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On-Line Waste Library V5.0 Supporting Information

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ABSTRACT

The On-Line Waste Library is a website that contains information regarding United States Department of Energy-managed high-level waste, spent nuclear fuel, and other wastes that are likely candidates for deep geologic disposal, with links to supporting documents for the data. This report provides supporting information for the data for which an already published source was not available.

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ACRONYMS AND DEFINITIONS

Abbreviation	Definition
DOE	United States Department of Energy
EBR-II	Experimental Breeder Reactor-II
EMT	electrometallurgical treatment
FFTF	Fast Flux Test Facility
HIP	hot isostatic pressing
HLW	high-level waste
INL	Idaho National Laboratory
MTHM	metric tons heavy metal
NNDC	National Nuclear Data Center
OWL	On-Line Waste Library
SNF	spent nuclear fuel

1. INTRODUCTION

The On-Line Waste Library (OWL) is a website that contains information regarding high-level waste (HLW), spent nuclear fuel (SNF), and other wastes that are managed by the United States Department of Energy (DOE) and are likely candidates for deep geologic disposal. In the interest of transparency and traceability, the website provides links to supporting documents for the data. In general, these supporting documents have already been published. However, in a few cases (e.g., calculation of a volume of waste based on published waste package dimensions), data in OWL could not be directly supported by a source that was already published. This report provides support for the data for which an already published source was not available.

Section 2 contains supporting information for calcine waste, Section 3 provides supporting information for strontium and cesium capsules, Section 4 provides supporting information for calculating radionuclide decay and production and calculating the heat generated by radioactive decay, and Section 5 provides supporting information for sodium-bonded spent fuel. Section 6 provides parameters values regarding glass waste that are needed to create input files for computer codes that perform repository safety analyses.

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2. CALCINE WASTE SUPPORTING INFORMATION

Calcine waste is a solid, dry, granular material derived from liquid wastes produced by reprocessing SNF. As a part of the final environmental impact statement for HLW currently stored in Idaho, several different options were proposed for treating the 4,400 m^3 (160,000 ft^3) of calcine HLW at Idaho National Laboratory (INL) so that it is ready for disposal in a national geologic repository (DOE, 2002a). Each option results in a different number of cans or canisters of waste. The discussion below explains the basis for the estimates of the number of cans or canisters of waste that will result from each of the treatment processes. The five distinct treatment processes for the calcine waste presented in the environmental impact statement are: 1) vitrification following separation, 2) hot isostatic pressing (HIP) without separation, 3) direct cementing without separation, 4) vitrification without separation, and 5) no treatment. The HIP option has since been developed further into either HIP with additives that eliminate the Resource Conservation and Recovery Act hazardous waste characteristics or without such additives (75 FR 137), for a total of six options. The number of canisters assumed to result from each of these six options is discussed below.

In addition, the average heat output of a canister produced by each of these six options is also discussed below. For all treatment processes, the average heat output of a canister is calculated by dividing the thermal output of all the waste, 92,600 watts (as of January 1, 2016), by the number of canisters or cans. This approach is valid for the treatment process that includes separating the waste prior to vitrification because the heat-generating radionuclides (cesium, strontium, and transuranic elements) remain in the waste that will be vitrified (DOE 2002a).

Vitrification Following Separation

In this option, cesium, strontium, and transuranic elements would be separated from the other constituents of the calcine waste. Cesium, strontium, and transuranic elements, called the "high-level waste fraction," account for most of the radioactivity, heat, and long-lived characteristics of HLW. The process stream remaining after separating out the mixed HLW fraction would be managed as low-level waste. The HLW fraction would be vitrified.

The environmental impact statement for the proposed Yucca Mountain Repository estimates that vitrifying the calcine waste after separation (i.e., vitrifying the HLW fraction) will result in 1,190 canisters of glass (Table A-26, DOE 2002b). The canisters are 2 feet in diameter and 10 feet tall, and each would be filled with approximately 22 ft^3 of waste. The average thermal output of each canister is about 78 watts. The total volume of glass would be 26,238 ft^3 (743 m^3) and the total volume of waste (including canisters) would be approximately 37,000 ft^3 (1,060 m^3). These cans would be stored pending disposal in a repository.

HIP Without Separation, With Additives

The current plan calls for the calcine to be placed in cans that are 60 inches (5 feet) in diameter and 30 inches (2.5 feet) high (CH2M WG Idaho 2012). The internal volume of this can, prior to HIP, is about 48 ft^3 (1.36 m^3), assuming a wall thickness of 0.125 inches and a bottom and top thickness of 0.25 inches. In previous tests with additives, the calcine waste loading was 77 wt% (waste loading is the weight of the calcine divided by the weight of the calcine and additives) (CH2M WG Idaho 2012). It is assumed, therefore, that approximately 1,100 m^3 of additive would be mixed with the waste for a combined volume of material to be treated of 5,500 m^3.

Under these assumptions, the 5,500 m^3 of calcine would fill 4,045 cans to undergo hot isostatic pressing. The average thermal output of each can to undergo the HIP process is about 23 watts. Once processed via HIP, these cans would then be placed in larger naval canisters for transportation and storage pending disposal in a repository. The larger naval canisters have a diameter of 5.5 feet and a height of 17.5 feet (CH2M WG Idaho 2012) with a usable interior height of about 16 feet because of a 3.5-inch thick bottom plate and a 15-inch thick shield plug (Section 1.5.1.4.1.2.1, DOE 2008). Assuming a 30% reduction in the height of a can that has undergone HIP (Bateman et al, 2013), such that a 2.5-foot high can is shortened to 1.75 feet high after HIP, nine of these shortened cans would fit in a single canister and approximately 450 canisters would be needed to store and dispose of the 4,045 cans of waste after HIP. The average thermal output of each canister is about 206 watts. The total volume of waste (including canisters) would be approximately 190,000 ft^3.

HIP Without Separation, Without Additives

As in the "HIP Without Separation, With Additives" treatment process, the current plan calls for the calcine to be placed in cans that are 60 inches (5 feet) in diameter and 30 inches (2.5 feet) high (CH2M WG Idaho 2012). The internal volume of this can, prior to HIP, is about 48 ft^3 (1.36 m^3), assuming a wall thickness of 0.125 inches and a bottom and top thickness of 0.25 inches. If additives are not mixed with the calcine waste, the 4,400 m^3 of calcine waste will fill 3,236 cans to undergo HIP. The average thermal output of each can to undergo HIP is about 29 watts. After treatment via HIP, these cans would then be placed in larger naval canisters for transportation and storage pending disposal in a repository. The larger naval canisters have a diameter of 5.5 feet and a height of 17.5 feet (CH2M WG Idaho 2012) with a usable interior height of about 16 feet because of a 3.5-inch thick bottom plate and a 15-inch thick shield plug (Section 1.5.1.4.1.2.1, DOE 2008). Assuming a 30% reduction in the height of a can that has undergone HIP (Bateman et al. 2013), such that a 2.5-foot high can is shortened to 1.75 feet high after HIP, nine of these shortened cans would fit in a single canister and approximately 360 canisters would be needed to store and dispose of the 3,236 cans of waste after HIP. The average thermal output of each canister is about 257 watts. The total volume of waste (including canisters) would be approximately 150,000 ft^3.

Direct Cement Without Separation

In this option, the calcine would be mixed with clay, blast furnace slag, caustic soda, and water. The resulting grout would be poured into stainless steel canisters (DOE 2002a). The Environmental Impact Statement for Idaho High-Level Waste and Facilities Disposition (DOE 2002a) gives the number of HLW canisters resulting from the direct cementing of the calcine waste as 18,000 (Table 3-2). The canisters are 2 feet in diameter and 10 feet tall (Section 3.1.4.2, DOE 2002a), and their average thermal output is about 5 watts. The total volume of cemented waste is 460,000 ft^3 (13,000 m^3) (Table 3-2, DOE 2002a) and the total volume (including canisters) would be approximately 570,000 ft^3. These canisters would be stored pending disposal in a repository.

Vitrification Without Separation

In this option, the calcine would be mixed with glass frit and fed to a melter to produce glass that would be poured into stainless steel canisters. The Environmental Impact Statement for Idaho High-Level Waste and Facilities Disposition (DOE 2002a) gives the number of HLW canisters

resulting from vitrification of the calcine waste as 12,000 (Table 3-2). The canisters are 2 feet in diameter and 10 feet tall (Section 3.1.4.3, DOE 2002a), and their average thermal output is about 8 watts. The total volume of vitrified waste is 300,000 ft^3 (8,500 m^3) (Table 3-2, DOE 2002a) and the total volume (including canisters) would be approximately 380,000 ft^3. These canisters would be stored pending disposal in a repository.

No Further Treatment

In this option, the calcine would be retrieved from the bins and packaged in stainless steel canisters for disposal in a geologic repository. The Environmental Impact Statement for Idaho High-Level Waste and Facilities Disposition (DOE 2002a) gives the number of HLW canisters resulting from packaging the calcine waste without further treatment (considered under the Steam Reforming Option for sodium-bearing waste) as 6,100 (Table 3-2). The canisters are 2 feet in diameter and 10 feet tall (Section 3.1.4.4, DOE 2002a), and their average thermal output is about 15 watts. The total volume of untreated calcine waste is 160,000 ft^3 (4,400 m^3) (Table 3-2, DOE 2002a) and the total volume (including canisters) would be approximately 190,000 ft^3. These canisters would be stored pending disposal in a repository.

Summary

Table 1 summarizes the treatment options, estimated number of canisters, and estimated total volume for each of the calcine treatment options.

Table 1. Summary of Estimated Number of Canisters and Estimated Total Volume for Each of the Calcine Treatment Options

Waste Treatment Option	Number of Cans or Canisters	Canister Dimensions	Total Volume of Waste (including canisters) (ft^3)	Average Thermal Output of a Can or Canister (watts)
Vitrification Following Separation	1,190	2 ft. diameter 10 ft. high	37,000	78
HIP Without Separation, With Additives	4,045 cans	5 ft. diameter 2.5 ft tall (prior to HIP)	190,000	23
	~450 canisters	5.5 ft. diameter 17.5 ft tall		206
HIP Without Separation, Without Additives	3,236 cans	5 ft. diameter 2.5 ft tall (prior to HIP)	150,000	29
	~360 canisters	5.5 ft. diameter 17.5 ft tall		257
Direct Cement Without Separation	18,000	2 ft. diameter 10 ft. high	570,000	5
Vitrification Without Separation	12,000	2 ft. diameter 10 ft. high	380,000	8

Waste Treatment Option	Number of Cans or Canisters	Canister Dimensions	Total Volume of Waste (including canisters) (ft^3)	Average Thermal Output of a Can or Canister (watts)
No Further Treatment	6,100	2 ft. diameter 10 ft. high	190,000	15

3. STRONTIUM AND CESIUM CAPSULES SUPPORTING INFORMATION

This waste consists of 1,335 CsCl capsules and 601 SrF_2 capsules, each about 21 inches tall and 3 inches in diameter. They are currently managed as high-level waste and stored in pools at the Waste Encapsulation and Storage Facility at Hanford. The preferred alternative for treating the waste in the capsules such that it can be disposed of is to open the capsules, remove the waste, and vitrify it. It is estimated that this treatment method would produce 340 canisters of glass waste, each 2 feet in diameter and 15 feet high (DOE 2012). The volume of the contents of the cesium and strontium capsules after vitrification, including the waste package, is therefore about 16,000 ft^3 ($\pi \times (1 \text{ foot})^2 \times 15 \text{ feet} \times 340$ packages).

The average thermal output of a cesium capsule is 118 watts while the average thermal output of a strontium capsule is 158 watts (Price 2018) as of January 2016. Therefore, the average thermal output of a canister of vitrified waste created from the cesium and strontium capsules would be about 743 watts (((118 × 1335) + (158 × 601))/340) as of that date.

An alternative method of disposal consists of disposing of the capsules as-is in waste packages designed for a deep borehole. The conceptual design for this approach calls for 18 capsules to be placed in each waste package and for each waste package to contain only cesium or only strontium capsules (Freeze et al. 2016). Each of the 108 waste packages is 8.625 inches in diameter and 15.6 feet tall (including impact limiter and fishing neck). The total volume of waste, including the waste package, is therefore about 686 ft^3 ($\pi \times (0.36 \text{ ft})^2 \times 15.6 \times 108$).

The average thermal output of a borehole-disposal waste package that contains cesium capsules is 2,124 watts (118 × 18), as of January 2016. The average thermal output of a borehole-disposal waste package that contains strontium capsules is 2,844 watts (158 × 18) as of January 2016.

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4. RADIONUCLIDE DATA

The OWL gives the radionuclide inventory as of a baseline date for each waste and also calculates the radionuclide inventory at a user-defined time in the future. It should be noted that, for radionuclides that have an inventory less than 1×10^{-7} curies, the inventory for that radionuclide is reported as 0. The actual inventory value for radionuclides present in a particular waste in quantities less than 1×10^{-7} curies can be found in the inventory supporting document associated with that waste.

The following sections discuss the assumptions made in performing radionuclide decay and ingrowth calculations and the basis for radionuclide decay heat calculations.

4.1. Radionuclide Decay and Ingrowth Assumptions

It is important to note that the decay and ingrowth calculations performed in OWL are intended to forecast the radionuclide inventory over the next few hundred years. As such, the quantities of daughter products that are in secular equilibrium with their parents is included in the inventory estimate, as is the ingrowth of ^{241}Am from ^{241}Pu; complex and long decay chains involving long-lived isotopes of U and Pu are not included in the inventory estimate.

In the OWL, a supporting document is provided for each radionuclide. This supporting document identifies modes of decay and decay product(s) for each radionuclide. In cases in which there is more than one decay product (i.e., daughter) for a given radionuclide, the daughter radionuclide entered into the OWL database is the radionuclide with the highest branching fraction; this daughter radionuclide is highlighted in yellow in each radionuclide supporting document.

The following assumptions are made in performing radionuclide decay calculations.

1. On the OWL Radionuclide Inventory Calculator webpage, the default target date is the current date.
2. On the OWL Radionuclide Inventory Calculator webpage, the day of the year for the user-selected target date is June 30.
3. On the OWL Radionuclide Inventory Calculator webpage, the earliest user-selected target date is 1992, which is the date the last production reactor (K reactor at Savannah River) was shut down.
4. The following table lists each radionuclide that is in secular equilibrium with another radionuclide, its half-life, the parent with which the radionuclide is in secular equilibrium, the half-life of the parent, the branching fraction, and the daughter radionuclide.

Table 2. Radionuclides in Secular Equilibrium for Decay and Ingrowth Calculations

Radionuclide	Half-life	Parent with which the radionuclide is in secular equilibrium	Branching Fraction	Parent Half-life	Daughter Radionuclide
^{90}Y	64 hours	^{90}Sr	1	28.9 years	^{90}Zr (stable)
^{106}Rh	30.1 seconds	^{106}Ru	1	1.02 years	^{106}Pd (stable)
^{125m}Te	57.4 days	^{125}Sb	0.09	2.76 years	^{125}Te (stable)
^{126}Sb	12.35 days	^{126}Sn	1	218,000 years	^{126}Te (stable)
^{126m}Sb	19.15 minutes	^{126}Sn	1	218,000 years	^{126}Sb
^{137m}Ba	2.552 minutes	^{137}Cs	0.95	30.08 years	^{137}Ba (stable)
^{144}Pr	17.28 minutes	^{144}Ce	0.989	284.91 days	^{144}Nd (stable)
^{208}Tl	3.05 minutes	^{228}Th	0.3594	1.9125 years	^{208}Pb (stable)
^{212}Pb	10.62 hours	^{228}Th	1	1.9125 years	^{212}Bi
^{214}Pb	27.06 minutes	^{226}Ra	1	1600 years	^{214}Bi
^{212}Bi	60.55 minutes	^{228}Th	1	1.9125 years	^{212}Po
^{214}Bi	19.71 minutes	^{226}Ra	1	1600 years	^{214}Po
^{212}Po	0.299 microseconds	^{228}Th	0.64	1.9125 years	^{208}Pb (stable)
^{214}Po	0.164 milliseconds	^{226}Ra	1	1600 years	^{210}Pb
^{216}Po	0.144 seconds	^{228}Th	1	1.9125 years	^{212}Pb

Radionuclide	Half-life	Parent with which the radionuclide is in secular equilibrium	Branching Fraction	Parent Half-life	Daughter Radionuclide
^{218}Po	3.097 minutes	^{226}Ra	1	1600 years	^{214}Pb
^{220}Rn	55.6 seconds	^{228}Th	1	1.9125 years	^{216}Po
^{222}Rn	3.82 days	^{226}Ra	1	1600 years	^{218}Po
^{224}Ra	3.66 days	^{228}Th	1	1.9125 years	^{220}Rn
^{228}Ac	6.15 hours	^{228}Ra	1	5.75 years	^{228}Th
^{231}Th	1.06 days	^{235}U	1	7.04×10^8 years	^{231}Pa
^{234}Th	24.1 days	^{238}U	1	4.468×10^9 years	^{234}Pa
^{233}Pa	26.98 days	^{237}Np	1	2.14×10^6 years	^{233}U
^{234}Pa	1.159 minutes	^{238}U	1	4.468×10^9 years	^{234}U
^{240}U	14.1 hours	^{244}Pu	1	8.11×10^7 years	^{240}Np
^{238}Np	2.099 days	^{242m}Am	0.005	141 years	^{238}Pu
^{239}Np	2.4 days	^{243}Am	1	7,364 years	^{239}Pu
^{240}Np	7.22 minutes	^{244}Pu	1	8.11×10^7 years	^{240}Pu
^{243}Pu	4.956 hours	^{247}Cm	1	1.56×10^7 years	^{243}Am
^{242}Am	16.02 hours	^{242m}Am	0.995	141 years	^{242}Cm
^{242}Cm	162.86 days	^{242m}Am	0.83	141 years	^{238}Pu

5. The inventory of ^{241}Am ($t_{½}$ = 432.7 years) includes production of ^{241}Am by decay of ^{241}Pu ($t_{½}$ = 14.4 years). ^{241}Am is the only actinide for which the inventory calculation includes both decay and production; for all other actinides, the inventory calculation includes only decay. That is, decay chains are not accounted for, except for decay of ^{241}Pu into ^{241}Am and those radionuclides that are in secular equilibrium. The equation used to calculate the quantity of ^{241}Am is:

$$Mass\ of\ Am241\ at\ target\ date = C2 \times T2 \times AM2 \times 2.7982 \times 10^{-6} \times e^{\left(\frac{-\ln(2)}{T2}\right)*(target\ date - baseline\ date)} + \frac{\frac{ln(2)}{T1} \times C1 \times T1 \times AM1 \times 2.7982 \times 10^{-6} \times (e^{\left(\frac{-\ln(2)}{T1}\right)\times(target\ date - baseline\ date)} - e^{\left(\frac{-\ln(2)}{T2}\right)\times(target\ date - baseline\ date)})}{\frac{\ln(2)}{T2} - \frac{\ln(2)}{T1}}$$

where:

C1 = curies of ^{241}Pu
C2 = curies of ^{241}Am
T1 = half-life of ^{241}Pu in years
T2 = half-life of ^{241}Am in years
AM1 = atomic mass of ^{241}Pu in grams
AM2 = atomic mass of ^{241}Am in grams
Target date and baseline date are in years

6. The conversion from curies to grams (or vice versa) is made using the following equation:

$$Grams\ of\ radionuclide\ i = Ci_i \times t_i \times AM_i \times 2.7982 \times 10^{-6}$$

where:

Ci_i = curies of radionuclide *i*
t_i = half-life of radionuclide *I*, years
AM_i = atomic mass of radionuclide *i*, grams

4.2. Radionuclide Decay Heat

The heat generated by radioactive decay is calculated for nine selected radionuclides. The nine radionuclides selected contribute the most to production of decay heat in spent fuel and high-level waste over the time scales of interest (between a few decades and a few hundred years out of reactor): ^{90}Sr, ^{90}Y, ^{137}Cs, ^{137m}Ba, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Am, and ^{244}Cm (Gauld and Murphy 2010). The calculation begins with the energy associated with the decay of a single atom of each radionuclide, which is taken from data sheets obtained on-line from the National Nuclear Data Center (NNDC) at Brookhaven National Laboratory (http://www.nndc.bnl.gov/). The energy (in

keV) is multiplied by 1000 to convert from keV to eV, multiplied by 1.602 x 10^{-19} (J/eV), multiplied by 3.7 x 10^{10} decays/second/Ci, and multiplied by 1000 Ci/kCi:

$$decay\ energy\ \left(\frac{keV}{decay}\right) \times 1000\,\frac{eV}{keV} \times 1.602 \times 10^{-19}\,\frac{J}{eV} \times 3.7 \times 10^{10}\,\frac{\frac{decays}{second}}{Curie} \times 1000\,\frac{Ci}{kCi} = decay\ heat\ \left(\frac{W}{kCi}\right)$$

Table 2 gives the decay energy per decay (as taken from the NNDC), the frequency of occurrence of that decay energy, the (weighted) average decay energy, and the calculated resulting decay heat for each of the nine radionuclides.

Table 3. Decay Energies and Decay Heat for Nine Selected Radionuclides

Radionuclide	Decay Energy (keV)	Frequency (%)	Average Decay Energy (keV)	Decay Heat (Watts/kCi)
^{90}Sr	195.8	100	195.8	1.16
^{90}Y	933.7	99.9885	933.61	5.53
	185.6	0.0115		
^{137}Cs	174.32	94.7	187.14	1.11
	416.26	5.3		
^{137m}Ba	661.657	89.9	661.657	3.92
^{238}Pu	5499.03	70.91	5486.22	32.52
	5456.3	28.98		
	5357.7	0.105		
^{239}Pu	5156.59	70.77	5139.11	30.46
	5144.3	17.11		
	5105.5	11.94		
^{240}Pu	5168.17	72.8	5150.95	30.53
	5123.68	27.1		
^{241}Am	5485.56	84.8	5474.72	32.45
	5442.8	13.1		
	5388	1.66		
	5544.5	0.37		
^{244}Cm	5762.64	23.10	5795.04	34.35
	5804.77	76.90		

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5. SODIUM-BONDED SPENT FUEL SUPPORTING INFORMATION

This waste consists of two types of SNF, i.e. driver fuel and blanket fuel. The driver fuel is highly enriched uranium (>20% enrichment) and the blanket fuel is depleted uranium alloy fuel. Most of the driver fuel consists of metal enriched to between 55% and 76% ^{235}U upon discharge alloyed with 5% to 10 wt.% zirconium or fissium. The fuel is completely surrounded by a layer of metallic sodium (for heat transfer) which is contained within steel cladding. Blanket fuel, which is fertile fuel that is placed at the perimeter of the core and is designed to breed fissile isotopes such as ^{239}Pu, consists of depleted uranium, also surrounded by a layer of sodium metal, in stainless-steel cladding. (DOE 2000a).

Most of this fuel was generated by research, development, and demonstration activities associated with three liquid metal fast breeder reactors over several decades: the Experimental Breeder Reactor-II (EBR-II) about 40 miles west of Idaho Falls, Idaho; the Enrico Fermi Atomic Power Plant in Monroe, Michigan; and the Fast Flux Test Facility (FFTF) at the Hanford site in Richland, Washington. Small quantities of sodium-bonded spent fuel were generated by other liquid metal reactor experiments at several DOE sites such as the Hanford site, Oak Ridge National Laboratory, the Savannah River site, Sandia National Laboratories, and Idaho National Laboratory. The current location of these small quantities is unknown. Quantities of sodium-bonded spent fuel as of 2000 are shown in Table 4 (DOE 2000b).

Table 4. Sodium-Bonded Spent Fuel Quantities as of 2000 (DOE 2000b, Table D-1)

Spent Nuclear Fuel Type	Canister Storage Volume (m^3)	Metric Tons of Heavy Metal (MTHM)
EBR-II Driver	58	3.1
EBR-II Blanket	13	22.4
Fermi-1 blanket	19	34.2
FFTF driver	8	0.3
Other	3	0.1
Total	101	60

The radionuclide inventories of the various types of spent sodium-bonded fuel vary widely due to difference in construction, function, and operational history of the fuel. Therefore, radionuclide inventory estimates were developed for the fuels that comprise the majority of the spent sodium-bonded spent fuel inventory: EBR-II driver (including a separate estimate for the experimental driver fuel), EBR-II blanket fuel, Fermi-1 blanket fuel, and FFTF driver fuel (DOE 2000b, Appendix D). It is the inventory of these fuels that is reported in the OWL.

The metallic sodium in the fuel presents challenges for management and disposal of this spent nuclear fuel without further treatment. Metallic sodium reacts with water to produce explosive hydrogen gas and corrosive sodium hydroxide, both of which could affect the performance of natural and engineered barriers that are part of a geologic repository. The DOE considered several options for treating sodium-bonded spent fuel to make it suitable for disposal and decided to treat all sodium-bonded spent fuel except the Fermi-1 blanket fuel using electrometallurgical treatment (EMT), also sometimes called “pyroprocessing” or “electrorefining” (DOE 2000a). The electrorefiner in which EMT occurs contains a molten mixture of primarily LiCl and KCl.

Chopped up fuel elements placed in stainless steel baskets to form the anode are lowered into the molten salt. Applying an electric voltage between the cathode (uranium) results in the dissolution of transuranic elements, fission products, and the sodium into the salt. The uranium is deposited at the cathode. Stainless-steel cladding hulls and noble metal fission products remain in the anode baskets (DOE 2000a). Salt is removed from the electrorefiner when either the sodium limit or the plutonium limit is reached (SNL 2014). The DOE is currently operating two electrorefiners, the Mark-IV, which is used to treat driver fuel, and the Mark-V, which is used to treat blanket fuel.

Thus, this treatment produces two forms of HLW: a salt waste and a metallic waste. It also produces a low-enriched uranium product that is not waste. The DOE preferred method for further treating the salt waste is to let it solidify, grind it to a desired size, mix it with zeolite, heat it, add glass frit, and hot press it to produce a ceramic HLW form that is expected to be suitable for disposal (DOE 2000a). An alternative, which has been proposed by others but is not DOE's preferred alternative, would be to not treat the salt waste and dispose of it as-is. The metallic waste is to be melted in a casting furnace to produce a metal HLW form that is expected to be suitable for disposal. The uranium is to be melted and solidified to form an ingot and stored until the DOE decides on a future use for this material (SNL 2014).

As of 2017, the DOE had processed 1.14 MTHM of EBR-II driver fuel and 0.22 MTHM of FFTF driver fuel in the Mark-IV electrorefiners and had processed 3.68 MTHM of EBR-II blanket fuel in the Mark-V electrorefiner (Rechard et al. 2017, Figure 1). The salt that was produced in each of the electrorefiner and which contains metal fission products and transuranic elements was still in the electrorefiners as of 2017; it had not yet been removed. The salt volume limit for the Mark-IV electrorefiner is 415 liters while the salt volume limit for the Mark-V electrorefiner is 449 liters (INL 2007). It is assumed for the OWL that the current volume of salt waste in each of the electrorefiners is equal to the salt volume limit for that electrorefiner.

As of 2017, three circular ingots of metallic waste had been cast (Rechard et al. 2017, Section 2.1), each 40 cm in diameter and 5 cm in height (Westphal et al. 2013, Section II.A). The ingot from the first production-scale metal waste run weighed 43.2 kg (Westphal et al. 2013, Section III).

EMT of all the sodium-bonded spent fuel from the EBR-II and the FFTF followed by treatment of the resulting salt waste to produce a ceramic HLW form suitable for disposal is expected to produce 96 canisters of HLW (DOE 2000c, Table 5-7). Each canister is 2 feet (0.61 m) in diameter and 10 feet (3 m) tall; the estimated total volume is 60 m^3 (DOE 2000c).

EMT of all the sodium-bonded spent fuel from the EBR-II and the FFTF followed by disposal of the salt waste with no further treatment is expected to produce 1,017 kg of Mark-IV salt waste and 699 kg of Mark-V salt waste. The proposed waste package configuration calls for the salt to be placed in small (25 cm diameter, 50.5 cm length) cylindrical stainless-steel containers; each container would hold about 40 kg of salt waste. Three of these containers would be stacked in a larger (27 cm diameter, 155 cm length) cylindrical stainless-steel disposal canister; each disposal canister would hold about 120 kg of salt waste. Each disposal canister would then be inserted into a cylindrical overpack to form a waste package. Thus, nine disposal canisters would be

needed for the Mark-IV salt and six disposal canisters would be needed for the Mark-V salt for a total of 15 waste packages. Each disposal canister is 27 cm in diameter and 155 cm long, yielding a total volume of approximately 1.3 m^3 (Lee et al. 2013, Section 4.3).

EMT of all the sodium-bonded spent fuel from the EBR-II and the FFTF followed by casting the metallic waste into circular ingots is expected to produce six canisters of HLW (DOE 2000c, Table 5-7). Each canister is 2 feet (0.61 m) in diameter and 10 feet (3 m) tall; the estimated total volume is 1.2 m^3 (DOE 2000c).

For the purposes of reporting quantities of the disposal waste forms associated with sodium-bonded spent fuels, the OWL team decided to report the quantities associated with EMT of all the EBR-II spent fuel (driver, experimental driver, and blanket) and the FFTF driver fuel as a single value for each disposal waste form. The data needed to estimate the quantity of each disposal waste form generated by each of the different sources and types of sodium-bonded spent fuel is not available. Furthermore, for the purposes of providing inventory information to those modeling the postclosure performance of the disposal waste forms associated with sodium-bonded spent fuel, it is not necessary to identify how much of each disposal waste form could be traced back to each source and type of sodium-bonded spent fuel. Therefore, in the OWL, quantities (e.g., mass, numbers of canisters, and volumes) of each disposal waste form associated with sodium-bonded spent fuel represent the amount of that disposal waste form generated by EMT of all the EBR-II spent sodium-bonded fuel and all the FFTF spent sodium-bonded fuel. Consequently, there are examples of numerical values that are the same for more than one disposal waste form. This is not because the disposal waste forms just happen to have the same values. The values are the same because they refer to the quantity of the disposal waste form associated with EMT of multiple sodium-bonded wastes, not just one of sodium-bonded wastes.

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6. DEVELOPMENT OF DATA FOR PFLOTRAN INPUT FILE

The OWL database contains most of the inventory data that are needed for performance assessment calculations, which is a set of analyses intended to demonstrate that a disposal site will comply with long-term safety requirements. One of the computer codes used in this type of analysis is PFLOTRAN, which is an open-source, state-of-the-art massively parallel subsurface flow and reactive transport code [PFLOTRAN, 2023]. This code requires input files that define the waste inventory; OWL has information that can be used to define the waste inventory, so it was decided to build within OWL the ability for PFLOTRAN users to query the OWL database and to create inventory-related input files formatted for use in PFLOTRAN. Some parameter values needed for the PFLTORAN input file were already available in OWL while other parameter values needed to be calculated using data already in OWL. The intention is for every piece of data needed for the PFLOTRAN input file to be in OWL so that a PFLOTRAN user can query the database and generate a PFLOTRAN input file for the wastes included in the PFLOTRAN analyses. This section describes the technical basis for the parameter values needed for the PFLOTRAN inventory input file that were not already available in the OWL database. This information is provided to enhance transparency and traceability for performance assessment calculations performed using PFLOTRAN.

6.1. Data Needs for PFLOTRAN Input File

In PFLOTRAN, the waste form is specified in the WASTE_FORM_GENERAL input deck card. For a glass waste, the following information is required within the MECHANISM block:

1. Name of waste (e.g., Hanford glass)
2. Specific surface area–- value and units
3. Matrix density – value and units
4. Species–- for each isotope
    a. Identification of isotope (e.g., ^{241}Am)
    b. atomic weight (grams/mole)
    c. decay constant (1/second) (equal to ln(2)/half-life (seconds))
    d. quantity of radionuclide (grams of species/gram bulk waste)
    e. instant release fraction (a number between 0 and 1)
    f. name of daughter (if it exists)

Several different waste form mechanisms are available in PFLOTRAN, but we decided to start with glass waste forms in developing the process of creating PFLOTRAN input files; input files for different waste forms can be developed in the future.

The “name” parameter is required for all mechanism types and should be unique. The “specific surface area” parameter is used for the glass mechanism (as well as other mechanisms) and specifies the specific surface area of the waste form bulk in units of area per mass. The “matrix density” parameter specifies the density of the bulk waste form in terms of mass per volume. The “species” parameters include the following information for each radionuclide species that the PFLOTRAN analyst wishes to include in the input file: the name of the isotope; its atomic

weight in grams/mole; its decay constant in s^{-1}, which is equal to the natural log of 2 divided by the half-life in seconds; the quantity of that radionuclide species in the waste in terms of grams of that radionuclide per gram of bulk waste; the instant release fraction (a number between 0 and 1 indicating what fraction of that radionuclide species is released immediately when the waste form begins to degrade; used only for spent nuclear fuel); and the name of the daughter radionuclide produced upon decay, if it exists, to account for radioactive decay and ingrowth.

Some of the parameter values needed for an inventory input file were already available in the OWL database while others required additional data or simple calculations or are to be provided by the PFLOTRAN analyst building the input file. Table 5 shows each parameter and indicates the source of that parameter value.

Table 5. Sources of Inventory Input File Parameter Values

Parameter	Value Already Available in OWL Database?	Comments
MECHANISM	N	Provided by PFLOTRAN analyst
NAME (of waste)	Y	
SPECIFIC SURFACE AREA	N	Additional information and calculations needed
MATRIX DENSITY	N	Additional information and calculations needed
SPECIES (for each waste)	Not applicable	
Name of Isotope	Y	
Formula Weight	Y	
Decay Constant	Y	Calculated by dividing ln(2) by the half-life in seconds.
Initial Mass Fraction	N	Additional information and calculations needed
Instant Release Fraction	N	Provided by PFLOTRAN analyst
Name of Daughter	Y	

6.2. Calculation of Parameter Values

The following sections explain how the parameter values for SPECIFIC SURFACE AREA, MATRIX DENSITY, and Initial Mass Fraction were calculated for each of the wastes that could be disposed of in a glass waste form.

6.2.1. Calcine Waste Vitrified Following Separation and Calcine Waste Vitrified Without Separation

The glass degradation model developed for the Yucca Mountain repository provides the density, surface area, volume, mass, and specific surface area for the HLW glass produced at the Defense Waste Processing Facility at Savannah River (BSC 2004). The canisters to be used for the glass produced by vitrifying the calcine waste (with or without separation) at Idaho National Engineering Laboratory are assumed to be similar to those used at the Defense Waste Processing Facility (DOE 2002a). Therefore, it is assumed that the parameter values provided in BSC (2004) for the glass waste produced by the Defense Waste Processing Facility can also be used for the glass produced by vitrifying the calcine waste (with or without separation). The values of the parameters are as follows (BSC 2004, Section 6.5.4):

- Matrix Density – 2,690 kg/m^3
- Surface Area – 4.74 m^2
- Volume – 0.626 m^3
- Mass of Glass – 1,682 kg
- Specific Surface Area – 2.8 x 10^{-3} m^2/kg

Based on the information provided in Section 2, the OWL database contains the number of glass canisters expected to be produced by vitrification of calcine without separation (12,000 canisters) and the number of glass canisters expected to be produced by vitrification of calcine following separation (1,190 canisters). The OWL database contains the total inventory of each reported radionuclide in the calcine waste as of the baseline date of 2016. Therefore, the initial mass fraction of each reported radionuclide in the calcine waste vitrified without separation can be calculated by dividing the mass of that radionuclide in the calcine waste by the mass of glass in a canister and the number of canisters. The results of that calculation are shown in Table 6. To determine the initial mass fraction of the HLW glass waste produced after separating the calcine waste into HLW and low-level waste, the inventory in Table A-29 of DOE (2002a), which gives the HLW radionuclide inventory in calcine waste after separation as of 2035, was divided by the mass of glass in a canister and the number of canisters. The results of that calculation are shown in Table 7.

Table 6. Initial Mass Fraction of Glass Waste Made from Calcine Without Separation as of 2016

Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction
^{60}Co	4.03E-08	^{144}Ce	2.16E-16	^{236}U	5.18E-04
^{63}Ni	8.35E-06	^{144}Pr	9.12E-21	^{237}U	5.42E-13
^{79}Se	8.43E-05	^{147}Pm	1.59E-08	^{238}U	1.75E-02

Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction
^{90}Sr	2.67E-03	^{151}Sm	1.62E-04	^{237}Np	5.11E-03
^{90}Y	6.75E-07	^{152}Eu	7.24E-08	^{238}Pu	2.28E-04
^{99}Tc	1.03E-02	^{154}Eu	2.75E-06	^{239}Pu	1.92E-03
^{106}Ru	8.26E-15	^{155}Eu	1.11E-07	^{240}Pu	3.54E-04
^{125}Sb	3.05E-09	^{230}Th	7.96E-07	^{241}Pu	1.95E-05
^{126}Sn	3.82E-04	^{231}Th	2.39E-14	^{242}Pu	4.89E-05
^{129}I	1.63E-05	^{233}Pa	1.73E-10	^{241}Am	1.24E-04
^{134}Cs	1.90E-09	^{232}U	6.34E-10	^{243}Am	2.63E-07
^{135}Cs	6.28E-03	^{233}U	5.61E-08	^{242m}Am	1.05E-08
^{137m}Ba	7.26E-10	^{234}U	1.87E-04	^{242}Cm	2.75E-11
^{137}Cs	4.76E-03	^{235}U	5.87E-03	^{244}Cm	5.17E-09

Table 7. Initial Mass Fraction of Glass Waste Made from Calcine After Separation as of 2035

Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction
^{3}H	1.86E-07	^{137m}Ba	5.30E-09	^{240}Pu	3.52E-03
^{14}C	3.12E-09	^{137}Cs	3.46E-02	^{241}Pu	9.17E-05
^{60}Co	1.41E-08	^{226}Ra	4.90E-09	^{242}Pu	4.31E-04
^{90}Sr	2.55E-02	^{230}Th	9.70E-06	^{241}Am	1.89E-03
^{90}Y	6.44E-06	^{232}Th	4.50E-07	$^{242/242m}$Am	7.16E-10
^{93}Nb	9.85E-07	^{232}U	1.03E-10	^{243}Am	3.51E-08
^{94}Nb	1.44E-08	^{233}U	6.74E-08	^{242}Cm	1.81E-12
^{99}Tc	9.93E-02	^{234}U	8.03E-03	^{243}Cm	4.61E-12
^{102}Rh	1.62E-15	^{235}U	1.36E-01	^{244}Cm	6.18E-11
^{106}Ru	1.66E-19	^{236}U	1.16E-02	^{245}Cm	1.07E-11
^{126}Sn	3.61E-03	^{238}U	4.31E-02	^{246}Cm	1.42E-13
^{129}I	1.59E-02	^{237}Np	4.48E-03	^{247}Cm	1.67E-16
^{134}Cs	1.28E-11	^{238}Pu	2.63E-03	^{248}Cm	1.13E-18
^{135}Cs	6.95E-02	^{239}Pu	1.45E-02		

6.2.2. Vitrified Hanford Tank Wastes

The glass degradation model developed for the Yucca Mountain repository provides the matrix density, surface area, volume, mass, and specific surface area for the expected Hanford HLW glass (BSC 2004). The values of the parameters in that glass degradation model are as follows (BSC 2004, Section 6.5.4):

- Matrix Density – 2,700 kg/m^3

- Surface Area – 8.5 m^2
- Volume – 1.19 m^3 (calculated by dividing the mass of glass by the matrix density)
- Mass of Glass – 3,210 kg
- Specific Surface Area – 2.6 x 10^{-3} m^2/kg

The initial mass fraction of the vitrified tank wastes cannot be calculated for individual radionuclides by dividing the known tank radionuclide inventory, which is in OWL, by the number of expected glass waste forms because some radionuclides will be removed from the waste stream as it goes from the tanks to the vitrification plant. Not enough is known about which radionuclides will be removed and the quantities in which they will be removed to provide a basis for that calculation, Therefore, the initial radionuclide mass fraction was calculated from the inventory information provided in the license application for the proposed Yucca Mountain repository (DOE 2008), which is in terms of curies per canister as of 2017. To calculate the expected initial radionuclide fraction in the vitrified HLW glass to be produced at Hanford, the grams per canister of each radionuclide was calculated based on the curies per canister information in DOE (2008). The mass fraction (grams per gram) of each radionuclide was then calculated by dividing the grams per canister of each radionuclide by the mass of glass in a canister, 3,210 kg, and making the appropriate unit conversions. The results of that calculation are shown in Table 8.

Table 8. Radionuclide Initial Mass Fraction of Glass Waste Made from Hanford Tank Waste as of 2017

Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction
^{241}Am	4.19E-05	^{238}Pu	3.95E-08	^{229}Th	2.22E-12
^{243}Am	1.56E-07	^{239}Pu	1.07E-04	^{233}U	6.79E-08
^{137m}Ba	3.26E-11	^{240}Pu	8.81E-06	^{234}U	7.31E-07
^{244}Cm	1.26E-09	^{241}Pu	2.62E-07	^{235}U	8.02E-05
^{60}Co	1.14E-10	^{242}Pu	7.84E-08	^{236}U	5.69E-06
^{137}Cs	2.14E-04	^{79}Se	2.05E-06	^{238}U	9.36E-03
^{93m}Nb	4.31E-09	^{151}Sm	4.06E-05	^{90}Y	3.56E-08
^{59}Ni	1.94E-06	^{126}Sn	1.45E-05	^{93}Zr	7.52E-04
^{63}Ni	2.72E-07	^{90}Sr	1.41E-04		
^{237}Np	1.11E-04	^{99}Tc	4.21E-04		

The current reference process for the eleven tanks containing remote-handed transuranic waste (RH-TRU) is to commingle it with the HLW and vitrify it in the vitrification facility. Therefore, this waste is also accounted for in Table 8 (SNL 2014).

6.2.3. Vitrified Cesium and Strontium Capsules

The current plan for treating the 1,335 CsCl and 601 SrF_2 capsules at Hanford is to remove the cesium and strontium from the capsules and prepare the cesium and strontium for vitrification at the Waste Treatment Plant, which is also being used to vitrify the tank wastes discussed in Section 6.2.2 (DOE, 2012). It is estimated that this process would produce 340 canisters. Because the cesium and strontium in the capsules will be treated using the same plant and process that will be used to treat the tank waste at Hanford, it is assumed that the glass produced from vitrifying the contents of the capsules would have the same matrix density, surface area, volume, mass of glass, and specific surface area as the glass produced from vitrifying the Hanford tank waste (Section 6.2.2):

- Matrix Density – 2,700 kg/m^3
- Surface Area – 8.5 m^2
- Volume – 1.19 m^3 (calculated by dividing the mass of glass by the matrix density)
- Mass of Glass – 3,210 kg
- Specific Surface Area – 2.6 x 10^{-3} m^2/kg

The average initial mass fraction of each radionuclide in the waste generated by vitrifying the capsule contents can be calculated by dividing the total mass of each radionuclide in all the capsules (as of 2016, given in OWL) by the number of glass canisters expected to be produced (340) and the mass of glass in a canister (3,210 kg), making the appropriate unit conversions. The results of that calculation are shown in Table 9.

Table 9. Radionuclide Initial Mass Fraction of Glass Waste Made from Strontium and Cesium Capsules as of 2016

Radionuclide	Initial Mass Fraction
^{90}Sr	9.47E-05
^{90}Y	2.40E-08
^{137m}Ba	5.42E-11
^{137}Cs	3.54E-04
^{135}Cs	3.91E-05

6.2.4. "German" Glass

In 1987, Pacific Northwest National Laboratory produced 30 heat and radiation source canisters for the Federal Republic of Germany (FRG). These sources contained ^{90}Sr and ^{137}Cs isotopes in a borosilicate glass and were intended to be used in the Asse salt mine (Brouns and Powell, 1988). However, the FRG never took possession of the source canisters, and they are now stored at Hanford and officially classified as remote-handled transuranic waste because of some contamination by transuranic isotopes (DOE, 1997). In addition to the 30 canisters filled with strontium and cesium, two production demonstration canister and two instrumented canisters were produced (DOE, 1997); the two production demonstration canisters are thought to contain depleted uranium and natural thorium but no strontium or cesium.

The average matrix density, average volume, and average mass of glass are given in Table 6.4 of Holton et al. (1989). The surface area of the glass and the specific surface area were not readily available in existing reports, so to calculate these values, the wall thickness was assumed to be 10 cm, the canister top lid was assumed to be 20 cm, the bottom lid thickness was assumed to be 10 cm, and the fraction of inner volume filled with glass was estimated to be 85% (Holton et al. 1989). The canister outer height (1.2 m) and outer diameter (0.30 m) are known (SNL, 2014). The matrix density, surface area, volume, mass of glass, and specific surface area for this glass are:

- Matrix Density – 2,610 kg/m^3
- Surface Area – 1.0 m^2
- Volume – 60.6 liters
- Mass of Glass – 158.3 kg
- Specific Surface Area – 6.3 x 10^{-3} m^2/kg

The average initial mass fraction of each radionuclide in the glass can be calculated by dividing the total mass of each radionuclide in all the canisters as of 1987 (given in OWL) by the number of glass canisters (32) and by the average mass of glass in a canister (158.3 kg), making the appropriate unit conversions. The results of that calculation are shown in Table 10.

Table 10. Radionuclide Initial Mass Fraction of "German" Glass Waste as of 1987

Radionuclide	Initial Mass Fraction
^{90}Sr	5.31E-03
^{90}Y	1.340E-06
^{137m}Ba	1.77E-09
^{137}Cs	1.15E-02
^{135}Cs	8.43E-03

6.2.5. Ceramic Waste from EMT

As discussed in Section 0, treatment of sodium-bonded spent fuel via EMT will produce a salt waste that will, in turn, be treated to produce a ceramic HLW form suitable for disposal. The total mass of ceramic waste form expected to be created from all the different reactors that generated sodium-bonded spent fuel (EBR-II and FFTF) is 50,950 kg (Ebert, 2005). As reported by Ebert (2005) the density of the ceramic waste form will be about 2,000 kg/m^2. Further, as reported by Ebert (2005, Section IV.D.2)), the ceramic waste forms will be cylinders about 1 m tall and 0.5 m in diameter, and the surface area of each waste form will be about 2 m^2. Assuming two ceramic waste forms can be placed in a waste package, the surface area of ceramic waste per waste package is 4 m^2. Each ceramic waste form will have a volume of about 0.2 m^2 and have a mass of about 400 kg (SNL, 2014). Assuming two ceramic waste forms per waste package, the volume and mass of the ceramic waste form per waste package are 0.4 m^2 and 800 kg, respectively. The specific surface area of the ceramic waste form in a waste package can then be calculated by dividing the surface area (4 m^2) by the mass of glass (800 kg), yielding 5.0 x 10^{-3} m^2/kg. In summary, the values of the parameters needed by PFLOTRAN for a waste package containing this ceramic HLW form are:

- Matrix Density – 2,000 kg/m^3
- Surface Area – 4.0 m^2
- Volume – 0.4 m^2
- Mass of Glass – 800 kg
- Specific Surface Area – 5.0 x 10^{-3} m^2/kg

The average initial mass fraction of each radionuclide can be calculated by taking the inventory reported in Table A-1 of Ebert (2005), which is in grams and is projected to 2040, and dividing by the total mass of ceramic waste expected to be produced, 50,950 kg (Ebert, 2005). Note that Table A-1 does not give a mass for ^{137m}Ba, but as it is produced by decay of ^{137}Cs, it is present. The mass of ^{137m}Ba present was calculated by assuming it is in secular equilibrium with ^{137}Cs. The results of the calculations described above are shown in Table 11.

Table 11. Radionuclide Initial Mass Fraction of Ceramic Waste as of 2040

Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction
Am-241	2.45E-06	Pu-239	5.12E-03
Am-243	8.56E-11	Pu-240	1.12E-04
Ba-137m	2.19E-11	Pu-241	1.61E-06
Cf-249	4.04E-22	Pu-242	3.45E-07
Cf-251	3.81E-26	Sm-151	1.32E-05
Cl-36	2.73E-20	Sr-90	7.87E-05
Cm-244	1.13E-10	Th-229	1.02E-12
Cm-245	2.51E-12	U-233	4.14E-10
Cm-246	1.31E-14	U-234	3.87E-06
Cm-247	4.06E-17	U-235	1.05E-03

Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction
Cm-248	1.62E-19	U-236	2.59E-05
Cs-137	1.43E-04	U-238	2.98E-03
Np-237	2.41E-05	Y-90	1.98E-08
Pu-238	6.63E-07		

6.2.6. Glass Waste Produced at Savannah River

The glass degradation model developed for the Yucca Mountain repository provides the density, surface area, volume, mass, and specific surface area for the HLW glass produced at the Defense Waste Processing Facility at Savannah River (BSC 2004). The values of the parameters are as follows (BSC 2004, Section 6.5.4):

- Matrix Density – 2,690 kg/m^3
- Surface Area – 4.74 m^2
- Volume – 0.626 m^3
- Mass of Glass – 1,682 kg
- Specific Surface Area – 2.8 x 10^{-3} m^2/kg

The average initial mass fraction of the radionuclides in the glass waste produced to-date (as of 2018) can be calculated by taking the mass of each radionuclide in all the glass waste produced to-date, which is in OWL, and dividing by the number of canisters to-date (4,125, also in OWL) and by the mass of glass in each canister (1,682 kg) and making the appropriate unit conversions. The results of that calculation are shown in Table 12.

Table 12. Radionuclide Initial Mass Fraction of Savannah River Glass Waste as of 2018

Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction
Am-241	2.86E-06	Pu-238	2.96E-06
Am-242m	3.75E-09	Pu-239	7.80E-05
Am-243	2.49E-06	Pu-240	7.25E-06
Ba-137m	4.43E-13	Pu-241	1.39E-07
Bk-247	6.91E-10	Pu-242	7.88E-07
Cf-249	1.01E-09	Se-79	6.96E-07
Cf-251	5.32E-09	Sm-151	2.40E-06
Cl-36	9.95E-07	Sn-121m	6.99E-09
Cm-244	1.78E-07	Sn-126	7.30E-06
Cm-245	2.97E-08	Sr-90	2.13E-05
Cm-246	3.37E-08	Tc-99	4.98E-06
Cm-247	2.70E-05	Th-229	2.07E-10
Cm-248	6.74E-07	U-233	1.86E-06

Radionuclide	Initial Mass Fraction	Radionuclide	Initial Mass Fraction
Co-60	1.22E-10	U-234	2.85E-06
Cs-137	3.12E-06	U-235	1.01E-04
Nb-93m	2.94E-10	U-236	5.75E-06
Ni-59	4.27E-06	U-238	2.16E-02
Ni-63	4.52E-07	Y-90	5.39E-09
Np-237	1.70E-05	Zr-93	5.31E-05

The Defense Waste Processing Facility continues to produce glass waste from the HLW in the tanks. In the absence of information regarding the composition of projected glass waste, it is reasonable to assume that the parameters describing the yet-to-be-produced glass (e.g., matrix density) and the initial mass fraction of radionuclides in the yet-to-be-produced glass will be the same as they are for the already-existing glass waste. Therefore, the above parameters and initial mass fractions can be used for the yet-to-be-produced glass waste.

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