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On-Line Waste Library 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.

ACKNOWLEDGMENTS

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TABLES

NOMENCLATURE

Abbreviation	Definition
DOE	United States Department of Energy
HIP	Hot isostatic pressing
HLW	high-level waste
INL	Idaho National Laboratory
NNDC	National Nuclear Data Center
OWL	On-Line Waste Library
SNF	spent nuclear fuel

CHANGE HISTORY

Version	Changes

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, and Section 4 provides supporting information for calculating radionuclide decay and production and calculating the heat generated by radioactive decay.

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
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 ($\pi \times (0.36\ ft)^2 \times 15.6 \times 108$).

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 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.

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 radionuclides are in secular equilibrium:
    a. ^{137m}Ba ($t_{½}$ = 2.552 minutes) is in equilibrium with ^{137}Cs ($t_{½}$ = 30.08 years). The activity of ^{137m}Ba is 95% of the activity of ^{137}Cs.
    b. ^{90}Y ($t_{½}$ = 64 hours) is in secular equilibrium with ^{90}Sr ($t_{½}$ = 28.9 years). The activity of ^{90}Y is equal to that of ^{90}Sr.
    c. ^{144}Pr ($t_{½}$ = 17.28 minutes) is in secular equilibrium with ^{144}Ce ($t_{½}$ = 284.91 days). The activity of ^{144}Pr is 98.9% of the activity of ^{144}Ce.
    d. ^{106}Rh ($t_{½}$ = 30.1 seconds) is in secular equilibrium with ^{106}Ru ($t_{½}$ = 1.02 years). The activity of ^{106}Rh is equal to that of ^{106}Ru.
    e. ^{126m}Sb ($t_{½}$ = 11 seconds) is in secular equilibrium with ^{126}Sn ($t_{½}$ = 230,000 years). The activity of ^{126m}Sb is equal to that of ^{126}Sn.

f. ^{126}Sb ($t_{½}$ = 12.35 days) is in secular equilibrium with ^{126}Sn ($t_{½}$ = 230,000 years). The activity of ^{126}Sb is equal to that of ^{126}Sn.

g. ^{125m}Te ($t_{½}$ = 57.4 days) is in equilibrium with ^{125}Sb ($t_{½}$ = 2.76 years). The activity of ^{125m}Te is 9% of the activity of ^{125}Sb.

h. ^{208}Tl ($t_{½}$ = 3.05 minutes) is in secular equilibrium with ^{232}U ($t_{½}$ = 68.9 years). The activity of ^{208}Tl is 35.94% that of ^{232}U.

i. ^{238}Np ($t_{½}$ = 2.117 days) is in secular equilibrium with ^{242m}Am ($t_{½}$ = 141 years). The activity of ^{238}Np is 0.5% of the activity of ^{242m}Am.

j. ^{239}Np ($t_{½}$ = 2.4 days) is in secular equilibrium with ^{243}Am ($t_{½}$ = 7,370 years). The activity of ^{239}Np is equal to that of ^{243}Am.

k. ^{242}Am ($t_{½}$ = 16.02 hours) is in secular equilibrium with ^{242m}Am ($t_{½}$ = 141 years). The activity of ^{242}Am is 99.5% of the activity of ^{242m}Am.

l. ^{242}Cm ($t_{½}$ = 162.8 days) is in secular equilibrium with ^{242m}Am ($t_{½}$ = 141 years). The activity of ^{242}Cm is 83% of the activity of ^{242m}Am.

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 2. 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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