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Defense High-Level Waste and Spent Fuel Characterization for Geologic Waste Repositories

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DEFENSE HIGH-LEVEL WASTE AND SPENT FUEL CHARACTERIZATION
FOR GEOLOGIC WASTE REPOSITORIES

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ABSTRACT

Results of SANDIA-ORIGEN calculations are given for the total thermal output and radionuclide content, up to 100 000 years decay time, for defense high-level waste expected from the Savannah River Plant, the Hanford Reservation and the Idaho National Engineering Laboratory, and for one pressurized water reactor and five boiling water reactor spent fuel types. These results are expected to be of value in evaluating the potential for storing defense high-level waste and/or spent fuel in geologic waste repositories.

CONTENTS

	<u>Page</u>
Abbreviations	6
Introduction	7
Defense High-Level Waste	8
Light Water Reactor Spent Fuel	11
Summary	13
Tables	14
Figures	25
References	28
APPENDIX -- Reading a SANDIA-ORIGEN Fiche	29
Microfiches -- Inside back cover	

ABBREVIATIONS

BWR	Boiling Water Reactor
DHLW	Defense High-Level Waste
GWR	Geologic Waste Repository
HR	Hanford Reservation
INEL	Idaho National Engineering Laboratory
LWR	Light Water Reactor
MTHM	Metric Tonne of Heavy Metal
MWd	Megawatt-days
PWR	Pressurized Water Reactor
SRP	Savannah River Plant

DEFENSE HIGH-LEVEL WASTE AND SPENT FUEL CHARACTERIZATION
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Introduction

Disposal of radioactive waste in geologic formations is being considered for the safe and permanent disposal of transuranic radioactive waste that has been generated in national defense programs over the past four decades. Current candidates for the geologic media in which radioactive waste repositories could be located include bedded salt, basalt, and granite.

Transuranic waste is process materials and trash contaminated with transuranic nuclides and their decay products (generally defined as > 10 nCi/g alpha activity). Most of this waste generates negligible quantities of decay heat and emits little damaging radiation. The principal design concern for disposing of such waste in a geologic waste repository (GWR) is immobilization of the radionuclides for a long time (up to 250 000 yr).

Two other types of radioactive materials currently being generated in the United States that eventually might be emplaced in a repository are defense high-level waste (DHLW) and commercial light water reactor (LWR) spent fuel. DHLW is currently in temporary storage at a number of locations with most of the waste at three sites:

1. the Savannah River Plant (SRP) near Aiken, SC
2. the Hanford Reservations (HR) near Richland, WA
3. the Idaho National Engineering Laboratory (INEL) near Idaho Falls, ID.

The LWR spent fuel consists of fuel assemblies from pressurized water reactors (PWRs) and boiling water reactors (BWRs). While LWR spent fuel is not normally considered a waste, it is being considered for direct emplacement in a repository without first being reprocessed. This spent fuel is currently being stored in spent-fuel pools located at the reactor plant sites.

One alternative for the permanent disposal of DHLW and LWR spent fuel is emplacement in a GWR. An assessment of the feasibility, suitability, and safety of this means of disposal for DHLW and LWR spent fuel requires, as a prerequisite, an estimation of the impact of storing these materials in a GWR.

The purpose of this report is to provide data necessary to obtain such an estimation. These data include heat generation rates, radionuclide content, and neutron and gamma source strengths as a function of decay time of DHLW and spent fuel. It is assumed in this study that the DHLW has been immobilized in a matrix (e.g., glass, ceramic) and the resulting mix is canistered prior to shipment to the GWR. LWR spent fuel would also be canistered with probably one PWR or one to three BWR assemblies per canister.

Defense High-Level Waste

Background

This report describes the analysis made to characterize the DHLW presently stored at SRP, HR, and INEL and provides estimates of the characteristics of waste which may be generated in the future. What may be termed the "worst" or "hottest" waste expected from each of the three sites will be described in this document. While much of the DHLW which may be sent to the GWR would have less "severe" characteristics than the "worst" case waste, the great uncertainties presently apparent in the characterization of on-hand DHLW and in waste concentration upon

immobilization provide a basis of justification for designing and analyzing the GWR with the "worst" case wastes as the basis.

The DHLW presently at SRP, HR, and INEL is stored in a number of forms. These include salts, sludges, liquids, and calcines. Table 1 lists estimates as of 1977 of the inventories of each waste form at the three sites.^{1 2 3} Sixty-seven percent of the DHLW at these locations is stored at HR and all the calcine waste (1% of the total inventory) is stored at INEL. The DHLW is contained at these sites in large volume (100 000s to 3 800 000 litre capacity) steel-lined tanks. Before being transported to the GWR, the DHLW at each site would be removed from the tanks, dried if necessary, immobilized (with the possible exception of calcine waste) in some as yet unspecified matrix, and canistered. Possible matrices include borosilicate glass, various ceramics, and metals.

Canister designs which would be acceptable at the GWR have yet to be determined. SRP has developed a reference storage canister concept⁴ which would be fabricated from Type 304-L stainless steel. The canister would be 60 cm (24 in.) in outer diameter with a wall thickness of 0.953 cm (0.375 in.) and a height of 2.4 m (8 ft). The canister would be filled with 0.63 m³ (22.1 ft³) of waste which would fill the canister to a height of 2.3 m (7.5 ft). For the purposes of this study, all DHLW is assumed to be contained in a SRP-type storage canister to obtain a comparison between the waste expected from each of the three storage sites.

Data Analysis and Results

The estimated radionuclide content of "worst" case waste expected from each of the DHLW sites differ considerably. Tables 2, 3, and 4 list estimates of the quantities in curies per cubic metre of DHLW of the more prominent radionuclides in SRP,⁴ INEL,⁵ and HR⁶ waste. The SRP and HR DHLW are assumed to be in a glass matrix, while INEL waste is considered in calcine form. These data were obtained from on-going studies by SRP, HR, and INEL personnel which are intended to characterize, in the short term, both the waste on hand and that expected to be generated in the future at each site.

Based on the information given in Tables 2, 3, and 4, a comparison of the quantities of radionuclides per cubic metre common to each site indicates that SRP produces the most radioactive DHLW. For instance, for Ru-106 the curies per cubic metre (Ci/m^3) are $2.54 \pm 3^*$, 1.3 ± 3 , and 5.48 ± 2 for SRP, INEL, and HR waste, respectively. The same trend is apparent for the high heat generating isotopes of Sr and Cs, except in this case the quantity of these elements at HR is even more depressed. The reason the HR waste is so deficient in Sr and Cs is that much of these elements has been chemically removed from the DHLW and encapsulated separately as CsCl and SrF_2 . The trend however is not maintained when the transuranics are considered. SRP DHLW still has the greatest quantity of the radionuclides per unit volume but is now followed by HR waste and then the INEL calcine. As an example, for Pu-239 the Ci/m^3 of waste are 3.10 ± 1 , 6.93 ± 0 , and 9.1 ± 1 for SRP, HR, and INEL waste, respectively. Obviously, the relationships as described above between the waste expected from each site may change in the event that one or more of the generating sites varies the concentration of radioactive material in its waste product.

The SANDIA-ORIGEN code⁷ was utilized to estimate the radionuclide and curie content, and heat generation rates associated with the "worst" case DHLW expected from each site. This code is a point code (no spatial dependence) which calculates detailed isotopic compositions as a function of time for a wide range of problems involving nuclear reactor fuel irradiation, neutron activation, and radioactive decay. The code uses a nuclear data file of 1062 isotopes. For this study, the wastes from the three sites were characterized for ages up to 10^5 yr, measured from the creation date of the solidified waste form.

Tables 5, 6 and 7 list the SANDIA-ORIGEN results for the heat generation rates in watts/cubic metre and watts/canister for DHLW from SRP, INEL, and HR, respectively. The heat generation rate data are

$$*2.54 \pm 3 \approx 2.54 \times 10^3$$

summarized graphically in Figure 1. As alluded to above, it is evident that of the three sites DHLW from SRP has the highest heat generation rate per unit volume for all times. For early times (up to approximately 200 yr), INEL waste has the second highest heat generation rate but for later times, it is surpassed by HR waste. These results reflect the fact, as mentioned before, that much of the Cs and Sr has been removed from HR DHLW and that the transuranic content of HR waste exceeds that of INEL waste.

The total curie content per cubic metre of waste and per SRP-type canister for SRP, INEL, and HR DHLW are summarized in Tables 8, 9, and 10 and in Figure 2. The curie content by individual isotope may be obtained from the microfiche at the end of this report.

The uncertainty in the integrated thermal (decay) power and total curie content for DHLW is estimated to be of the order of 10%, given the initial nuclide concentrations as a starting point for the calculations. Higher uncertainties would be expected for individual isotopes. No estimate is made of the uncertainties in the initial waste compositions.

Light Water Reactor Spent Fuel

Background

The determination of the thermal and radionuclide characteristics of LWR spent fuel needed for the analysis of spent-fuel interment, in a GWR is complicated by the wide variety of LWR spent-fuel types generated in the nuclear power industry. The irradiation histories of individual spent-fuel assemblies can be reasonably well characterized. Therefore, since the spent-fuel assemblies will be treated on an individual basis (one or a few per canister), the use of a single set of typical LWR spent-fuel characteristics for evaluating spent-fuel emplacement in a GWR may be inappropriate. In the following analysis a number of "typical" spent-fuel types were examined.

Data and Analysis

The parameters describing the irradiation of one PWR and five BWR typical fuel types are summarized in Table 11. These parameters were obtained by using current, standardized nuclear power plant designs and incore fuel management plans. The typical PWR irradiation cycle analyzed involves a 3-yr, 3-cycle exposure of 3.3 w/o* ^{235}U enriched fresh fuel. The total burnup in this case is approximately 15 000 megawatt-days (MWd)/assembly or 33 000 MWd/metric ton of heavy metal (MTHM). Typical BWR spent fuel has fresh fuel enrichments of 2.66 w/o or 2.83 w/o ^{235}U and undergoes three or four irradiation cycles in the reactor. Four cycle BWR fuel experiences widely varying power levels (48% or 96% of the core average power) during the final cycle. Total burnups ranging from 4500 to 5800 MWd/assembly or 24 600 to 30 600 MWd/MTHM are thus obtained.

Table 12 presents the fresh fuel compositions assumed for each spent-fuel type. The compositions given include the cladding and structural materials contained in fresh fuel assemblies in addition to the uranium content.

The SANDIA-ORIGEN code was utilized to model the irradiation and subsequent radioactive decay of all six spent-fuel types. The calculated thermal power (up to 100 000 yr decay time) in watts per fuel assembly for the six spent-fuel types are listed in Table 13. In the table the terms "low" and "high" burnup refer to the percent of the core average power (48% and 96%, respectively) experienced by the fuel during the final irradiation cycle. Figure 3 presents the time variation of decay thermal power for the PWR fuel, the 4-cycle, 2.66 w/o ^{235}U , "low" burnup BWR fuel and the 4-cycle, 2.83 w/o ^{235}U , "high" burnup BWR fuel. The thermal history projections of the other BWR fuel types, if plotted, would be enclosed in the envelope formed by the two BWR histories shown. From the figure it is apparent that typical PWR spent fuel is characterized by a higher thermal output for all times than typical BWR spent fuel. This is due to the higher initial fuel concentration, greater fissile and fertile

*w/o = weight percent

material content, and higher burnups experienced by PWR fuel relative to typical BWR fuel. Also, from Table 13 and Figure 3 it is seen that the thermal output of the PWR spent fuel exceeds that of all the BWR types by a factor of 2.5 to 3.5 for cooling times greater than 4 yr. Thus, the thermal output of a spent-fuel canister containing two or three BWR assemblies may be considered essentially equivalent to that of a canister with one PWR assembly.

The radionuclide content of LWR spent fuel is too extensive to list here. This information may be obtained from the microfiche located at the end of this report.

For LWR spent fuel, uncertainties in the range of 10% to 20% percent would be expected for integrated or total quantities.⁷ Specific concentrations of less common isotopes would have higher uncertainties due to uncertainties in the neutron flux, fission yields, and neutron cross sections.

Summary

In conclusion, reasonably accurate characterizations of the radionuclide content of and heat generation rates associated with DHLW expected from SRP, INEL, and HR and for one PWR and five BWR typical spent-fuel types have been obtained for ages from 1 through 100 000 yr decay time. The characterizations have been based on the best information available concerning the current radiolytic composition of DHLW at the three sites and on realistic LWR fuel cycle data. Subsequent redefinition of the radionuclide composition of waste expected from defense program sites probably will not significantly alter the DHLW results reported in this document, since the "hottest" DHLW was used as the basis.

Table 1

Waste Quantities of DHLW
(millions of gallons)

<u>Site</u>	<u>Salts</u>	<u>Sludges</u>	<u>Liquid Waste</u>	<u>Calcline Waste</u>
SRP	13.3	3.4	5.6	--
Hanford	25.0	11.0	11.0	--
INEL	--	--	2.9	0.33

Table 2

Radionuclide Content of Savannah River
Defense High-Level Waste in Glass Form

<u>Isotope</u>	<u>Curies/m³**</u>	<u>Isotope</u>	<u>Curies/m³</u>
Co-60	3.96 + 2*	Pu-147	4.63 + 4
Se-79	3.39 - 1	Sm-151	1.35 + 2
Sr-90	7.51 + 4	Eu-152	5.98 + 0
Y-90	7.83 + 4	Eu-154	3.26 + 3
Y-91	2.72 - 4	Eu-155	3.72 + 2
Zr-91	2.40 + 0	Tl-208	1.82 + 0
Zr-95	3.04 - 3	U-232	1.49 + 0
Nb-95	6.62 - 3	U-233	6.39 - 7
Ru-106	2.54 + 3	U-234	4.23 - 2
Rh-106	2.56 + 3	U-235	2.14 - 5
Pd-107	2.27 - 2	U-236	2.72 - 4
Ag-110	3.99 + 1	U-238	2.14 - 4
Sn-117m	1.43 - 1	Np-237	1.31 - 3
Sn-121m	5.19 + 1	Pu-236	1.63 + 1
Sn-123	2.80 - 1	Pu-238	2.78 + 3
Sn-126	3.04 - 2	Pu-239	3.10 + 1
Sb-125	1.43 + 3	Pu-240	2.43 + 1
Sb-126m	1.38 - 1	Pu-241	4.43 + 3
Sb-126	1.38 - 1	Pu-242	3.45 - 2
Te-125m	3.31 + 2	Am-241	4.84 + 1
Te-127m	5.54 - 2	Am-242	3.56 - 2
Te-127	5.63 - 2	Am-242m	3.56 - 2
Cs-134	2.21 + 4	Am-243	1.12 - 1
Cs-135	1.92 - 1	Cm-242	7.80 - 2
Cs-137	8.49 + 4	Cm-243	2.89 - 2
Ba-137m	8.18 + 4	Cm-244	3.15 + 0
Ce-142	2.54 - 5	Cm-245	1.44 - 4
Ce-144	1.74 + 4	Cm-246	1.34 - 5
Pr-144	1.77 + 4		

* $3.96 + 2 = 3.96 \times 10^2$

**Ci/m³ of the final product

Table 3

Radionuclide Content of INEL Defense
High-Level Waste in Calcine Form

<u>isotope</u>	<u>Curies/m³</u>	<u>isotope</u>	<u>Curies/m³</u>
Ni-59	8.3 ± 2	Pm-147	1.6 ± 4
Rb-87	4.7 ± 6	Sr-90	2.2 ± 2
Sr-90	1.7 ± 4	Ku-154	2.3 ± 2
Y-90	1.7 ± 4	U-233	1.6 ± 9
Zr-93	4.0 ± 1	U-234	5.6 ± 1
Nb-93m	9.8 ± 2	U-235	2.3 ± 6
U-99	2.7 ± 0	U-236	1.3 ± 5
Ku-106	1.3 ± 3	U-237	6.2 ± 9
Ru-106	1.3 ± 3	U-238	1.3 ± 11
Pd-107	2.6 ± 3	Np-237	6.2 ± 5
Sn-126	4.2 ± 2	Pu-238	9.1 ± 1
Nb-126m	4.2 ± 2	Pu-239	9.1 ± 1
Nb-126	4.2 ± 2	Pu-240	6.5 ± 1
CS-134	4.3 ± 3	Pu-241	2.1 ± 2
CS-135	9.8 ± 2	Pu-242	2.3 ± 3
CS-137	1.7 ± 4	Am-241	1.3 ± 0
Ba-137m	1.6 ± 4	Am-243	1.1 ± 2
Co-144	1.1 ± 6	Cm-242	8.5 ± 1
Pr-143	1.1 ± 4	Cm-244	6.8 ± 1

^a Ci/m³ of the final product

Table 4

Radionuclide Content of Hanford Defense
High-Level Waste in Glass Form

Isotope	Curies/m ³ ^a	Isotope	Curies/m ³ ^a
U-14	4.46 $\pm$ 1	Ce-144	7.99 $\pm$ 2
Ni-59	3.35 $\pm$ 2	Pm-147	1.65 $\pm$ 5
Ni-63	2.21 $\pm$ 1	Sm-151	3.45 $\pm$ 2
Co-60	5.23 $\pm$ 1	Eu-152	1.55 $\pm$ 6
Se-79	4.85 $\pm$ 1	Eu-154	4.15 $\pm$ 2
Si-99	1.50 $\pm$ 3	Eu-155	3.95 $\pm$ 1
Zr-93	4.66 $\pm$ 3	U-235	9.50 $\pm$ 3
Ti-99	1.81 $\pm$ 1	Np-237	7.60 $\pm$ 2
Ru-106	5.48 $\pm$ 2	Po-238	1.16 $\pm$ 6
Pd-107	5.60 $\pm$ 2	Po-239	6.95 $\pm$ 5
Ag-110m	4.15 $\pm$ 1	Pu-240	3.46 $\pm$ 6
Cd-113m	3.95 $\pm$ 6	Pu-241	3.16 $\pm$ 2
Sn-119m	2.66 $\pm$ 2	Am-241	3.15 $\pm$ 2
Sn-121m ^b	1.58 $\pm$ 1	Am-242m	2.55 $\pm$ 6
Sn-126	4.70 $\pm$ 2	Am-243	1.72 $\pm$ 6
Sb-123	3.99 $\pm$ 2	Cm-242	2.67 $\pm$ 5
I-129	1.55 $\pm$ 2	Cm-243	2.05 $\pm$ 5
Cs-134	2.17 $\pm$ 5	Cm-244	2.95 $\pm$ 5
Cs-137	5.05 $\pm$ 3	Cm-245	3.96 $\pm$ 5

^a Ci/m³ of the final product

Table 3

Savannah River Defense High-Level Waste

<u>Waste Age (yr)</u>	<u>Watts/m³</u>	<u>Watts/Canister</u>	<u>Waste Age (yr)</u>	<u>Watts/m³</u>	<u>Watts/Canister</u>
1	1.30 + 3	8.13 + 2	1000	2.95 + 0	1.84 + 0
2	1.18 + 3	7.36 + 2	2000	1.80 + 0	1.15 + 0
3	1.09 + 3	6.84 + 2	3000	1.52 + 0	9.49 - 1
4	1.03 + 3	6.47 + 2	4000	1.40 + 0	8.76 - 1
5	9.88 + 2	6.18 + 2	5000	1.37 + 0	8.26 - 1
6	9.50 + 2	5.94 + 2	6000	1.25 + 0	7.85 - 1
7	9.17 + 2	5.74 + 2	7000	1.19 + 0	7.47 - 1
8	8.89 + 2	5.56 + 2	8000	1.14 + 0	7.12 - 1
9	8.63 + 2	5.40 + 2	9000	1.09 + 0	6.80 - 1
10	8.41 + 2	5.26 + 2	10000	1.04 + 0	6.50 - 1
20	6.65 + 2	4.16 + 2	20000	7.06 - 1	4.42 - 1
30	5.35 + 2	3.35 + 2	30000	5.25 - 1	3.28 - 1
40	4.34 + 2	2.72 + 2	40000	4.15 - 1	2.59 - 1
50	3.54 + 2	2.21 + 2	50000	3.42 - 1	2.14 - 1
60	2.90 + 2	1.81 + 2	60000	2.91 - 1	1.82 - 1
70	2.39 + 2	1.49 + 2	70000	2.56 - 1	1.60 - 1
80	1.98 + 2	1.24 + 2	80000	2.30 - 1	1.44 - 1
90	1.65 + 2	1.03 + 2	90000	2.12 - 1	1.31 - 1
100	1.39 + 2	8.67 + 1	100000	1.99 - 1	1.24 - 1
200	3.58 + 1	2.24 + 1			
300	1.62 + 1	1.01 + 1			
400	9.52 + 0	5.96 + 0			
500	6.53 + 0	4.09 + 0			
600	5.00 + 0	3.13 + 0			
700	4.14 + 0	2.59 + 0			
800	3.60 + 0	2.25 + 0			
900	3.22 + 0	2.02 + 0			

Table 5

INEL Defense High-Level Waste

<u>Years from Calcification</u>	<u>Watts/m²</u>	<u>Watts/Canister</u>	<u>Years from Calcification</u>	<u>Watts/m³</u>	<u>Watts/Canister</u>
1	2.73 + 2	1.71 + 2	1000	1.09 - 1	6.85 - 2
2	2.33 + 2	1.46 + 2	2000	6.30 - 2	3.94 - 2
3	2.11 + 2	1.32 + 2	3000	5.17 - 2	3.23 - 2
4	1.97 + 2	1.23 + 2	4000	4.74 - 2	2.97 - 2
5	1.87 + 2	1.17 + 2	5000	4.47 - 2	2.79 - 2
6	1.79 + 2	1.12 + 2	6000	4.24 - 2	2.65 - 2
7	1.73 + 2	1.08 + 2	7000	4.03 - 2	2.52 - 2
8	1.67 + 2	1.05 + 2	8000	3.85 - 2	2.41 - 2
9	1.62 + 2	1.02 + 2	9000	3.67 - 2	2.30 - 2
10	1.58 + 2	9.88 + 1	10000	3.51 - 2	2.20 - 2
20	1.22 + 2	7.69 + 1	20000	2.41 - 2	1.51 - 2
30	9.69 - 1	6.07 + 1	30000	1.83 - 2	1.15 - 2
40	7.66 - 1	4.79 + 1	40000	1.49 - 2	9.31 - 3
50	6.06 + 1	3.79 + 1	50000	1.26 - 2	7.90 - 3
60	4.81 + 1	3.01 + 1	60000	1.11 - 2	6.92 - 3
70	3.81 + 1	2.39 + 1	70000	9.95 - 3	6.23 - 3
80	3.03 + 1	1.89 + 1	80000	9.14 - 3	5.72 - 3
90	2.42 + 1	1.51 + 1	90000	8.55 - 3	5.35 - 3
100	1.93 + 1	1.21 + 1	100000	8.11 - 3	5.08 - 3
200	2.50 + 0	1.57 + 0			
300	6.63 - 1	4.15 - 1			
400	3.44 - 1	2.15 - 1			
500	2.38 - 1	1.49 - 1			
600	1.85 - 1	1.16 - 1			
700	1.55 - 1	9.67 - 2			
800	1.35 - 1	6.42 - 2			
900	1.20 - 1	7.53 - 2			

Table 7

Hanford Defense High-Level Waste

<u>Waste Age (yr)</u>	<u>Watts/m³</u>	<u>Watts/Canister</u>	<u>Waste Age (yr)</u>	<u>Watts/m³</u>	<u>Watts/Canister</u>
1	1.03 + 2	6.42 + 1	1000	2.70 + 0	1.69 + 0
2	9.60 + 1	6.01 + 1	2000	8.35 - 1	5.23 - 1
3	9.15 + 1	5.72 + 1	3000	4.45 - 1	2.79 - 1
4	8.79 + 1	5.50 + 1	4000	3.53 - 1	2.21 - 1
5	8.50 + 1	5.32 + 1	5000	3.23 - 1	2.02 - 1
6	8.24 + 1	5.16 + 1	6000	3.05 - 1	1.91 - 1
7	8.01 + 1	5.01 + 1	7000	2.91 - 1	1.82 - 1
8	7.79 + 1	4.88 + 1	8000	2.79 - 1	1.74 - 1
9	7.59 + 1	4.75 + 1	9000	2.67 - 1	1.67 - 1
10	7.40 + 1	4.63 + 1	10000	2.56 - 1	1.60 - 1
20	5.90 + 1	3.69 + 1	20000	1.77 - 1	1.11 - 1
30	4.82 + 1	3.01 + 1	30000	1.31 - 1	8.17 - 2
40	3.99 + 1	2.50 + 1	40000	1.01 - 1	6.33 - 2
50	3.35 + 1	2.10 + 1	50000	8.12 - 2	5.08 - 2
60	2.85 + 1	1.78 + 1	60000	6.69 - 2	4.19 - 2
70	2.45 + 1	1.53 + 1	70000	5.65 - 2	3.54 - 2
80	2.14 + 1	1.34 + 1	80000	4.90 - 2	3.07 - 2
90	1.89 + 1	1.18 + 1	90000	4.35 - 2	2.72 - 2
100	1.69 + 1	1.06 + 1	100000	3.95 - 2	2.47 - 2
200	9.40 + 0	5.38 + 0			
300	7.59 + 0	4.75 + 0			
400	6.48 + 0	4.05 + 0			
500	5.56 + 0	3.48 + 0			
600	4.79 + 0	3.00 + 0			
700	4.14 + 0	2.59 + 0			
800	3.58 + 0	2.24 + 0			
900	3.10 + 0	1.94 + 0			

Table 8

Savannah River Defense High-Level Waste Curie Content

<u>Waste Age (yr)</u>	<u>Curies/m³</u>	<u>Curies/Canister</u>	<u>Waste Age (yr)</u>	<u>Curies/m³</u>	<u>Curies/Canister</u>
1	3.88 + 5	2.43 + 5	1000	1.00 + 2	5.27 + 1
2	3.57 + 5	2.24 + 5	2000	6.56 + 1	4.10 + 1
3	3.36 + 5	2.10 + 5	3000	5.68 + 1	3.55 + 1
4	3.19 + 5	2.00 + 5	4000	5.31 + 1	3.32 + 1
5	3.06 + 5	1.92 + 5	5000	5.06 + 1	3.17 + 1
6	2.95 + 5	1.84 + 5	6000	4.85 + 1	3.03 + 1
7	2.85 + 5	1.78 + 5	7000	4.65 + 1	2.91 + 1
8	2.76 + 5	1.72 + 5	8000	4.48 + 1	2.80 + 1
9	2.67 + 5	1.67 + 5	9000	4.31 + 1	2.70 + 1
10	2.60 + 5	1.63 + 5	10000	4.16 + 1	2.60 + 1
20	2.01 + 5	1.26 + 5	20000	3.10 + 1	1.94 + 1
30	1.58 + 5	9.91 + 4	30000	2.53 + 1	1.58 + 1
40	1.25 + 5	7.82 + 4	40000	2.19 + 1	1.37 + 1
50	9.88 + 4	6.18 + 4	50000	1.96 + 1	1.23 + 1
60	7.82 + 4	4.89 + 4	60000	1.80 + 1	1.13 + 1
70	6.19 + 4	3.88 + 4	70000	1.69 + 1	1.06 + 1
80	4.91 + 4	3.07 + 4	80000	1.62 + 1	1.01 + 1
90	3.90 + 4	2.44 + 4	90000	1.56 + 1	9.75 + 0
100	3.10 + 4	1.94 + 4	100000	1.52 + 1	9.49 + 0
200	3.63 + 3	2.27 + 3			
300	7.51 + 2	4.70 + 2			
400	3.28 + 2	2.05 + 2			
500	2.14 + 2	1.34 + 2			
600	1.64 + 2	1.02 + 2			
700	1.37 + 2	8.54 + 1			
800	1.20 + 2	7.50 + 1			
900	1.09 + 2	6.79 + 1			

Table 9

INEL Defense High-Level Waste Curie Content

Waste Age (yr)	Curies/m ³	Curies/Canister	Waste Age (yr)	Curies/m ³	Curies/Canister
1	9.09 + 4	5.69 + 4	1000	7.70 + 0	4.82 + 0
2	7.98 + 4	5.00 + 4	2000	6.16 + 0	3.86 + 0
3	7.31 + 4	4.58 + 4	3000	5.81 + 0	3.63 + 0
4	6.85 + 4	4.29 + 4	4000	5.66 + 0	3.54 + 0
5	6.50 + 4	4.07 + 4	5000	5.57 + 0	3.48 + 0
6	6.22 + 4	3.89 + 4	6000	5.48 + 0	3.43 + 0
7	5.99 + 4	3.75 + 4	7000	5.40 + 0	3.38 + 0
8	5.78 + 4	3.62 + 4	8000	5.33 + 0	3.34 + 0
9	5.60 + 4	3.50 + 4	9000	5.27 + 0	3.30 + 0
10	5.43 + 4	3.40 + 4	10000	5.21 + 0	3.26 + 0
20	4.18 + 4	2.62 + 4	20000	4.75 + 0	2.98 + 0
30	3.29 + 4	2.06 + 4	30000	4.47 + 0	2.80 + 0
40	2.60 + 4	1.62 + 4	40000	4.27 + 0	2.67 + 0
50	2.05 + 4	1.28 + 4	50000	4.11 + 0	2.57 + 0
60	1.62 + 4	1.01 + 4	60000	3.97 + 0	2.49 + 0
70	1.28 + 4	7.99 + 3	70000	3.85 + 0	2.41 + 0
80	1.01 + 4	6.31 + 3	80000	3.75 + 0	2.34 + 0
90	7.98 + 3	4.99 + 3	90000	3.65 + 0	2.28 + 0
100	6.31 + 3	3.95 + 3	100000	3.56 + 0	2.23 + 0
200	6.52 + 2	4.08 + 2			
300	9.68 + 1	6.05 + 1			
400	3.04 + 1	1.90 + 1			
500	1.72 + 1	1.08 + 1			
600	1.24 + 1	7.77 + 0			
700	1.01 + 1	6.34 + 0			
800	8.90 + 0	5.57 + 0			
900	8.17 + 0	5.11 + 0			

Table 10
Hanford Defense High-Level Waste Curie Content

<u>Waste Age</u> <u>(yr)</u>	<u>Curies/m³</u>	<u>Curies/Canister</u>	<u>Waste Age</u> <u>(yr)</u>	<u>Curies/m³</u>	<u>Curies/Canister</u>
1	4.10 + 4	2.57 + 4	1000	1.09 + 2	6.84 +
2	3.65 + 4	2.28 + 4	2000	5.29 + 1	3.31 + 1
3	3.31 + 4	2.07 + 4	3000	4.11 + 1	2.57 + 1
4	3.05 + 4	1.91 + 4	4000	3.81 + 1	2.38 + 1
5	2.84 + 4	1.77 + 4	5000	3.70 + 1	2.32 + 1
6	2.66 + 4	1.67 + 4	6000	3.63 + 1	2.27 + 1
7	2.52 + 4	1.58 + 4	7000	3.57 + 1	2.24 + 1
8	2.40 + 4	1.50 + 4	8000	3.52 + 1	2.20 + 1
9	2.29 + 4	1.43 + 4	9000	3.47 + 1	2.17 + 1
10	2.20 + 4	1.38 + 4	10000	3.42 + 1	2.14 + 1
20	1.64 + 4	1.02 + 4	20000	3.07 + 1	1.92 + 1
30	1.28 + 4	8.04 + 3	30000	2.84 + 1	1.78 + 1
40	1.02 + 4	6.36 + 3	40000	2.68 + 1	1.68 + 1
50	8.09 + 3	5.07 + 3	50000	2.56 + 1	1.60 + 1
60	6.47 + 3	4.05 + 3	60000	2.46 + 1	1.54 + 1
70	5.18 + 3	3.24 + 3	70000	2.37 + 1	1.48 + 1
80	4.18 + 3	2.61 + 3	80000	2.29 + 1	1.43 + 1
90	3.38 + 3	2.12 + 3	90000	2.22 + 1	1.39 + 1
100	2.75 + 3	1.22 + 3	100000	2.16 + 1	1.35 + 1
200	5.76 + 2	3.61 + 2			
300	3.11 + 2	1.94 + 2			
400	2.42 + 2	1.51 + 2			
500	2.04 + 2	1.28 + 2			
600	1.76 + 2	1.10 + 2			
700	1.54 + 2	9.65 + 1			
800	1.36 + 2	8.54 + 1			
900	1.22 + 2	7.62 + 1			

Table 11

Spent Fuel Parameters

Fuel Type	Fresh Fuel Enrichment w/o ^{235}U	Number of Irradiation Cycles	% of Core Average Power During Final Cycle	Total Burnup	
				Mwd / Assembly	Mwd/MTHM
PWR	3.30	3	100	15200	33000
BWR	2.66	3	110	4500	74600
BWR	2.66	4	46	4800	26200
BWR	2.66	4	96	5400	29600
BWR	2.83	4	48	5000	27200
BWR	2.83	4	96	5800	30600

Cycle involves 100 days irradiation followed by 20 days at zero power repeated 3 times

*Mwd = megawatts-days

**MTHM = metric ton of heavy metal initially charged to the reactor

Table 12

Fresh Fuel Elemental Composition

Structural Material	Element Content (grams/assembly)		
	PWR	BWR	
Si	99.97	80.27	
Cr	3110	1590	
Mn	198.9	159.9	
Fe	8105	5602	
Ni	3825	719.0	
Zr	125800	53920	
Nb	278.3	--	
Mo	168.2	--	
Sn	2054	879.8	
Fuel	3.3 w/o ^{235}U	2.66 w/o ^{235}U	2.83 w/o ^{235}U
^{234}U	127.2	58.74	62.02
^{235}U	15070	4872	5171
^{236}U	--	525.2	556.4
^{238}U	441300	177500	177200
Total Weight per Assembly	600.2 kg	245.9 kg	245.9 kg

Table 13

Spent Fuel Thermal Output in Watts Per Assembly

Years Out of Reactor	PWR	3-Cycle BWR	2.66 w/a		2.83 w/a	
			4-Cycle BWR Low Burnup	4-Cycle BWR High Burnup	4-Cycle BWR Low Burnup	4-Cycle BWR High Burnup
1	5030	1400	1100	1500	1100	1500
2	2590	760	630	830	650	860
3	1710	480	430	550	440	570
4	1230	350	330	410	340	430
5	970	280	270	330	280	350
6	820	230	240	290	250	300
7	720	210	220	260	220	270
8	660	190	200	240	210	250
9	610	180	190	230	200	230
10	580	170	180	220	190	220
20	430	130	140	170	150	170
30	360	110	120	140	120	140
40	300	94	100	120	100	120
50	250	81	87	100	90	100
60	220	71	76	88	78	91
70	190	62	66	77	69	80
80	170	55	59	69	61	71
90	150	49	53	61	55	63
100	130	44	48	56	49	57
200	70	24	26	30	27	31
300	54	18	20	23	20	23
400	46	15	17	19	17	19
500	40	13	14	16	14	16
600	35	12	13	14	13	14
700	31	11	11	13	11	13
800	28	9.6	10	11	10	11
900	26	8.7	9.2	10	9.3	10
1000	23	8.0	8.5	9.4	8.5	9.4
2000	13	4.7	4.8	5.2	4.8	5.2
3000	10	3.8	3.9	4.1	3.9	4.2
4000	9.3	3.4	3.5	3.7	3.5	3.7
5000	8.6	3.2	3.3	3.5	3.3	3.5
6000	8.1	3.0	3.1	3.2	3.1	3.3
7000	7.6	2.8	2.9	3.1	2.9	3.1
8000	7.1	2.7	2.7	2.9	2.7	2.9
9000	6.7	2.5	2.6	2.7	2.6	2.7
10000	6.3	2.4	2.4	2.5	2.4	2.6
20000	3.8	1.4	1.5	1.5	1.5	1.5
30000	2.5	0.98	0.99	1.0	0.99	1.0
40000	1.8	0.71	0.72	0.74	0.72	0.75
50000	1.4	0.54	0.55	0.57	0.55	0.57
60000	1.1	0.42	0.43	0.45	0.43	0.45
70000	0.85	0.34	0.34	0.36	0.35	0.36
80000	0.69	0.28	0.28	0.30	0.29	0.30
90000	0.57	0.23	0.24	0.25	0.24	0.26
100000	0.49	0.20	0.21	0.22	0.21	0.22

Figure 1. High-Level Defense Waste Thermal Histories

Figure 2. High-Level Defense Waste Total Curie Content

Figure 3. Spent Fuel Thermal Histories

References

- ¹Alternatives for Long-Term Management of Defense High-Level Radioactive Waste, ERDA-77-4211, Vol. 1, Savannah River Plant, Aiken, GA, May 1977.
- ²Alternatives for Long-Term Management of Defense High-Level Radioactive Waste ERDA-77-44, Hanford Reservations, September 1977.
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- ⁶D. D. Wodrich, ltr to S. H. Sutherland, Subject: Radionuclide Constituents of Hanford Defense Waste, Rockwell Hanford Operations, Richland, WA, November 8, 1978.
- ⁷D. E. Bennett, SANDIA-ORICEN Users Manual, SAND79-0299, (Albuquerque: Sandia Laboratories, May 1979).

APPENDIX

Reading a SANDIA-ORIGEN Fiche

In general, a typical SANDIA-ORIGEN output includes a load map, an echo of the user's input data deck, a summary of nuclear data parameters used in the code, and SANDIA-ORIGEN generated data listed by subcase.

The load map occupies the first six pages of each set of fiche at the end of this report. The seventh page indicates the user's data and the eighth page provides a summary of nuclear data parameters. The next page designates the beginning of subcase one. The SANDIA-ORIGEN generated data for that subcase is given in the following pages up to the page designating the start of subcase two data. The process is then repeated until all the data generated for the user specified subcases have been printed and then output is terminated.

The time duration of a subcase is user specified. The code user also specifies the number and termination time of subintervals (one to ten are allowed) in each subcase, whether irradiation or strictly decay occurs during the subinterval, and from which subinterval of the previous subcase (if there is one) the code is to obtain continuation data. As an example, the waste and spent fuel decay intervals and subintervals utilized in this report are given in Table A-1. In each case in this work the continuation data for a subcase was obtained from the data associated with the final subinterval in the immediately preceeding subcase. At the top of each column of data is the time at the end of the subinterval. For instance, if the top of a column reads 5000 Y, it means the data in that column is for waste formed 5000 yr previously or for spent fuel 5000 yr out of the reactor.

For each subcase the data is output in the following format:

1. Structural Materials

a. Nuclide listing in gram-atoms

- b. Element listing in gram-atoms
 - c. Nuclide listing in grams
 - d. Element listing in grams
 - e. Nuclide listing in curies
 - f. Element listing in curies
 - g. Nuclide listing thermal output in watts
 - h. Element listing thermal output in watts
- II. Actinides
- a-h as above
- III. Fission Products
- a-h as above
- IV. Summary Table Showing Totals for Gram-Atoms, Grams, Curies and Watts
- V. Alpha-N Neutron Production Rate
- VI. Spontaneous Fission Neutron Production Rate
- VII. Structural Materials - Gamma Spectrum
- VIII. Actinides - Gamma Spectrum
- IX. Fission Products - Gamma Spectrum
- X. Summary Table - Gamma Spectrum

At the top center of each page of data in a subcase is a title, the subcase number, and the calculation basis (i.e., fuel assembly, cubic feet of glass, etc.). At the top left corner of each page is a description of what data appears on the page. If irradiation occurs during the subcase, relevant information concerning the irradiation appears in the top right corner of the page.

Table A-1

Subcase Time Range (yr)	Time at the End of Each Subinterval in the Subcase (yr)									
0-1	1	2	3	4	5	6	7	8	9	10
10-100	10	20	30	40	50	60	70	80	90	100
100-1000	100	200	300	400	500	600	700	800	900	1000
1000-10000	1000	2000	3000	4000	5000	6000	7000	8000	9000	10000
10000-100000	10000	20000	30000	40000	50000	60000	70000	80000	90000	100000