 800 °C (1,290 to 1,470 °F) and under a slight vacuum (–20- to –40-inch water column). Additional coreactants would be added to optimize the incorporation of radionuclides, alkali metals, sulfates, chlorides, fluorides, phosphates, and heavy metals into a solid granular product.

The second fluidized-bed vessel (the carbon reduction reformer) would be used to complete the destruction of nitrogen compounds. The upper zone of the carbon reduction fluid bed would be operated under oxidizing conditions by injection of oxygen. This oxidizing zone would convert residual carbon reductants and organics into carbon dioxide and water vapor. In the steam reformer vessels, the granular/particle bed material would be fluidized with low-pressure, super-heated steam and recycled offgas.

A continuous emission monitor system would monitor the offgas at the outlet of the steam reformer for the purpose of process control and would include online indication of total nitrogen oxides, oxygen, and carbon monoxide. The nitrogen oxides monitor would be used to control the level of reductants in the fluid bed.

### **Product Packaging System and Interim Storage**

The product packaging system would receive the cooled processed mineralized waste (product) solids from the conveyor valve and cooler/separator. The granular waste product would be collected from the bottom of each steam reforming vessel, the cyclone, or other particulate collectors. The product packaging system would convey the granular/powdery final product as a free-flowing solid into bulk containers for disposal or storage. Product solids would be intermittently withdrawn from the bottom of the de-nitrator reformer as needed to maintain the solids level in the fluid bed at the desired bed depth. The outlet conveyor valve would be used to control the discharge rate of the product solids. The product solids from the cooler/separator would be transferred to a holdup tank. Product solids from the holdup tank would be metered to the container fill port where solids would drop by gravity into the bulk container and the container would be sealed.

The processed, mineralized, granular steam reforming waste would be placed in 2.25-cubic-meter (3.0-cubic-yard) steel packages for disposal. At the assumed density of 1,000 kilograms per cubic meter (62.4 pounds per cubic foot), each steel waste box would contain approximately 2.25 metric tons of granular waste. Other container sizes may be evaluated during design.

The steam reforming process facility would require sufficient storage capacity for approximately 200 waste packages, corresponding to the number of waste packages that would be produced on average in a 30-day operating period.

### **Offgas Scrubber and Filtration System**

The offgas scrubber and filtration system would provide components to cool, scrub, filter, and monitor the offgas from the reformer before its discharge up the plant stack. The offgas scrubber and filtration system would consist of the following major components:

- Quencher/scrubber
- Condensers and demister
- HEPA filter
- Continuous emissions and radionuclide monitors

Offgas from the reformer cyclones would flow directly to a submerged-bed quencher/scrubber/evaporator vessel to remove acid gas and particulates and to cool the hot offgases. Sodium hydroxide would be metered to the scrubber solution to neutralize acid gases removed from the offgases.

The cooled and scrubbed offgases would exit the quencher/scrubber and flow to a condenser. The condenser would serve as the process heat sink and control water balance in the process. A high-efficiency mist eliminator would be downstream of the condenser to remove any fine mist that did not coalesce in the condenser. A second-stage condenser would remove 99 percent of the water/tritium.

The quencher/scrubber and a downstream condenser and demister also would function to remove product fines (grains less than 1 to 2 millimeters in diameter) carried over from the reformer. The product fines and neutralized acid gas salts removed by the scrubber and condenser/demister system would be periodically returned to the reformer waste feed tank as a recycle stream for incorporation into the solid processed reformer waste product.

The offgas stream would then be passed through a HEPA filter bank and discharged to the stack. Redundant HEPA filters would be provided so that filter elements could be removed and new elements could be installed while the process system remains online.

Mercury is expected to volatilize in the steam reforming process and be present in the offgas. An activated carbon or gold-impregnated media system could be added to capture mercury from the main process vent downstream of the HEPA filter to remove essentially all volatile mercury from the offgas stream. For analysis purposes, this EIS assumed a mercury abatement technology would be applied, resulting in a solid-waste stream that would be disposed of in an IDF. In addition, for the purpose of estimating air emissions impacts, it was assumed that no mercury abatement would occur.

Two continuous emission monitor systems and a radionuclide monitor system would monitor the offgas from the process. The second continuous emission monitor system would monitor the outlet of the HEPA filters to document that discharges met applicable environmental standards and that process equipment operated as intended.

## **PROCESS SUPPORT AREAS**

The steam reforming process would require auxiliary and utility systems and services to support continuous operations. An additional area would provide space for the control room, change room, standby generator, electrical equipment building, cooling water system, nitrogen supply, instrument air, cooling water, boiler/steam supply, and other equipment. Temporary facilities may be erected by the construction contractor during construction of the Steam Reforming Facility. These facilities would be typically limited to trailers used as construction site offices and change and restroom facilities for the construction workers. The Steam Reforming Facility would be deactivated at the end of the process mission.

### **E.1.2.3.8.4 Waste Form/Disposal Package**

#### **PRIMARY WASTE**

The LAW solution or WTP pretreatment intermediate product would be processed in the steam reforming process and converted into processed, mineralized, granular steam reforming waste with a density of 1,000 kilograms per cubic meter (62.4 pounds per cubic foot). During the steam reforming process, clay ( $\text{Al}_2\text{O}_3\text{SiO}_2$ ), sugar, iron, and oxygen would be added to the system under high-temperature, low-pressure, and fluidizing conditions to chemically bind radionuclides into the processed, mineralized waste and to decompose nitrate, nitrite, and organic compounds in the feed. The principal minerals formed would be nepheline ( $\text{NaAlSiO}_4$ ), nosean ( $\text{Na}_8[\text{AlSiO}_4]_6\text{SO}_4$ ), mullite ( $\text{Al}_6\text{Si}_2\text{O}_{13}$ ), hematite ( $\text{Fe}_2\text{O}_3$ ), magnetite ( $\text{Fe}_3\text{O}_4$ ), and corundum ( $\text{Al}_2\text{O}_3$ ). Hematite and magnetite would form from the added iron. Smaller quantities of other minerals also would be formed, depending upon the composition of the LAW solutions fed to the steam reformer. For example, technetium would substitute for sulfate in the nosean mineral.



The average concentration of sodium oxide in the processed mineralized waste is estimated to be approximately 20 weight-percent sodium oxide loading (CEES 2010d).

The processed, mineralized, granular steam reforming waste would be placed in 2.25-cubic-meter (3.0-cubic-yard) steel packages for disposal. The density of the final waste form was assumed to be 1,000 kilograms per cubic meter (62.4 pounds per cubic foot), resulting in each container weighing approximately 2.25 metric tons. The packages would have a void space of approximately 10 percent. Other container sizes may be evaluated during design.

Quantities of primary waste generated by the steam reforming process are provided in Section E.1.2.4.

#### **WASTE FORM PERFORMANCE**

Packaged, granular waste product from the steam reforming process would be disposed of in an IDF. Final waste acceptance criteria have not been established; however, it was assumed that the criteria would be consistent with the HSSWAC (Fluor Hanford 2005a).

Waste form testing was conducted at the Hazen Research Facility in Golden, Colorado, in December 2001 using a small-scale (15-centimeter-diameter [5.9-inch-diameter]) FBSR unit that produced the mineral waste form from approximately 568 liters (150 gallons) of Hanford 241-AN-107 LAW simulant (Jantzen 2002; McGrail et al. 2003). The principal minerals formed were determined to be nepheline ( $\text{NaAlSiO}_4$ ), nosean ( $\text{Na}_8[\text{AlSiO}_4]_6\text{SO}_4$ ), and other aluminosilicate derivatives. All of the sodium present in the simulated LAW feed solution was determined to be present in the steam reformer mineral waste form derived from small-scale tests with simulated LAW solution. The sodium oxide loading in the steam reformer mineral waste form was determined to be 19.82 weight-percent in this small-scale test with simulated LAW solution. The reducing environment in the steam reformer was demonstrated to convert certain hazardous heavy metals to nonhazardous valence states. For example, hexavalent chromium would be reduced to trivalent chromium and would be incorporated into the mineralized processed waste (sodium-alumina-silicate product). Tritium, carbon-14, and iodine-129 would be volatilized to the offgas treatment system. Nonradioactive simulant has been used in small-scale steam reforming testing. Based on test data, it was assumed that 100 percent of the radionuclides of cesium, cobalt, europium, strontium, and TRU waste elements would be retained in the steam-reformer-processed, mineralized waste. On the first pass, technetium capture rates are expected to be lower. Approximately 66 percent of technetium-99 surrogates are initially captured in the waste form, and 34 percent are retained in the scrubber solution. The scrubber solution would be recycled to the reformer vessel, thereby achieving capture rates approaching 100 percent for technetium into the waste form. Testing on Hanford LAW surrogates has shown that more than 91 percent of the sulfates, more than 92 percent of fluorides, and more than 93 percent of chlorides in the simulated LAW feed react in the steam reformer with the clay coreactant and become an integral part of the final mineralized waste product's crystalline structure. Small amounts of the sulfates, chlorides, and fluorides in the simulated LAW feed are volatilized as acid gases. The offgas treatment system could be designed to compensate for a small difference in retention rates (Jantzen 2002). For analytical purposes, this EIS assumes that 95 percent of the sulfates, chlorides, and fluorides present in the LAW feed would be retained in the mineralized waste processed by the steam reformer, with the remainder volatilized to the offgas treatment system.

Subsequent studies of the performance of the mineralized waste form conducted in 2003 suggest favorable retention rates for surrogate sodium-bearing waste. Furthermore, analyses indicate steam reforming mineral waste product forms comply with the U.S. Environmental Protection Agency's Universal Treatment Standards, making delisting of the final waste form an option for this technology (Jantzen 2003).

The steam reforming process recently demonstrated an approximate 5.5:1 reduction in waste feed to final product mass. In a recent pilot-scale demonstration (two separate runs) conducted at the Hazen Research Facility in Golden, Colorado, in 2006, treatment of 40,400 kilograms (89,000 pounds) of waste feed, including chemical additives, produced approximately 7,350 kilograms (16,200 pounds) of mineralized waste. The waste feed consisted of a simulant for INL sodium-bearing waste (THOR and WGI 2006).

The nature of the steam reforming waste, analysis of its long-term performance, methods used to estimate release rates from the steam reforming waste, and other related parameters are presented in Appendix M.

## SECONDARY WASTE

Liquid waste effluent streams, such as tritium condensed from the offgas, would be treated at the existing TEDF or ETF. Secondary waste associated with the operations of the Steam Reforming Facilities that is not recycled within the process (e.g., spent HEPA filters, liquid effluents, and miscellaneous solid wastes) would also be generated and sent to the ETF for treatment, with the resultant grouted waste form disposed of in an IDF.

### E.1.2.3.8.5 Assumptions and Uncertainties

Final IDF waste acceptance criteria have not been established; however, it was assumed that the criteria would be consistent with the HSSWAC (Fluor Hanford 2005a).

The capture and immobilization of several select radionuclides in the final waste form product are important considerations when evaluating the performance of the steam reforming process as a potential supplemental thermal LAW treatment option. The high temperatures associated with this thermal treatment process, while not as high as bulk vitrification, would cause some of the select radionuclides to offgas and be captured in secondary-waste streams. The estimated fractions of select radionuclides captured between the final waste form product and secondary-waste streams as a percentage of the feed stream to the steam reforming process are provided in Table E-10.

The behavior of technetium-99 and iodine-129 in thermal processes and the fractions that would be captured in the final waste form also are hard to predict. Therefore, these fractions were assumed based on ORP guidance (CEES 2007:Attachments 3 and 4). This is in contrast to the use of the partitioning factors of HTWOS simulations for the other radioactive COPCs. Further demonstration and testing of this technology should provide the necessary performance data to confirm these assumptions. Design changes may have to be implemented should the actual fractions in secondary-waste streams be demonstrated to be higher than anticipated. Carbon-14 is highly volatile and would be produced in very high fractions in the offgas. Incorporation of treatment technologies may be required in the final facility design to reduce these emissions. The select radionuclides that exist in particulate form would be recycled back into the waste feed.

Demonstration of steam reforming technology for treating sodium-bearing tank waste beyond bench scale is very limited. However, bench-scale testing using Hanford tank waste surrogates and recent pilot-scale testing leading to full-scale implementation of steam reforming to treat sodium-bearing tank waste at INL have continued to produce favorable results.

**Table E-10. Partitioning Factors of Radioactive Constituents of  
Potential Concern in Steam Reforming Waste<sup>a</sup>**

Radioactive Constituent of Potential Concern	Percentage of SR Feed in SR Waste	Percentage of SR Feed in Liquid Secondary Waste
Iodine-129 <sup>b</sup>	20.0	80.0
Cesium-137	100.0	0.0
Carbon-14	0.0	0.1
Uranium-233	99.9	0.1
Uranium-234	99.9	0.1
Uranium-235	99.9	0.1
Neptunium-237	99.9	0.1
Uranium-238	99.9	0.1
Plutonium-239	100.0	0.0
Plutonium-240	100.0	0.0
Hydrogen-3 (tritium)	0.0	100.0
Strontium-90	100.0	0.0
Technetium-99 <sup>c</sup>	99.9	0.1

<sup>a</sup> Totals may not equal 100 percent due to rounding.

<sup>b</sup> Partitioning factors were directed by ORP (CEES 2007:Attachment 4).

<sup>c</sup> Partitioning factors were directed by ORP (CEES 2007:Attachment 3).

**Key:** ORP=Office of River Protection; SR=steam reforming.

**Source:** Data extracted from CEES 2007.

Early engineering-scale tests using Hanford LAW surrogates were performed in December 2001 at the Hazen Research Facility. The small-scale tests were performed using 568 liters (150 gallons) of Hanford LAW 241-AN-107 simulant. The data supported the conclusion that steam reforming warranted additional evaluation as a technology for treating sodium-bearing tank waste (Jantzen 2002). Subsequent to the engineering-scale tests at Hazen, a series of pilot-scale demonstrations were performed at the Science and Technology Applications Research (STAR) facility in Idaho Falls, Idaho, on both INL sodium-bearing waste and Hanford LAW 241-AN-107 simulants. One of the tests, conducted using Hanford LAW simulants in August 2004, produced 148 kilograms (326 pounds) of mineralized product. The initial mass of simulated waste feed and additives was 364 kilograms (802 pounds), resulting in a 60 percent reduction in mass. The test was conducted in a 15-centimeter-diameter (0.50-foot-diameter) reactor vessel and lasted 68.4 hours (Olson et al. 2004).

Following the tests conducted at the Hazen Research Facility in Colorado and the STAR facility in Idaho, the Savannah River National Laboratory developed the Bench-Top Steam Reformer to correlate and confirm waste form performance at the pilot-scale level. Generally, the Bench-Top Steam Reformer product was found to be consistent with results achieved during pilot-scale tests at Hazen and STAR using INL and Hanford LAW simulants. In addition, the final waste form performance characteristics continued to show promise (Burket et al. 2005).

Recently, the Hazen Research Facility concluded a series of pilot-scale production runs using INL sodium-bearing waste simulants. This demonstration used a 38-centimeter-diameter (1.25-foot-diameter) reactor vessel to represent a 1:10-scale version of the full-scale production reactors (1.83-meter-diameter [6.0-foot-diameter]) assumed to be needed for the supplemental treatment of Hanford LAW. The first test, completed in February 2006, involved treatment of 20,630 liters (5,450 gallons) of continuous waste feed over 334 hours; the second, completed in June 2006, 9,275 liters (2,450 gallons) over 164 hours.

The vessel size (38-centimeter-diameter [1.25-foot-diameter]) and length of test runs (up to 334 hours continuous) performed at Hazen Research Facility represent the largest and longest pilot-scale tests completed to date. The primary focus of the pilot-scale test was to demonstrate the integration of the entire steam reforming treatment process and to obtain operational data for scale-up, final design, and air emissions. Waste form performance analysis and assessment is pending additional study (THOR and WGI 2006).

The technology development needs for steam reforming include engineering-scale tests that use actual Hanford tank waste and continued assessment of waste product performance. Additional pilot- and full-scale testing is needed to demonstrate long-term bed sustainability and mitigate operational challenges, such as feed nozzle plugging, bridging, and agglomeration, which are inherent with fluidized-bed technologies.

The selection of steam reforming to treat sodium-bearing waste from tanks at INL would lead to progressive testing and full-scale production data on sodium-bearing tank waste that would become available in the near term.

#### **E.1.2.3.9 Sulfate Removal**

High concentrations of sulfate in the WTP LAW feed solutions present processing problems for the LAW vitrification process that uses joule-heated melters. Preliminary testing of the LAW vitrification system indicated that a separate molten sulfur layer could form in the melter at the maximum sulfate-to-sodium ratios found in Hanford tank waste. This molten sulfur layer would be highly corrosive to the melter components. The presence of higher concentrations of sulfate in LAW vitrification process feed can also make the final ILAW glass form more brittle. Formation of the sulfur layer can be avoided by reducing the amount of sulfate in the melter feed. The sulfate removal process would therefore be beneficial in increasing LAW vitrification throughput (in terms of quantity of waste sodium vitrified), increasing LAW melter service life, and decreasing maintenance requirements and downtime for installing LAW melter replacements. Application of the sulfate removal process would increase incorporation of waste into the ILAW glass form from approximately 14 weight-percent to approximately 20 weight-percent sodium oxide (CEES 2007).

The sulfate removal process is an additional pretreatment process step that is not part of the current WTP Pretreatment Facility, but is proposed and evaluated under Tank Closure Alternative 5. However, the addition of sulfate removal as pretreatment technology would require the construction of two additional processing facilities adjacent to the existing WTP facility in the 200-East Area: a waste processing facility (including waste receipt and precipitation process areas) and a grouting facility. The sulfate removal process would be conducted following existing WTP pretreatment processing steps (solid-liquid separations and ion exchange). The liquid product from sulfate removal would contain primarily sodium salts (except sulfate) and would be routed back to the existing WTP LAW melter feed evaporator for vitrification as ILAW. The secondary-waste-stream solid output (sulfate grout) would be routed to an IDF for disposal.

The sulfate removal process is not proposed for use on waste provided as feed to supplemental technologies such as the cast stone, bulk vitrification, or steam reforming processes because there would be no benefit for these technologies. For this reason, Sulfate Removal Facilities are not proposed for the 200-West Area. In addition, low-sulfate waste streams that undergo WTP pretreatment may not need the sulfate removal process (DOE 2003c).

### E.1.2.3.9.1 Technology Description

The sulfate removal supplemental pretreatment process would separate sulfate from LAW that has already been pretreated within the existing WTP Pretreatment Facility to separate entrained solids and select radionuclides (i.e., pretreated LAW). The sulfate would be separated from the pretreated LAW solution by a precipitation process that would use strontium nitrate as the precipitation reagent. The resulting strontium sulfate precipitate would be immobilized in a grout waste form or other suitable containerized waste form. The sulfate-depleted LAW solution would be returned to the WTP for evaporation and subsequent LAW vitrification.

A conceptual diagram of the sulfate removal process is shown in Figure E-21. Entrained solids, strontium-90, TRU waste elements, and cesium-137 would be initially separated from the LAW in the existing WTP Pretreatment Facility. Following processing in the WTP Pretreatment Facility, the LAW solution would be adjusted to pH 1.0 by adding nitric acid. A strontium nitrate solution would be added to the waste to precipitate strontium sulfate, which would be separated from the LAW using conventional solid-liquid separations equipment and immobilized as a grouted waste form (sulfate grout waste). The immobilized strontium sulfate precipitate would be disposed of in an IDF.

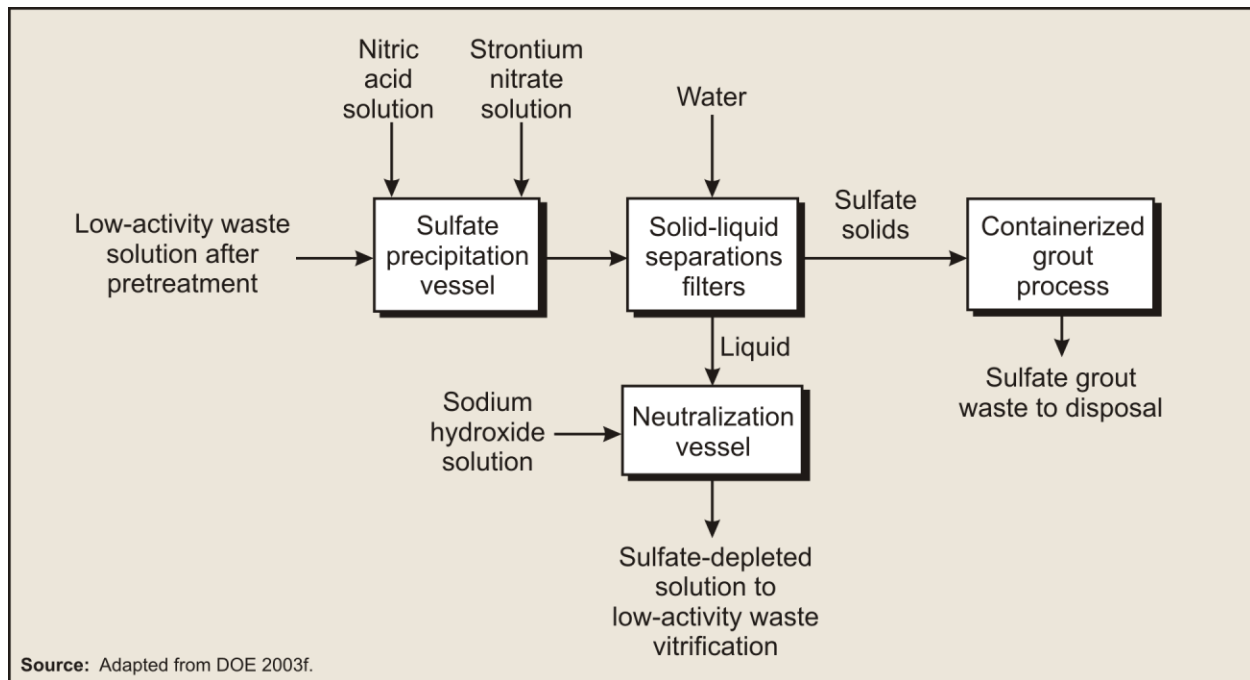


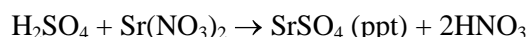
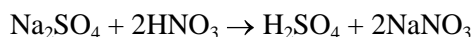
Figure E-21. Sulfate Removal Conceptual Process Diagram

The acidic filtrate (i.e., LAW) from the solid-liquid separations step would be neutralized by adding sodium hydroxide solution prior to transferring the sulfate-depleted LAW back to the WTP for vitrification.

Sulfate precipitation would be a batch-type process that would involve selectively precipitating sulfate using a strontium nitrate reagent. The LAW solution contains carbonate that, if present in sufficient concentration, would preferentially precipitate the added strontium nitrate as strontium carbonate, requiring excess reagent to precipitate the sulfate. To minimize the use of strontium nitrate, the feed must be acidified to decompose carbonate to carbon dioxide according to the following reaction:



This would be followed by further acidification and the addition of the strontium nitrate reagent. Near-minimum solubility of strontium sulfate is attained at approximately pH 1.3 and 1.0-molar sulfate. Precipitation would then occur (approximately 1-hour digestion) by the following reactions:



This process is expected to remove 90 to 95 percent of the sulfate present in the incoming pretreated LAW. Previous testing of a sulfate removal process with simulated LAW indicated that strontium fluoride and some strontium chromate would likely precipitate. Prior experience at Hanford's B Plant indicates that barium, lead, and other components would also precipitate if they were present in the pretreated LAW feed (DOE 2003f).

To evaluate radionuclide partitioning into the strontium sulfate precipitate, Westinghouse Savannah River Company conducted screening tests in which barium nitrate solution was added to a pretreated LAW solution derived from Hanford tank 241-AN-102 supernatant that had been acidified by the addition of nitric acid. The following percentages of radionuclides were removed from the acidified LAW (tank 241-AN-102 supernatant):

- Strontium-90                      89 percent
- Technetium-99                    0 percent
- Cesium-137                        11 percent
- Neptunium-237                    4 percent
- Uranium-238                       0 percent
- Plutonium-239                    14 percent
- Americium-241                    33 percent
- Curium-244                        89 percent

Although barium nitrate was used in these tests, the radionuclide partitioning is expected to be similar using strontium nitrate, except for strontium-90. Because any strontium in solution would be isotopically diluted by the addition of nonradioactive strontium nitrate, it was assumed that essentially all of the strontium-90 would precipitate and end up in the grouted waste form. Note that little, if any, of the technetium-99 is expected to be present in the sulfate grout waste. It was further assumed that iodine-129 would not precipitate or volatilize, but that all of the incoming carbon-14 would be volatilized as carbon dioxide (DOE 2003f).

Following precipitation, the sulfate precipitate would be separated from the liquid stream using an ultrafiltration system similar to that used in the WTP.

A preliminary process flow diagram of the sulfate removal process is provided in Figure E-22. The main inputs to the process include the following:

- LAW from the WTP Pretreatment Facility that has been pretreated to remove solids and cesium
- A 57 weight-percent nitric acid solution that would be added to acidify the pretreated LAW
- A 41.5 weight-percent strontium nitrate solution to precipitate sulfate, as well as other cations and anions
- A 30 weight-percent sodium hydroxide solution to neutralize the sulfate-depleted LAW liquid stream
- Portland cement and other grout mix components to produce the sulfate grout waste form

The primary output streams from the sulfate removal process would be the following:

- Sulfate-depleted pretreated LAW solution, which would be returned to the WTP for vitrification as ILAW.
- The grouted waste form (sulfate grout waste) containing the strontium sulfate precipitate. For analysis purposes, it was assumed that the grout waste form would be placed in steel containers approximately 3.6 cubic meters (4.7 cubic yards) in volume.

The addition of nitric acid to the pretreated LAW is expected to result in the generation of heat (heat of solution and reaction) and acid vapors. As shown in Figure E-22, acid vapors would be absorbed from the offgas system and recycled to the feed stream. The heat evolved would have beneficial effects on the precipitation reaction (the reaction rate and degree of completion). However, heat exchange capability may need to be incorporated into the process as required to ensure safe operation. Neutralization, if necessary, would involve similar considerations.

#### **E.1.2.3.9.2 Facilities**

As previously discussed, use of the sulfate removal process would involve construction of two new facilities in the 200-East Area near the existing WTP, a Sulfate Removal Facility and an associated Sulfate Waste Grout Facility.

The Sulfate Removal Facility would house the acidification, precipitation, and solid-liquid separations processes previously described. The Sulfate Removal Facility would be approximately 130 meters long by 48 meters wide by 11 meters high (426 feet long by 157 feet wide by 35 feet high). These dimensions correspond to a facility footprint and volume of 6,200 square meters (66,900 square feet) and 66,000 cubic meters (2.34 million cubic feet), respectively.

The Sulfate Waste Grout Facility would house grout mixing and curing operations. The grout facility would be approximately 40 meters long by 38 meters wide by 11 meters high (130 feet long by 126 feet wide by 35 feet high). These dimensions correspond to a facility footprint and volume of 1,520 square meters (16,400 square feet) and 16,200 cubic meters (573,000 cubic feet), respectively.

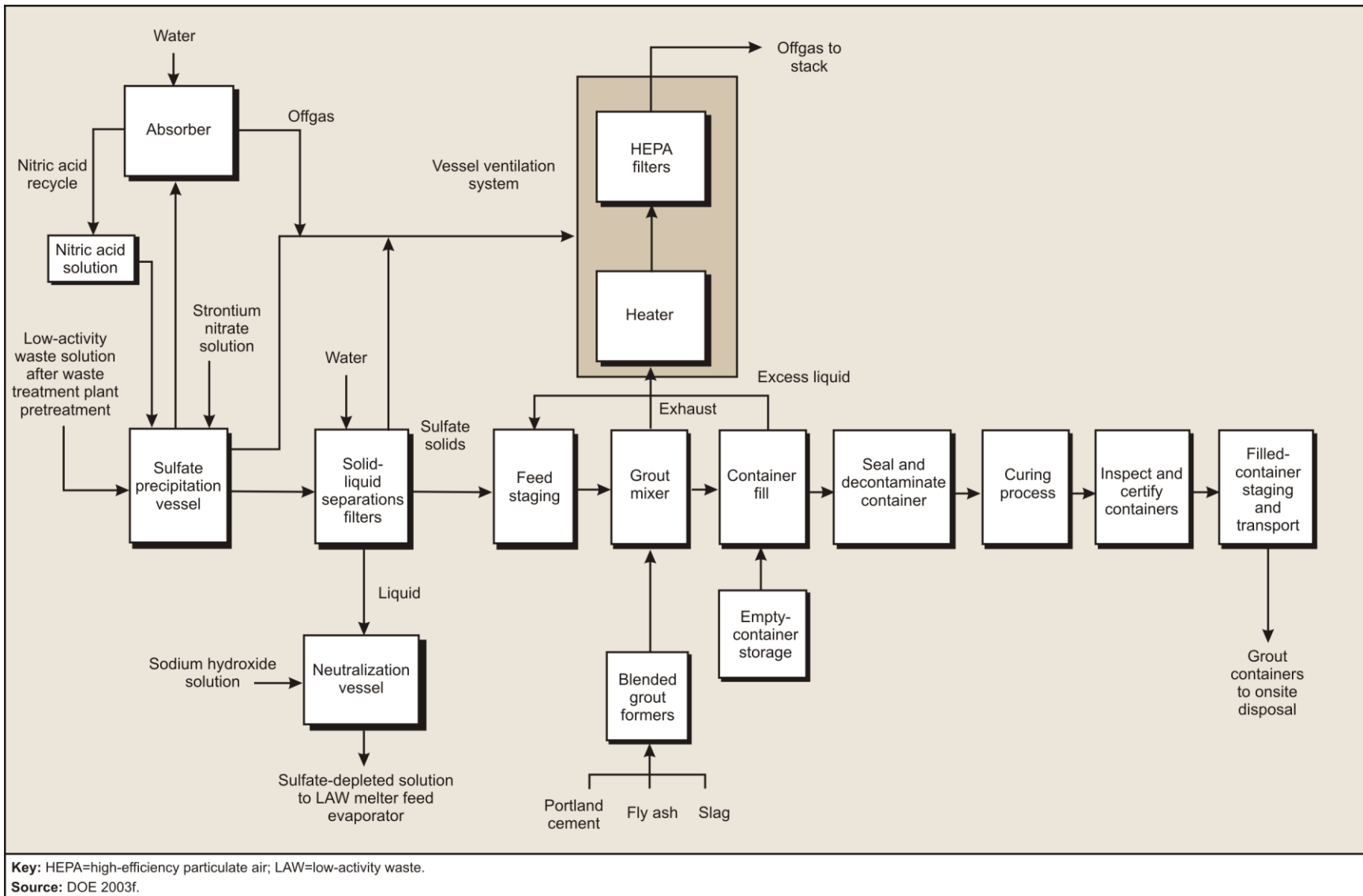


Figure E-22. Sulfate Removal Process Flow Diagram



### **E.1.2.3.9.3 Waste Form/Disposal Package**

#### **PRIMARY WASTE**

The strontium sulfate precipitate would be immobilized in a sulfate grout waste. The formulation of the grout was assumed to be 28 weight-percent sulfate precipitate, 33.8 weight-percent water, and 38.2 weight-percent Portland cement (DOE 2003f). Additional grout formulation would occur in the course of process development to maximize the contaminant retention capability and optimize operational parameters (e.g., heat evolved as a result of hydration and curing).

The sulfate removal process would generate two waste streams: (1) a sulfate-depleted effluent that would be directed to the WTP LAW melter, and (2) a strontium sulfate precipitate that would be directed to the grout facility. The precipitate would be grouted in 3.6-cubic-meter (4.7-cubic-yard) steel containers, allowed to cure, and disposed of in an IDF. For analysis purposes, this EIS assumes that, under Tank Closure Alternative 5, the sulfate removal process would generate approximately 6,100 containers of grouted precipitate filled to 90 percent capacity. The density of the sulfate grout waste was assumed to be 1,800 kilograms per cubic meter (112.4 pounds per cubic foot). The resultant volume of sulfate grout waste would be 19,840 cubic meters (25,950 cubic yards) and the weight would be 35,700 metric tons, which would be disposed of in an IDF.

The quantities of primary waste generated by the sulfate removal process and a more detailed description of the disposal package are provided in Section E.1.2.4, Waste Post-Treatment Storage and Disposal.

#### **WASTE FORM PERFORMANCE**

The sulfate grout waste form would need to comply with applicable regulations (e.g., WAC 173-303, “Dangerous Waste Regulations”). Significant amounts of select radionuclides (e.g., TRU waste, cesium) would be removed within the WTP Pretreatment Facility prior to being introduced into the Sulfate Removal Facility. The performance of this waste form would be further enhanced over cast stone waste because only very low fractions of the select radionuclides that exhibit appreciable mobility (technetium-99 and iodine-129) would be partitioned to the sulfate precipitate (DOE 2003f).

### **E.1.2.3.9.4 Assumptions and Uncertainties**

Although work has been conducted to date regarding the performance of the sulfate precipitation and grouting processes, additional R&D would be necessary to optimize the performance of these processes.

Sulfate removal process development would need to be optimized within the constraints of the existing WTP process design.

The sulfate removal process would add nitrate and some additional sodium to the LAW feed to WTP vitrification. This would be offset by a substantial reduction in the quantity of ILAW glass produced.

Melter replacement frequency was assumed to be independent of the application of sulfate removal to the pretreated LAW solution. However, sulfate removal could increase WTP LAW melter service life, particularly for feeds with high sulfate-to-sodium ratios.

ILAW glass formulations with sulfate removal were assumed to meet the minimum performance requirements of the *Hanford Immobilized Low-Activity Waste Performance Assessment: 2001 Version* (Mann et al. 2001).

Partial precipitation of other cations (e.g., chromium, barium, and lead) and radionuclides (e.g., strontium-90, technetium-99, cesium-137, uranium-238, and TRU waste elements) is expected to occur during sulfate removal, which would decrease the concentration of these species in the ILAW glass produced from sulfate-depleted LAW. For analysis purposes, these components were conservatively assumed to remain in the liquid stream returned to the WTP for incorporation into the ILAW glass. These partially precipitated components were also assumed to be present in the sulfate grout waste.

#### **E.1.2.3.10 Technetium-99 Removal**

The Pretreatment Facility was originally designed to remove technetium and blend the technetium from the LAW vitrification feed with HLW solids for feed to HLW vitrification. However, based on reviews of technetium-99 in ILAW glass, DOE and Ecology agreed to delete technetium removal from the WTP permit (Hedges 2008). With this modification, the technetium content of IHLW glass would decrease, while the technetium concentration in the ILAW glass would increase (DOE 2003c).

Tank Closure Alternatives 2B and 3B are the only alternatives analyzed in this *TC & WM EIS* that would remove technetium in the WTP (see Table E-3). If the technetium were not removed in pretreatment, it would be included in the resulting ILAW glass; if technetium removal occurred during pretreatment, it would be included in the IHLW glass.

Tank Closure Alternatives 2A and 2B represent the conclusions of the *TWRS EIS* ROD (62 FR 8693) with modifications. Tank Closure Alternative 2A does not include technetium-99 removal in pretreatment; as noted above, Tank Closure Alternative 2B does. The only waste forms produced under both Tank Closure Alternatives 2A and 2B would be the vitrified products of the WTP. Therefore, comparison of the two alternatives reflects long-term waste performance on site and shows the impact of the decision to eliminate the technetium-99 removal capability from pretreatment.

Under Tank Closure Alternative 3A, technetium-99 would not be removed from the waste prior to bulk vitrification. The long-term waste performance of the bulk vitrification waste form could then be compared with that of the ILAW glass because technetium-99 would be present in both glass waste forms. Under Tank Closure Alternative 3B, technetium-99 would be removed prior to incorporating the LAW into cast stone waste in the 200-East Area because previous grout data showed that technetium-99 removal would be required for acceptable long-term waste performance. For analysis purposes, no technetium-99 removal was assumed to be performed in the 200-West Area.

##### **E.1.2.3.10.1 Technology Description**

Technetium-99 removal by ion exchange was originally included in the WTP pretreatment process. In that process, waste feed to the WTP would first undergo ultrafiltration to separate liquids from the HLW solids or precipitated strontium and TRU waste. The liquid portion would then be processed through cesium ion exchange to remove cesium-137, which, if not removed, would result in high dose rates from the ILAW glass and require special handling and shielding. After cesium ion exchange, the waste would be processed through technetium ion exchange to remove technetium-99. Technetium-99 has a very long half-life (213,000 years) and requires durable waste forms for disposal. The technetium removed from the feed to LAW vitrification would be blended with HLW solids for feed to HLW vitrification. For analysis purposes, this EIS assumes a technetium-99 removal efficiency of approximately 99 percent (CEES 2007).

**E.1.2.3.10.2 Facility**

The Pretreatment Facility was originally designed to remove technetium. Based on reviews of technetium-99 in ILAW glass, DOE and Ecology decided to delete technetium removal from the WTP permit (Hedges 2008). Construction of the Pretreatment Facility to date has eliminated the capability to remove technetium-99 from the LAW stream. This *TC & WM EIS*, however, assumes that technetium-99 removal could be completed in the existing Pretreatment Facility. Design and construction modifications would need to be made later to add the technetium-99 removal capacity, if required.

**E.1.2.3.10.3 Assumptions and Uncertainties**

The ability of the ion exchange materials to remove technetium-99 to the desired level is currently uncertain. The WTP project has selected an alternative ion exchange resin in the event that the previously identified material did not perform adequately. Some of the technetium-99 present in the tank waste is not in the anionic pertechnetate form; thus, it would not be removed by the ion exchange process.

**E.1.2.3.11 Tank-Derived Mixed Transuranic Waste Processing**

This section describes the processing and packaging of tank-derived mixed TRU waste for disposal. There are currently 20 tanks (17 SSTs and 3 DSTs) that are candidates for classification as mixed TRU waste. Tank inventory data indicate that approximately 11.8 million liters (3.1 million gallons) total of waste could be treated and managed as mixed TRU waste. Waste from 11 of the SSTs may be processed by CH methods, while the waste in the remaining 6 SSTs and 3 DSTs would likely need to be processed by RH methods (Certa et al. 2008).

Table E–11 presents these tanks and their associated waste volumes.

**Table E–11. Mixed Transuranic Waste Tanks and Associated Volumes**

Tank	Waste Type	Waste Volume <sup>a</sup> (kiloliters)
<b>Single-Shell Tanks</b>		
241-B-201	CH-TRU waste	115
241-B-202	CH-TRU waste	111
241-B-203	CH-TRU waste	193
241-B-204	CH-TRU waste	189
241-T-201	CH-TRU waste	110
241-T-202	CH-TRU waste	81
241-T-203	CH-TRU waste	140
241-T-204	CH-TRU waste	140
241-T-104	CH-TRU waste	1,199
241-T-110	CH-TRU waste	1,397
241-T-111	CH-TRU waste	1,692
241-T-105	RH-TRU waste	371
241-T-107	RH-TRU waste	655
241-T-112	RH-TRU waste	226
241-B-107	RH-TRU waste	611
241-B-110	RH-TRU waste	925
241-B-111	RH-TRU waste	910

**Table E–11. Mixed Transuranic Waste Tanks and Associated Volumes**  
(continued)

Tank	Waste Type	Waste Volume <sup>a</sup> (kiloliters)
<b>Double-Shell Tanks</b>		
241-AW-103	RH-TRU waste	1,184
241-AW-105	RH-TRU waste	999
241-SY-102	RH-TRU waste	550
<b>Total Mixed TRU Waste Volume</b>		<b>11,798</b>

<sup>a</sup> Includes salt cake and sludge phases.

**Note:** To convert kiloliters to gallons, multiply by 264.17.

**Key:** CH=contact-handled; RH=remote-handled; TRU=transuranic.

**Source:** Certa et al. 2008; DOE 2003b.

#### E.1.2.3.11.1 Technology Description

The mixed TRU waste would be processed and packaged in accordance with applicable regulations and waste acceptance criteria. The tank waste is considered to be mixed waste and is therefore subject to regulation under the Atomic Energy Act of 1954 and RCRA. Appropriate permits for these facilities would be obtained. All TRU waste packages would be designed as required to meet WIPP Waste Acceptance Criteria.

The CH-TRU waste may be appropriately processed using simple processes such as dewatering and packaging. For RH-TRU waste, an additional step, sludge washing, may be applied to remove the soluble cesium-137 and thereby reduce the radiation dose rate of the packaged waste; however, for analysis purposes, it was assumed that this step would not occur.

#### E.1.2.3.11.2 Contact-Handled Mixed Transuranic Waste

For CH-mixed TRU waste, the process is based on packaging the waste using transportable treatment systems located at each tank farm (in both the 200-East and 200-West Areas). These systems are also referred to as “CH-Mixed TRU Waste Facilities.” Packaged waste would then be sent to interim storage or disposal.

Feed to the CH-Mixed TRU Waste Facilities would be supplied from the MRS skid. Feed would be pumped in batch transfers as a slurry or sludge to the packaging skid. Each facility would receive waste at a maximum rate of 300 liters (80 gallons) per minute in a receiver tank.

The waste would be pumped into a dewatering unit (e.g., centrifuge) that would produce a liquid waste stream (MLLW) and a solid TRU waste stream for packaging. The solid-waste stream would be mixed with an absorbent material to ensure that final waste packaging meets the disposal criteria. The mixed product (TRU waste and absorbent) would then be metered into 208-liter (55-gallon) drums or other approved containers. An appropriate quantity of absorbent material would be added to each waste container to ensure no free liquids are present.

After being filled, the containers would be closed with a bolted lid. The exterior surface of the container would be checked for contamination and decontaminated as necessary. Completed packages would be sent to interim storage for eventual disposal at WIPP.

Offgas generated by the process would be passed through a HEPA filtration system and then discharged through a stack to the atmosphere. Liquid effluent from the facility would be recycled for use in retrieval

processes, forwarded to the DST system pending further treatment, or directed to onsite liquid effluent treatment or disposal facilities, as appropriate.

### E.1.2.3.11.3 Remote-Handled Mixed Transuranic Waste

For RH-mixed TRU waste, the process is assumed to be located in the 200-East Area at a single, fixed facility for treatment and packaging of waste before storage and disposal. See Figure E-11 for the proposed location. The RH-Mixed TRU Waste Facility would be functionally similar to the CH-Mixed TRU Waste Facilities. Differences include the following:

- Waste feed to the RH system would come from the DST system.
- All operations would be remote.
- The RH-mixed TRU waste system would be housed in a fixed facility located in the 200-East Area.

Additionally, the higher radiation levels in the DST waste would require remote-impact wrenches and cranes to access equipment and piping, which could involve using specialized fasteners, seals, connectors, and skid arrangements. Some of the shielding would need to be removable to allow access to equipment. Moreover, during operational testing, it would be necessary to evaluate access spacing, laydown logistics, ability to repair equipment, and remote operability. The facility design would need to incorporate a controlled laydown area that allows for shielding and contamination control. The facility would also need to have extra flushing and decontamination capability.

Figures E-23 and E-24 present simplified process-block-flow diagrams of the packaging process for CH-mixed TRU and RH-mixed TRU waste, respectively.

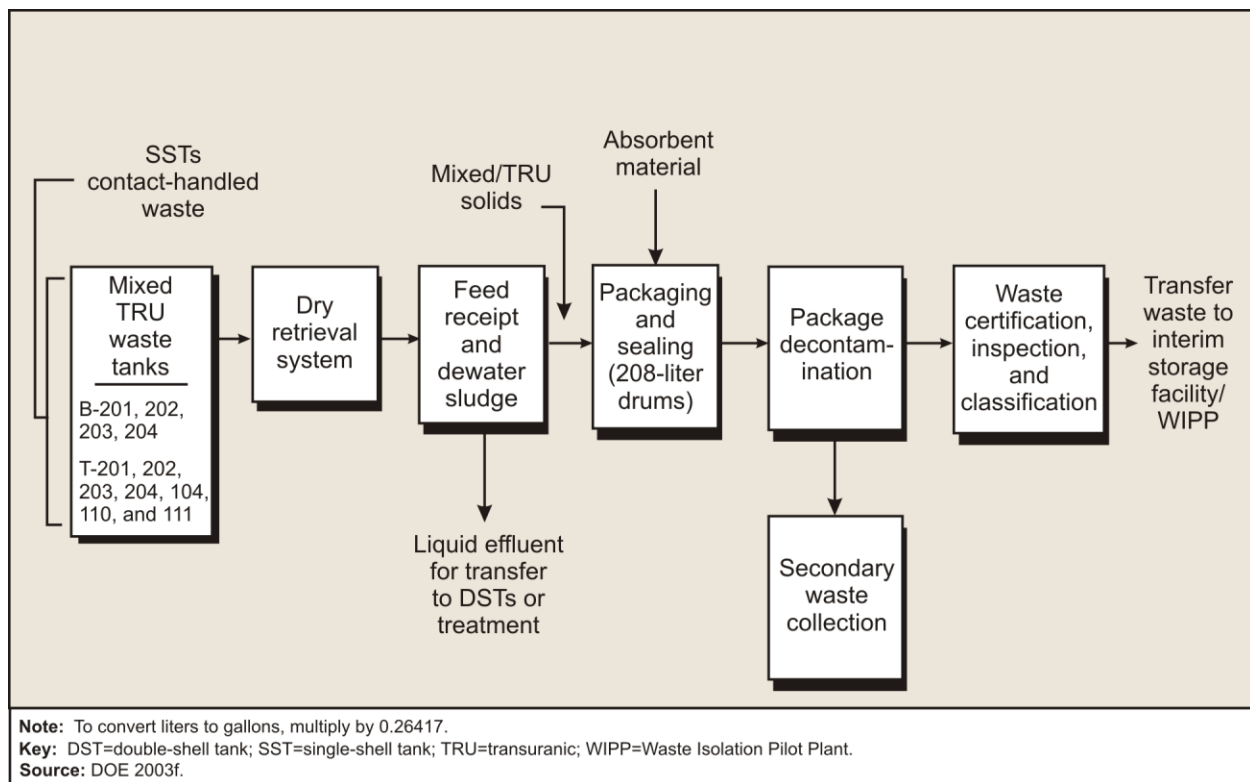
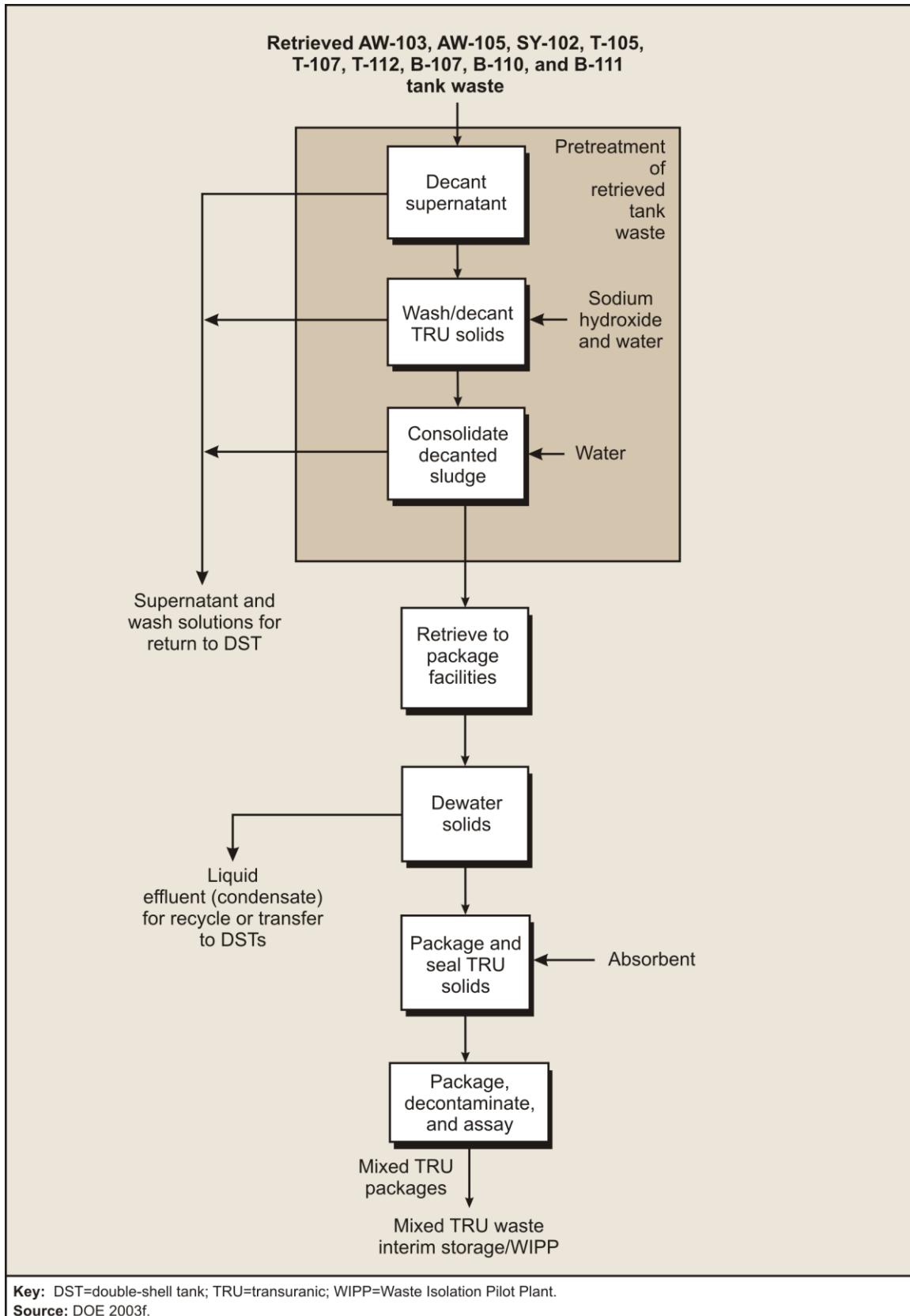


Figure E-23. Contact-Handled Mixed Transuranic Waste Packaging Process Flow Diagram



**Figure E-24. Remote-Handled Mixed Transuranic Waste Packaging Process Flow Diagram**

#### **E.1.2.3.11.4 Transuranic Waste Interim Storage Facility**

Given that the demand for transport and disposal could exceed the delivery capacity of WIPP and the availability of transportation casks for the timeframe that the Hanford tank-derived mixed TRU waste may be treated, onsite interim storage and staging facilities were analyzed. Moreover, logistics and prioritization constraints related to TRU waste transport from Hanford to WIPP could necessitate an extended period of onsite interim storage for tank-derived mixed TRU waste. In addition to storage, characterization and certification of the waste may also be needed. Storage time can exceed 90 days; therefore, the TRU Waste Interim Storage Facility would need to be permitted under RCRA (WAC 173-303).

The proposed TRU Waste Interim Storage Facility is expected to provide only a passive storage function and could store both mixed and nonmixed TRU waste. The supplemental treatment facility that would generate tank-derived TRU waste was assumed to provide a TRU waste package that is compliant with the WIPP Waste Acceptance Criteria. This could necessitate that the supplemental treatment facility provide capabilities for the following:

- Decontamination
- Nondestructive assay (NDA)
- NDE
- Headspace testing
- Certification
- Preparation for transport

Tank-derived mixed CH-TRU and RH-TRU waste would be generated and packaged as discussed in Sections E.1.2.3.11.2 and E.1.2.3.11.3, then transferred to the proposed TRU Waste Interim Storage Facility for staging. This proposed facility would be a steel-walled building on a concrete slab (DOE 2003e). The interior floor of the building would be sloped, with a raised perimeter curb to contain and direct spilled liquids to a collection sump into which a portable sump pump could be installed as required. The floor would be sealed with an impervious epoxy sealant to prevent contaminants from entering the concrete.

The building would be accessible through two rollup truck doors and two personnel doors. The building interior would contain a central aisle that facilitates loading and unloading operations. Access and maneuverability areas around the building would be stabilized with asphalt or gravel. Given that the mixed TRU waste drums would be sealed and their external surfaces decontaminated, the building ventilation would not have to perform a confinement function. Therefore, the building would be maintained at atmospheric pressure, and heating and cooling would not be required for operations. The building would not be insulated except for two rooms, connected through doorways, that would contain the fire riser and electrical/telecommunication systems. The storage area would be provided with overhead lighting.

The TRU Waste Interim Storage Facility footprint was assumed to be 75 meters wide by 174 meters long (247 feet wide by 571 feet long) and to occupy a space envelope of 13,050 square meters (140,500 square feet). To facilitate cost-effectiveness and facility operations, the interim storage building was assumed to be adjacent to the RH-Mixed TRU Waste Facility.

To provide the necessary shielding, RH-TRU waste is overpacked within a high-integrity container, a large, reinforced-concrete cylinder that can be sealed with a lid, at the RH-Mixed TRU Waste Facility. The high-integrity container is packed with 208-liter (55-gallon) drums in two layers of seven drums each. Once filled, it is emplaced in the TRU Waste Interim Storage Facility. Eventually the RH-TRU waste would be retrieved from interim storage and emplaced in a TRU Waste Package

Transporter II (TRUPACT-II) at the Waste Receiving and Processing Facility (WRAP) or a similar facility in preparation for offsite shipping and disposal at WIPP.

As the TRU Waste Interim Storage Facility was assumed to be adjacent to the stationary tank-derived RH-Mixed TRU Waste Facility, the same staff would operate both. Given the minimal work scope of interim storage operations (emplacement/retrieval of drums, periodic inspections, etc.), additional supplemental treatment plant operations and maintenance staff would not be required to support the TRU Waste Interim Storage Facility.

The CH-mixed TRU waste drums and RH-mixed TRU waste packages delivered for interim storage are ready for disposal (i.e., the drums are already sealed and external surfaces do not exceed contamination limits). Therefore, the TRU Waste Interim Storage Facility would not be a credible source of significant hazardous or radioactive emissions during operation.

The TRU Waste Interim Storage Facility would only serve a passive storage function. Operations and maintenance activities within this facility would not be a credible source of significant secondary waste.

#### **E.1.2.3.11.5 Facilities**

The mixed TRU waste packaging process would include two CH-mixed TRU waste systems and one RH-mixed TRU waste system. Each CH-mixed TRU waste system would consist of approximately three mobile trailers, and the RH-mixed TRU waste system would be housed in a shielded facility with support structures. The two CH-Mixed TRU Waste Facilities would be used simultaneously in the 200-East and 200-West Areas, and the RH-Mixed TRU Waste Facility would be constructed in the 200-East Area. In the 200-West Area, the mobile CH-Mixed TRU Waste Facilities could be located at the T tank farm. In the 200-East Area, a mobile CH-Mixed TRU Waste Facility could be located at the B tank farm, and the RH-Mixed TRU Waste Facility could be located to the south of the AP tank farm. Potential locations of the facilities appear in Figure E-11. Final locations for these facilities have not been determined.

The waste packaging systems for CH-mixed TRU and RH-mixed TRU wastes would be similar and would include the following elements:

- Waste receipt tanks
- Dewatering
- Drum filling
- Decontamination
- Empty-container staging area
- Filled-container staging area
- Absorbent storage area
- Secondary confinement and air filtration
- Instrumentation and control



General descriptions of the CH-TRU and RH-TRU waste packaging systems and facilities are as follows:

- **Waste Receipt Tanks.** The waste receipt tanks would receive batch transfers of waste slurry from the MRS process skid. Waste stored in the two receipt tanks would be pumped with progressive cavity pumps to the dewatering system. Waste receipt tanks would have a nominal capacity of 5,700 liters (1,500 gallons) each.
- **Dewatering System.** The dewatering system would remove all drainable liquid from the waste by using, for example, a centrifuge. The system would be nominally rated at 11,300 liters (3,000 gallons) of dewatered waste per day. The dewatering system, drum-filling system, and decontamination system would be housed in a glovebox-like containment structure. Recovered supernatants would be recycled within the retrieval and dewatering systems. Excess water would be disposed of through the DST system or the ETF. Excess water would be transferred using tanker trucks or through pipelines.
- **Drum-Filling System.** The dewatered waste would be transferred to drums. Concurrent with the waste addition to the drum, absorbent material would be metered in to ensure that the final waste package would contain no free liquids.
- **Decontamination System.** The decontamination system would provide carbon dioxide pellets or other suitable media to facilitate container decontamination.
- **Empty-Container Staging Area.** The empty-container staging area would provide sufficient storage capacity for 20 empty containers. The drums would be stacked four to a pallet, three pallets high, on a concrete pad. A forklift would be used to transfer empty drums to the packaging system. This area of the facility would not be a radiologically controlled area (RCA). The staging area is expected to be a graded and graveled area with access control, but no weather covering would be provided.
- **Filled-Container Staging Area.** A single filled-container staging area would provide interim storage for filled waste drums. The storage area would accommodate 4 days of production. The staging area would be a concrete pad that is curbed and sloped with access controls and weather protection. Drums from the staging area would be loaded on trucks for shipment to the TRU Waste Interim Storage Facility.
- **Absorbent Staging Area.** The absorbent staging area would provide temporary storage for absorbent materials received by truck at the RH-Mixed TRU Waste Facility. The absorbent staging area would be a nonradiological operation adjacent to the waste packaging system.
- **Air Filtration System.** The air filtration system process would maintain negative air pressure within process vessels and areas. Air would flow from those areas with the least potential for contamination to areas with the highest potential for contamination. Air collected by the air filtration system would be passed through HEPA filters before being discharged to a stack. The air filtration system would include heaters to prevent condensation in the HEPA filters, redundant banks of HEPA filters, exhaust fans, and a stack with a continuous air radiation monitor and a gas sampler train.
- **Instrumentation and Control System.** The instrumentation and control system would provide process control and monitoring of waste packaging process equipment and support systems.

Designation and certification of packaged mixed TRU waste would include characterization of results, identification of procedures, preparation of reports, and conduct of interviews to properly collect the required data and systems information to support the TRU waste designation. These data would include WIPP-approved documentation, hazardous waste codes, acceptable knowledge, waste history, and isotopic and chemical constituent information. When necessary, sampling and analysis would be done to support the data packages. Certification and inspection would include the use of audited programs and trained personnel to perform the required certification and quality assurance work.

All CH-Mixed TRU Waste Facilities, except for waste storage pads, would be mobile and temporary. The RH-Mixed TRU Waste Facility was assumed to be a temporary, but not mobile, facility.

Other temporary facilities may be erected during construction of the RH-Mixed TRU Waste Facility. Typically, these facilities are limited to trailers used as construction site offices and change and restroom facilities for the construction workers.

Upon completion of processing, the mobile and temporary facilities would be deactivated, and the waste inventories would be removed to the extent practical, stabilized, and prepared for disposal.

#### **E.1.2.3.11.6 Waste Form/Disposal Package**

The waste would be pumped from the underground tanks into a dewatering unit (e.g., a centrifuge) that would produce both an MLLW liquid waste stream and a solid mixed TRU waste stream for packaging. The solid-waste stream would be mixed with an absorbent material to ensure that the final packaged waste would meet the WIPP Waste Acceptance Criteria. The blended product (waste and absorbent) would then be metered into drums or other approved containers.

The TRU waste would be processed and packaged in accordance with applicable regulations and waste acceptance criteria. The tank waste is considered mixed waste and would therefore be subject to regulation under RCRA. Appropriate RCRA permits for the packaging facilities would be obtained. All TRU waste packages would be designed as required to meet the WIPP Waste Acceptance Criteria.

#### **E.1.2.3.11.7 Assumptions and Uncertainties**

The mixed TRU waste packaging system would not require the development of any unique process equipment or systems. The technical approach to processing and packaging the subject waste is based on information from other sites within the DOE complex, where a substantial amount of TRU waste processing has already occurred. Disposal of the packaged waste would be in accordance with existing, well-understood requirements.

There is waste in certain HLW storage tanks that ORP believes are candidates for classification as TRU waste based on the origin of the waste. This *TC & WM EIS* evaluates the environmental impact of handling this tank waste as TRU waste under some Tank Closure alternatives, thus assuming the historical processing data to support this classification. However, DOE has not defined a process for making such determinations and has not decided whether WIPP can accept such waste. The TRU waste must meet the disposal criteria described in Section E.1.2.4.3.1.

The excess water from dewatering operations should be acceptable for treatment at the ETF. If it were unacceptable, this waste stream would be sent to the DSTs for processing in the WTP.

#### **E.1.2.4 Waste Post-Treatment Storage and Disposal**

Many waste disposal aspects of the proposed actions have been addressed in previous EISs. DOE has evaluated the programmatic aspects of waste management across the DOE complex (DOE 1997a). WIPP has an EIS that addressed transportation and disposal of a given waste quantity at WIPP (DOE 1997b). These EISs adopted assumptions and methodologies for assessing waste transportation to and disposal at the respective facilities and reported resultant environmental impacts. This *TC & WM EIS* was developed to be as consistent as possible with these adopted assumptions and methodologies to avoid contradictions in the impacts reported for overlapping activities.

##### **E.1.2.4.1 Immobilized High-Level Radioactive Waste Glass**

This section provides information used to assess the impacts of interim storage of IHLW glass in specially designed facilities on Hanford.

###### **E.1.2.4.1.1 Technology Description**

Based on the Preferred Alternative selected in the *TWRS EIS* (DOE and Ecology 1996) for the treatment of Hanford tank waste and a supplement analysis (DOE 1998), treatment at Hanford's WTP is expected to involve separation of the waste into high- and low-activity fractions by removal of selected radionuclides (e.g., cesium-137, strontium-90, and TRU waste). The high-activity portion of the waste would be vitrified in borosilicate glass and placed in welded stainless steel canisters measuring 0.6 meters in diameter by 4.6 meters high (2 feet in diameter by 15 feet high). Each IHLW glass canister would contain an average of 3.2 metric tons of glass IHLW. The low-activity fraction is discussed in Section E.1.2.4.2.

The IHLW glass canisters would be placed in interim storage on Hanford. A number of factors influence the volume of IHLW glass produced, and thus the number of canisters required, under each alternative analyzed in this *TC & WM EIS*. The number of canisters produced was assumed to be directly proportional to the percentage of waste retrieved from the tanks for processing. For the retrieved waste, the amount of waste that could be immobilized in a given volume of glass depends on the composition of the waste, the combination of glass-forming materials used, and the desired properties of the IHLW glass to be produced. For example, the presence of chromium in the waste at minor concentrations (e.g., less than 1 weight-percent chromium oxide by weight) could promote the formation of solid spinel phases that could create difficulties in processing the glass in typical joule-heated melters. Elevated quantities of aluminum, iron, and zirconium in the waste could also limit the processibility and durability of the final waste form. Other minor constituents such as sulfate have limited solubilities in glass at 1,050 to 1,150 °C (1,922 to 2,100 °F) and are therefore important factors in determining the amount of waste loading that could be achieved for the IHLW glass.

For analysis purposes in this *TC & WM EIS*, glass-property constraints that are less conservative than those previously employed were utilized. See Section E.1.2.3.1.7 for additional information. The resulting estimates of total IHLW glass volume are still considered to be conservative because of the glass-property predictions and assumptions regarding the extent of removal of constituents that influence the overall IHLW glass volume in pretreatment. The resulting estimates of IHLW glass volumes under each of the alternatives analyzed appear in Table E-12.

Table E-12. Summary Waste Masses/Volumes/Waste Containers for Tank Closure Alternatives

Tank Closure Alternative	Immobilized High-Level Radioactive Waste <sup>a, b</sup>			Immobilized Low-Activity Waste <sup>a</sup>			Bulk Vitrification Glass <sup>a</sup>			Cast Stone Waste <sup>a</sup>			Steam Reforming Waste <sup>a</sup>		
	Mass (MTG)	Glass Volume (m <sup>3</sup> )	No. of Cont. <sup>c</sup>	Mass (MTG)	Glass Volume (m <sup>3</sup> )	No. of Cont. <sup>d</sup>	Mass (MTG)	Glass Volume (m <sup>3</sup> )	No. of Cont. <sup>e</sup>	Mass (MT Grout)	Grout Volume (m <sup>3</sup> )	No. of Cont. <sup>f</sup>	Mass (MT Product)	Product Volume (m <sup>3</sup> )	No. of Cont. <sup>g</sup>
2A	38,400	14,220	12,000	553,510	212,890	92,250	–	–	–	–	–	–	–	–	–
2B	38,400	14,220	12,000	553,510	212,890	92,250	–	–	–	–	–	–	–	–	–
3A <sup>h</sup>	27,840	10,310	8,700	171,040	65,780	28,510	200-W: 100,350 200-E: 156,490	200-W: 40,140 200-E: 62,600	200-W: 2,360 200-E: 3,670	–	–	–	–	–	–
3B <sup>h</sup>	27,840	10,310	8,700	171,040	65,780	28,510	–	–	–	200-W: 181,870 200-E: 283,690	200-W: 90,940 200-E: 141,840	200-W: 9,090 200-E: 14,180	–	–	–
3C <sup>h</sup>	27,840	10,310	8,700	171,040	65,780	28,510	–	–	–	–	–	–	200-W: 101,950 200-E: 158,970	200-W: 101,950 200-E: 158,970	200-W: 45,310 200-E: 70,650
4h, i WTP: PPF Feed: Total:	27,840 6,730 34,570	10,310 2,490 12,800	8,700 2,100 10,800	171,040 1,100 172,140	65,780 420 66,200	28,510 180 28,690	200-W: 101,340	200-W: 40,540	200-W: 2,380	200-E: 287,540	200-E: 143,770	200-E: 14,380	–	–	–
5h, j, k	24,960	9,240	7,800	186,590	71,760	31,100	200-W: 91,490	200-W: 36,590	200-W: 2,150	200-E: 100,080 <sup>j</sup>	200-E: 50,040 <sup>j</sup>	200-E: 5,000 <sup>j</sup>	–	–	–
6A, Base Case <sup>l, m</sup>	548,260	203,060	171,330	– PPF Glass: 4,170	– PPF Glass: 1,600	– PPF Glass: 700	–	–	–	–	–	–	–	–	–
6A, Option Case <sup>l, m</sup>	548,260	203,060	171,330	– PPF Glass: 109,910	– PPF Glass: 42,270	– PPF Glass: 18,320	–	–	–	–	–	–	–	–	–
6B, Base Case <sup>k, m</sup>	38,400	14,220	12,000	557,990 PPF Glass: 4,170	214,610 PPF Glass: 1,600	93,000 PPF Glass: 700	–	–	–	–	–	–	–	–	–
6B, Option Case <sup>k, m</sup>	38,400	14,220	12,000	557,990 PPF Glass: 109,910	214,610 PPF Glass: 42,270	93,000 PPF Glass: 18,320	–	–	–	–	–	–	–	–	–
6C <sup>k</sup>	38,400	14,220	12,000	553,510	212,890	92,250	–	–	–	–	–	–	–	–	–

**Table E-12. Summary Waste Masses/Volumes/Waste Containers for Tank Closure Alternatives (continued)**

- a Per R. Wilson to B. Gannon email, dated June 5, 2003, "Preliminary Alt. 2B Glass Production Quantities (RPP System Plan, Rev. 1 Draft Conditions)" (Wilson 2003), the following values were used:
- Mass of IHLW glass: 3.2 metric tons of glass per canister (this assumes use of the thin-wall HLW canister)
  - Mass of ILAW glass: 6 metric tons of glass per container
  - Density of IHLW glass: 2,700 kilograms per cubic meter
  - Density of ILAW glass: 2,600 kilograms per cubic meter
- Per Mass Balance Calculation referenced above, the following values were used:
- Mass of bulk vitrification glass: 42.6 metric tons of glass per container
  - Mass of cast stone waste: 20.0 metric tons of grout per container
  - Mass of steam reforming waste: 2.25 metric tons of waste per container
  - Density of bulk vitrification glass: 2,500 kilograms per cubic meter
  - Density of cast stone waste: 2,000 kilograms per cubic meter
  - Density of steam reforming waste: 1,000 kilograms per cubic meter
- b The IHLW estimates exclude the estimated mass, volume, and number of canisters generated by the treatment of the cesium and strontium capsules in the WTP. Estimates for this waste stream are (ref: Data Request No. 199, Calculation Package WT-ST-053, Rev. 1, September 29, 2006) (CEES 2006c):
- Mass (metric tons of glass): 1,090
  - Volume (cubic meters): 400
  - Number of IHLW canisters: 340
- c Per DOE/ORP-2003-08, Rev. 0, the IHLW canisters are to be 0.6 meters in diameter by 4.5 meters long (DOE 2003c). Per Alternative 3 – Mass Balance Updated to *River Protection Project System Plan*, Rev. 1 – Case 3, Effective Date: May 21, 2003, Section 4.3, the waste loading for each canister was assumed to be 3.2 metric tons of glass, which reflects use of the thin-wall IHLW canister. Canister volume approximately 1.27 cubic meters. Glass volume per canister = 1.18 cubic meters (DOE 2003g). Per email, R. Wilson to M. Burandt, "Estimated IHLW Canister Reduction from 20 TRU Tanks," dated April 2, 2004, the number of IHLW canisters is to be reduced by 500 canisters to account for tank TRU waste (Wilson 2004). Tank Closure Alternative 4 includes an additional 2,102 (rounded to 2,100) IHLW canisters resulting from the treatment of BX and SX tank farms' "tanks, ancillary equipment removal, and deep soils removal" recovered in the PPF (ref: Calculation Package WT-ST-040, Rev. 5, August 27, 2010) (CEES 2010e).
- d Per DOE/ORP-2003-08, Rev. 0, the ILAW containers are to be 1.22 meters in diameter by 2.3 meters long and contain 6 metric tons of glass (DOE 2003c). Canister volume approximately 2.69 cubic meters. Glass volume/canister = 2.31 cubic meters. **Tank Closure Alternative 4** includes an additional 183 (rounded to 180) ILAW canisters resulting from the treatment of BX and SX tank farms' deep soil removal that is recovered in the PPF (ref: Calculation Package WT-ST-040, Rev. 5, August 27, 2010) (CEES 2010e). **Tank Closure Alternatives 6A, Base Case, and 6B, Base Case**, include 696 (rounded to 700) LAW containers resulting from the treatment of "deep soils removal" from the 12 SST farms. **Tank Closure Alternatives 6A, Option Case, and 6B, Option Case**, include 18,319 (rounded to 18,320) LAW containers resulting from the treatment of deep

- soils removal at the 12 SST farms and the six sets of cribs and trenches (ditches) in the B and T Areas. The waste is recovered and thermally treated in the PPF (ref: Calculation Packages WT-ST-040, Rev. 5, SAIC-updated December 2, 2010, and WT-ST-052, Rev. 5, SAIC-updated December 7, 2010) (SAIC 2010b, 2010c).
- e Per response to *Data Request 202 – Supplemental Treatment Data Review for the Tank Closure EIS*, dated April 17, 2006, the bulk vitrification containers are to be boxes with nominal dimensions of 10 feet × 24 feet × 8 feet, 71.1 cubic yards or approximately 54.3 cubic meters. Per the mass balances referenced above, glass mass per box = 42.6 metric tons of glass (CH2M HILL 2006b).
- f Per response to *Data Request 202 – Supplemental Treatment Data Review for the Tank Closure EIS*, dated April 17, 2006, the cast stone containers are to be boxes with nominal dimensions of 9 feet × 9 feet × 5 feet, 15 cubic yards or approximately 11.5 cubic meters. Per mass balances referenced above, grout volume per box = 10 cubic meters with a mass = 20 metric tons (CH2M HILL 2006b).
- g Per the mass balances referenced above, the steam reforming waste (assumed to be granular solids, not a grouted monolithic solid) containers would contain a waste volume of 2.25 cubic meters per container at 19.8 percent sodium oxide. At the density of 1,000 kilograms per cubic meter, each container's mass = 2.25 metric tons.
- h **Tank Closure Alternatives 3A, 3B, 3C, 4, and 5** include the treatment of tank TRU waste. This TRU waste would be treated in new treatment facilities and disposed of at WIPP. Waste in the following tanks is assumed to be managed as TRU waste (Certa et al. 2008):
- CH SST: B-201, B-202, B-203, B-204, T-201, T-202, T-203, T-204, T-104, T-110, T-111
  - RH SST: T-112, T-105, T-107, B-107, B-110, B-111
  - RH DST: AW-103, AW-105, SY-102

**Tank Closure Alternatives 1, 2A, 2B, 6A, 6B, and 6C** do not include the treatment of tank TRU waste and therefore do not assume disposal at WIPP. Under these Tank Closure alternatives, the waste in the 20 tanks would be treated as HLW.

Per the mass balances referenced above, the tank TRU waste mass, the waste volumes and numbers of containers (for Tank Closure Alternatives 3A, 3B, 3C, 4, and 5) are as follows (assumes 100 percent of the original SST TRU inventory is CH-TRU waste):

Tank Closure Alternatives	Mass (metric tons)	Waste Volumes (cubic meters)	Numbers of Containers
<b>3A, 3B, and 3C</b>	CH: 1,740 RH (SST): 2,210 RH (DST): 710	CH: 1,500 RH (SST): 1,620 RH (DST): 520	CH: 7,120 RH (SST): 7,710 RH (DST): 2,480
<b>4</b>	CH: 1,760 RH (SST): 2,230 RH (DST): 720	CH: 1,510 RH (SST): 1,630 RH (DST): 530	CH: 7,190 RH (SST): 7,780 RH (DST): 2,500
<b>5</b>	CH: 1,590 RH (SST): 2,010 RH (DST): 650	CH: 1,360 RH (SST): 1,470 RH (DST): 470	CH: 6,480 RH (SST): 7,010 RH (DST): 2,260

**Table E–12. Summary Waste Masses/Volumes/Waste Containers for Tank Closure Alternatives (continued)**

h (continued)

Per WT-ST-046, Rev. 0, dated April 27, 2004, *Optimized Drum Estimates for RH-TRU Single-Shell Tanks*, the CH-TRU and RH-TRU waste are to be packaged in 55-gallon (~ 0.21-cubic-meter) drums (CEES 2004). Each drum would be filled with 151 liters of sludge and 57 liters of sorbent material. Assuming the sorbent is mixed with the waste prior to loading of drums. Density of CH-TRU waste sludges = 1,230 kilograms per cubic meter. Density of dried RH-TRU waste sludges = 1,500 kilograms per cubic meter. Densities of sorbent/waste:

- CH-TRU waste:
- $40/55 \times 1,230$  kilograms per cubic meter +  $15/55 \times 1,000$  kilograms per cubic meter = 1,167 kilograms per cubic meter.
- RH-TRU waste:
- $40/55 \times 1,500$  kilograms per cubic meter +  $15/55 \times 1,000$  kilograms per cubic meter = 1,364 kilograms per cubic meter.

i The Tank Closure Alternative 4 totals for IHLW and ILAW include the tank waste and PPF feed contributions resulting from clean closure of the BX and SX tank farms (ref: Calculation Package WT-ST-040, Rev. 5, August 27, 2010) (CEES 2010e).

j Per the mass balances referenced above, Tank Closure Alternative 5 produces a sulfate grout waste to be disposed of on site with a density of 1,800 kilograms per

cubic meters; at a 90 percent waste retrieval, the mass, volume, and number of containers are respectively:

- Mass (metric tons) = 35,700
- Volume (cubic meters) = 19,840
- Number of containers = 6,120 (3.6 cubic meter capacity boxes, 90 percent filled)

This waste form is excluded from the cast stone estimates.

k Mass of IHLW for Tank Closure Alternative 6A is derived from Calculation Package WT-ST-025, Rev. 6, dated October 18, 2006. Mass =  $6.45 \times 10^8$  kilograms  $\times$  0.85 (ratio of IHLW mass, relaxed Glass Properties Model (GPM), *River Protection Project System Plan*, Draft Rev. 1 (Guidance) to GPM) =  $5.48 \times 10^8$  kilograms (CEES 2006d; DOE 2003g).

l **Tank Closure Alternatives 6A and 6B, Base and Option Cases**, produce  $8.82 \times 10^5$  cubic yards or approximately 147,000 shielded boxes of tanks/ancillary equipment that are to be managed as HLW and are to be disposed of off site. The number of shielded boxes assumes  $2 \times$  (swell factor) of the excavated volume without compaction.

m ILAW produced under **Tank Closure Alternative 6B, Base Case and Option Case**, is to be managed as IHLW and would be disposed of off site.

**Note:**

–Due to rounding, the values listed in this table may not calculate precisely.

–Table reflects the mass balances included with Calculation Package WT-ST-056, Rev. 2, dated March 14, 2007, *Revision of PCAL 17284-2 Mass Balance*, (CEES 2007) and “Mass Balance Calculation for Supplemental Treatment Technologies,” Rev. 10, for the following Tank Closure Alternatives: 2A, 2B, 3A, 3B, 3C, 4, 5, and 6B (CEES 2010d).

–Mass balances were not provided for Tank Closure Alternative 1, 6A, or 6C.

–To convert cubic meters to cubic yards, multiply by 1.308; kilograms to pounds, by 2.2046; liters to gallons, by 0.26417; meters to feet, by 3.281.

**Key:** 200-E=200-East Area; 200-W=200-West Area; CH=contact-handled; Cont.=containers; DST=double-shell tank; HLW=high-level radioactive waste; IHLW=immobilized high-level radioactive waste; ILAW=immobilized low-activity waste; LAW=low-activity waste; m<sup>3</sup>=cubic meter; MT=metric tons; MTG=metric tons of glass; PPF=Preprocessing Facility; RH=remote-handled; SST=single-shell tank; TRU=transuranic; WIPP=Waste Isolation Pilot Plant; WTP=Waste Treatment Plant.

#### **E.1.2.4.1.2 Facilities**

The facilities required for interim storage of IHLW glass are described in Section E.1.2.1.3. The current planned capacity of the CSB is 880 canisters of IHLW glass. Eventually, structures similar to the CSB would be constructed as storage modules, each able to hold 2,640 canisters. Each module would be the same as the first storage module to be constructed as part of the IHLW Shipping/Transfer Facility (DOE 2003e). The number of IHLW Interim Storage Modules to be constructed varies under each Tank Closure alternative evaluated in this *TC & WM EIS*, depending on the number of canisters produced.

#### **E.1.2.4.1.3 Transportation**

*Project W-464, Immobilized High-Level Waste Interim Storage Facility, Preliminary Design Report* (Colosi 2002) provides data regarding onsite transportation of IHLW glass canisters from the WTP to the IHLW Shipping/Transfer Facility. Based on possible site locations recommended for the IHLW Shipping/Transfer Facility, the longest onsite travel distance would be 7.7 kilometers (4.8 miles). Transported waste would be sufficiently shielded to comply with radiation-level restrictions (49 CFR 173).

#### **E.1.2.4.1.4 Disposal**

The Secretary of Energy has determined that a Yucca Mountain repository is not a workable option for permanent disposal of SNF and HLW. However, DOE remains committed to meeting its obligations to manage and ultimately dispose of these materials. The Administration has convened the Blue Ribbon Commission on America's Nuclear Future (BRC) to conduct a comprehensive review of policies for managing the back end of the nuclear fuel cycle, including all alternatives for the storage, processing, and disposal of SNF and HLW. The BRC's final recommendations will form the basis of a new solution to managing and disposing of SNF and HLW.

#### **E.1.2.4.2 Immobilized Low-Activity Waste Glass**

This section provides information used to assess the impacts of the transportation and disposal of ILAW glass generated under the various alternatives evaluated in this *TC & WM EIS*, including wastes from supplemental treatment of Hanford tank waste. This *TC & WM EIS* also addresses disposal of the ILAW glass produced by the WTP and the glass melters from the WTP that would be taken out of service and replaced.

##### **E.1.2.4.2.1 Technology Description**

Vitrification at the WTP, as well as one or more supplemental treatment technologies, could be used to process LAW currently stored in the Hanford tank farms. Supplemental treatment technologies being considered include bulk vitrification, cast stone, steam reforming, sulfate removal, technetium removal, solid-liquid separations, and mixed TRU waste processing and packaging.

Disposal capabilities would be required for the following waste forms that could be produced by the WTP and supplemental treatment technologies (DOE 2003f):

- **ILAW Glass.** This waste form is glass produced by the WTP. The ILAW glass packages are stainless steel cylinders (2.7-cubic-meter [3.5-cubic-yard] capacity) that have been filled with vitrified LAW, cooled, and sealed. The packages are approximately 2.3 meters in height and 1.22 meters in diameter (7.5 feet in height and 4.0 feet in diameter). The density of ILAW glass is assumed to be 2.6 metric tons per cubic meter with each package containing approximately 6 metric tons of glass.

- **Bulk Vitrification Glass.** This waste form is assumed to be a monolithic glass with a density of 2.5 metric tons per cubic meter. The waste would be placed into a roll-off box for disposal. The 54.3-cubic-meter (71.1-cubic-yard) roll-off box would contain 17 cubic meters (22.3 cubic yards) of glass waste form and the remaining box space would be filled with soil or sand.
- **Cast Stone Waste.** This waste form is grout with a cured density of approximately 2.0 metric tons per cubic meter. The grout would be cast in 11.5-cubic-meter (15-cubic-yard) containers.
- **Steam Reforming Waste.** This waste form was assumed to be a mineralized granular product suitable for packaging. The final product was assumed to have a density of 1.0 metric ton per cubic meter. The waste would be placed into steel containers for disposal. The 2.25-cubic-meter (2.9-cubic-yard) containers would weigh approximately 2.25 metric tons.

A pretreatment process for the WTP, sulfate removal, would generate about 20,000 cubic meters (26,200 cubic yards) of precipitated strontium sulfate blended with grout (sulfate grout waste). This waste form would be cast in 3.6-cubic-meter (4.7-cubic-yard) containers. This option would have the net effect of reducing the volume of ILAW glass produced in the WTP for high-sulfate waste feeds and is considered in the evaluation of Tank Closure Alternative 5.

In addition to the primary products (ILAW glass), the actions proposed in this *TC & WM EIS* would generate numerous LLW and MLLW streams as secondary waste. Further discussion of secondary waste can be found in Section E.1.2.4.5.

For analysis purposes, disposal of all ILAW glass and STP wastes was assumed to be on site in an IDF. See Section E.3.4 for a description and status of the IDF(s) evaluated in this *TC & WM EIS*.

#### **E.1.2.4.2.2 Transportation**

ILAW glass, STP wastes, and secondary waste (LLW or MLLW) would be transported from the WTP, supplemental treatment technology sites, or individual tank farms to onsite disposal facilities. Transport of these containers would be performed by the use of a lowboy trailer attached to a semitractor and loaded with a single waste container per shipment (Colosi 2002). Transported waste would be sufficiently shielded to comply with radiation-level restrictions imposed by Pipeline and Hazardous Materials Safety Administration shipment and packaging requirements (49 CFR 173). See Appendix H for a detailed description of the transportation analysis.

#### **E.1.2.4.2.3 Disposal**

Based on the waste form performance information provided in the *Waste Treatment and Supplemental Technology Data Package* (DOE 2003f), the expected release rates of contaminants from the ILAW glass should be less than indicated by information provided in the *Hanford Immobilized Low-Activity Waste Performance Assessment: 2001 Version* (Mann et al. 2001). (See Appendix M for details on waste form performance.)

Release rates from cast stone waste were based on diffusion of immobilized contaminants from the waste form under conditions of near-surface disposal at Hanford. Thus, parameters affecting the source term include items such as the time the release begins, the corrosion rate, and the waste form surface area.

Although the ILAW glass package that would be placed in an IDF would consist of a steel container holding the ILAW glass, the steel container was not assumed to provide a barrier to water intrusion after it is placed into the disposal facility (Mann et al. 2001:Section 3.5.3.1.2).



Alternatives for onsite disposal are evaluated in this EIS and discussed in detail in Section E.1.3.4. As a result of the evaluation of alternatives in this *TC & WM EIS*, different combinations of waste forms and volumes were considered.

#### **E.1.2.4.2.4 Assumptions and Uncertainties**

STPs would produce waste forms that meet regulatory requirements for disposal and disposal facility acceptance criteria.

#### **E.1.2.4.3 Tank-Derived Mixed Transuranic Waste**

This section describes information used to assess the impacts of mixed TRU waste that would be generated from actions taken under Tank Closure Alternatives 3 through 5.

##### **E.1.2.4.3.1 Technical Description**

Tank-derived mixed TRU waste is divided into two types: CH-mixed TRU and RH-mixed TRU waste. CH-mixed TRU waste has dose rates equal to or less than 200 millirem per hour on the surface of the package and can be stored in facilities similar to those used for TRU waste and LLW at Hanford. RH-mixed TRU waste has dose rates above 200 millirem per hour on the surface of the package and requires storage in facilities with shielding capabilities.

The technology analyzed in this EIS for mixed TRU waste retrieval is a vacuum-type method capable of retrieving waste from Hanford's SSTs and DSTs. After retrieval, the waste would be processed (dewatered and absorbent added) to yield CH- or RH-mixed TRU waste that is compliant with the WIPP Waste Acceptance Criteria described in the *Contact-Handled Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant* (DOE 2002a).

The quantities of CH- and RH-mixed TRU waste that supplemental treatment is estimated to produce are listed in the notes to Table E-12. Processing rates are expected to be two drums per hour for CH-mixed TRU waste, and one drum per hour for RH-mixed TRU waste.

##### **E.1.2.4.3.2 Facilities**

The TRU Waste Interim Storage Facility for CH- and RH-mixed TRU waste would provide the same capabilities as Hanford's CWC. The capabilities of the CWC are described in Section E.3.2.2. The TRU Waste Interim Storage Facility is described in Section E.1.2.3.11.4.

The shipment of mixed TRU waste generated at Hanford is limited by the receipt rates of WIPP. The CH-Mixed TRU Waste Facilities analyzed in this *TC & WM EIS* would operate from 2009 through 2010. The RH-Mixed TRU Waste Facility analyzed in this *TC & WM EIS* would operate from 2015 through 2019. The TRU Waste Interim Storage Facility would be operational for the storage of CH- and RH-mixed TRU waste from 2009 through 2034, pending transportation to the WIPP disposal facility. It was assumed all mixed TRU waste would be shipped off site by 2035, when WIPP is expected to close and discontinue receipt of TRU waste.

The facilities for treatment and storage of mixed TRU waste would obtain applicable permits. Furthermore, any onsite interim storage of mixed TRU waste would have to be permitted under WAC 173-303.

##### **E.1.2.4.3.3 Transportation**

Filled drums could be transported from the packaging site to the TRU Waste Interim Storage Facility adjacent to the RH-Mixed TRU Waste Facility. The longest distance used for evaluating transportation

impacts would be that from near the T tank farm, near the northern corner of the 200-West Area, to the storage facility located in the 200-East Area south of AP tank farm. The distance traveled between these sites is about 16 kilometers (10 miles). Drum transport could be by flatbed truck or a flatbed trailer attached to a semitractor. Depending on the concentration and quantity of radionuclides, TRU waste packages may require overpacks during transportation to provide additional safety or lower dose rates (DOE 2003e).

Type B containers certified by NRC would be used for transport to WIPP. Four packaging systems meet this criterion: TRUPACT-II containers; RH-72B (for RH-TRU waste); HalfPACT (for heavy packages of CH-TRU waste); and CNS10-160B (for RH-TRU waste, with plutonium limited to 20 grams [0.71 ounces] per shipment). Additional packages are under consideration. At this time, the WIPP transportation system is constraining the rate at which DOE sites can dispose of CH-TRU waste. Transported waste must be sufficiently shielded to comply with radiation-level restrictions imposed by Pipeline and Hazardous Materials Safety Administration shipment and packaging requirements (49 CFR 173).

#### **E.1.2.4.3.4 Disposal**

Eventual disposal of the mixed TRU waste would be performed at WIPP. The volumes of CH- and RH-mixed TRU waste estimated to be produced under each Tank Closure alternative are listed in Table E-12.

#### **E.1.2.4.3.5 Assumptions and Uncertainties**

- For analysis purposes, it was assumed that WIPP (or some other fully authorized facility) would be available and would have sufficient disposal capacity to eventually meet the demand for all CH- and RH-TRU waste generated at Hanford. Any differences in reported capacity and demand for disposal volume of TRU waste would be resolved without impact on Hanford's treatment, storage, and disposal operations.
- The WIPP original RH-TRU waste design capacity of 7,080 cubic meters (9,260 cubic yards) may not be realized because of original design assumptions and the delay of receipt of RH-TRU waste, according to the *National TRU Waste Management Plan, Corporate Board Annual Report* (DOE 2002b). Whether this shortcoming would affect the shipment of Hanford's RH-TRU waste is unknown.
- WIPP has a planned 35-year operational life; thus, it may not be available after 2034. Additionally, estimates of anticipated disposal capacity may not be accurate because of the unknowns inherent in waste characterization and remediation actions.

#### **E.1.2.4.4 Waste Treatment Plant Melters**

This section describes information used to assess the impacts of onsite transportation and interim storage of WTP melters that would be taken out of service. The LAW melters from WTP operations would be managed and disposed of as MLLW.

For analysis purposes in this *TC & WM EIS*, it was assumed that WTP HLW melters would be packaged within an overpack at the WTP to provide shielding and confinement. The overpack would be 5.29 meters long by 5.29 meters wide by 4.38 meters high (17.4 feet long by 17.4 feet wide by 14.4 feet high). The overpack containing a melter would weigh approximately 263 metric tons (Lowe and Haigh 2003). The dose rate at 30 centimeters (11.8 inches) from the overpack surface would be less than 16 millirem per hour. The radionuclide inventory within a WTP melter would be equal to or less than that provided in *HLW Melter Radioactive Inventory at End of Life* (BNI 2004).

The HLW melters have not been installed or operated, and a high degree of uncertainty exists about their operation and lifespan. Due to the uncertainty surrounding information about the melters at the WTP, this EIS assumed that the HLW melters would be stored on site.

The LAW melters from WTP operations are expected to be classified as MLLW and would be disposed of in an IDF. As such, WTP LAW melters would require treatment to meet land disposal restrictions under RCRA. The overpack would be 6.79 meters long by 9.38 meters wide by 4.86 meters high (22.3 feet long by 30.8 feet wide by 15.9 feet high). The weight of the overpack containing a melter is unknown, but it is expected to be significantly higher than that of the overpacked HLW melter (Lowe and Haigh 2003).

Tank Closure Alternatives 2 through 5 would require disposition of LAW melters that have been taken out of service in an IDF as MLLW. At an estimate of 31 LAW melters, Tank Closure Alternative 2B represents the maximum quantity of LAW melters that would be disposed of on site. Tank Closure Alternative 2B's quantity of melters would be driven by the expanded ILAW glass operations and vitrification treatment processes (SAIC 2010a).

#### **E.1.2.4.4.1 Onsite Transportation**

Conveyance of melters from the WTP to either interim storage (HLW melters) or the MLLW disposal facility (LAW melters) would require specialized equipment for handling these very heavy loads. This could be accomplished using either the WTP onsite transporter or the transporter equipment described in *Failed Melter Disposal – Alternative Generation and Analysis, and Decision Report* (Calmus and Baker 2001).

#### **E.1.2.4.4.2 Interim Storage**

The HLW Melter Interim Storage Facility is a storage pad that would provide staging and temporary storage for WTP melters until they could be removed for disposition and final disposal. The pad would consist of a reinforced-concrete slab at grade level. The pad would be sealed to prevent contaminants from entering the concrete and would be sloped to allow precipitation to collect in a trench located in the center of the pad. The collection trench would drain to a sump into which a portable sump pump could be installed as required. The area around the pad would be sloped to prevent rainwater from draining onto the pad. To facilitate cost-effectiveness and facility operations, the pad would be as close as possible to the WTP. The largest HLW Melter Interim Storage Facility footprint (Tank Closure Alternative 6A) would include seven pads, each measuring 38.3 meters wide by 38.3 meters long (126 feet wide by 126 feet long) and occupying a space envelope of 1,467 square meters (1,920 square yards). Under Tank Closure Alternative 6A, each HLW Melter Interim Storage Facility pad would be constructed as needed for the storage of HLW melters taken out of service. It was assumed that a new pad, up to the required seven, would be constructed every 20 years to support the operational life of the WTP for this alternative.

#### **E.1.2.4.4.3 Assumptions and Uncertainties**

- HLW melters would be placed in interim onsite storage until disposition decisions are made and implemented.
- LAW melters would contain residual ILAW and would be disposed of as MLLW. Adequate disposal capacity would be provided in an IDF, which is discussed in Section E.3.4.
- The number of HLW melters requiring disposition varies by alternative, depending on the duration of HLW vitrification operations. All HLW melters taken out of service would be interim-stored on site until disposition decisions are made and implemented. The HLW melter design life is 5 years per melter.

- The number of LAW melters requiring disposition varies by alternative, depending on the duration of LAW vitrification operations. LAW melters taken out of service would be disposed of in an IDF. The assumed LAW melter design life is 5 years.
- Melters would be overpacked in containers that provide confinement and shielding (DOE 2003e).

#### **E.1.2.4.5 Secondary Waste**

This section provides information used to assess the impacts of the transportation and disposal of secondary waste generated from construction, operations, deactivation, and closure activities under the various alternatives. Secondary waste is waste generated as a result of another activity. Secondary waste includes TRU waste, LLW, MLLW, hazardous waste, and nonhazardous waste.

##### **E.1.2.4.5.1 Secondary Transuranic Waste**

Secondary TRU waste (e.g., equipment, tools, filters, and empty containers) would be generated during waste retrieval and operation of tanks and treatment facilities. The secondary TRU waste would be treated if necessary, packaged, certified (in WRAP or a mobile facility), stored in existing facilities at the CWC pending transportation, and disposed of at WIPP. Projections for secondary TRU waste generation under each Tank Closure, FFTF Decommissioning, and Waste Management alternative are presented in Chapter 4.

Secondary TRU waste does not include the tank-derived TRU waste that would be generated by treating waste currently stored in the SSTs and DSTs. A discussion of tank-derived TRU waste is presented in Section E.1.2.3.11.

##### **E.1.2.4.5.2 Secondary Low-Level Radioactive Waste**

Secondary LLW includes personal protective equipment, tools, filters, empty containers, and contamination control materials (e.g., plastic, paper, wood) that become radioactively contaminated during construction, operations, deactivation, and closure activities, and would be disposed of on site in an IDF. See Chapter 4 for projected volumes under each Tank Closure, FFTF Decommissioning, and Waste Management alternative. The characteristics of this secondary LLW were derived from the Solid Waste Information and Tracking System (Friday and Sterling 2002) or the *Solid Waste Integrated Forecast Technical (SWIFT) Report, FY2006–FY2035 (SWIFT Report)* (Barcot 2005). Most of the LLW would be packaged in standard waste drums or waste boxes. Waste container transport would be by flatbed truck or a semitractor–flatbed trailer combination.

##### **E.1.2.4.5.3 Secondary Mixed Low-Level Radioactive Waste**

Secondary MLLW includes personal protective equipment, tools, filters, empty containers, and contamination control materials (e.g., plastic, paper, wood) from construction, operations, deactivation, and closure activities that have become contaminated with radioactive materials or a hazardous component subject to RCRA. In addition to the examples of MLLW cited above, it is anticipated that a secondary MLLW stream containing iodine-129 would be generated by the WTP offgas treatment system. MLLW would be disposed of on site in an IDF. See Chapter 4 for projected volumes under each Tank Closure, FFTF Decommissioning, and Waste Management alternative. The characteristics of this secondary MLLW were derived from the Solid Waste Information and Tracking System (*Tri-Party Agreement Databases, Access Mechanism and Procedures* [Friday and Sterling 2002]) or the *SWIFT Report* (Barcot 2005). MLLW would be treated using a combination of on- and offsite capabilities. Most of the MLLW would be packaged in standard waste drums or waste boxes. Waste container transport would be via flatbed truck or a semitractor–flatbed trailer combination.

Other MLLW would also be generated from closure activities. This waste would include soil excavated from around the tanks, ancillary equipment removed from tank farms, rubble, and tank material and soil that have been processed in a facility to remove some of the radioactive and chemical contaminants under the Tank Closure alternatives that consider clean closure (Tank Closure Alternatives 6A and 6B) or selective clean closure (Tank Closure Alternative 4). This waste would be disposed of on site in the new River Protection Project Disposal Facility (RPPDF), which would be similar to an IDF and would be between the 200-East and 200-West Areas. See Figure E-11 for the proposed location of the RPPDF. See Section E.3.5 for a description of the RPPDF.

#### **E.1.2.4.5.4 Hazardous Waste**

Hazardous waste generated during construction, operations, and closure would be packaged in U.S. Department of Transportation-approved containers, stored on site for a short time, and shipped off site to permitted commercial recycling, treatment, and disposal facilities. See Chapter 4 for projected volumes under each Tank Closure, FFTF Decommissioning, and Waste Management alternative.

#### **E.1.2.4.5.5 Nonhazardous Waste**

Any nonhazardous solid waste generated during construction, operations, and deactivation would be packaged and transported in conformance with standard industrial practice. Solid waste such as office paper, metal cans, and plastic and glass bottles that can be recycled would be sent off site for that purpose. Under the Tank Closure alternatives that generate large volumes of nonhazardous solid waste in the form of soils, the material may be stockpiled for use in closure activities. The remaining solid nonhazardous waste would be sent off site for disposal in a local landfill.

#### **E.1.2.4.5.6 Risks and Uncertainties Related to Treatment and Disposal of Secondary Waste**

As discussed above, there are risks and uncertainties associated with treatment and disposal of secondary waste produced by the WTP and the supplemental treatment technologies, particularly the impacts this waste may have on long-term human health at an IDF. To address these issues, DOE, along with the U.S. Environmental Protection Agency, Ecology, the Oregon Department of Energy, NRC, technical experts from the DOE national laboratories and academia, and private consultants, participated in a Hanford Site Secondary Waste Roadmap Workshop over the period July 21–23, 2008, to develop a roadmap. DOE's Office of Environmental Management describes roadmapping as "a disciplined, consensus building, analysis, solution development, and decisionmaking methodology that supports strategic programmatic and project planning." The objectives of the workshop were to develop a roadmap outlining the technical and programmatic steps necessary to design, develop, demonstrate, and accept a baseline waste form for the treatment and disposal of secondary waste associated with treatment of tank waste at Hanford.

The Secondary Waste Roadmap Workshop focused on the waste streams that are expected to contain the largest fractions of iodine-129 and technetium-99, which the IDF risk assessment analyses showed may have the largest contributions to the estimated IDF disposal impacts on groundwater. For example, the roadmapping effort focused on the scrubber/offgas treatment process, which would send liquids with technetium-99 to the ETF for treatment and solidification and subsequently dispose of the solidified waste in an IDF. In addition, the silver mordenite and carbon beds with captured iodine-129 would be packaged and disposed of in an IDF.

The workshop culminated in development of the following distinct program needs elements. Programmatic/regulatory needs include the following (PNNL 2009):

- Select and deploy Hanford tank waste supplemental treatment technology.
- Provide treatment capability for secondary-waste streams resulting from tank waste treatment.
- Develop consensus on secondary-waste form acceptance.

Technology needs are as follows (PNNL 2009):

- Define secondary-waste composition ranges and uncertainties.
- Identify and develop waste forms for secondary-waste immobilization and disposal.
- Develop test methods to characterize secondary-waste-form performance.

Chapter 7 and Appendix M of this EIS provide more-detailed discussion, sensitivity analysis, and potential mitigation strategies for the treatment and disposal of secondary waste.

### **E.1.2.5 Tank System Closure**

This section describes the technologies, facilities, uncertainties, and assumptions associated with closure of Hanford waste tanks. The alternatives evaluated in this *TC & WM EIS* include various options for closure of the tanks and related equipment. Tank Closure Alternatives 1 and 2A do not include closure of the tanks. Tank Closure Alternatives 2B through 6C include different options for closure. Options for closure vary depending on factors such as the extent of waste retrieval from the tanks and regulatory decisions regarding the method of closure (e.g., clean closure, landfill closure, combination of clean and landfill closure) (DOE 2003d).

#### **E.1.2.5.1 Tanks**

The largest components of the Hanford tank farm system are the tanks themselves. As noted previously, the tanks vary in terms of capacity and age, but have some similarities. All were constructed as underground storage tanks for radiation shielding purposes; all have access risers (pipes) that extend vertically from the tank dome to the tank farm surface; and all 100-series tanks have a 22.9-meter (75-foot) internal diameter.

Options for closure of the tanks themselves involve stabilization of the tanks and any residual waste for closure as a landfill or removal of the tanks and residual waste to regulatory standards for clean closure.

##### **E.1.2.5.1.1 Grout Fill**

Stabilizing residual waste and filling the tanks with grout is an option for closure of the tanks as a landfill.

#### **TECHNOLOGY DESCRIPTION**

This section describes the process and activities associated with filling the 149 SSTs with grout. Grout is a material formed from cement, fly ash, fine aggregate, sodium bentonite clay, and water to create a free-flowing material that can be used to fill the tanks after waste retrieval is completed. The grout hardens in the tanks to stabilize the residual waste and provide structural stability for landfill closure of the tank farms.

The tanks would be filled with grout in a series of lifts in two separate phases. Each lift would deposit a layer of grout within the tank. The time allotted between each would allow the added grout to set up (cure).

Phase 1 would involve initial grout placement to stabilize the residual waste heel that is expected to remain following retrieval. Materials called sequestering agents could be added to immobilize specific contaminants (technetium and uranium) in residual waste. Phase 2 would involve filling the remaining tank void space up to the tank dome. This would prevent long-term degradation of the tank farm surface barrier due to structural subsidence, minimize water infiltration, and discourage intruder access.

The fill material for the tanks was assumed to be similar to the cold-cap grout formulation developed by the U.S. Army Corps of Engineers for the Hanford Grout Vault Program. This formulation exhibits a rather low-hydration heat and is free-flowing, self-leveling, and designed to generate little or no free water during curing. This cold-cap grout formulation would include the following materials:

- American Petroleum Institute Class H cement (180 kilograms per cubic meter)
- American Society for Testing and Materials Class F fly ash (660 kilograms per cubic meter)
- Natural fine aggregate (100 percent passing No. 8 sieve) (780 kilograms per cubic meter)
- Sodium bentonite clay (23 kilograms per cubic meter)
- Water (330 kilograms per cubic meter)
- High-range, water-reducing admixture (0.17 kilograms per cubic meter)

A portable, continuous-mixing grout plant would be mobilized outside the RCA at a staging area adjacent to each tank farm. Portable generators would provide electric power, site services would provide water to the mixing plant, and dry-grout mix components would be trucked in from offsite suppliers. The portable grout plant would have two production lines, each with a nominal production rate of 57 cubic meters (75 cubic yards) per hour. The twin grout production lines could be run independently or simultaneously for a combined nominal grout production rate of 115 cubic meters (150 cubic yards) per hour. Both grout production lines would feed into a single pump. The grout would be pumped through a slickline (pipeline that would be used to deliver grout to the tanks) constructed between the grout pump and the tank riser. Scaffolding support may be required adjacent to the riser pit and other locations where the ground would not provide continuous support of the slickline. The grout slickline would cross the radiological control barrier. The slickline within the RCA would be constructed, maintained, and operated by Hanford forces; the slickline outside the RCA could be constructed, maintained, and operated by the grout vendor.

Following residual waste heel stabilization (Phase 1 grouting), the remaining tank void space would be filled with grout (Phase 2). Phase 2 grouting would be accomplished in the same manner as Phase 1 grouting, though lift heights would now be constrained by dissipation of heat generated from the setting grout, tank wall loading associated with hydrostatic pressures, and grout production capability.

The free-flowing grout would be placed into the tank through (and all the way up to) a riser (or risers, depending on the tank configuration) using a tremie (a long tube used to deliver grout into the tank) to limit the drop height. The grout would flow over and cover the residual waste.

Grout lift height limitations and required cure time between lifts would be determined during design. Lift heights would be monitored during the grouting process using the in-tank video system or the existing level gauges. Tank operating pressures would be controlled within specified limits using a portable exhauster. Air emissions would be controlled by HEPA filters on the portable exhauster and by installation of portable confinement structures, if required.

The temperature of the grout during curing would be managed within specified limits by controlling the grout formulation, lift height, and period between lifts. Monitoring instruments for in-tank temperatures would not be required for process control. The existing data collection system may be used to verify that in-tank temperatures are within specifications.

Once the specified volume of grout has been produced at the grout plant, the plant would be shut down and the volume of grout in the slickline would be delivered to the tank using the slickline cleaning procedure. Cleaning of the grout plant mixing chamber, pump, and slickline would be required at the completion of a grouting campaign or during extended delays. The crushed rock and grout mixture would be disposed of as solid waste at a nonradioactive, nonhazardous waste disposal facility at Hanford, and the wastewater would be treated and disposed of in compliance with Hanford facility regulations. Weather-sealed protective foam coating would then cover the exposed grout surfaces, which is the current practice for water intrusion prevention (DOE 2003d).

### RESIDUAL WASTE STABILIZATION

A volume of residual waste would remain in the tanks for closure. Physical stabilization of the residual waste would be the preferred approach for treatment. Grout has physical, as well as chemical, waste stabilization properties that would make it an effective technology for stabilization of residual waste. However, chemical stabilization using sequestering agents may also be considered if needed to further immobilize specific contaminants.

Sequestering agents may be used to chemically alter or bind potentially mobile contaminants within the grout matrix. Selection of sequestering agents would depend on the chemical form of the contaminant and its concentration in the residual waste.

For example, Sandia National Laboratories conducted bench-scale testing of inorganic sequestering agents (apatite and bone char) for immobilization of technetium and uranium in Hanford soils, groundwater, and simulated tank waste during fiscal year 2001. Technetium and uranium were expected to represent a major fraction of the radiological risk to groundwater. Apatite performed better in terms of irreversibly adsorbing both technetium and uranium. Very little desorption was detected. The apatite process requires a reducing agent (stannous chloride) (DOE 2003d). The success of this and other work has led to the consideration of apatite as a sequestering agent for tank waste residuals in enhanced grout formulations. Development work has focused on grout formulations that incorporate apatite for use in residual waste immobilization in support of the Accelerated Tank Closure Demonstration Project.

Residual waste can be physically stabilized in one of the following three ways:

- **Microencapsulation.** Residual waste would be mixed with the grout-fill material to achieve dispersion of waste within the grout matrix (as much as technically possible).
- **Macroencapsulation.** Residual waste would be displaced by the grout-fill material and sandwiched between a lower level containing sequestering agents and an upper layer.
- **Isolation.** Residual solid waste not displaced by the grout-fill material would remain adhered to the steel liner. The grout-fill material would minimize water infiltration through the waste and thereby reduce contaminant mobility.

The purpose of waste stabilization is to assist in reducing residual waste constituent mobility by physically isolating the residual waste from the environment and/or treating the waste chemically to reduce its mobility.

The *Accelerated Tank Closure Demonstration Alternatives Generation and Analysis Report* (Riess 2002) selected the use of a layer of low-strength grout for waste stabilization in the tanks. The *Engineering Report for Interim Closure for Tank No. 241-C-106 and the 241-C Farm 200-Series Tanks* (Fredenburg 2003) selected the addition of dry solids (granular material) to immobilize the waste heel and eliminate any freestanding liquid in the bottom of the tank. This selection was also based on lessons learned from tank closures at the SRS (DOE 2003d).



A small volume of free liquid may remain after waste retrieval from the tanks. The liquid waste fraction would be immobilized within the granular material; this layer would be covered with grout. Immobilization of the liquid waste is significant because it is expected to contain the highest concentration of mobile constituents. Further, the residual solid waste adhering to the steel liner of the tank bottom would be blanketed with the first grout layer. Although the solid waste would not be fully encapsulated, the overlying grout is expected to minimize contact with infiltration water and maintain a high pH environment that may be conducive to minimizing contaminant solubility. Chemical sequestering agents may be added (to the granular material and grout) as part of Phase 1. The Accelerated Tank Closure Demonstration Project funded development by the Savannah River Technology Center and Sandia National Laboratories during fiscal year 2003 of enhanced grout formulations that incorporate sequestering agents (e.g., apatite) to immobilize residual waste (DOE 2003d).

Physical encapsulation or stabilization is intended to reduce contact between the waste and the environment by isolating the contaminants and contaminated media. This can be accomplished on a macroscale where the residual tank waste would be encased, surrounded by, or sandwiched between solid media such as grout. The placement of grout over residual waste with the tank bottom beneath is also a form of waste stabilization, though less effective than entirely surrounding a waste volume with grout. In addition, the stabilization of residual waste can be accomplished by placing two grout lifts at the bottom of a tank. The initial lift would generally have a higher density than the residual waste, thereby displacing or floating it. Following partial curing of the initial lift, the second lift of grout would be placed over the portion of the residual waste displaced by the first lift.

A granular layer may be placed in the tank bottom following waste retrieval. The granular material may consist of sand, gravel, dry bentonite, absorbent zeolite material, apatite, or apatite blended with another granular material. The purpose of the aggregate material would be to wick and contain the liquid fraction of the residual waste. Zeolites have the added benefit of providing selective ion exchange (i.e., immobilization) for certain constituents (e.g., uranium and technetium-99). The use of dry bentonite (or similar swelling clay material) would create a low-permeability capping layer that physically incorporates the liquid waste fraction and effectively seals the solid waste fraction from future water infiltration. Coarser aggregate material may be added to increase the structural performance of the bentonite layer. Additional study is required to identify design parameters for using granular materials.

The system evaluated in this *TC & WM EIS* includes the following:

- **Waste Immobilization.** This process involves pneumatic delivery of dry powder to the wet tank heel.
- **Dry Granular Material.** The purpose of introducing the dry granular material would be to absorb and immobilize both the anticipated liquids and the nonadherent solids remaining in the tank bottom after completion of the waste retrieval effort.
- **Waste Stabilization.** This process involves delivery of the grout to the tanks by pumping the grout through a slickline to the tank.
- **Wet Grout.** A layer of free-flowing grout approximately 30 centimeters (12 inches) thick (Phase 1) would be placed into the tank (approximately 126 cubic meters [165 cubic yards] for 100-series tanks and 9.2 cubic meters [12 cubic yards] for 200-series tanks) through a single tank riser using a pipe extension to limit the drop height. The grout would mix with or encapsulate the residual waste not fixed to the bottom of the tank. Multiple grout lifts may be used to form the waste stabilization layer. Each successive grout lift would capture a portion of the unbound residual waste at the interface between the grout and residual waste.

- **Tank Fill.** The remainder of the void space in the tank would also be filled using a series of grout lifts until the entire tank (including the riser) is full of grout.

For the purpose of this *TC & WM EIS*, 30 centimeters (12 inches) of grout would be provided as a minimum to establish a cost basis for scaling to other volumes. The results of regulatory analysis and risk assessments would determine the ultimate volume of grout required to ensure protection of human health and the environment (DOE 2003d).

The performance of the stabilized residual waste form used for analysis in this *TC & WM EIS* was based on typical grouted waste performance without sequestering agents for any specific constituents added. Thus, releases from the immobilized waste form used in the analysis would be greater than those for actual waste forms where sequestering agents would be employed.

## **FACILITIES**

Temporary facilities would include mobile grout facilities such as a dry mix plant and a grout batch plant. Offices and bathroom, shower, and changing facilities may be set up to accommodate the employees required for operation of the grouting activity. Additionally, several material stockpiles for the various components of the grout mixture may be needed. The stockpiles may be covered with tarps to reduce airborne emissions caused by wind and other elements.

Existing grout plant infrastructure such as permanent storage silos may be used to supplement the temporary facilities described above.

## **ASSUMPTIONS AND UNCERTAINTIES**

- The grout-filling option could be readily implemented. This option would use standard construction equipment, commercially available materials, and portable dry mix and batch plant facilities for grout production.
- The long-term effectiveness of the sequestering agent technology to immobilize contaminants is unclear. It is also uncertain whether stabilization would yield a waste form with performance equivalent to LLW.
- The physical form of the residual waste that would remain after retrieval is uncertain. To minimize the amount of liquid or loose sludge pushed to the outer edge of the tank, it was assumed that dry granular material (possibly containing sequestering agents) would be added to the tank before adding grout. It is not known whether dry powder would be required for all tank waste or only for those tanks containing free liquids.
- Possible sequestering agents and their methods of application are currently uncertain. It was assumed that sequestering agents would be applied to the tank as an additive to the dry powder and to the wet grout materials. The sequestering agents may or may not be the same materials for the powder and the grout (DOE 2003d).

### **E.1.2.5.1.2 Tank Removal**

Details regarding the tank removal technology required to support clean closure of a tank farm or multiple tanks are described in Section E.1.2.5.3.2.

### **E.1.2.5.2 Ancillary Equipment**

This section addresses the disposition of ancillary equipment associated with tank farm operations under alternatives involving landfill closure or clean closure of the tank farms. For analysis purposes, WRFs would be closed as part of the SST system in the same manner as ancillary equipment.

Tank Closure Alternatives 2B, 3A, 3B, 3C, 5, and 6C in this *TC & WM EIS* would employ grout-fill stabilization of ancillary equipment. Tank Closure Alternative 4 would involve grout-fill stabilization of ancillary equipment associated with landfill closure of all tank farms except BX and SX. The BX and SX tank farms would be “clean-closed” with ancillary equipment removal. Tank Closure Alternatives 6A and 6B would involve clean closure of all SST farms, including removal of ancillary equipment. Tank Closure Alternatives 6A, Option Case, and 6B, Option Case, add removal of the contaminated soils caused by the liquid releases from the six sets of contiguous cribs and trenches (ditches). See Appendix D, Section D.1.5, for a discussion of the six sets of contiguous cribs and trenches (ditches). All Tank Closure alternatives except 1, 2A, and 5 assumed ancillary equipment outside the projected closure barriers would be remediated or removed; under these alternatives, this ancillary equipment would be left as is, with no remediation actions.

### **TECHNOLOGY DESCRIPTION**

Tank farm ancillary equipment includes MUSTs; tank-related equipment (pump pits, risers, in-tank equipment); the waste transfer system (pipelines, diversion boxes, valve pits); and miscellaneous facilities used to store, retrieve, or treat tank waste. For long-term impacts, the greatest consequences would result from abandoning the ancillary equipment in place with no remediation. The radionuclide and chemical inventories for this option are described in Appendix D, Section D.1.2, and are included in the analysis of long-term impacts. The following paragraphs include a breakdown of options for ancillary equipment closure considering short-term impacts, as well as treatment options for each category of such equipment:

- **MUSTs.** Generally, landfill closure of the MUSTs would be similar to landfill closure of the SSTs, particularly in terms of residual waste treatment and stabilization of the tank structure. Stabilization of the residual waste heel would be accomplished by adding grout. Grout would also be used for structural stabilization of the tank and as a barrier to water intrusion.
- **Diversion Boxes.** Diversion boxes provide secondary containment for leaking transfer piping jumpers and collection of fluids from transfer piping encasements. By design, the diversion boxes are not normally exposed to tank waste. However, a percentage of the diversion boxes have been contaminated by waste that has adhered to the internal concrete box surfaces.

For clean closure alternatives that require component removal, scabbling—a scabber is a metallic rotating device fitted onto a track hoe used to grind material from the surface of concrete—to a depth of 0.63 centimeters (0.25 inches) over the internal concrete surfaces would be performed.

For landfill closure, the diversion boxes would be filled with sand, gravel, or grout to prevent excessive soil subsidence. Not all of the diversion boxes would be within the projected landfill closure barrier and, as such, not all may be landfill-closed. Removal or in situ remediation of diversion boxes outside of the closure barrier was assumed.

- **Waste Transfer Piping and Encasements.** Underground waste transfer pipeline configurations include (1) direct-buried, single-walled pipelines; (2) steel pipelines in concrete encasements; and (3) double-walled pipelines providing double containment. There are many miles of pipe connecting tanks and ancillary equipment within a tank farm to processing facilities, laboratories, disposal sites (e.g., cribs, trenches, ponds), treatment facilities, and other tank farms. Not all of the waste transfer pipes would be within the projected landfill closure barrier and, as such, some

may be removed or remediated in place, and these are addressed in the cumulative impacts analysis.

- **Valve Pits and Tank Pits.** Similar in function to the diversion boxes, these pits may be contaminated because of transfer piping and jumper connection leaks. Although not primary-waste containments, a percentage of these pits have been contaminated with tank waste that has adhered to internal concrete surfaces.

For clean closure alternatives that require component removal, scabbling to a depth of approximately 0.63 centimeters (0.25 inches) over the internal concrete surfaces would be performed.

All but one of the valve pits and all of the tank pits would be landfill-closed (filled with grout to prevent excessive soil subsidence) within the projected landfill closure barrier. Removal or in situ remediation of the valve pit outside of the closure barrier may be required.

- **Flush Pits, Cleanout Boxes, and Leak Detection Pits.** Generally, these pits would be within the projected landfill closure barrier and are not expected to have significant levels of contamination.

For clean closure, both equipment and concrete would be removed. All contaminated debris would be disposed of in an IDF.

For the landfill closure options, the flush pits, cleanout boxes, and leak detection pits would be filled with grout to prevent excessive soil subsidence.

- **Tank Risers.** Access to the interior of the waste tanks is provided through vertical pipes that penetrate the tank roof and terminate either above grade or within an at-tank pit. Risers also provide a means for installation of in-tank equipment.

Landfill closure options would require filling the tank risers with grout to prevent excessive soil subsidence.

- **In-Tank Equipment.** In-tank equipment, installed in the tank risers, includes pumps, instrumentation trees, airlift circulators, and retrieval equipment. Except when required for retrieval operations, in-tank equipment would not be removed and would be landfill-closed. In-tank equipment that presents a preferential pathway for infiltrating water to reach stabilized residual waste would be filled with grout for stabilization. Grout fill is considered to have the greatest impacts for landfill closure of in-tank equipment.

- **Miscellaneous Facilities.** By design, the structure of the miscellaneous facilities would not normally be exposed to the tank waste. These structures would be within the footprint of the projected closure barrier and would be landfill-closed. Under the alternatives that include landfill closure, miscellaneous facilities would have contaminated equipment removed and would be demolished to grade level. The below-grade structures would be filled with grout to prevent excessive soil subsidence.

All contaminated debris would be disposed of in permitted mixed waste disposal facilities (an IDF) at Hanford. Ancillary equipment would be characterized and disposed of as appropriate.

## **ASSUMPTIONS AND UNCERTAINTIES**

- Sequestering agents, if required to meet performance objectives, would be incorporated into the formulation of the stabilization grout. The analysis of the stabilized-waste-form performance in this *TC & WM EIS* was based on typical grouted waste performance without sequestering agents for any specific constituents added.
- The residual waste inventory in the miscellaneous tanks and other ancillary equipment following waste retrieval operations would not preclude landfill closure or disposal of removed components as mixed waste in permitted onsite disposal facilities after DOE Order 435.1 and RCRA requirements are satisfied.
- In-tank equipment not required for retrieval operations would be landfill-closed in place within the tank and not removed.
- Some ancillary equipment would not be capped by the closure barrier and, as such, may not be closed under landfill closure.
- Grout-filling of pipes (if required) could be accomplished through access to existing pits; no tapping sleeves would be required. There is some uncertainty as to the adequacy of existing access points for grouting long lengths of pipe and lines that may be plugged.
- Closure activities involving ancillary equipment close to an SST would not proceed until stabilization activities for the SST were completed.
- Grout for the filling of tanks and ancillary equipment would be produced at either of the two portable grout plants located in the 200-East and 200-West Areas of Hanford and trucked to the local site for placement into the ancillary equipment.
- The outer walls of the double-walled pipes and encasements, which do not provide primary containment, would not require grout-filling to prevent subsidence (DOE 2003d).

### **E.1.2.5.2.1 Grout Fill**

Filling ancillary equipment with grout is considered to have the greatest impacts for landfill closure of ancillary equipment due to the additional expenditure of resources for this option.

## **TECHNOLOGY DESCRIPTION**

Filling the ancillary equipment with grout may be required for stabilization of residual waste, prevention of water infiltration, and structural stabilization to preclude subsidence of the closure barrier cap.

The following ancillary equipment may need to be filled with grout depending on the alternative:

- MUSTs, double-contained receiver tanks, vaults, and tank pits
- Diversion boxes and valve, flush, and leak detection pits
- Waste transfer piping and cleanout boxes
- Tank risers and in-tank equipment
- Below-grade portions of miscellaneous facilities

For placement of large volumes of grout, an overground grout delivery line (slickline) would be temporarily installed from a grout pump outside the tank farm RCA to ancillary equipment within an RCA. Grout would be trucked from the nearest portable grout plant and discharged into the hopper of

the grout pump. The grout would then be pumped and gravity-discharged into the ancillary equipment or into a second grout pump located within the RCA for pressure-injection grouting of the system.

For placement of smaller grout volumes, a self-contained unit with a continuously circulating holding tank may be used. Grout would be trucked from the nearest plant and discharged into the holding tank of the self-contained grouting unit.

## **FACILITIES**

Necessary temporary facilities would include (1) two portable grout plants located in the 200-East and 200-West Areas to service the closure needs for both ancillary equipment and SSTs; (2) grout delivery slicklines within the tank farm RCA for transferring grout from the pump to ancillary equipment; and (3) portable generators for operating the grout pumps.

## **ASSUMPTIONS AND UNCERTAINTIES**

- Dose rates from the contaminated ancillary equipment are based on *Single-Shell Tank System Description* (Field 2003) estimates, as referenced in the *Inventory and Source Term Data Package* (DOE 2003b). Although the *AX Tank Farm Ancillary Equipment Study* (Skelly 1998) cites much higher dose rates, available field data provided in the *Single-Shell Tank System Description* suggest that the values in the *AX Tank Farm Ancillary Equipment Study* are conservatively high.
- This analysis assumes that the pipelines can be grout-filled by accessing the nozzles in the various pits. Data regarding the flow distance of grout in pipes are not available; however, grout was pumped over 305 meters (1,000 feet) through a 13-centimeter (5-inch) slickline for closure of an underground waste tank at DOE's SRS.
- Grout filling of pipelines that were not used for transfer of waste was not included.
- Ancillary equipment lists were based on currently available data (DOE 2003d). These lists are likely to change as integration with other remediation efforts at Hanford matures.
- There is uncertainty associated with the quantities of secondary waste (e.g., lead used for shielding and abandoned equipment in pits and ancillary equipment).

### **E.1.2.5.2.2 Removal**

Removal of ancillary equipment is considered to have the greatest impacts for clean closure and partial clean closure due to the additional expenditure of resources for removal and disposal of equipment under this option.

## **TECHNOLOGY DESCRIPTION**

This section describes technology required for the complete removal of ancillary equipment. Ancillary equipment removal would be a necessary component under any alternative that includes near-surface soil or tank removal. Removal of the following ancillary equipment may be required for closure:

- MUSTs, double-contained receiver tanks, vaults, and tank pits
- Diversion boxes and valve, flush, and leak detection pits
- Waste transfer piping and cleanout boxes
- Tank risers and in-tank equipment
- Below-grade portions of miscellaneous facilities

The approach for ancillary equipment removal would involve segregating equipment and materials that have residual waste from materials with low levels of contamination and packaging them for proper disposal.

The ancillary equipment would be removed using shielded cab excavators with various end attachments for excavating, shearing, pulverizing, grappling, and scabbling. The components would be reduced in size and placed in appropriate packages for onsite permitted landfill disposal. All operations would be performed within actively ventilated enclosures.

Contaminated concrete would be decontaminated by removal of surface material using scabbling. Scabbling is expected to remove 95 percent of residual contamination. Size reduction of concrete structures would be performed using concrete pulverizer attachments on excavation equipment. Size reduction of metal items would be performed using shear attachments on excavation equipment.

Waste with radiation levels low enough to allow contact handling would be loaded into roll-off boxes in preparation for disposal. Waste requiring remote handling would be packaged in shielded disposal boxes. Soil would be packaged with the CH concrete debris to minimize void space within the disposal box. Packaged pipe, metal debris, and other contaminated debris would be grouted within the disposal box to minimize void space.

## **FACILITIES**

Portable enclosures with active ventilation similar to those currently used on Hanford would be erected locally over the site of remediation.

A portable grout plant may be erected near the disposal facility for void-space filling of disposal boxes.

## **ASSUMPTIONS AND UNCERTAINTIES**

- Dose rates from the contaminated ancillary equipment are based on *Single-Shell Tank System Description* (Field 2003) estimates, as referenced in the *Inventory and Source Term Data Package* (DOE 2003b). Although the *AX Tank Farm Ancillary Equipment Study* (Skelly 1998) cites much higher dose rates, available field data provided in the *Single-Shell Tank System Description* suggest that the values in the *AX Tank Farm Ancillary Equipment Study* are conservatively high.
- Ancillary equipment lists were based on currently available data (DOE 2003d; SAIC 2010a). These lists are likely to change as integration with other remediation efforts at Hanford matures.
- Five process pipelines are plugged with residual waste. There is a potential for this ancillary equipment to be classified as RH-TRU waste (DOE 2003d).
- It is uncertain whether removal of 0.63 centimeters (0.25 inches) from the internal surface of unlined pits would adequately decontaminate the pits for removal and disposal as MLLW.
- Primary Waste: For Tank Closure Alternatives 2B, 3A, 3B, 3C, and 6C, where 4.6 meters (15 feet) of assumed contaminated soil and ancillary equipment would be removed from SST farms BX and SX, the waste consists of contaminated ancillary equipment that would be removed and packaged for onsite disposal.

- Secondary Waste: Items that may become contaminated during removal of the contaminated ancillary equipment include hydraulic excavators with shielded cabs and attachments, concrete scabbler systems, front-end loaders, filter media, decontamination fluids, personal protective equipment, rags, and wipes.

### **E.1.2.5.3 Tank and Soil Removal**

Options for closure evaluated in this *TC & WM EIS* include removal of soil within the tank farms that is contaminated or suspected of being contaminated as a result of spills during routine tank farm operations or leaks from SSTs. Tank Closure Alternatives 2B, 3A, 3B, 3C, and 6C involve partial removal of soils from the BX and SX tank farms along with removal of ancillary equipment. The BX and SX tank farms were chosen for this option because (1) their tank waste inventories are well characterized and the nature and extent of past leaks and spills are documented; (2) their current in-tank inventories include substantial amounts of long-lived, highly mobile constituents and short-term health risks; and (3) they are located in both the 200-East and 200-West Areas.

Tank Closure Alternatives 4, 6A (Base and Option Cases), and 6B (Base and Option Cases) involve removal of the soils and ancillary equipment and removal of tanks and contaminated soils beneath the tanks for selected tank farms.

#### **E.1.2.5.3.1 Near-Surface Soil Removal**

Removal of near-surface soils was analyzed for the environmental impacts associated with the removal of localized areas of soil contamination. Based upon eventual soil characterization data, some tank farms may require less than 4.6 meters (15 feet) of soil excavation, while others may need deeper excavation; therefore, a 4.6-meter (15-foot) average was used for analysis purposes in this *TC & WM EIS*. The basis for extensive removal of near-surface soil considered in this *TC & WM EIS* involves closing the SSTs in place, with partial removal of the ancillary equipment and removal of the top 4.6 meters (15 feet) of soil.

### **TECHNOLOGY DESCRIPTION**

The confinement concept for extensive near-surface soil removal is provided in Figure E-25.

The SSTs and DSTs are found within 18 separate tank farms on Hanford. In addition to the tanks themselves, the tank farms include ancillary equipment such as MUSTs; waste transfer system components (pipelines, diversion boxes, valve pits); at-tank equipment (pits, risers, pumps, instrumentation); and miscellaneous components used in the treatment, transfer, or storage of tank waste. The ancillary equipment, along with the soil, would all be removed to a depth of 4.6 meters (15 feet) from the surface of the tank farm.

As an option, 4.6 meters (15 feet) of clean soil could be added over the top of the tank farms without first removing the contaminated soil. However, the impacts would be lower than those of the removal/replacement option evaluated.

It is assumed that the tanks and in-tank equipment would be stabilized before removal of near-surface soils. Closure of the ancillary equipment is described in detail in Section E.1.2.5.2. For analysis purposes, it was assumed all ancillary equipment is accessible and would be removed with the 4.6 meters (15 feet) of soil excavated.



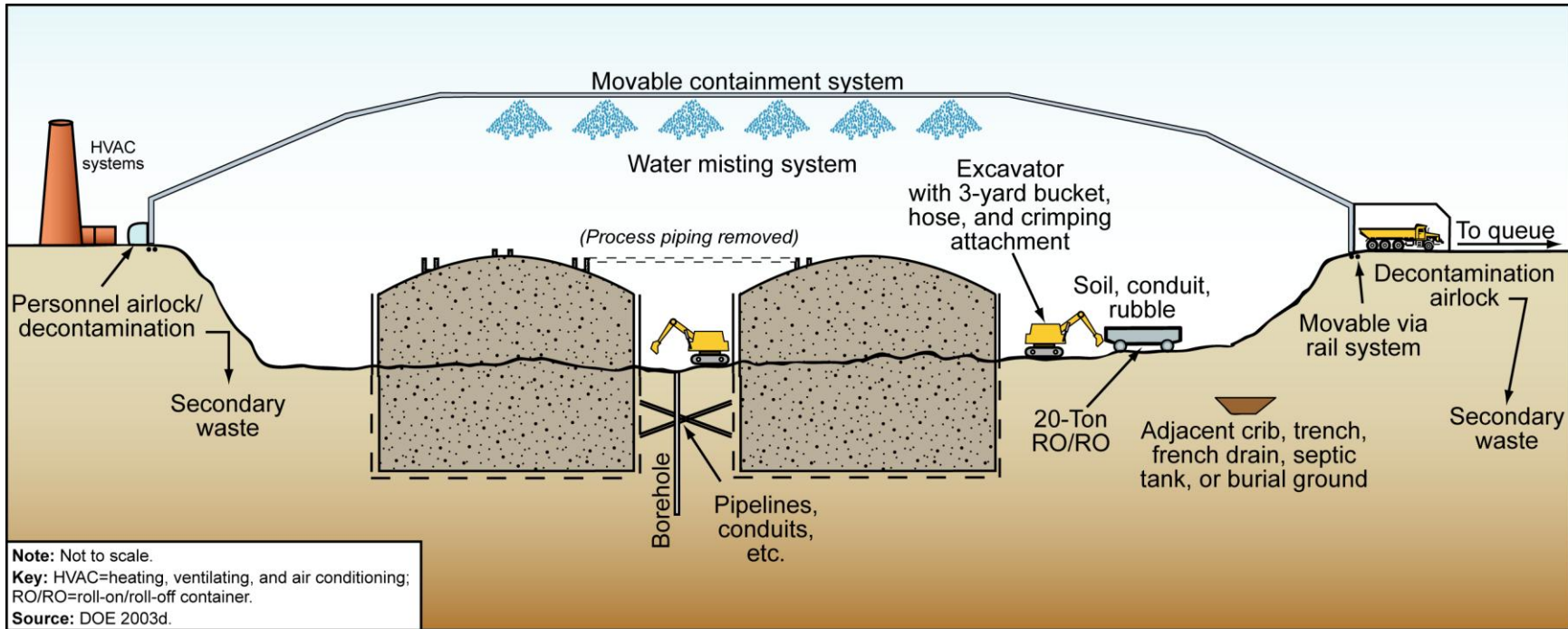


Figure E-25. Confinement Concept for Near-Surface Soil Removal

Hot spots are defined as near-surface sites of spills from piping or ancillary equipment with elevated levels of contamination. Tank farm closure actions could involve taking extra measures to minimize the source term remaining in the tank farm at the time of closure. The practical application of near-surface soil removal would focus on removal of hot spots. However, the characterization data are insufficient to accurately quantify the extent of contamination. Thus, the concept assumed was one that involves removal of all soil within the upper 4.6 meters (15 feet) of each tank farm.

Because of confinement requirements, the soil removal activity would be conducted within a confinement structure. Soil removal would require two large confinement structures. These structures are further described in Section E.1.2.5.3.2.

Excavated soil would be placed in large roll-off boxes for transport to the RPPDF.

Clean soil assumed to be provided from Borrow Area C would be used as backfill. The soil would be placed by heavy equipment, and soil compaction would result only from the equipment placing the soil.

## **FACILITIES**

An overarching confinement structure, or “bubble,” would be placed over all or part of a tank farm. The structure’s ventilation system would be equipped with filters and would have at least two zones of negative pressure, each with personnel and equipment air locks. The bubble would be used to keep fugitive dusts containing hazardous or radioactive particles from escaping to the environment. Shielded excavation and size-reduction equipment would be used under the bubble.

The site enclosure concept includes two connected structures. One is the large enclosure structure that would span the entire length and width of the excavation. The other would house support facilities for the operations. Site personnel and waste containers would enter and exit through air-lock access points within the connected support facility structure.

Temporary facilities would include the set of mobile grout plants to stabilize debris and/or soil in disposal boxes. These facilities would include a dry mix plant and a grout batch plant. Additionally, development of several material stockpiles for the various components of the grout mixture would be needed. The stockpiles would be covered with tarps to reduce airborne emissions caused by wind and other elements.

Equipment staging facilities, waste container and laydown areas, material storage areas, and waste storage areas would be required. Temporary greenhouse confinement structures would be set up as well.

A container storage building would be constructed to house containers filled with either soil or material/debris. This facility would store the containers until they were ready for transport to the disposal facility, a staging area, and a final decontamination and package preparation area.

In addition to the structures, utilities and other site services would be provided. These include power, water, exhaust blowers, HEPA filter housings, ducting, stacks, and fuel storage.

Temporary office and bathroom/shower/changing facilities would be set up to accommodate employees performing closure activities.

## **ASSUMPTIONS AND UNCERTAINTIES**

- The tanks would remain in place after waste retrieval. The waste residue heel remaining in the tanks would be stabilized, and the void space would be filled with a structural fill to prevent dome collapse.
- A large domelike structure spanning a tank farm would be erected to provide a means of confinement for the soil excavation activities around the tank farms.
- Soil and ancillary equipment would be removed to a depth of 4.6 meters (15 feet) from the surface. Decontamination, size reduction, and packaging of the soil and ancillary equipment for disposal would be performed within the confinement structure.
- Contaminated soil would be disposed of on site in the RPPDF. Excavated areas would be backfilled with clean or native soil.
- Groundwater, within the confines of the tank farm envelope, would be treated as part of the postclosure care of the tank farms.
- Soil removal would be performed under containment housing, either the large containment structure or smaller greenhouse-type structures. Safety would be a concern, as an effort of this scale has never been undertaken in the United States.
- All decontaminated concrete and waste debris that can be CH would meet onsite disposal (RPPDF) requirements without additional treatment.
- Borrow materials were assumed to be taken from Borrow Area C, near the intersection of Beloit Avenue and State Route 240.
- The quantity of material retrieved from the ancillary equipment is uncertain. Access to the piping and equipment components may be difficult because of insufficient waste retrieval. Liquid holdup may create hot spots.
- Additional disposal capability would be required to handle the anticipated waste volumes generated by closure activities. Resources required for this are accounted for in evaluating alternatives that involve soil removal.
- The enclosure structure identified for the 241-AX Tank Farm Closure Demonstration was a free-span structure that would cover an area 152 meters (500 feet) wide by 168 meters (550 feet) long. Most SST farms include three or four rows of tanks, and this enclosure concept may not be adaptable to these larger areas.
- Applying the 241-AX tank farm study data sets to all other tank farms is technically uncertain.
- Technologies used to excavate and treat soils would not be used until after the tank dome fill is completed to prevent dome collapse and to comply with existing safety load restrictions within tank farms (DOE 2003d).

#### E.1.2.5.3.2 Tank, Ancillary Equipment Removal, and Deep Soil Removal

Selective or full clean closure of the SST farms is analyzed under the following three Tank Closure alternatives:

- **Tank Closure Alternative 4:** The 27 SSTs in the BX and SX tank farms and the 3 meters (10 feet) of soil directly beneath these tanks would be removed. The highly contaminated rubble, soil, and equipment (RSE) from this removal action, which are expected to include tank steel, concrete, ancillary equipment, and contaminated soils, would be transported to the Preprocessing Facility (PPF) in shielded boxes and subjected to a strong acid wash. The resultant liquid waste stream would be neutralized and piped to the DSTs for treatment as HLW in the WTP, and the decontaminated (washed) RSE would be packaged and disposed of on site in the RPPDF. Similarly, the contaminated soils from deep soil excavation (tank leak plumes) would be removed, transported to the PPF in shielded boxes, and subjected to a weak acid wash. The resultant liquid waste stream would be neutralized and piped to the DSTs for treatment in the WTP as LAW, and the decontaminated (washed) rubble and soil would be packaged and disposed of on site in the RPPDF.
- **Tank Closure Alternatives 6A and 6B:** All 144 SSTs in the 12 SST farms and the 3 meters (10 feet) of soil directly beneath these tanks would be removed. The highly contaminated and moderately contaminated RSE from this removal action, which are expected to include tank steel, concrete, ancillary equipment, and contaminated soils, would be compacted to the extent practical and boxed in shielded storage boxes for disposal as HLW. To accommodate the long-term storage of these estimated 147,000 boxes, 35 covered concrete storage pads would be constructed near the PPF. Similar to Tank Closure Alternative 4, the contaminated soils from deep soil excavation (plumes) would be removed, transported to the PPF in shielded boxes, and subjected to a weak acid wash. However, unlike Tank Closure Alternative 4, this resultant liquid waste stream would be further treated in the PPF and immobilized by treatment in a glass melter system to produce an immobilized waste form that would be equivalent to ILAW glass in its long-term performance. This PPF glass would be disposed of in an IDF. The decontaminated (washed) rubble and soil from these PPF decontaminated operations would be packaged and disposed of on site in the RPPDF.

The above description applies to both the Base Cases and the Option Cases for Tank Closure Alternatives 6A and 6B. For the Option Cases, the deep soil wash activities in the PPF would include contaminated soils and rubble from the six sets of cribs and trenches (ditches) near the B/BX/BY and T/TX/TY waste management areas. See Appendix D, Section D.1.5, for a more complete description of these cribs and trenches (ditches); see Section D.1.7 for additional discussion and simplified mass balance flowsheets for Tank Closure Alternatives 6A and 6B.

Generic features of selective or full clean closure of these farms include the following:

- Routine work away from the dig face could be performed under the enclosure without respiratory protection. Numbers of workers involved and total exposure to radiological and industrial hazards would increase if supplied air were required.
- All contaminated material and soil would be disposed of either in an IDF or the RPPDF or would be placed in onsite interim storage and managed as HLW, as described above.

## **TECHNOLOGY DESCRIPTION**

The scope of the tank, ancillary equipment, and deep soil removal alternatives and associated technology would involve the complete removal of the tanks and ancillary equipment, the excavation of all soil in the tank farms down to the soil/groundwater interface, and disposal of the debris and soil. The excavation of all soil in the tank farms down to the soil/groundwater interface represents the most extreme implementation of these alternatives. DST farms and some SST farms may have considerable uncontaminated soils. Some would be directly disposed of as is, while others may require additional treatment to meet RCRA/dangerous waste disposal requirements (WAC 173-303). The resulting excavation would then be backfilled with clean soil from Borrow Area C.

The overall concept is depicted in Figures E-26 and E-27. A containment structure would be built over a group of tanks. Then the ancillary equipment, tanks, and contaminated soil down to the groundwater (70 to 76 meters [200 to 250 feet] below the ground surface in most tank farms) would be completely removed. The technology would require an overarching containment structure (or “bubble”) to be placed over all or part of a tank farm prior to tank or soil removal. The structure would be filtered and have at least two zones of negative pressure, each with personnel and equipment air locks.

The bubble would be used to keep fugitive dusts containing hazardous or radioactive particles from escaping to the environment. Shielded cab excavation and size-reduction equipment would be used under the bubble.

Use of a containment structure represents a reasonable and responsible case and the greatest impacts with respect to resource use. The Hanford environmental restoration contractor has been routinely cleaning up Columbia River corridor sites without the use of containment structures. A containment structure, such as that described herein, may be required for soils containing large inventories of contaminants (e.g., in tank farms where SSTs have leaked).

**Ancillary Equipment Removal.** All ancillary equipment components would be removed, reduced in size, packaged, and prepared for disposal. Whenever necessary, this work would be performed remotely. Concepts developed for demolishing and removing the ancillary equipment include using shielded heavy equipment and hydraulic excavators with specialized implements designed for demolition and material handling. To reduce the volume of heavily contaminated debris, a vacuum scabbler would be used to remove thin, highly contaminated layers of concrete. A truck-mounted vacuum system, similar to the system already in use in the tank farms, would be used for shallow excavations in areas that are otherwise limited to hand excavation.

Ancillary equipment removal concepts and information are taken from the *AX Tank Farm Ancillary Equipment Study* (Skelly 1998). The number of different types and the large number of pieces of ancillary equipment combine to make removal a substantial effort.

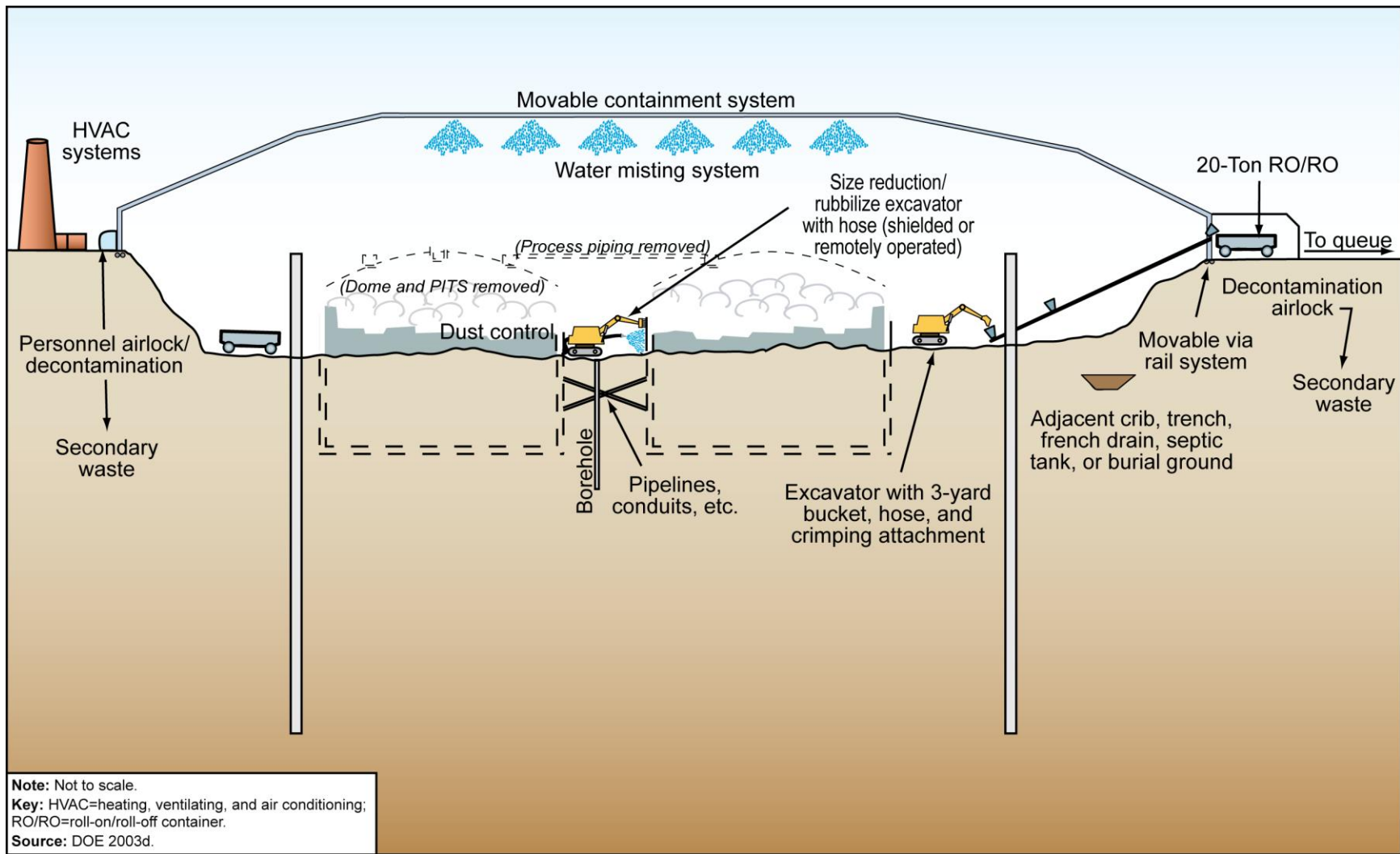


Figure E-26. Conceptual Drawing of Clean Closure Showing Domes and Pits Partially Removed

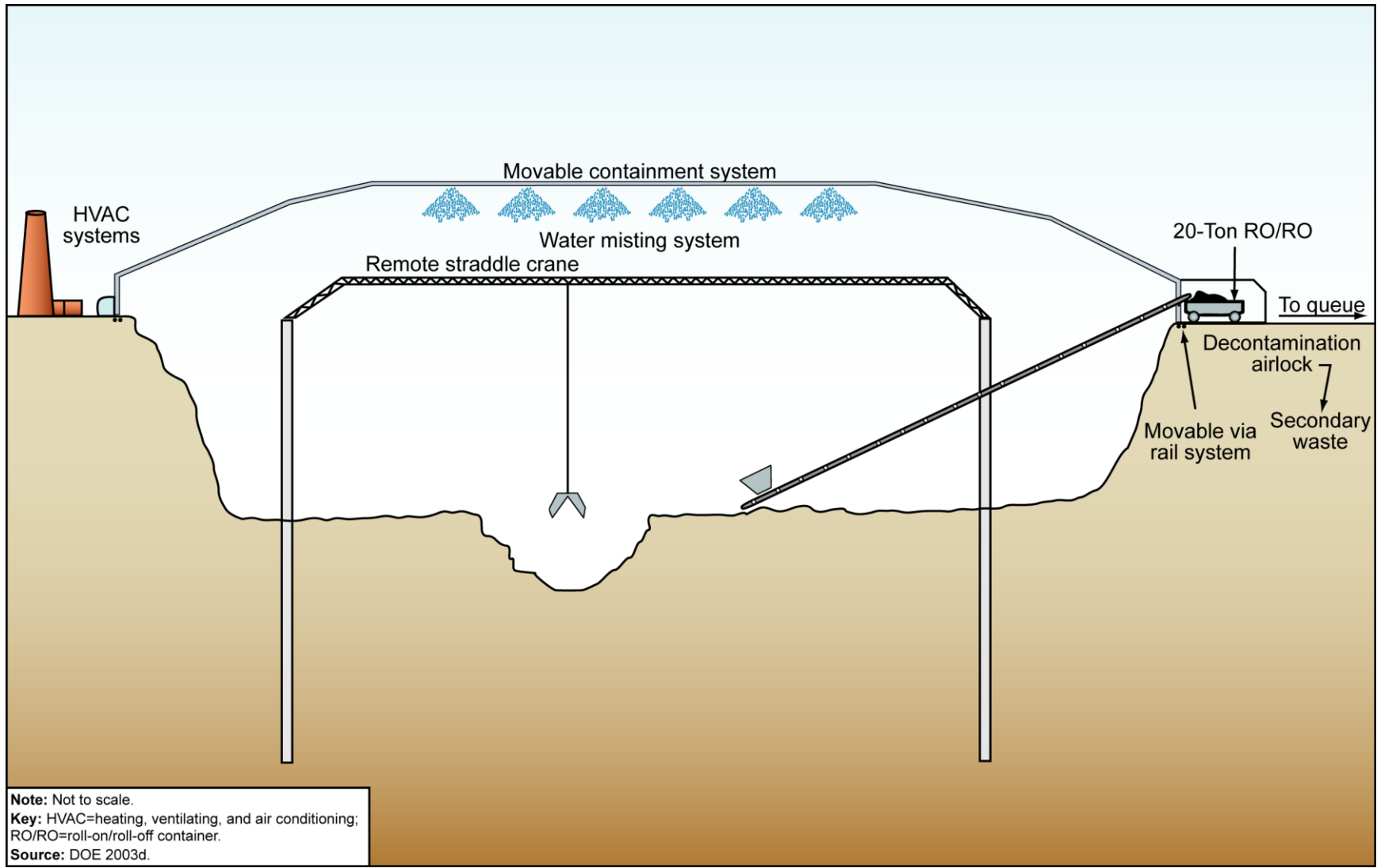


Figure E-27. Conceptual Drawing of Clean Closure After Removal of Tanks

**Tank Removal.** Removal of the tanks under Tank Closure Alternatives 4 (BX and SX tank farms only), 6A (Base and Option Cases), and 6B (Base and Option Cases) would involve the following activities:

- All contaminated tanks, ancillary equipment, and contaminated soils would be removed.
- Several large, movable enclosures spanning the width of each tank farm would be required for containment of radioactive and hazardous materials.
- The vadose zone and groundwater would be characterized to assess the nature and extent of contamination from losses during tank waste retrieval and to gather geotechnical data.
- Retaining walls would be installed. The outer wall would extend around the perimeter of the tank farm and would support excavation down to the tank base elevation. This wall would be installed before construction of the tank farm enclosure. The remaining walls would be installed following tank removal and would extend down from the tank base elevation to support excavation of tank leak plumes. A 30-centimeter-thick (1-foot-thick) layer of high-flow, low-strength grout would be placed in each of the tanks to reduce worker exposures from residual waste in the tanks.
- Soil cover would be removed from grade level down to the top of the tank domes. The excavated, lightly contaminated soil would be placed into containers for transport to and disposal in the onsite RPPDF.
- The tank domes would be demolished and the rubble placed inside the tank for additional radiation shielding.
- The soil surrounding the tanks would be excavated, debris would be placed in containers, and lightly contaminated soil would be transported to the onsite RPPDF.
- Tank sidewalls would be demolished and rubble placed in containers. Under Tank Closure Alternative 4, this waste would be transported for decontamination to the PPF. Under Tank Closure Alternatives 6A (Base and Option Cases) and 6B (Base and Option Cases), this waste would be packaged in shielded containers for disposition as HLW.
- The tank slab and footing would be demolished, and the debris and the 3 meters (10 feet) of soil under the tank slab would be placed in shielded boxes. Under Tank Closure Alternative 4, the highly contaminated portion of this waste would be decontaminated in the PPF. Under Tank Closure Alternatives 6A, Base and Option Cases, and 6B, Base and Option Cases, this waste would be packaged in shielded containers for disposition as HLW.
- Demolishing the base slabs would involve the most adverse working and material-handling conditions encountered during tank removal. Base slab demolition would involve the use of 5-centimeter-thick (2-inch-thick) sheets of steel laid down on the soil as the slab is removed to provide shielding and reduce worker radiological exposure rates to manageable levels (DOE 2003d).

**Deep Soil Removal.** After the tanks are removed, contaminated soil from past tank leaks would be excavated. To accommodate larger excavation volumes, a conveyor system would be constructed to move the soil from the excavation area to the container loading area. A straddle crane and rail system would be installed within the confinement facility to support soil excavation. The crane, with a clamshell attached, would be fully operable from a remote location and would use cameras to view the excavation area. The straddle crane would be mounted on three parallel sets of east-west rails supported by steel



girders. To increase the production by the crane, a belt conveyor system would be constructed to move the contaminated soil from the floor of the excavation to the container loading zone.

Lightly contaminated soil meeting waste acceptance criteria would be placed into roll-on/roll-off containers and transported to the onsite RPPDF for disposal. Soil with higher concentrations of radionuclides is expected immediately beyond the 3-meter (10-foot) layer of soil below the tank bases. These soils and other highly contaminated soils would be placed in shielded boxes and transported to the PPF for decontamination under Tank Closure Alternatives 4; 6A, Base and Option Cases; and 6B, Base and Option Cases (DOE 2003d).

Under Tank Closure Alternatives 6A and 6B, Option Cases, an additional volume of contaminated soil would be either decontaminated at the PPF (highly contaminated soil) or disposed of at the RPPDF (lightly contaminated soil). The sources of this volume and its radionuclide/chemical inventory are the six sets of contiguous cribs and trenches (ditches) in the B and T Areas. The balance of cribs and trenches (ditches) at Hanford are evaluated in Chapter 6, "Cumulative Impacts." Specifically analyzed are the following 33 cribs and trenches (ditches): 2 cribs in the B tank farm, 8 trenches in the BX tank farm, 7 cribs in the BY tank farm, 2 cribs in the T tank farm, 6 trenches (ditches) in the T tank farm, 5 trenches (ditches) in the TX tank farm, and 3 cribs in the TY tank farm. (Note: T and TX trenches are considered one set.) The total estimated volume of soil added by these six sets of cribs and trenches (ditches) is approximately  $6.50 \times 10^6$  cubic meters ( $8.50 \times 10^6$  cubic yards). See Appendix D, Section D.1.5, for additional discussion on the cribs and trenches (ditches).

**Preprocessing/Packaging Contaminated Soil and Debris.** Under Tank Closure Alternative 4, more than 310,000 cubic meters (405,000 cubic yards) of soil and debris recovered from the ancillary equipment, tank, and deep soil removal efforts are expected to be heavily contaminated with tank waste. Because this material would likely exceed waste acceptance criteria for onsite disposal, the material would be treated at a standalone, 4-hectare (10-acre) PPF (DOE 2003h). In the PPF, the highly contaminated tank debris, equipment, soil, and rubble from tank removal would be treated using a strong acid wash. The washed soils and debris would be packaged and disposed of on site in the RPPDF. The contaminated liquid waste stream from the acid wash would be neutralized and sent to the DSTs for treatment in the WTP. The contaminated soils from deep soil excavation would be treated in the PPF using a weak acid wash. The washed soils would be disposed of on site in the RPPDF, and the contaminated liquid waste stream from the acid washing would be sent to the DSTs for treatment in the WTP.

Under Tank Closure Alternatives 6A, Base and Option Cases, and 6B, Base and Option Cases, more than 670,000 cubic meters (880,000 cubic yards) of tank debris, equipment, soil, and rubble from tank removal are expected to be heavily contaminated with tank waste. However, these materials would be considered HLW and would be packaged in approximately 147,000 shielded storage boxes for disposal as HLW. To accommodate the shielded storage boxes, 35 covered concrete pads would be constructed near the PPF (SAIC 2010a). It was assumed that the boxed HLW would require long-term onsite storage until disposition decisions are made and implemented. It was also assumed that the radionuclide and chemical inventories in this waste would be contained during onsite storage. Therefore, this waste would not represent a source of groundwater contamination. Highly contaminated soil removed from deep soil excavation would be treated in the PPF using a weak acid wash. The washed soils would be disposed of on site in the RPPDF, and the contaminated liquid waste stream from the acid washing would be further treated in the PPF using a glass melter. The PPF melter would produce an immobilized waste form that would be equivalent to ILAW glass in waste form performance. Under Tank Closure Alternatives 6A and 6B, Base Cases, the volume of PPF glass produced would fill approximately 700 canisters, and under the Option Cases, the volume would fill approximately 18,320 canisters. This PPF glass would be disposed of on site in an IDF.

## FACILITIES

The facilities involved in the removal of tanks, ancillary equipment, and deep soil would include the containment structures, retaining walls for excavation, and the PPF. Adequate space must also be available for the disposal or interim storage of excavated soil and debris.

Construction-related activities would involve setting up field offices and utilities; obtaining characterization data; preparing the site; and erecting the containment structure, its support facility, and the PPF.

Under Tank Closure Alternatives 6A and 6B, Base Cases, six containment enclosures would be required. The enclosures would be around the following sets of tanks:

- T, TY, and TX tank farms
- U tank farm
- S and SX tank farms
- B, BY, and BX tank farms
- C tank farm
- A and AX tank farms

Under Tank Closure Alternatives 6A and 6B, Option Cases, four additional containment enclosures would be required because many of the cribs and trenches (ditches) are located in the B and T Areas (DOE 2003d).

Under all three Tank Closure alternatives involving “clean closure” (4, 6A, and 6B), the PPF would be a soil-washing facility. However, the original PPF was designed to clean equipment, not soils, and thus required a strong acid (57 percent nitric acid) solution to etch the material being treated. Because no etching would be required of the PPF for soil-washing purposes, a weaker acid (3 percent) would be adequate to treat contaminated soils. This weaker acid concentration would be comparable to those used in hydrometallurgy industries to remove the majority of removable contaminants. For Tank Closure Alternative 4, an additional equipment treatment system would be included because this alternative would involve treating the highly contaminated RSE generated from the removal of the SST and ancillary equipment. This additional treatment system would use a strong acid wash to remove 85 percent of the residual contamination to permit packaging and disposal of the decontaminated debris and soils in the RPPDF. Under Tank Closure Alternatives 6A and 6B, this highly contaminated waste stream would be packaged in shielded HLW containers for interim onsite storage.

The size of the PPF varies among the Tank Closure alternatives to optimize the facility’s throughput. Under Tank Closure Alternative 4, which excludes a PPF melter but would include both a strong and weak acid wash system, the estimated footprint of the facility would be approximately 13,000 square meters (140,000 square feet). This facility size would support a continuous feed to the HLW and LAW melter operations at the WTP. Under Tank Closure Alternatives 6A and 6B, Base Cases, the same facility footprint of 13,000 square meters (140,000 square feet) would be adequate to support the weak acid wash operations with one PPF melter. Estimated annual throughput would be 2.5 metric tons of glass per day. Under the Option Cases, the facility footprint would be approximately double, or 26,000 square meters (280,000 square feet), which would support six melters operating simultaneously for an estimated throughput of 15 metric tons of glass per day (CEES 2010e; SAIC 2010b, 2010c).

## **ASSUMPTIONS AND UNCERTAINTIES**

- Because an effort of this scale in a radioactive environment has not been undertaken in the United States, it is unclear whether the operation could be conducted with adequate considerations for worker safety. Although the technology related to the installation of movable containment structures is clearly understood, some uncertainty exists regarding the feasibility of installing these structures over the tank farms.
- For analysis purposes, it was assumed that 99.9 percent of the waste would be removed from the tanks, before tank removal operations begin, but leakage of 15,140 liters (4,000 gallons) per SST would occur during retrieval of waste from the SSTs.
- Tank removal would include removal of all contaminated soils in a tank farm, including excavation of plumes to groundwater, if necessary.
- The equipment used for soil excavation and tank removal would require additional radiation shielding to protect the equipment operators.
- Tank base materials are expected to have high radionuclide concentrations, so a new design for RH equipment would be required.
- An uncertainty associated with the PPF would be related to remote operation of the debris- and soil-cleaning process.
- The PPF process is an immature design with very little data available and would probably change as more-detailed design efforts are undertaken (DOE 2003d).

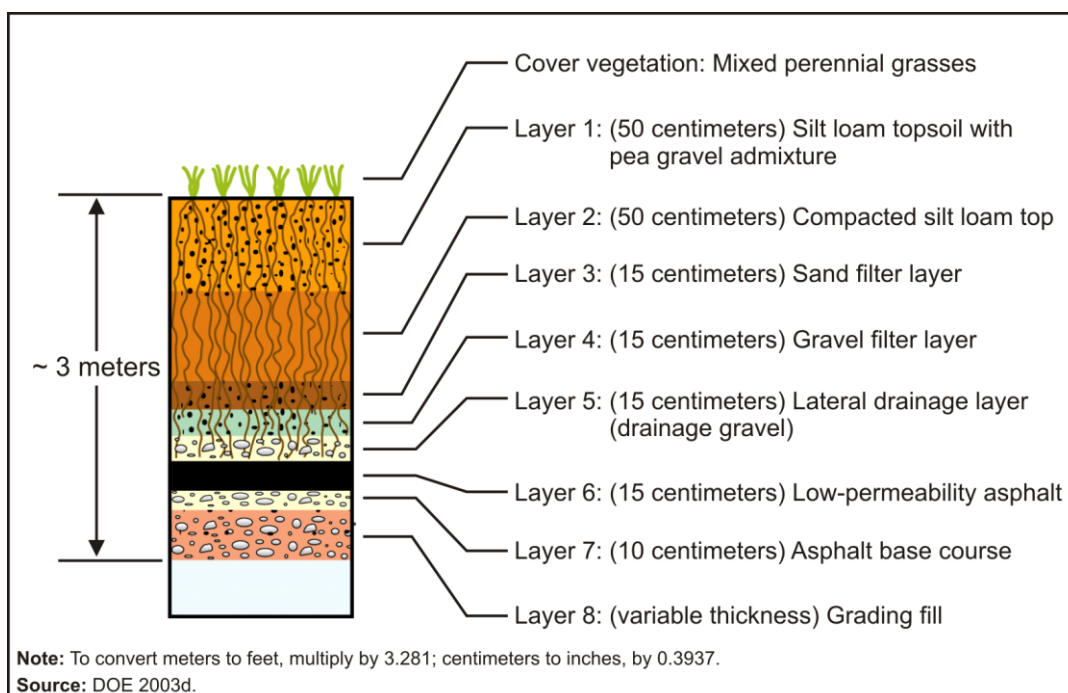
### **E.1.2.5.4 Tank Farm Closure System**

The tank farm closure system combines surface barrier technology with postclosure care. Installation of surface barriers and monitoring systems was included in the analysis of impacts under Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, 5, and 6C in this *TC & WM EIS*, all of which involve closure of the tank farms as landfills.

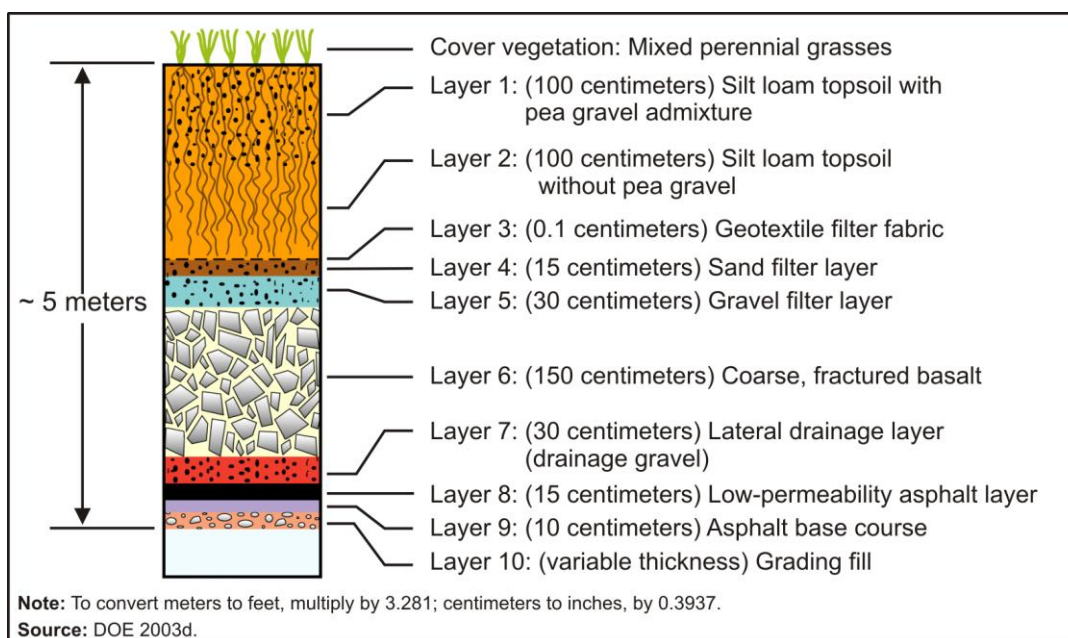
#### **E.1.2.5.4.1 Surface Barriers**

The reference surface barrier discussed here is the modified RCRA Subtitle C barrier. Geologic resource requirements for the modified RCRA Subtitle C barrier are essentially 40 percent less than those required for the Hanford barrier. This is due primarily to the difference in barrier installation depth. The modified RCRA Subtitle C barrier is an eight-layer barrier with a combined thickness of 2.7 meters (9.0 feet) versus 4.6 meters (15 feet) for the Hanford barrier and, also unlike the Hanford barrier, has no basalt riprap layer.

Substantial research and testing have been conducted on the Hanford barrier as part of a CERCLA treatability study. For analysis purposes, it was assumed that the Hanford barrier is designed for 1,000 years of performance without maintenance to control water infiltration; plant, animal, and human intrusion; and wind and water erosion. Performance results of both barrier prototypes would be evaluated further to confirm their suitability for closure of the tank farms as landfills. Cross sections of both types of barriers are shown in Figures E-28 and E-29. Both barrier designs are described in greater detail in the *Focused Feasibility Study of Engineered Barriers for Waste Management Units in the 200 Areas* (DOE 1996).



**Figure E-28. Modified Resource Conservation and Recovery Act Subtitle C Barrier**



**Figure E-29. Hanford Barrier Profile Technology Description**

All alternatives that use landfill closure as part of their proposed activities, except Tank Closure Alternative 5, entail the construction of a modified RCRA Subtitle C barrier. Tank Closure Alternative 5 entails the construction and operation of the Hanford barrier design.

### TECHNOLOGY DESCRIPTION

The modified RCRA Subtitle C barrier would cover each tank farm and the ancillary equipment within it. This barrier is designed to provide long-term containment and hydrologic protection for a performance

period of 500 years. The design incorporates an asphalt layer to inhibit both bio-intrusion (plants and burrowing animals) and human intrusion.

The Hanford barrier is a 10-layer design with a combined thickness of 4.6 meters (15 feet). It is designed to remain functional for a performance period of 1,000 years and to provide maximum containment and hydrologic protection. This barrier includes a layer of coarse, fractured basalt intended to provide primary bio-intrusion control and to limit inadvertent human intrusion. Criteria common to both barrier designs include the following:

- Minimize moisture infiltration through the cover.
- Design a multilayer cover of materials resistant to natural degradation processes.
- Design a durable cover requiring minimal maintenance during its functional design life.
- Prevent plants from accessing and mobilizing contaminants (i.e., prevent root penetration into the waste zone).
- Prevent burrowing animals from accessing and mobilizing contamination.
- Facilitate drainage and minimize surface erosion by wind and water.
- Design the low-permeability layer of the cover to have a permeability less than or equal to any natural subsoil present.
- Design the cover to prevent the migration and accumulation of topsoil material within the lateral drainage layer.

A conceptual representation of the five surface barrier “lobes” required for closure of Hanford tanks is presented in Figures E-30 and E-31. (A lobe is a section of a barrier that covers a tank farm or an area of contiguous tank farms.) Three barrier lobes are anticipated in the 200-West Area and two much larger lobes in the 200-East Area, including a large lobe over the C tank farm and the A tank farm complex. All contaminated areas under these five lobes are considered in this analysis. The perimeter of each lobe is considered to be the maximum distance within which soils would be removed under alternatives that call for either near-surface or deep soil removal.

All SSTs, ancillary equipment, facilities, and those existing DSTs within the barrier lobe area would be removed or isolated and stabilized prior to barrier construction. WTP closure is not part of the proposed actions because it is an active component needed to complete waste treatment. The existing 28 DSTs, also active components, are included in the closure scenario for each alternative presented in this *TC & WM EIS* that includes landfill closure. When the closure barrier is placed over the SSTs, it would also need to cover proximal DSTs, given the engineering design and proximity of the DSTs to the SSTs. Therefore, the decision was made to include existing DSTs in the closure configuration. In contrast, new or replacement DSTs proposed as part of the infrastructure required to support certain alternatives would not be closed, as these new DSTs would be physically separate from the original 177 tanks (149 SSTs and 28 DSTs) built at Hanford and outside the real extent of the SST closure barriers. Although a closure configuration for the DSTs is evaluated in this EIS, this EIS is not intended to support a decision on the closure of the new or replacement DSTs. Closure decisions related to both the new or replacement DSTs and the WTP would have to be addressed at a later date after appropriate NEPA analysis. See Figure E-11 for the locations of proposed facilities.

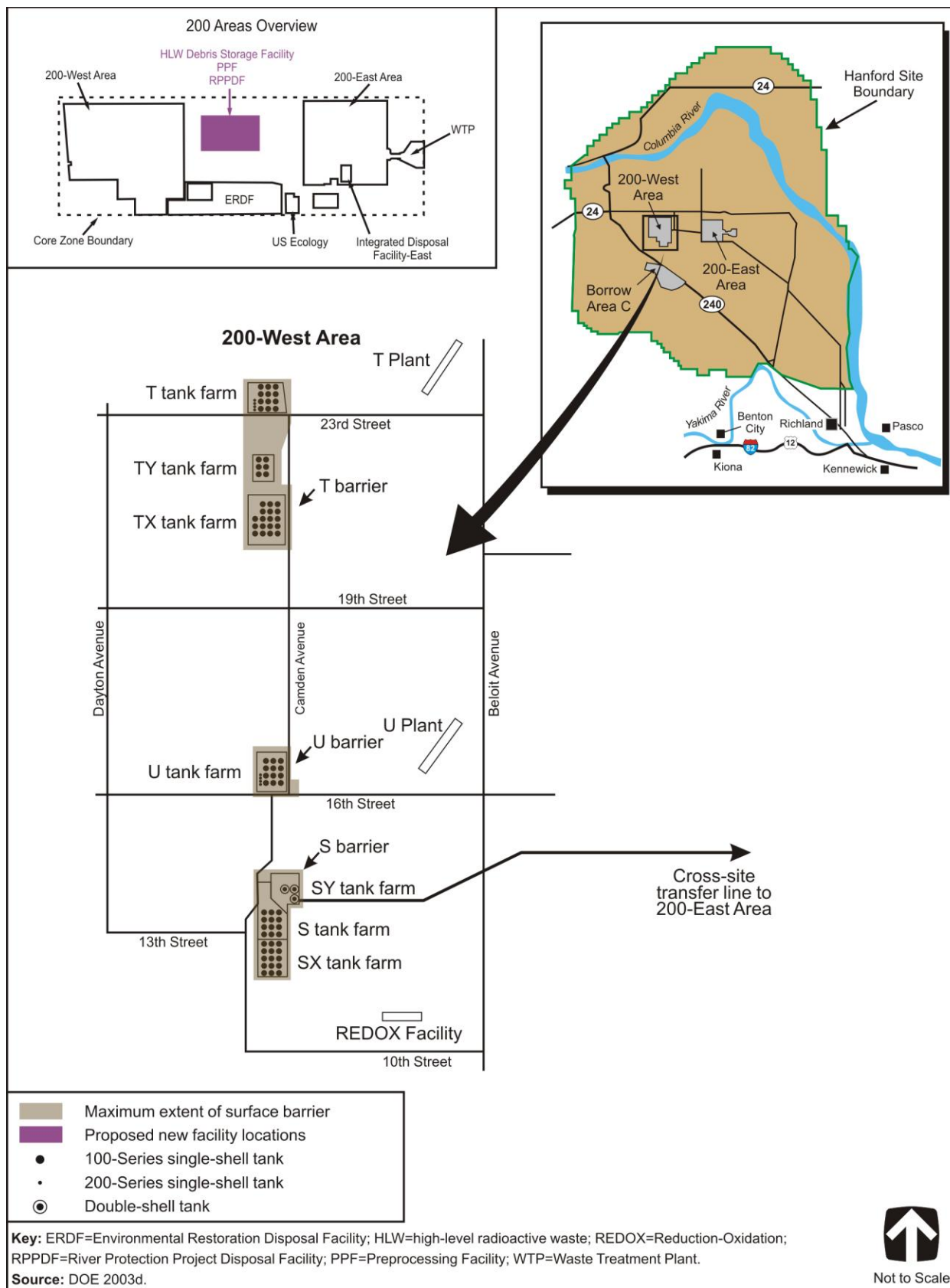


Figure E-30. Conceptualized Closure Surface Barriers in the 200-West Area

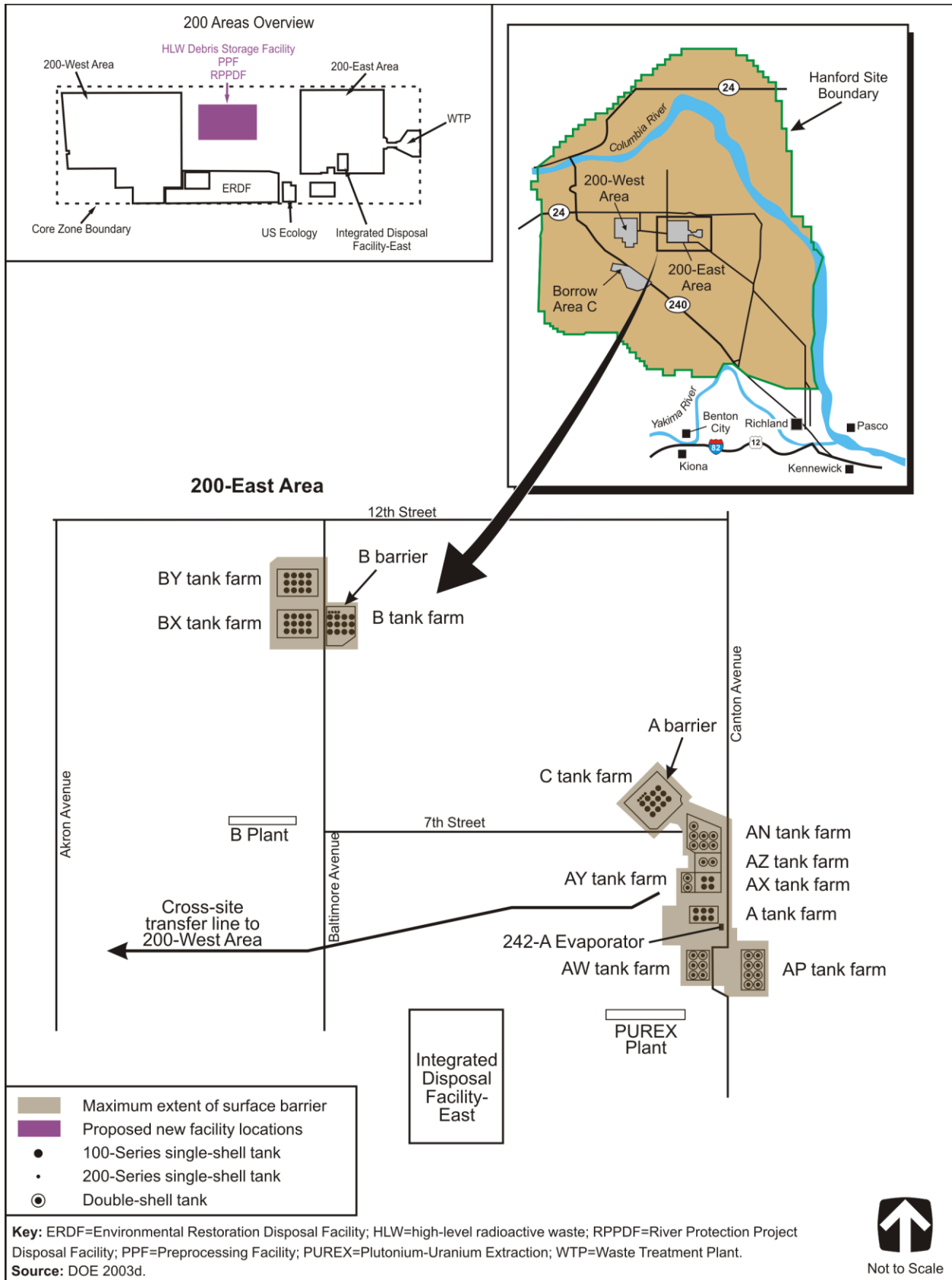


Figure E-31. Conceptualized Closure Surface Barriers in the 200-East Area

Under the Tank Closure alternatives where the tanks are left in place, the tanks would be filled with grout to stabilize the tank domes for subsequent barrier emplacement. Standard earthmoving equipment (e.g., graders, scrapers, bulldozers, compactors) would be used to install the barriers; the barriers themselves involve massive amounts of soil and gravel. Any tanks and equipment left in place would need to be stabilized to prevent collapse during or after barrier emplacement.

Surface barriers would be installed using a three-step process: (1) excavation of material from Borrow Area C, (2) transportation of materials to the tank farms, and (3) emplacement of the barrier over the tank farms. All resource material except the asphalt would be hauled from Borrow Area C, a 930-hectare (2,300-acre) area near the intersection of Beloit Avenue and State Route 240, just south of the 200-West Area. Conventional excavation and transportation equipment would be used to remove and haul the material to the tank farms.

Topography of the tank farms would be considered in the design of the individual barrier lobes. The initial barrier layer would be clean soil fill material of varying thickness to establish grade. This would be followed by two courses of asphalt, which would serve as the primary permeability barrier to prevent water intrusion. Above the asphalt layers would be three layers of sand and gravel to serve as a capillary break and as a drainage layer for water that percolates from the upper soil and vegetation layers. Little if any drainage is expected, as the upper soil and vegetation layers, each a minimum of 0.5 meters (20 inches) thick, are designed to promote evapotranspiration of moisture from precipitation.

The soil and gravel layers would be placed using standard construction equipment such as scrapers, bulldozers, graders, and compactors. Minimal quantities of water may be needed to minimize windblown particulates. Asphalt layers would be constructed using typical road-paving techniques. Vegetation would be planted in the top layer by drilling and crimping or possibly hydroseeding. Typical farm or landscaping equipment would be used. There are several different options for seeding and/or reseeding surface barriers. These options vary based on the type of vegetation planted and the contractor that does the planting. Previous drill planting and crimping using tractor implements have been successful (DOE 2003d).

## **FACILITIES**

Borrow Area C, discussed in Section E.1.2.5.5, provides the facilities and borrow material for closure of solid-waste burial grounds. Surface barriers for the tank farms covering approximately 60 hectares (150 acres) would be similar to those installed over more than 120 hectares (300 acres) of burial grounds. Resource material for burial ground and tank farm surface barriers would come from the same Borrow Area C.

In the process of barrier placement over tank farms and their periphery, several liquid waste disposal sites currently not assigned to the RPP would be covered.

## **ASSUMPTIONS AND UNCERTAINTIES**

The placement of a surface barrier would be rather straightforward. Construction of the Hanford barrier over the 216-B-57 Crib in the 200-East Area successfully demonstrated that surface barriers could be built using existing construction equipment and practices.

For analysis purposes, the surface barriers were assumed to be effective for minimizing the infiltration of precipitation through the tank farms covered by those barriers.

It was further assumed that, because the existing DST farms are adjacent to SST farms, any practical installation of surface barriers over the SST farms would also cover the adjacent DST farms. The new or replacement DSTs required by some alternatives would not be covered by barriers.

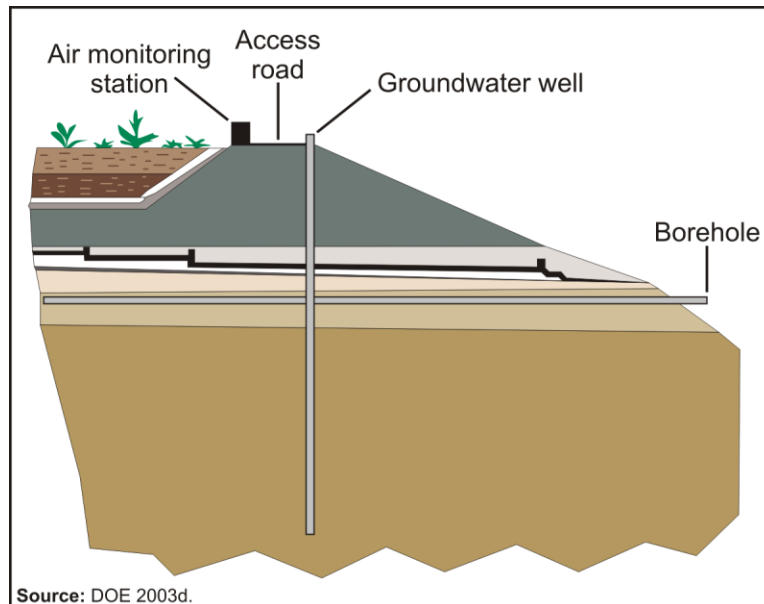


#### **E.1.2.5.4.2 Postclosure Care**

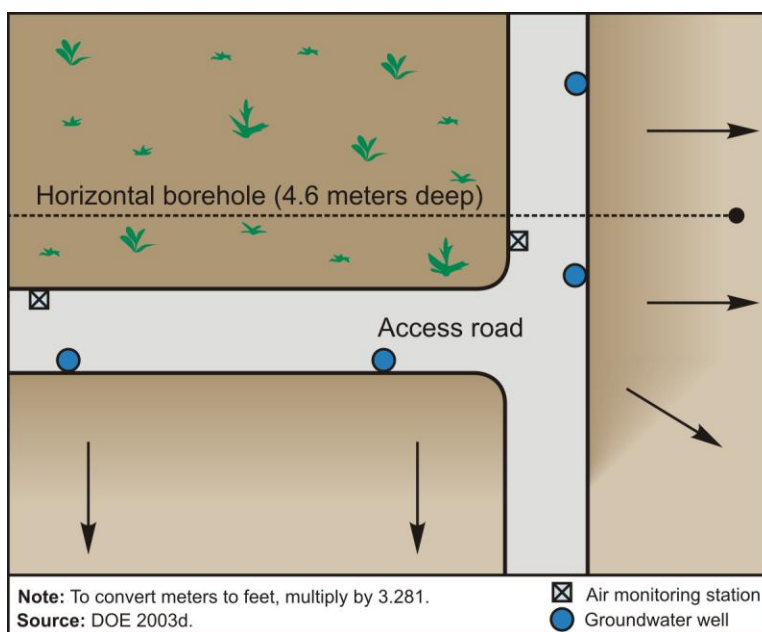
Under alternatives where there would not be total removal of all tanks, ancillary equipment, and contaminants in the vadose zone, postclosure care of residual waste would be required for 100 years after the tank farms are closed and the surface barriers installed. Confirmation that the barriers are performing as designed is especially important (i.e., vegetation planted in the cover is evapotranspiring rainwater and snowmelt such that very little moisture can move through the vadose and mobilize contaminants). Therefore, a major portion of this section addresses vadose zone monitoring. Risk-based clean closure, where contaminants would be removed or cleaned up to a specific risk level, may not require postclosure care.

#### **TECHNOLOGY DESCRIPTION**

Previous documents published by Hanford and supporting organizations have examined tank closure and associated alternatives, but have not included postclosure care in detail. Until now, long-term monitoring plans for the time beyond tank farm closure have been delayed until RCRA closure plans are written. Although these monitoring activities would not be performed for many years, it is important that general information on the various technologies and alternatives for monitoring be identified here. The three primary areas identified for long-term postclosure care would be air, groundwater, and the vadose zone. Figures E-32 and E-33 illustrate both a cross section and a plan view of a barrier with appropriate air, groundwater, and vadose zone monitoring equipment.



**Figure E-32. Cross Section of Proposed Postclosure Monitoring Systems**



**Figure E-33. Plan View of a Barrier Lobe Corner with Monitoring Systems Installed**

**Air Monitoring.** Atmospheric releases of radioactive material from Hanford to the surrounding region are a potential source of human exposure. Radioactive and chemical constituents in air are monitored at a network of air sampling locations on and around Hanford, as well as in nearby and distant communities.

There are currently 45 continuously operating samplers. Of these, 24 samplers are on Hanford, 11 are near the site perimeter, 8 are in nearby communities, and 2 are in distant communities. Air samples collected are analyzed for gross beta and gross alpha radioactivity, specific gamma-emitting radionuclides, strontium-90, plutonium and uranium isotopes, and tritium.

Postclosure care plans would evaluate and/or incorporate the existing air monitoring program. It was assumed that atmospheric radioactive and chemical constituents would decline following barrier placement, which would result in a less vigorous ambient air monitoring program. It is expected that the number of near-field monitoring stations would be reduced almost to zero for long-term monitoring, giving way to at-surface barrier monitoring stations. It was also assumed that technologies would advance considerably in the next few years, and that the site would take advantage of those improvements.

Air monitoring technologies would concentrate on sampling, detecting, and/or analyzing volatiles that may be moving up through the surface barrier. As the asphalt layer prevents upward movement of volatile vapors, this monitoring would primarily address barrier performance. A number of technologies are currently available to fulfill this mission, including high-volume samplers that pass the air stream through activated charcoal and perforated shallow-soil gas probes. When coupled with photo ionization chambers and telemetry, it was anticipated that data could be continuously logged in a remote computer powered by small solar panels (DOE 2003d).

**Groundwater Monitoring.** DOE has monitored groundwater on Hanford since the 1940s to help determine which chemical and radioactive contaminants have made their way into the vadose zone and groundwater. In 1996, DOE established the Groundwater Monitoring Project, under which current groundwater monitoring continues at 24 waste management sites. Each year, monitoring wells are evaluated for performance and new wells are installed as needed. Old wells that are in poor condition or

wells that are not being used for monitoring activities are decommissioned according to WAC 173-160, "Minimum Standards for Construction and Maintenance of Wells."

There are several regulations and statutes that govern groundwater monitoring on Hanford, including the Atomic Energy Act, RCRA, CERCLA, and the *Washington Administrative Code*. The TPA (Ecology, EPA, and DOE 1989) is used to coordinate groundwater protection and remedial action efforts. Groundwater monitoring technologies currently in use include chemical and radiochemical analysis of groundwater samples, piezometers, laminated steel electronic sounding tapes, graduated steel tapes, pressure gauges, data loggers, and pressure transducers.

The sitewide groundwater monitoring strategy is to continue incorporating more cost-efficient methods as they are developed. An evaluation of the regulations governing groundwater monitoring activities would need to be performed in conjunction with development of a groundwater postclosure care plan. This evaluation would need to determine whether all wells would be monitored or whether a restriction to selected wells surrounding barriers would be more appropriate. It is anticipated that, by the time barriers are constructed, any groundwater wells currently in place would need to be replaced or upgraded. It was therefore assumed that all new wells would be needed for 100 years of postclosure care, and that these wells would be drilled at the interface between the slope and the perimeter road built on top of each barrier or just outside the barrier perimeter.

**Surface Barrier Monitoring.** Surface barrier monitoring would include surveillance of the surface barrier for structural integrity, animal burrowing, soil erosion and deposition, and vegetation status. Currently, barrier monitoring can be done in several ways, and the resources needed are difficult to estimate until a specific method of monitoring is selected. Resources would vary from field investigation on foot to aerial photography and computer diagnostics.

**Vadose Zone Monitoring.** Vadose zone monitoring is currently conducted in tank farms using gamma logging and neutron probes in dry wells. Alternative vadose zone monitoring technology demonstrations are also being conducted on a research level.

Monitoring of the vadose zone after tank farm closure is the most important mechanism for ensuring that the closure activities are successful. Particularly important is confirmation that the surface barrier is performing as designed. Under complete clean closure (i.e., tank and deep soil removal), the tank farm areas would be considered clean after closure, and it was assumed that no postclosure care would be required. Monitoring would, however, be needed under any of the landfill alternatives (Tank Closure Alternatives 2B through 5 and 6C), where tanks, residual waste, ancillary equipment, or contaminated soils remain.

Recommendations cannot be made at this time regarding the best technologies for long-term monitoring of the vadose zone and surface barriers because the technologies are evolving at a rapid pace. Several research projects are under way to evaluate new electrical, geophysical, and photogrammetric/remote sensing methods and technologies for monitoring the vadose zone. These new technologies would offer more easily implemented and economical monitoring options than the historical and current field monitoring technologies. Before monitoring technologies are chosen, the results for those currently in R&D would be evaluated against the performance of the currently existing technologies. Regardless of the technology chosen to monitor the vadose zone, boreholes would be needed (DOE 2003d).

## **FACILITIES**

No facilities would be required for postclosure care.

## ASSUMPTIONS AND UNCERTAINTIES

- Technologies are developing at a rapid pace, so it is uncertain what technologies would actually be deployed for monitoring.
- The number and locations of the monitoring wells are uncertain at this time and could be affected by changes in the groundwater level and/or flow direction that could occur by the time postclosure care is initiated.
- All interim monitoring would continue until all tank farms were cleaned and/or areas were ready for barrier placement.
- Evaluation of current air monitoring stations (number and locations) would be conducted, and existing stations would be used whenever possible.
- New vadose zone monitoring boreholes would be needed at the perimeters of the barriers.
- Borehole spacing for this analysis was based on engineering judgment. Actual spacing would be determined during barrier design.
- Horizontal boreholes would be incorporated into the surface barrier design.
- The vegetation cover would be planted with a mixture of native perennial grass species by disking, seeding, and crimping, and would be reseeded when necessary (e.g., after range fires, which were assumed to occur every 10 years) (DOE 2003d).

### E.1.2.5.5 Borrow Area C Operations

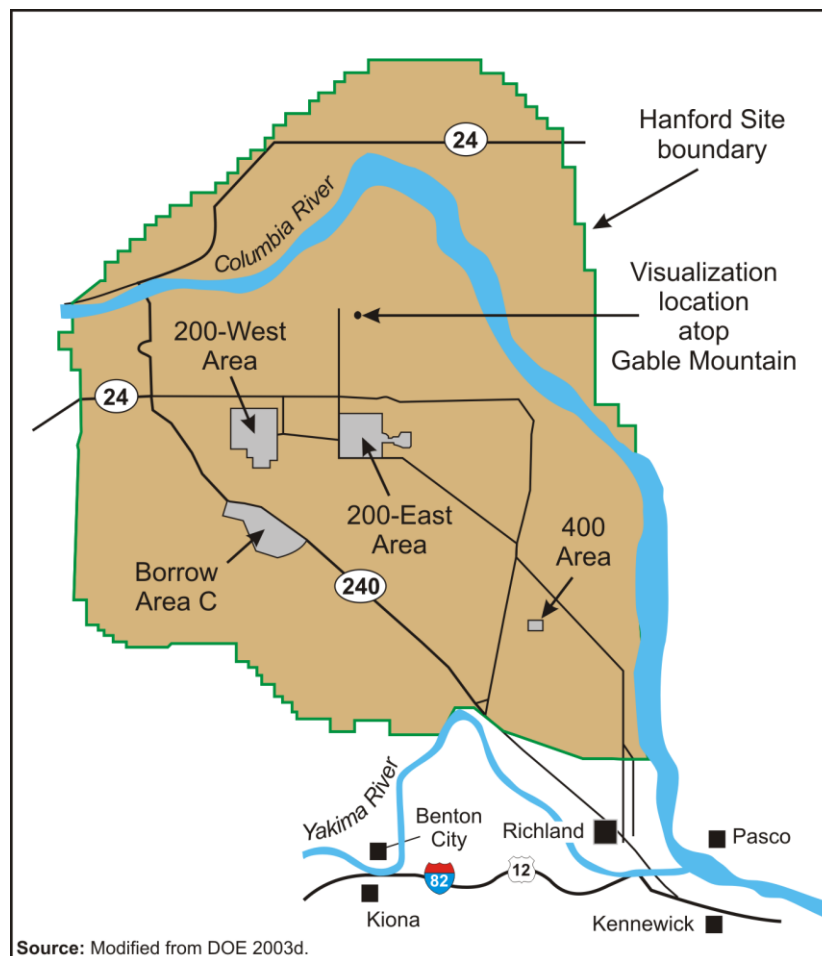
Borrow Area C is an onsite pit to be used to supply sand, soil, and gravel for environmental remediation activities throughout Hanford. Area C has been included in several NEPA reviews, beginning with the *Final Hanford Comprehensive Land-Use Plan Environmental Impact Statement (Hanford Comprehensive Land-Use Plan EIS)* (DOE 1999). The *Hanford Comprehensive Land-Use Plan EIS* set aside a portion of the Fitzner-Eberhardt Arid Lands Ecology Reserve (including Area C) as a quarry site instead of the McGee Ranch. The latter location was originally included as part of DOE's *Hanford Comprehensive Land-Use Plan EIS* Preferred Alternative due to the occurrence there of extensive basalt rock and silty soil materials, which would be needed for Hanford remediation activities. However, due to the concerns of the public and cooperating agencies, such as the U.S. Fish and Wildlife Service (USFWS) and the Washington Department of Fish and Wildlife, about the importance of a wildlife corridor and shrub-steppe habitat throughout the McGee Ranch/Umtanum Ridge area, DOE modified its Preferred Alternative. In the final Preferred Alternative, the McGee Ranch land use designation was revised to Preservation and the ranch was included within a USFWS-managed wildlife refuge. In exchange, and to support the need for appropriate borrow materials, a portion of the Fitzner-Eberhardt Arid Lands Ecology Reserve was set aside and designated as Conservation (Mining). This tradeoff was subsequently acknowledged by USFWS in its *Hanford Reach National Monument Final Comprehensive Conservation Plan and Environmental Impact Statement* (USFWS 2008). Note that DOE intends to reanalyze sources of borrow material at a later date, but, for analysis purposes, this EIS analyzed Borrow Area C as described in this section.

Specific alternatives discussed in this *TC & WM EIS* require the utilization of borrow materials from Borrow Area C. Resource material from Borrow Area C would primarily be used for construction of new facilities, backfilling and regrading where facilities and/or contaminated soils were removed from the ground, and creation of modified RCRA Subtitle C or Hanford barriers.

Conventional excavation, loading, and transportation equipment would be used at Borrow Area C. Conveyor systems may be employed to move excavated material to stockpile areas or load trucks. Conveying systems may be outfitted with crushing, sorting, and screening systems to segregate rock and fines according to Hanford's needs. Basalt, when encountered, would be blasted with controlled, subsurface detonations.

#### E.1.2.5.5.1 Description

Borrow Area C is near the intersection of Beloit Avenue and State Route 240. The site is a large polygon of approximately 930 hectares (2,300 acres) bounded by State Route 240 and the Fitzner-Eberhardt Arid Lands Ecology Reserve. Borrow Area C is not part of the Hanford Reach National Monument. It was selected for use as a borrow area because of its proximity to most of the environmental activities that would require borrow materials (e.g., onsite disposal facilities, tank farms, other 200 Area treatment facilities). Figure E-34 illustrates the location of Borrow Area C relative to the 200 Area and tank farms. Borrow material needed to support *TC & WM EIS* alternatives was assumed to come from Borrow Area C because it can provide the variety of gravel, sand, and soil types necessary to support environmental activities.



**Figure E-34. Borrow Area C Pit Location and Access Routes**

Assuming the entire Borrow Area C footprint of approximately 930 hectares (2,300 acres) is excavated to a depth of 4.6 meters (15 feet), Borrow Area C would yield up to 42.6 million cubic meters (55.7 million cubic yards) of resource material to support Hanford activities, including those activities analyzed in this EIS.

For analysis purposes, the baseline operational data for Borrow Area C were based on an assumed excavation rate of 150 cubic meters (200 cubic yards) per hour. Available hours of operation would be a standard 2,080 hours per year. This assumed rate and hours of operation would be capable of producing approximately 318,000 cubic meters (416,000 cubic yards) per year of borrow material for use at Hanford.

A hydraulic excavator with a 1.5-cubic-meter (2-cubic-yard) shovel excavates sand and gravel at a rate of 122 cubic meters (160 cubic yards) per hour. Therefore, conservatively, two hydraulic excavators would be needed to achieve the operational rate of 150 cubic meters (200 cubic yards) per hour. A loader with a 3-cubic-meter (4-cubic-yard) bucket can load the sand and gravel into dump trucks at a rate of 245 cubic meters (320 cubic yards) per hour. Based on a 7.5-cubic-meter (10-cubic-yard) capacity dump truck making a single round trip in 1 hour, 20 dump trucks would be required to sustain a production rate of 150 cubic meters (200 cubic yards) per hour.

Two water trucks would be utilized to maintain dust control. Borrow Area C water usage for dust control would be approximately 190,000 liters (50,000 gallons) per day (SAIC 2010a).

#### **E.1.2.5.5.2 Resources**

The total borrow material demand from all tank closure, Fast Flux Test Facility (FFTF) decommissioning, and waste management activities analyzed under the *TC & WM EIS* alternatives is provided in Table E-13. The estimated volumes in this table are for analysis purposes only and assume that Borrow Area C would be operated at full production rates every day for the duration of the alternative, regardless of the actual need for borrow material. This table lists all the Borrow Area C demand activities and volumes that are expected to be implemented and provides four combinations of Tank Closure, FFTF Decommissioning, and Waste Management alternatives:

1. Alternative Combination 1: Tank Closure Alternative 1, FFTF Decommissioning Alternative 1, and Waste Management Alternative 1 (No Action Alternatives).

This combination estimates the demand for Borrow Area C resource material when all the No Action Alternatives are analyzed together.

2. Alternative Combination 2: Tank Closure Alternative 2B, FFTF Decommissioning Alternative 2 with the Hanford Option for disposition of remote-handled special components (RH-SCs), and Waste Management Alternative 2 in combination with Disposal Group 1.

This combination estimates the demand for Borrow Area C resource material resulting from a combination of the alternatives and represents DOE's Preferred Alternative.

3. Alternative Combination 3: Tank Closure Alternative 6B, Base Case; FFTF Decommissioning Alternative 3 with the Hanford Option for disposition of RH-SCs; and Waste Management Alternative 2 in combination with Disposal Group 2.

This combination estimates the demand for Borrow Area C resource material due to a combination of the alternatives and represents an upper limit for analyzing the demands on Borrow Area C resource materials.

4. Alternative Combination 4: Tank Closure Alternative 6A, Base Case; FFTF Decommissioning Alternative 3 with the Hanford Option for disposition of RH-SCs; and Waste Management Alternative 2 in combination with Disposal Group 3.

This combination estimates the highest demands for Borrow Area C borrow materials.

Table E-13. Demand on Borrow Area C Resources from Alternative Activities

Alternative	Total (yd <sup>3</sup> )	Total (m <sup>3</sup> )	Alternative Combinations							
			Combination 1		Combination 2		Combination 3		Combination 4	
			TC Alt. 1, FFTF Dec. Alt. 1, WM Alt. 1 (yd <sup>3</sup> )	TC Alt. 1, FFTF Dec. Alt. 1, WM Alt. 1 (m <sup>3</sup> )	TC Alt. 2B; FFTF Dec. Alt. 2 (Hanford); WM Alt. 2, DG 1 (yd <sup>3</sup> )	TC Alt. 2B; FFTF Dec. Alt. 2 (Hanford); WM Alt. 2, DG 1 (m <sup>3</sup> )	TC Alt. 6B, Base; FFTF Dec. Alt. 3 (Hanford); WM Alt. 2, DG 2 (yd <sup>3</sup> )	TC Alt. 6B, Base; FFTF Dec. Alt. 3 (Hanford); WM Alt. 2, DG 2 (m <sup>3</sup> )	TC Alt. 6A, Base; FFTF Dec. Alt. 3 (Hanford); WM Alt. 2, DG 3 (yd <sup>3</sup> )	TC Alt. 6A, Base; FFTF Dec. Alt. 3 (Hanford); WM Alt. 2, DG 3 (m <sup>3</sup> )
1	1.21×10 <sup>5</sup>	9.28×10 <sup>4</sup>	1.21×10 <sup>5</sup>	9.28×10 <sup>4</sup>	N/A	N/A	N/A	N/A	N/A	N/A
2A	1.64×10 <sup>6</sup>	1.25×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
2B	5.66×10 <sup>6</sup>	4.33×10 <sup>6</sup>	N/A	N/A	5.66×10 <sup>6</sup>	4.33×10 <sup>6</sup>	N/A	N/A	N/A	N/A
3A	6.03×10 <sup>6</sup>	4.61×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
3B	5.38×10 <sup>6</sup>	4.11×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
3C	5.61×10 <sup>6</sup>	4.29×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
4	6.09×10 <sup>6</sup>	4.66×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
5	7.03×10 <sup>6</sup>	5.38×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6A, Base Case	2.93×10 <sup>7</sup>	2.24×10 <sup>7</sup>	N/A	N/A	N/A	N/A	N/A	N/A	2.93×10 <sup>7</sup>	2.24×10 <sup>7</sup>
6A, Option Case	2.90×10 <sup>7</sup>	2.21×10 <sup>7</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6B, Base Case	1.42×10 <sup>7</sup>	1.08×10 <sup>7</sup>	N/A	N/A	N/A	N/A	1.42×10 <sup>7</sup>	1.08×10 <sup>7</sup>	N/A	N/A
6B, Option Case	1.39×10 <sup>7</sup>	1.06×10 <sup>7</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
6C	6.21×10 <sup>6</sup>	4.75×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Tank Closure

Table E-13. Demand on Borrow Area C Resources from Alternative Activities (continued)

Alternative	Total (yd <sup>3</sup> )	Total (m <sup>3</sup> )	Alternative Combinations							
			Combination 1		Combination 2		Combination 3		Combination 4	
			TC Alt. 1, FFTF Dec. Alt. 1, WM Alt. 1 (yd <sup>3</sup> )	TC Alt. 1, FFTF Dec. Alt. 1, WM Alt. 1 (m <sup>3</sup> )	TC Alt. 2B; FFTF Dec. Alt. 2 (Hanford); WM Alt. 2, DG 1 (yd <sup>3</sup> )	TC Alt. 2B; FFTF Dec. Alt. 2 (Hanford); WM Alt. 2, DG 1 (m <sup>3</sup> )	TC Alt. 6B, Base; FFTF Dec. Alt. 3 (Hanford); WM Alt. 2, DG 2 (yd <sup>3</sup> )	TC Alt. 6B, Base; FFTF Dec. Alt. 3 (Hanford); WM Alt. 2, DG 2 (m <sup>3</sup> )	TC Alt. 6A, Base; FFTF Dec. Alt. 3 (Hanford); WM Alt. 2, DG 3 (yd <sup>3</sup> )	TC Alt. 6A, Base; FFTF Dec. Alt. 3 (Hanford); WM Alt. 2, DG 3 (m <sup>3</sup> )
1	0	0	0	0	N/A	N/A	N/A	N/A	N/A	N/A
2	1.59×10 <sup>5</sup>	1.21×10 <sup>5</sup>	N/A	N/A	1.59×10 <sup>5</sup>	1.21×10 <sup>5</sup>	N/A	N/A	N/A	N/A
2, Hanford options (SRF and RTP)	2.14×10 <sup>3</sup>	1.63×10 <sup>3</sup>	N/A	N/A	2.14×10 <sup>3</sup>	1.63×10 <sup>3</sup>	N/A	N/A	N/A	N/A
3	1.88×10 <sup>5</sup>	1.43×10 <sup>5</sup>	N/A	N/A	N/A	N/A	1.88×10 <sup>5</sup>	1.43×10 <sup>5</sup>	1.88×10 <sup>5</sup>	1.43×10 <sup>5</sup>
3, Hanford options (SRF and RTP)	2.14×10 <sup>3</sup>	1.63×10 <sup>3</sup>	N/A	N/A	N/A	N/A	2.14×10 <sup>3</sup>	1.63×10 <sup>3</sup>	2.14×10 <sup>3</sup>	1.63×10 <sup>3</sup>
2, T&S	1.39×10 <sup>4</sup>	1.06×10 <sup>4</sup>	N/A	N/A	1.39×10 <sup>4</sup>	1.06×10 <sup>4</sup>	1.39×10 <sup>4</sup>	1.06×10 <sup>4</sup>	1.39×10 <sup>4</sup>	1.06×10 <sup>4</sup>
2, DG 1	2.46×10 <sup>6</sup>	1.88×10 <sup>6</sup>	N/A	N/A	2.46×10 <sup>6</sup>	1.88×10 <sup>6</sup>	N/A	N/A	N/A	N/A
2, DG 2	9.46×10 <sup>6</sup>	7.23×10 <sup>6</sup>	N/A	N/A	N/A	N/A	9.46×10 <sup>6</sup>	7.23×10 <sup>6</sup>	N/A	N/A
2, DG 3	9.46×10 <sup>6</sup>	7.23×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	9.46×10 <sup>6</sup>	7.23×10 <sup>6</sup>
3, T&S	1.39×10 <sup>4</sup>	1.06×10 <sup>4</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
3, DG 1	2.17×10 <sup>6</sup>	1.66×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
3, DG 2	9.37×10 <sup>6</sup>	7.16×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
3, DG 3	9.37×10 <sup>6</sup>	7.16×10 <sup>6</sup>	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
<b>Total</b>			<b>1.30×10<sup>5</sup></b>	<b>9.90×10<sup>4</sup></b>	<b>8.30×10<sup>6</sup></b>	<b>6.34×10<sup>6</sup></b>	<b>2.38×10<sup>7</sup></b>	<b>1.82×10<sup>7</sup></b>	<b>3.90×10<sup>7</sup></b>	<b>2.98×10<sup>7</sup></b>

**Note:**

–The Idaho options under FFTF Decommissioning Alternatives 2 and 3 do not require Borrow Area C materials.

–All borrow material resources are assumed to be supplied by Borrow Area C.

–Available Borrow Area C resources are estimated to be 4.26×10<sup>7</sup> cubic meters (5.57×10<sup>7</sup> cubic yards). This estimate assumes approximately 930 hectares (2,300 acres) are available and the average excavated depth is 4.6 meters (15 feet).

**Key:** Alt.=Alternative; Base=Base Case; Dec.=Decommissioning; DG=Disposal Group; FFTF=Fast Flux Test Facility; Hanford=Hanford Site; m<sup>3</sup>=cubic meters; N/A=not applicable; RTP=Remote Treatment Project; SRF=Sodium Reaction Facility; T&S=treatment and storage; TC=Tank Closure; WM=Waste Management; yd<sup>3</sup>=cubic yards.

**Source:** SAIC 2010a.



As shown in Table E-13, the demands for all the alternative combinations above do not exceed the 42.6 million cubic meters (55.7 million cubic yards) of available borrow material volume. Alternative Combination 4 requires the highest volume of borrow material: approximately 70 percent of the available borrow material.

Alternatively, based on an estimated total available volume of borrow material of 42.6 million cubic meters (55.7 million cubic yards), Borrow Area C would be able to support Hanford operations for up to 134 years at the maximum production rate of 318,000 cubic meters (416,000 cubic yards) per year. In practice, the resource demands for each *TC & WM EIS* alternative on Borrow Area C would be intermittent, nonexistent during lengthy intervals, and significantly less than the volumes estimated in Table E-13. However, for the purpose of analyzing air emissions generated from operations of heavy equipment and wind erosion (e.g., dust), each *TC & WM EIS* alternative conservatively assumes that Borrow Area C would be operated at full production rates every day for the duration of the alternative, regardless of the actual needs for borrow material.

#### **E.1.2.5.5.3 Assumptions and Uncertainties**

- The analysis of available resources (e.g., borrow material) and the consumption of these resources in support of the alternatives analyzed in this *TC & WM EIS* are based on several assumptions related to the operations of the Borrow Area C pits. Operation of a Borrow Area C pit is purely an earthmoving exercise utilizing traditional mechanical means and methods. The production rate of borrow material is based on a modest amount of equipment (e.g., 1.5- to 3.1-cubic-meter [2- to 4-cubic-yard] excavators, 0.7- to 3.1-cubic-meter [1- to 4-cubic-yard] loaders, 7.6- to 15.3-cubic-meter [10- to 20-cubic-yard] dump trucks) and a standard 2,080-hour work year. If necessary to meet peak resource demands to support environmental activities at Hanford, the large size of Borrow Area C (approximately 930 hectares [2,300 acres]) could easily accommodate an increase in heavy equipment or daily work shifts to temporarily increase production rates.
- Borrow Area C operations would generate no LLW, MLLW, TRU waste, or hazardous waste.
- Borrow Area C infrastructure upgrades prior to operation and restoration activities after operations are not included in this analysis.
- Borrow Area C would produce equal amounts of sand and gravel.
- Radiological exposure to workers (radiation workers) would be zero.
- The average Borrow Area C excavation depth is 4.6 meters (15 feet).
- Controlled blasting may be required for excavation of basalt to meet gravel/rock requirements (SAIC 2010a).

#### **E.1.2.5.5.4 Viewshed at Borrow Area C**

Impacts on the Borrow Area C viewshed of excavations for Alternative Combination 2 and the most conservative alternative combination were evaluated through a comparison with current conditions. A description of Alternative Combination 2 is provided in Section E.1.2.5.5.2. The most conservative alternative combination, as described in Chapter 4, Section 4.4.4, includes Tank Closure Alternative 6A, Option Case; FFTF Decommissioning Alternative 3 (with all facilities to be built at Hanford); and Waste Management Alternative 2 (with waste Disposal Group 3). Resources at Borrow Area C are assumed to extend to approximately 930 hectares (2,300 acres) and to an average excavation depth of 4.6 meters (15 feet). The area of disturbance for each combination was calculated by dividing the volume of the

combined geologic and soil resource requirements by the assumed excavation depth. Table E-14 provides the Borrow Area C land area and volume demands for Alternative Combination 2 and the most conservative alternative combination.

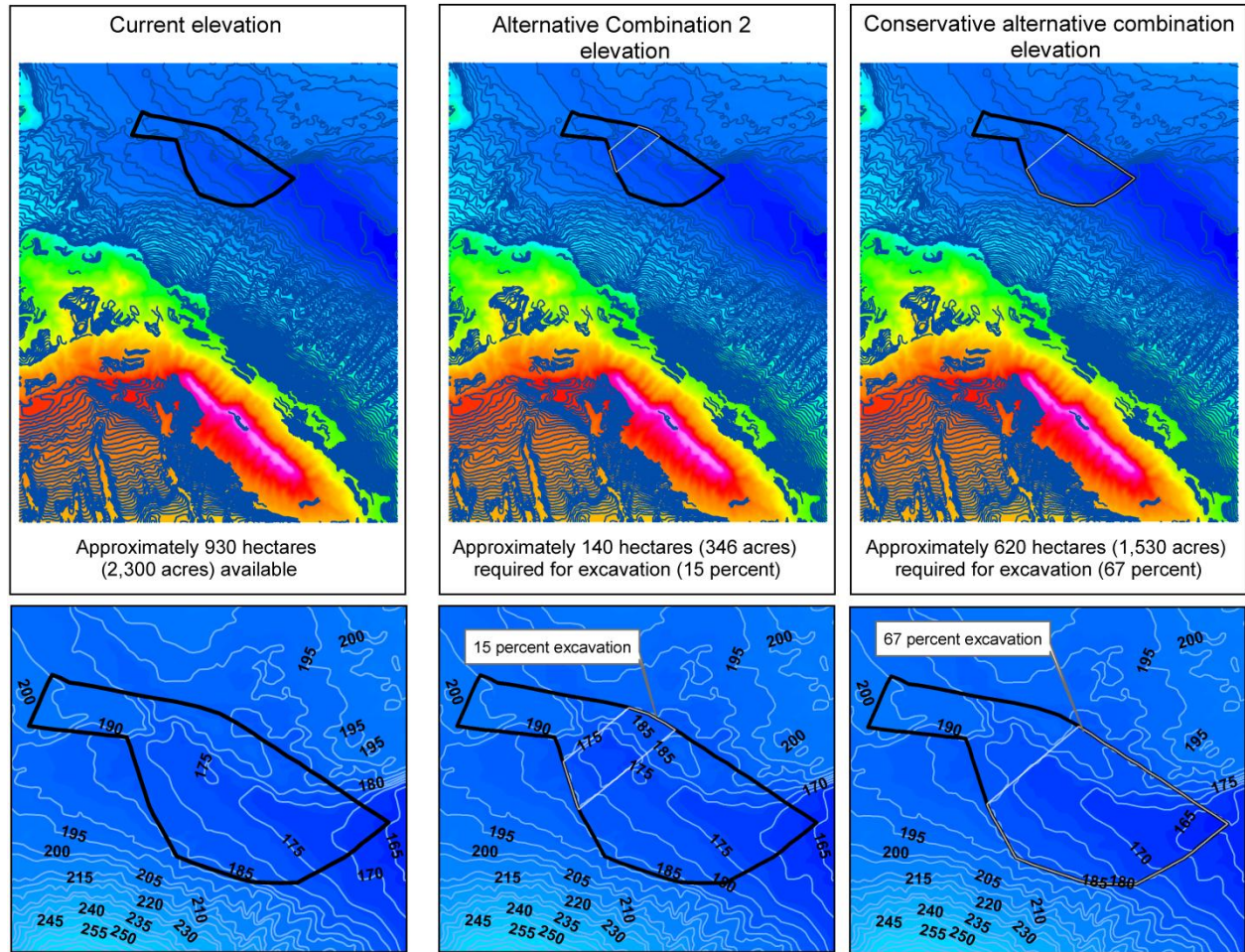
**Table E-14. Borrow Area C Land and Resource Demands for Alternative Combination 2 and Most Conservative Alternative Combination**

Combination Component	Area (hectares)	Volume (cubic meters)
<b>Alternative Combination 2</b>		
Tank Closure Alternative 2B	9.41×10 <sup>1</sup>	4.33×10 <sup>6</sup>
FFTF Decommissioning Alternative 2, Hanford Reuse Option	2.76	1.27×10 <sup>5</sup>
Waste Management Alternative 2, Disposal Group 1	4.33×10 <sup>1</sup>	1.99×10 <sup>6</sup>
<b>Total Demand</b>	<b>1.40×10<sup>2</sup></b>	<b>6.45×10<sup>6</sup></b>
<b>Percentage of Available Resources</b>	<b>15</b>	
<b>Most Conservative Alternative Combination</b>		
Tank Closure Alternative 6A, Option Case	4.58×10 <sup>2</sup>	2.11×10 <sup>7</sup>
FFTF Decommissioning Alternative 3, Hanford Reuse Option	3.2	1.47×10 <sup>5</sup>
Waste Management Alternative 2, Disposal Group 3	1.59×10 <sup>2</sup>	7.31×10 <sup>6</sup>
<b>Total Demand</b>	<b>6.20×10<sup>2</sup></b>	<b>2.86×10<sup>7</sup></b>
<b>Percentage of Available Resources</b>	<b>67</b>	

**Note:** To convert cubic meters to cubic feet, multiply by 35.315; hectares to acres, by 2.471.

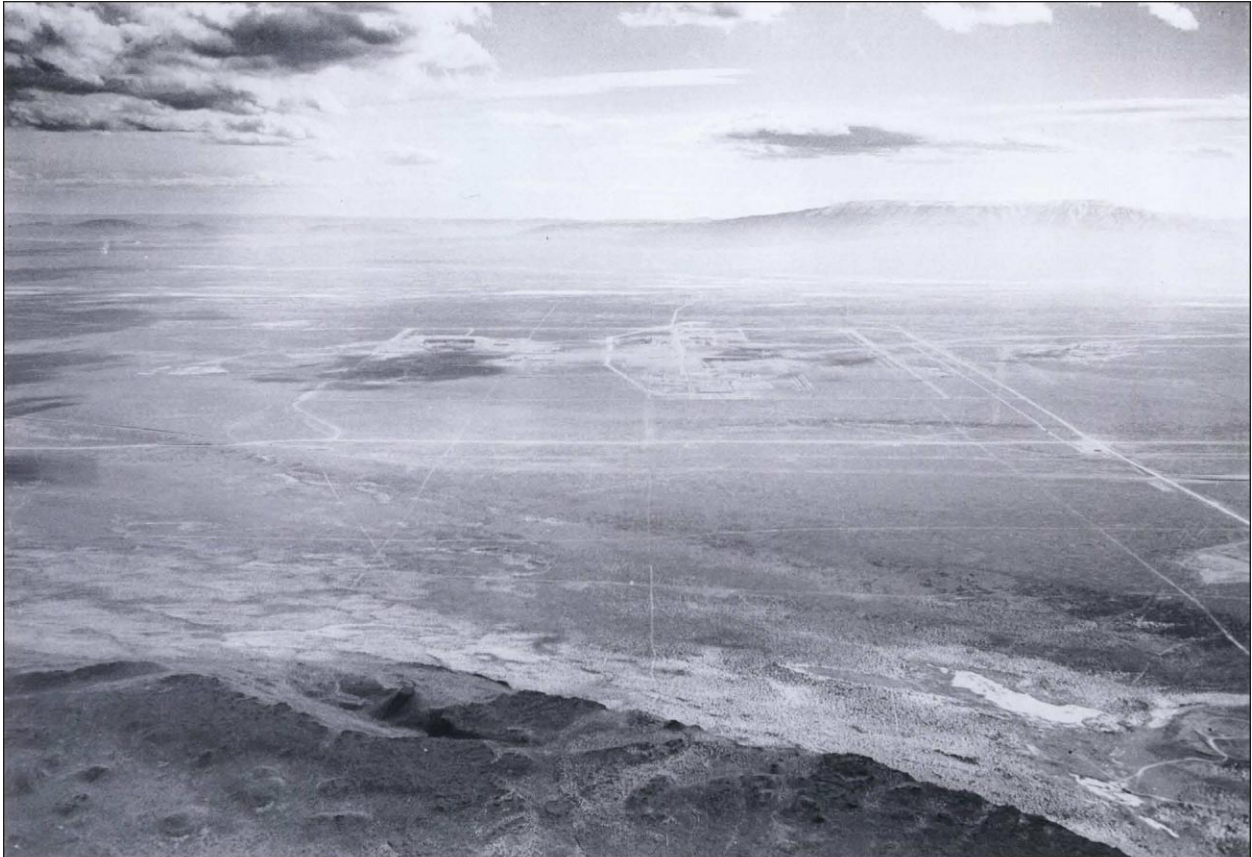
**Key:** FFTF=Fast Flux Test Facility.

Figure E-35 presents aerial views of Borrow Area C and the surrounding area for the three sets of conditions that were compared: (1) current conditions, (2) conditions representative of Alternative Combination 2, and (3) conditions representative of the most conservative alternative combination. The figure's top row depicts footprints that represent the total land areas within Borrow Area C required for excavation under each condition. The figure's bottom row depicts the corresponding changes in elevation that would result from the removal of the total volumes of material shown in Table E-14.



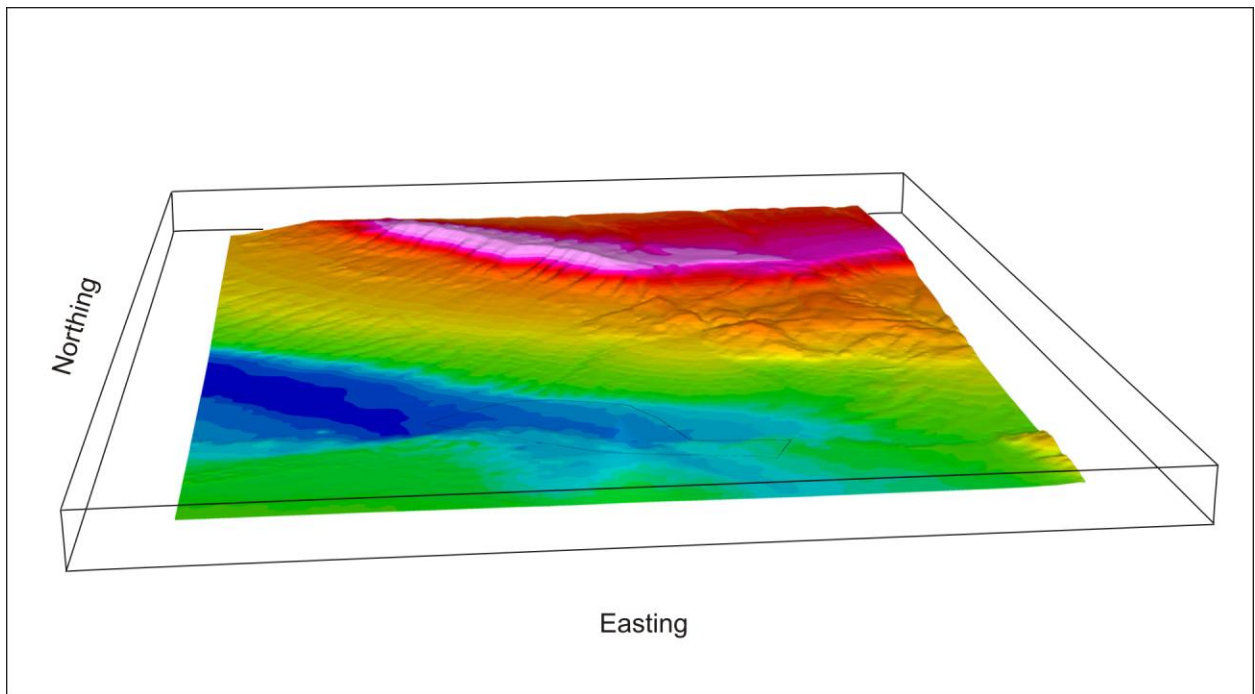
**Figure E-35. Comparison of Excavation Areas at Borrow Area C Under Alternative Combination 2 and the Most Conservative Alternative Combination**

The preceding analysis was used to develop and compare changes in topography within Borrow Area C. To supplement the evaluation of topography, a comparison based on visualization of the potential landform changes for the viewshed from a visualization location atop Gable Mountain, looking south, was developed using geographic information systems (GISs). Figure E-34 shows Borrow Area C, south of the Core Zone, as well as the location of Gable Mountain, just north of the Core Zone, and the visualization location that was used for this analysis. Figure E-36 is a ground-based photograph of the Borrow Area C vicinity taken from the visualization location that illustrates the current viewshed. Figure E-37 is the corresponding GIS depiction of the current topography (note that the color scale indicates elevation). Figures E-38 and E-39 are GIS depictions of the potential topography under Alternative Combination 2 and the most conservative alternative combination, respectively. These GIS depictions use the Washington State Plane Coordinate System of 1983, south zone.



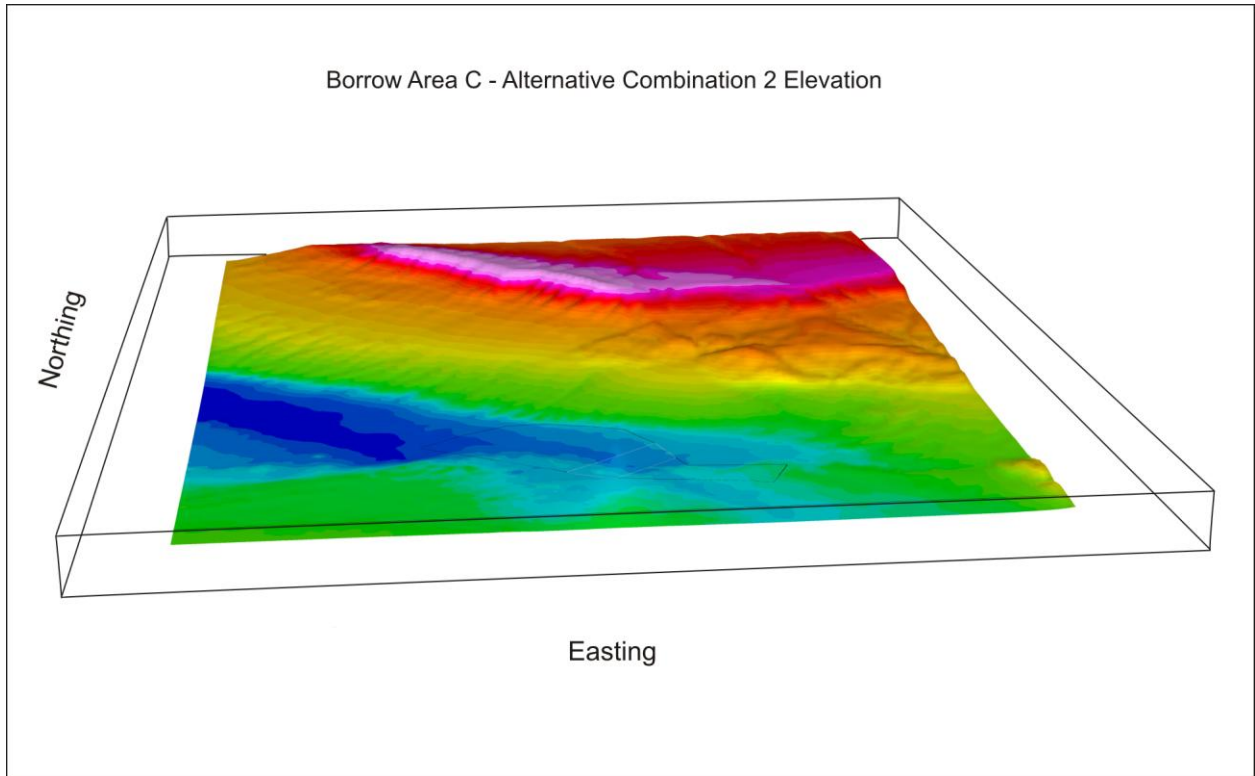
Source: Shoop 2007.

**Figure E-36. Ground-Based Photograph of Borrow Area C, from Gable Mountain Looking South**

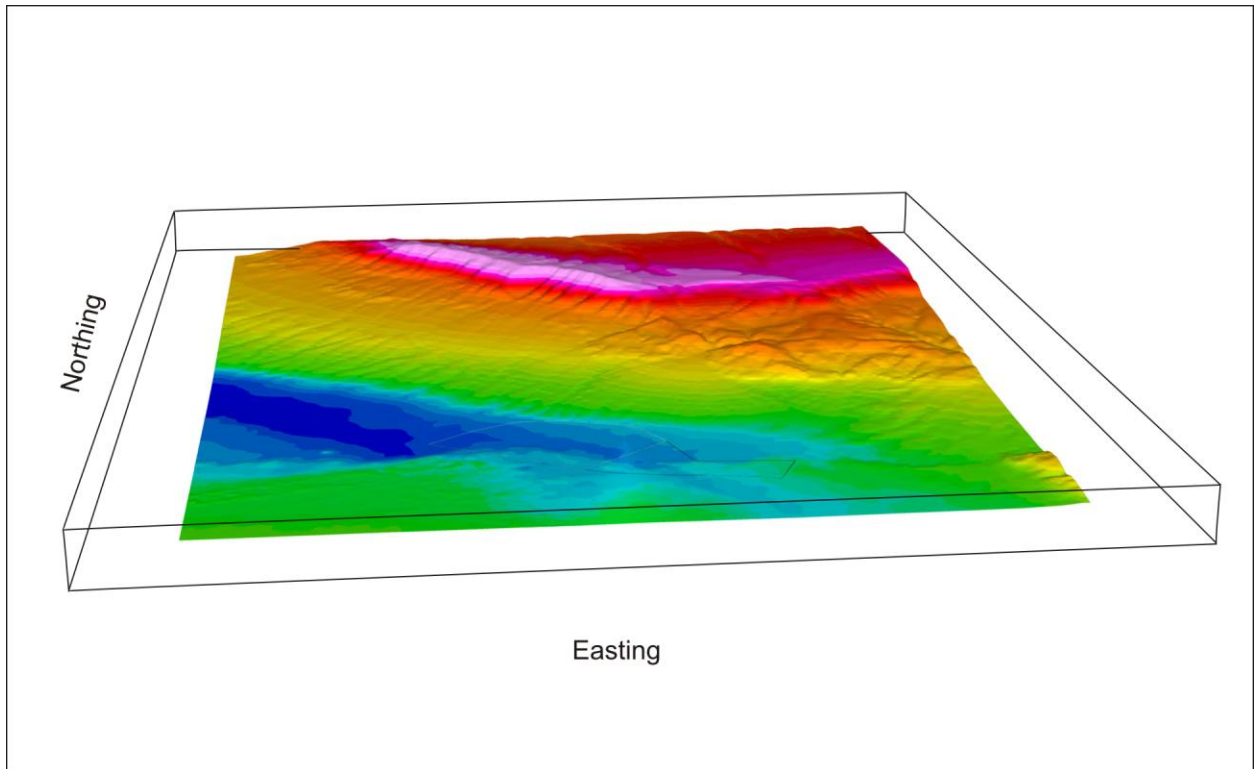


**Figure E-37. Geographic Information System Depiction of Current Topography in the Vicinity of Borrow Area C**





**Figure E-38. Geographic Information System Depiction of Potential Topography in the Vicinity of Borrow Area C Under Alternative Combination 2**



**Figure E-39. Geographic Information System Depiction of Potential Topography in the Vicinity of Borrow Area C Under the Most Conservative Alternative Combination**

### E.1.2.6 Facility Decontamination and Decommissioning

This *TC & WM EIS* specifically evaluates the D&D of only the following 10 existing Hanford facilities that would be required prior to final closure of the SST system:

- **242-S Evaporator.** The 242-S Evaporator, north of the S tank farm, was used to concentrate tank waste. Operation of the evaporator began in 1973 and continued until 1980. The facility was shut down in 1980 and placed in a standby mode in 1981.
- **242-T Evaporator.** The 242-T Evaporator is adjacent to the TX tank farm. Operation of the evaporator began in 1952 and continued intermittently until 1980. In April 1981, a shutdown/standby plan was written, and a final waste transfer out of the facility was made in 1982.
- **204-AR Receiver Station.** The 204-AR Receiver Station is west of the AX tank farm. The facility was designed to receive liquid waste from rail tank cars or tank trailers and to pump this waste to a designated 200-East Area tank farm. The facility was constructed in 1981 and is still operational.
- **241-A-431 Vent Building.** The 241-A-431 Vent Building was constructed in 1953 to provide offgas de-entrainment for the six tanks in the A tank farm and to receive drainage from the 296-A-11 stack. It began operation in 1955 and was shut down in 1969.
- **241-AX-IX Ion Exchange.** The 241-AX-IX Ion Exchange is east of the A tank farm and was used to treat condensate from the waste facility exhausters between the A and AX tank farms. The column was designed and built in the late 1960s and was operated routinely from 1973 to 1976.
- **241-BY-ITS1 In-Tank Solidification System.** The 241-BY-ITS1 In-Tank Solidification System is located in the BY tank farm. The system was constructed in the late 1950s and operated until the mid-1970s to concentrate waste in the BY tanks.
- **241-C-801 Cesium Loadout Facility.** The 241-C-801 Cesium Loadout Facility is located in the C tank farm. It commenced operation in 1962 and served as a cesium processing transfer facility until 1976.
- **241-SX-401 and 241-SX-402 Condenser Shielding Buildings.** These condenser shielding buildings were built in 1954 and are located within the SX tank farm. Building 241-SX-401 was used as designed to cool some of the tanks in the SX tank farm. This continued until 1975, when the facility's use was ended.
- **241-AX-WT-SP-137 Seal Pot.** The 241-AX-WT-SP-137 Seal Pot is an underground structure located in the AX tank farm. The seal pot would be grouted and abandoned in place.

D&D of these 10 facilities would include the following operations: decontamination of building surfaces and equipment; removal of major vessels from inside each facility; demolition of each facility to ground level; and transfer of waste, rubble, and debris into containers or shielded burial boxes for shipment to appropriate disposal locations (DOE 2003d).

This *TC & WM EIS* evaluates the deactivation of the WTP and other proposed waste treatment and interim storage facilities (i.e., the CSB, IHLW Shipping/Transfer Facility, IHLW Interim Storage Modules, and ILAW Interim Storage Facilities) at the end of their operational lives. However, closure and D&D of these facilities are not within the scope of this *TC & WM EIS*.

### **E.1.3 Tank Closure Technologies Considered, but Not Analyzed in Detail**

This section describes the technologies that were initially considered, but were not evaluated, in this *TC & WM EIS*.

#### **E.1.3.1 Waste Storage**

Some of the alternatives may require additional storage capacity beyond the current DST capacity. The proposed storage arrangement is the construction of new below-grade DSTs. The following storage options were considered but not evaluated:

**Modification of Existing Canyon Facilities.** This concept involves the modification of existing canyon facilities (e.g., the PUREX [Plutonium-Uranium Extraction] Plant, B Plant) to store HLW. Existing equipment would be removed and new tank capacity would be installed. This option was not evaluated because (1) existing canyon facilities are not designed for storage of large volumes of liquid waste, (2) the existing radiation and contamination levels would result in elevated personnel exposure, (3) the low volume of storage space would not be cost-effective, and (4) environmental permitting is highly uncertain (DOE 2003a).

**New Above-Grade DSTs.** This concept involves construction of new above-grade DSTs similar to the WTP waste receipt tanks. This option would also require construction of a facility surrounding the tanks to provide shielding. This option was not evaluated because (1) there are technical disadvantages associated with shielding large (3.8-million-liter [1-million-gallon]) aboveground tanks, and (2) resources associated with new aboveground tanks would be similar to those associated with the below-grade tanks (DOE 2003a).

**Staging of Retrieved Waste in SSTs.** This option was not evaluated in detail in this *TC & WM EIS* due to several factors, one being that the SSTs have been declared unfit for use and cannot readily be made compliant with current regulations. However, DOE is considering staging waste in SSTs as an option to building additional DSTs. Ecology has identified a number of factors that would influence its potential acceptance of this approach, including: (1) upgrades of systems with additional leak detection, monitoring, and mitigation capabilities; (2) replacement of waste transfer pumps, transfer lines, and ventilation systems; (3) maintenance of the interim stabilization criteria after the waste is staged; (4) development of a liquid-waste management plan; and (5) agreement on selection criteria for the tanks to be used. At present, criteria for determining which tanks are suitable for staging have not been identified. Infrastructure needs have been identified at a system level, but specific design information related to a particular tank or tank farm has not been identified. In addition, liquid-waste management issues associated with meeting the interim stabilization criteria would need to be addressed. If these issues were addressed, SST staging would be similar to the proposed waste transfers and waste storage activities for WRFs and/or DSTs. Near-term actions associated with these activities, as well as their impacts, are evaluated under Tank Closure Alternatives 2 through 6 (CEES 2010f).

#### **E.1.3.2 Waste Retrieval**

The following technologies could be used in the SSTs to retrieve waste. All of the technologies are flexible with regard to the general configuration of the equipment, fluid velocities and flow rates, and methods of operation. As such, tank-specific considerations such as riser availability, waste condition, or in-tank interferences offer advantages to one retrieval technology over other technologies and lead to the selection of that technology to retrieve waste from a particular tank. The following retrieval technologies have been used or are in use and were addressed in the *TWRS EIS*. However, they are very similar to, and effectively encompassed by, the retrieval technologies evaluated in this *TC & WM EIS*. Therefore, DOE did not consider them reasonable for further consideration and evaluation in this *TC & WM EIS*.

**Past-Practice Sluicing.** Past-practice sluicing is the introduction of liquid (water or liquid waste from another tank) at moderate pressures and high volumes into the waste. Pressure of 1.2 million pascals (180 pounds per square inch) and flows of 1,300 liters (350 gallons) per minute are typical. The liquid dissolves and breaks apart solid materials and suspends them in a waste slurry. A transfer pump inside the tank pumps the slurry to a receiver tank at flows of approximately 1,300 liters (350 gallons) per minute. Past-practice sluicing has been used to remove waste from more than 50 tanks at Hanford and was most recently used on tank 241-C-106. Modified sluicing offers improvements over past-practice sluicing (e.g., less water, better distribution systems) and was selected for analysis in this *TC & WM EIS*.

**Fluidic Mixing.** Fluidic mixing systems consist of in-tank mixing vessels connected to submerged jets or orifices via a pipe. A vacuum is created in the mixing vessel, filling it with solid and liquid waste. The vacuum is released and the mixing vessel is charged, driving the fluid in the mixing vessel back into the tank through the jet and imparting a mixing action on the tank waste. This process is repeated until the tank waste is sufficiently mixed. The waste is pulled into the mixing vessel and driven out of the tank through a pipeline to a receiver tank. The fluidic mixing system has been used to retrieve radioactive waste at ORNL (DOE 2003a).

**Salt Cake Dissolution.** The salt cake dissolution method sprays a solvent (typically water) onto the salt cake surface using a sprinkler-type device. The sprinkler device is designed to distribute the solvent in a uniform manner across the waste surface. The solvent dissolves the salt cake and drains to a centrally located pump. This pump is housed in a salt well screen specifically designed to prevent passage of the larger-sized particles that are inherent with sludge and insoluble waste. The salt cake dissolution retrieval system is designed to retrieve primarily soluble wastes and not to retrieve sludge and other insoluble waste components. A proof-of-concept salt cake dissolution system was designed, tested, and used in tank 241-U-107 (DOE 2003a).

### **E.1.3.3 Treatment Technologies**

Section E.1.2.3.5.1 describes the process DOE used to identify and select supplemental LAW treatment and immobilization techniques. From the dozens of treatment options considered, seven were selected for further consideration based on their expected viability for accelerating cleanup and reducing risk while maintaining cleanup quality.

After further data development and evaluation, the C3T MAI subgroup compared and evaluated the seven options based on criteria developed by the team. The selection criteria addressed worker and public safety, environmental protection, schedule acceleration, cost-effectiveness, operability, and system interface effects. Using both quantitative and qualitative measures for scoring, the subgroup selected three immobilization techniques, bulk vitrification, cast stone, and steam reforming, as well as one pretreatment technology, sulfate removal, for further evaluation and prototype testing. These are the supplemental treatment technologies identified in the Notice of Intent for potential analysis in this *TC & WM EIS*.

The three immobilization techniques, bulk vitrification, cast stone, and steam reforming, as well as the pretreatment technology, sulfate removal, are included under the alternatives analyzed in this EIS and are described in Sections E.1.2.3.6 through E.1.2.3.9.

In addition to the technologies selected by the subgroup for further evaluation, other pretreatment and immobilization technologies may become viable for use in place of or in addition to the technologies analyzed in detail in this *TC & WM EIS*. Decisions on the implementation of these technologies are outside the scope of this *TC & WM EIS*. A future decision to implement any of these technologies would require analysis to demonstrate that the technology performs within the bounds of this EIS.



### **E.1.3.3.1 Technology Readiness Assessment**

In 2007, DOE conducted a Technology Readiness Assessment (TRA) to determine the maturity level of the technologies applied to treatment of Hanford tank LAW. The results of this assessment are documented in DOE's *Hanford River Protection Project Low Activity Waste Treatment: A Business Case Evaluation* (Wade et al. 2007). The following is an excerpt from Appendix B, "Technology Readiness Assessment Results," to this document:

#### **B.1 Description of Technology Assessment**

In 1999, the U.S. General Accounting Office (GAO) produced a report (GAO/NSIAD-99-162) that examined the differences in technology transition between the U.S. Department of Defense (DoD) and private industry. The GAO concluded that the DoD took greater risks and attempted to transition emerging technologies at lesser degrees of readiness compared to private industry and that the use of immature technology increased the overall program risk and led to substantial cost and schedule overruns. The GAO recommended that the DoD adopt the use of National Aeronautics and Space Administration's (NASA) Technology Readiness Levels (TRL) as a means of assessing technology readiness before design transition. In 2001, the DoD Deputy Undersecretary for Science and Technology issued a memorandum that endorsed the use of the TRA process to develop TRLs for new major programs. Guidance for assessing technology readiness was incorporated into the *Defense Acquisition Guidebook* (DoD 2004), which defines a TRA as follows:

"A TRA is a systematic, metric-based process and accompanying report that assesses the readiness of certain technologies [called Critical Technology Elements (CTEs)] used in systems."

The TRA process consists of three parts: (1) identifying critical technology elements (CTE), (2) assessing the TRL of each CTE using an established readiness scale, and (3) preparing the TRA report. The CTE identification process involves breaking the project under evaluation into its component systems and subsystems and determining which of these are essential to project success, and either represent new technologies, are combinations of existing technologies in new or novel ways, or will be used in a new environment. If some of the CTEs are judged to be below the desired level of readiness, the TRA is followed by development of a technology maturation plan that identifies the additional development required to attain the desired level of readiness.

DOE is conducting a pilot program that includes this study to evaluate the use of TRAs in its projects. The TRA process is being adapted for use by DOE from applications by other agencies (e.g., DOE, NASA). Notably, the TRA process is being adapted from use in product development applications to nuclear-chemical engineering process development applications for U.S. Department of Energy, Office of Environmental Management (EM). Use of the TRA process for evaluating LAW alternative technologies represents the first TRA application by EM and therefore is a pilot study that will serve as a basis for subsequent revisions to the use of the TRA process by DOE. The purpose of the TRAs conducted in support of this report was to determine the readiness level of the technologies as applied to the treatment of Hanford Site tank LAW.

The TRL scale used for the Hanford Site TRAs is shown in Table B.1. The DoD policy requires that testing of a prototypical design in a relevant environment be completed before incorporation of the technology into the final design of the facility. Thus, for DoD, a TRL 6

must be achieved prior to proceeding to Milestone B, which is the entrance point for the initiation of a system acquisition program.

The testing requirements used for the Hanford Site TRAs are compared to the TRLs in Table B.2. These definitions provide a convenient means to display the relationship between the scale of testing, fidelity of testing system, testing environment, and the TRL. The goal is to achieve a TRL 6 prior to incorporation of the technology into the final design. In order to attain a TRL 6, testing must be completed at an engineering or pilot scale, with testing of the system fidelity that is similar to the actual application and with a range of simulated waste and/or limited range of actual waste, if applicable.

DOE's assessment of the TRLs was aided by a TRL Calculator that was originally developed by the U.S. Air Force (Nolte et al. 2003) and modified by the DOE Assessment Team. This tool is a standard set of questions addressing hardware, software program, and manufacturability questions and is implemented in Microsoft Excel™. The set of questions provide the criteria used to assess and determine the TRL numerical value. The TRL Calculator produces a graphical display of the TRLs achieved and was used by the Assessment Team in establishing TRLs.

**Table B.1. Technology Readiness Levels Used in this Assessment**

<b>Relative Level of Technology Development</b>	<b>Technology Readiness Level</b>	<b>TRL Definition</b>	<b>Description</b>
System Operations	9	Actual system operated over the full range of expected conditions	The technology is in its final form and operated under the full range of operating conditions. Examples include using the actual system with the full range of wastes.
System Commissioning	8	Actual system completed and qualified through test and demonstration	The technology has been proven to work in its final form and under expected conditions. In almost all cases, this TRL represents the end of true system development. Examples include developmental testing and evaluation of the system with real waste in hot commissioning.
	7	Full-scale, similar (prototypical) system demonstrated in relevant environment	This represents a major step up from TRL 6, requiring demonstration of an actual system prototype in a relevant environment. Examples include testing the prototype in the field with a range of simulants and/or real waste and cold commissioning.
Technology Demonstration	6	Engineering/pilot-scale, similar (prototypical) system validation in relevant environment	Engineering-scale models or prototypes are tested in a relevant environment. This represents a major step up in a technology's demonstrated readiness. Examples include testing a prototype with real waste and a range of simulants.

*Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington*

<b>Relative Level of Technology Development</b>	<b>Technology Readiness Level</b>	<b>TRL Definition</b>	<b>Description</b>
Technology Development	5	Laboratory scale, similar system validation in relevant environment	The basic technological components are integrated so that the system configuration is similar to (matches) the final application in almost all respects. Examples include testing a high-fidelity system in a simulated environment and/or with a range of real waste and simulants.
	4	Component and/or system validation in laboratory environment	The basic technological components are integrated to establish that the pieces will work together. This is relatively "low fidelity" compared with the eventual system. Examples include integration of ad hoc hardware in a laboratory and testing with a range of simulants.
Research to Prove Feasibility	3	Analytical and experimental critical function and/or characteristic proof of concept	Active research and development (R&D) is initiated. This includes analytical studies and laboratory-scale studies to physically validate the analytical predictions of separate elements of the technology. Examples include components that are not yet integrated or representative. Components may be tested with simulants.
Basic Technology Research	2	Technology concept and/or application formulated	Once basic principles are observed, practical applications can be invented. Applications are speculative, and there may be no proof or detailed analysis to support the assumptions. Examples are still limited to analytic studies.
	1	Basic principles observed and reported	This is the lowest level of technology readiness. Scientific research begins to be translated into applied R&D. Examples might include paper studies of a technology's basic properties.

**Table B.2. Relationship of Testing Requirements to the Technology Readiness Level**

TRL	Scale of Testing <sup>1</sup>	Fidelity <sup>2</sup>	Environment <sup>3</sup>
9	Full	Identical	Operational (Full Range)
8	Full	Identical	Operational (Limited Range)
7	Full	Similar	Relevant
6	Engineering/Pilot	Similar	Relevant
5	Laboratory	Similar	Relevant
4	Laboratory	Pieces	Simulated
3	Laboratory	Pieces	Simulated
2	Paper	Paper	Paper
1	Paper	Paper	Paper

- <sup>1</sup> Full Scale = Full plant scale that matches final application  
 1/10 Full Scale < Engineering/Pilot Scale < Full Scale (Typical)  
 Lab Scale < 1/10 Full Scale (Typical)
- <sup>2</sup> Identical System – configuration matches the final application in all respects  
 Similar System – configuration matches the final application in almost all respects  
 Pieces System – matches a piece or pieces of the final application  
 Paper System – exists on paper (no hardware)
- <sup>3</sup> Operational (Full Range) – full range of actual waste  
 Operational (Limited Range) – limited range of actual waste  
 Relevant – range of simulants + limited range of actual waste  
 Simulated – range of simulants

## B.2 Summary of LAW Treatment TRA Results

The TRA conducted to support this report were performed by DOE following the process described above. Results from the two TRAs identified below were utilized to support the business case study presented in this report. Table B.3 provides a summary of the TRLs related to each business case.

**Table B.3. Business Case Technology Readiness Level Summary**

Business Case and SI Approach	Critical Technology Elements (Systems)	TRLs
<b>1 – WTP LAW Only</b>	Container Sealing	5
	Decontamination	4
	LAW Melter Feed	6
	LAW Melter	6
	Melter Offgas and Vessel Vent Process	6
<b>2 – 2nd WTP LAW</b>	Same as Case 1	Same as Case 1
<b>3 – BV</b>	Feed Receipt, Feed Preparation, and Feeding	5
	In-Container Vitrification	5
	Offgas Treatment	4
<b>4 – CS</b>	Feed Receipt, Feed Preparation, and Feeding	3
	Mixing and Casting	3
<b>5 – SR</b>	Feed Receipt, Feed Preparation, and Feeding	4
	Thermal Reformer	5
	Offgas Treatment	4
	Container Handling and Waste Qualification	3
<b>6 – BV East and West w/TF Pretreatment Facility</b>	BV Same as Case	3
	Rotary Filtration	3
	Cesium ion exchange	3
<b>7 – LAW First and BV w/TF Pretreatment Facility</b>	Same as Cases 1 and 6	Same as Cases 1 and 6

Note: SI=Supplemental Immobilization. In Business cases 6 and 7, tank farm-based pretreatment TRLs are also provided.

Table B.4 provides a summary of the Waste Treatment and Immobilization Plant (WTP) Low-Activity Waste (LAW) results from 07-DESIGN-042, *Technology Readiness Assessment for the Waste Treatment Plant (WTP) Analytical Laboratory, Balance of Facilities and LAW Waste Vitrification*.

**Table B.4. Technology Readiness Level Summary WTP LAW  
Critical Technology Elements**

<b>Critical Technology Element/Description</b>	<b>Technology Readiness Level</b>	<b>Rationale</b>
<p><b>ILAW Container Finishing Handling System (LFH) Container Sealing Subsystem</b> The ILAW container sealing subsystem press fits and locks a flat circular lid into a circular groove in the container neck.</p>	<b>5</b>	The container sealing system design is based on existing technologies but has not been demonstrated as an integrated prototypical system in an operating environment.
<p><b>LFH Decontamination Subsystem</b> The LFH decontamination subsystem sprays carbon dioxide pellets at ILAW container surfaces to remove radioactive contamination. The sublimed carbon dioxide and dislodged contamination are contained by a vacuum system and shroud.</p>	<b>4</b>	The ILAW container decontamination design is based on existing technology concepts, but has not been demonstrated as an integrated, prototypical system in a relevant environment. Testing on a laboratory scale of the carbon dioxide spray to decontaminate flat-metal specimens has been completed; testing did not demonstrate the WTP Project's requirement on surface decontamination levels. Integrated testing of the robot, carbon dioxide spray, and shrouding system has not been carried out on the complex surfaces of the ILAW container.
<p><b>LAW Melter Feed Process System (LFP)</b> The LFP mixes LAW Facility waste and glass formers to provide feed for the LAW melters.</p>	<b>6</b>	There has been extensive WTP and vendor testing to demonstrate the adequacy of the mixing systems.
<p><b>LAW Melter System (LMP)</b> The LMP is the LAW melter system that melts mixtures of LAW and glass formers.</p>	<b>6</b>	The LAW melter has a significant development basis in previous DOE projects and developmental tests for the WTP. However, risk remains with the availability of MA758, a high chromium alloy used for the LAW bubbler assembly. An alternate bubbler material of construction should be identified.
<p><b>LOP/LVP</b> The LOP/LVP is the LAW Melter Offgas and Vessel Vent Process Systems that remove aerosols, gases, and particulates generated by the LAW melters and vessel vent streams.</p>	<b>6</b>	The LOP/LVP have a significant technology basis. Two of 12 MACT DRE tests for naphthalene conducted on a prototypical system did not attain the required destruction efficiency. Engineering analysis shows that the WTP system should attain MACT standards based on higher capacities of the plant unit operations as compared to the pilot plant unit operations.

The DOE TRA cited above assesses Business Case No. 7 (LAW First and Bulk Vitrification with Tank Farm Pretreatment) at Technology Readiness Levels of 3 to 6. However, at the time of the preparation of this *TC & WM EIS*, DOE had not made a decision on whether to support implementation of this Business Case. Such a decision requires, minimally, completion and DOE approval of its project

management system Critical Decision 0 (CD-0) requirements. These requirements include, for example, approval of a Mission Need for the project and the completion of preconceptual design planning and analysis. On December 21, 2007, DOE approved the Justification of Mission Need Statement for an Interim Pretreatment System Project (Rispoli 2007). However, no design information was included in this project documentation to support analyzing tank farm pretreatment in the *Draft TC & WM EIS*.

Differences between the LAW First and Bulk Vitrification with Tank Farm Pretreatment Business Case and this EIS identified at this time include the following:

- Solids and cesium separation using rotary microfiltration and ion exchange, respectively
- Secondary waste, which would include additional inventories of technetium-99 and iodine-129 that would be disposed of in an IDF
- Upgrades to the ETF

However, it is recognized that the above differences could change as design information matures.

In 2008, DOE commissioned an external technical review of the system planning for alternative supplemental treatment of LAW at Hanford. The report from this review concluded that, although the current schedule for completion of the WTP LAW Vitrification Facility and supporting facilities could support early treatment of LAW in 2014, such early startup would require an interim pretreatment capability and means for disposition of secondary waste (Kosson et al. 2008). Since 2008, DOE has been evaluating the transition of the WTP from construction to commissioning. Section E.1.3.3.2 provides the current status of this evaluation.

#### **E.1.3.3.2 Vision for WTP Project Transition to Operations**

As of the end of calendar year 2010, WTP design and construction were 81 and 57 percent complete, respectively. In response to DOE's vision for transitioning the WTP to operations, Washington River Protection Solutions, LLC, and Bechtel National, Inc., submitted a proposal in March 2011 (WRPS and BNI 2011) for the early commissioning and operation of the WTP. The following summarizes the scope of this proposal.

The WTP consists of several facilities that vary in construction progress and their ability to begin operations; farthest along are the LAW Vitrification Facility, the Analytical Laboratory, and the BOF. To accomplish this strategy to successfully and effectively start up and operate all of the WTP facilities, beginning with the LAW Vitrification Facility, the Analytical Laboratory, and the BOF, DOE has identified the following actions that will support the vision of WTP hot operations beginning earlier, i.e., 2016:

- Use a phased turnover approach that will allow WTP facilities to be transitioned to an operations state on as short a timeline as credible.
- Align tank farm operations and WTP objectives so that waste feed can be supplied to the WTP when it is required.
- Ensure immobilized waste and recycle streams can be received by the tank farms to support the December 2016 production of ILAW.
- Ensure funding is available to support these efforts.
- Modify the tank farm and WTP contracts to support the early startup of these facilities.

One of the regulatory elements to be addressed is the NEPA evaluation. The required NEPA analysis of the WTP facilities has been updated from the *TWRS EIS* in both the *Draft* and this *Final TC & WM EIS*. Both versions provide discussion and analysis of the infrastructure needed to support early operations of the identified facilities within the WTP, including the following:

**INTERIM PRETREATMENT SYSTEM (CESIUM AND SOLIDS REMOVAL)**

The scope of this project is to upgrade the AP tank farm to include rotary microfiltration and cesium ion exchange pretreatment of tank waste to feed the LAW Vitrification Facility. The focus of these technologies is solids filtration and cesium removal.

All Tank Closure alternatives except the No Action Alternative include two facilities that could support these functions. Section E.1.2.3.1.1 describes the WTP Pretreatment Facility, which includes cesium ion exchange, filtration, and solids separations capabilities. Section E.1.2.3.5.2, Separations Activities, describes a potential new facility in the 200-West Area, the Solid-Liquid Separations Facility, whose purpose is to provide solids separations through decanting. Cesium ion exchange capability was not assumed to be included in this facility, but adequate space is available if needed.

The analysis in this *TC & WM EIS* treats cesium ion exchange and solids filtration as inherent to all waste processing through the Pretreatment Facility for the life of the plant. Accordingly, the impacts of that analysis are expected to bound the impacts of planned waste treatment within the AP tank farm to feed the LAW Vitrification Facility. Emissions in fiscal year 2017 resulting from waste pretreatment in the AP tank farm and those in fiscal year 2018 from Pretreatment Facility operations as analyzed in this *EIS* would be comparable.

**INTERIM LAW WASTE FEED DELIVERY**

The scope of this project is to provide staging tanks to support feeding waste to the LAW Vitrification Facility. Staging capacity needed is approximately 170,343 liters (45,000 gallons), or three 56,781-liter (15,000-gallon) tanks.

**Storage Capacity.** All Tank Closure alternatives except the No Action Alternative involve the construction, operations, and deactivation of additional storage capacity. Section E.1.2.2.8 discusses the storage capacity of four WRFs. A single WRF can store approximately 1.7 million liters (450,000 gallons) of waste. Section E.1.2.2.9 discusses the storage capacity for a new DST. A single DST can store approximately 4.5 million liters (1.2 million gallons) of waste.

It is also possible to use tanks procured for the discontinued DBVS project, which are currently stored on site and have adequate capacity. The impacts analysis of the four WRFs in this *TC & WM EIS* is expected to bound the impacts of providing the needed storage capacity for feed to the LAW Vitrification Facility.

**Transfer Lines.** The scope of this project is to provide transfer lines to support feeding waste to the LAW Vitrification Facility. This *TC & WM EIS* evaluates the construction, operations, and deactivation of transfer lines.

All Tank Closure alternatives except the No Action Alternative involve new transfer lines. Section E.1.2.2.7.3 discusses the use of hose-in-hose transfer lines; Section E.1.2.2.7.5, new underground transfer lines.

The impacts analysis of transfer lines in this *TC & WM EIS* is expected to bound the impacts of those lines needed for tank farm operations to support feeding waste to the LAW Vitrification Facility.

### **INTERIM SECONDARY-LIQUID-WASTE HANDLING**

The scope of this project is to return the secondary liquid waste generated in the LAW Vitrification Facility and the Analytical Laboratory to the DST farms after volume reduction in the 242-A Evaporator.

**Liquid Waste Management.** All Tank Closure alternatives except the No Action Alternative involve replacement of the 242-A Evaporator. Section E.1.2.3.2 describes replacements for the 242-A Evaporator and provides assumptions as to their operation. This EIS assumes evaporator campaign runs of approximately 42 days each year. The 42-day runs are expected to provide adequate capacity to support the early startup of the LAW Vitrification Facility and the Analytical Laboratory.

**Interim Secondary Waste.** All Tank Closure alternatives except the No Action Alternative involve the handling of secondary liquid waste. The *Draft TC & WM EIS* noted potential issues related to disposal of secondary waste from the WTP and other processes during the period when only the LAW Vitrification Facility and the Analytical Laboratory would operate. However, because such wastes would be transferred back to the DST farms, it has been concluded that long-term disposal performance of secondary-liquid-waste forms would not be an issue.

### **LAW PRODUCT AND SECONDARY-SOLID-WASTE HANDLING AND DISPOSAL**

The scope of this project is to provide support for the handling and disposal of ILAW and WTP-generated secondary solid waste in an IDF.

All Tank Closure alternatives except the No Action Alternative involve disposal of ILAW glass and WTP-generated secondary solid waste.

Waste Management Alternatives 2 and 3 include the disposal of ILAW and secondary solid waste at IDF-East. The impacts analysis of the disposal of ILAW and WTP-generated secondary solid waste in this *TC & WM EIS* is expected to bound the impacts of similar handling and disposal operations included in the March 2011 proposal (WRPS and BNI 2011).

Transportation of various waste streams generated at Hanford and to be stored or disposed of at Hanford is analyzed in this EIS. This applies to waste that is designated for disposal in IDF (e.g., ILAW glass and secondary solid waste) and to IHLW, which would be stored in the IHLW Interim Storage Facilities until a disposal path is found. The impacts analysis of the transportation of various waste streams in this *TC & WM EIS* is expected to bound the impacts of similar transportation activities included in the March 2011 proposal (WRPS and BNI 2011).

### **SECONDARY-LIQUID-WASTE DISPOSAL/EFFLUENT TREATMENT FACILITY UPGRADES**

The scope of these projects is to upgrade the ETF and the LERF. The baseline case is to support WTP operations in 2018.

All Tank Closure alternatives except the No Action Alternative involve the construction, operation, and deactivation of an ETF replacement. The LERF was assumed to be adequate as currently configured; therefore, no upgrades were analyzed. Although not analyzed, ongoing studies by DOE to support management of future WTP secondary-liquid-waste streams, including the addition of a solidification capability at the ETF, are discussed in Section E.1.2.3.3.4. As stated in that section, DOE believes potential impacts of ETF replacement would bound the impacts of any addition of a solidification capability at the ETF.



### **HLW FEED DELIVERY SYSTEM**

The scope of this element is to upgrade the DST farms to support delivery of HLW to the WTP facilities. While the specifics of these upgrades are not well defined in the March 2011 proposal (WRPS and BNI 2011), it is expected that their impacts would be bounded by those of a number of activities analyzed in this EIS: tank system upgrades (Section E.1.1.2), waste retrieval and storage (Section E.1.2.2), hose-in-hose transfer lines (Section E.1.2.2.7.3), and new underground transfer lines (Section E.1.2.2.7.5).

### **IHLW INTERIM STORAGE FACILITIES**

The scope of this project is to provide interim storage of IHLW canisters at Hanford pending a decision by DOE related to the ultimate disposal of IHLW.

All Tank Closure alternatives except the No Action Alternative involve the storage of IHLW canisters. Section E.1.2.1.3 describes the CSB and IHLW Interim Storage Facilities. The *TWRS EIS* evaluated storage capacity for two CSBs; the *Draft TC & WM EIS*, a range of alternatives addressing storage of most of the IHLW canisters; and this *Final TC & WM EIS*, the storage of all IHLW canisters produced under each alternative. The number of canisters to be stored depends on the specifics of the alternative.

### **E.1.3.3.3 Low-Activity Waste Treatment and Immobilization Technologies**

The following subsections describe potential LAW treatment and immobilization technologies that were not included under the alternatives analyzed in this *TC & WM EIS*. Each discussion addresses the reasons the technology was not analyzed.

#### **E.1.3.3.3.1 Active-Metal Reduction**

The active-metal reduction process was one of the potential LAW treatment and immobilization technologies evaluated by the MAI subgroup. This technology was not selected for further analysis. The technology scored lower than the other technologies considered, primarily on the basis of its technical immaturity, complexity, and the operational safety issues involved because of its flammable gas generation.

The active-metal reduction process uses a stirred tank or a fluidized-bed reactor to decompose nitrates and nitrites present in the salt cake waste. The sodium aluminate–aluminum hydroxide product from the active-metal reduction reactor is mixed with phosphoric acid or silica to produce a sodium aluminosilicate or aluminophosphate ceramic waste form. The active-metal reduction process has been demonstrated on a laboratory scale, but the process chemistry has not been entirely confirmed. This process is a potentially complete LAW treatment option with no streams returning to DSTs. Further research is required to confirm chemical reactions on actual waste, the liquid effluent stream and offgas compositions, and the performance of the phosphate-bonded ceramic waste form produced (DOE 2003f).

#### **E.1.3.3.3.2 Fractional Crystallization**

The WTP Pretreatment Facility currently being constructed in the 200-East Area would separate tank waste into HLW and LAW fractions. Fractional crystallization is a technology being considered as a potential supplemental pretreatment process in the 200-West Area that would primarily target medium-curie tank waste from SSTs. As a supplemental separations technology, the clean salt technology would be employed after selective dissolution, settling, decanting, and ion exchange processes—steps that are already assumed as a baseline 200-West Area separations process for pretreating tank waste streams. The ion exchange pretreatment process would require the construction of a new facility. The selective dissolution, settling, and decanting steps would not require new facilities

and would be incorporated into the retrieval process and the proposed 200-West Area Solid-Liquid Separations Facility, which is described in Section E.1.2.3.5.2.

The medium-curie tank waste, characterized by high sodium and sulfate content, presents operational challenges for the WTP LAW melters (Hamilton 2005). Supplemental treatment technologies are being considered that could increase WTP LAW processing capacities. In contrast, when considering final waste performance and worker exposures, it is more desirable to divert medium-curie waste with certain radionuclides (e.g., cesium-137 and technetium-99) to the WTP LAW melters (Hamilton 2006b). The proposed 200-West Area Solid-Liquid Separations Facility already includes methods for fractioning radionuclides and reducing sulfate content. Fractional crystallization would go beyond this separations technology by also reducing the sodium content of the waste stream being directed back to the WTP process. This would increase the waste loading in the WTP LAW melters. In essence, the objectives of the fractional crystallization would be to (1) reduce the radioactivity (e.g., cesium-137) in the LAW feed into the supplemental treatment facility, and (2) reduce the sodium content of the WTP LAW feed (Hamilton 2006b). Sulfate removal by precipitation, if necessary, could also be incorporated into the fractional crystallization process facility to accomplish a third objective.

The fractional crystallization process was not evaluated in detail due to the lack of available data demonstrating this process on actual tank wastes. If new R&D suggests favorable results, the implementation of fractional crystallization for separations may be evaluated in detail at a later date. The core process for the options uses the same precipitation process to achieve separation. The options are as follows:

- Fractional crystallization for radionuclide separations and sodium reduction
- Fractional crystallization for radionuclide separations with sodium and sulfate reduction

For the fractional crystallization option (without sulfate removal), the sodium nitrate crystals are formed by evaporating the liquid waste below the sodium nitrate solubility in a reduced atmosphere. These crystals are separated from the highly radioactive liquid and washed. The liquid fraction, containing most of the radionuclides (e.g., cesium-137, iodine-129, and technetium-99) would be returned to the DST system for eventual processing in the WTP and the washed salts would be re-dissolved with the process condensate and immobilized as ILAW glass.

For the option with sulfate removal, an additional precipitation step utilizing strontium nitrate would be used to remove sulfate from the waste stream. Following the sulfate reduction, the separated supernatant would be treated by fractional crystallization as described above. The re-dissolved solids from fractional crystallization would then be mixed with the precipitated solids from the sulfate removal step. This slurry would be immobilized with a LAW supplemental treatment technology.

Fractional crystallization, a commercially proven process that is typically used for pharmaceuticals (purifying drugs) and industrial chemicals (cleansers, fertilizers, etc.), works by evaporating feed stocks and selectively forming pure crystalline products (Hamilton 2006b).

Laboratory-scale testing of fractional crystallization was completed in 2005 on simulated tank waste. Phase I laboratory testing of fractional crystallization using simulated waste is part of a two-phase Pretreatment Testing and Demonstration Program designed to determine whether fractional crystallization is a viable supplemental pretreatment process for Hanford medium-curie waste. The laboratory study was conducted by a technical team commissioned by CH2M HILL Hanford Group, Inc., that was composed of COGEMA, Inc.; Georgia Institute of Technology; AREVA Framatome ANP; and Swenson Technologies, Inc. The results of the study are presented in the *Hanford Medium/Low Curie Waste Pretreatment Project – Phase I Laboratory Report* (Hamilton 2006b).

Simulants tested during Phase I of the program represented the following anticipated tank liquid waste streams: (1) early SST feed, (2) late SST feed, and (3) DST feed. The purpose of demonstrating feasibility on early versus late SST feed is to determine the ability of fractional crystallization to cope with variability in liquid waste feeds. In general, the fractional crystallization process desires homogenous waste feed. The purpose of using DST tank waste feed simulants is to determine the effects of high aluminum ion content on the process, which typically characterizes DST feed. The higher concentration of aluminum in DST feed may lead to the formation of aluminum-based gels prior to satisfactory sodium recovery. The formation of gels can lead to difficulties in slurry handling (Hamilton 2006b).

The results of the Phase I laboratory-scale tests were very favorable for continued development of the technology for potential use on Hanford tank waste. Laboratory-scale tests demonstrated successful exclusion of cesium from crystalline product and successful reduction of sulfate-to-sodium ratios for all three waste simulants, as well as sufficient sodium recovery for both SST feeds, but not quite for the DST feed. Further testing and manipulation of the waste feed pH through reactions involving hydroxyl ions and carbon dioxide reduced the formation of aluminum-based gels and increased sodium recovery. Therefore, unless sodium recovery objectives can be achieved through process design refinements, the pretreatment of waste using fractional crystallization may be limited to SST feed (Hamilton 2006b).

Phase I testing, although found to meet the objectives for cesium exclusion from crystalline product, suggested that some cesium-bearing liquor may become entrapped within inclusions and interstitial spaces of the crystallized salt. It is expected that washing the crystals would remove the cesium residing in the interstitial spaces. However, cesium retained in inclusions would likely remain within the crystal phase (Hamilton 2005). The significance of the effect of inclusions on radionuclide separations would need to be evaluated during subsequent laboratory-scale testing.

Continuation of the Pretreatment Testing and Demonstration Program under Phase II occurred in 2006 and 2007. Phase II of the program includes laboratory-scale testing of fractional crystallization on actual radioactive Hanford tank waste and pilot-scale operational testing and integration of process equipment. Phase II testing continued to show fractional crystallization as a viable technology. However, in 2008, CH2M HILL Hanford Group selected ion exchange for cesium separation over fractional crystallization and caustic-side solvent extraction for their interim pretreatment system in the Hanford tank farms. Therefore, only limited testing of fractional crystallization will be continued to ensure an alternative cesium removal technology for providing a waste feed supply to the WTP. Ion exchange was selected over caustic-side solvent extraction and fractional crystallization because it would provide the earliest possible deployment of the interim pretreatment system. Implementation schedules showed the interim pretreatment system could be implemented approximately 1 year earlier if ion exchange were selected over fractional crystallization, and approximately 2 years earlier if ion exchange were selected over caustic-side solvent extraction. Further, the ion exchange capital and life-cycle costs were estimated to be significantly lower than those of the other two technologies (CEES 2010g).

A different option, called “clean salt,” was considered by the MAI subgroup under two different flowsheet options; however, neither of the clean salt options were selected by the MAI subgroup for further evaluation (CH2M HILL 2003a).

The MAI subgroup evaluated clean salt, assuming that the sodium nitrate product from the clean salt options (without sulfate removal) would be immobilized in a phosphate-bonded ceramic. The strontium sulfate precipitate from the clean salt with sulfate removal options would be incorporated into a grout matrix. The sodium nitrate crystals from the clean salt with sulfate removal options would be immobilized by microencapsulation using a polyethylene polymer (DOE 2003f).

After the tank waste underwent selective dissolution and solid-liquid separations, the liquid waste stream would be sent to an acid reactor where it would be acidified with nitric acid. Acidification neutralizes sodium hydroxide and sodium carbonate, dissolves aluminum hydroxide, and converts all the nitrite to nitrate and nitrogen oxide. The primary form of sodium leaving the acid reactor would be sodium nitrate. Crystallization of sodium nitrate would be performed in two steps. First, the acidified liquid stream would be evaporated at approximately 110 °C (230 °F) at atmospheric pressure until about 50 percent by weight of the sodium nitrate was crystallized out. Second, the liquid stream would be cooled, resulting in the crystallization of another 20 percent by weight of sodium nitrate (Gasper et al. 2002). Fractional crystallization was assumed to be adequate to produce sufficient radionuclide decontamination to meet applicable requirements. Select radioactive components (particularly cesium-137) would be excluded from the crystal matrices, thus yielding a radioactively enriched effluent that can be directed to the WTP process. Because of the inherent ability of this technology to separate radionuclides from the clean salt, a separate ion exchange facility for the removal of cesium and technetium may not be necessary.

The washed crystals would be separated from the filtrate in a solid-liquid separations step and then sent to the immobilization step. Waste sodium nitrate needs to be immobilized because, as a waste form, it does not comply with land disposal restrictions due to its toxicity and ignitability. The key problem identified would be achieving an immobilized waste form for the crystallized sodium nitrate that could meet disposal requirements and did not result in a significant increase in the volume of immobilized waste to be disposed of at Hanford (DOE 2003f).

#### **E.1.3.3.3.3 High-Level Radioactive Waste and Low-Activity Waste Vitrification and Phosphate Glass**

It has been proposed that the use of a phosphate glass formula for Hanford waste vitrification would have some advantages over the current baseline borosilicate glass. Hanford tank waste has some chemical constituents that are troublesome to incorporate into the base program ILAW and IHLW borosilicate glasses. The low solubility of sulfate in silicate glasses limits the concentration of sodium oxide in the ILAW glass. Without the sulfate problem, an increase in waste loading would be possible for ILAW glass. Sulfate incorporation and chemical durability have been demonstrated in the laboratory for phosphate glasses formulated for Hanford ILAW. Similarly, for IHLW glass, the chromium solubility limits the waste loading in the baseline borosilicate glass. High chromium content may be incorporated by adding phosphate to the waste feed and operating at 1,200 to 1,250 °C (2,190 to 2,280 °F). Increased waste loading can be accommodated, and the lower viscosity of the resulting melt allows a shorter residence time in the melter. These factors offer the potential for improved IHLW glass throughput at the WTP. This option was not considered for evaluation in this *TC & WM EIS* because the phosphate glass formula has not been proven to be compatible with production-scale melters, and the resulting product glass has not been shown to meet the waste acceptance technical requirements for DOE's Civilian Radioactive Waste Management System (DOE 2007). DOE recently reviewed the most recent technical data and concluded that there are no referenceable data that address issues that need to be addressed, such as the potential impacts on the current WTP flowsheet, waste throughput, offgas system requirements, and physical space requirements for phosphate melters (CEES 2010h).

#### **E.1.3.3.3.4 Preprocessing Tank Waste with a Plasma Mass Separator (Archimedes Technology)**

DOE, as part of the accelerated cleanup program, issued a Request for Information in October 2003 seeking data on potential HLW processing options. The Request for Information targeted technical application areas for an enhanced separation process, low-temperature immobilization, thermal immobilization, waste retrieval, and tank closure. The Archimedes Technology Group provided a response to the Request for Information.

The plasma mass separation process physically separates elements based on their atomic mass. The process relies on the physics of rotating plasmas to establish a so-called “mass cutoff,” which is indifferent to the complexity of waste input. Elements with an atomic mass unit above the mass cutoff are simply sent in one direction, while elements below the cutoff are sent in another.

Waste is injected into a vacuum chamber. Radio frequency power converts the injected material into a plasma (an atom transforms into a plasma state when an outer electron leaves its atomic orbit). The resulting positively charged ions in the plasma respond to electric and magnetic forces. Electric and magnetic fields combine to rotate or spin the plasma much like these two force fields spin an electric motor. When the plasma rotates at high speeds, the magnetic and electric fields can be adjusted to create a mass cutoff that keeps light ions confined to the plasma while heavy ions rotate instantaneously to the side wall collector.

This *TC & WM EIS* does not evaluate preprocessing tank waste with a plasma mass separator due to the present immaturity of the technology. The project team recommended that DOE conduct further testing and demonstration of the process and a thorough evaluation of the business model prior to considering implementation. The Archimedes Technology Group is currently implementing a demonstration program for processing Hanford tank waste (DOE 2003f).

#### **E.1.3.3.3.5 Additional Technologies**

Section B.9.0 of the *TWRS EIS* includes additional potentially viable pretreatment and immobilization technologies that are available for use should they prove to be technically viable and perform within the bounds of the environmental impacts addressed in this *TC & WM EIS*.

Some technologies were rejected from further analysis because of the inability to adequately treat Hanford waste. For example, in situ (in place) vitrification was eliminated from consideration as an option for supplemental treatment because vitrification of the tank waste in situ (in the tanks) would not achieve the objective of removing 99 percent of the waste from the tanks (DOE 2003f).

#### **E.1.3.4 Disposal**

Options for offsite disposal of ILAW glass (MLLW) and secondary waste (LLW and MLLW) and onsite disposal of the HLW melter taken out of service were considered. The following sections provide information about these disposal options and the reasons they are not included in the alternatives analyzed in this *TC & WM EIS*.

##### **E.1.3.4.1 ILAW Glass and Secondary-Waste Disposal**

A primary advantage of the offsite disposal of ILAW glass (MLLW) and secondary waste (LLW and MLLW) from tank waste treatment would be that it would not require construction, operation, and closure of waste disposal facilities at Hanford. The *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste (WM PEIS)* (DOE 1997a) provided analysis of potential environmental impacts of broad alternatives for DOE’s waste management program to provide a basis for DOE decisions on programmatic configurations of sites for waste management activities. One of DOE’s decisions based on the *WM PEIS* addressed disposal of LLW and MLLW: DOE decided that Hanford would dispose of its own LLW and MLLW on site (65 FR 10061). As a result, the option of disposing of these wastes off site was eliminated from further consideration in this EIS.

#### **E.1.3.4.2 High-Level Radioactive Waste Melter Disposal**

Onsite disposal of the HLW melter taken out of service was considered. WTP HLW melter taken out of service would be packaged within an overpack at the WTP. The overpack would provide shielding and confinement. The overpack would be 5.29 meters long by 5.29 meters wide by 4.38 meters high (17.4 feet long by 17.4 feet wide by 14.4 feet high). The overpack containing a melter would weigh approximately 263 metric tons, and the dose rate at 30 centimeters (11.8 inches) from the overpack surface would be less than 16 millirem per hour (Lowe and Haigh 2003).

As the HLW melter has not been installed or operated, a high degree of uncertainty exists about their operation, lifespan, waste characterization, and waste classification. As a result, this *TC & WM EIS* assumed a conservative (i.e., economical and with consideration of the human health impacts of melter storage, transportation, and disposal) disposition of the melter: the HLW melter would be stored on site until disposition decisions are made and implemented. Thus, onsite disposal was eliminated from further consideration in this EIS.

#### **E.1.3.5 Tank System Closure**

Several technologies providing in situ soil remediation and alternatives to support tank farm closure were considered but not selected for detailed analysis in this *TC & WM EIS*. The following provides an overview of these technologies and the rationale for not analyzing each.

##### **E.1.3.5.1 Subsurface Barriers**

Underground containment barriers could be an important method for limiting and/or eliminating the movement of contaminants through the subsurface and minimizing impacts on human health and the environment. Subsurface barriers could contain the volume of waste and reduce the potential for migration through the vadose zone and into the groundwater.

Many subsurface barrier technologies are commercially available and others are in various stages of development. The purpose and function of the subsurface barrier system must be determined prior to designing and constructing the barrier. Site characterization is an essential part of choosing an appropriate barrier.

Several factors should be considered when designing a subsurface barrier. First, it is important to establish the barrier geometry (e.g., alignment, depth, and thickness). Second, a stress-deformation analysis should be performed on the surrounding area to assess the potential impacts of barrier construction. Third, compatibility testing must be done to select the most effective barrier materials and, when necessary, appropriate mixture combinations. Fourth, it is necessary to determine the most effective and feasible construction methods. Finally, construction quality assurance/quality control, along with monitoring, is a crucial component of subsurface barrier design.

Construction quality assurance and quality control are essential for the successful design, implementation, and performance of subsurface barrier systems. Different types of subsurface barriers have different construction quality control criteria; however, there are two primary concerns. First, the installed barrier must have a hydraulic conductivity equal to or less than that specified in the design. The second concern is barrier continuity.

The use of subsurface barriers in the vadose zone below and around the sides of Hanford SSTs has previously been evaluated. Two types of potential installations were studied: “close-coupled” and “mega” (massive) applications. Barrier installation would be difficult because of the proximity to other tanks; application of close-coupled barriers would therefore be limited. Installation of mega subsurface barriers would cocoon entire tank farms. In addition, upon construction of the subsurface barrier, no

method is available to measure the completeness of the subsurface barrier or methods to determine how much leaks through.

Previous studies to evaluate, among other things, the application of existing subsurface barriers and their potential for the reduction of environmental impacts have concluded the following:

- The use of subsurface barrier concepts in general tank farm applications would result in only a small reduction of the risk associated with waste retrieval, tank stabilization, and surface barrier technologies.
- Uncertainty about the performance of subsurface barriers (e.g., verification of placement and performance in immobilizing contaminants) is high.
- Potential risks to workers involved in implementing subsurface barrier alternatives increase by factors of up to 15 compared with those of surface barriers and waste retrieval.

Recent developments regarding the use of reactive barriers in soils to immobilize contaminants appear promising. However, detailed evaluation of reactive barriers for closure of Hanford tanks is not practical because limited information is currently available for this application (DOE 2003d).

#### **E.1.3.5.2 In Situ Soil Remediation**

In situ (in place) methods that are generally less expensive and disruptive to the natural landscape, hydrology, and ecosystems than conventional excavation, treatment, and disposal methods are options for soil remediation. The main advantage of in situ remediation is that it allows soil to be treated without being excavated and transported. Consequently, in situ remediation offers a likely reduction of environmental disturbance and personnel exposure to hazardous materials. However, in situ treatment generally requires long periods of time, there is uncertainty about the uniformity of treatment because of the variability in soil and aquifer characteristics, and the efficacy of the process is difficult to verify. In situ soil remediation of metals and radionuclides using inexpensive additives such as minerals (apatite, zeolite, or clay minerals) or waste byproducts (steel shot, beringite, or iron-rich biosolids) is a promising alternative to current remediation methods. In situ remediation techniques rely on a fundamental understanding of the natural geochemical processes governing the speciation, migration, and bioavailability of metals and radionuclides in the environment. In contaminated soils, metals and radionuclides can be dissolved in solution, held on inorganic soil particles, complexed with organic soil components, or precipitated as pure or mixed solids. Soluble contaminants are subject to migration with soil water, uptake by plants, or loss due to volatilization into the atmosphere. Metals and radionuclides in soil may be associated with various phases that are reactive, semireactive, or nonreactive. Soil amendments used with in situ remediation techniques decrease the mobility of metals and radionuclides by increasing retention in the nonmobile solid phase.

Although numerous methods exist or are being developed for the in situ remediation of contaminated soils, their application can be problematic. The primary mechanism for contaminant migration in the vadose zone under arid site conditions typical of Hanford is the infiltration of precipitation. Consequently, properly designed and constructed surface covers offer the greatest control of water infiltration and ultimate protection of human health and the environment. A variety of in situ soil remediation technologies were considered and subsequently rejected because of difficulties and uncertainties associated with the placement of treatment zones and their performance verification.

The majority (greater than 90 percent) of the radioactivity in Hanford tanks can be attributed to strontium-90, cesium-137, and their daughter compounds. These radioactive contaminants are relatively insoluble, tend to be trapped in the soil matrix, are not easily transported by groundwater movement, and tend to remain in localized areas. Although small in terms of total inventory, mobile, long-lived

radionuclides present the greatest concerns from a groundwater risk perspective. The primary mobile, long-lived COPCs are technetium-99, iodine-129, neptunium-237, and uranium-238.

Factors that contribute to the difficulty in performing in situ remediation of contaminated soils include location; identification and quantification of the contaminants listed above; site lithology, geology, and hydrology; and the differing properties of the contaminants within the soil structure and matrix. The remediation technology for each selected COPC can be highly specialized, emphasizing the need for selectivity. Generally speaking, no single technology can remediate an entire site because multiple contaminants, both organic and inorganic, as well as soil and groundwater, can be involved. Several treatment technologies must be typically combined at a single site to form an effective treatment train (DOE 2003d).

Three primary strategies are typically used separately or in combination to achieve remediation of most sites. These strategies are as follows:

- Destruction or alteration of contaminants
- Extraction or separation of contaminants from environmental media
- Immobilization of contaminants

Available in situ soil remediation technologies include various chemical, physical, and biological treatment methods including, but not limited to, the following:

- In situ vapor extraction
- In situ solidification/stabilization
- In situ bioremediation
- In situ bioventing and biosparging
- In situ soil washing/flushing
- In situ vitrification
- In situ electrochemical remediation
- In situ REDOX control
- In situ enhanced soil mixing

While these treatment technologies can be effectively used for in situ remediation of contaminated soils, only solidification/stabilization, soil washing/flushing, and vitrification are commonly applied to the in situ treatment of radioactive materials and heavy metals (the primary contaminants at Hanford). In situ REDOX control has been demonstrated for some heavy-metal remediation and has potential application to soils contaminated with radioactive materials. Other in situ remediation technologies are better suited to work with halogenated volatiles and semivolatiles, nonhalogenated volatiles and semivolatiles, polychlorinated biphenyls (PCBs), pesticides, and cyanides.

The potential applicability of land disposal restrictions to waste left in the tanks after closure and to the contaminated soil that remains in place is an important area of regulatory uncertainty. The “Land Disposal Restrictions” (40 CFR 268) impose strict requirements on waste management operations and environmental restoration activities at DOE sites. Under the land-disposal-restriction regulatory framework, the disposal of RCRA hazardous waste in or on land is prohibited unless the waste has been treated to meet applicable treatment standards or it has been demonstrated to a reasonable degree of certainty that there would be no migration of hazardous constituents from the disposal unit or injection zone for as long as the waste remains hazardous (i.e., a “no-migration” petition has been submitted [40 CFR 268.6]) (DOE 2003d).



For mixed waste, the hazardous components are subject to regulation under RCRA, while the radioactive components are regulated by the Atomic Energy Act of 1954. Because RCRA requirements apply to the hazardous portion of the waste, mixed waste may be subject to the land disposal restrictions. Uncertainties as to the ability to verify in situ soil remediation technology placement and subsequent performance make it difficult to demonstrate compliance with the no-migration requirements.

If land disposal restrictions are determined to be applicable to tank waste residuals and contaminated soils, and the waste cannot be treated to meet a total concentration level or the treatment technology is not appropriate for the waste, DOE could petition for a variance from the treatment standards. A treatability variance does not remove the requirement to treat waste residuals and contaminated soil. Rather, alternative treatment standards based on data from actual treatment of soils and waste residuals become the treatment standard that must be met (DOE 2003d).

### **E.1.3.5.3 Gravel Filling of Tanks**

After the residual waste has been stabilized, filling the void spaces in the tanks with gravel could be used as an alternative to grouting to stabilize the tanks structurally. This process would use a gravel slinger to uniformly distribute sized, crushed rock throughout the tank, including the tank dome. This commercially proven technology is used to fill ship holds and silos with materials such as grain or cement. Tests performed at Hanford have verified the use of this technology with local materials in a tank-like environment.

Gravel would be distributed in the tank with a slinger, a mechanism suspended in the tank from the center riser. The slinger would capture gravel on a fast-moving horizontal belt, then throw it as it is slowly rotated. The belt speeds, belt angle, gravel feed rate, and rotational speed would be the primary controlling parameters. A hopper, mounted directly above the slinger, would be fed from the conveyor system and, in turn, would feed the slinger through a quick-acting isolation valve.

Tank conditions could require more than one slinger in a tank. The availability of existing risers versus the difficulty of installing new risers also could drive the decision to use more than one slinger. These somewhat smaller slingers could operate like the larger, center-mounted unit. For analysis purposes, it is assumed that all tanks would use a larger, center-mounted unit. Sacrificial material vibrators could be strategically placed within the tanks to ensure maximum fill for critical areas.

While it is clear that the gravel-filling option would adequately stabilize the tank structurally, the gravel would not prevent water intrusion and possible mobilization of contaminants from stabilized residual waste. In addition, the option of placing grout in the tanks to prevent structural subsidence represents the most conservative estimate for commitment of resources and was used to analyze the alternatives in this *TC & WM EIS* (DOE 2003d).

## **E.2 FAST FLUX TEST FACILITY DECOMMISSIONING**

### **E.2.1 Fast Flux Test Facility Background**

FFTF is a DOE-owned, formerly operating, 400-megawatt (thermal) liquid-metal (sodium)-cooled research and test reactor located in the 400 Area of Hanford near the city of Richland, Washington (see Figures E-40 and E-41). A detailed description of the FFTF complex is provided in the *Technical Information Document for the Fast Flux Test Facility Closure Project Environmental Impact Statement* (Fluor Hanford 2005c).



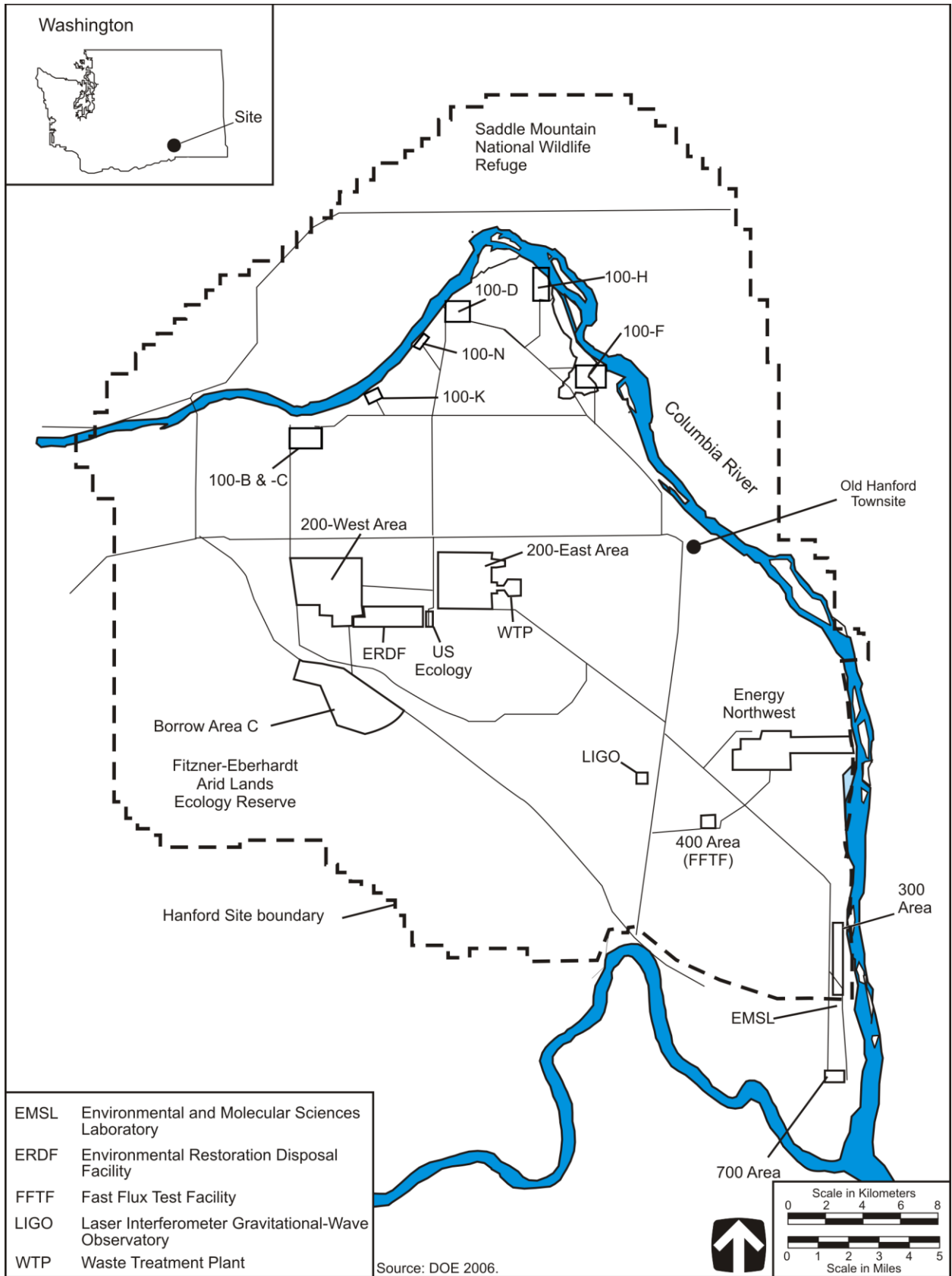
**Figure E-40. Fast Flux Test Facility**

The purpose of the facility was to develop and test advanced fuels and materials for the Liquid Metal Fast Breeder Reactor Program (FFTF is a liquid-metal reactor) and to serve as a prototype facility for future Liquid Metal Fast Breeder Reactor Program facilities; other missions were subsequently pursued. Construction of FFTF was completed in 1978. Initial criticality was achieved on February 9, 1980, and full power was initially achieved on December 21, 1980. Following an additional year of extensive acceptance testing, FFTF operated safely and successfully from 1982 to 1992 and provided the nuclear industry with significant advances in fuel performance, medical isotope production, materials performance, and passive and active safety system testing. In December 1993, DOE decided to discontinue FFTF operations because of a lack of economically viable missions at that time and issued a shutdown order for FFTF.

In May 1995, DOE prepared the *Environmental Assessment, Shutdown of the Fast Flux Test Facility, Hanford Site, Richland, Washington* (DOE 1995), evaluating the potential impacts associated with actions necessary to place FFTF in a radiologically and industrially safe permanent shutdown and deactivation condition (Phase I), suitable for long-term S&M (Phase II) prior to decommissioning (Phase III). The 1995 environmental assessment (EA) did not evaluate Phase III. DOE determined that an EIS was not required for the permanent shutdown and deactivation of FFTF and issued a Finding of No Significant Impact (FONSI) in May 1995.

In January 1997, DOE decided to maintain FFTF in standby pending an evaluation of a future role in DOE's national tritium production strategy. In December 1998, DOE decided FFTF should not play a role in production of the Nation's tritium stockpile. Facility deactivation work continued under the 1995 EA, but the work was limited to activities that would not preclude reactor restart.

*Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington*



**Figure E-41. Hanford Site**

In December 2000, DOE published the *Final Programmatic Environmental Impact Statement for Accomplishing Expanded Civilian Nuclear Energy Research and Development and Isotope Production Missions in the United States, Including the Role of the Fast Flux Test Facility (NI PEIS)* (DOE 2000). The *NI PEIS* evaluated the role of FFTF as an alternative nuclear irradiation services facility to accomplish civilian nuclear energy R&D, medical and industrial radioisotope production, and production of plutonium-238 to support future National Aeronautics and Space Administration space exploration missions. Also evaluated was an alternative to permanently deactivate FFTF. Based on the *NI PEIS*, DOE decided, as reflected in the *NI PEIS* ROD (66 FR 7877) issued on January 26, 2001, that permanent deactivation of FFTF was to be resumed, and no new missions proposed. Since that time, deactivation has continued, consistent with the 1995 EA and FONSI, the 2000 *NI PEIS*, and the 2001 *NI PEIS* ROD.

In December 2003, DOE issued a final request for proposals to “clean up and take down” the FFTF complex. On December 22, 2005, DOE canceled the solicitation for the Hanford Site FFTF Closure Project. Cancellation of the solicitation was deemed necessary because of budget constraints and the need to support higher-risk/higher-priority Hanford cleanup projects. In February 2006, DOE announced its intention to prepare this *TC & WM EIS* (71 FR 5655). DOE decided to merge the scope of the proposed but canceled “Environmental Impact Statement for the Decommissioning of the Fast Flux Test Facility at the Hanford Site, Richland, Washington,” described in DOE’s Notice of Intent (69 FR 50176), to further coordinate resources and ensure a comprehensive look at environmental impacts at Hanford. In this *TC & WM EIS*, the potential decision for final D&D of FFTF would identify the final end state for the aboveground, belowground, and ancillary support structures.

In March 2006, DOE published the *Environmental Assessment, Sodium Residuals Reaction/Removal and Other Deactivation Work Activities, Fast Flux Test Facility (FFTF) Project, Hanford Site, Richland, Washington* (DOE 2006). This EA was an interim action that examined the environmental consequences of an expanded deactivation work scope involving a different approach to sodium residuals management than was previously analyzed in the 1995 EA. The 1995 EA provided the foundation for most of the analyses included in the 2006 EA, which addressed potential additional environmental impacts. There had been minor changes in environmental conditions at the 400 Area of Hanford since 1995. The affected environment is described in Section 3.0 of the 2006 EA, which updates the description provided in the 1995 EA (as documented in current 2005 reviews of Hanford environmental conditions). As such, the 2006 EA supplements the 1995 EA with regard to deactivation actions. Under the criteria of the Council on Environmental Quality’s Regulations for Implementing the Procedural Provisions of the National Environmental Policy Act (40 CFR 1500–1508), these actions are not expected to have adverse environmental impacts or limit the choice of reasonable alternatives under consideration in this *TC & WM EIS* (DOE 2006).

In the 2006 EA, DOE proposed a different approach to the ongoing deactivation work at FFTF, one that was not extensively discussed and analyzed in the 1995 EA. DOE proposed to remove radioactively contaminated sodium residuals left over from the draining of Hanford’s radioactively contaminated sodium inventory (i.e., in FFTF, the Hallam Reactor, and the Sodium Reactor Experiment [SRE]) by reacting the sodium metal with water (as superheated steam) to produce caustic sodium hydroxide solution; to remove associated equipment/components to allow removal of the sodium; and to remove, dispose of, and stabilize miscellaneous hazards and waste streams left over after the draining of sodium. These activities would further support low-cost, environmentally safe S&M activities at FFTF.

Some of the specific issues discussed and evaluated in the 2006 EA include the following (DOE 2006):

- The use of the superheated steam process in place or at designated cleaning locations to remove sodium residuals. (Superheated steam occurs when steam is heated well above the boiling point of water before being injected into the preheated equipment/components [e.g., piping, valves, tanks, etc.] at controlled rates.)

- The locations where the reaction of sodium or sodium residuals associated with the sodium systems and equipment can be performed (i.e., in place or at designated cleaning locations).
- The use of alternative technology/technologies in select situations for small-scale reaction of sodium residuals.

Other deactivation work activities discussed and evaluated in the 2006 EA include removal of associated equipment/components to facilitate removal of the sodium residuals and removal, disposition, and stabilization of the miscellaneous hazards and waste streams resulting from the draining of sodium. These activities include the following (DOE 2006):

- Clean in-place vessels, components, and large-bore pipe (equal to or greater than 20.3-centimeter-diameter [8-inch-diameter]) in primary and secondary sodium cooling systems.
- Remove small-bore pipe (less than 20.3-centimeter-diameter [8-inch-diameter]), valves, and other components for reaction in a cleaning station.
- Remove large components for cleaning.
- Remove FFTF RH-SCs (the primary cold trap, cesium trap, and two vapor traps) and package them for storage in the 400 Area pending final disposition.
- Remove/dispose of asbestos.
- Remove/stabilize existing hazards in conjunction with deactivating systems and equipment associated with sodium residuals.
- Remove/recycle/dispose of excess deactivated equipment and components as necessary.

On March 31, 2006, DOE issued a FONSI (DOE 2006:Appendix B) containing the following conclusion:

“Based on the analysis in the EA, and considering preapproval comments received (Appendix A of DOE/EA-1547F [the 2006 EA]), DOE has determined that the proposed action is not a major federal action significantly affecting the quality of the human environment within the meaning of the “*National Environmental Policy Act of 1969 (NEPA)*, 42 U.S.C. 4321, et seq.” Therefore, the preparation of an Environmental Impact Statement (EIS) is not required.”

In addition, the FONSI stated:

“The DOE/EA-1547F [the 2006 EA] does not address FFTF decommissioning activities, i.e., final end state of the FFTF. That scope of work will be addressed in the Tank Closure and Waste Management Environmental Impact Statement” (DOE 2006:Appendix B).

Since June 2007, the major deactivation activities that have been completed at FFTF include shipment of fuel off site and deactivation of auxiliary plant systems. Approximately 916,000 liters (242,000 gallons) of the 958,000 liters (253,000 gallons) of radioactively contaminated bulk sodium have been drained from the FFTF reactor vessel, three primary and three secondary heat transport system loops, the Fuel Storage Facility, and the interim decay storage vessel and associated auxiliary systems and transferred to the Sodium Storage Facility (SSF), which is adjacent to FFTF. Associated trace heat systems have been de-energized (Chapin 2007). See Chapter 3, Section 3.2.13, for information on the status of FFTF SNF.

In April 2009, FFTF auxiliary plant systems deactivation activities were completed, except for sodium residuals reaction/removal deactivation activities. In June 2009, the facility transitioned into the surveillance and maintenance phase (McCormick 2009).

## E.2.2 Fast Flux Test Facility Description

The existing buildings and structures within the FFTF complex, including the Reactor Containment Building (RCB), reactor support buildings, and auxiliary buildings, are identified in Table E–15. The RCB is a large, round building measuring 41.1 meters (135 feet) in diameter with a domed roof. The building rises 56.7 meters (186 feet) above grade and extends 24 meters (78 feet) below grade. The RCB is constructed from carbon steel, reinforced concrete, and magnetite concrete (used for shielding). The numerous reactor and auxiliary buildings are primarily steel frame buildings with metal siding on reinforced-concrete foundations or pads. Other structures are also associated with FFTF, including storage tanks, electrical/power transformer substations, and cooling towers. The SSF and the Fuel Storage Facility are also a part of FFTF. Figure E–42 is a map of the FFTF complex and associated facilities.

This *TC & WM EIS* includes alternatives for final disposition of all these buildings and associated structures, as well as the bulk sodium material in the SSF. It does not, however, include disposition of the fuel stored in the Fuel Storage Facility. Previous NEPA documentation addresses the alternatives for handling and disposition of the fuel removed from FFTF (Fluor Hanford 2005c).

**Table E–15. Fast Flux Test Facility Complex Facilities and Structures**

Building Number	Building Name	Equipment	Construction Materials
405	Reactor Containment Building	Reactor vessel, maintenance and storage cells, sodium-cooling system, inert cover gas system	Carbon steel, reinforced concrete, magnetite concrete (shield)
<b>Reactor Support Buildings</b>			
4703	FFTF Control Building	Reactor control room, telephone and computer equipment, offices, etc.	Reinforced concrete, steel frame, metal-and-concrete roof and siding
4621E	Auxiliary Equipment Building, East	Plant control and auxiliary equipment, radiological monitoring, access control, diesel generator	Steel frame, metal siding above, reinforced concrete below
4621W	Auxiliary Equipment Building, West	Electrical switchgear and equipment, diesel generator	Steel frame, metal siding above, reinforced concrete below
491E	HTS Service Building, East	Secondary sodium piping, pumps, switchgear controls, and equipment	Steel frame, metal siding above, reinforced concrete below
491W	HTS Service Building, West	Secondary sodium piping, pumps, switchgear controls, and equipment	Steel frame, metal siding above, reinforced concrete below
491S	HTS Service Building, South	Primary sodium-sampling equipment, small hot cell, cesium trap, cover gas sampling, decontamination, inert gas cooling	Steel frame, metal siding above, reinforced concrete below

**Table E-15. Fast Flux Test Facilities and Structures (continued)**

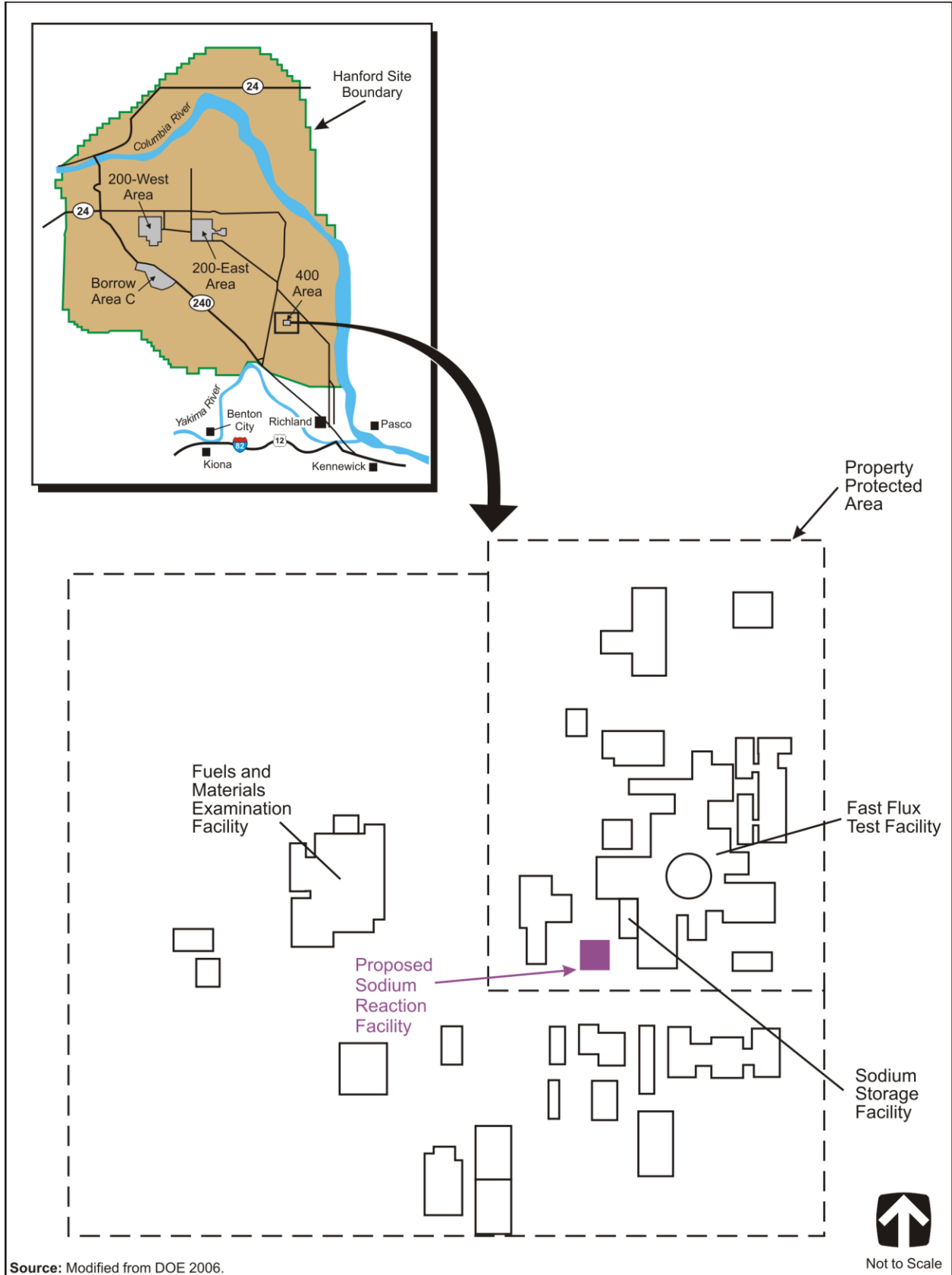
Building Number	Building Name	Equipment	Construction Materials
<b>Reactor Support Buildings (continued)</b>			
408A	HTS DHX, East	Heat exchanger modules, fans, motors, secondary sodium piping	Reinforced concrete, steel frame, metal siding (tornado-hardened)
408B	HTS DHX, South	Heat exchanger modules, fans, motors, secondary sodium piping	Reinforced concrete, steel frame, metal siding
408C	HTS DHX, West	Heat exchanger modules, fans, motors, secondary sodium piping	Reinforced concrete, steel frame, metal siding
409A	Closed Loop Heat Dump, East 1	Sodium-to-air forced-convection heat exchangers (never operational)	Steel frame, metal siding
409B	Closed Loop Heat Dump, East 2	Sodium-to-air forced-convection heat exchangers (never operational)	Steel frame, metal siding
4717	Reactor Service Building	Cask loading, radioactive gas and waste management	Steel frame, metal siding above, reinforced concrete below
403	Fuel Storage Facility	Spent nuclear fuel storage vessel and associated sodium potassium heat exchangers	Steel frame, metal siding above, reinforced concrete below
402	Sodium Storage Facility	Storage vessels for secondary and primary sodium coolant	Reinforced concrete
484	FFTF In-Containment Chiller Water Equipment Building	Chiller units, pumps, tanks, piping	Steel frame, metal siding, on reinforced-concrete pad
483	Cooling Towers Chemical Addition Building	Eight modular, forced-draft, evaporative, closed-loop cooling towers and adjacent chemical addition building	Modular cooling towers on reinforced-concrete pad
4716	FFTF Rigging Loft	Maintenance equipment and tool storage	Steel frame, metal siding
4721	FFTF Emergency Turbine Generator Building	Oil-fired turbine generator and switchgear, fuel oil storage tank	Reinforced concrete (tornado-hardened)
451A	Substation	Electrical substation and associated equipment	Concrete pads
<b>Auxiliary Buildings</b>			
432A	Interim Covered Equipment Storage	Storage of fuel cask handling equipment	Steel frame, metal siding
4718	400 Area Interim Storage Area	Spent nuclear fuel cask storage area	Concrete pad with perimeter fence and lighting
436	Training Facility	Classrooms, reactor control room simulator	Wood frame, composite siding
437	Maintenance & Storage Facility	Multipurpose maintenance and storage, equipment testing, training, equipment decontamination, sodium cleaning, radioactive liquid storage	Steel frame, metal siding
440	90-day Storage Pad	Hazardous waste storage	Steel frame, metal siding, concrete foundation

**Table E–15. Fast Flux Test Facilities and Structures (continued)**

<b>Building Number</b>	<b>Building Name</b>	<b>Equipment</b>	<b>Construction Materials</b>
<b>Auxiliary Buildings (continued)</b>			
453A	Transformer Station, East DHX A1 2.4kV	Transformer support pad	Concrete
453B	Transformer Station, South DHX A2 2.4kV	Transformer support pad	Concrete
453C	Transformer Station, West DHX A3 2.4kV	Transformer support pad	Concrete
480A	Water Supply Well House (P-14)	Well pumps and chlorination equipment	Wood frame, aluminum siding
480B	Water Supply Well House (P-15)	Well pumps and chlorination equipment	Wood frame, aluminum siding
480D	Water Supply Well House (P-16)	Well pumps and chlorination equipment	Concrete masonry
482A	Water Storage Tank (T-58)	Water supply storage	Tank on reinforced-concrete foundation
482B	Water Storage Tank (T-87)	Water supply storage	Tank on reinforced-concrete foundation
482C	Water Storage Tank (T-330)	Water supply storage	Tank on reinforced-concrete foundation
481	Water Pump House	Distribution pumps for 400 Area water supply	Concrete masonry
481A	Water Pump House	Distribution pumps for 400 Area water supply	Concrete masonry
4842B	Switchgear Building for Pump Houses	Switchgear for well pumps and associated equipment	Wood frame, metal siding
4701	Kentucky Boulevard Guard Building	Security, access control to 400 Area Property Protected Area	Wood frame, composite siding, concrete masonry
4710	FFTF Office Building	Offices for operations and support staff	Wood frame, composite siding
4713A	Riggers and Drivers Operations Facility	Offices for maintenance staff	Steel frame, metal siding
4713B	FFTF Maintenance Shop	Maintenance operations	Shop: steel frame, metal siding Annex: wood frame, composite siding
4713C	Contaminated Storage Warehouse	Equipment storage	Steel frame, metal siding
4713D	Interim Maintenance and Storage Facility	Maintenance operations, equipment storage	Steel frame, metal siding
4734A	FFTF Argon/Nitrogen Dewar Pad	Mounting and foundation for gas storage tanks	Concrete

**Key:** DHX=Dump Heat Exchanger; FFTF= Fast Flux Test Facility; HTS=Heat Transport System; kV=kilovolt.





Source: Modified from DOE 2006.

Figure E-42. Fast Flux Test Facility and Associated Facilities Location

### E.2.3 Summary Description of FFTF Decommissioning Alternatives

DOE needs to decommission FFTF and its support facilities at Hanford; to manage waste associated with decommissioning the facilities, including certain FFTF wastes designated as RH-SCs; and to manage the disposition of the radioactively contaminated bulk sodium inventory at Hanford. Those actions are necessary to protect human health and the environment; facilitate cleanup at Hanford; comply with decisions reached by DOE as a result of previous NEPA reviews (DOE 1995, 2000; 66 FR 7877), as well as Federal and Washington State laws and regulations; and meet enforceable milestones established in the TPA (Ecology, EPA, and DOE 1989, 2002, 2003).

To address anticipated needs for decommissioning FFTF, DOE proposes to complete the following actions:

- Decommission FFTF and associated support facilities within the Property Protected Area (PPA) at the Hanford 400 Area, and manage radioactive and hazardous wastes using existing capabilities.
- Manage certain FFTF wastes designated as RH-SCs using management capabilities that do not currently exist at Hanford or elsewhere.
- Dispose of the inventory of radioactively contaminated bulk sodium resulting from deactivation of FFTF, as well as sodium from the Hallam Reactor and the SRE that is now in storage at the Hanford 200-West Area.

Aspects of the alternatives for accomplishing these proposed actions are as follows. A No Action Alternative is also evaluated, as required by NEPA.

- **Demolition of the facilities within the 400 Area of Hanford.** Demolition of all or part of the facilities in the 400 Area would be required under each of the alternatives evaluated in this *TC & WM EIS*, except for the No Action Alternative. Demolition would result in radioactive and chemically hazardous waste requiring disposal. Disposal of the bulk of this demolition waste would occur on site in disposal facilities approved for Hanford's operational waste (e.g., an onsite IDF). Waste volumes would vary among the alternatives. This *TC & WM EIS* provides the environmental impact information needed for DOE to make informed decisions regarding preferred alternatives based in part on waste volumes generated, worker safety and other environmental risks/impacts, appropriate D&D technologies, and disposal requirements (e.g., appropriate modified RCRA Subtitle C, D, or other barrier designs).
- **Management and disposition of the FFTF RH-SCs.** Currently, no facility exists without modification within the DOE complex for handling or treating the RH-SCs. In February 2009, under the 2009 FONSI (DOE 2009a), DOE selected the Preferred Alternative of decontamination, as necessary, and modifications to one of two existing facilities within the Idaho Nuclear Technology and Engineering Center (INTEC) at INL. The analyses in this *TC & WM EIS* evaluate two options for processing the RH-SCs in addition to the No Action Alternative, as follows:
  - Idaho Option: Removal and shipment of the RH-SCs to INL for treatment in either the New Waste Calcining Facility (NWCF) or the Fluorinel Dissolution Process (FDP) within the Fluorinel Dissolution Process and Fuel Storage Facility (FAST Facility), both of which are at INTEC (see Section E.2.4.4), followed by shipment to and disposal at either NNSS or Hanford.

- Hanford Option: Removal and storage of the RH-SCs on site at Hanford until a new facility, the Remote Treatment Project (RTP), is permitted and built, followed by disposal at Hanford.

A description of both of these facilities can be found in Section E.2.4.4. This EIS provides the environmental impact information on both options for disposition of RH-SCs to allow DOE to make informed, programmatic decisions on treatment capability, construction, location, and operation with respect to the FFTF RH-SCs, as well as other materials requiring remote handling and processing.

- **Management and disposition of radioactively contaminated bulk sodium.** FFTF reactor coolant systems and storage vessels contained about 984,200 liters (260,000 gallons) of radioactively contaminated sodium. Management and disposition of this sodium, along with about 128,700 liters (34,000 gallons) and about 26,500 liters (7,000 gallons) of radioactive sodium from the Hallam Reactor and SRE, respectively, are addressed in this *TC & WM EIS*. Additionally, radioactively contaminated piping and other general demolition wastes would result from decommissioning FFTF. Processing the radioactively contaminated bulk sodium coolant from FFTF, as well as sodium from the Hallam Reactor and SRE, is required under the scope of the FFTF decommissioning actions. This *TC & WM EIS* analyzes the following two options, in addition to the No Action Alternative, for processing this bulk sodium:

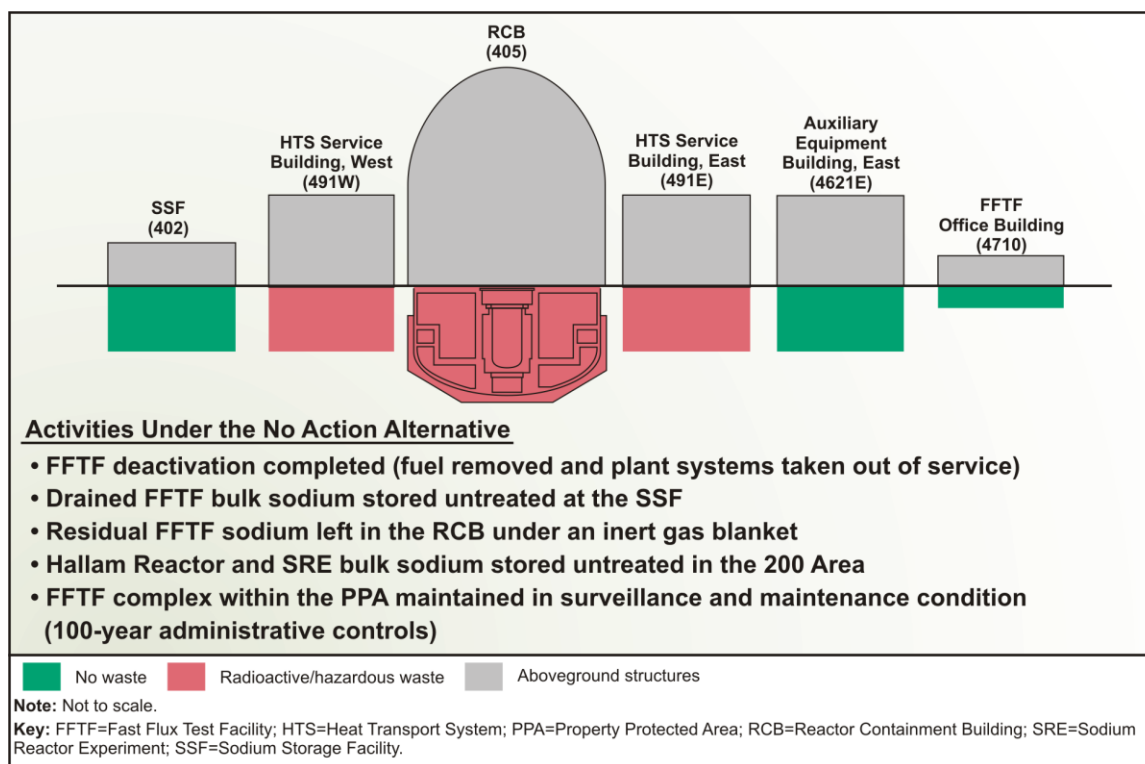
- Hanford Reuse Option: Store the bulk sodium on site at Hanford until a new Sodium Reaction Facility (SRF) is built and the sodium can be processed into a caustic sodium hydroxide solution for product reuse by ORP for the WTP or Hanford tanks corrosion control. This alternative requires permitting and construction of a new facility to convert the reactive radioactive sodium to a caustic sodium hydroxide solution on site at Hanford.
- Idaho Reuse Option: Prepare and ship the sodium to INL for processing into a caustic sodium hydroxide solution and shipment back to Hanford for product reuse by ORP for the WTP or Hanford tanks corrosion control. This alternative requires that the Sodium Processing Facility (SPF) at INL's Materials and Fuels Complex (MFC) be restarted and available for use when required.

In May 2007, the *Hanford Site Sodium Disposition Evaluation Report* (Burke 2007) was issued to document current planning by the TPA agencies for the management and disposition of the radioactively contaminated bulk sodium currently stored at Hanford. Based on planning at the time, the document concluded that conversion of the sodium to caustic sodium hydroxide solution would not utilize the SPF located at INL and anticipated construction of a new conversion facility adjacent to the SSF located in the 400 Area at Hanford. However, as noted in the document, that conclusion does not preclude or predetermine sodium management decisions that may be reached in the DOE NEPA *TC & WM EIS* ROD. Thus, DOE has decided to retain for analysis purposes the option of processing the Hanford radioactively contaminated bulk sodium at the INL SPF.

Descriptions of the SRF and SPF can be found in Sections E.2.4.2 and E.2.4.3, respectively. This EIS provides the environmental impact information for DOE's programmatic decisions on whether to transport the sodium to the SPF at INL's MFC or to permit and construct a new facility, the SRF, at Hanford for processing the bulk sodium.

### E.2.3.1 FFTF Decommissioning Alternative 1: No Action

Council on Environmental Quality regulations require that NEPA analyses include a No Action Alternative. Under this alternative, deactivation of the FFTF complex and support buildings would be completed as specified by previous FFTF NEPA decisions (DOE 2006:Appendix B), so that FFTF can be maintained in a long-term S&M condition for the foreseeable future. The facility would be monitored and periodic S&M performed to ensure that environmental or safety issues are minimized and addressed. Figure E-43 is a graphic representation of this FFTF Decommissioning No Action Alternative.



**Figure E-43. FFTF Decommissioning Alternative 1: No Action**

The impacts of continuing to implement previous decisions would be compared with the potential impacts of the proposed actions under the action alternatives. The following sections provide additional descriptions of activities to be conducted under the No Action Alternative:

**Facility Disposition.** FFTF (the RCB, Building 405), along with the rest of the buildings within the 400 Area PPA, would be maintained in a long-term S&M condition after completion of all deactivation activities. The buildings would be left standing, with a maintained exterior capable of protecting them from the elements. They would be unoccupied, with essential safety-related systems left operational. Such systems could include, but would not be limited to, fire protection, emergency lighting, ventilation, air monitoring, and inert gas systems used to isolate piping and equipment containing sodium residuals.

**Process Components.** The reactor vessel, piping systems, and tanks (contained above and below grade within the RCB and the immediately adjacent buildings) would be left in place under an inert gas (e.g., argon) blanket. Deactivation activities would already be completed, including removal of SNF and lead shielding; draining of the bulk sodium; and removal of small-bore piping, valves, and other components. Some systems (e.g., those not associated with maintaining safety-related functions) would be deactivated/de-energized and isolated per the deactivation plans.

**Sodium Residuals.** Sodium residuals in the RCB vessels and cooling system piping would be left in place, untreated, but under an inert gas (e.g., argon) blanket. During deactivation activities, the FFTF bulk sodium would be drained from the reactor systems and stored as a solid in tanks in the SSF within the 400 Area. The small amount of sodium-potassium alloy would be blended with the content of the bulk sodium storage containers. The sodium from the Hallam Reactor and SRE would remain in its current storage location (Hanford's 200-West Area).

**Demolition and Other Waste.** There would be no demolition under the No Action Alternative; hence, no demolition waste would be generated. Solid and liquid radioactive and/or hazardous waste generated during deactivation would be managed and disposed of on site. Activities associated with the No Action Alternative would not generate substantial additional quantities of solid waste for disposal. The small amounts of radioactive solid waste generated during S&M activities would be disposed of on site in disposal facilities approved for Hanford's operational waste in the existing LLBG trenches. Other regulated waste, such as PCBs, asbestos, and hazardous waste, would be handled in a similar manner under all of the alternatives. The volume of that waste is expected to be small, and it would be disposed of in accordance with existing Hanford facility acceptance criteria or offsite treatment contracts.

**End State.** The facilities and infrastructure within the 400 Area PPA, including the RCB, would be maintained in a long-term S&M condition using appropriate monitoring and controls (to ensure that environmental or safety concerns are minimized) for the foreseeable future.

### **E.2.3.2 FFTF Decommissioning Alternative 2: Entombment**

Under this alternative, the FFTF RCB (and structures within) above grade level (i.e., 168 meters [550 feet] above mean sea level) would be decontaminated as necessary, dismantled, and removed. The RCB structures below grade level, as well as the FFTF reactor vessel and radioactive and contaminated equipment, components, piping, and other materials that have become radioactive or otherwise contaminated, would remain in place. Sodium residuals would be either removed from the RCB and treated in existing 400 Area facilities or treated in place. In addition, the RCB below grade level would be filled with grout or other suitable fill material to immobilize remaining hazardous chemicals and radioactive materials to the maximum extent practicable and to prevent subsidence. The RCB fill material may include other demolition debris containing hazardous or radioactive materials, as allowed by regulations. A modified RCRA Subtitle C barrier would be constructed over the filled area. The barrier, together with the lower RCB and adjacent structures and the immobilized internal structures, would be within the entombed area. A preliminary, conceptual view of the Entombment Alternative is presented in Figure E-44.

The FFTF support buildings would be decontaminated as needed and demolished, as noted in Table E-16. The area previously occupied by the facilities would then be backfilled with soil to eliminate void spaces and then compacted, contoured, and revegetated. An appropriate monitoring program for the PPA would also be established. The following sections provide additional descriptions of activities to be conducted under the Entombment Alternative.

**Facility Disposition.** Table E-16 summarizes the proposed decommissioning activities for each building under both the Entombment and the Removal Alternatives. For the Entombment Alternative, the main RCB and the two immediately adjacent support facilities (Buildings 491E and 491W) would have all above-grade structures (e.g., the RCB dome) dismantled, and the demolition waste would be disposed of in an IDF or consolidated in the below-grade spaces. Below-grade structures would be filled with demolition waste as practicable and stabilized with suitable fill material (e.g., grout) to immobilize hazardous chemical and radioactive materials and prevent subsidence in the future.

All other ancillary buildings, including their internal equipment and components, would be demolished, as noted in Table E-16, and the contaminated demolition debris would be disposed of in an IDF or consolidated within available below-grade spaces within the RCB or Buildings 491E and 491W. All radioactive and/or hazardous material would be removed. Wood and large steel components would also be removed. Foundation rubble (e.g., concrete and rebar) would remain. The area previously occupied by these facilities would be backfilled, compacted, contoured, and revegetated. As indicated in Table E-16, some of these buildings would be either completely or partially within the footprint (including side slope) of the engineered barrier over the RCB.

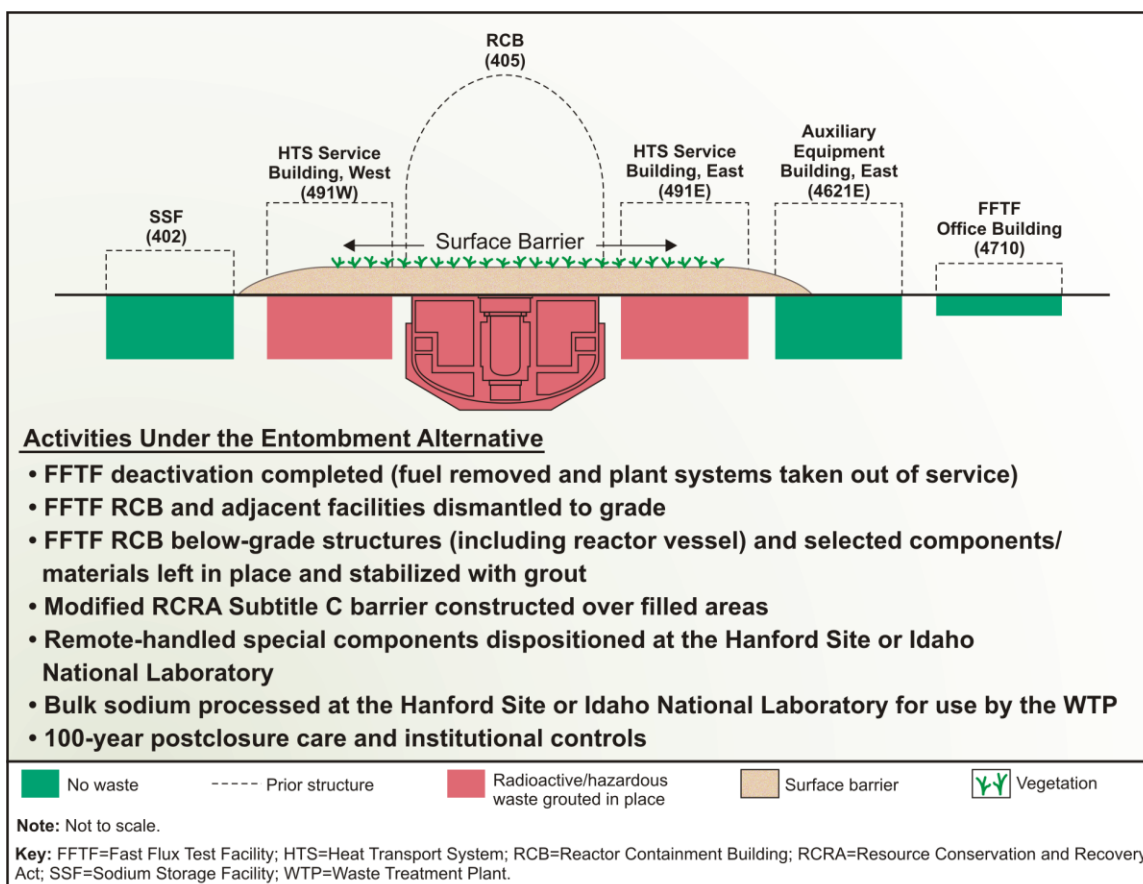


Figure E-44. FFTF Decommissioning Alternative 2: Entombment

Table E-16. Proposed Decommissioning Actions for Hanford Site Fast Flux Test Facility Complex Facilities and Structures

Building Number	Building Name	Action Alternative	
		Alternative 2: Entombment	Alternative 3: Removal
405	FFTF Reactor Containment Building	F	E
491E	HTS Service Building, East	F	C
491W	HTS Service Building, West	F	C
4621E	Auxiliary Equipment Building, East	D	C
4621W	Auxiliary Equipment Building, West	D	C
4703	FFTF Control Building	D	C
4717	Reactor Service Building	D	C

**Table E–16. Proposed Decommissioning Actions for Hanford Site  
Fast Flux Test Facility Complex Facilities and Structures (continued)**

Building Number	Building Name	Action Alternative	
		Alternative 2: Entombment	Alternative 3: Removal
491S	HTS Service Building, South	D	C
408A	Main Heat Dump, East	B	A
408B	Main Heat Dump, South	B	A
408C	Main Heat Dump, West	B	A
409A	Closed Loop Heat Dump, East 1	B	A
409B	Closed Loop Heat Dump, East 2	B	A
403	Fuel Storage Facility	C	C
402	Sodium Storage Facility	A	A
432A	ISA Covered Equipment Storage	A	A
436	Training Facility	A	A
437	Maintenance and Storage Facility	A	A
440	90-day Covered Storage Pad	A	A
451A	Substation	A	A
453A	Transformer Station, East DHX A1 2.4kV	A	A
453B	Transformer Station, South DHX A2 2.4kV	A	A
453C	Transformer Station, West DHX A3 2.4kV	A	A
4701	Former FFTF Guard Station	A	A
4710	FFTF Office Building	A	A
4713A	Riggers and Drivers Operations Facility	A	A
4713B	FFTF Maintenance Shop	A	A
4713C	Contaminated Storage Warehouse	A	A
4713D	Interim Maintenance and Storage Facility	A	A
4716	FFTF Rigging Loft	A	A
4718	400-Area Interim Storage Area Pad	A	A
4721	FFTF Emergency Generator Building	A	A
4734A	FFTF Argon/Nitrogen Pad	A	A
480A	Water Supply Well House (P-14)	A	A
480B	Water Supply Well House (P-15)	A	A
480D	Water Supply Well House (P-16)	A	A
481	Water Pump House	A	A
481A	Water Pump House	A	A
482A	Water Storage Tank (T-58)	A	A
482B	Water Storage Tank (T-87)	A	A
482C	Water Storage Tank (T-330)	A	A
483	Cooling Towers Chemical Addition Building	A	A
484	FFTF In-Containment Chiller Water Equipment Building	A	A
4842B	Switchgear Building for Pump Houses	A	A
SRFa	Sodium Reaction Facility (Proposed)	A	A

**Table E–16. Proposed Decommissioning Actions for Hanford Site Fast Flux Test Facility Complex Facilities and Structures (continued)**

<sup>a</sup> If the U.S. Department of Energy decides to process the bulk sodium at an existing INL facility, the SRF would not be constructed. Decommissioning of the INL facility is not addressed in this *Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington*.

**Note:** Gray shading indicates buildings with reinforced-concrete basements.

A = Demolish and remove building and soils, down to 0.91 meters (3 feet) below grade (if present, subsurface floors and interior walls would be collapsed into the below-grade space; basement exterior walls below 0.91 meters [3 feet] and basement floor and foundations would remain). Backfill to grade with soil, then compact and contour surface and revegetate. Remove all radioactive and/or hazardous material, as well as wood and large steel components. Foundation rubble (e.g., concrete and rebar) would remain.

B = Same as A, except the building footprint would be partially covered by the engineered barrier system.

C = Demolish and remove building down to grade. Remove above- and below-grade components and systems, then collapse floors and walls into the below-grade space at least down to 0.91 meters (3 feet) below grade (basement exterior walls below 0.91 meters [3 feet] and basement floor and foundations would remain). Backfill to grade with soil, then compact and contour surface and revegetate. Remove all radioactive and/or hazardous material, as well as wood and large steel components. Foundation rubble (e.g., concrete and rebar) would remain.

D = Same as C, except the building footprint would be partially covered by the engineered barrier system.

E = Same as C, except small amounts of radioactive activation products in structural concrete and steel would remain.

F = Remove above-grade structures and systems. Contaminated equipment and systems below grade would remain. Consolidate waste and demolition debris below grade, then backfill with grout and cover entirely as part of the engineered barrier system. Radioactive and hazardous waste would remain entombed.

**Key:** DHX=Dump Heat Exchanger; FFTF=Fast Flux Test Facility; HTS=Heat Transport System; INL=Idaho National Laboratory; ISA=Interim Storage Area; kV=kilovolts; SRF=Sodium Reaction Facility.

**Process Components.** The reactor vessel, piping systems, and tanks (contained above and below grade within the RCB and the immediately adjacent buildings) would have all above-grade systems dismantled and placed in below-grade spaces as practicable or transported to an IDF for disposal. Systems that are below grade (including regulated waste) would be grouted in place after treatment of sodium residuals. The small-diameter piping (less than 20.3 centimeters [8 inches] in diameter) would be removed, treated (cleaned of sodium) in the 400 Area, and disposed of on site in an IDF or placed in below-grade spaces within the RCB.

**Sodium Residuals.** All sodium residuals would be removed from the RCB systems or treated in place. The analyses assumed that sodium would be drained from plant systems to the extent practicable, followed by passivation and/or flushing with water to stabilize sodium residuals. Sodium residuals in small-diameter piping would be treated in the 400 Area after removal of the components from the reactor plant.

**Demolition and Other Waste.** Demolition debris from facility decommissioning (chemically hazardous or radioactive solid waste) would be handled in the same way for both action alternatives, except that the disposition of the volumes of debris would change. The debris not placed in the RCB or other voids or used as backfill would be transported to an IDF for disposal. Analyses of solid waste resulting from any of the processing options (for sodium residuals, bulk sodium, etc.) were included in the analyses of those options, specific to the appropriate processing activities.

Radioactive liquid waste resulting from treatment of the sodium residuals would also be handled in the same way for both FFTF Decommissioning Alternatives 2 and 3. The liquid volume would be reduced at FFTF (either through ion exchange and reuse or evaporation), and the remaining liquids would be transported to the 200 Area ETF for processing and disposal. For analysis purposes, it was assumed that a 90 percent reduction in volume could be achieved prior to shipment of the liquid to the ETF for processing. Any other sources of radioactive waste (such as decontamination solutions) are expected to result in very small volumes compared with those produced as a result of treating sodium residuals.

Other regulated waste, such as PCBs, asbestos, and nonradioactive hazardous waste, would be handled in a similar manner under all of the FFTF Decommissioning alternatives. The volume of that waste is



expected to be small, and it would be disposed of in accordance with existing Hanford facility waste acceptance criteria or offsite treatment contracts.

**End State.** For the analyses in this *TC & WM EIS*, an engineered barrier that is compliant with regulations (see Figure E-28), such as a modified RCRA Subtitle C barrier, was assumed to be constructed over the remaining below-grade portions of the RCB and Buildings 491E and 491W, which would contain radioactive and/or hazardous waste. The barrier also would extend over part or all of the immediately adjacent facility footprints. The barrier would be circular with a radius of about 39.2 meters (128.5 feet), not including the side slope used for drainage. The side slope would be about 5.2 meters (17.1 feet), using a 3H:1V slope. Minimal site postclosure care and maintenance would be required. The remainder of the PPA would be backfilled with soil, compacted, contoured, and revegetated.

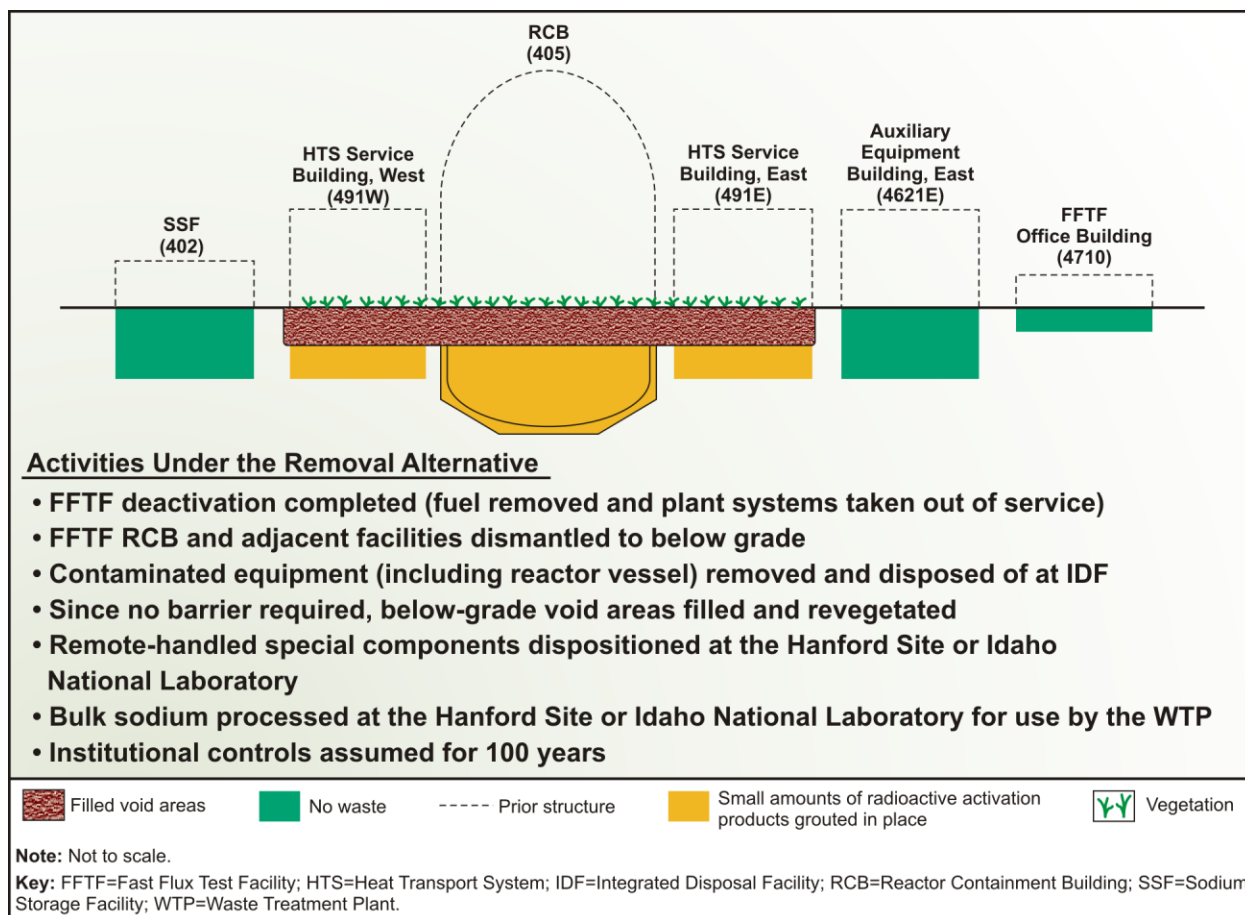
The modified RCRA Subtitle C barrier would be designed to provide long-term containment and hydrologic protection for a performance period of 500 years, assuming no maintenance is performed after a 100-year institutional control period. This performance period is conservatively based on radionuclide concentration and activity limits for Category 3 LLW. The modified RCRA Subtitle C barrier would be composed of eight layers of durable material with a combined minimum thickness of about 1.7 meters (5.7 feet), excluding the grading fill layer, which would range from zero at the edge to around 0.8 meters (2.6 feet) at the center for a 2 percent drainage slope. This design incorporates RCRA “minimum technology guidance” (EPA 1989) with modifications for extended performance. One deviation from the guidance consists of elimination of the clay layer, which may desiccate and crack over time in an arid environment. The geo-membrane component was also eliminated because of its uncertain long-term durability (hundreds to thousands of years). The design incorporates an asphalt layer to inhibit bio-intrusion or inadvertent human intrusion (SAIC 2010d).

### **E.2.3.3 FFTF Decommissioning Alternative 3: Removal**

Under the Removal Alternative, the RCB (and structures within) above grade level would be decontaminated as necessary, dismantled, and removed. All sodium residuals would be removed from the RCB or treated in place to neutralize its chemical reactivity. Below grade, the reactor vessel and its contaminated internals, as well as other radioactively contaminated equipment, components, piping, and materials, including any asbestos, depleted uranium shielding, and lead shielding, would also be removed. Such radioactively contaminated equipment, components, piping, and materials would include the intermediate heat exchangers, primary pumps, primary isolation valves, primary overflow tanks, interim examination and maintenance cell equipment, 8.5- to 12.2-meter-long (28- to 40-foot-long) test assembly hardware, and the interim decay storage vessel. Additional radioactively contaminated equipment from the RCB and the FFTF heat transport system would also be removed. Upon removal, this equipment and material would be disposed of in appropriate Hanford 200 Area disposal units in an IDF. The below-grade RCB and the structures within, as well as the FFTF support buildings outside the RCB area, would be decontaminated as necessary and demolished. The area previously occupied by the facilities would then be backfilled with soil to eliminate void spaces, compacted, contoured, and revegetated. An appropriate monitoring program would be established. Figure E-45 is a graphic representation of the Removal Alternative. The following sections describe the activities to be conducted under the Removal Alternative.

**Facility Disposition.** Table E-16 summarizes the proposed decommissioning activities for each building under both the Entombment and the Removal Alternatives. Under the Removal Alternative, the main RCB and the immediately adjacent support facilities with substructures (basements) would have all their above-grade structures dismantled and the contaminated demolition debris would be disposed of in an IDF. The RCB would be demolished to grade and the support facilities would be demolished to 0.91 meters (3 feet) below grade. Below grade, radioactively contaminated components and equipment

(including the reactor vessel) would be removed. However, the reinforced-concrete cavity in the RCB would remain to be backfilled with either soil or grout to minimize void space, and the surface would be contoured and revegetated. Small amounts of radioactive activation products in structural concrete and steel would remain. As discussed in the following sections, all small-diameter pipes would be removed, and sodium residuals would be either treated in place or removed from the RCB for treatment at an onsite facility to neutralize the chemical reactivity of the metallic sodium.



**Figure E-45. FFTF Decommissioning Alternative 3: Removal**

All other ancillary buildings, including their internal equipment and components, would be demolished and removed (down to a depth of 0.91 meters [3 feet] below grade). The contaminated demolition debris would be removed to an IDF for disposal, and the vacated spaces would be backfilled, compacted, contoured, and revegetated. All radioactive and/or hazardous material would be removed. Wood and large steel components would also be removed. Foundation rubble (e.g., concrete and rebar) would remain.

**Process Components.** The above- and below-grade reactor vessel, piping systems, and tanks within the RCB and the immediately adjacent buildings would be dismantled and transported to an IDF for disposal. Radioactively contaminated equipment, components, piping, tanks, and hazardous materials (including asbestos and lead shielding) would also be removed for disposal in an IDF. The reactor vessel (along with any internal piping and equipment and attached depleted uranium shielding) would be filled with grout, removed, packaged, and transported to an IDF for disposal. Uncontaminated (clean of radioactive or hazardous substances) material would not be removed, and, as previously stated, the reinforced-concrete cavity would remain. All small-diameter piping would be removed and treated in the

400 Area to remove sodium residuals, and the decontaminated equipment piping would be disposed of on site in an IDF.

**Sodium Residuals.** In the same manner as for the Entombment Alternative, all sodium residuals would be removed from the RCB systems or treated in place under the Removal Alternative. It was assumed that sodium would be drained from the plant systems to the extent practicable, followed by passivation and/or flushing with water to stabilize sodium residuals. Sodium residuals in small-diameter piping would be treated in the 400 Area after the components are removed from the reactor plant.

**Demolition and Other Waste.** Demolition debris, radioactive solid waste, radioactive liquid waste, and other regulated hazardous waste would be handled in the same manner for both action alternatives. Under the Removal Alternative, the demolition waste would be disposed of in an IDF.

**End State.** Below-grade portions of structures would be backfilled with soil, compacted to eliminate void spaces, contoured such that natural settling would not result in depressions (to avoid the potential for ponding of water), and revegetated. An appropriate site postclosure care program would be established.

## **E.2.4 FTFF Decommissioning Alternative Process Descriptions**

### **E.2.4.1 Hanford Bulk Sodium Processing**

There are approximately 1.1 million liters (300,000 gallons) of Hanford radioactively contaminated sodium that need to be disposed of. This inventory consists of three separate categories of sodium, as follows:

- **FTTF sodium.** The FTFF reactor, located in the 400 Area of Hanford, ceased operation in April 1992 and was maintained in a standby condition from January 1997 until December 2001, when DOE issued a decision to permanently deactivate the facility. Since that time, activities have focused on preparing the facility for deactivation. Historically, FTFF has reported a sodium inventory of approximately 984,200 liters (260,000 gallons) (Burke 2007). This estimate was revised and the current sodium inventory estimate is approximately 958,000 liters (253,000 gallons). As of June 2007, approximately 916,000 liters (242,000 gallons) of radioactively contaminated bulk sodium have been drained and transferred to the SSF (Chapin 2007). The SSF tanks have a total capacity of approximately 1.1 million liters (300,000 gallons).
- **Hallam sodium.** The Hallam Reactor, located in Hallam, Nebraska, shut down in 1964, and its approximately 128,700 liters (34,000 gallons) of sodium were received at Hanford in 1967. This sodium is stored in solid form under an inert cover gas in five storage tanks at the 2727-W Hallam Sodium Storage Building in the 200-West Area at Hanford (Burke 2007).
- **SRE sodium.** The SRE sodium, approximately 26,500 liters (7,000 gallons), was received at Hanford in 1975 from the SRE, located at the Santa Susanna Field Laboratory, California. This sodium is stored in solid form in 158 208-liter (55-gallon) drums sealed within 322-liter (85-gallon) overpacks. The SRE sodium is stored in eight South Alkali Metal Storage Modules in the 200-West Area CWC at Hanford (Burke 2007).

Based on the historical estimate (i.e., 1.1 million liters [300,000 gallons]), the approximate quantities of Hanford sodium remaining to be processed and the resulting quantity of 50 weight-percent caustic sodium hydroxide solution that would be produced are shown in Table E-17. The 2.5 million liters (667,000 gallons) of sodium represent less than 40 percent of the caustic sodium hydroxide solution additions required by the WTP pretreatment processes (waste processing and caustic leaching) (ANL-W and Fluor Hanford 2002).

**Table E–17. Hanford Site Radioactive Sodium Inventory**

Sodium Category	Sodium		50 Weight-Percent Caustic Sodium Hydroxide Solution	
	Liters	Metric Tons	Liters	Metric Tons
FFTF	984,200	950	2,176,600	3,360
Hallam	128,700	130	287,700	450
SRE	26,500	30	60,600	90
<b>Total</b>	<b>1,139,400</b>	<b>1,110</b>	<b>2,524,900</b>	<b>3,900</b>

**Note:** To convert liters to gallons, multiply by 0.26417.

**Key:** FFTF=Fast Flux Test Facility; SRE=Sodium Reactor Experiment.

**Source:** ANL-W and Fluor Hanford 2002.

Two options for disposal of Hanford's sodium inventory are being considered: the Hanford Reuse Option and the Idaho Reuse Option. To understand whether the radiation levels in Hanford's sodium inventory could affect the feasibility of the caustic conversion process for either option, sodium activity levels were obtained and decayed to October 2008, the assumed date used for the start of processing in the *Hanford Site Sodium Disposition Trade-Off Study* (ANL-W and Fluor Hanford 2002). Table E–18 summarizes the sodium activity levels for the major contaminants, decayed appropriately, and compares them with Experimental Breeder Reactor II (EBR-II) and Enrico Fermi Nuclear Generating Station (Fermi) sodium activity levels (at time of processing). Though the activity levels are low, this material does not qualify as nonradioactive material according to the U.S. Department of Transportation.

**Table E–18. Significant Radioisotopes in Sodium**

	Isotope (nCi/gram)			Date of Activity
	Sodium-22	Cesium-137	Hydrogen-3 (Tritium)	
FFTF primary sodium	5.6	$4.8 \times 10^{-2}$	$5.2 \times 10^1$	October 2008
Hallam sodium	$4.6 \times 10^{-4}$	$4.9 \times 10^{-4}$	$1.2 \times 10^1$	October 2008
SRE sodium	$1.8 \times 10^{-4}$	$8.5 \times 10^{-1}$	N/A	October 2008
Fermi sodium	$2.2 \times 10^{-2}$	$5.6 \times 10^{-1}$	1.2	December 1998
EBR-II primary sodium	$2.05 \times 10^1$	$1.09 \times 10^1$	$2.52 \times 10^2$	September 2000
DOT limit <sup>a</sup>	2	2	2	–

<sup>a</sup> Maximum activity for nonradioactive material shipments (49 CFR 173.403).

**Key:** DOT=U.S. Department of Transportation; EBR-II=Experimental Breeder Reactor II; FFTF=Fast Flux Test Facility; N/A=not applicable; nCi=nanocuries; SRE=Sodium Reactor Experiment.

**Source:** ANL-W and Fluor Hanford 2002.

Elemental sodium is a silver, soft, and ductile alkali metal at room temperature and has a density slightly less than that of water. Sodium reacts vigorously with water and steam and is extremely reactive, oxidizing rapidly when exposed to air. It melts at about 190 °C (208 °F) to form a silvery liquid. The normal boiling point of sodium is 1,600 °C (1,618 °F). The basic chemical reaction is an exothermic reaction with water that, for excess water, produces a caustic sodium hydroxide solution and the evolution of hydrogen gas:



Liquid sodium would be transferred from a storage tank into the facility where the reaction would take place and would be controlled by adjusting the injection rate of the liquid reactants. The process would occur in the reactor vessel, which is a nickel pressure vessel (4.6 meters [15 feet] tall by 0.8 meters [30 inches] in diameter). The entire system would use nitrogen as an inert cover and pressurizing gas.

For a 50 weight-percent caustic sodium hydroxide solution, the reaction would occur at approximately 138 °C (280 °F). Offgases emitted during the process would contain hydrogen, nitrogen, and water vapor. They would be exhausted from the vessel, dried, scrubbed, HEPA-filtered, and monitored before venting as a nonflammable nitrogen/hydrogen mixture. The final caustic sodium hydroxide solution would be pumped from the reaction vessel to a fill station where transportation tanks or drums would be used to contain it for storage before shipment off site (ANL-W and Fluor Hanford 2002).

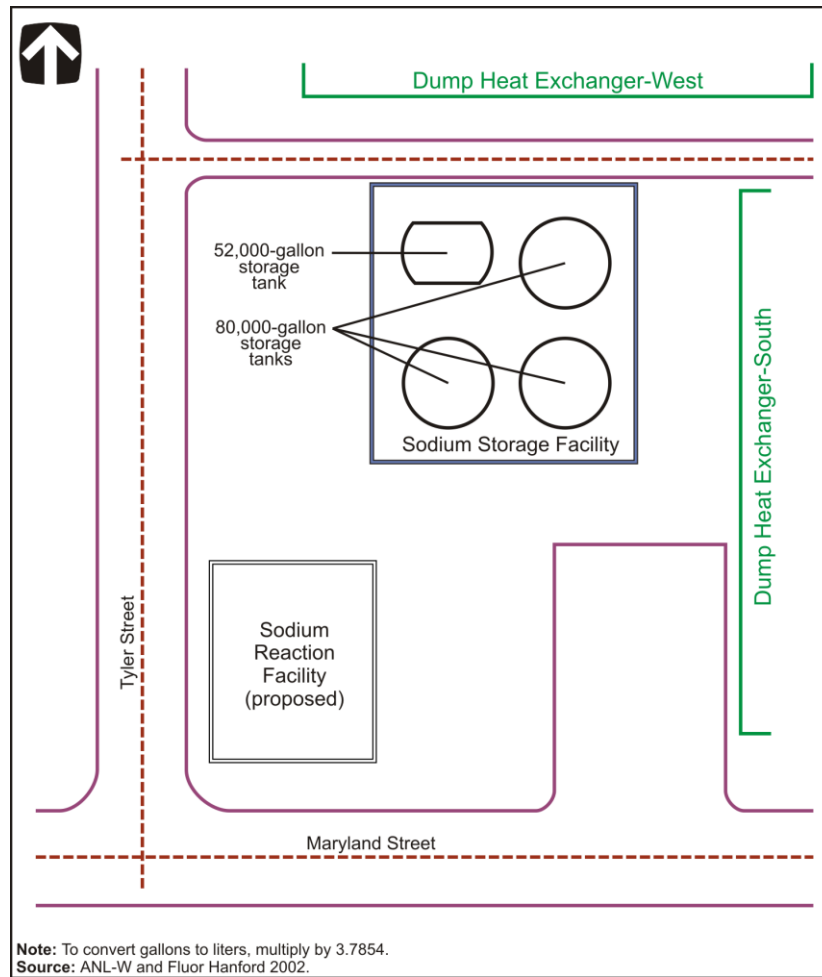
#### **E.2.4.2 Sodium Reaction Facility—Hanford Reuse Option**

This section provides background information on processing of the Hanford bulk sodium at Hanford. The inventory of the Hanford bulk sodium to be processed is described in detail in Section E.2.4.1. The sodium reaction process utilized by the SPF at INL, the model for processing at the FFTF SRF, is described in detail in Section E.2.4.3.

##### **E.2.4.2.1 Description of the Sodium Reaction Facility**

The SRF at Hanford would be used to house the process for converting sodium into a caustic sodium hydroxide solution. The SRF would be directly adjacent to the existing SSF, as shown in Figure E-46. Locating the system adjacent to the SSF would reduce construction and operation costs through the sharing of utilities and operational integration. The SSF, an existing building west of the FFTF south dump heat exchanger, consists of three 300,000-liter (80,000-gallon) tanks and one 200,000-liter (52,000-gallon) tank. The SSF structure is 28 meters long by 27 meters wide by 9.1 meters high (91 feet long by 90 feet wide by 30 feet high). The SSF would be used to store the bulk sodium until its transfer to the SRF for treatment. An exterior photo of the SSF is shown in Figure E-47.

For analysis purposes, this *TC & WM EIS* assumes that the process to be used at the SRF to produce the caustic sodium hydroxide solution would be identical to the process used during the processing of the EBR-II bulk sodium at the SPF at INL. The following subsections describe the individual SRF systems/components.



**Figure E-46. Location of the Sodium Reaction Facility and the Sodium Storage Facility**



Source: ANL-W and Fluor Hanford 2002.

**Figure E-47. Sodium Storage Facility at Hanford**

#### **E.2.4.2.2 Sodium Barrel Melt-and-Drain System**

SRE sodium would arrive at the SRF in its currently packaged 208-liter (55-gallon) drums, each contained in its own overpack, and would be transferred into the SRF day tanks by the barrel melt-and-drain system.

System components would include the following:

- Barrel melt-and-drain stations
- Clamp-on band heaters and thermocouples
- Associated piping (nitrogen supply lines, sodium transfer lines)
- Control panels for each drain station
- HEPA-filtered room exhaust system
- Radiation and contamination detectors

#### **E.2.4.2.3 Sodium Reaction Facility Transfer Line**

The SRF transfer line would be an approximate 3.8-centimeter (1.5-inch) heated, stainless steel, insulated pipeline used to transfer sodium from the SSF to the SRF. An isolation valve would be installed at each end of the SRF transfer line. The transfer line would be designed with a downward slope to maximize draining of the pipe when pumping ceases. As with the SPF transfer line, secondary containment of sodium in double-walled piping would not be required for sodium transfer lines. The transfer line would be trace-heated with resistive heaters and insulated and sealed with a closure system and vapor barrier that would be finished with a weatherproof jacketing material.

#### **E.2.4.2.4 Sodium Transfer System**

This system would transfer sodium from the SSF to the SRF's sodium day tanks. During processing of the sodium, the day tanks would receive sodium from one of two sources:

- SRE sodium from the barrel melt-and-drain system
- FFTF and Hallam sodium from the SSF via the transfer line

There would be two identical carbon-steel sodium day tanks. The SSF nitrogen blanket would be pressurized to push sodium from the SSF to fill one of the day tanks at approximately 110 liters (30 gallons) per minute, while the other day tank would be used for processing.

#### **E.2.4.2.5 Sodium Reaction System**

The reaction system would be used to perform the chemical conversion of liquid metallic sodium to a caustic sodium hydroxide solution. This reaction would take place in the reaction vessel when sodium is transferred from the day tanks to the reaction vessel. This transfer would be accomplished by pressurizing the in-service day tank with nitrogen gas.

The reaction vessel would be a 0.8-meter-diameter by 4.6-meter-high (30-inch-diameter by 15-foot-high) vertical cylinder constructed from caustic corrosion-resistant nickel alloy 200. After passing through an injection nozzle, the sodium would react with water to produce a caustic sodium hydroxide solution and hydrogen gas.

By controlling the atmospheric boiling point of the solution with the periodic addition of water, the concentration of the sodium hydroxide product would be fixed. Sodium would be injected into the reaction vessel through specially designed nozzles capable of adding steam or nitrogen gas (or both) simultaneously at the point of injection. Nitrogen would be introduced into an annulus area at the nozzle

tip to aid in atomizing the sodium upon injection into the reaction vessel. The more completely the sodium was atomized, the greater would be the surface area of sodium exposed and the quieter, less forceful the resulting reaction between the water and sodium as experienced by the SPF. This would also ensure completion of the sodium reaction beneath the surface of the hydroxide solution where the energy release can be readily absorbed, and thus would eliminate the possibility of a carryover of sodium into the offgas. In addition, the capability for initiating a flow of nitrogen or steam to the nozzles after sodium flow was terminated would minimize plugging; if plugging occurred, steam could be used to clear the plug. Each nozzle would have its own electromagnetic flow meter. The nitrogen would be vented from the reaction vessel via the caustic offgas system along with the reaction-produced hydrogen and some water vapor.

The resulting concentration of caustic sodium hydroxide solution in the reaction vessel would be circulated continuously using a caustic recirculation pump. This recirculation pump would take suction from the bottom of the reaction vessel and discharge it near the top of the reaction vessel. Circulation of the caustic sodium hydroxide solution combined with the vigorous nature of the sodium/water reaction would ensure uniform mixing of the solution.

#### **E.2.4.2.6 Caustic Transfer System**

The caustic transfer system would consist of the following components:

- Caustic recirculation pump
- Caustic recirculation line
- 3,800-liter (1,000-gallon) caustic cooling tank
- Caustic product transfer line (with concentric heat exchanger)
- Caustic metering pump
- Caustic transfer pump
- 15,000-liter (4,000-gallon) caustic storage tank

The caustic recirculation pump would take suction from the bottom of the reaction vessel and discharge to piping that would either (1) return the solution to the vessel, (2) divert some of the solution to the product transfer line, or (3) pump down the contents of the vessel to the caustic cooling tank.

The 3,800-liter (1,000-gallon) caustic cooling tank would measure approximately 1.2 meters in diameter by 3.4 meters long (4 feet in diameter by 11 feet long). It would contain a heat exchanger that would reduce the caustic temperature below the levels necessary for caustic corrosion when the product was being pumped from the reaction vessel to the caustic storage tank. The caustic cooling tank also would be used to store some caustic sodium hydroxide solution when the reaction vessel is drained for maintenance. One of the first evolutions to be performed during process startup would involve using the caustic metering pump to transfer this initial charge of caustic sodium hydroxide solution from the caustic cooling tank directly to the reaction vessel. Sodium would be injected into this volume of caustic sodium hydroxide solution to resume processing operations. All piping would be made of nickel because of its corrosion resistance to high-temperature sodium hydroxide.

The caustic transfer system would be equipped with two automatic flow-control valves, as follows:

- One automatic flow-control valve would divert approximately 3.8 to 7.6 liters (1 to 2 gallons) per minute of the caustic sodium hydroxide solution from the caustic recirculation line to the product fill line. This diversion would be set to maintain a specified level in the reaction vessel.
- The other flow-control valve would direct caustic to the caustic sodium hydroxide solution cooling tank during process shutdown.



#### **E.2.4.2.7 Product System**

When the level of caustic sodium hydroxide solution in the reaction vessel reached the control set point, an automatic valve would open, diverting this product solution from the discharge of the caustic recirculation pump to the product container via a product fill line. The product fill container would be determined later by Hanford personnel. The product fill line would be a concentric pipe, counter-flow heat exchanger, which would have an inner pipe (nickel) utilized for high-temperature sodium hydroxide and an outer pipe (stainless steel) containing a coolant (chemically treated water). The fill line would be routed from the process area to the product area.

After a product container was placed in the product area and the fill line nozzle was connected, an operator would actuate a switch on the offload station control panel to begin the product container fill process. If the level of caustic sodium hydroxide solution in the reaction vessel was above the level control set point, it would enable the opening of an automatic valve that would divert caustic sodium hydroxide solution to the product area. The caustic sodium hydroxide solution passing through the product fill line concentric heat exchanger would be cooled prior to entering the product container. When sensors indicated that the product container was full, the operator would actuate another switch on the offload station control panel to temporarily reroute the product to the caustic storage tank. This lineup would allow processing to continue until another product container was connected to the fill-line nozzle. At this point, an operator would sample the full product container for product quality, and concentration would be adjusted as necessary. It would then be sealed and surveyed by the health physics technician to verify outer cleanliness and contact radiation levels. It would then be moved aside to one of the storage bays, and another product container would be put in place to begin the filling process.

#### **E.2.4.2.8 Service and Suspect Water Systems**

Two separate water systems—the service and suspect water systems—would make up the SRF water system. The service water system would be a clean potable water system that functions (1) to cool the offgas condenser and (2) to serve as a heat sink for the product-fill heat exchanger. Service water would be supplied to the SRF from a sitewide source at approximately 690,000 pascals, gauge (100 pounds per square inch, gauge). The primary purpose of the suspect water system would be to provide the water that is injected into the reaction vessel to control the main processing temperature. It would also provide baffle sprays in the reaction vessel and makeup water to the offgas scrubber. The primary source of the water in this system would be the condensation and drains from the offgas system. This water would be potentially contaminated with radioactive constituents. The drains would be collected in a small collection vessel and pumped to a 1,900-liter (500-gallon) water holding tank. A pump would then take suction on the water holding tank and discharge it to the reaction vessel. Makeup water to the system would be supplied to the water holding tank from a deionized water system.

#### **E.2.4.2.9 Caustic Offgas System**

The system would be designed to perform the following actions:

- Contain and recycle the water vapor and caustic carryover from the reaction vessel.
- Remove the gases produced during the process of converting sodium to sodium hydroxide.
- Remove the nitrogen that collects in the vessel from the various purges in the reaction system.
- Provide for filtration, cleaning, and monitoring of gases and particulates carried over in the offgas stream in accordance with environmental and engineering standards prior to their discharge to the environment.

The system would be composed of the following components, as well as interconnecting piping (components are listed in flow path order):

- Reaction vessel baffles
- Reaction vessel demister
- Offgas condenser
- Mist eliminator (mesh-type)
- Scrubber
- Moisture separator (vane-type)
- Parallel prefilters (with polypropylene material)
- Parallel HEPA filters
- Building discharge piping

The principal constituents in the caustic offgas stream as it exited the reaction vessel would be water vapor, hydrogen, and nitrogen. Hydrogen is a reaction product of sodium and water and would make up approximately 60 to 70 percent of the offgas-stream volume at the anticipated SRF sodium injection rate of 2.3 liters (0.6 gallons) per minute to the reaction vessel. Nitrogen would be used to atomize the sodium in the reaction vessel injection nozzles, to do the initial purge of the reaction vessel and offgas system, and to provide an inert cover gas in all process system tanks and vessels to eliminate oxygen. The caustic offgas system would process this gas stream and ultimately release hydrogen and nitrogen outside the SRF building to the atmosphere and return the condensed water vapor to the reaction process. Condensate would be returned to the water holding tank through a series of drain lines.

#### **E.2.4.2.10 Vent Systems**

Two independent vent systems would be associated with the sodium process area and the 15,000-liter (4,000-gallon) caustic sodium hydroxide solution storage tank. All vent system piping would be carbon steel. The vent systems would collect gaseous effluents from all tanks in the sodium process area, including the following:

- 2,760-liter (730-gallon) sodium day tanks
- 3,800-liter (1,000-gallon) caustic cooling tank
- 15,000-liter (4,000-gallon) caustic storage tank
- 1,900-liter (500-gallon) water holding tank

#### **E.2.4.2.11 Steam System**

Steam would be used to clear the injection nozzles in the reaction vessel in the sodium process area.

#### **E.2.4.2.12 Nitrogen System**

Because of the reactive nature of sodium, nitrogen gas would be utilized in the SRF as the primary source of inert gas supplied for all applications requiring a cover gas or motive force.

#### **E.2.4.2.13 Control Air System**

The compressed-air system would be used to perform the following actions:

- Operate the barrel-tilting mechanism for the melting-draining operation.
- Operate the pneumatically actuated valves in the sodium, caustic sodium hydroxide solution, and vacuum piping.
- Operate the pneumatic tilting device to move drums of hydroxide to the drum palletizing area if drums are used as the product fill containers.

#### **E.2.4.2.14 Process Computer System**

The SRF processes would be monitored and controlled using a computer control system consisting of the following components:

- A main control computer
- A graphics computer
- Two bus computers

Process instrumentation and actuators would be connected to the computer control system through input/output channels on the two bus computers. Operators would interact with computer color graphics screens to control and monitor the processes. For certain portions of the process system, operators would interact with the control system through pushbutton/indicator panels located in specific process areas.

#### **E.2.4.2.15 Sodium Throughput**

The SRF is modeled directly on the SPF for throughput. The SPF processed sodium at a nominal rate of 2.3 liters (0.6 gallons) per minute, resulting in about 136 liters (36 gallons) per hour, or 3,220 liters (850 gallons) per day. At this rate, approximately 7,600 liters (2,000 gallons) of 50 weight-percent caustic sodium hydroxide solution could be produced each day (ANL-W and Fluor Hanford 2002).

#### **E.2.4.3 Sodium Processing Facility—Idaho Reuse Option**

The SPF (see Figures E-48 and E-49) was originally constructed in the mid-1980s to convert sodium from the Fermi reactor plant into 50 weight-percent caustic sodium hydroxide solution for use in the PUREX process at Hanford. This use was abandoned after the SPF was constructed, but before it began operations. Once the EBR-II was ordered shut down, defueled, and prepared for deactivation, the SPF was used as a means of preparing the Fermi and EBR-II sodium for disposal at the Radioactive Waste Management Complex at INL. Production operations with radioactive sodium began on December 20, 1998. Processing of all EBR-II and Fermi sodium was completed on March 5, 2001, and the facility was placed in a standby condition. To date, approximately 662,445 liters (175,000 gallons) of radioactive sodium have been processed in the SPF.



Source: ANL-W and Fluor Hanford 2002.

**Figure E-48. Experimental Breeder Reactor II/  
Sodium Processing Facility Complex**



Source: ANL-W and Fluor Hanford 2002.

**Figure E-49. Sodium Processing Facility at the  
Materials and Fuels Complex**

The purpose of the SPF is to react sodium with water to produce a caustic sodium hydroxide solution. The process has the capability of producing any concentration of this solution simply by changing the processing temperature.

This would allow conversion of the Hanford sodium to a 50 weight-percent caustic sodium hydroxide solution, which is specified for use by ORP at Hanford.

The SPF can receive sodium in the following ways:

- The 208-liter (55-gallon) barrels can be delivered directly to the SPF, where the contents can be melted and drained to a sodium storage tank. The Fermi sodium was received in this manner, which would also be the method of transferring the SRE sodium into the facility.
- Sodium can be transferred via a heated pipeline that runs from the secondary sodium drain tank located in the EBR-II secondary sodium boiler building basement. This method was used to transfer EBR-II primary and secondary sodium, and could also be used to transfer the Hallam and FFTF sodium into the SPF.

Once the sodium is in the SPF and in the sodium storage tank, it can be transferred to one of two day tanks, each with a working volume of 2,570 liters (680 gallons), by pressurizing the sodium storage tank with nitrogen gas. During normal operations, one of the day tanks can be filled from the sodium storage tank while the other is used to supply sodium to the reaction vessel, which also can be accomplished by pressurizing the tank with nitrogen gas.

In the reactor vessel, the sodium would react with the water in the caustic sodium hydroxide solution used to initially charge the vessel. This reaction would release heat, which would increase the temperature of the caustic sodium hydroxide solution in the vessel until it reached the control set point. As this would be a saturated boiling system, the end caustic product concentration would be determined by this temperature set point. A 50 weight-percent liquid product would require an operating temperature of 138 °C (280 °F), as opposed to the 186 °C (367 °F) used to process EBR-II sodium into a 70 weight-percent (solid) caustic waste form. Water would be injected into the reactor vessel intermittently to maintain the control set point within  $\pm 0.5$  °C ( $\pm 0.5$  °F).

As the sodium reacts and new caustic sodium hydroxide solution is produced, the level in the reaction vessel would increase. When it reaches the level control set point, operators would begin filling a 15,000-liter (4,000-gallon) tank or direct the 50 weight-percent caustic sodium hydroxide solution to the caustic storage tank until another tank was readied for filling. The majority of instrumentation and system controls at the SPF are coordinated through a control computer in the SPF control room, which permits system operations with minimal operator input (ANL-W and Fluor Hanford 2002).

#### **E.2.4.3.1 Facility Description**

The SPF complex consists of several buildings. These include the original SPF building (and a large addition to it), as well as the caustic storage tank room, an operations support trailer, the EBR-II sodium boiler building, and the sodium transfer line located in the yard area between the sodium boiler building and the SPF.

The SPF currently consists of a four-room metal building housing the barrel melt-and-drain room, barrel holding room, equipment room, control room, and a carbon steel-lined concrete pad on which the process equipment (process area) is located. This part of the building is 20.4 meters long by 17.4 meters wide (67 feet long by 57 feet wide). A newer, large addition to the original building contains the product area. This addition is approximately 22.6 meters long by 7.6 meters wide (74 feet long by 25 feet wide). Two attached storage bays are, in combined outside dimensions, 9.8 meters long by 7.3 meters wide (32 feet long by 24 feet wide).

The SPF is supported on a thickened-edge, reinforced-concrete pad. Most of the exterior is constructed of galvanized-steel siding and roof panels on a structural-steel frame. However, the barrel melt-and-drain room has 30.5-centimeter-thick (12-inch-thick) reinforced-concrete block walls and a 20.3-centimeter-thick (8-inch-thick) reinforced-concrete slab roof. All sections of the building meet the requirements of the Uniform Building Code and Seismic Zone 2 or 2B.

A small metal-sided building, constructed over a lined concrete secondary-containment basin, is just west of the original SPF building. It houses the caustic storage tank. An operations support trailer provides office space, a lunchroom, a locker room, and showers for the operating crews. The EBR-II sodium boiler building houses the secondary sodium drain tank, a recirculation system, and pumps used to transfer sodium to the SPF.

#### **E.2.4.3.2 Barrel Melt-and-Drain Room**

This room contains the eight melt-and-drain stations that would be used to melt the sodium in the 208-liter (55-gallon) SRE drums.

#### **E.2.4.3.3 Barrel Holding Room**

The 208-liter (55-gallon) sodium drums would be brought into this area through a sliding service door in the east exterior wall. This room is a staging area that supports the barrel melt-and-drain room.

#### **E.2.4.3.4 Equipment Room**

This room houses several electrical panels, the remote Met-L-X fire control station for the barrel melt-and-drain room, the constant electric power unit, and several sitewide radio networks utilized by the facility area supervisor. It also is used by the facility area supervisor as office space for conducting facility business, such as work control and lockout/tagout.

#### **E.2.4.3.5 Control Room**

This area contains the main control computer that is used to control and monitor all process functions. It also contains controls for a remote video camera monitoring the process area, three backup computers used to monitor process parameters, and the rest of the emergency communications equipment used by the facility area supervisor for emergency response.

#### **E.2.4.3.6 Process Area**

This room contains all of the major equipment necessary to convert sodium to caustic sodium hydroxide solution. There is a steel-lined secondary-containment basin below all tanks and most piping containing caustic or sodium in this area, as well as systems to detect hydrogen leaks and fires.

#### **E.2.4.3.7 Product Area**

This area houses equipment that was used for filling waste drums when EBR-II sodium was processed and now would be used for filling 15,000-liter (4,000-gallon) tanks after modifications are made. This area also contains chemical analysis equipment for product quality verification, the ventilation system equipment for the main facility, and two bays that can be used for heated storage of product or waste containers. The building height in the main part of the product area is approximately 9.4 meters long and about 3.6 meters wide (31 feet long and about 12 feet wide) in the storage bays. A floor area just inside a rollup door on the east side of the building has an open workspace approximately 9.1 meters long by 7.3 meters wide (30 feet long by 24 feet wide) that would be used to house an International Organization for Standardization (ISO) tank while it is being filled with caustic sodium hydroxide solution (ANL-W and Fluor Hanford 2002).

### **E.2.4.3.8 Sodium Processing Facility Basic System Descriptions**

#### **E.2.4.3.8.1 Sodium Barrel Melt-and-Drain System**

The SRE sodium would arrive at the SPF in its currently packaged 208-liter (55-gallon) drums, each contained in its own overpack, and would be transferred into the SPF sodium storage tank by the barrel melt-and-drain system.

System components would include the following:

- Eight-barrel melt-and-drain stations
- Clamp-on band heaters and thermocouples
- Associated piping (nitrogen supply lines, sodium transfer lines)
- Control panels for each drain station
- HEPA-filtered room exhaust system
- Radiation and contamination detectors
- Fire suppression system (Ansul Met-L-X) for the barrel container assemblies

#### **E.2.4.3.8.2 Sodium Processing Facility Transfer Line**

The SPF transfer line is a heat-traced, 2.5-centimeter-diameter (1-inch-diameter) stainless steel, insulated pipeline that would be used to transfer sodium a distance of approximately 270 meters (900 feet) from the secondary sodium drain tank in the basement of the EBR-II sodium boiler building to the SPF. An isolation valve is installed at each end of the SPF transfer line (i.e., in the sodium boiler building and the SPF). The transfer line is routed out of the sodium boiler building just above grade level and thence west and north toward the SPF. The transfer line enters the SPF through the west wall of the process area and connects to the top of the sodium storage tank at a flange. The transfer line is designed to maximize draining of the pipe when pumping ceases due to its downward slope from the high point just west of the SPF down to the sodium boiler building basement. The transfer line is trace-heated with resistive heaters and insulated and sealed with a closure system and vapor barrier that is finished with a weatherproof jacketing material.

#### **E.2.4.3.8.3 Sodium Transfer System**

This system would transfer sodium from the SPF sodium storage tank to the sodium day tanks. During processing of the Hanford sodium, the sodium storage tank would receive sodium from one of two sources:

- SRE sodium transferred from the barrel melt-and-drain system
- FFTF and Hallam sodium transferred from the secondary sodium drain tank via the transfer line

The secondary sodium drain tank has an effective volume of 56,800 liters (15,000 gallons), and the sodium in this tank could be either recirculated or pumped to the SPF using two annular, linear induction pumps at about 90.8 liters (24 gallons) per minute. The sodium storage tank is a carbon steel storage tank that has a working volume of 16,300 liters (4,300 gallons). A vacuum system is used to create a vacuum in the tank to provide the motive force for sodium transfers into the tank. A nitrogen blanket is also maintained over the tank contents to keep the contents of the tank inert.

There are two identical carbon steel sodium day tanks, each with a working volume of 2,570 liters (680 gallons). The sodium storage tank nitrogen blanket is pressurized to push sodium from the sodium storage tank to fill one of the day tanks at approximately 110 liters (30 gallons) per minute, while the other day tank is used for processing.

#### **E.2.4.3.8.4 Sodium Reaction System**

The purpose of the reaction system is to perform the chemical conversion of liquid metal sodium to a caustic sodium hydroxide solution. Sodium transfer would be accomplished by pressurizing the in-service day tank with nitrogen gas, which provides the driving force for the injection of sodium into the reaction vessel, where the reaction takes place.

The reaction vessel is a 0.8-meter-diameter (30-inch-diameter) by 4.6-meter-high (15-foot-high) vertical cylinder constructed from caustic, corrosion-resistant, alloy 200 nickel. After passing through an injection nozzle, the sodium would react with water to produce a caustic sodium hydroxide solution and hydrogen gas.

By controlling the boiling point of the solution with the periodic addition of water, the concentration of the sodium hydroxide product would be fixed. Sodium would be injected into the reaction vessel through specially designed nozzles capable of adding steam or nitrogen gas (or both) simultaneously at the point of injection. Nitrogen would be introduced into an annulus area at the nozzle tip to aid in atomizing the sodium upon injection into the reaction vessel. The more completely the sodium is atomized, the greater the surface area of sodium exposed and the quieter, less forceful the resulting reaction between the water and sodium. This also would ensure completion of the sodium reaction beneath the surface of the hydroxide solution where the energy release can be readily absorbed, and thus would eliminate the possibility of the carryover of sodium into the offgas. In addition, the capability to initiate a flow of nitrogen or steam to the nozzles after sodium flow was terminated would minimize plugging; if plugging occurred, the steam could be used to clear the plug. Each nozzle has its own electromagnetic flow meter. Nitrogen would be vented from the reaction vessel via the caustic offgas system along with the reaction-produced hydrogen, water vapor, and some caustic carryover.

The resulting concentration of caustic sodium hydroxide solution in the reaction vessel would be circulated continuously using a caustic recirculation pump. Circulation of the caustic sodium hydroxide solution combined with the vigorous nature of the sodium/water reaction would ensure uniform mixing of the solution.

#### **E.2.4.3.8.5 Caustic Transfer System**

The caustic transfer system consists of the following components:

- Caustic recirculation pump
- Caustic recirculation line
- 3,800-liter (1,000-gallon) caustic cooling tank
- Caustic product transfer line (with concentric heat exchanger)
- Caustic metering pump
- Caustic transfer pump
- 15,000-liter (4,000-gallon) caustic storage tank

The caustic recirculation pump would take suction from the bottom of the reaction vessel and discharge to piping that can either (1) return the solution to the vessel, (2) divert some of the solution to the product transfer line, or (3) pump down the contents of the vessel to the caustic cooling tank.

The 3,800-liter (1,000-gallon) caustic cooling tank is 3.4 meters long by 1.2 meters in diameter (11 feet long by 4 feet in diameter). It contains a heat exchanger, which would reduce the caustic temperature below the levels necessary for caustic corrosion when the product is being pumped from the reaction vessel to the caustic storage tank. The caustic cooling tank also would be used to store some caustic sodium hydroxide solution when the reaction vessel is drained for maintenance. One of the first



evolutions performed during process startup would involve using the caustic metering pump to transfer this initial charge of caustic sodium hydroxide solution from the caustic cooling tank directly to the reaction vessel. Sodium would be injected into this volume of caustic sodium hydroxide solution to resume processing operations.

The caustic metering pump is a sealless, magnetic-drive, centrifugal pump with a rated flow of 7.6 liters (2 gallons) per minute. All piping is made of nickel because it resists corrosion by high-temperature caustic. The caustic transfer system is equipped with two automatic flow control valves that function as follows:

- One valve would divert approximately 3.8 to 7.6 liters (1 to 2 gallons) per minute of the caustic sodium hydroxide solution from the caustic recirculation line to the product fill line. This diversion would be set to maintain a specified level in the reaction vessel.
- The other valve would direct caustic sodium hydroxide solution to the caustic cooling tank during process shutdown.

#### **E.2.4.3.8.6 Product Offload**

When the level of caustic sodium hydroxide solution in the reaction vessel reached the control set point, an automatic valve would open to divert this product solution from the discharge of the caustic recirculation pump to the ISO tank via a product fill line. The product fill line is a concentric pipe, counter-flow heat exchanger that has an inner pipe (nickel) utilized for high-temperature sodium hydroxide and an outer pipe (stainless steel) containing a coolant (chemically treated water). The fill line is routed from the process area to the product area.

After an ISO tank was placed in the product area and the fill line nozzle was connected, an operator would actuate a switch at the offload station control panel to begin the ISO tank fill process. If the level of caustic sodium hydroxide solution in the reaction vessel were above the level control set point, it would enable the opening of an automatic valve that would divert caustic sodium hydroxide solution to the product area. The caustic sodium hydroxide solution passing through the product fill line's concentric heat exchanger would be cooled prior to entering the ISO tank.

When sensors indicated that the ISO tank was full, the operator would actuate another switch at the offload station control panel to temporarily reroute the product to the caustic storage tank. This lineup would allow processing to continue until another ISO tank was connected to the fill-line nozzle. At this point, the full ISO tank could be sampled by an operator to verify product quality. It would then be sealed and surveyed by a health physics technician to verify outer cleanliness and contact radiation levels. It would then be moved aside to one of the storage bays, and another ISO tank would be put in place to begin the filling process.

#### **E.2.4.3.8.7 Service and Suspect Water Systems**

Two separate water systems—the service and suspect water systems—make up the SPF water system. The service water system is a clean potable water system. Its functions are to cool the offgas condenser and to serve as a heat sink for the product fill heat exchanger. The service water is supplied to the SPF from a sitewide source at approximately 690,000 pascals, gauge (100 pounds per square inch, gauge) through galvanized-steel piping. The primary purpose of the suspect water system would be to provide the water injected into the reaction vessel that controls the main processing temperature. It also would provide baffle sprays in the reaction vessel and makeup water to the offgas scrubber. The primary sources of the suspect water in this system would be the condensation and drainage from the offgas system. This water would be potentially contaminated with radioactive constituents. The drainage would be collected in a small collection vessel and pumped to the 1,900-liter (500-gallon) water holding tank. A

pump would then take suction on the water holding tank and discharge it to the reaction vessel. Makeup water to the system would be supplied to the water holding tank from the MFC's deionized water system.

#### **E.2.4.3.8.8 Caustic Offgas System**

The system is designed to perform the following actions:

- Contain and recycle water vapor and caustic carryover from the reaction vessel.
- Discharge the gases produced during the process of converting sodium to sodium hydroxide.
- Discharge the nitrogen that collects in the vessel from the various purges in the reaction system.
- Provide filtration, cleaning, and monitoring of gases and particulates carried over in the offgas stream in accordance with environmental and engineering standards prior to their discharge to the environment.

The caustic offgas system has the following components, as well as interconnecting piping (components are listed in flow path order):

- Reaction vessel baffles
- Reaction vessel demister
- Offgas condenser
- Mist eliminator (mesh-type)
- Scrubber
- Moisture separator (vane-type)
- Parallel prefilters (with polypropylene material)
- Parallel HEPA filters
- Building discharge piping

The principal constituents in the caustic offgas stream as it exited the reaction vessel would be water vapor, hydrogen, and nitrogen. Hydrogen is a reaction product of sodium and water and would make up approximately 60 to 70 percent of the offgas stream volume at the rated sodium injection rate of 2.3 liters (0.6 gallons) per minute to the reaction vessel. Nitrogen would be used to atomize the sodium in the reaction vessel injection nozzles, to do the initial purge of the reaction vessel and offgas system, and to provide an inert cover gas in all process system tanks and vessels to eliminate oxygen. The caustic offgas system would process this gas stream and ultimately release hydrogen and nitrogen outside the SPF building to the atmosphere and return the condensed water vapor to the reaction process. Condensate would be returned to the water holding tank through a series of drain lines.

#### **E.2.4.3.8.9 Vent Systems**

Two independent vent systems are associated with the sodium process area and 15,000-liter (4,000-gallon) caustic storage tank. All vent system piping is carbon steel. The vent systems would collect gaseous effluents from all tanks in the sodium process area, including the following:

- 16,300-liter (4,300-gallon) sodium storage tank
- 2,570-liter (680-gallon) sodium day tanks
- 3,800-liter (1,000-gallon) caustic cooling tank
- 15,000-liter (4,000-gallon) caustic storage tank
- 1,900-liter (500-gallon) water holding tank

#### **E.2.4.3.8.10 Steam System**

Steam would be used to clear the injection nozzles in the reaction vessel in the sodium process area. The 1.2-million-pascal, gauge (175-pound-per-square-inch, gauge), steam from the site steam supply would be reduced in pressure to 200,000 to 350,000 pascals, gauge (30 to 50 pounds per square inch, gauge) (measured in respect to atmospheric pressure).

#### **E.2.4.3.8.11 Nitrogen System**

Because of the reactive nature of sodium, nitrogen gas would be utilized in the SPF as the primary source of inert gas supplied for all applications requiring a cover gas or motive force.

#### **E.2.4.3.8.12 Control Air System**

The compressed-air system is supplied by a 25-horsepower air compressor that would provide cool, filtered air at more than 690,000 pascals, gauge (more than 100 pounds per square inch, gauge), for the following uses:

- Operation of the barrel-tilting mechanism for the melting-draining operation
- Operation of the pneumatically actuated valves in the sodium, caustic sodium hydroxide solution, and vacuum piping
- Operation of the pneumatic tilting device to move drums of caustic sodium hydroxide solution to the drum palletizing area

#### **E.2.4.3.8.13 Process Computer System**

The SPF processes would be monitored and controlled using a computer control system consisting of the following components:

- A main control computer
- A graphics computer
- Two bus computers

Process instrumentation and actuators are connected to the computer control system through input/output channels on the two bus computers. Operators would interact with computer color graphics screens to control and monitor the processes. For certain portions of the process system, operators would interact with the control system through pushbutton/indicator panels located in specific process areas (ANL-W and Fluor Hanford 2002).

#### **E.2.4.3.8.14 Sodium Throughput**

The SPF processed sodium at a nominal rate of 2.3 liters (0.6 gallons) per minute, resulting in about 136 liters (36 gallons) per hour, or 3,220 liters (850 gallons) of sodium processed per day. At this rate, approximately 7,600 liters (2,000 gallons) of 50 weight-percent caustic sodium hydroxide solution could be produced each day; thus, 2 days would be required to fill a 15,000-liter (4,000-gallon) ISO tank.

The SPF processed approximately 658,660 liters (approximately 174,000 gallons) of EBR-II and Fermi sodium in just over 2 years. The plant utilization factor steadily increased over this period. During the initial startup of the SPF, the plant factor was low while engineering personnel refined the process's performance and operations personnel gained experience in how to run the process most efficiently. This is represented by a plant utilization factor of only 11 percent achieved in the first 3 months of process

operation. As experience was gained and lessons learned were implemented, the plant utilization factor increased until an average of approximately 62 percent was achieved for the last 6 months of operation. A projected plant utilization factor of 65 percent for resumed operation of the SPF, or the operation of a completely new facility, is well within expectations due to further implementation of lessons learned. In addition, production of 50 weight-percent caustic sodium hydroxide solution rather than a 70 weight-percent caustic would be much easier on systems and components, resulting in less downtime for repairs and maintenance (ANL-W and Fluor Hanford 2002).

#### **E.2.4.4 Remote-Handled Special Components Processing**

##### **E.2.4.4.1 Fast Flux Test Facility Remote-Handled Special Components Description**

Removal of FFTF RH-SCs, which would be completed under the deactivation work, was evaluated in the 2006 EA (DOE 2006). The removed RH-SCs would be stored within the FFTF complex until the selected treatment facility had been built and was ready to receive them.

As background, FFTF RH-SCs include the primary cold trap (N-5), the cesium trap (N-3), two sodium condenser vapor traps (U-527 and U-532), and the associated filter vapor traps (VT-61, VT-62, VT-63, and VT-64). Each of these components has a high-radiation-dose level due to the presence of high-energy, gamma-emitting fission products (primarily cesium-137). The primary cold trap and the cesium trap both contain sodium residuals (Fluor Hanford 2005c). Each of these components would require remote operations to disconnect and isolate the traps from process system piping, to cap or blind off inlets and outlets, and to remove them from the facility. Isolation and removal of these components is a major activity that must be completed before other D&D activities can occur.

The current plan is to leave the sodium residuals frozen in the traps until after removal and to transport the traps to an interim storage facility (Fluor Hanford 2005c). Due to the inventory uncertainty, it was assumed for analysis purposes that the two vapor traps (U-527 and U-532) would also include their respective filter vapor traps. Two alternatives were analyzed for treatment of these RH-SCs. The first alternative is treatment at INL. The *Final Environmental Assessment for the Remote-Handled Waste Disposition Project* (DOE 2009b), issued February 2009, analyzed four alternatives for the treatment of INL and FFTF RH-SC waste streams, ranging from use of existing facilities at INTEC (the Preferred Alternative) to use of new facilities at the MFC (EA Alternative 4). DOE issued a FONSI (DOE 2009a) on February 18, 2009, selecting Alternative 1: INTEC Existing Facilities Alternative (the Preferred Alternative). The second alternative is to treat these components at a new facility constructed at Hanford. This new facility would be designed and constructed to be the same as the INL facility.

There is currently no NRC-licensed transportation cask with the capacity to handle these traps for shipment to INL. Therefore, this EIS assumes that a transportation cask or other shielded container would exist at the time of removal to transport the RH-SCs to an interim storage facility either at Hanford or at INL. Vehicle emissions for transport of the RH-SCs on site at Hanford or off site to INL were calculated and included in this EIS (SAIC 2010d).

Disposal of the decontaminated RH-SCs would be handled in a number of ways. Under the Hanford Option, the RH-SCs would be disposed of in an IDF at Hanford. However, under the Idaho Option, the decontaminated RH-SCs would be either packaged and sent to Hanford for disposal in an IDF or packaged and sent to NNSS for disposal. For analysis purposes, it was assumed that, under both options (i.e., Hanford or Idaho Options), disposal of the decontaminated RH-SCs would take place at Hanford in an IDF or at NNSS.

The radionuclide inventory of the FFTF RH-SCs is summarized in Table E-19.

**Table E-19. Fast Flux Test Facility Remote-Handled Special Component Inventory Summary**

Component	Sodium Residuals Volume (gallons)	Contact Dose Rate <sup>a</sup> (rem per hour)	Inventory <sup>a</sup> (curies)	Isotope
Cesium Trap (N-3)	80	60	210	cesium-137
Cold Trap (N-5)	710	10	470	cesium-137
			70	cobalt-60
5-scfm Vapor Trap (U-527) and Filter Vapor Trap (VT-63 and VT-64)	<1	5	90	cesium-137
1-scfm Vapor Trap (U-532) and Filter Vapor Trap (VT-61 and VT-62)	<1	0.6	5	cesium-137

<sup>a</sup> The contact dose rates and inventories are assumed or estimated. The isotope inventory is based on the assumed/estimated contact dose rate (i.e., an inventory of either 94 curies of cesium-137 or 14 curies of cobalt-60 would result in the estimated 2-rem-per-hour contact dose rate for the cold trap). Date of estimate was 2005.

**Note:** To convert gallons to liters, multiply by 3.7854.

**Key:** scfm=standard cubic feet per minute.

**Source:** CEES 2006e; DOE 2009b.

Currently, no facility exists without modification within the DOE complex for handling or treating the RH-SCs. In February 2009, under the 2009 FONSI (DOE 2009a), DOE selected the Preferred Alternative of decontamination, as necessary, and modifications in either the NWCF (CPP-659) or the FDP within the FAST Facility (CPP 666) to process the RH-SCs if DOE decides through *TC & WM EIS* decisionmaking that processing of the FFTF RH-SCs should take place at INL. Therefore, this EIS analyzes the following two options under FFTF Decommissioning Alternatives 2 and 3 for processing the RH-SCs:

- Hanford Option: Removal and storage of the RH-SCs on site at Hanford until a new facility is permitted and built, followed by disposal in an IDF at Hanford
- Idaho Option: Removal and shipment of the RH-SCs to INL for treatment in either the NWCF or the FDP within the FAST Facility, both of which are at INTEC, followed by shipment to and disposal in an IDF at Hanford or NNSS

The *Idaho National Laboratory, Conceptual Design Report for the Remote Treatment Project, Annex to the Hot Fuel Examination Facility* (ANL-W 2004) describes a workable concept, an annex to the Hot Fuel Examination Facility (HFEF) at the MFC, for processing RH waste from INL and for handling and shipping the output waste streams. This concept has been applied to both INL and Hanford for the analyses described in this *TC & WM EIS*. Specifically, for the Idaho Option, it was determined that the *Final Environmental Assessment for the Remote-Handled Waste Disposition Project* (DOE 2009b) provides adequate NEPA coverage for construction activities at INTEC and, therefore, this EIS analyzes only the environmental impacts of operation and deactivation of activities similar to those originally planned for the RTP in either the NWCF or the FDP within the FAST Facility. It was also assumed that the environmental impacts of processing the FFTF RH-SCs at INTEC and deactivating the facility would be equal to or bounded by the impacts of the same activities at the MFC. For the Hanford Option, this EIS analyzed the environmental impacts of all three activities (construction, operations, and deactivation) at the RTP, which would be located in the Hanford 200-West Area near the T Plant complex.

At Hanford, the RTP would have the same design as the INL facilities, except that a new high-bay cask-unloading area would be required at Hanford. The INL operation would use hot cells at either the NWCF or the FAST Facility for cask unloading. Thus, equipment and resource utilization (e.g., structural steel) at the Hanford RTP would be higher than that at INL because of the new construction required at Hanford (SAIC 2010d).

#### **E.2.4.4.2 Remote Treatment Project Process and Facility Descriptions**

At INL, either the hot cells located in the NWCF or the FDP cell located in the FAST Facility would be used to perform the necessary processing actions. Modifications to the NWCF cells would involve several hot cells, including Cell 308 and Cell 306. Modifications to the FDP cell would include upgrading to support the necessary remote-handling equipment. Additional details for the necessary modifications are included in the *Final Environmental Assessment for the Remote-Handled Waste Disposition Project* (DOE 2009b).

A description of the proposed remote treatment process for either of the two selected INTEC facilities is included in the *Final Environmental Assessment for the Remote-Handled Waste Disposition Project* (DOE 2009b) and is not repeated here. The following is a brief description of the RTP and the process that would be constructed in the 200-West Area at Hanford.

##### **E.2.4.4.2.1 Process Flow Description**

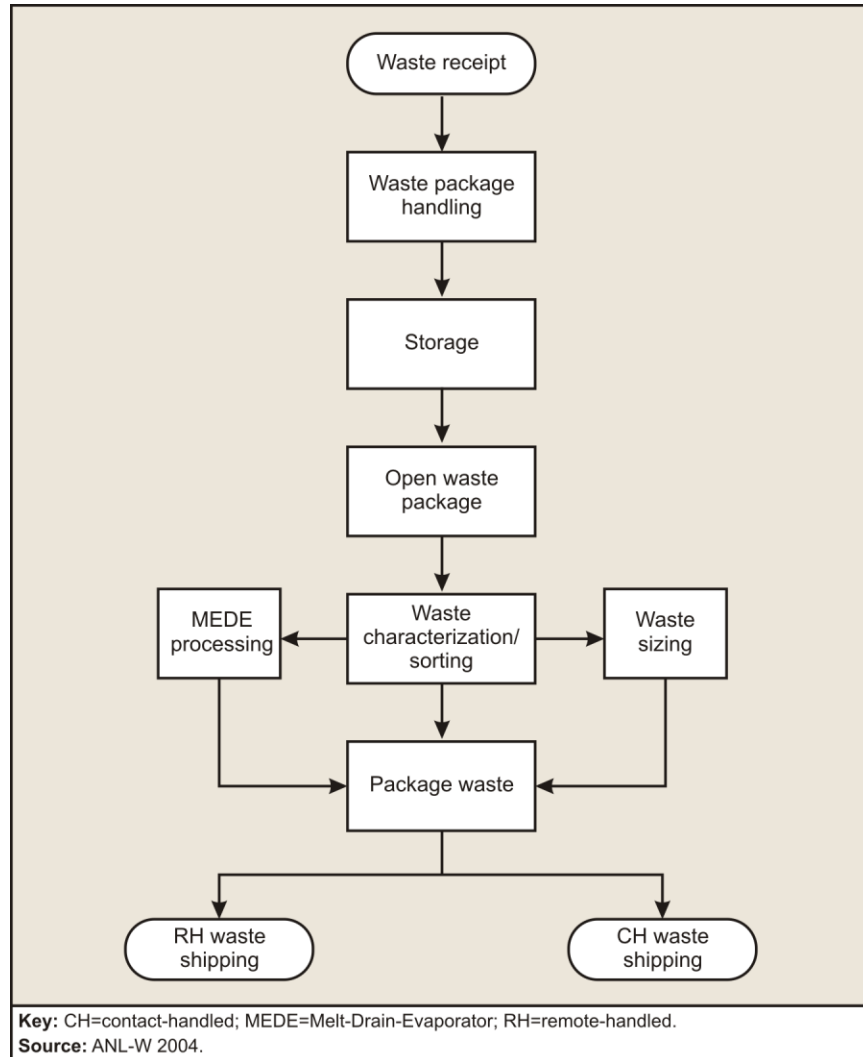
Waste would always enter the processing cell in some type of packaging. The first task would be to open the package and extract the waste. CH debris created during disassembly would be placed into 208-liter (55-gallon) drums, which would then be placed into standard waste boxes for transport and disposal at an appropriate CH disposal facility, depending on the character of the waste. Any RH debris that is generated would be transferred to the RH-waste processing area, sorted at the waste sorter station, and size-reduced so that it could be packaged for removal and disposal. Technicians working with remote manipulators at the hot cell windows would perform visual segregation and characterization. Samples would be extracted and sent to the Analytical Laboratory for quantitative analysis. Figure E-50 shows a simplified waste processing flow diagram.

Other processes would involve use of specialized handling equipment to open specific types of waste containers. There also would be a need to remove sodium from waste components in much of the waste. A melt-and-drain station would be provided at a window for this purpose. The separated sodium would be treated either within the RTP or at another facility. Use of an induction melter station based on an existing design is foreseen for volumetric consolidation of wastes that require deep geologic disposition. Other waste handling and treatment equipment would be installed in the cell as needed for more-specific or future waste processing campaigns.

Waste package handling would occur when waste shipments are received by over-the-road trucks with trailers carrying shielded casks or waste containers. The over-the-road truck and trailer would be backed into the truck lock. The payload would be accessible to a high-bay overhead bridge crane. After proper dismantling of any impact limiters and other protection devices on the payload, it would be lifted off the trailer and placed upright onto a cask cart in the cask tunnel.

A detailed process flow diagram is provided in Appendix C of the *Idaho National Laboratory, Conceptual Design Report for the Remote Treatment Project, Annex to the Hot Fuel Examination Facility* (ANL-W 2004). After the cask was loaded onto the cask cart, a shield ring or bagging ring would be placed on top of the cask. The cart would be moved along its rails into the cask tunnel. Once in the tunnel, the cask cart would be positioned under a cask penetration leading up into the waste-processing cell. The bottom side of the cask penetration in the cask room would have an integral bagging ring. A polyvinyl chloride (PVC) plastic sleeve would be connected between the bagging ring on the cask top and the bagging ring on the cask penetration to provide a contamination barrier against the release of contamination when the top of the cask and the cask penetration are opened. Actuation of the cart's cask-elevating mechanism would lift the top-loaded casks and mate them to the cask penetration in the waste processing cell floor to enable the in-cell crane to extract the waste container from the cask.

After unloading, the cask would be lowered and the bagging sleeve would be heat-sealed between the two sleeve ends. The interior surfaces of the exposed cask top would be cleaned. The emptied cask would then be returned on the cart to the truck lock where the cask's lid would be reinstalled. The overhead crane would lift the cask off the cask cart and place it on the transport trailer. The cask would be prepped for shipment, and the over-the-road truck and trailer with the empty cask would be moved out of the truck lock to the exterior for parking or transport to an offsite location.



**Figure E-50. Simplified Waste Processing Flow Diagram for Remote Treatment Project**

After the waste has been transferred into the waste processing cell, a variety of processing equipment would be used to store, sort, size, process, and repackage the waste. These processes are described in detail in the sections that follow.

The process for transferring processed waste out of the cell would utilize a bagging penetration located in the floor of the waste processing cell. The penetration would be similar to the cask penetration in that it would have a shield plug with a bagging ring on the underside of the penetration to allow the sealed bagging-in or -out of waste and materials.

Below the penetration would be a bag-out room, which would have an elevating mechanism, such as a scissor-lift, to support the shipping container in which the waste is loaded. This room would

accommodate packaging of an LLW 170-liter (45-gallon) inner waste canister, a 110-liter (30-gallon) sodium waste container, or a 208-liter (55-gallon) plastic drum liner into a 208-liter (55-gallon) overpack drum or a designated waste box. The penetration port in the ceiling of this room would have a bagging ring for use with PVC bags sized to fit into the overpack container. The overpack container would rest on a movable electric forklift, and a scissor-lift platform would be underneath the transfer port. The forklift and scissor-lift would have a 900-kilogram (2,000-pound) safe working load. The bottom of the PVC bag would be placed inside the overpack container.

Waste canisters would be lowered down into the overpack container using the cell bridge crane. After the bag has been sealed and cut, a lid would be placed on the overpack container, and it would be moved to the storage area to await transfer out of the facility. A motor-operated lid on the bag-out room side would cover the transfer port in the roof of the drum-bagging room when the port is not being used. The bag-out room would include a jib-mounted plastic heat-sealer that would swing out from the wall to seal the PVC bags. The drum or container would be moved out of the bag-out room and then loaded onto an over-the-road truck and trailer in a fashion similar to that for cask transfers.

#### **E.2.4.4.2.2 In-Cell Equipment and Operations**

Equipment necessary to process initial waste from the Radioactive Scrap and Waste Facility (RSWF) (i.e., liners and their contents, waste cans, and other material) would be similar to other equipment designed and long used at various INL facilities. Maintaining commonality in this fashion would enhance the effectiveness and safety of operations.

The proposed equipment includes the following:

- Liner-disassembly station
- NDA
- Waste can size-reducing device
- Sodium removal (melt-drain-evaporate [MEDE]) system
- Waste-sorting station
- Induction melter
- Melter equipment handling station
- General-purpose work tables

Equipment necessary to process the remaining RH waste is expected to be limited to sorting and size reduction; therefore, a sorting table with size reduction capability would be provided. Additional equipment would be provided based on waste treatment needs.

#### **E.2.4.4.2.3 Liner Disassembly Station**

The liner disassembly station would be capable of handling, unloading, and disassembling both the 41-centimeter-diameter (16-inch-diameter) and 61-centimeter-diameter (24-inch-diameter) RSWF liners and the waste cans. The station would consist of an elevator and a rotational drive assembly, a table with limited vertical and horizontal motions, and a multiaxis, remotely controlled robot arm that operates a cutting tool. The liner disassembly station would be located at a window.

When a liner is brought into the hot cell, it would be transferred to the liner disassembly station, where it would be placed on the elevator stand/rotational drive and clamped into place. The liner would be lowered into the pit using the disassembly station elevator assembly to align the cutting tool with the top of the liner. The liner would then be rotated and the lid of the liner would be cut free.



A pair of PaR Systems 6350 telerobotic manipulators would be near the liner disassembly station to aid in disassembly of the liners. These manipulators may be table- or wall-mounted. The manipulators would provide a 113-kilogram (250-pound) lifting capacity to aid in the automated cutting and handling of the liners. These manipulators have a 1.8-meter (6-foot) reach, are resistant to a cumulative radiation dose of  $1.0 \times 10^9$  rad, have 6 degrees of freedom, and are remotely maintainable. A variety of end-effectors would be used to help cutting tools and grippers lift and move heavy components.

The liner lid would be removed and placed in a CH waste container using the PaR telerobotic manipulators. Once the lid has been removed, and depending on the configuration, either the shield plug would be removed and set aside or a portion of the gravel would be vacuumed out of the 41-centimeter (16-inch) liner. The cutting tool would then be used to cut the top portion of the liner, which would be placed into a CH waste container. This process would continue until the inner waste storage can was exposed and could be removed. Figure E-51 shows a sketch of the liner disassembly station.

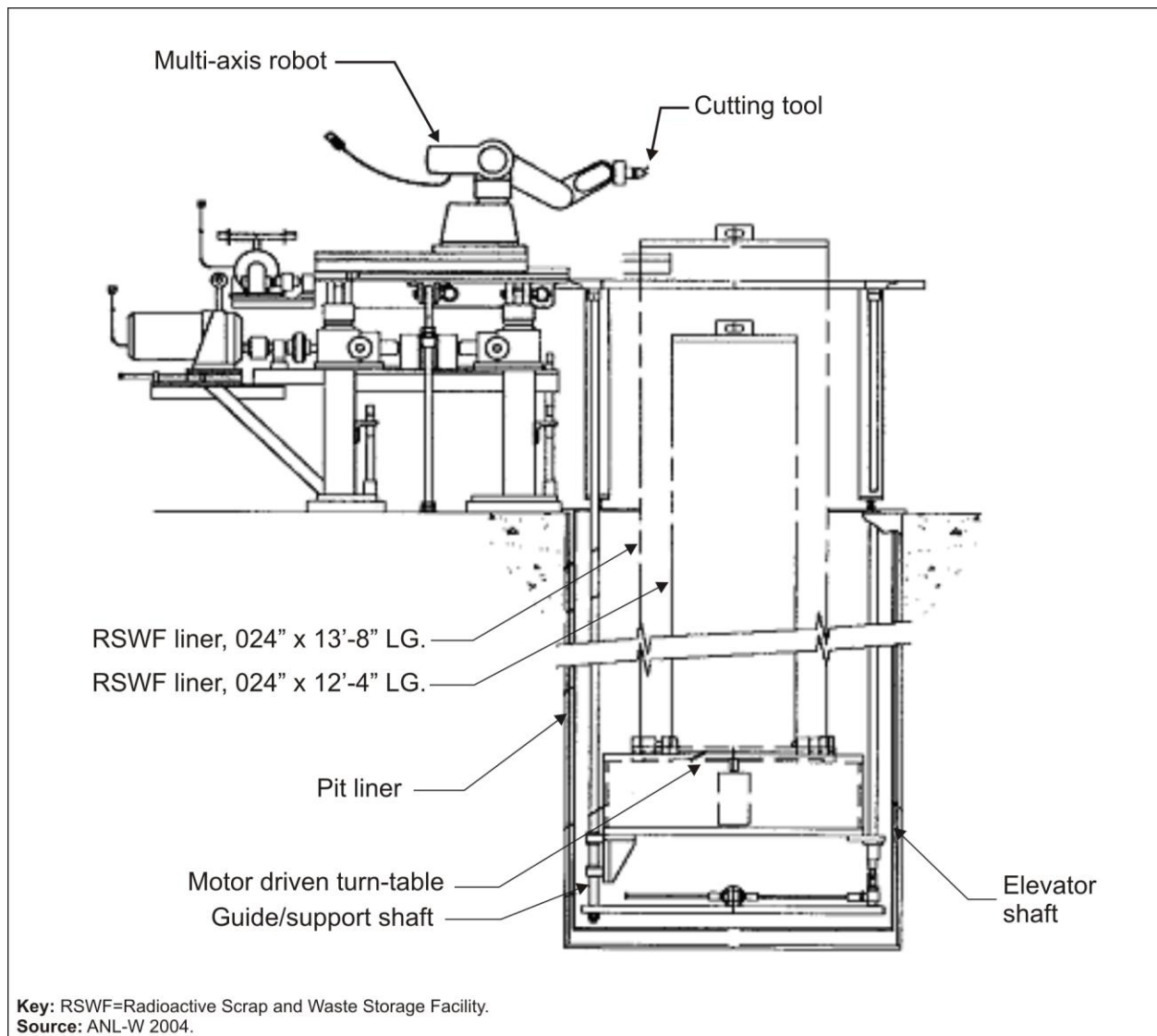


Figure E-51. Liner Disassembly Station

#### **E.2.4.4.2.4 Nondestructive Assay**

Two major NDA process categories would be needed in the RTP: item assays and package assays. Item assays would be used to quantify identifiable, separate items encountered in the repackaging process. Package assays would be used after items have been packaged for shipment. These would be primarily confirmatory measurements. Both categories would use the same basic NDA techniques, but item assays would be more accurate because uncertainties arising from the unknown geometries of packaged items would be either reduced or eliminated.

Both categories would use qualitative gamma-ray spectroscopy (for isotope identification) and quantitative gamma-ray spectroscopy, such as segmented gamma scanning, tomographic gamma scanning, and whole-item-corrected assays (to quantify isotopes whose gamma rays are detectable). Fissile isotopes are not expected to be directly detected in this manner, however.

Both categories would use passive and active neutron measurement methods to quantify fissile materials. Passive methods would count coincident neutrons primarily from plutonium-240 decay, while active methods would induce and count fission neutrons (usually with a coincidence counter) from plutonium-239 and uranium-235. Specific instruments would include the passive-active neutron counter and the shuffler (which is usually active, but can be passive also). A shielded measurement system with a passive coincidence counter may also be used.

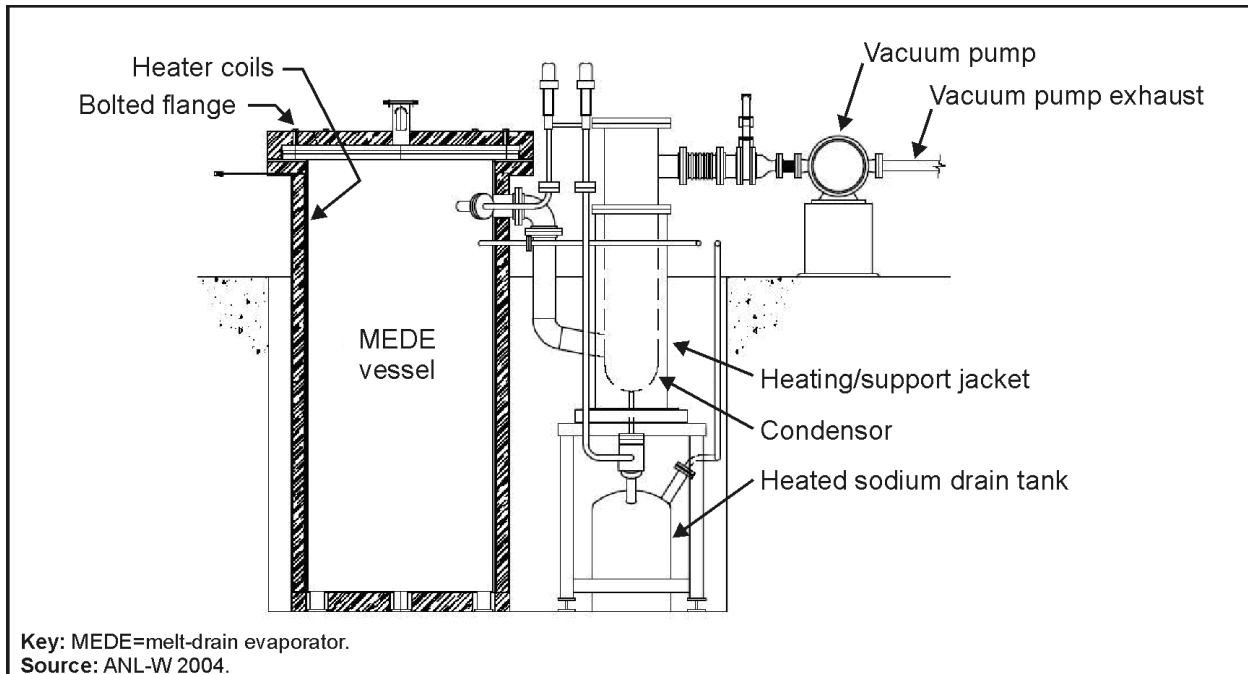
The NDA system would consist of a shielded pipe in the NDA subcell. This pipe would contain an elevator and a turntable to position and rotate the item or waste can within the counters, as well as a shield cover at the top opening at floor level in the RH room. The item or can would be placed on the elevator by the electromechanical manipulator or overhead crane; the elevator would be lowered until the shield cover could be closed, and the object would be lowered and/or rotated in the detectors. After counting, the shield cover would be opened and the object would be retrieved. The instrumentation room could not be manned during operation.

#### **E.2.4.4.2.5 Waste Can Size-Reducing Device**

The waste can size-reducing device would be located in the air cell at a window. The waste can size-reducing device would either compact the waste can tubes or cut the tubes into smaller pieces suitable for denser packing in waste containers such as 208-liter (55-gallon) drums. This machine would be used for CH waste only.

#### **E.2.4.4.2.6 Sodium Removal System**

Much of the RH waste from the RSWF is contaminated with sodium that must be removed or treated prior to shipment. The MEDE system is currently being evaluated for this purpose. The main components of the MEDE system are the vessel, heater, vacuum pump, and sodium storage tank. The MEDE would be located at a window. Figure E-52 is a sketch of the MEDE process equipment.



**Figure E-52. Melt-Drain-Evaporate Process Equipment**

By means of the MEDE process, material to be processed would be placed in an evaporation vessel where it would be heated to melt and drain the sodium. The vessel would be heated further under vacuum to remove the sodium from crevices. Tests completed at INL's MFC showed the MEDE system to be capable of removing sodium from metal samples that had 20-centimeter-deep (8-inch-deep) crevices and from a slot that was 5 centimeters (2 inches) long and only 0.01 centimeters (0.005 inches) wide. Tests completed as part of a Fermi reactor fuel-processing cost estimate at INL's MFC demonstrated greater than 99 percent removal from a fuel cladding annulus over 30 centimeters (12 inches) long with approximately a 0.02-centimeter (0.010-inch) clearance. This was noteworthy in that higher temperatures, approximately 650 °C (1,200 °F), were used to decrease the vacuum required, allowing rather simple and inexpensive commercial dry vacuum pumps to be used instead of oil-sealed or cryogenic pumps (ANL-W 2004).

The majority of the sodium and sodium-potassium present in the waste streams would be contained in bulk in a few of the stored components and could be largely removed using the melt-and-drain step. Sodium metal can be removed by the distillation cycle, but oxide or hydroxide films cannot. Therefore, the system would be provided with a moist carbon dioxide gas stream to react either sodium residual films or reaction product films to produce sodium carbonate, the desired final stable form, and the reaction product could be removed by water-washing if required. CH sodium and sodium-potassium metal removed from the waste streams would be commingled and removed from the cell and deactivated at another facility, while RH sodium and sodium-potassium would be accumulated and later remotely deactivated using equipment.

#### **E.2.4.4.2.7 Waste-Sorting Station**

The waste-sorting station would be used to disassemble waste cans and to remove, resize, and sort the waste into various waste containers. The waste-sorting station would be composed of a support-table assembly, a stand with a vertically driven waste can clamp, a rotating waste can cutting head, a sorting table, a chop saw, and storage locations for various waste containers.

When a waste can (approximately 183 centimeters [72 inches] long and 30 centimeters [12 inches] in diameter) is transferred to the waste-sorting station, it would be placed in the clamp assembly. The cutting head would be moved into position and attached to the waste can, then rotated to cut off the lid. The cut lid would be removed using the electromechanical manipulator and placed into a waste container. The electromechanical manipulator would lift the inner waste can out of the outer waste can and place it in a temporary storage compartment in the table. The outer can would be inverted using the rotating clamp, and the bottom of the can would be removed using the cutter. The can bottom would be placed into a waste container. The electromechanical manipulator would be used to pick up the waste can tube and transfer it to the waste can size-reducing device located in the CH room. The size-reducing device would compact the tube or cut it into pieces suitable for denser packing in a waste container.

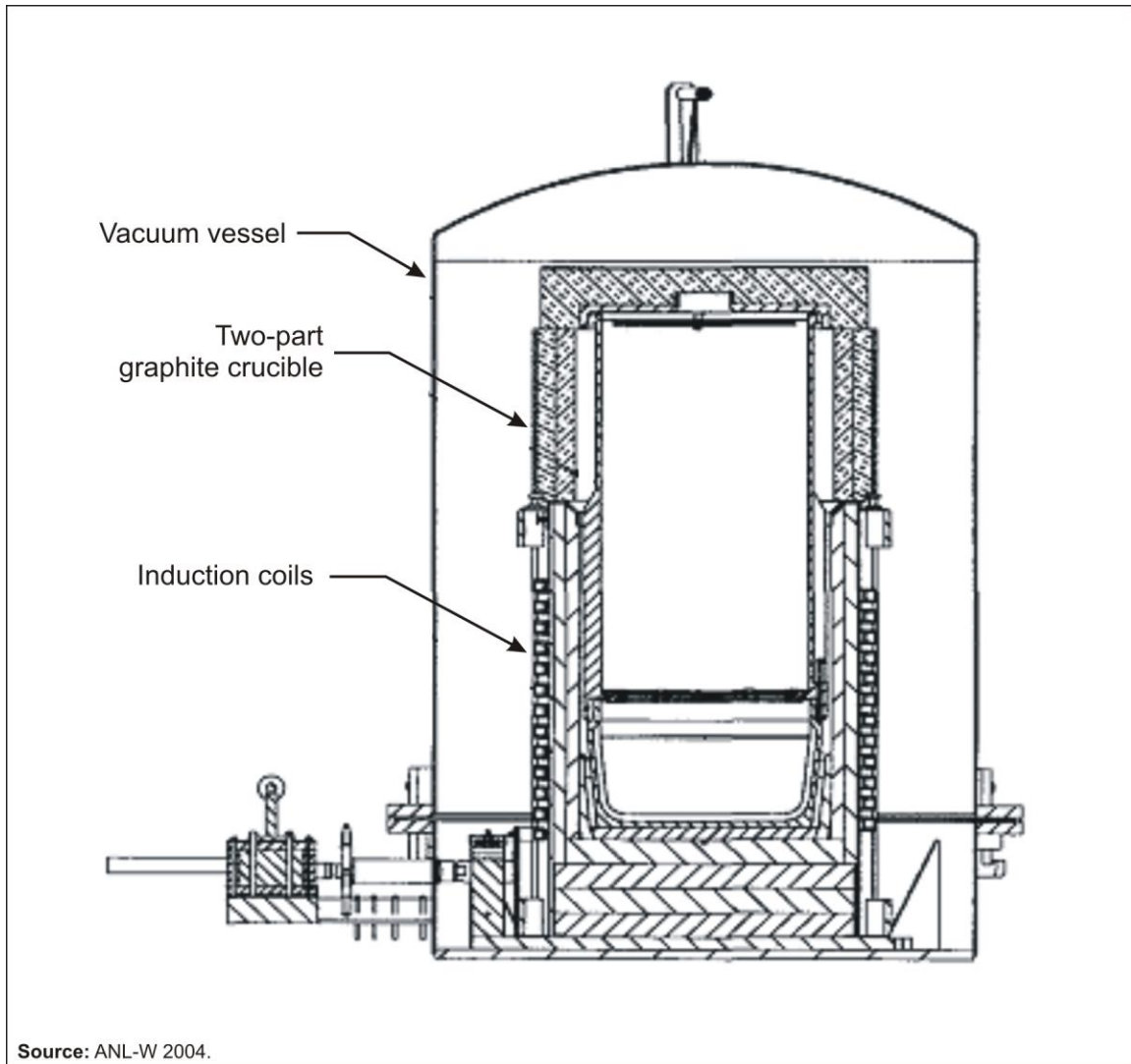
The inner waste can that was temporarily stored in an insert in the table would be lifted using the electromechanical manipulator and set into the clamp. The top lid would be either unbolted or removed using the cutter, and the lid would be placed into a waste container. The inner waste can would be lifted and rotated so that the contents would fall into the sorting tray. The inner waste can would be completely inverted, and the cutter would be used to remove the bottom of the container. The bottom of the inner waste can would be placed into a waste container. The electromechanical manipulator would be used to pick up the inner waste can tube and transfer it to the waste can size-reducing device in the CH room. The size-reducing device would either compact the tube or cut it into pieces suitable for denser packing in a waste container.

The waste in the sorting tray would be segregated into the various types of waste and placed into appropriate waste containers for each waste type. If the waste were subassembly hardware, it might be resized using the chop saw and the pieces placed into the melter crucible. Some of the waste might require additional NDA for accurate characterization. These items could be placed into a new inner waste can and transferred to the NDA cell, after which they would be returned and sorted.

#### **E.2.4.4.2.8 Induction Melter**

The induction melter is used to consolidate irradiated and contaminated metal components, including zircaloy and stainless steel. The melter consists of an insulation assembly, surrounded by a coil assembly, surrounded by the removable vacuum vessel. Graphite crucibles would be placed into and removed from the insulation assembly from the top. An insulated lid would cover the top of the crucibles. This new insulation assembly would be capable of melting waste at 1,700°C (3,092 °F). A typical crucible run would follow these steps: (1) prepare the crucible assembly; (2) load charge materials into the crucible assembly; (3) lower the crucible assembly into the melter; (4) place the lid on the crucible assembly; (5) install the vacuum vessel lid; and (6) evacuate the vessel and backfill with argon, if needed. One melter heatup-cooldown cycle would take approximately 24 hours.

If constructed at Hanford, the induction melter design would follow that of the HFEF Metal Waste Melter at INL, with the addition of an insulation extension and a modified crucible assembly to accommodate longer feed materials. Use of the induction melter would improve volumetric packaging in waste containers while avoiding the particulate contamination created by other mechanical size reduction techniques. Figure E-53 is a sketch of the induction melter.



**Figure E-53. Induction Melter**

#### **E.2.4.4.2.9 Melter-Equipment-Handling Station**

The melter-equipment-handling station would be used to prepare and load crucibles, to dump and sample the ingots, and to package the ingots into waste cans. The station would consist of a modular worktable, including a crucible-dumping/cleaning fixture; sampling equipment; a crucible-loading station; a balance; and ingot, crucible, and tool storage locations. The processed ingots would typically weigh approximately 100 kilograms (220 pounds) based on a high-void-fraction charge.

After completion of the melt-processing, the crucible containing the ingot would be removed from the melter and transferred to the melter-equipment-handling station. The crucible would be placed in a crucible inverter, and the top portion of the crucible would be removed and set down on the table. The bottom portion of the crucible would be clamped into the inverter. An ingot catch-plate would be attached to the inverter, and the crucible and ingot would be inverted. Pins would be removed, and the inverter with the bottom crucible would be lifted off the inverted ingot and catch-plate. The ingot would be weighed and then placed in a waste can using a tong-type lifting device and the electromechanical manipulator.

#### **E.2.4.4.2.10 General-Purpose Worktables**

The worktable would be used to perform equipment repairs using the remote manipulators and for general tasks. The worktable would have a scale for weighing containers and equipment for resizing hardware. The tables would be made of structural steel tubing and would have removable lids covering insert locations used to store tools and small items of equipment.

#### **E.2.4.4.2.11 High-Efficiency Particulate Air Filter Processing**

A HEPA-filter-preparation station would be used to separate the filter media from the filter frames. The radioactively contaminated HEPA filters are approximately 610 millimeters long by 610 millimeters wide by either 15 centimeters or 30 centimeters thick (24 inches long by 24 inches wide by either 6 inches or 12 inches thick). The frames are either steel or wood, and the media consist of aluminum separators and fiberglass paper. The HEPA-filter-processing equipment would consist of a staging area table, a punch press and die assembly, a size-reducing device, and a receiver-can-positioning assembly. The removed filter media would be size-reduced and transferred to additional equipment that would transform the filter media into a powder and mix it with concrete in 208-liter (55-gallon) rigid-plastic drum liners. This additional equipment would consist of a melter, a grinding machine, a hammer mill, and a mixing station to mix the filter media powder with concrete.

The HEPA filter frames would be resized at the waste can size-reducing device if the contamination levels were sufficiently low, and the debris would be placed in the appropriate waste containers. If the contamination levels were high, additional size-reducing equipment would be required for the RH-waste processing area (ANL-W 2004).

#### **E.2.4.4.2.12 Remote Treatment Project Description**

The RTP would provide a shielded, air-atmosphere processing cell designed to ensure complete and reliable containment of radioactive particulates. In addition, the facility would provide support areas, including an administrative area, restrooms and locker rooms, a hot-equipment repair area, a loading dock, a cask-handling area, and adequate space for equipment (e.g., mechanical equipment, electrical equipment, material and waste handling/processing equipment, and backup diesel-fuel-powered generator). The RTP design also would incorporate flexibility to accommodate potential mission changes during the life of the facility.

Plans call for the facility to be a concrete and steel structure approximately 29 meters long by 22 meters wide (94 feet long by 72 feet wide) with an approximate height of 20 meters (66 feet) above grade. The facility would consist of four separate floors: the service floor (basement), the operating floor (grade level), the utility floor, and the high-bay floor. The total floor area (all four floors) would be approximately 2,600 square meters (28,000 square feet). Each of the floors is described in the following paragraphs.

The main features and equipment of the service floor would include a cask tunnel; an NDA cell; heating, ventilating, and air-conditioning supply and exhaust systems; an air compressor; and miscellaneous electrical and water systems. If constructed at Hanford, the cask tunnel would be an approximately 16.8-meter-long (55-foot-long) extension of the HFEF cask tunnel at INL and would provide access to the facility for cask-compatible RH waste.

The NDA cell port would be at the south end of the waste processing cell. This penetration would allow RH waste to be lowered into a shielded cavity where active systems such as a pulsed-neutron source could be set up. In addition, a passive system such as gamma spectroscopy or coincidence-neutron measurements could also be used to interrogate RH-waste streams. This penetration would be sealed to

maintain the cell environment and to prevent the release of contamination to the NDA cell. A data acquisition room would be located outside the NDA cell in the service-floor area.

The operating floor would house the heavily shielded waste processing cell, the loading dock, and the administrative areas. Office space for engineers and technicians would be provided in other existing, nearby facilities. An enclosed personnel passage at ground level would provide sheltered access between the truck lock and the RTP.

The utility floor would house the majority of the electrical panels and cabinets, thereby freeing up space on the operating floor around the processing cell. Restrooms and locker rooms also would be located on this floor. In addition, this floor would have a shielded personnel-access door leading to an upper gallery of the processing cell to allow hands-on repair of in-cell crane drive modules.

The high-bay floor would house the hot repair area, which would consist of a decontamination chamber, a glove wall, a remote-manipulator repair glovebox, an equipment storage area, and a horizontal car transfer station. A hatch on the roof of the waste processing cell would enable equipment and waste to be lowered directly into the cell. The glove wall would be used to repair small components, prepare samples, and conduct bench-top experiments. In addition, there would be space for staging areas, dressing rooms, and a Health Physics counting room.

A diesel generator for standby electric power would be housed apart from the main structure in a lightweight pre-engineered enclosure. The foundation and floor for the diesel generator building would be totally separate from that of the main structure.

A security vulnerability would be eliminated from the storage tank relocation by locating the fuel oil unloading station outside of the security interest area. The pumping station would be located in a new, small building outside of the fenced complex, meaning that fuel trucks would not have to enter the main laboratory area. Fuel oil would be piped from the unloading station to the new location of the fuel oil storage tanks. The access area around the pumping station would be paved and sized to accommodate the maneuvering of a semitrailer truck with another tanker trailer (ANL-W 2004).

### **E.3 WASTE MANAGEMENT**

#### **E.3.1 Current Hanford Site Solid Waste Operations Complex**

Each current Solid Waste Operations Complex (SWOC) facility performs duties to achieve Hanford's waste management goals. These duties are generally complementary, and each facility contributes to the overall process. However, some processes and activities are performed at more than one facility, either because it is necessary or because it maximizes flexibility and project efficiency. The primary processes for each facility include receipt, staging, storage, repackaging, treatment, and shipment of waste, all of which must comply with the waste acceptance criteria. By facility, those activities are as follows:

- **LLBGs:** Retrieve suspect TRU waste, LLW, and MLLW; dispose of LLW and MLLW; and store certain RH wastes.
- **CWC:** Provide interim storage of LLW, MLLW, TRU waste, mixed TRU waste, and certain RH wastes (limited treatment and drum processing is planned for the CWC).
- **T Plant:** Clean out stored waste and contaminated equipment; store waste; decontaminate equipment; and verify, sample, treat, and repack LLW, MLLW, TRU waste, and RH wastes.
- **WRAP:** Confirm, repackage, certify, sample, assay, perform NDE of, and treat LLW, MLLW, and TRU waste for disposal.

The overall SWOC process flow is based on waste type, i.e., LLW or TRU waste. The process flow follows each waste type through generation, storage, treatment, and disposal. LLW or MLLW can be generated either on or off site. Once generated, the waste can be staged or stored at the CWC, LLBG, or T Plant until it is treated, analyzed, or directly disposed of on site at the LLBGs or off site at a compliant facility. If the waste requires treatment, it usually is performed within an SWOC facility or at an offsite facility. Once the waste is treated, it is staged or stored at an SWOC facility until disposal can be arranged.

TRU waste, mixed CH-TRU waste, and RH-TRU waste can be generated either on or off site. TRU waste can be either staged or stored within an SWOC facility until it is treated, or it can be sent directly from generation to treatment. Once the waste is treated, it can be disposed of at WIPP if it meets the WIPP Waste Acceptance Criteria (DOE 2002a), or it can be stored at an SWOC facility until disposal can be arranged (Weidert 2003).

### **E.3.1.1 Low-Level Radioactive Waste Burial Grounds**

The operations at the LLBGs involve primarily LLW and MLLW disposal, TRU waste retrieval, and special waste considerations. Special waste considerations include management of containers with RH-waste and encasement of Category 3 waste in high-integrity containers. RH-waste operations include emplacement in buried, specially designed concrete or steel vaults or caissons.

LLW is received from on- or offsite generators for disposal. TRU waste retrieval operations occur within the LLBG sites, and waste containers are transferred to the CWC, T Plant, WRAP, or other LLBG sites.

Currently, LLW and MLLW are sent to RCRA-compliant trenches in LLBG 218-W-5 (trenches 31 and 34) or the Environmental Restoration Disposal Facility, and reactor compartments are disposed of in LLBG 218-E-12B (trench 94). The activities at these trenches, whether for LLW or MLLW, involve several common steps, as follows:

- Waste transfer to a disposal trench area
- Waste receipt
- Container handling
- Inspection and survey
- Staging and disposal
- Trench construction, backfilling, and capping
- Stabilization and grouting
- Waste treatment

TRU waste retrieval activities are currently ongoing. These activities involve uncovering and moving the waste containers that were retrievably stored in the following locations:

- LLBG 218-W-4C: trenches 4C-T01, 4C-T04, 4C-T07, 4C-T20, 4C-T24 (EBR-II casks), and 4C-T29
- LLBG 218-W-4B: trenches 4B-TV7, 4B-T07, and 4B-T11
- LLBG 218-E-12B: trenches 12B-T17 and 12B-T27
- LLBG 218-W-3A: trenches 3A-T01, 3A-T04, 3A-T05, 3A-T06, 3A-T08, 3A-T10, 3A-T15, 3A-T17, 3A-T23, 3A-T30, 3A-T32, 3A-T34, 3A-T6S, and 3A-T9S



For suspect TRU waste retrieval, the activities at these trenches include the following:

- Trench backfill removal (where applicable)
- Container handling, including drum venting
- Inspection and survey
- Decontamination of spills (when applicable)
- NDEs and NDAs
- Waste transfers within the LLBGs or to other SWOC facilities
- Container venting
- Container placement in specially designed concrete or steel vaults or caissons
- Soil sampling and placement

Additional activities include continued management of the material currently exposed in LLBG 218-W-4C, trenches 4C-T01, 4C-T04, 4C-T07, 4C-T20, 4C-T24, and 4C-T29. Trench 4C-T24 has vaults for EBR-II cask storage and is used for staging waste containers. It is not backfilled except around the vault sides.

Preliminary site investigations are conducted in the LLBG as needed to obtain in situ information regarding the current physical condition of buried TRU waste containers and to determine the status of the environmental conditions immediately surrounding the stored waste. Once the stored waste locations have been confirmed and conditions assessed, a few selected waste containers may be retrieved and characterized to provide additional information for the preliminary site investigation in preparation for the full-scale retrieval operations that would follow.

In general, most types of waste packages are received, stored, or disposed of in the same manner. Active trenches are backfilled as needed to minimize operator dose rates and exposure to waste. Backfilling a trench also minimizes the amount of waste exposed to conditions that could cause package degradation, waste handling accidents, and severe environmental conditions (fires, water intrusion, and earthquakes) (Weidert 2003).

### **E.3.1.2 Central Waste Complex**

Receiving and storing wastes are the primary activities at the CWC. This section describes the facility's process flow and activities by operational area.

The CWC provides storage and staging for waste containers that are awaiting waste processing operations at other waste management facilities. The CWC receives waste from both onsite and offsite generators. Four types of waste are processed or stored at the CWC: LLW, MLLW, TRU waste, and mixed TRU waste. The CWC would receive, as necessary, unvented containers from retrieval operations for staging prior to venting. Venting also may occur at the CWC.

Personnel receive and inspect waste packages at the Waste Receiving and Staging Area. In accordance with all applicable procedures, transport offloading operations are performed using handtrucks, forklifts, or cranes operated by qualified personnel. Packages are transferred from the offloading area to the appropriate CWC storage building or other storage area. Alternatively, waste packages may be received, inspected, and unloaded at the specific CWC building or storage area where the waste would be stored. Typical stored waste packages include a 208-liter (55-gallon) drum; a 322-liter (85-gallon) overpack; and a fiberglass-reinforced plywood, plywood, or metal box. Atypical packages include, but are not limited to, radioisotopic thermoelectric generators, vault tank filter assemblies, steel waste disposal boxes, overpacks, and pipe overpacks in 208-liter (55-gallon) drums.

Planned activities at the CWC include performing headspace gas sampling (HSGS) on containers; NDEs and NDAs using portable units to characterize container contents; intrusive sampling operations to characterize or verify contents; minimal waste treatment (encapsulating, absorbing, stabilizing, neutralizing, and venting); and packaging and repackaging (adding shielding inside containers, filling voids, removing noncompliant items, and decontaminating shipping container interiors).

The above discussion provides a general description of the CWC waste management process flow. The following paragraphs provide a brief discussion of activities by operational area.

#### **E.3.1.2.1 Waste Receiving and Staging Area**

Primary activities in the Waste Receiving and Staging Area include shipping, receiving, staging, storing, and handling closed containers of LLW, MLLW, TRU waste, and mixed TRU waste. Activities for any of these wastes may include the following:

- Receiving and shipping waste containers
- Receiving waste from onsite and offsite generators
- Transferring waste from the CWC to onsite waste management facilities (LLBG, WRAP, T Plant)
- Handling containers (drums, solid-waste boxes [SWBs], boxes, and other acceptable containers)
- Staging waste pending other CWC activities
- Inspecting and surveying waste containers and equipment
- Conducting intrusive sampling
- Decontaminating waste containers and work areas

#### **E.3.1.2.2 Mixed Waste Storage Pad**

The Mixed Waste Storage Pad is used for staging and storing wastes. Activities for any wastes can include the following:

- Handling containers (drums, SWBs, boxes, and other acceptable containers)
- Staging waste pending other CWC activities
- Inspecting and surveying waste containers and equipment
- Performing NDEs and NDAs in mobile units near loading docks
- Sampling headspace gas inside or near the area
- Treating waste
- Placing waste into overpacks, SWBs, or other acceptable containers
- Assembling payloads

#### **E.3.1.2.3 Low-Flashpoint Mixed Waste Storage Modules**

The Low-Flashpoint Mixed Waste Storage Modules are used for staging, storing, and analyzing waste. Activities for any waste can include the following:

- Handling containers (drums, boxes, and other acceptable containers)
- Staging waste pending other CWC activities
- Inspecting and surveying waste containers and equipment
- Treating waste
- Placing waste into overpacks, SWBs, or other acceptable containers

#### **E.3.1.2.4 Alkali Metal Waste Storage Modules**

The Alkali Metal Waste Storage Modules are used for staging, storing, and analyzing MLLW. Activities for any waste can include the following:

- Handling containers (drums and other acceptable containers)
- Staging waste pending other CWC activities
- Inspecting and surveying waste containers and equipment
- Treating waste
- Placing waste into overpacks, SWBs, or other acceptable containers

#### **E.3.1.2.5 Mixed Waste Storage Buildings (2402-W, 2403-W, and 2404-W Series)**

The 2402-W-, 2403-W-, and 2404-W-series buildings are used for staging, storing, and analyzing waste. Activities for any waste can include the following:

- Handling containers (drums, SWBs, boxes, and other acceptable containers)
- Staging waste pending other CWC activities
- Inspecting and surveying waste containers and equipment
- Performing NDEs and NDAs in mobile units in the building or nearby
- Sampling headspace gas inside or near the area
- Treating waste
- Placing waste containers into overpacks
- Removing long-stem filters
- Placing waste into overpacks, SWBs, or other acceptable containers
- Venting containers

#### **E.3.1.2.6 2420-W Cask Storage Pad**

The 2420-W Cask Storage Pad is used for storage of casks containing TRU waste. Most of the waste is not readily dispersible. Activities for any waste can include the following:

- Receiving and shipping waste containers
- Receiving waste from onsite and offsite generators
- Transferring waste from the CWC to other SWOC facilities
- Handling containers
- Inspecting and surveying waste containers and equipment
- Decontaminating waste containers and work areas

#### **E.3.1.2.7 South Alkali Metal Storage Modules**

The South Alkali Metal Storage Modules are used for staging, storing, and analyzing radioactively contaminated alkali metal. They also store sodium product.

#### **E.3.1.2.8 Miscellaneous Outside Storage Areas**

The miscellaneous outside storage areas are used for staging and storing waste. Activities involving any waste can include the following:

- Handling containers (drums, SWBs, boxes, and other acceptable containers)
- Staging waste pending other CWC activities
- Inspecting and surveying waste containers and equipment

- Performing NDEs and NDAs in mobile units
- Sampling headspace gas inside or near the area
- Treating waste
- Placing waste into overpacks, SWBs, or other acceptable containers
- Applying polyurea coatings (Weidert 2003)

Additional descriptions of the CWC and proposed expansion of the CWC can be found in Section E.3.2.2.

### **E.3.1.3 T Plant**

Decontamination and waste storage, treatment, repackaging, and verification are the central activities at the T Plant. The facility has unique capabilities that provide flexibility in managing waste. This section describes the facility's process flow and activities by operational area.

Before equipment or waste containers are sent to the T Plant, they are evaluated for acceptance at 221-T Canyon or the 2706-T/TA/TB Facility. If equipment or waste is sent to the T Plant that is not in accordance with facility acceptance criteria, it is returned to the customer or action is initiated to rectify the discrepant condition. Once accepted, the process flow might take several routes, based on the type of waste or contaminated equipment.

LLW (including MLLW), TRU waste (including mixed TRU waste), and contaminated equipment are typically received at the 2706-T/TA/TB Facility or at the 221-T building's railway tunnel. They may be staged in a yard or building storage area prior to processing at the 2706-T/TA/TB Facility, the 221-T railway tunnel, or the 221-T Canyon, including the head-end. MLLW that does not meet Hanford disposal requirements may be sent off site for treatment and disposal or to other SWOC facilities for storage and treatment.

TRU (including mixed TRU) waste received for HSGS is sent to the 2706-T/TA/TB Facility. After HSGS is completed, the waste is sent to the CWC or WRAP. Other TRU waste or TRU waste-contaminated equipment could be sent to the 221-T railway tunnel, the 221-T Canyon, or the 2706-T/TA/TB Facility. It could be staged in a yard or building storage area prior to processing at the 2706-T/TA/TB Facility, the 221-T railway tunnel, or the 221-T Canyon. The primary access for transferring waste containers into the 221-T Canyon is through the 221-T head-end. The head-end is used as an interim waste container staging area for moving containers into and out of the canyon. The T Plant currently contains RH-TRU waste received from offsite sources. In the future, it is possible that other RH waste would also come to the T Plant.

The T Plant treated sludge by solidification near sections 7, 8, and 9 of the canyon deck area. The sludge/water mixture was evacuated from a large-diameter container overpack and conveyed to the sludge grouting system where the mixture was fed into 208-liter (55-gallon) drums, and blended with water, Portland cement, and bentonite clay to create grouted sludge.

The sludge transfer system fluidizes the sludge material and then transfers it to the buffer tank. The amount of sludge added to each drum is based on a correlation between the gamma monitor reading and the volume of sludge with that reading, which, when grouted, results in a product with a contact dose rate of less than 200 millirem per hour. Efforts are made to minimize the number of packages produced while controlling the process at a set point below the acceptance criteria for CH packages.

To reduce worker dose during processing, the system uses shielding for each drum that allows access to the drum for dose measurements, smears, and easy drum removal.

The system design incorporates a grout-mixing paddle that is lowered into the drum and a cover plate that seals the top to prevent sludge from splashing outside the drum during sludge loading, grout addition, and

blending. It also incorporates a process for transferring a specified volume of slurry into the drum. Water is added to the sludge, then grout is added as the mixture is blended. After the cement/clay mixture is added and the mixture is blended, the mixing paddle is dropped into the mixture in the drum. The drum is moved along the conveyor to install the vented lid. The final conveyor location provides a scale for weighing the drum. Measurement of the drum dose rate is required during this process to ensure that contact dose rates are not exceeded. The final grouted drum of sludge-type material is then ready for disposal as either LLW or TRU waste.

It is anticipated that additional sludge-type material, i.e., K Basin sand filter media and sludge, could be received at the T Plant for treatment using the described solidification process. Any waste received for treatment must meet the T Plant waste acceptance criteria and must not exceed the source strength control limit of 82.5 dose-equivalent curies for a single container.

#### **E.3.1.3.1 221-T Railway Tunnel**

The 221-T railway tunnel provides access for waste received at the 221-T Canyon. Current and planned tunnel activities include the following:

- Receiving and shipping LLW and TRU waste containers and equipment
- Receiving large-diameter container overpacks in shipping casks
- Handling LLW and TRU waste drums, boxes, and sludge casks
- Inspecting and surveying LLW and TRU waste drums, boxes, equipment, and sludge casks
- Staging, storing, and venting LLW and TRU waste drums, boxes, and sludge casks
- Conducting intrusive sampling of LLW and TRU waste drums and boxes
- Packaging and repackaging LLW and TRU waste drums, boxes, and equipment
- Performing routine decontamination of residual waste
- Performing facility repair and process equipment maintenance
- Treating LLW and TRU waste and equipment

#### **E.3.1.3.2 221-T Canyon**

The 221-T Canyon is a reinforced-concrete building used for processing and storage. Current and potential activities include the following:

- Handling LLW and TRU waste drums, boxes, sludge casks, and RH-TRU waste
- Inspecting and surveying LLW and TRU waste drums, boxes, equipment, sludge casks, and RH-TRU waste
- Staging, storing, and venting LLW and TRU waste containers, including MLLW, equipment, sludge casks, and RH-TRU waste
- Filling voids and stabilizing, neutralizing, grouting, and removing noncompliant items
- Sampling headspace gas
- Performing mobile NDAs and NDEs
- Conducting intrusive sampling of LLW and TRU waste containers and equipment, including MLLW
- Reducing size of LLW and TRU waste, including MLLW and equipment

- Treating LLW and TRU waste and equipment
- Packaging and repackaging LLW and TRU waste drums, boxes, and equipment
- Decontaminating large equipment
- Performing routine decontamination of residual waste
- Performing facility repair and process equipment maintenance
- Compacting waste drums

#### **E.3.1.3.3 2706-T/TA/TB Facility**

The 2706-T/TA/TB Facility is used primarily for sampling, treating, and packaging and repackaging waste, as well as decontaminating equipment. Current and potential future activities include the following:

- Transferring LLW and TRU waste containers and equipment
- Receiving and shipping LLW and TRU waste containers and equipment
- Handling LLW and TRU waste containers and equipment
- Inspecting and surveying LLW and TRU waste containers and equipment
- Staging, storing, and venting LLW and TRU waste containers and equipment
- Sampling headspace gas
- Performing mobile NDAs and NDEs
- Conducting intrusive sampling of LLW and TRU waste containers
- Treating LLW and TRU waste containers
- Packaging and repackaging LLW and TRU waste containers and equipment
- Decontaminating equipment
- Performing routine decontamination of residual waste
- Transferring decontaminated liquids to storage tanks
- Performing facility repair and process equipment maintenance
- Compacting waste drums

#### **E.3.1.3.4 Yard Storage**

Yard storage areas are used primarily for storage and for limited packaging and repackaging activities. Current and planned activities include the following:

- Transferring LLW and TRU waste containers and equipment
- Receiving and shipping LLW and TRU waste containers and equipment
- Handling LLW and TRU waste containers and equipment, including overpacks
- Inspecting and surveying LLW and TRU waste containers and equipment
- Staging and storing LLW and TRU waste containers and equipment
- Packaging by welding LLW and TRU waste containers

#### **E.3.1.3.5 Building Storage**

Building storage areas are used primarily for storage and limited packaging and repackaging activities. Current and planned activities include the following (Weidert 2003):

- Transferring LLW and TRU waste drums, boxes, and equipment
- Receiving and shipping LLW and TRU waste drums, boxes, and equipment
- Handling LLW and TRU waste drums, boxes, and equipment, including overpacking LLW and TRU waste drums and boxes in the 214-T building
- Inspecting and surveying LLW and TRU waste drums, boxes, and equipment
- Staging and storing LLW and TRU waste drums, boxes, and equipment

Additional descriptions of the T Plant complex and proposed expansion can be found in Section E.3.2.3.

#### **E.3.1.4 Waste Receiving and Processing Facility**

The primary WRAP activities are to confirm, sample, repackage, certify, store, and treat waste for shipment to a treatment, storage, and disposal unit. WRAP contains three operational areas: the Shipping and Receiving Area, the NDE and NDA Area, and the Process Area. WRAP process flow begins with receipt of CH waste (CH-LLW, CH-MLLW, CH-TRU waste, and CH-Mixed TRU waste) containers at the loading docks. Containers (drums or boxes) of CH waste may be received from Hanford generators, the CWC, waste retrieval operations, and offsite generators. Containers are inspected and documentation is verified. Waste that meets WRAP acceptance criteria is officially received in the facility and input into the computerized management system. Waste that does not meet the criteria is returned to the sender or isolated pending management instructions.

Items are tracked, routed, and processed independently and might undergo an activity more than once. Drums or waste boxes undergo NDEs and NDAs for characterization against the waste acceptance criteria. Suspect containers or containers received for routine verification are sent to the Process Area and inspected. Waste items that do not comply with the waste acceptance criteria are removed and/or treated to achieve compliance. Waste is repacked and returned as necessary for NDE and NDA, HSGS, storage, or shipment. Containers are either staged in the Shipping and Receiving Area just prior to preparation for TRUPACT-II loading for shipment to WIPP or transferred to other SWOC facilities.

##### **E.3.1.4.1 Shipping and Receiving Area**

Primary activities in the Shipping and Receiving Area include shipping, receiving, staging, storing, and handling of closed waste containers. Activities for any waste can include the following:

- Receiving and shipping CH waste (CH-LLW, CH-MLLW, CH-TRU waste, and CH-Mixed TRU waste) containers
- Transferring waste to WRAP from onsite generators
- Transporting TRUPACT-IIs from WRAP
- Handling containers (drums, SWBs, boxes, and other acceptable containers)
- Inspecting and surveying waste containers and equipment

- Decontaminating waste containers and work areas
- Staging waste pending other WRAP activities
- Sampling headspace gas
- Venting containers
- Packaging and repackaging containers
- Decontaminating equipment
- Performing facility repair and maintenance

Drums are typically received in the Shipping and Receiving Area, transferred to a containment pallet, and placed in storage. Before entering the NDE and NDA Area, each drum is removed from the pallet and the bar code is read. Drums requiring assay are typically weighed, and the appropriate weight is recorded prior to NDA. Waste received is either newly generated waste requiring verification, waste retrieved from the LLBG, or waste that has previously been accepted and shipped to WRAP for processing or certification.

#### **E.3.1.4.2 Nondestructive Examination and Nondestructive Assay Area**

The purpose of NDE and NDA operations is to characterize waste contents using noninvasive techniques such as real-time radiography, gamma energy analysis, and imaging passive/active neutron analysis. The operations use the following analytical equipment:

- Box real-time radiography NDE
- Box integrated gamma energy analysis-imaging passive/active neutron (inactive)
- Drum real-time radiography NDE
- Drum gamma energy analysis to quantify gamma-emitting radionuclides
- Drum imaging passive/active neutron to quantify fissile contents

Activities conducted in the NDE and NDA Area include the following:

- Performing NDEs and NDAs of drum and boxed waste
- Handling containers
- Staging in-process waste drums and boxes
- Decontaminating waste containers and NDE and NDA equipment
- Performing facility repair and maintenance
- Sampling headspace gas

Waste drums and boxes received at WRAP are subjected to NDEs or NDAs to verify compliance with waste certification requirements and (in the case of drums) to determine the appropriate method for processing any restricted waste present. For example, LLW received for verification purposes might require only NDE.

Waste containers determined by NDE or NDA to meet appropriate disposal criteria are either certified and shipped to a disposal facility or volume-reduced by supercompaction, certified, and shipped for disposal. Under normal operating conditions, waste containers determined by NDE and NDA to contain restricted waste (or those requiring additional verification) are opened, sorted, sampled, and treated. The restricted waste is either processed in WRAP or repackaged for storage pending treatment at an



appropriate facility. The remaining compliant waste is repackaged, certified, and shipped to storage or disposal. Waste boxes determined by NDE or NDA to meet the disposal criteria are certified and shipped to storage or disposal. Boxed waste that does not meet the disposal requirements is shipped to storage, pending treatment at other facilities.

#### **E.3.1.4.3 Process Area**

Process enclosures in the Process Area are used to open waste containers for verification, characterization, sorting, and treatment to evaluate whether TRU waste and LLW comply with appropriate waste acceptance criteria.

Process operations occur in the following separate process enclosures (gloveboxes) in the waste processing area:

- TRU waste and LLW process enclosure
- TRU waste process enclosure
- LLW restricted-waste management (RWM) process enclosure
- TRU waste RWM process enclosure

Activities conducted to achieve compliant waste in the process areas include the following:

- Handling drums being moved to and from staging and the enclosures
- Inspecting and surveying waste in the enclosures and containers in the area
- Staging waste on carousels or lifts in association with processing
- Sampling headspace gas
- Conducting intrusive sampling of opened drums inside the enclosures, including separation and sorting, visual examination, and sample collection
- Adding shielding to RH waste to achieve CH requirements
- Packaging and repackaging through loadout from the enclosures
- Decontaminating waste
- Treating noncompliant waste
- Performing facility repair and maintenance

Waste drums that do not meet the appropriate disposal criteria are opened and sorted in a process enclosure where the restricted waste or waste types prohibited by the applicable waste acceptance criteria are removed or processed. Restricted and prohibited wastes that are not certifiable or are noncompliant are either processed in WRAP or returned to the generator for resolution. Two process lines perform these waste processing operations: one is dedicated to TRU waste; the other can process either TRU waste or LLW. Waste drums may be taken to a process glovebox for verification purposes.

In the sorting process, specific hazardous materials and other regulated waste (e.g., lead bricks, PCBs) may be removed from the drum for processing in the RWM enclosures. Restricted-waste items are segregated and placed in containers in the waste process enclosures before being transferred to the RWM enclosures. One enclosure is dedicated to LLW and the other to TRU waste material to avoid the

possibility of cross-contamination. Samples of restricted-waste items are collected in the RWM enclosures for analysis elsewhere. The enclosures also provide limited restricted-waste processing, such as absorbing small quantities of liquids, immobilizing particulates, and depressurizing aerosol cans. After waste processing is completed, outgoing waste containers might undergo NDEs and/or NDAs for final certification before shipment.

In addition, waste containers are generated as products of the processing enclosures. There are a number of other categories of waste drums in WRAP. There are mixed or hazardous waste accumulations resulting from mixed or hazardous waste management, as well as waste drums for waste generated outside the process enclosures during operations in the WRAP Process Area. Additionally, there are some waste drums, such as pipe overpack containers, overpacked drums with long-stem Nucfil filters, and drums with added filler material—all requiring minimal processing and sampling within a confinement ventilation zone. These drums may be opened within the WRAP Process Area, but outside the glovebox, to perform this limited processing and sampling. Radiation protection controls are established, as appropriate (Weidert 2003).

#### **E.3.1.4.4 Accelerated Process Lines**

The accelerated process lines (APLs) use mobile units such as trailers or relocatable structures to house the equipment used to characterize and process candidate CH-TRU waste containers for disposal at WIPP. These units can be near or within any SWOC building and are positioned to increase TRU waste characterization and processing throughput.

APL capabilities include any or all of the following (Weidert 2003):

- NDE using real-time radiography
- NDA using the segmented gamma system, imaging passive/active neutron analysis, or other similar neutron- or gamma-counting systems
- HSGS, either by taking samples through the existing filter or filter port or by using a drum-venting system and sample analysis system combination
- Drum venting
- Gas generation testing
- Glovebox processing (not approved at this time)

Additional descriptions of WRAP and the proposed expansion can be found in Section E.3.2.4.

### E.3.2 Waste Management Alternatives and Treatment Facilities Analyzed in This Environmental Impact Statement

#### E.3.2.1 Waste Management Alternatives

This *TC & WM EIS* analyzes three Waste Management alternatives. A summary of each follows and a more detailed discussion of each treatment facility is included in Sections E.3.2.2 through E.3.2.4. Further discussions of IDF(s) and the RPPDF are included in Sections E.3.4 and E.3.5, respectively.

- **Waste Management Alternative 1: No Action.** Under this alternative, storage and treatment of LLW, MLLW, and TRU waste would continue at the CWC, and disposal would continue at the LLBG 218-W-5, trenches 31 and 34, until an estimated operational closure date of 2035. Likewise, storage and treatment of onsite LLW, MLLW, and TRU waste would continue at WRAP and the T Plant complex. No shipments of offsite LLW, MLLW, or TRU waste would be accepted, except as allowed as part of DOE's January 6, 2006, Settlement Agreement with the State of Washington (as amended on June 5, 2008) regarding *State of Washington v. Bodman* (Civil No. 2:03-cv-05018-AAM), signed by DOE, Ecology, the Washington State Attorney General's Office, and the U.S. Department of Justice. Further construction of IDF-East would be discontinued in 2008. Administrative controls would be carried on for 100 years following operational closure of the disposal trenches. Figure E-54 is a graphic representation of the Waste Management No Action Alternative.

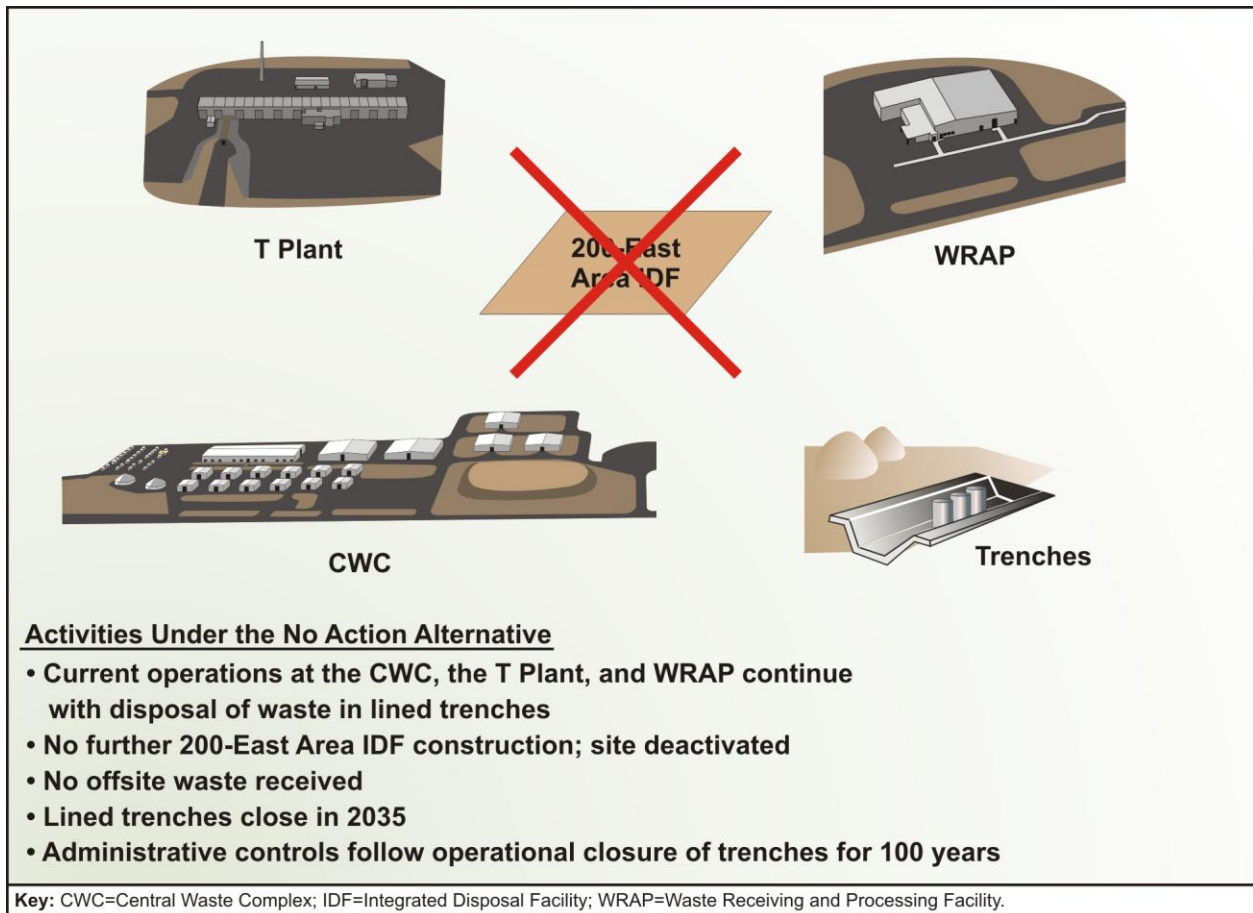
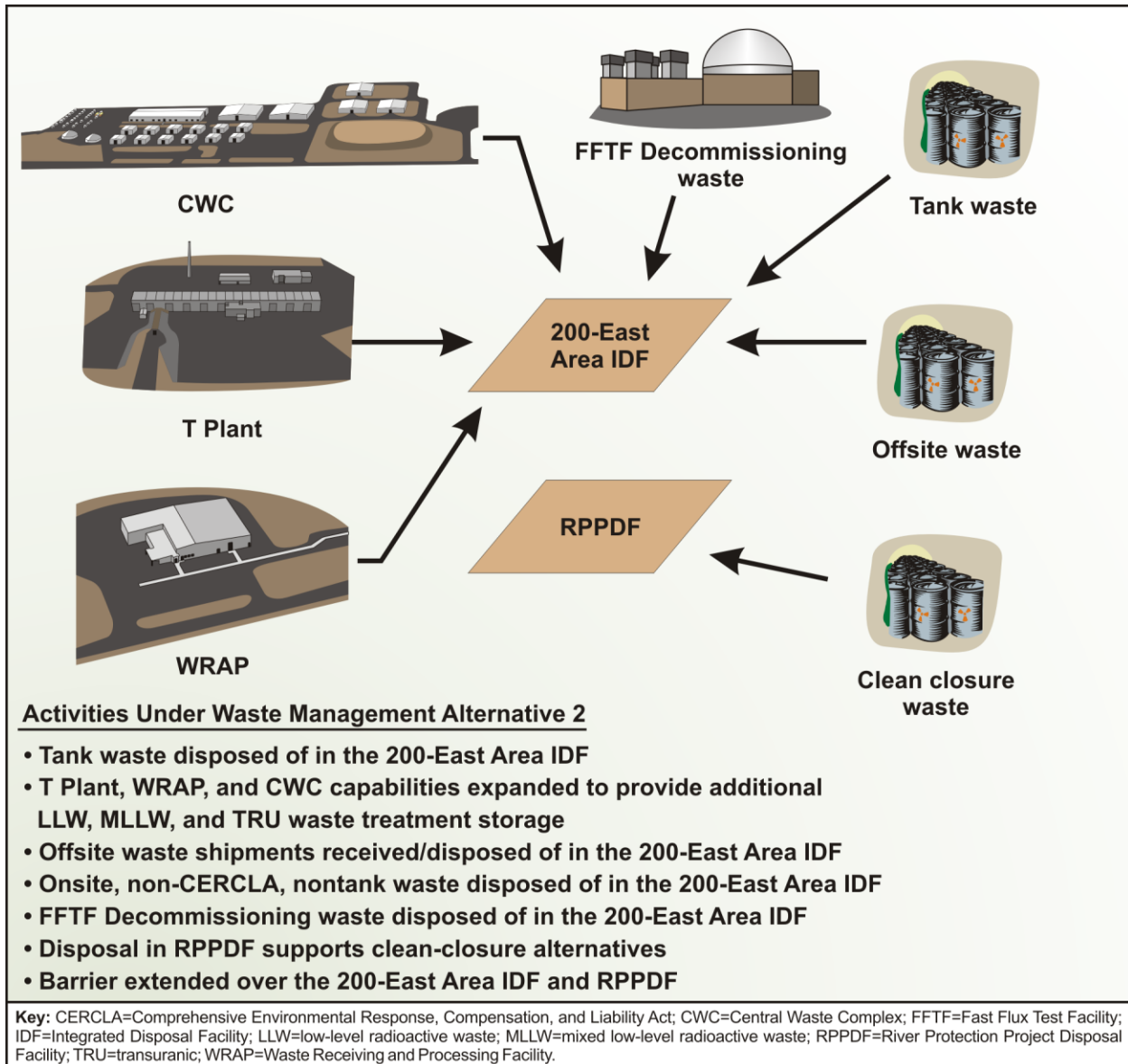


Figure E-54. Waste Management Alternative 1: No Action

- **Waste Management Alternative 2: Disposal in IDF, 200-East Area Only.** Under this alternative, storage and treatment of LLW, MLLW, and TRU waste would continue, using both the existing and the proposed expanded capabilities at the CWC, T Plant complex, and WRAP facilities. Offsite waste received would be limited to 62,000 cubic meters (81,000 cubic yards) of LLW and 20,000 cubic meters (26,000 cubic yards) of MLLW, with shipments estimated to be received from 2010 through 2046. Onsite nontank waste from non-CERCLA sources would be generated through 2035. For analysis purposes, operation of the LLBG 218-W-5, trenches 31 and 34, would continue through 2050. However, all waste streams requiring onsite disposal were assumed to be disposed of in IDF-East. Under this alternative, IDF-East, which is partially constructed in the 200-East Area, would accept the following: tank treatment and FFTF decommissioning waste; waste management facility-generated (secondary) waste; onsite non-CERCLA, nontank waste; and projected waste shipments from other DOE sites. IDF-East was assumed to be fully constructed and ready to begin operations in 2009. A new disposal facility, the RPPDF, would be constructed for disposal of lightly contaminated RSE resulting from closure of the tank farm facilities. To reduce the combinations of IDF and RPPDF configurations, three disposal groups are analyzed:
  - **Disposal Group 1:** IDF-East would have a capacity of 1.2 million cubic meters (1.57 million cubic yards) and the RPPDF, 1.08 million cubic meters (1.41 million cubic yards). Both facilities would operate through 2050. This grouping supports Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, 5, and 6C; FFTF Decommissioning Alternatives 2 and 3; and disposal of onsite non-CERCLA, nontank waste and projected waste shipments from other DOE sites.
  - **Disposal Group 2:** IDF-East would have a capacity of 425,000 cubic meters (556,000 cubic yards) and the RPPDF, 8.37 million cubic meters (10.9 million cubic yards). Both facilities would operate through 2100. This grouping supports Tank Closure Alternatives 2A and 6B; FFTF Decommissioning Alternatives 2 and 3; and disposal of onsite non-CERCLA, nontank waste and projected waste shipments from other DOE sites.
  - **Disposal Group 3:** IDF-East would have a capacity of 425,000 cubic meters (556,000 cubic yards) and the RPPDF, 8.37 million cubic meters (10.9 million cubic yards). Both facilities would operate through 2165. This grouping supports Tank Closure Alternative 6A; FFTF Decommissioning Alternatives 2 and 3; and disposal of onsite non-CERCLA, nontank waste and projected waste shipments from other DOE sites.

Figure E-55 is a graphic representation of Waste Management Alternative 2: Disposal in IDF, 200-East Area only.



**Figure E-55. Waste Management Alternative 2: Disposal in IDF, 200-East Area Only**

- **Waste Management Alternative 3: Disposal in IDF, 200-East and 200-West Areas.** Under this alternative, the activities would be the same as those under Waste Management Alternative 2, except that onsite waste disposal would be split between the current IDF-East site and a new IDF site in the 200-West Area (IDF-West). Tank waste would still be disposed of in IDF-East, but all the other waste streams disposed of in IDF-East under Waste Management Alternative 2 would be disposed of in IDF-West. The three disposal groups under Waste Management Alternative 3 are as follows:
  - **Disposal Group 1:** IDF-East would have a capacity of 1.1 million cubic meters (1.43 million cubic yards); IDF-West, 90,000 cubic meters (118,000 cubic yards); and the RPPDF, 1.08 million cubic meters (1.41 million cubic yards). All three facilities would operate through 2050. This grouping supports Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, 5, and 6C; FFTF Decommissioning Alternatives 2 and 3; and disposal of onsite non-CERCLA, nontank waste and projected waste shipments from other DOE sites.

- **Disposal Group 2:** IDF-East would have a capacity of 340,000 cubic meters (445,000 cubic yards); IDF-West, 90,000 cubic meters (118,000 cubic yards); and the RPPDF, 8.37 million cubic meters (10.9 million cubic yards). IDF-East and the RPPDF would operate through 2100. IDF-West would operate through 2050. This grouping supports Tank Closure Alternatives 2A and 6B; FFTF Decommissioning Alternatives 2 and 3; and disposal of onsite non-CERCLA, nontank waste and projected waste shipments from other DOE sites.
- **Disposal Group 3:** IDF-East would have a capacity of 340,000 cubic meters (445,000 cubic yards); IDF-West, 90,000 cubic meters (118,000 cubic yards); and the RPPDF, 8.37 million cubic meters (10.9 million cubic yards). IDF-East and the RPPDF would operate through 2165. IDF-West would operate through 2050. This grouping supports Tank Closure Alternative 6A; FFTF Decommissioning Alternatives 2 and 3; and disposal of onsite non-CERCLA, nontank waste and projected waste shipments from other DOE sites.

Figure E-56 is a graphic representation of Waste Management Alternative 3: Disposal in IDF, 200-East and 200-West Areas.

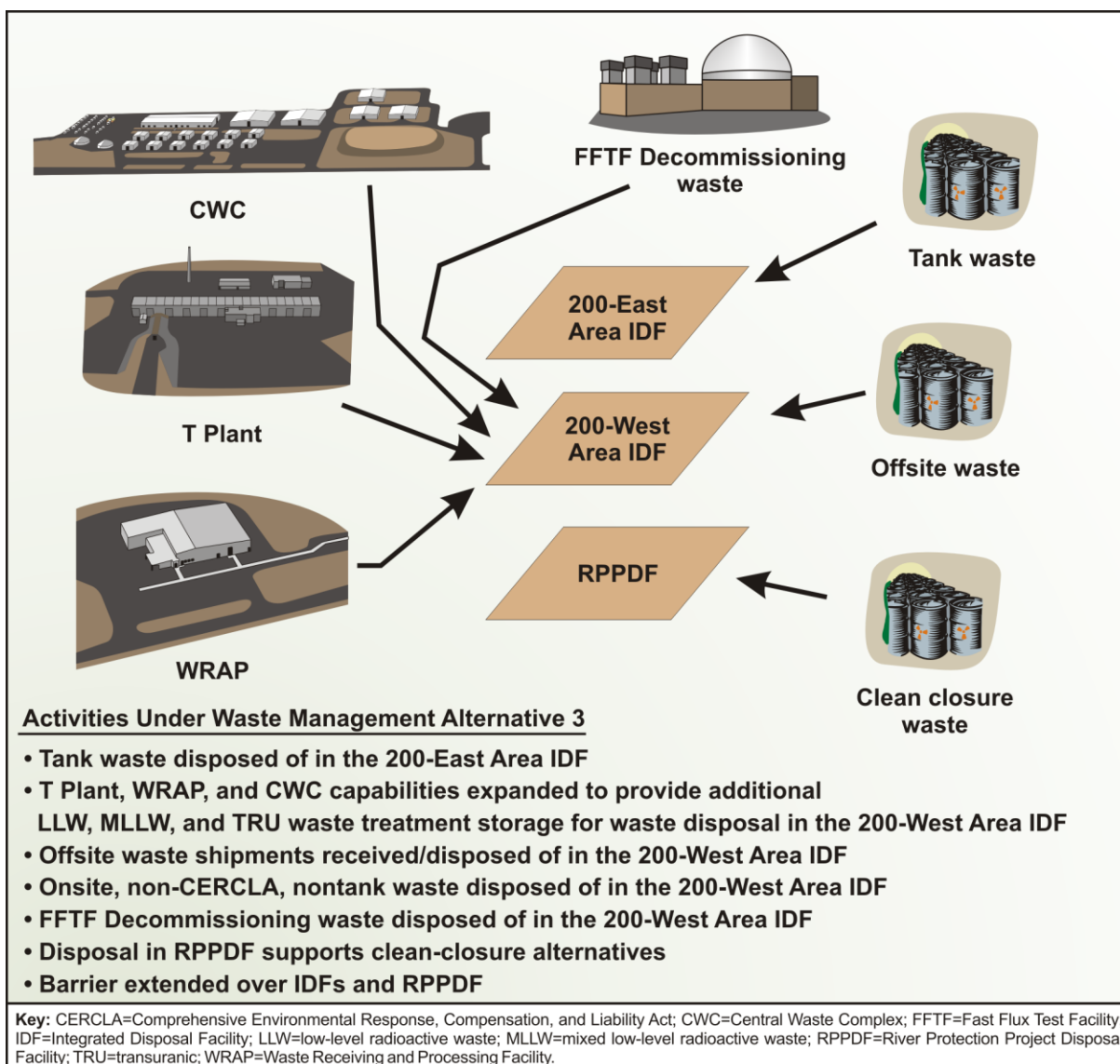


Figure E-56. Waste Management Alternative 3: Disposal in IDF, 200-East and 200-West Areas

### **E.3.2.2 Central Waste Complex**

#### **E.3.2.2.1 Description**

The CWC is designed and authorized to receive and store CH-LLW, CH-MLLW, CH-TRU waste, and CH-mixed TRU waste. Current and future activities at the CWC for these waste packages include shipment and receipt of waste containers, staging/storing of waste, nonintrusive surveying and inspection, treatment of waste, intrusive sampling, verification, packaging, repackaging, loading, HSGS, drum venting, and decontamination. The CWC main buildings are the 2401-series, 2402-series, 2403-series, and 2404-series buildings. Other CWC buildings include the Low-Flashpoint Mixed Waste Storage Modules (FS-1 through FS-27), Alkali Mixed Waste Modules (AWM-1 through AWM-4), South Alkali Metal Storage Modules, and Mixed Waste Storage Modules (MWS-1 through MWS-11). Other facilities include the Waste Receiving and Staging Area, Mixed Waste Storage Pad, and 2420-W Cask Storage Pad.

- All newly generated Hanford waste is required to meet the HSSWAC (Fluor Hanford 2005a) established by the Hanford Solid Waste Acceptance Program. Waste is generally packaged in 208-liter (55-gallon) drums or other packages of approved size, shape, and form. Each container is handled individually using a handtruck or dolly, a forklift, or a crane. Drums are placed on pallets, a maximum of four per pallet, and the pallets are stacked in three tiers. The storage buildings or pads have physical features that provide for appropriate separation between incompatible wastes. The 2403-WD building footprint is 52 meters wide by 99 meters long by 6 meters high (170 feet wide by 325 feet long by 20 feet high) for a total area of 5,150 square meters (55,400 square feet), and 31,000 cubic meters (1.1 million cubic feet) of volume.
- The 2403-WD building drum capacity is 17,500 drums.
- The total CWC drum capacity is 82,480 drums (CEES 2006f).

| Figure E-57 is a site plan of the current CWC, including WRAP.



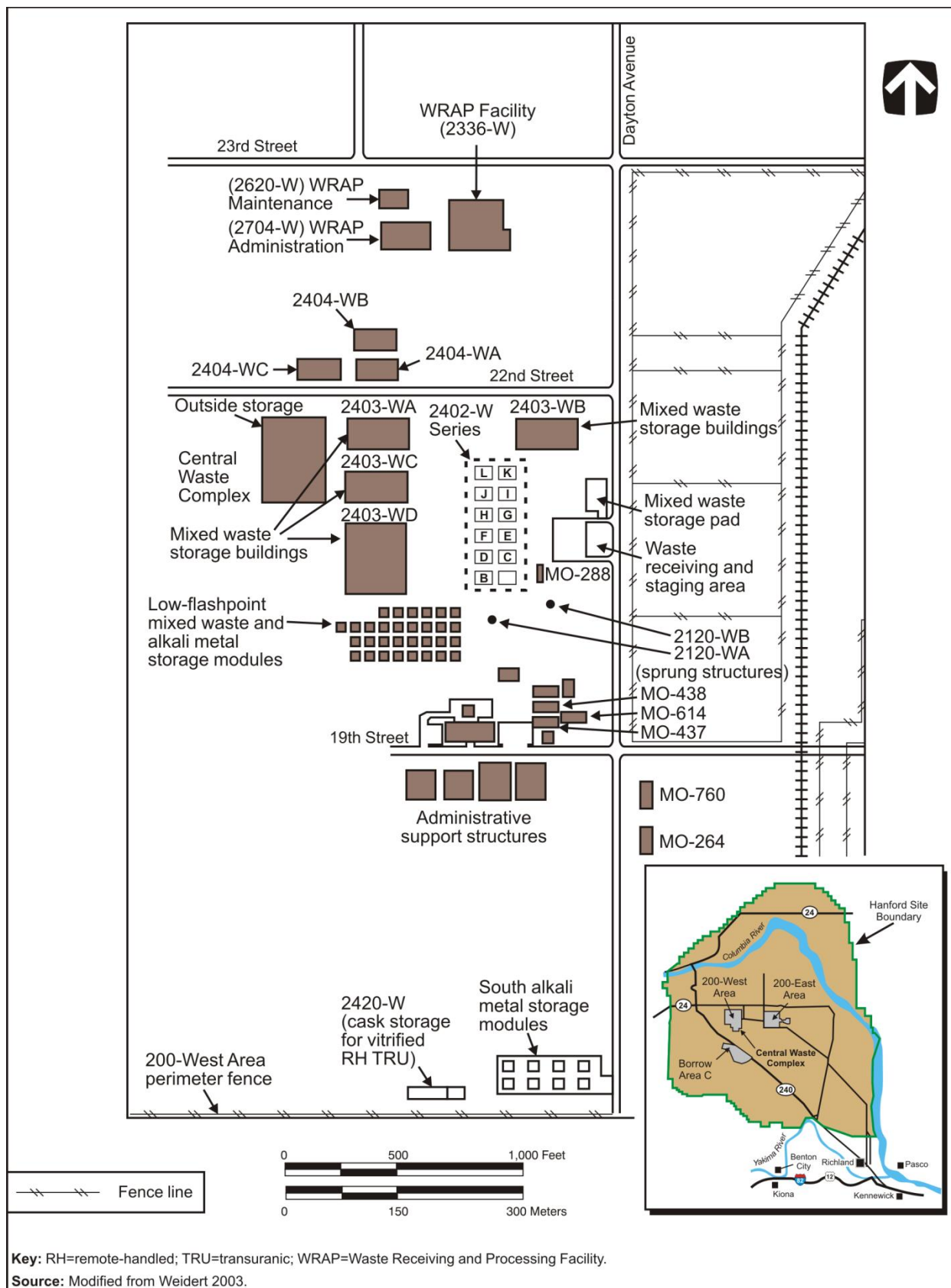


Figure E-57. Current Central Waste Complex Site Plan



#### **E.3.2.2.2 Central Waste Complex Expansion**

Due to the uncertainty of the waste forecasts for Hanford, it was assumed for analysis purposes that additional solid-waste storage capacity would be required at the CWC as soon as possible following issuance of the ROD regarding the proposed actions. This capacity would be gained by construction of another 2403-WD facility at the CWC. Thus, under Waste Management Alternatives 2 and 3, the current capabilities of the CWC would be expanded to support additional storage capacity for radioactive waste. Specifically, the current storage building 2403-WD, with a drum storage capacity of 17,500 drums, is duplicated under both Alternatives 2 and 3. Some of the more pertinent assumptions made to support the expansion are as follows:

- The CWC expansion is based on building 2403-WD and its capacity of 17,500 drums.
- Steel waste capacity for roll-on/off containers is estimated to be 13.6 metric tons (DOE 2003a).
- The medium-duty work trucks are used for training and operator rounds; trucks are used 0.5 hours per day each day of the year (182.5 hours per year).
- Ninety percent of the total operational labor hours are radiation worker labor hours; the remaining 10 percent are nonradiation worker labor hours.
- The operational resource personnel vehicle, forklifts, mechanics' trucks, crane, and light plants operate an average of 8 hours per day during normal operations. The backhoe operates an average of 1 hour per day, and the portable generators operate an average of 12 hours per day.
- Waste generated from consumables during solid-waste processing, such as gloves and tape, is incorporated into the repackaging operation without an increase to the source term, volume, or number of containers.
- Deactivation of the CWC assumes 100 percent of normal operating staffing during the deactivation period (CEES 2006f).
- Closure and postclosure care of this CWC expansion is not within the scope of Waste Management Alternatives 2 and 3 but is analyzed in Chapter 6, "Cumulative Impacts."

Waste Management Alternative 1 does not analyze the expansion of the CWC; under this alternative, the existing, permitted CWC would operate to support onsite waste.

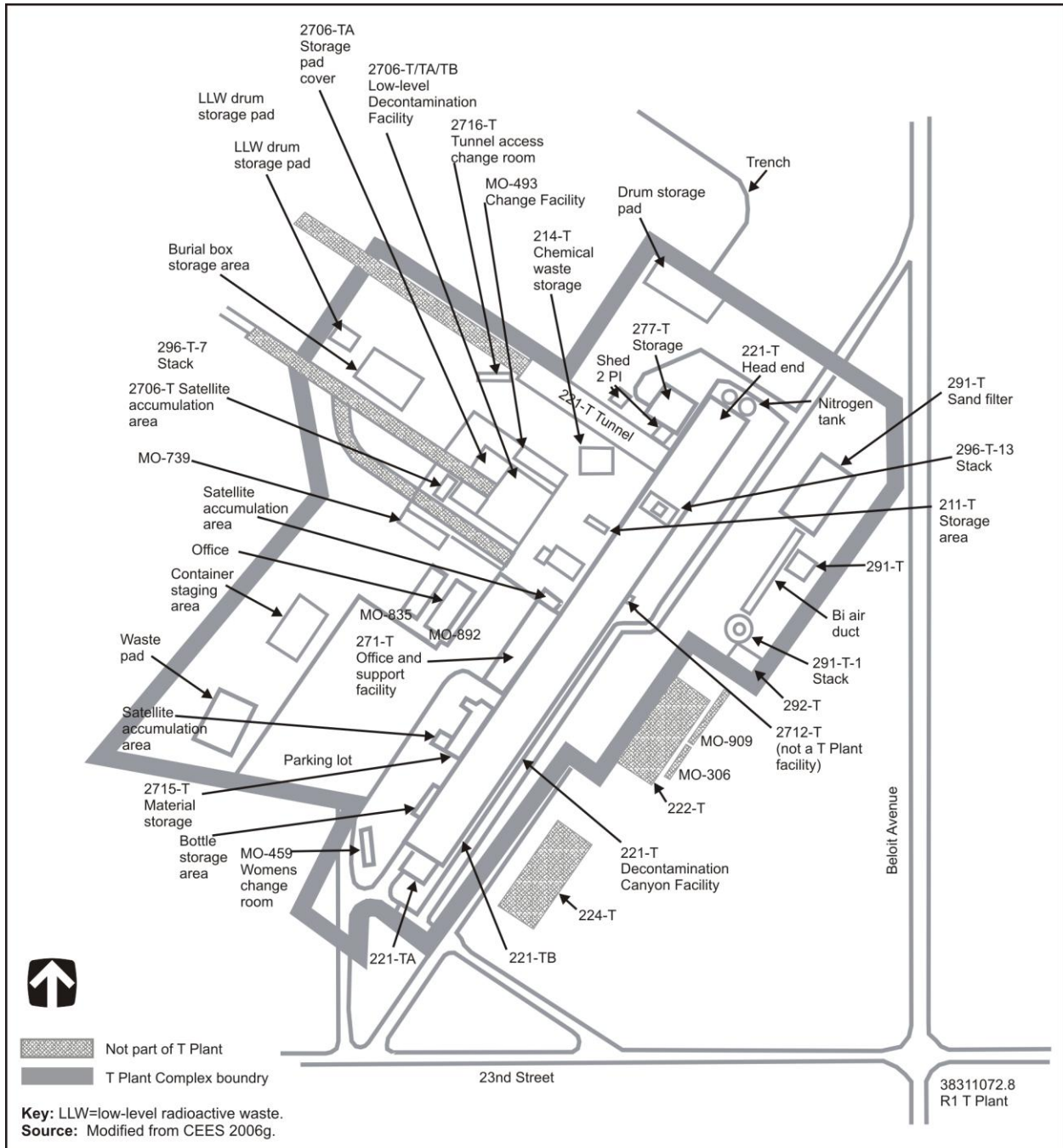
### **E.3.2.3 T Plant Complex**

#### **E.3.2.3.1 Description**

According to current estimates, there are approximately 8,000 cubic meters (10,500 cubic yards) of high-dose (i.e., RH) or oversized waste packages that may not meet the HSSWAC criteria for disposal (Johnson and Parker 2004). A new facility or facility modifications may be required to process these waste volumes. Waste processing could be accomplished through the use of new modules either within or adjacent to an existing facility, e.g., the 2706-T/TA/TB Facility, 221-T Canyon, WRAP, or mobile TRU waste processing units. The T Plant could be modified to remotely open, sort, size-reduce, and repackage MLLW and TRU/mixed TRU waste that is either CH or RH and in variously sized packages (Bailey et al. 2006). The T Plant would need to accommodate waste containers sized anywhere from 3.8-liter (1-gallon) cans to boxes measuring 6.1 meters (20 feet) long by 4 meters (13 feet) wide by 3.4 meters (11 feet) high. The waste containers could be as heavy as 37,600 kilograms (83,000 pounds), and the radiation levels could be as high as 20,000 rem per hour at the container surface. These waste types currently cannot be processed using existing site capabilities. This additional waste processing capability could be included in an expansion of either WRAP or the T Plant.

The T Plant is operating under interim status as an RCRA treatment, storage, and disposal unit with no current RCRA permit capacity limit. The T Plant includes the 221-T Canyon and 2706-T/TA/TB Facility, which conduct segregation, treatment, repackaging, verification, classification, and general venting and sampling activities on CH-waste containers (drums and boxes). The T Plant also has limited RH processing capabilities in the 221-T Canyon. T Plant solid-waste processing consists of the addition of absorbent or grout material to the waste matrix, neutralization, and amalgamation of mercury or other metals. Additional services include sampling of drum headspace to support the TRU waste program and management of analytical samples returned from commercial laboratories (CEES 2006g).

Figure E-58 is the current T Plant complex site plan.



**E.3.2.3.1.1 221-T Canyon**

The 221-T Canyon currently has the capability for some TRU waste processing, including some RH-TRU waste. The capacity of the 221-T Canyon may change over time, depending on plans for storing and treating K Basin sludge. The 221-T Canyon continues to be decontaminated, and mobile contamination is being fixed; however, the 221-T Canyon still currently requires an air-purifying respirator for manned entries.

The 221-T Canyon is part of the T Plant, which was constructed in 1944. The 221-T Canyon is a heavily shielded, reinforced-concrete structure. The overall area of the 221-T Canyon is 5,400 square meters (57,800 square feet), with a height of 22.6 meters (74 feet) (CEES 2006g).

#### **E.3.2.3.1.2 2706-T/TA/TB Facility**

The 2706-T/TA/TB Facility includes the 2706-T building, 2706-TA building addition, 2706-TB building addition, MO-433 Change Facility, and two hazardous waste storage modules. The 2706-T building, 2706-TA building addition, and 2706-TB building additions make up a single building and are referred to in this document as the “2706-T/TA/TB building.” The surrounding grounds and supporting building are referred to as the “2706-T/TA/TB Facility.” The 2706-T/TA/TB Facility is surrounded by the 2706-T yard area, measuring approximately 67 meters long by 61 meters wide (220 feet long by 200 feet wide) (CEES 2006g).

The 2706-T/TA/TB Facility currently has limited capability to process MLLW and TRU waste. Modifications may be made to the 2706-T/TA/TB Facility to allow additional stabilization, macroencapsulation, sorting, sampling, repackaging, NDE, and NDA capabilities. All wastes managed by the T Plant are required to meet the T Plant waste acceptance criteria (Fluor Hanford 2005a).

The 2706-T/TA/TB building is a pre-engineered metal building. The foundation is concrete slab on grade throughout. The overall area of the 2706-T/TA/TB building is 900 square meters (9,700 square feet). The 2706-T/TA/TB building has a slanted roof with eaves 7.6 meters (25 feet) above grade and a peak at 9.8 meters (32 feet) above grade. The average height of the 2706-T/TA/TB building is 9 meters (30 feet), giving the 2706-T/TA/TB building an overall facility volume of 8,210 cubic meters (290,000 cubic feet).

Current operations in the 2706-T/TA/TB building include waste sampling, packaging/repackaging, HSGS, and management of “problem” waste containers. To support some of these operations, greenhouses (relocatable confinement structures) are used as necessary in the 2706-T/TA/TB building. An example of a greenhouse erected in the 2706-T/TA/TB building includes two work areas separated by a common entry vestibule. Each work area has an isolatable air space (CEES 2006g).

#### **E.3.2.3.2 T Plant Expansion**

Due to the uncertainty of the Hanford waste forecasts, it was assumed for analysis purposes that additional solid-waste processing capabilities would be required as soon as possible following issuance of the ROD and would be gained by construction of another 2706-T/TA/TB-type building near the T Plant. This would allow processing of high-dose-equivalent curie or high-dose waste in the 221-T Canyon, while all other CH waste processing could be shifted to the 2706-T/TA/TB building. The average 2706-T/TA/TB building throughput was assumed to be 900 cubic meters (1,200 cubic yards) of LLW and MLLW per year (based on previous performance [Barcot 2002], but limited to a potential source term of 3.0 dose-equivalent curies per year). The decision to expand the T Plant was based on the following assumptions:

- The average T Plant (221-T Canyon and 2706-T/TA/TB Facility) solid-waste processing throughput is 1,800 cubic meters (2,400 cubic yards) of LLW and MLLW, based on previous performance. The 2706-T/TA/TB building processed 500 cubic meters (650 cubic yards) of LLW and MLLW in 2005, 1,800 cubic meters (2,400 cubic yards) in 2004, and 1,910 cubic meters (2,500 cubic yards) in 2003 (Barcot 2002).
- Solid-waste processing at the T Plant is split between the 221-T Canyon and 2706-T/TA/TB Facility (Fluor Hanford 2005a). The 2706-T/TA/TB Facility solid-waste processing operations typically do not require supplied air or shielding, and there are no high-dose areas within the 2706-T/TA/TB Facility. Therefore, the 2706-T/TA/TB Facility is the

preferred location to process CH solid waste at the T Plant, based on operational requirements. However, some solid waste exceeds the limitation on the 2706-T/TA/TB Facility of 3.0 dose-equivalent curies per year (some exceptions have been made for vented HEPA containers) and must be processed in the 221-T Canyon along with RH solid waste. The processing split between the two processing areas changes with time, depending on the needs of the greater SWOC at any given time. The average processing capacity is assumed to be evenly split between the 221-T Canyon and 2706-T/TA/TB Facility (based on engineering judgment). Therefore, the average processing capacity of the 2706-T/TA/TB Facility is assumed to be 900 cubic meters (1,200 cubic yards) of LLW and MLLW.

- The waste-container-processing capacity of the 2706-T/TA/TB building is in linear relation to the number of units available for processing (CEES 2006g).

Under Waste Management Alternatives 2 and 3, three T Plant expansion activities are analyzed: construction, operations, and deactivation of another 2706-T/TA/TB-type building near the T Plant. Closure and postclosure care of this T Plant expansion is not within the scope of the Waste Management alternatives, but is analyzed in Chapter 6, “Cumulative Impacts.” Construction activities include the deployment of the resources and materials necessary to construct a replacement of the 2706-T/TA/TB building. Operations include deployment of the resources, materials, and labor necessary to conduct solid-waste processing of up to 900 cubic meters (1,200 cubic yards) per year of CH-LLW and CH-MLLW. Deactivation activities, including the removal of all containers containing radioactive and hazardous materials to ensure adequate protection of workers, public health and safety, and the environment, would place the facility in a stable and known condition. The T Plant deactivation assumes that 1 percent of 1 year’s throughput remains in the facility, and that it takes 6 months to clean up the facility with no waste processing during that period.

Additional solid waste NDE and NDA capacity is gained by the use of APLs containing NDE and/or NDA equipment supporting solid-waste processing activities in the 2706-T/TA/TB building for CH-TRU waste drums and standard waste boxes. The APLs may be operated from within one of the 2706-T/TA/TB building bays and may use greenhouse isolation for contamination control, similar to the greenhouse currently employed at the T Plant. Currently, the T Plant has some capability to perform macroencapsulation activities to support management of problem waste containers. Additional solid-waste macroencapsulation capacity may be gained by the use of APLs containing macroencapsulation trains to support solid waste processing activities in the 2706-T/TA/TB building. The APLs may be operated from within one of the 2706-T/TA/TB building bays and may use greenhouse isolation for contamination control, similar to the greenhouse currently employed at the T Plant (CEES 2006g).

Waste Management Alternative 1 does not analyze expansion of the T Plant; under this alternative, the existing, permitted T Plant would operate to support onsite waste.

### **E.3.2.4 Waste Receiving and Processing Facility**

#### **E.3.2.4.1 Description**

WRAP, located in the 200-West Area of Hanford, was completed in the late 1990s. WRAP comprises three buildings: 2336W, the main processing facility; 2740W, the administrative support building; and 2620W, the maintenance support building. WRAP is designed to process certain sizes of CH-LLW, CH-MLLW, CH-TRU waste, and CH-mixed TRU waste. A second building was planned for processing RH waste, but was never constructed. The original design life of WRAP was 30 years. WRAP is a Hazard Category 2 nuclear facility (CEES 2006h; Weidert 2003).

The main objectives of WRAP are receiving, verifying, repackaging/treating (if necessary), certifying, and shipping TRU waste to WIPP. WRAP receives containers of CH waste from onsite Hanford generators, including, among others, the CWC, Waste Retrieval Operations, LLBG, and T Plant, as well as from offsite generators. Radioactive waste is processed in three operational areas within WRAP's main processing facility. These are the shipping and receiving area, NDE/NDA area, and process area. Current and future activities at WRAP for all types of waste packages include shipping, receiving, waste container handling, waste container staging/storage, nonintrusive surveying, and inspection. Inspections include NDEs, NDAs, and visual examination of open waste containers in gloveboxes. Additional activities at WRAP include waste treatment, intrusive sampling, packaging, repackaging, loading, HSGS, drum venting, and decontamination.

When operating at two full shifts per day, WRAP processes 300 waste containers (LLW or CH-TRU waste) per month, or 3,600 containers per year. WRAP provides storage and staging of up to 348 drums and 16 SWBs inside the facility, and 50 LLW containers in an outside storage area.

The maximum radioactive material inventory for a single container (drum or waste box) within the SWOC is 500 grams (18 ounces) at 0.165 dose-equivalent curies (based on the inhalation equivalent of 1 curie of plutonium-239) per gram (0.04 curies per ounce), or a maximum of 82.5 dose-equivalent curies per container. The administrative control limit on radioactive source strength at WRAP is 5,700 dose-equivalent curies (CEES 2006h). Provided as Figures E-59 and E-60, respectively, are layouts of the WRAP 2336W Building and of the shipping and receiving area within the 2336W Building (CEES 2006h).

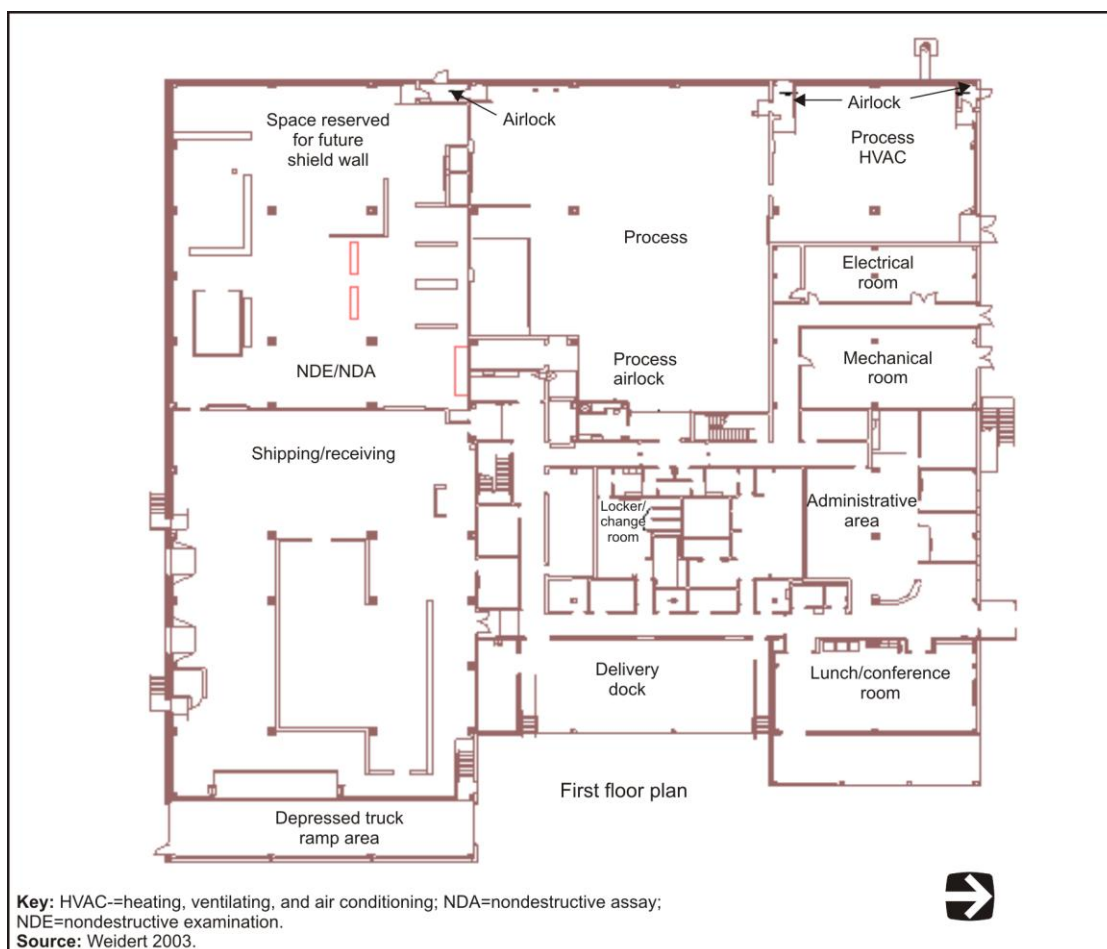
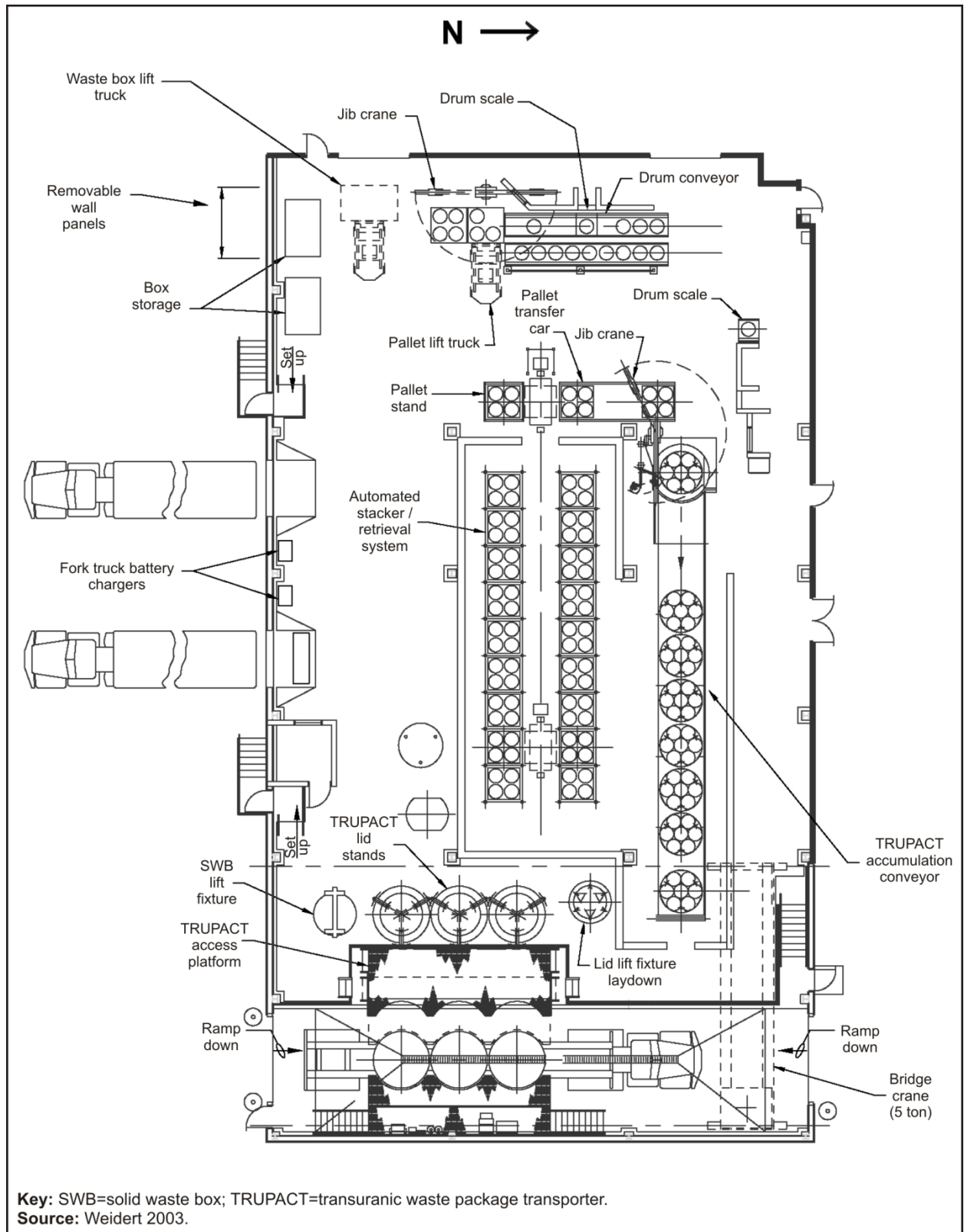


Figure E-59. Waste Receiving and Processing Facility Structure Floor Plan



**Figure E-60. Waste Receiving and Processing Facility Shipping and Receiving Area Floor Plan and Equipment Layout**

#### E.3.2.4.2 Waste Receiving and Processing Facility Expansions

Due to the uncertainty of the waste forecasts for Hanford, it was assumed for analysis purposes that additional solid-waste processing capability would be required in WRAP as soon as possible following issuance of the ROD, and that this capability would be gained by the addition of two capabilities, as follows:

- **Expanded CH-LLW, CH-MLLW, and CH-TRU Waste Processing Capability.** The existing WRAP 2336W main processing facility has no vacant area for expansion of LLW, MLLW, or CH-TRU waste processing activities. To increase the throughput of waste operations at WRAP, additional structures would be needed. Addition of structures was previously considered as part of proposed SWOC accelerated-closure initiatives. Accelerated-closure analysis determined that the best method of increasing throughput at WRAP was to construct new, or modify existing, storage buildings at the CWC for expanded waste processing activities. The CWC buildings can be modified to accept the APL trailers within the building to perform NDE, NDA, and HSGS activities that would allow processing of up to a box of waste measuring 1.5 meters (5 feet) long by 1.5 meters (5 feet) wide by 2.7 meters (9 feet) high. Although no APL trailers are presently installed within the buildings, the infrastructure needed for their installation already exists.

To increase CH-LLW, CH-MLLW, or CH-TRU waste processing capabilities beyond the level achieved through installation of APL trailers in the CWC buildings, new CWC buildings would need to be constructed called the CH-Mixed TRU/TRU Waste Facilities. For each new CWC building constructed, the waste throughput would increase by approximately 40 percent, because the new CWC would have HSGS capability and one unit each for NDE and NDA—though no glovebox for visual examinations. Resource impacts associated with construction, operation, and deactivation of CWC expansion buildings were estimated as part of the CWC expansion analysis. The CWC expansion analysis revealed resource impacts for facility construction and deactivation identical to those for increased WRAP CH-LLW, CH-MLLW, or CH-TRU waste processing throughput. Operational resource impacts for increased CWC waste processing throughput were based on the same scale (40 percent container throughput increase per CWC building) as that adopted for the present WRAP. Thus, to increase the current WRAP throughput capacity by 100 percent, i.e., to double it to 7,200 containers per year, the current WRAP operations data resources were scaled by a factor of 2.5. For analysis purposes, it was assumed that three additional CWC waste storage buildings identical to 2403-WD would be constructed within the CWC site. These three structures, totaling a building footprint of 15,450 square meters (166,300 square feet), would house the proposed CH-LLW, CH-MLLW, and CH-TRU waste processing equipment (CEES 2006h).

- **RH-TRU Waste Processing Capability.** The existing WRAP has no capability to process RH-TRU waste. Originally, a second phase of WRAP (WRAP II) was to be constructed for processing RH-TRU waste. That facility was never constructed. The WRAP II Facility was to be approximately the same size as the existing WRAP in dimensions, staffing, and resource usage. Because of the similar size and function of WRAP and the proposed WRAP II Facility, data from the current WRAP data sets were used to determine operation and deactivation resource impacts associated with establishing a new WRAP RH-Mixed TRU/TRU waste facility capable of processing several different sizes of RH-TRU waste packages. This new facility would be constructed adjacent to the existing WRAP (CEES 2006h).

Waste Management Alternatives 2 and 3 require WRAP expansion analyses addressing construction, operations, and deactivation of CH-Mixed TRU/TRU and RH-Mixed TRU/TRU waste facilities. Closure and postclosure care of WRAP expansions are not within the scope of the Waste Management alternatives, but are analyzed in Chapter 6, “Cumulative Impacts.” Construction includes the resources,



materials, and labor necessary to construct these facilities; operations, the resources, materials, and labor necessary to conduct the additional waste processing services; and deactivation, the resources required to place the facilities in a stable and known condition, including the removal of all radioactive and hazardous materials to ensure adequate protection of workers, public health and safety, and the environment. WRAP deactivation assumes that 1 percent of 1 year's throughput remains in each facility, and that it takes 6 months to clean up each facility without incurring new waste (CEES 2006h).

Waste Management Alternative 1 does not involve the expansion of WRAP; under this alternative, the existing, permitted WRAP would operate to support onsite waste.

### **E.3.3 Low-Level Radioactive Waste Burial Grounds**

#### **E.3.3.1 Description**

The LLBGs consist of eight separate waste disposal areas consolidated administratively into a single radioactive waste unit (see Table E-20). Six LLBGs are located in the 200-West Area (see Figure E-61) and two in the 200-East Area (see Figure E-62). The combined area of the LLBGs totals about 220 hectares (544 acres) (DOE 1997c). LLBG 218-W-6 is reserved for future use; no trenches have been excavated and no waste has been received. All other LLBGs have received waste, although several contain sizable portions where no trenches have been excavated and no waste has been received (CEES 2006i).

**Table E-20. Low-Level Radioactive Waste Burial Ground Locations,  
Service Dates, Areas, and Waste Types**

LLBG	Location	Service Date	Area		Waste Types
			Hectares	Acres	
218-E-10	200-East	1960	36	89	RH- and CH-LLW, MLLW, most in concrete boxes
218-E-12B	200-East	1967	68	168	LLW, TRU waste, defueled RCs
218-W-3A	200-West	1970	20	49	LLW, MLLW, TRU waste
218-W-3AE	200-West	1981	20	49	RH-LLW, MLLW, large equipment
218-W-4B	200-West	1968	4	10	TRU waste (in trenches and caissons)
218-W-4C	200-West	1978	20	49	TRU waste, MLLW, test reactor fuel waste
218-W-5	200-West	1986	37	91	LLW, MLLW (two mixed waste trenches)
218-W-6	200-West	Not in service	16	40	Reserved for MLLW trenches

**Key:** CH=contact-handled; LLBG=low-level radioactive waste burial ground; LLW=low-level radioactive waste; MLLW=mixed low-level radioactive waste; RC=reactor compartment; RH=remote-handled; TRU=transuranic.

**Source:** DOE 1997c.

The LLBGs contain lined and unlined trenches of varying size and depth that are used for disposal of LLW and for retrievable storage of TRU waste. Both CH and RH wastes are disposed of and retrievably stored at the LLBGs. All waste accepted for disposal must meet the requirements specified in the HSSWAC (Fluor Hanford 2005a).

Portions of the LLBGs contain mixed waste that is subject to WAC 173-303. Other portions contain only LLW and are designated as solid-waste management units. The LLBGs have been permitted under an RCRA Part A permit since 1985 and operate as an interim-status facility pending final administrative disposition of the LLBG Part B permit application (DOE 1997c).

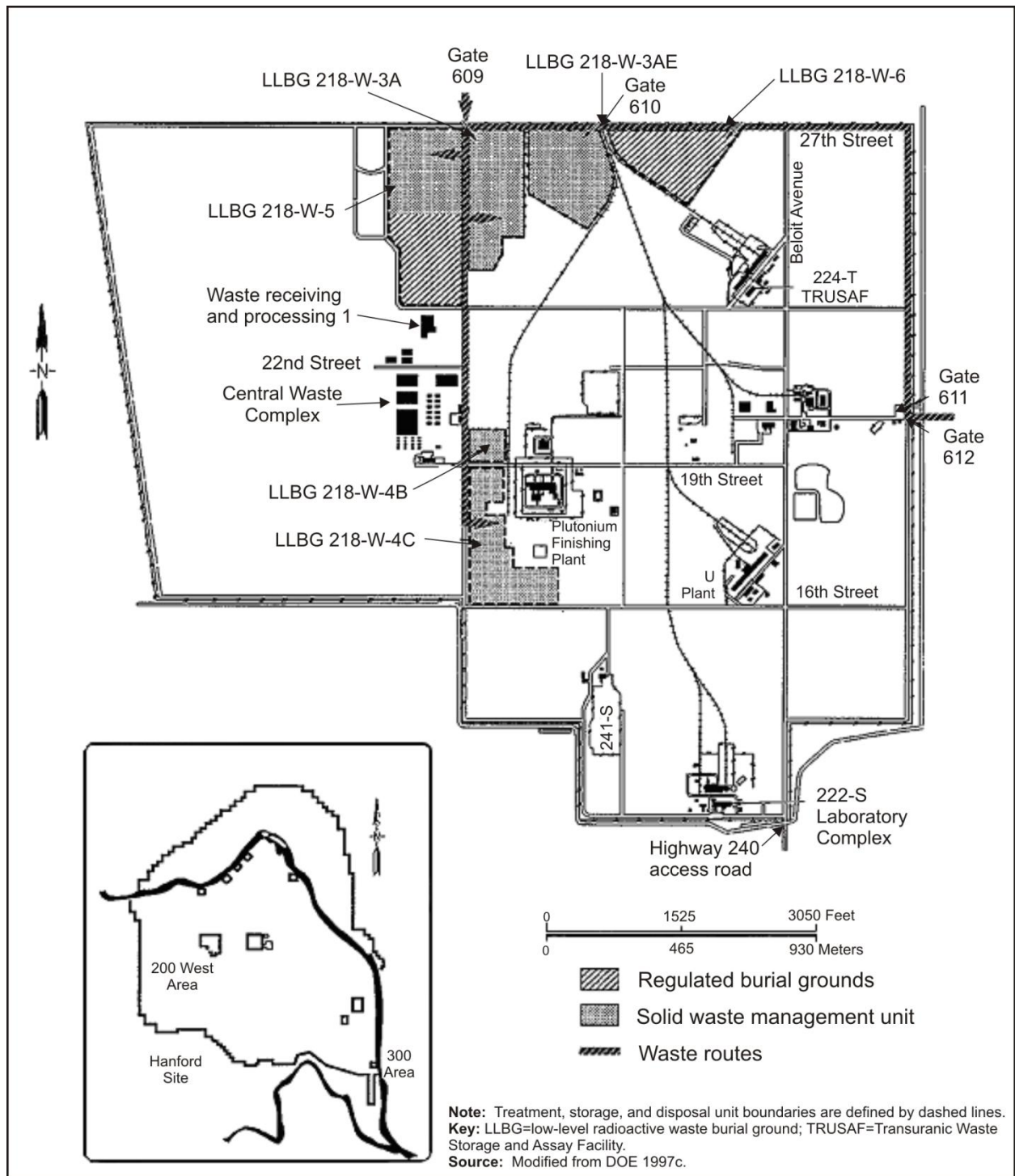
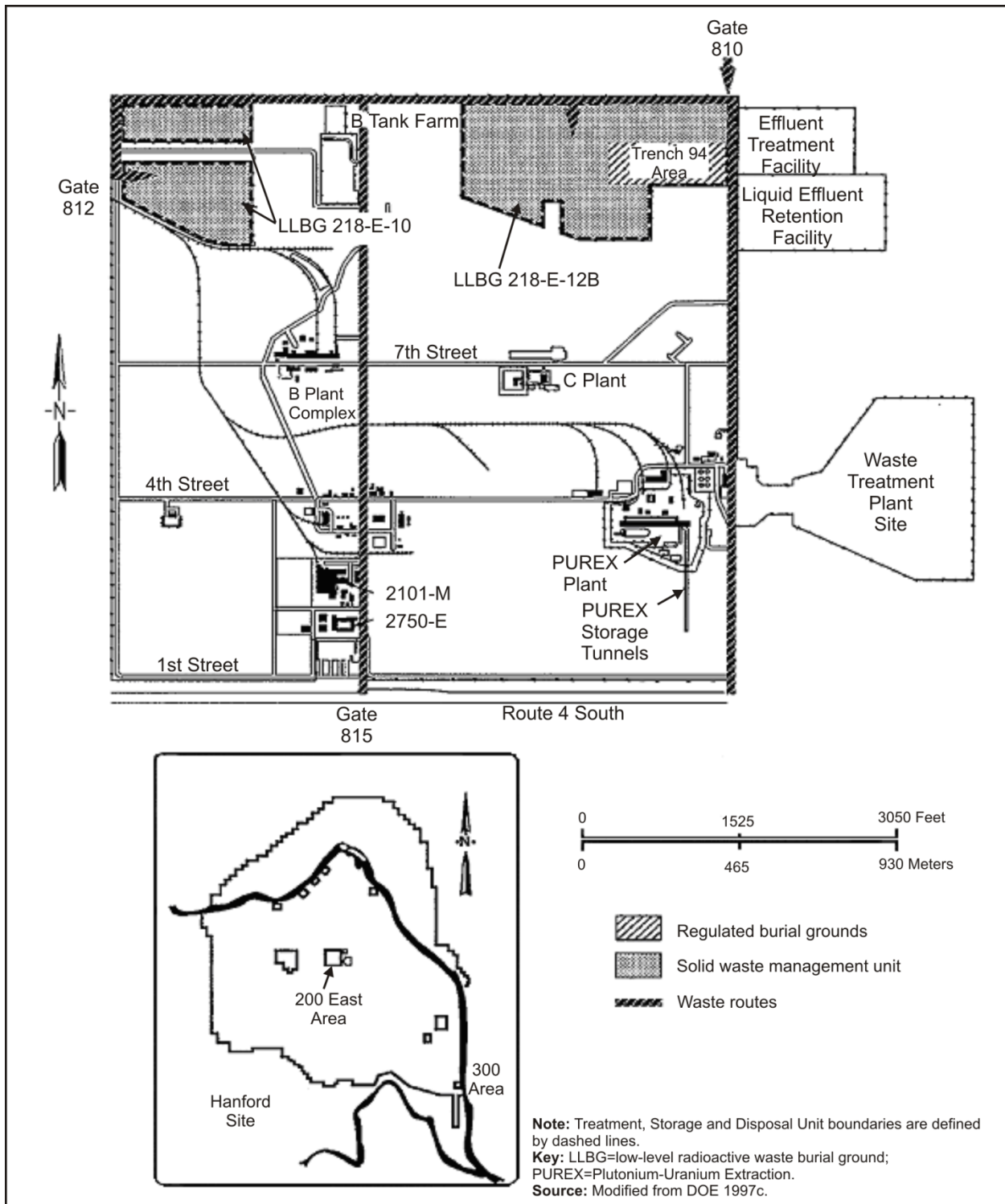


Figure E-61. 200-West Area Low-Level Radioactive Waste Burial Grounds



**Figure E-62. 200-East Area Low-Level Radioactive Waste Burial Grounds**

The earliest active burial ground operations began in 1960. From 1960 to 1970, waste was buried in shallow, unlined trenches without regard to radionuclide content or hazardous components. Beginning in 1970, suspect TRU waste was segregated from other radioactive waste and placed in retrievable storage configurations in various unlined LLBG trenches. Both soil-covered and uncovered storage configurations were used. Some RH-TRU waste in small containers was placed in underground

Alpha (TRU waste) Caissons in LLBG 218-W-4B. TRU waste has not been placed in the LLBGs since 1987 (CEES 2006i).

Burial of MLLW in the LLBGs was halted in 1987; from 1987 until the opening in 1999 of a lined, RCRA-compliant trench in LLBG 218-W-5 (trench 34), MLLW was sent to storage in the CWC. Since 1999, MLLW meeting the RCRA land-disposal-restriction treatment standards has been disposed of in trench 34. A second lined, RCRA-compliant trench (trench 31), adjacent to trench 34, became operational for disposal in 2004. Trenches 31 and 34 are the only lined trenches in the LLBGs (CEES 2006i).

In 2004, DOE announced its decision to cease disposing of radioactive waste in unlined trenches (69 FR 39449). Since 2004, all waste received for disposal at the LLBG, whether LLW or MLLW, has been disposed of in trenches 31 and 34. Disposal operations at all other existing LLBG trenches have ceased. The only exception is trench 94 in LLBG 218-E-12B, which is an unlined trench currently used for disposal of defueled Navy reactor components. Trench 94 is not affected by the DOE 2004 decision and continues to receive reactor components (CEES 2006i).

TRU waste retrieval activities are ongoing at trenches in LLBGs 218-W-3A, 218-W-4B, and 218-W-4C. The TPA (Ecology, EPA, and DOE 1989) requires that DOE retrieve all CH-TRU waste from LLBGs 218-E-12B, 218-W-3A, 218-W-4B, and 218-W-4C by 2010 (Milestone M-91-40); retrieve noncaisson RH-TRU waste by 2014 (M-91-41); and retrieve caisson RH-TRU waste by 2018 (M-91-41) (CEES 2006i).

Although not part of the proposed actions in this EIS, but added for information, an analysis of the retrieval of the RH-TRU waste currently retrievably stored in the Alpha Caissons in LLBG 218-W-4B was conducted. Included in this analysis were the resource and labor requirements for construction, operation, and deactivation of the retrieval facilities necessary to support retrieval of the estimated 5,534 waste items contained in Alpha Caissons 1 through 4. These waste items consist primarily of 3.8-, 7.6-, and 19-liter (1-, 2-, and 5-gallon) cans containing RH-TRU waste from hot-cell examinations that originated as irradiated fuel and nonfuel scrap. The Alpha Caissons are reinforced-concrete cylinders buried approximately 4 meters (13 feet) below grade with 0.9-meter-diameter (3-foot-diameter) inlet chutes that are offset to eliminate direct radiation shine from the stored wastes. Wastes are no longer added to these caissons, and TPA (Ecology, EPA, and DOE 1989) Milestone M-91-41 requires retrieval of the RH-TRU waste by December 31, 2018.

The results of the analysis indicate that required resources and air emissions would be minimal over the 2-year period estimated for completion of retrieval when compared with those of other waste retrieval activities at the site. One of a number of reasons is the plan to purchase a single pre-engineered retrieval enclosure, containing a hot-cell enclosure, from offsite sources and to move it from one caisson site to another without disassembly. After removal of the waste, each caisson would be remotely cleaned of any residue and stabilized with grout. The use of onsite equipment, fuel, electric power, water, and labor would be minimal, as would both radioactive and nonradioactive air emissions due to the use of HEPA filtration at the enclosure and minimal use of fueled equipment (CEES 2010i).

Retrieval of radioactive waste buried before 1970 is not part of the proposed actions and is not analyzed in the alternatives section of this *TC & WM EIS*. Chapter 6, "Cumulative Impacts," includes analysis of this waste.

### **E.3.3.2 Low-Level Radioactive Waste Burial Ground Activities**

Continued use of the two lined mixed waste trenches (ditches), 31 and 34, at LLBG 218-W-5 was analyzed under all three Waste Management alternatives. Activities analyzed include the following disposal operations:

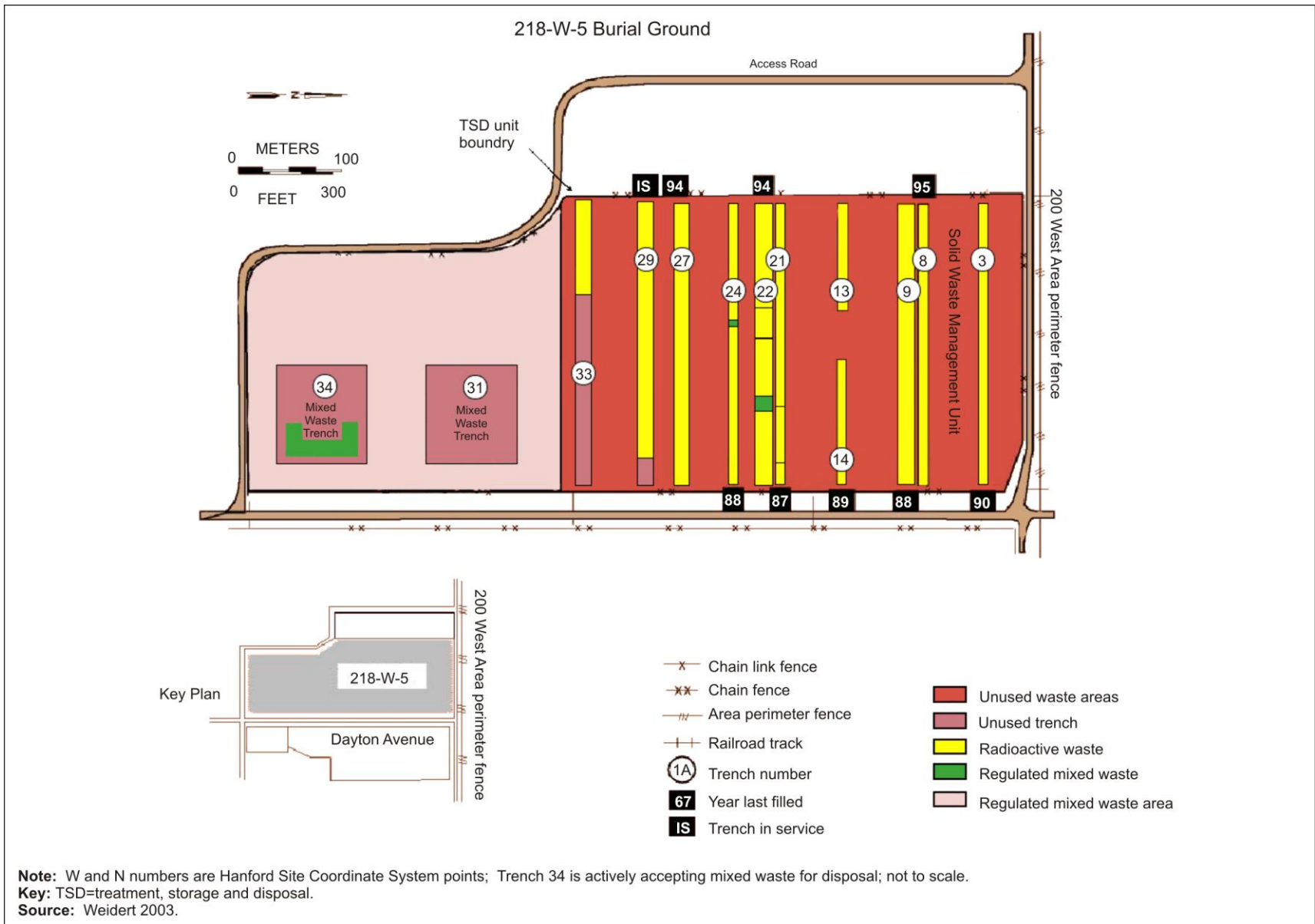
- Placement of disposal layers within the trenches
- Placement of intermediate soil cover between disposal layers
- Placement of final soil cover to grade level
- Collection and removal of leachate (CEES 2006i)

Rain or snowmelt captured by the trench liner systems is drained to a sump, pumped to a holding tank, and removed by tanker truck for treatment at the ETF.

To meet HSSWAC requirements (Fluor Hanford 2005a), Category 3 LLW/MLLW must be disposed of either in a high-integrity container or an in-trench-grouted encasement (monolith). Interviews with the current facility contractor indicated that high-integrity container use for disposal of Category 3 waste in trenches 31 and 34 has been discontinued in favor of in-trench grouting. Typical disposal practice for in-trench grouting, or concrete encasement, involves placing concrete forms and rebar around a stack of waste containers and pouring concrete to encase the waste. This results in a solid freestanding monolith. For analysis purposes, construction resource information for monolith construction was included in the resource requirements for LLBG operations. The stack of waste destined for concrete encasement is typically 39.6 meters long by 3.4 meters wide by 2.1 meters high (130 feet long by 11 feet wide by 7 feet high). The effective disposal volume of a typical monolith is approximately 116 cubic meters (152 cubic yards) of waste. The number of grout campaigns completed annually will depend on Category 3 waste receipts. Because the waste volume projections are not yet finalized, it was assumed that one concrete monolith would be constructed during each year of operation (CEES 2006i).

Construction activities are not included because the trenches are in current operation. Deactivation is not applicable because operations include filling the landfill with soil to grade level following completion of waste disposal activities, which is normally a deactivation activity. Closure and postclosure care are not analyzed for the trenches or the other LLBGs because these activities are not included in the scope of the *TC & WM EIS* alternatives analysis. However, LLBG closure and postclosure care impacts are analyzed in Chapter 6, "Cumulative Impacts." Likewise, retrieval of retrievably stored TRU waste from the LLBGs is analyzed in Chapter 6.

Trenches 31 and 34 are rectangular units with depths of 9.1 to 12.2 meters (30 to 40 feet). Both have surface dimensions of about 137 meters long by 91 meters wide (450 feet long by 300 feet wide) and base dimensions of about 76 meters long by 30 meters wide (250 feet long by 100 feet wide). The total finished airspace volume of each trench from base to ground surface is approximately 53,000 cubic meters (69,000 cubic yards) (WHC 1994). The excavation spoils from construction of trenches 31 and 34 were stockpiled within LLBG 218-W-5 for use as operational cover. Presented as Figure E-63 is a layout of LLBG 218-W-5 depicting the location of trenches 31 and 34 (CEES 2006i).



**Figure E-63. Low-Level Radioactive Waste Burial Ground 218-W-5**

The following is a description of the three configurations under each Waste Management alternative:

- **Waste Management Alternative 1.** Under the No Action Alternative, the two lined trenches, 31 and 34, would continue to receive LLW and MLLW from onsite non-CERCLA sources through 2035. It was assumed that, by the year 2035, no additional LLW or MLLW would be generated under the No Action Alternative.
- **Waste Management Alternatives 2 and 3.** Under both alternatives, the two lined trenches in LLBG 218-W-5, 31 and 34, would continue to receive LLW and MLLW from onsite non-CERCLA sources until they are filled, which is estimated to be in the year 2050. The current remaining space in the two trenches is approximately 17,215 cubic meters (22,520 cubic yards); thus, at the projected emplacement rate, the trenches would be filled to capacity by no later than 2050 (CEES 2006i). For analysis purposes, this EIS assumed the trenches would continue to operate even after IDF-East begins operations in 2009; however, all waste generated after the opening of IDF-East was assumed to be disposed of in an IDF, not trenches 31 and 34.

### **E.3.4 Integrated Disposal Facility**

#### **E.3.4.1 Description**

IDF-East, as it currently exists, is partially constructed and permitted for only limited use. Assuming completion (construction would cease under the No Action Alternative), IDF-East would consist of a single landfill with two separate, expandable cells. The landfill would be divided lengthwise, with one cell for disposal of LLW and the other permitted as an RCRA-compliant landfill system for disposal of MLLW and ILAW. The mission of IDF-East includes the following functions:

- Provide an approved disposal facility for the permanent, environmentally safe disposition of MLLW, including vitrified ILAW packages that meet the environmental requirements and are approved by DOE and Ecology.
- Provide an approved disposal facility for LLW that meets the requirements of DOE.
- Conduct solid waste operations to receive and dispose of onsite and offsite mixed waste and solid LLW (CEES 2006j).

The currently planned IDF-East would dispose of approximately 900,000 cubic meters (1.18 million cubic yards) of LLW and MLLW. However, this EIS analyzes different configurations and capacities, as described in Section E.3.4.2. IDF-East would not accept CERCLA waste, but would dispose of ILAW glass from the WTP, STP wastes (e.g., bulk vitrification glass, cast stone waste, steam reforming waste), mixed waste generated through waste operations, other LLW, offsite MLLW and LLW, and the LAW melters taken out of service (CEES 2006j).

The IDF cells would include a double liner, a leachate collection and removal system, and a leak detection system. The liner system would comply with RCRA requirements for hazardous waste landfills. When completed, IDF-East would be 555 meters long by 442 meters wide and 13.1 meters deep (1,820 feet long by 1,450 feet wide and 43 feet deep), with current planned capacity of 900,000 cubic meters (1.18 million cubic yards) of LLW and MLLW (CEES 2006j). As of April 2006, two cells had been constructed. This completed section of IDF-East is approximately 233 meters (765 feet) long by 457 meters (1,500 feet) wide and 12.8 meters (42 feet) deep (DOE 2011).

The following four IDF activities are analyzed under the Waste Management alternatives. Deactivation is not applicable because the proposed operations would include filling the landfill with soil to grade level following completion of waste disposal activities, which is normally a deactivation activity.

- **Construction.** Construction would consist of soil excavation and placement of the RCRA-compliant liner system. IDF-East's location in the 200-East Area and its dimensions were obtained from the *Preliminary Closure Plan for the Integrated Disposal Facility* (Baune 2004). IDF-East construction was assumed to occur from 2006 through 2008.
- **Operations.** Operations would consist of placement of disposal units within the landfill, soil cover between the disposal layers, and final soil cover to grade level. IDF-East operations resources were obtained from *Integrated Disposal Facility (IDF) Operations and Maintenance Philosophy, Final Report* (Dehner and Comstock 2004).
- **Closure.** Closure would consist of placement of a modified RCRA Subtitle C barrier over the landfill (CEES 2006j).
- **Postclosure Care.** Postclosure care would consist of air, groundwater, and vadose zone monitoring. As planned, postclosure care would occur for 100 years following completion of the modified RCRA Subtitle C barrier (CEES 2006j).

#### E.3.4.2 Integrated Disposal Facility Configurations

A number of different IDF configurations are analyzed within the three Waste Management alternatives. The following is a brief description of the configurations analyzed under each Waste Management alternative:

- **Waste Management Alternative 1.** Following a *TC & WM EIS* ROD of No Action, current construction operations at IDF-East would cease and the construction site would be deactivated. Deactivation activities would include liner removal and backfilling of the site to restore the area to its natural grade (SAIC 2010e).
- **Waste Management Alternative 2.** Construction of IDF-East would continue at the current site (see Figure E-11). For analysis purposes, three disposal groups have been identified to support Hanford waste management needs. Each disposal group would support a specific number of Tank Closure, FFTF Decommissioning, and Waste Management alternatives. These groupings were developed to limit the number of iterations of analysis and to support reader understanding. While it was recognized that some of the Tank Closure, FFTF Decommissioning, or Waste Management alternatives may be penalized due to oversizing of IDF-East to support all Tank Closure alternatives within each disposal group, it was concluded that such oversizing would be minor and the benefits would outweigh the negatives. Each disposal group under Waste Management Alternative 2 is summarized in Table E-21.



**Table E–21. Waste Management Alternative 2 Disposal Groups**

Parameters	IDF-East		
	Group 1	Group 2	Group 3
Tank Closure alternatives supported	2B, 3A, 3B, 3C, 4, 5, 6C	2A, 6B Base, 6B Option	6A Base, 6A Option
FFTF Decommissioning alternatives supported	All	All	All
Offsite waste	Yes	Yes	Yes
Onsite non-CERCLA waste	Yes	Yes	Yes
Capacity (cubic meters)	1,200,000	425,000	425,000
Size (hectares)	32.8	11.3	11.3
Last year of operations	2050	2100	2165

**Note:** To convert cubic meters to cubic yards, multiply by 1.308; hectares to acres, by 2.471.

**Key:** Base=Base Case; CERCLA=Comprehensive Environmental Response, Compensation, and Liability Act; FFTF=Fast Flux Test Facility; IDF-East=200-East Area Integrated Disposal Facility; Option=Option Case.

**Source:** SAIC 2010e.

- **Disposal Group 1.** This disposal group would support Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, 5, and 6C, as well as any of the three FFTF Decommissioning alternatives and disposal of onsite non-CERCLA LLW and MLLW and projected waste shipments from other DOE sites. A capacity of 1.2 million cubic meters (1.6 million cubic yards) would be required for IDF-East operations continuing through 2050. This capacity would exceed the currently planned capacity of 900,000 cubic meters (1.18 million cubic yards) by approximately 33 percent.
  - **Disposal Group 2.** This disposal group would support Tank Closure Alternatives 2A and 6B (Base and Option Cases) and any of the three FFTF Decommissioning alternatives, as well as disposal of onsite non-CERCLA LLW and MLLW and projected waste shipments from other DOE sites. A capacity of 425,000 cubic meters (555,900 cubic yards) would be required for IDF-East operations continuing through 2100. This capacity would be approximately 53 percent less than the currently planned capacity of 900,000 cubic meters (1.18 million cubic yards).
  - **Disposal Group 3.** This disposal group would support Tank Closure Alternative 6A (Base and Option Cases) and any of the three FFTF Decommissioning alternatives, as well as disposal of onsite non-CERCLA LLW and MLLW and projected waste shipments from other DOE sites. A capacity of 425,000 cubic meters (556,000 cubic yards) would be required for IDF-East operations continuing through 2165. This capacity would be approximately 53 percent less than the currently planned capacity of 900,000 cubic meters (1.18 million cubic yards) (SAIC 2010e).
- **Waste Management Alternative 3.** Two IDFs would be constructed under this alternative. IDF-East would receive only waste generated by the Tank Closure alternatives. IDF-West (north of WRAP and northwest of LLBG 218-W-5) would receive the balance of the waste that is not disposed of in IDF-East (FFTF Decommissioning alternative waste; onsite non-CERCLA waste; and projected waste shipments from other DOE sites). See Figure E–11 for the proposed locations of IDF-East and IDF-West. As with Waste Management Alternative 2, for analysis purposes, three disposal groups were identified to support Hanford waste management needs. These groupings were developed to limit the number of iterations of analysis and to support reader understanding. While it is recognized that some of the Tank Closure, FFTF Decommissioning, or Waste Management alternatives may be penalized due to oversizing of IDF-East to support all Tank Closure alternatives within each disposal group, it was concluded

that such oversizing would be minor and the benefits would outweigh the negatives. Each disposal group for Waste Management Alternative 3 is summarized in Table E–22.

**Table E–22. Waste Management Alternative 3 Disposal Groups**

Parameters	IDF-East			IDF-West
	Group 1	Group 2	Group 3	All Disposal Groups
Tank Closure alternatives supported	2B, 3A, 3B, 3C, 4, 5, 6C	2A, 6B Base, 6B Option	6A Base, 6A Option	N/A
FFTF Decommissioning alternatives supported	N/A	N/A	N/A	All
Offsite waste	N/A	N/A	N/A	Yes
Onsite non-CERCLA waste	N/A	N/A	N/A	Yes
Capacity (cubic meters)	1,100,000	340,000	340,000	90,000
Size (hectares)	29.9	9.3	9.3	2.4
Last year of operations	2050	2100	2165	2050

**Note:** To convert from cubic meters to cubic yards, multiply by 1.308; hectares to acres, by 2.471.

**Key:** Base=Base Case; CERCLA=Comprehensive Environmental Response, Compensation, and Liability Act; FFTF=Fast Flux Test Facility; IDF-East=200-East Area Integrated Disposal Facility; IDF-West=200-West Area Integrated Disposal Facility; N/A=not applicable; Option=Option Case.

**Source:** SAIC 2010e.

- **Disposal Group 1.** This disposal group would support Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, 5, and 6C. IDF-East would have a capacity of 1.1 million cubic meters (1.43 million cubic yards), exceeding the currently planned capacity of 900,000 cubic meters (1.18 million cubic yards) by approximately 22 percent, and would be required to operate through 2050. IDF-West would have a capacity of 90,000 cubic meters (118,000 cubic yards), approximately 90 percent less than that currently planned for IDF-East, and would be required to operate through 2050. IDF-West would support any of the three FFTF Decommissioning alternatives, as well as disposal of onsite non-CERCLA LLW and MLLW and projected waste shipments from other DOE sites.
- **Disposal Group 2.** This disposal group would support Tank Closure Alternatives 2A and 6B (Base and Option Cases). IDF-East would have a capacity of 340,000 cubic meters (445,000 cubic yards), approximately 62 percent less than the currently planned capacity of 900,000 cubic meters (1.18 million cubic yards), and would be required to operate through 2100. IDF-West would have a capacity of 90,000 cubic meters (118,000 cubic yards), approximately 90 percent less than that currently planned for IDF-East, and would be required to operate through 2050. IDF-West would support any of the three FFTF Decommissioning alternatives, as well as disposal of onsite non-CERCLA LLW and MLLW and projected waste shipments from other DOE sites.
- **Disposal Group 3.** This disposal group would support Tank Closure Alternative 6A (Base and Option Cases). IDF-East would have a capacity of 340,000 cubic meters (445,000 cubic yards), approximately 62 percent less than the currently planned capacity of 900,000 cubic meters (1.18 million cubic yards), and would be required to operate through 2165. IDF-West would have a capacity of 90,000 cubic meters (118,000 cubic yards), approximately 90 percent less than that currently planned for IDF-East, and would be required to operate through 2050 (SAIC 2010e). IDF-West would support any of the three FFTF Decommissioning alternatives, as well as disposal of onsite non-CERCLA LLW and MLLW and projected waste shipments from other DOE sites.

### **E.3.5 River Protection Project Disposal Facility**

#### **E.3.5.1 Description**

Another onsite disposal facility analyzed under the Waste Management alternatives is the RPPDF. The RPPDF would support disposal of moderately and lightly contaminated RSE resulting from closure activities at the SST farms under many of the Tank Closure alternatives. For analysis purposes, it was assumed that the design, construction, and operations activities necessary for the RPPDF would be identical to those needed for IDF-East (CEES 2006j). However, the capacity of the RPPDF would differ from the IDF-East capacities described above. Therefore, scaling factors were applied to the IDF-East estimates to support the projected RPPDF estimates.

#### **E.3.5.2 River Protection Project Disposal Facility Activities**

As with the IDFs, four RPPDF activities were analyzed under the Tank Closure alternatives: construction, operations, closure, and postclosure care. Construction would include all the necessary actions to build the disposal site and all the required resources and materials; operations, all the necessary resources to operate the landfill, including the disposition of leachate at the ETF; closure, placement of a modified RCRA Subtitle C barrier over the landfill (CEES 2006j); and postclosure care, air, groundwater, and vadose zone monitoring. Postclosure care is planned for 100 years following completion of the modified RCRA Subtitle C barrier (CEES 2006j). Deactivation is not applicable because operations would include filling the landfill with soil to grade level following completion of waste disposal activities, which is normally a deactivation activity.

The Tank Closure alternatives that generate moderately and lightly contaminated RSE are Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, 6A (Base and Option Cases), 6B (Base and Option Cases), and 6C. Tank Closure Alternatives 2B, 3A, 3B, 3C, and 6C would generate RSE from the removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms. Tank Closure Alternative 4 would generate RSE from the clean closure of the BX and SX tank farms, which are SST tank farms that have had past, unintentional releases. Finally, Tank Closure Alternatives 6A, Base Case, and 6B, Base Case, would generate RSE from the clean closure of all 12 SST farms, and Tank Closure Alternatives 6A, Option Case, and 6B, Option Case, would generate RSE from clean closure of all 12 SST farms and the six sets of contiguous cribs and trenches (ditches).

- **Waste Management Alternative 1.** Following a *TC & WM EIS* ROD of No Action, there would be no need for the RPPDF because no clean closure activities would be conducted. Therefore, no construction activities would begin on the RPPDF.
- **Waste Management Alternative 2.** A single RPPDF would be constructed at a site near IDF-East. See Figure E-11 for the proposed location. As with the IDFs, for analysis purposes, three disposal groups were identified to support clean closure activities under the Tank Closure alternatives. Each disposal group would support a number of Tank Closure alternatives. These groupings were developed to limit the number of iterations of analysis and to aid in reader understanding. While it is recognized that some of the Tank Closure alternatives may be penalized due to the resulting oversizing of the RPPDF to support all of the Tank Closure alternatives within a disposal group, it was concluded that such oversizing would be minor and

the benefits would outweigh the negatives. For analysis purposes, it was assumed the RPPDF would be the same depth as the IDF, 13.1 meters (43 feet). Each disposal group for Waste Management Alternative 2 is defined as follows (SAIC 2010e):

- **Disposal Group 1.** The RPPDF would have a capacity of approximately 1.08 million cubic meters (1.41 million cubic yards) and would be required to operate through 2050. Disposal Group 1 would support Tank Closure Alternatives 2B, 3A, 3B, 3C, 4, and 6C, all of which would generate RSE from the removal of 4.6 meters (15 feet) of soil and ancillary equipment at the BX and SX tank farms.
- **Disposal Group 2.** The RPPDF would have a capacity of approximately 8.37 million cubic meters (10.9 million cubic yards) and would be required to operate through 2100. This group would support Tank Closure Alternative 6B (Base and Option Cases).
- **Disposal Group 3.** The RPPDF would have a capacity of approximately 8.37 million cubic meters (10.9 million cubic yards) and would be required to operate through 2165. Disposal Group 3 would support Tank Closure Alternative 6A (Base and Option Cases).
- **Waste Management Alternative 3.** Sizing and location of the RPPDF would be the same as described under Waste Management Alternative 2, above.

Table E–23 summarizes the disposal groups for the RPPDF under Waste Management Alternatives 2 and 3.

**Table E–23. Waste Management Alternatives 2 and 3 RPPDF Disposal Groups**

<b>Waste Management Alternatives 2 and 3</b>			
<b>Parameters</b>	<b>RPPDF Disposal Group 1</b>	<b>RPPDF Disposal Group 2</b>	<b>RPPDF Disposal Group 3</b>
Tank Closure alternatives supported	2B, 3A, 3B, 3C, 4, 6C	6B Base and 6B Option	6A Base and 6A Option
Capacity (cubic meters)	1,080,000	8,370,000	8,370,000
Size (hectares)	29.5	228.2	228.2
Last year of operations	2050	2100	2165

**Note:** To convert cubic meters to cubic yards, multiply by 1.308; hectares to acres, by 2.471.

**Key:** Base=Base Case; Option=Option Case; RPPDF=River Protection Project Disposal Facility.

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