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Shielding Analysis for the Sealed Storage Cask Concept

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October 1975

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SHIELDING ANALYSIS FOR THE
SEALED STORAGE CASK CONCEPT

By

M. G. Zimmerman

October 1975

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Several people have contributed to this project. Specifically S. W. Heaberlin (BNW) performed many S_N Transport calculations and along with D. H. Thomsen (BNW) converted a more efficient version of the S_N code ANISN for use in this project. D. E. Larson (ARHCO) provided all data on the RSSF storage modules for use in the calculations and provided overall direction for the shielding design effort.

ABSTRACT

A shielding study in support of the sealed storage cask concept for the Retrievable Surface Storage Facility has been completed. This study consists of an analysis of dose rates from solidified waste under various shielding conditions. The radiation sources considered were high level radioactive wastes from PWR-uranium cycle, PWR-Pu recycle, LMFB, and HTGR fuels. The types of shielding considered include massive shields of steel, concrete or water for the cylindrical solidified waste canisters, end shielding of the canisters, streaming through the air cooling inlet and outlet ports of the shields, and large distances through air from a waste storage array.

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SHIELDING ANALYSIS FOR THE SEALED STORAGE CASK CONCEPT

M. G. Zimmerman

SUMMARY

A shielding study in support of the Sealed Storage Cask Concept for the Retrievable Surface Storage Facility (RSSF) has been completed. Calculations were performed to determine radiation source strengths and dose rates from canisters of solidified high-level wastes from spent reactor fuel processing in various configurations and with various types and amounts of shielding. Wastes from PWR-uranium, PWR-plutonium recycle, HTGR and LMFBR reactor types were included. Calculations were generally based on solidified waste aged 10 years after fuel removal from the reactor which released 5 kW radioactive decay energy from a standard canister 10 feet long and 1 foot in diameter. Information to convert the results to other waste "ages" is provided in the report. Results are based on shielding of individual canisters with steel, water, a commercial concrete-like shield material, ordinary concrete and high density concrete. Concrete shields were analyzed for four different weight per cents of water in both ordinary and high-density concretes.

Other problems considered included:

- lead, concrete, water or steel shielding of the ends of the waste canisters
- dose rates above an array of canisters in a water-filled storage basin
- dose rates under an RSSF storage unit while being lifted into place
- streaming dose rates from the air-cooling inlet and outlet ports of an RSSF storage unit
- dose rates from an array of RSSF modules.

The calculations were performed using the ORIGEN, ANISN and Battelle-Monte Carlo computer codes. ORIGEN was used to calculate the source terms; the other codes were used for the actual shielding calculations. A

sophisticated calculational technique was unavailable to calculate dose rates at large distances from an RSSF storage array. Therefore, we used literature sources and hand calculational techniques to estimate dose rates.

INTRODUCTION

A shielding study was performed to support the design and development of the Sealed Storage Cask Concept of the Retrievable Surface Storage Facility. The purpose of the study was to provide shielding data applicable to the solidified high-level reprocessed wastes from fuel discharged from the various types of power reactors.

Such a study was needed because solidified wastes reprocessed from spent fuel produce concentrated radioactive sources which must be shielded to reduce dose rates to safe levels and also allow reasonable movement in the storage array. Knowledge of the expected dose rates is also necessary to guide the design of handling equipment and other facilities.

The scope of the study included the prediction of source terms and shielding of these sources. Source terms were calculated for PWR's for both the uranium cycle and plutonium recycle, HTGR, and LMFBR reactor wastes. The source terms considered included primary gamma, neutrons and secondary gammas. Alpha and beta radiation is assumed to be shielded totally by the waste canister. The shielding calculations were performed mainly for concrete of two compositions and for 4 amounts of water retention, though calculations were also performed for other shielding materials. Other calculations were performed to determine dose rates from radiation streaming through the air inlet and outlet ports. The dose rates were also predicted at a distance from the storage array. Other calculations were performed as needed for the design effort and are also included in the report.

CONCEPT

For the purposes of this study the design of the Sealed Storage Cask Concept consisted of a cylindrical steel canister enclosed by a steel cask

which is contained in concrete shielding. Figure 1 shows a photograph of a model design. For most of the calculations of this study we assumed a solidified waste in a canister 10 feet long and 15 inches in diameter having a wall of 3/8 in. steel. Surrounding this was a 2 in. layer of steel, a 6 in. air gap and finally about 38 in. of concrete shielding. These approximate the dimensions of the prototype design.

This report will refer to one canister and its associated shielding as a RSSF storage unit (Retrievable Surface Storage Facility). RSSF will be used to refer to the entire storage array of RSSF storage units and/or its associated equipment and buildings.

METHODS

The methods discussed in this section include the computer codes used and the associated data used with them. In some cases data from the open literature were used to estimate dose rates. This estimation procedure will be discussed when the computation model for the specific shielding problem is presented later.

The codes used for this study included ORIGEN⁽¹⁾, ANISN⁽²⁾, and BMC-MG⁽³⁾. ORIGEN is a radioisotope buildup and decay code used to calculate radiation sources based on reactor parameters. ANISN is a one-dimensional transport code used to calculate primary gamma, secondary gamma and neutron dose rates from solidified waste cylinders. BMC-MG is a general geometry, three-dimensional, multigroup Monte Carlo code used to calculate dose rates where more than one dimension is needed to adequately predict neutron dose rates or where scattering effects are important.

BASIC DATA

The basic data needed for this study consisted of neutron and gamma cross-sections, neutron and gamma source strengths and spectra, reactor parameters for input to the ORIGEN code, and flux-to-dose conversion factors.

The nuclear data needed for the calculations consisted of the following items:

1. Neutron cross-sections
2. Gamma cross-sections
3. Neutron-gamma production cross-sections
4. Neutron yields and spectra
5. Gamma yields and spectra

The neutron cross-sections are produced from the DLC-2D⁽⁴⁾ set of 100-group P_8 cross-sections based on ENDFB version III data, obtained from the Radiation Shielding Information Center (RSIC) at the Oak Ridge National Laboratory.

The one-dimensional transport code, ANISN, was used to collapse the cross-sections down to 27 groups. The ANISN collapse was performed over a combination fission and (α, n) source spectrum. The P_3 legendre expansion representation was used for the cross-sections. The neutron group structure, shown in Table 1, is a standard structure used at PNL and elsewhere for many shielding problems.

The gamma cross-sections were obtained from the Storm-Israel library⁽⁵⁾ available from RSIC; and the code MUG⁽⁶⁾ was used to produce multigroup P_3 legendre expansion cross-sections from the Storm-Israel data. The gamma energy group structure is shown in Table 1. This group structure was obtained from that given in CTC-42⁽⁷⁾ minus the two top groups. An insignificant number of gammas were produced in these two top groups by the sources considered.

Neutron-gamma production cross-sections are produced by combining neutron gamma yields and cross-sections. Secondary gamma rays can be produced from many neutron interactions including for example (n, γ) , (n, α) , $(n, 2n)$, and inelastic scatter. Only those yields obtainable from the POPOP4⁽⁸⁾ library were used. All cross-section sets included, at a minimum, the secondary gamma production from (n, γ) reactions. The secondary gamma yield and some inelastic gamma production cross-sections were obtained from the POPOP4 library and the (n, γ) cross-sections were obtained from GAM-II^(a).

a. Information from J. L. Carter, Battelle, Pacific Northwest Laboratories.

Essentially, the same procedures as those described in the document CTC-20⁽⁹⁾ available from RSIC were followed in producing the (n,γ) production cross-sections. Many of the same approximations and questionable techniques described as such in the document were used. These included the use of thermal neutron-induced (n,γ) yields for all neutron energies, spectrum weighting of the (n,γ) cross-sections not precisely consistent with the weighting used for the neutron cross-sections, and some inelastic gamma production cross-sections used directly from the POPOP4 library, which may not have been properly spectrum weighted for our purpose.

A small code to combine neutron, gamma and neutron-gamma production cross-sections⁽⁹⁾ was used to obtain coupled neutron-gamma cross-section sets. Figure 2 indicates the steps followed and described above.

The spectra used in the calculation are shown in Table 1. A CF-252 spectrum was used for the neutron calculations. This spectrum was used to represent the spontaneous fission and (α,n) source from Cm-252 and Cm-254 in the waste. Since it is harder, the CF-252 spectrum is conservative for the PWR and LMFBW wastes. If the HTGR waste contains enough carbon to give rise to a significant (α,n) source from carbon, the CF-252 spectrum may not be conservative for the HTGR cases. This is true because the neutrons from the (α,n) reaction with carbon have a much higher average energy than that for a fission spectrum or an (α,n) spectrum from oxygen (α,n) reactions.

The gamma spectra were based on 10-year-old PWR uranium cycle waste. This type of waste has the highest gamma source per kilowatt. We assumed that the gamma spectra would be similar to that for other reactor types and over the time range (10-100 years after discharge) of interest.

The data needed for the ORIGEN code to calculate yields and spectra were built in as part of the code^(a) except for certain reactor parameters. Table 2 shows the reactor types and parameters considered in this study. The neutron and gamma source strength and the gamma spectra were obtained from the ORIGEN output. The sources and spectra are based on waste aged 10

a. Certain plutonium cross-sectional data was changed by E. T. Merrill (BNW) based on data available at PNL.

years after chemical separation which was assumed to take place immediately upon removal from the reactor core. During chemical separation it is assumed that 99.5% of the uranium and plutonium is removed from the waste for the PWR and LMFBR cases and that 99.5% of the thorium and uranium is removed from the HTGR wastes. Table 3 shows the neutron and primary gamma source strengths. The decay heat sources for the different fuels are also shown because all results in the report are based on the amount of solidified waste producing 5 kW of energy.

Table 1 also shows the flux-to-dose conversion factors used. The gamma conversion factors came from Rockwell.⁽¹⁰⁾ The neutron flux-to-dose factors came from the Code of Federal Regulations.⁽¹¹⁾

CORRELATION

An ANISN calculation was performed for comparison with data available in the literature.⁽¹³⁾ in order to provide a cross-section check. The comparison consists of a point fission source in infinite concrete. Table 4 shows the concrete composition used. A standard U-235 fission spectrum was used for the source, and Henderson⁽¹²⁾ conversion factors were used to convert flux to dose. Figure 3 shows the comparison of our ANISN calculations with results from ORNL-1167.⁽¹³⁾ The ORNL-1167 neutron results were calculated by the moments method. Our neutron results agree very well but the (n, γ) results do not. Even assuming our (n, γ) results might be as low as the ORNL-1167 results indicate, the (n, γ) dose rate only approaches that due to neutrons or primary gammas for shield thicknesses of primary interest in this study.

COMPUTATION MODELS

The computational model elements consist of source and shield compositions, geometrical descriptions, and other data used in the computer and hand calculation models.

The shield materials used in the computer calculations consisted of the solidified waste, mild steel, water, a commercial heat resistant concrete-

like shielding material, ordinary concrete and a high density concrete. Table 5 shows the waste compositions used in the calculations. Three types of solidified waste are shown: calcine, fluid bed calcine, and borosilicate glass. Since cross-sections for the radioactive waste products were not available and would be difficult to approximate because of the great diversity of nuclides involved, their contribution to the shielding was ignored. The compositions of the shield and structural materials, except for the concrete, appear in Table 6; Table 7 shows the concrete compositions used for both the ordinary and high density concrete shields. Because the waste will give off significant amounts of heat which may drive water off from the concrete, calculations were performed with varying amounts of water. Table 8 shows the water retentions assumed.

The computational geometries were simplified representations of the actual retrievable surface storage system storage unit. A conceptual design of one of the storage units is shown in Figure 1 and the geometrical model of the storage unit as used in the ANISN calculations is shown in Figure 4. Since ANISN is a one-dimensional code, the cylinder must be assumed to be infinite. However, the error caused by this assumption will be small and conservative since we are calculating the dose rate at the surface of the shield, a distance from the waste canister less than half the height of the canister. The error is estimated to be on the order of or less than 10%.

ANISN was also used to predict dose rates above a water-filled storage basin containing vertical waste canisters. The infinite slab ANISN geometry used to calculate this is shown in Figure 5. The waste canisters and water surrounding them were smeared together to model the source slab, and the water above the canisters is the upper slab.

Various types of geometries were used in the BMC-MG calculations depending on the problem to be solved. These consisted of multi-dimensional systems where it was necessary to predict dose rates due in large part to scattering or streaming.

One of the shielding problems of a proposed design was analyzed using both the BMC-MG code and ANISN. The problem consisted of estimating the dose rate above the canister surrounded by the shielding of the storage unit without a plug in the top. This was necessary to determine the amount of shielding necessary to protect a welder while installing the plug. Figure 6 shows the geometry of this problem. In order to reduce the cost of analysis BMC-MG calculations were used to predict the dose rate above the canister with no shielding plug in place and then ANISN slab geometry cases were used to predict the effectiveness of several different shields for the plug. A buckling factor was used in ANISN to approximate leakage to the side.

The streaming problems were also analyzed with the BMC-MG code. These problems consisted of predicting the dose rates (a) under a RSSF storage unit while it is being lifted without a base during the receiving and placement of the unit; and (b) from radiation streaming through the air cooling inlet and outlet ports of the unit. Figure 7 shows the geometry used in the lift calculation. The floor was assumed to be concrete and only the bottom foot of the vertical length of the source was used for the calculation in order to reduce computer time to a reasonable level. Other calculations to evaluate the effect of the rest of the source were performed using the next foot vertically upward. From this the effect of the rest of the canister on the dose rates under the module while being lifted was conservatively estimated. Only a 6 in. thickness of shield material around the streaming volume was used to account for scattering effects. It was assumed that if a particle penetrated 6 in. away from the scattering surface, it would not return. Calculations were done with different thicknesses to show that 6 in. is adequate.

Approximately the same procedure was used to predict dose rates from radiation streaming through the air-cooling inlet and outlet ports. The analysis included six different designs. The first design is the same as the lift cases discussed above and shown in Figure 7. The distance from base to the RSSF storage unit was assumed to be 6 in. for this case. Figures 8-12 show the other designs which were conceived in order to reduce

the streaming dose rate. The design shown in Figure 6 was tentatively selected for the RSSF because of its ease of fabrication and its reduction of the streaming dose rate.

The problems in obtaining good statistical results with a reasonable amount of computer time with BMC-MG for the streaming cases were more difficult than for the lift cases. In the streaming cases it was necessary to consider the problem in several parts. As an example consider Figure 12. Detector points were placed in the streaming area at various points radially outward from the waste canister. Adequate results could be obtained at a point in the streaming area about 20 in. from the centerline of the module. Another calculation was performed to determine the streaming down the straight duct and then another to determine the streaming around a right angle bend to the port opening. By this method we felt a reasonable estimate of the dose rate could be obtained. The scattering area was considered to include only the first 6 in. of concrete next to the streaming volume. Any radiation penetrating beyond 6 in. was assumed not to return.

Several limitations need to be noted concerning the streaming calculations. Because of the need to stay within a reasonable computer budget, the approximations pointed out above were made. Also, secondary gamma production by the streaming neutrons was not considered. The secondary gamma dose rate due to streaming is thought to be small compared to dose rates from both the neutron and primary gamma streaming but no data has been obtained to prove this. In some cases neutron streaming cases were not run because it became obvious that the dose rates from primary gamma streaming were greater. Some of the streaming dose rate was contributed by direct transmission through the concrete, but this was small compared to the streaming component. For the above reasons, a possible error of a factor of two should be assigned to these results. For design purposes this is probably adequate. Also, the error should be in a conservative direction.

Another problem analyzed was the prediction of dose rates as a function of distance from an array of 8-in. thick steel shielded waste canisters on 10 foot centers from the short side of an array 2500 feet by

5000 feet in size. The direct radiation attenuation factors were obtained from an ANISN slab case for 40 meters of air. A simple point kernel computer program was then written to calculate the direct dose rate from the first two rows assuming no shielding of the second row by the first row. The rest of the rows were assumed to contribute only by skyshine. To be conservative the ground surface was assumed to be a mirror reflector of the radiation from the first two rows.

The skyshine was estimated as a function of distance from information available in references 14 and 15. For the gamma skyshine, a skyshine correction factor based on the direct scattered radiation component in infinite media for ^{137}Cs and ^{60}Co found in the Reactor Handbook⁽¹⁴⁾ was used. All of the direct radiation was assumed to have been scattered at the large distances being considered. The Reactor Handbook⁽¹⁴⁾ indicates that the gamma skyshine correction factor is a slowly varying function of energy; therefore, these correction factors can be used to estimate the skyshine for the spectra considered in this study. The skyshine for neutrons was estimated from data in Jaeger, et. al.⁽¹⁵⁾ pertaining to neutron flux as a function of energy, distance and angle between the source direction and a line connecting the source point and detector. Inherent in this analysis are a large number of approximations. The results therefore can be used only as rough estimates. This is especially true for the skyshine estimate at large distances.

RESULTS

The results are based on radiation source strengths from the code ORIGEN. Table 3 shows the radiation source strengths and energy release from irradiated fuel waste 10 years after shutdown and reprocessing. These source strengths are based on a metric ton of heavy metal charged to the reactor and form the basis of all of the following results. All results, unless otherwise noted, are based on a total amount of calcine waste producing 5 kW radioactive decay energy.

The results can be adjusted by using simple ratios to approximate other waste ages and fuel types. The source strength variations as a function of

time to 100 years are shown in Figures 13 and 14. The results can be approximately adjusted for any time after shutdown by using these graphs. The results presented are based on PWR-U cycle waste for gamma dose rates and PWR-Pu recycle waste for neutron dose rates. Ratios for converting the results to dose rates from different reactor wastes are given in Table 9. These ratios are calculated from Table 3 based on a source strength per kW radioactive decay energy. The results can also be easily adjusted for waste producing other amounts of energy by using simple ratios.

The first set of results consists of dose rates from solidified waste canisters shielded by various materials. Surface dose rates for bare solidified waste canisters are presented in Tables 10 and 11 and make clear the necessity for shielding. Surface dose rates for various shield types based on the ANISN calculational geometry (shown in Figure 4) are presented in Figures 15 through 30.

Figures 15-24, 27 and 28 show neutron, primary gamma, secondary gamma and total surface dose rates for shields of water, mild steel, the commercial shielding material, ordinary cured concrete and high density cured concrete from both PWR-U cycle and PWR-Pu recycled waste. Figures 25-26 and 29-30 show total surface dose rates for shields of ordinary concrete and high density concrete for four different weight percents of water for both Pu-U cycle and PWR-Pu recycle wastes. Also shown in these figures, are total surface dose rates for different thicknesses of mild steel inside the concrete shields. Tables 12-25 present the results some of the foregoing figures are based on. Included are dose rates due to the separate types of radiation (neutron, primary gamma, and secondary gamma) which are not all shown separately on the figures.

The dose rates at the top of the water storage basin are presented next. These results are based on waste cylinders on a 1-ft. pitch, standing vertically under water. (The calculation made using a smeared slab source was described earlier in the computation model section.) Within reason the results can be adjusted to other pitch spacings by assuming different source densities. These results are shown in Figure 31. The gamma results are for PWR-uranium cycle waste and the neutron results are for PWR-plutonium recycled waste.

The results of shielding the cylinder ends for welding are presented next. It was assumed that the ends were shielded by lead or steel followed by a material similar in shielding properties to water such as polyethylene. Figures 32 and 34 show the dose rates from the cylinder ends as a function of lead or steel thickness. Figures 33 and 35 present the shielding effect of water after a nominal thickness of lead or steel, respectively. The ordinate of these graphs is a ratio of the dose rate for each radiation type at the surface of a given water thickness over the dose rate at the surface of a nominal steel or lead thickness. This ratio corresponding to a given water thickness can be multiplied by the results in Figures 32 and 34 for a given lead or steel thickness to obtain approximate dose rates from the canister end shielded by lead and water or by steel and water.

The next results show the dose rates under a RSSF storage unit while it is being lifted without a base. The results consist of dose rates radially out from the centerline of the module and halfway between the ground and base of the unit for various heights of the module above the ground. Tables 26 and 27 show these results.

Table 28 presents the dose rates calculated at the streaming exits from the air cooling inlet and outlet ports. The results are for the various designs discussed previously. Neutron calculations for some of the designs were not performed when it became clear that the gamma streaming was more important. Gamma results are based on PWR-uranium cycle waste and neutron results on PWR-plutonium recycle wastes.

Figures 36 and 37 show the dose rates as a function of distance from a large array of RSSF storage units shielded by 8 in. of steel. Because of methods used to obtain these results, the dose rates obtained must be considered very approximate. As before, gamma results are from PWR-uranium cycle wastes and neutron results are for PWR-plutonium recycle wastes.

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TABLE 1. Flux-to-Dose Conversion Factors and Spectra

Group No.	Upper Energy MeV	mrem/hr per Neutrons or gammas/cm ² sec.	Neutrons or ^(a) Gammas/sec
1	14.62	1.97×10^{-1}	4.65×10^{-4}
2	12.21	1.63×10^{-1}	1.88×10^{-3}
3	10.0	1.47×10^{-1}	5.76×10^{-3}
4	8.187	1.47×10^{-1}	1.57×10^{-2}
5	6.703	1.48×10^{-1}	2.87×10^{-2}
6	5.488	1.48×10^{-1}	4.23×10^{-2}
7	4.993	1.40×10^{-1}	5.78×10^{-2}
8	3.679	1.33×10^{-1}	1.08×10^{-1}
9	3.012	1.28×10^{-1}	9.50×10^{-2}
10	2.466	1.27×10^{-1}	9.40×10^{-2}
11	2.019	1.28×10^{-1}	9.25×10^{-2}
12	1.653	1.30×10^{-1}	8.71×10^{-2}
13	1.353	1.31×10^{-1}	7.11×10^{-2}
14	1.108	1.31×10^{-1}	6.89×10^{-2}
15	9.072×10^{-1}	1.16×10^{-1}	1.03×10^{-1}
16	6.081×10^{-1}	9.00×10^{-2}	6.30×10^{-2}
17	4.076×10^{-1}	4.55×10^{-2}	8.94×10^{-2}
18	1.119×10^{-1}	1.19×10^{-2}	1.21×10^{-2}
19	1.503×10^{-2}	3.76×10^{-3}	1.46×10^{-3}
20	3.355×10^{-3}	3.70×10^{-3}	0.0
21	5.829×10^{-4}	4.16×10^{-3}	0.0
22	1.103×10^{-4}	4.46×10^{-3}	0.0
23	2.902×10^{-5}	4.46×10^{-3}	0.0
24	1.068×10^{-5}	4.46×10^{-3}	0.0
25	3.059×10^{-6}	4.46×10^{-3}	0.0
26	1.125×10^{-6}	4.30×10^{-3}	0.0
27	4.139×10^{-7}	3.68×10^{-3}	0.0
28	7.0	7.94×10^{-3}	0.0
29	6.0	7.04×10^{-3}	0.0
30	5.0	5.99×10^{-3}	0.0
31	4.0	5.24×10^{-3}	0.0
32	3.5	4.69×10^{-3}	2.88×10^2
33	3.0	4.10×10^{-3}	9.10×10^3
34	2.5	3.51×10^{-3}	1.12×10^5
35	2.0	2.98×10^{-3}	1.10×10^6
36	1.6	2.44×10^{-3}	2.78×10^6
37	1.2	2.00×10^{-3}	9.20×10^6
38	9.0×10^{-1}	1.54×10^{-3}	2.88×10^8
39	6.0×10^{-1}	1.05×10^{-3}	0.0
40	4.0×10^{-1}	6.17×10^{-4}	3.74×10^7
41	2.1×10^{-1}	2.92×10^{-4}	0.0
42	1.2×10^{-1}	1.48×10^{-4}	0.0
43	7.0×10^{-2}	1.67×10^{-4}	0.0

(a) Un-normalized, total neutrons are not in proper ratio to total gammas.

TABLE 2. Reactor Parameters Used To Calculate Waste Characteristics

	<u>Burnup (MWD/MT)</u>	<u>Power (MW)</u>	<u>Removed During Reprocessing</u>	<u>Initial Enrichment</u>
PWR-Uranium	33,000	30	99.5 wt% U and Pu	3.3% U-235
PWR-Pu Recycle	33,000	30	99.5 wt% U and Pu	3.93 wt% fissile
LMFBR				
Core	105,000	125.7	99.5 wt% U and Pu	12.6 wt% fissile
Axial Blanket	3,100	9.7	99.5 wt% U and Pu	Depleted U
16 Radial Blanket	7,400	5.6	99.5 wt% U and Pu	Depleted U
HTGR	94,300	64.6	99.5 wt% U and Th	7 wt% fissile

All results are based on equilibrium cycles.

TABLE 3. Source Terms 10 Years After Separation

Reactor Type	Energy Release (kw per tonne)*	Gamma-MeV/Sec		Neutrons/Sec	
		Per KW	Per Tonne*	Per KW	Per Tonne*
PWR-Uranium Cycle	1.09	2.43×10^{15}	2.65×10^{15}	2.28×10^8	2.49×10^8
PWR-Pu Recycle	4.21	6.57×10^{14}	2.77×10^{15}	3.16×10^{10}	1.33×10^{10}
LMFBR	1.12	2.53×10^{15}	2.82×10^{15}	1.81×10^8	2.03×10^8
HTGR	3.81	1.89×10^{15}	7.19×10^{15}	4.99×10^8	1.90×10^8

*Per tonne heavy metal charge

TABLE 4. Normal Concrete Composition From ORNL-TM-1167⁽¹⁴⁾

Element	Density	
	<u>g/cm³</u>	<u>atoms/bn-cm</u>
H	0.0200	1.205×10^{-2}
O	1.116	4.201×10^{-2}
Si	0.491	1.056×10^{-2}
Ca	0.612	9.214×10^{-3}
C	0.118	5.922×10^{-3}
Total	<u>2.357</u>	

TABLE 5. Solidified Waste Compositions Assumed For Calculations

<u>Element</u>	<u>Weight Percent</u>		<u>Borosilicate Glass</u>
	<u>Calcine</u>	<u>Fluid Bed Calcine</u>	
O	100	47	55
Al		53	
B			22
Na			12
Fe			11
Density G/cc	1.3	1.3	3.0

TABLE 6. Shield Compositions

<u>Element</u>	<u>Mild Steel</u>	<u>Density-g/cc</u>	
		<u>Water</u>	<u>Commercial Material (a)</u>
H		0.111	0.050
B			0.023 ^(b)
Na			0.007
O		0.889	0.87
S			0.0029 ^(c)
Mg			0.0075 ^(c)
Fe	7.86		0.004
Al			0.38
Si			0.02
Ca			0.14
Total	7.86	1.0	1.50

- a. Special concrete-like refractory shielding material sold commercially.
- b. Twenty percent B-10 was used for the elemental boron in the calculations.
- c. These were left out of calculations because cross-sections were not readily available.

TABLE 7. Concrete Compositions

<u>Element.</u>	<u>Ordinary Poured Concrete Gm/cc</u>	<u>High Density Poured Concrete Gm/cc</u>
H	0.023	0.0217
O	1.22	1.255
C	0.0023	-----
Al	0.078	0.0838
Si	0.780	0.1284
Ca	0.1299	0.2506
Cr	-----	0.2091
Fe	0.032	1.6817
Na	0.0368	-----
 TOTAL	 2.3	 3.62

TABLE 8. Concrete Water Content

<u>Concrete Type</u>	<u>wt% Water</u>
• Ordinary Concrete	
Poured	9.0
Cured	6.0
Heated	2.0
Dry	0.0
• High Density Concrete	
Poured	5.4
Cured	3.6
Heated	2.4
Dry	0.0

TABLE 9. Conversion Factors for Reactor Types Based on 10-Year-Old Waste Producing 5 kW Radioactive Decay Energy and Using the PWR-Pu Case as the Base Case for Neutrons and the PWR-U Case as the Base Case for Gammas.

<u>Cases</u>	<u>Gamma Dose Rate</u>	<u>Neutron Dose Rate</u>
PWR-U	1.0	0.072
PWR-Pu	0.27	1.0
GEFO ^(a)	1.08	0.057
AIFO ^(a)	1.05	0.041
HTGR	0.79	0.016

(a) LMFBR

TABLE 10. Dose Rates From Bare Solidified Waste Canisters
Containing Waste Producing 5 KW Thermal

Canister Type and Diameter	Surface Dose Rate mrem/hr		Dose Rate 10 Feet From C mrem/hr	
	Neutron (a)	Gamma (b)	Neutron	Gamma
1' Calcine	8.04×10^4	3.81×10^8	3.65×10^3	1.74×10^7
2' Calcine	3.36×10^4	1.11×10^8	2.89×10^3	9.63×10^6
1' Glass	6.05×10^4	2.44×10^8	2.75×10^3	1.11×10^7
2' Glass	2.02×10^4	6.58×10^7	1.72×10^3	5.72×10^6
1' Fluid Bed	8.29×10^4	4.53×10^8	3.79×10^3	2.08×10^7
2' Fluid Bed	3.62×10^4	1.36×10^8	3.11×10^3	1.19×10^7

(a) Neutron source based on PWR-Pu recycle.

(b) Gamma source based on PWR-U

TABLE 11. Dose Rates From Bare Solidified Waste Canisters
(vol weighted source)

Canister Type and Diameter	Surface Dose Rate mrem/hr		Dose Rate 10 Feet From Q mrem/hr	
	Neutron	Gamma	Neutron	Gamma
1' Calcine (a)	8.04×10^4	3.81×10^8	3.65×10^3	1.34×10^7
2' Calcine (b)	1.34×10^5	4.44×10^8	1.16×10^4	3.85×10^7
1' Glass	6.05×10^4	2.44×10^8	2.75×10^3	1.11×10^7
2' Glass	8.10×10^4	2.63×10^8	6.90×10^3	2.29×10^7
1' Fluid Bed	8.29×10^4	4.51×10^8	3.79×10^3	2.08×10^7
2' Fluid Bed	1.45×10^5	5.44×10^8	1.25×10^4	4.58×10^7

(a) Source basis for 1' diameter cases same as in Table 2.

(b) Source for 2' diameter cases volume weighted to maintain same source density.

TABLE 12. Dose Rates Through Composite Shields Due to High-Level Waste from Spent PWR-U Fuel: 8" Iron Primary Shield and Three Secondary Shields

Secondary Shield Material	Shield Thickness (Inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Water	12	1.97×10^{-1}	2.15	4.90×10^2	4.92×10^2
	30	3.71×10^{-4}	8.01×10^{-2}	8.15	8.23
	48	1.99×10^{-6}	4.77×10^{-3}	1.46×10^{-1}	1.51×10^{-1}
Concrete (poured)	12	5.07	2.04	4.44×10^1	5.15×10^1
	30	9.37×10^{-3}	1.34×10^{-2}	2.00×10^{-2}	4.28×10^{-2}
	48	4.37×10^{-5}	7.01×10^{-5}	1.10×10^{-5}	1.25×10^{-4}
Commercial Shielding Material	12	1.29	3.23×10^{-1}	2.26×10^2	2.28×10^2
	30	4.43×10^{-3}	2.62×10^{-3}	1.15	1.16
	48	4.13×10^{-5}	7.36×10^{-5}	7.19×10^{-3}	7.30×10^{-3}
No Secondary Shield 8" Iron Primary Shield Only		9.87×10^2	1.62	1.32×10^4	1.42×10^4

TABLE 13. Dose Rates Through Composite Shields Due to High-Level Waste from Spent PWR-Pu Fuel: 8" Iron Primary Shield and Three Secondary Shields

Secondary Shield Material	Shield Thickness (Inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Water	12	2.76	30.1	1.31×10^2	1.64×10^2
	30	5.19×10^{-3}	1.12	2.18	3.31
	48	2.78×10^{-5}	6.67×10^{-2}	3.92×10^{-2}	1.06×10^{-1}
Concrete (poured)	12	70.9	28.5	11.9	1.11×10^2
	30	1.31×10^{-1}	1.87×10^{-1}	5.36×10^{-3}	3.23×10^{-1}
	48	6.11×10^{-4}	9.81×10^{-4}	2.94×10^{-6}	1.59×10^{-3}
Commercial Shielding Material	12	18.0	4.52	60.5	83.0
	30	6.20×10^{-2}	3.66×10^{-2}	3.08×10^{-1}	4.07×10^{-1}
	48	5.77×10^{-4}	1.03×10^{-3}	1.90×10^{-3}	3.51×10^{-3}
No Secondary Shield 8" Iron Primary Shield Only		1.38×10^4	22.6	3.53×10^3	1.74×10^5

TABLE 14. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-U Fuel: 12" Iron Primary Shield and Three Secondary Shields

Secondary Shield Material	Shield Thickness (Inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Water	12	2.67×10^{-2}	1.33	4.37	5.73
	30	5.39×10^{-5}	5.25×10^{-2}	9.61×10^{-2}	1.49×10^{-1}
	48	3.12×10^{-7}	3.11×10^{-3}	2.29×10^{-3}	5.40×10^{-3}
Concrete (poured)	12	2.08	1.27	4.92×10^{-1}	3.84
	30	2.00×10^{-3}	7.94×10^{-3}	3.71×10^{-4}	1.03×10^{-2}
	48	7.36×10^{-6}	4.13×10^{-5}	2.55×10^{-7}	4.89×10^{-5}
Commercial Shielding Material	12	2.24×10^{-1}	1.93×10^{-1}	2.22	2.64
	30	5.91×10^{-4}	1.66×10^{-3}	1.70×10^{-2}	1.93×10^{-2}
	48	5.83×10^{-6}	5.00×10^{-5}	1.43×10^{-4}	1.99×10^{-4}
No Secondary Shield 12" Iron Primary Shield Only		5.33×10^2	0.808	1.51×10^2	6.85×10^2

TABLE 15. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-Pu Fuel: 12" Iron Primary Shield and Three Secondary Shields

Secondary Shield Material	Shield Thickness (Inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Water	12	3.73×10^{-1}	1.85×10^1	1.17	2.00×10^1
	30	7.54×10^{-4}	7.34×10^{-1}	2.58×10^{-2}	7.61×10^{-1}
	48	4.37×10^{-6}	4.35×10^{-2}	6.15×10^{-4}	4.41×10^{-2}
Concrete (poured)	12	2.91×10^1	1.78×10^1	1.32×10^{-1}	4.70×10^1
	30	2.80×10^{-2}	1.11×10^{-1}	9.94×10^{-5}	1.39×10^{-1}
	48	1.03×10^{-4}	5.78×10^{-4}	6.83×10^{-8}	6.81×10^{-4}
Commercial Shielding Material	12	3.13	2.70	5.95×10^{-1}	6.43
	30	8.27×10^{-3}	2.32×10^{-2}	4.55×10^{-3}	3.60×10^{-2}
	48	8.15×10^{-5}	6.99×10^{-4}	3.83×10^{-5}	8.19×10^{-4}
No Secondary Shield 12" Iron Primary Shield Only		7.45×10^3	11.3	40.5	7.50×10^3

TABLE 16. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-U Fuel: 16" Iron Primary Shield and Three Secondary Shields

Secondary Shield Material	Shield Thickness (Inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Water	6	3.93×10^{-1}	3.00	3.12×10^{-1}	3.71
	21	1.15×10^{-4}	1.52×10^{-1}	1.14×10^{-2}	1.64×10^{-1}
	36	1.34×10^{-6}	1.24×10^{-2}	4.89×10^{-4}	1.29×10^{-2}
Concrete (poured)	6	1.26×10^1	1.71	9.43×10^{-2}	1.44×10^1
	21	2.85×10^{-2}	7.29×10^{-2}	2.15×10^{-4}	1.02×10^{-1}
	36	7.29×10^{-5}	8.08×10^{-4}	5.83×10^{-7}	8.81×10^{-4}
Commercial Shielding Material	6	2.40	7.58×10^{-1}	2.11×10^{-1}	3.37
	21	1.08×10^{-3}	9.30×10^{-3}	3.12×10^{-3}	1.35×10^{-2}
	36	1.54×10^{-5}	3.16×10^{-4}	5.82×10^{-5}	3.90×10^{-4}
No Secondary Shield 16" Iron Primary Shield Only		3.08×10^2	0.478	2.73	3.11×10^2

TABLE 17. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-Pu Fuel: 16" Iron Primary Shield and Three Secondary Shields

Secondary Shield Material	Shield Thickness (Inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Water	6	5.49	4.20×10^1	8.37×10^{-2}	4.76×10^1
	21	1.61×10^{-3}	2.13	3.06×10^{-3}	2.13
	36	1.87×10^{-5}	1.73×10^{-1}	1.31×10^{-4}	1.73×10^{-1}
Concrete (poured)	6	1.76×10^2	2.39×10^1	2.53×10^{-2}	2.00×10^2
	21	3.99×10^{-1}	1.02	5.76×10^{-5}	1.42
	36	1.02×10^{-3}	1.13×10^{-2}	1.56×10^{-7}	1.23×10^{-2}
Commercial Shielding Material	6	3.36×10^1	1.06×10^1	5.66×10^{-2}	4.43×10^1
	21	1.51×10^{-2}	1.30×10^{-1}	8.37×10^{-4}	1.46×10^{-1}
	36	2.16×10^{-4}	4.42×10^{-3}	1.56×10^{-5}	4.65×10^{-3}
No Secondary Shield 16" Iron Primary Shield Only		4.31×10^3	6.68	0.723	4.32×10^3

TABLE 18. Dose Rates Through Concrete Shields Due to High-Level Waste From Spent PWR-U Fuel

<u>Shield Material</u>	<u>Shield Thickness (In.)</u>	<u>Neutron Dose Rate (mrem/hr)</u>	<u>Sec. Gamma Dose Rate (mrem/hr)</u>	<u>Pri. Gamma Dose Rate (mrem/hr)</u>	<u>Total Dose Rate (mrem/hr)</u>
Ordinary Concrete (poured)	12	1.07×10^2	6.01	8.05×10^5	8.05×10^5
	24	3.29	0.443	2.50×10^3	2.50×10^3
	36	8.87×10^{-2}	1.86×10^{-2}	7.45	7.56
	48	2.43×10^{-3}	6.79×10^{-4}	2.75×10^{-2}	3.06×10^{-2}
Ordinary Concrete (cured)	12	1.59×10^2	5.01	8.29×10^5	8.29×10^5
	24	6.16	0.572	2.72×10^3	2.73×10^3
	36	0.246	3.65×10^{-2}	8.87	9.15
	48	8.58×10^{-3}	1.73×10^{-3}	2.91×10^{-2}	3.94×10^{-2}
Ordinary Concrete (heated)	12	3.20×10^2	3.03	8.53×10^5	8.53×10^5
	24	22.78	0.884	2.91×10^3	2.93×10^3
	36	1.71	0.132	12.30	14.14
	48	0.123	1.40×10^{-2}	6.82×10^{-2}	0.205
Ordinary Concrete (dried)	12	6.72×10^2	1.31	9.69×10^5	9.70×10^5
	24	1.09×10^2	1.16	3.73×10^3	3.84×10^3
	36	26.15	0.916	17.96	45.03
	48	5.67	0.196	0.110	5.98
High Density Concrete (poured)	12	54.34	1.88	5.41×10^4	5.42×10^4
	24	0.307	0.114	15.65	16.07
	36	7.15×10^{-3}	7.22×10^{-4}	1.41×10^{-2}	2.20×10^{-2}
	48	8.65×10^{-5}	9.30×10^{-6}	2.10×10^{-5}	1.17×10^{-4}
High Density Concrete (cured)	12	94.38	2.45	6.41×10^4	6.42×10^4
	24	1.72	8.37×10^{-2}	22.28	24.08
	36	3.40×10^{-2}	2.15×10^{-3}	2.28×10^{-2}	5.90×10^{-2}
	48	6.52×10^{-4}	4.83×10^{-5}	3.84×10^{-5}	7.39×10^{-4}
High Density Concrete (heated)	12	1.39×10^2	2.67	7.27×10^4	7.28×10^4
	24	3.51	0.147	27.95	31.61
	36	9.22×10^{-2}	5.29×10^{-3}	3.09×10^{-2}	0.128
	48	2.29×10^{-3}	1.57×10^{-4}	5.81×10^{-5}	2.51×10^{-3}
High Density Concrete (dried)	12	4.00×10^2	1.56	9.19×10^4	9.23×10^4
	24	34.89	0.513	44.34	79.74
	36	3.37	0.117	5.81×10^{-2}	3.55
	48	0.324	1.92×10^{-2}	1.30×10^{-4}	0.343

TABLE 19. Dose Rates Through Concrete Shields Due to High-Level Waste From Spent PWR-Pu Fuel

Shield Material	Shield Thickness (In.)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Ordinary Concrete (poured)	12	1.50×10^3	84.1	2.16×10^5	2.18×10^5
	24	4.60×10^1	6.20	6.7×10^2	7.22×10^2
	36	1.24	0.26	2.00	3.50
	48	0.034	0.0095	7.4×10^{-3}	1.69×10^{-2}
Ordinary Concrete (cured)	12	2.22×10^3	70.0	2.23×10^5	2.25×10^5
	24	86.1	8.0	7.31×10^2	8.25×10^2
	36	3.44	0.51	2.38	6.33
	48	0.12	0.0242	7.80×10^{-3}	0.152
Ordinary Concrete (heated)	12	4.48×10^3	42.39	2.29×10^5	2.33×10^5
	24	318.6	12.36	7.79×10^2	1.11×10^3
	36	23.93	1.84	3.30	29.07
	48	1.72	.196	0.0183	1.93
Ordinary Concrete (dried)	12	9.40×10^3	18.30	2.59×10^5	2.68×10^5
	24	1.52×10^3	16.17	1.00×10^3	2.54×10^3
	36	365.76	12.81	4.82	3.83×10^2
	48	79.25	2.74	2.96×10^{-2}	82.02
High Density Concrete (poured)	12	7.60×10^2	26.30	1.45×10^4	1.52×10^4
	24	4.30	1.6	4.20	10.10
	36	0.10	1.01×10^{-2}	3.78×10^{-3}	0.114
	48	1.21×10^{-3}	1.3×10^{-4}	5.66×10^{-6}	1.35×10^{-3}
High Density Concrete (cured)	12	1.32×10^3	34.24	1.72×10^4	1.86×10^4
	24	24.03	1.17	5.98	31.18
	36	0.475	3.00×10^{-2}	6.10×10^{-3}	0.511
	48	9.12×10^{-3}	6.76×10^{-4}	1.03×10^{-5}	9.81×10^{-3}
High Density Concrete (heated)	12	1.95×10^3	37.33	1.95×10^4	2.15×10^4
	24	49.09	2.05	7.50	58.64
	36	1.29	.074	8.28×10^{-3}	1.37
	48	.032	.0022	1.56×10^{-5}	3.42×10^{-2}
High Density Concrete (dried)	12	5.6×10^3	21.83	2.47×10^4	3.03×10^4
	24	4.88×10^2	7.18	11.90	5.07×10^2
	36	47.09	1.63	1.56×10^{-2}	48.74
	48	4.53	0.268	3.49×10^{-5}	4.80

TABLE 20. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-U Fuel: 4" Iron Primary Shield and Concrete Secondary Shields

<u>Secondary Shield Material</u>	<u>Shield Thickness (Inches)</u>	<u>Neutron Dose Rate (mrem/hr)</u>	<u>Sec. Gamma Dose Rate (mrem/hr)</u>	<u>Pri. Gamma Dose Rate (mrem/hr)</u>	<u>Total Dose Rate (mrem/hr)</u>
Ordinary Poured Concrete	6	1.57×10^2	6.97	8.60×10^4	8.62×10^4
	21	0.941	0.345	80.41	81.70
	36	9.58×10^{-3}	4.59×10^{-3}	8.23×10^{-2}	9.64×10^{-2}
Ordinary Cured Concrete	6	2.24×10^2	5.21	9.69×10^4	9.71×10^4
	21	2.49	0.705	1.13×10^2	1.16×10^2
	36	3.90×10^{-2}	1.68×10^{-2}	0.148	0.204
Ordinary Heated Concrete	6	3.95×10^2	1.98	1.08×10^5	1.08×10^5
	21	13.44	1.58	1.75×10^2	1.90×10^2
	36	0.623	0.184	0.309	1.12
High Density Poured Concrete	6	34.53	6.23×10^{-2}	2.66×10^4	2.66×10^4
	21	0.335	4.88×10^{-2}	2.07	2.45
	36	1.50×10^{-3}	2.47×10^{-4}	4.67×10^{-4}	2.21×10^{-3}
High Density Cured Concrete	6	1.84×10^2	4.61	2.92×10^4	2.94×10^4
	21	0.743	7.79×10^{-2}	2.81	3.63
	36	4.48×10^{-3}	4.79×10^{-4}	7.52×10^{-4}	5.71×10^{-3}
High Density Heated Concrete	6	2.48×10^2	4.20	3.07×10^4	3.09×10^4
	21	1.72	0.174	3.37	5.26
	36	1.57×10^{-2}	1.72×10^{-3}	1.03×10^{-3}	1.85×10^{-2}
No Secondary Shield 4" Iron Primary Shield Only		2.02×10^3	4.11	1.66×10^6	1.66×10^6

TABLE 21. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-Pu Fuel: 4" Iron Primary Shield and Concrete Secondary Shields

<u>Secondary Shield Material</u>	<u>Shield Thickness (inches)</u>	<u>Neutron Dose Rate (mrem/hr)</u>	<u>Sec. Gamma Dose Rate (mrem/hr)</u>	<u>Pri. Gamma Dose Rate (mrem/hr)</u>	<u>Total Dose Rate (mrem/hr)</u>
Ordinary Poured Concrete	6	2.19×10^3	97.53	2.30×10^4	2.53×10^4
	21	13.16	4.83	21.50	39.49
	36	0.134	6.42×10^{-2}	2.20×10^{-2}	0.220
Ordinary Cured Concrete	6	3.13×10^3	72.90	2.59×10^4	2.91×10^4
	21	34.87	9.86	30.32	75.05
	36	0.546	0.235	3.95×10^{-2}	0.821
Ordinary Heated Concrete	6	5.53×10^3	27.76	2.91×10^4	3.47×10^4
	21	1.88×10^2	22.06	46.88	2.57×10^2
	36	8.72	2.58	8.28×10^{-2}	11.38
High Density Poured Concrete	6	4.83×10^2	0.872	7.15×10^3	7.63×10^3
	21	4.69	0.683	0.556	5.93
	36	2.10×10^{-2}	3.45×10^{-3}	1.25×10^{-4}	2.46×10^{-2}
High Density Cured Concrete	6	2.58×10^3	64.49	7.80×10^3	1.04×10^4
	21	10.39	1.09	0.750	12.23
	36	6.26×10^{-2}	6.70×10^{-3}	2.01×10^{-4}	6.95×10^{-2}
High Density Heated Concrete	6	3.47×10^3	58.74	8.25×10^3	1.18×10^4
	21	24.02	2.44	0.905	27.37
	36	0.22	0.024	2.75×10^{-4}	0.244
No Secondary Shield 4" Iron Primary Shield Only		2.83×10^4	57.5	4.45×10^5	4.73×10^5

TABLE 22. Dose Rates Through Composite Shields Due to High-Level Wastes From Spent PWR-U Fuel: 8" Iron Primary Shield and Concrete Secondary Shields

Secondary Shield Material	Shield Thickness (inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Ordinary Poured Concrete	6	57.92	2.09	5.98×10^2	6.58×10^2
	21	0.225	0.271	0.823	1.32
	36	1.43×10^{-3}	2.36×10^{-3}	1.04×10^{-3}	4.83×10^{-3}
Ordinary Cured Concrete	6	82.94	3.34	6.69×10^2	7.55×10^2
	21	0.666	0.418	1.23	2.31
	36	7.44×10^{-3}	8.51×10^{-3}	2.43×10^{-3}	1.84×10^{-2}
Ordinary Heated Concrete	6	1.68×10^2	1.18	8.45×10^2	1.01×10^3
	21	5.43	1.01	2.12	8.56
	36	0.256	0.111	5.72×10^{-3}	0.373
High Density Poured Concrete	6	41.97	3.71	2.45×10^2	2.91×10^2
	21	5.15×10^{-2}	2.34×10^{-2}	4.23×10^{-2}	0.117
	36	2.12×10^{-4}	9.80×10^{-5}	1.33×10^{-5}	3.23×10^{-4}
High Density Cured Concrete	6	67.35	2.93	2.66×10^2	3.36×10^2
	21	0.136	3.88×10^{-2}	5.57×10^{-2}	3.31×10^{-1}
	36	6.46×10^{-4}	1.72×10^{-4}	2.14×10^{-5}	8.39×10^{-4}
High Density Heated Concrete	6	92.95	2.65	2.80×10^2	3.76×10^2
	21	0.380	9.37×10^{-2}	6.66×10^{-2}	0.540
	36	2.41×10^{-3}	6.56×10^{-4}	2.86×10^{-5}	3.09×10^{-3}
No Secondary Shield 8" Iron Primary Shield Only		9.87×10^2	1.62	1.40×10^4	1.50×10^4

TABLE 23. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-Pu Fuel: 8" Iron Primary Shield and Concrete Secondary Shields

Secondary Shield Material	Shield Thickness (inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Ordinary Poured Concrete	6	8.10×10^2	29.2	1.60×10^2	1.0×10^3
	21	3.15	3.80	0.22	7.17
	36	0.020	0.033	2.80×10^{-4}	5.05×10^{-2}
Ordinary Cured Concrete	6	1.16×10^3	46.76	1.79×10^2	1.39×10^3
	21	9.31	5.84	0.329	15.48
	36	0.104	0.119	6.51×10^{-4}	0.224
Ordinary Heated Concrete	6	2.35×10^3	16.5	2.26×10^2	2.59×10^3
	21	75.9	14.1	0.568	90.57
	36	3.58	1.55	1.53×10^{-3}	5.13
High Density Poured Concrete	6	5.87×10^2	51.9	65.55	7.04×10^2
	21	0.720	0.327	1.13×10^{-2}	1.06
	36	2.97×10^{-3}	1.37×10^{-3}	3.57×10^{-6}	4.34×10^{-3}
High Density Cured Concrete	6	9.42×10^2	40.94	71.10	1.05×10^3
	21	1.90	0.543	1.49×10^{-2}	2.46
	36	9.04×10^{-3}	2.41×10^{-3}	5.72×10^{-6}	1.15×10^{-2}
High Density Heated Concrete	6	1.30×10^3	37.1	75.04	1.41×10^3
	21	5.32	1.31	1.78×10^{-2}	6.65
	36	3.37×10^{-2}	9.17×10^{-3}	7.65×10^{-6}	4.29×10^{-2}
No Secondary Shield 8" Iron Primary Shield Only		1.38×10^4	22.6	3.73×10^3	1.76×10^5

TABLE 24. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-U Fuel: 12" Iron Primary Shield and Concrete Secondary Shields

<u>Secondary Shield Material</u>	<u>Shield Thickness (inches)</u>	<u>Neutron Dose Rate (mrem/hr)</u>	<u>Sec. Gamma Dose Rate (mrem/hr)</u>	<u>Pri. Gamma Dose Rate (mrem/hr)</u>	<u>Total Dose Rate (mrem/hr)</u>
Ordinary Poured Concrete	6	35.04	1.29	5.34	41.67
	21	6.44×10^{-2}	0.177	1.34×10^{-2}	0.255
	36	2.79×10^{-4}	1.22×10^{-3}	3.29×10^{-5}	1.53×10^{-3}
Ordinary Cured Concrete	6	38.54	2.09	5.91	46.54
	21	0.277	0.259	1.77×10^{-2}	0.554
	36	2.20×10^{-3}	4.99×10^{-3}	5.46×10^{-5}	7.24×10^{-3}
Ordinary Heated Concrete	6	83.66	0.722	6.58	90.96
	21	2.91	0.648	2.58×10^{-2}	3.58
	36	0.141	6.96×10^{-2}	1.05×10^{-4}	0.211
High Density Poured Concrete	6	17.59	2.31	2.24	22.14
	21	8.58×10^{-3}	1.32×10^{-2}	8.49×10^{-4}	2.26×10^{-2}
	36	2.85×10^{-5}	5.18×10^{-5}	3.80×10^{-7}	8.07×10^{-5}
High Density Cured Concrete	6	30.24	1.83	2.44	34.51
	21	3.12×10^{-2}	2.20×10^{-2}	1.11×10^{-3}	5.43×10^{-2}
	36	9.22×10^{-5}	8.51×10^{-5}	6.06×10^{-7}	1.78×10^{-4}
High Density Heated Concrete	6	45.83	1.66	2.54	50.03
	21	0.118	5.51×10^{-2}	1.30×10^{-3}	0.174
	36	3.98×10^{-4}	3.20×10^{-4}	7.99×10^{-7}	7.19×10^{-4}
No Secondary Shield 8" Iron Primary Shield Only		5.33×10^2	0.808	1.51×10^2	6.85×10^{-2}

TABLE 25. Dose Rates Through Composite Shields Due to High-Level Waste From Spent PWR-Pu Fuel: 12" Iron Primary Shield and Concrete Secondary Shields

Secondary Shield Material	Shield Thickness (inches)	Neutron Dose Rate (mrem/hr)	Sec. Gamma Dose Rate (mrem/hr)	Pri. Gamma Dose Rate (mrem/hr)	Total Dose Rate (mrem/hr)
Ordinary Poured Concrete	6	4.9×10^2	18.0	1.43	5.09×10^2
	21	0.90	2.48	3.58×10^{-3}	3.38
	36	3.90×10^{-3}	0.017	8.80×10^{-6}	2.09×10^{-2}
Ordinary Cured Concrete	6	5.39×10^2	29.24	1.58	5.70×10^2
	21	3.88	3.62	4.72×10^{-3}	7.50
	36	3.08×10^{-2}	6.98×10^{-2}	1.46×10^{-5}	1.01×10^{-1}
Ordinary Heated Concrete	6	1.17×10^3	10.1	1.76	1.18×10^3
	21	40.7	9.06	6.89×10^{-3}	49.77
	36	1.97	0.974	2.82×10^{-5}	2.94
High Density Poured Concrete	6	2.46×10^2	32.25	0.601	2.79×10^2
	21	0.120	0.184	2.27×10^{-4}	0.304
	36	3.98×10^{-4}	7.24×10^{-4}	1.02×10^{-7}	1.12×10^{-3}
High Density Cured Concrete	6	4.23×10^2	25.60	0.653	4.49×10^2
	21	0.437	0.308	2.96×10^{-4}	0.745
	36	1.29×10^{-3}	1.19×10^{-3}	1.62×10^{-7}	2.48×10^{-3}
High Density Heated Concrete	6	6.41×10^2	23.2	0.682	6.65×10^2
	21	1.65	0.77	3.49×10^{-4}	2.42
	36	5.56×10^{-3}	4.48×10^{-3}	2.14×10^{-7}	1.00×10^{-2}
No Secondary Shield 8" Iron Primary Shield Only		7.45×10^3	11.3	40.5	7.50×10^3

TABLE 26. Neutron Dose Rates Under the Lifted RSSF Module
Half-Way Between the Ground and the Module

Radial Distance from Centerline (cm)	mrem/hr		
	6" Lift	18" Lift	3' Lift
0.0	1.19×10^4	4.75×10^3	1.72×10^3
15.5	6.36×10^3	3.11×10^3	1.31×10^3
23.62	1.94×10^3	2.15×10^3	9.93×10^2
53.54	49.	1.35×10^2	1.45×10^2
65.35	21.	62.	54.

TABLE 27. Gamma Dose Rates Under The Lifted RSSF Module
Half-Way Between the Ground and the Module

Radial Distance from Centerline (cm)	mrem/hr		
	6" Lift	18" Lift	3' Lift
0.0	1.79×10^7	6.50×10^6	1.86×10^6
15.5	3.18×10^6	1.66×10^6	9.12×10^5
23.62	4.77×10^5	6.05×10^5	4.59×10^5
53.54	1.28×10^4	1.95×10^4	4.88×10^4
65.35	6.96×10^3	8.39×10^3	2.41×10^4

TABLE 28. Dose Rate at the Streaming Exit of
the RSSF Module Designs

<u>Design</u>	mrem/hr	
	<u>PWR-Pu Neutron Source</u>	<u>PWR-U Gamma Source</u>
Figure 1	49.	1.4×10^4
Figure 2	15.	75.
Figure 3	2.	34.
Figure 4	-	4.
Figure 5	-	300.
Figure 6	-	8.

FIGURE 1. Model of Retrievable Surface Storage Facility Module

CALCULATION PROCEDURE

FIGURE 2. Computer Calculation Procedure

FIGURE 3. Dose Rates in Infinite Concrete From a Point Fission Source

FIGURE 4. Geometry of the RSSF Storage Unit Used for ANISN Calculations

FIGURE 5. ANISN Geometry for a Water Basin Storage Array

FIGURE 6. Calculation Geometry for the Welding Problem

FIGURE 7. Streaming Model for RSSF Storage Unit

FIGURE 8. First Anti-Stream Design For RSSF Storage Unit

FIGURE 9. Second Anti-Stream Design for RSSF Storage Unit

FIGURE 10. Third Anti-Stream Design for RSSF Storage Unit

FIGURE 10. (Cont.)

FIGURE 11. Fourth Anti-Stream Design for RSSF Storage Unit
(see Figure 1)

FIGURE 12. Fifth Anti-Stream Design For RSSF Storage Unit

FIGURE 13. Gamma Source Vs Time After Shutdown Per Tonne Heavy Metal Charge

FIGURE 14. Neutron Source Vs Time After Shutdown Per Tonne Heavy Metal Charge

FIGURE 15. Dose Rates at the Surface of an Iron Shield for PWR-U Cycle Waste

FIGURE 16. Dose Rates at Surface of an Iron Shield for PWR-Pu Recycle Waste

FIGURE 17. Dose Rates at Surface of a Water Shield for PWR-U Cycle Waste

FIGURE 18. Dose Rates at Surface of a Water Shield for PWR-Pu Recycle Waste

FIGURE 19. Total Dose Rates at the Surface of a Water Shield for PWR-Pu Recycle Waste

FIGURE 20. Total Dose Rates at the Surface of a Water Shield for PWR-U Cycle Waste

FIGURE 21. Total Dose Rates at the Surface of a Commercial Shielding Material for a PWR-U Cycle Waste

FIGURE 22. Total Dose Rates at the Surface of a Commercial Shielding Material for PWR-Pu Recycle Waste

FIGURE 23. Dose Rates at the Surface of an Ordinary Cured Concrete Shield for PWR-U Cycle Wastes

FIGURE 24. Dose Rates at the Surface of an Ordinary Cured Concrete Shield for PWR-Pu Recycle Wastes

FIGURE 25. Total Dose Rates at the Surface of an Ordinary Concrete Shield for PWR-U Waste

FIGURE 26. Total Dose Rates at the Surface of an Ordinary Concrete Shield for PWR-Pu Waste

FIGURE 27. Dose Rates at the Surface of a High Density Cured Concrete Shield for PWR-U Waste

FIGURE 28. Dose Rates at the Surface of a High Density Cured Concrete Shield for PWR-Pu Waste

FIGURE 29. Total Dose Rates at the Surface of a High Density Concrete Shield for PWR-U Waste.

FIGURE 30. Total Dose Rates at the Surface of a High Density Concrete Shield for PWR-Pu Waste

FIGURE 31. Dose Rate at the Surface of a Water Filled Storage Basin

FIGURE 31. (Continued)

FIGURE 32. Dose Rates From Canister End as a Function of Pb Thickness

FIGURE 32. (Continued)

FIGURE 33. Relative Shield Effectiveness of Water After 25 cm Lead

FIGURE 34. Dose Rates From Canister End as a Function of Steel Thickness

FIGURE 34. (Continued)

FIGURE 35. Relative Shield Effectiveness of Water After 25 cm of Mild Steel

FIGURE 36. Gamma Dose Rate Vs Distance From Storage Array PWR-U

FIGURE 37. Neutron Dose Rate Vs Distance From Storage Array PWR-Pu

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