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On-Line Waste Library V4.0 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.

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 m³ (160,000 ft³) 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 ft³ of waste. The average thermal output of each canister is about 78 watts. The total volume of glass would be 26,238 ft³ (743 m³) and the total volume of waste (including canisters) would be approximately 37,000 ft³ (1,060 m³). 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 ft³ (1.36 m³), 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 m³ of additive would be mixed with the waste for a combined volume of material to be treated of 5,500 m³. Under these assumptions, the 5,500 m³ 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 ft³.

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 ft³ (1.36 m³), 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 m³ 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 ft³.

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 ft³ (13,000 m³) (Table 3-2, DOE 2002a) and the total volume (including canisters) would be approximately 570,000 ft³. 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 ft³ (8,500 m³) (Table 3-2, DOE 2002a) and the total volume (including canisters) would be approximately 380,000 ft³. 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 ft³ (4,400 m³) (Table 3-2, DOE 2002a) and the total volume (including canisters) would be approximately 190,000 ft³. 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) (ft³)	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
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 SrF₂ 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 ft³ ($\pi \times (1 \text{ foot})^2 \times 15 \text{ feet} \times 340 \text{ packages}$).

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 ($((118 \times 1335) + (158 \times 601))/340$) 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 ft³ ($\pi \times (0.36 \text{ ft})^2 \times 15.6 \times 108$).

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.

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)
$^{125\text{m}}\text{Te}$	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)
$^{126\text{m}}\text{Sb}$	19.15 minutes	^{126}Sn	1	218,000 years	^{126}Sb
$^{137\text{m}}\text{Ba}$	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
^{218}Po	3.097 minutes	^{226}Ra	1	1600 years	^{214}Pb

Radionuclide	Half-life	Parent with which the radionuclide is in secular equilibrium	Branching Fraction	Parent Half-life	Daughter Radionuclide
^{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	$^{242\text{m}}\text{Am}$	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	$^{242\text{m}}\text{Am}$	0.995	141 years	^{242}Cm
^{242}Cm	162.86 days	$^{242\text{m}}\text{Am}$	0.83	141 years	^{238}Pu

5. The inventory of ^{241}Am ($t_{1/2} = 432.7$ years) includes production of ^{241}Am by decay of ^{241}Pu ($t_{1/2} = 14.4$ 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

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

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:

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

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 , $^{137\text{m}}\text{Ba}$, ^{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 1.602×10^{-19} (J/eV), multiplied by 3.7×10^{10} decays/second/Ci, and multiplied by 1000 Ci/kCi:

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

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		

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% ²³⁵U 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 ²³⁹Pu, 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 3 (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 electrorefiner 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 electrorefiners 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 m³ (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 m³ (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 m³ (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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