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CONTRACTOR REPORT

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Effects of Various Radiation Source Characteristics on Shielding Requirements at the Potential Yucca Mountain Repository

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EFFECTS OF VARIOUS RADIATION SOURCE
CHARACTERISTICS ON SHIELDING REQUIREMENTS
AT THE POTENTIAL YUCCA MOUNTAIN REPOSITORY

by

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ABSTRACT

This radiation shielding study provides dose rate information that can be used to estimate required shielding thicknesses for different repository configurations, including various hot cells and vaults in the waste-handling building, the boreholes in the underground emplacement area, and the transfer casks. The study determines gamma and neutron source strengths for various waste types and source geometries representative of conditions at the repository and determines dose rates as a function of shielding thickness for selected materials.

This work was completed November 1989.

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1.0 INTRODUCTION

1.1 Purpose

The purpose of the study is to evaluate radiation shielding requirements for selected gamma and neutron source strengths and geometries that are representative of conditions that may be encountered at the repository.

1.2 Scope

This study determines gamma and neutron radiation source strengths for selected waste types, develops source geometries representative of conditions at the repository, and determines dose rate profiles as a function of shielding thickness for selected materials.

1.3 Organizational Approach

These shielding analyses use the following approach:

- o Determine gamma and neutron radiation source strengths for selected waste types,
- o Develop geometric models of the sources and shielding,
- o Calculate dose rate versus shield thickness for each geometric model, and
- o Identify any significant conclusions resulting from the calculations

This report consists of eight sections. Section 1.0 is the introduction. Section 2.0 presents the bases for this study,

including methods of analysis, waste type descriptions, reference facility description, materials description, and assumptions. Section 3.0 describes the computer programs used in the gamma and neutron analyses, and includes QA material. Section 4.0 identifies the source strength data used in shielding calculations. Section 5.0 presents the gamma dose rate analysis. Section 6.0 presents the neutron dose rate analysis, which also includes calculations for the transfer cask. Section 7.0 summarizes the conclusions of this study and identifies further considerations and recommendations. And Section 8.0 lists the references used in the report.

2.0 STUDY BASES

2.1 General Methodology

The shielding design for the repository should consider only two types of radiation: gamma rays and neutrons. Alpha and beta radiation do not exhibit significant penetrating power. Both gamma rays and neutrons interact in many ways with shielding media, but for the purpose of this study, a simplified concept of these processes is sufficient. Figure 2-1 illustrates this concept. The primary gamma flux has two components--scattered and unscattered. The neutron flux has an additional component--capture gammas or other gammas generated from neutron interactions with atoms of the shielding material.

2.1.1 Gamma Ray Attenuation

Attenuation of the primary gamma unscattered flux follows the general equation (Schaeffer, 1973),

$$F = \frac{S}{4\pi r^2} \exp(-uX),$$

where F = flux at the receiver point;
 S = source strength;
 u = linear attenuation coefficient, which is
 characteristic of the shielding medium and is
 a function of gamma energy;
 X = distance through the shield; and
 r = distance from the source to the receiver point.

The $1/4\pi r^2$ term relates to the geometric attenuation due to the spreading of the gamma ray wavefront over a sphere of area $4\pi r^2$ as the gamma rays cover the distance r from the flux source to the receiver. The $\exp(-uX)$ term relates to the attenuation due to the interaction of the gamma rays with matter that lies between the flux source and the receiver.

Figure 2-1. Radiation Interactions Within Shield Walls

Gamma attenuation due to interaction with matter can be described by three distinct processes (Schaeffer, 1973), commonly called photoelectric capture, pair production, and Compton scattering. The first two of these are annihilation processes in which the gamma ray is destroyed. In Compton scattering, the original gamma interacts with an electron, is deflected from its path, and in the process loses part of its energy.

For gamma scattering interactions (primarily Compton scattering), the contribution to the total flux at the receiver point is taken into account by multiplying the unscattered flux, determined in the above equation by a dimensionless number, the buildup factor. The buildup factor is the ratio of the actual gamma flux to that which would be calculated using a simple exponential attenuation with the linear attenuation coefficient. Since flux is directly proportional to dose rate, this buildup factor is also applied to dose rate calculations.

2.1.2 Neutron Attenuation

Neutrons can interact with matter in several ways:

- o Collisions with hydrogen nuclei, which reduce the neutron's energy by an amount ranging from 0% to 100% and change its direction from 0° to 90°;
- o Elastic (billiard ball) collisions with other heavier nuclei;
- o Inelastic collisions with nuclei; and
- o Capture by hydrogen nuclei or other shield material, usually at or near thermal neutron energies.

Gamma rays may result from the capture of neutrons or the inelastic scattering of fast neutrons. For this study, all gammas resulting from neutron interactions with matter are called capture gammas.

Fast, intermediate, and thermal energy (slow) neutrons contribute to the total neutron dose rate. In determining the neutron dose rates, this study takes into account all energy groups.

2.1.3 Source and Shield Geometries

The geometry of the radiation source and shield is another consideration in shielding calculations. The source is modeled as accurately as the computer code permits. Diagrams are provided in the gamma and neutron analyses that illustrate how source and shield geometries are treated for each specific case.

For fuel assemblies, a right parallelepiped with actual source dimensions was selected in the gamma analysis. For single containers, right circular cylinders in the source region were chosen. For storage racks and the surface storage vault (SSV), source homogenization was used and modeled as right parallelepipeds. In most cases, except for the SSV, the source container was assumed to be intimate with the shielding.

Because of the one-dimensional capability of the ANISN (Engle, 1967) computer program, the neutron source geometry is modeled either as an infinitely long cylinder of finite radius (for radial dose rate calculations) or as an infinite slab with a finite slab thickness (for axial dose rate calculations), as described in Subsection 3.2.2.

2.1.4 Computer Codes

Computer codes that employ the theory described above were used in these analyses to provide gamma and neutron dose rates in various shielding media. CYLSEC is a three-dimensional primary gamma computer program used for the gamma analysis. The one-dimensional ANISN (Engle, 1967) computer program was used for the neutron analysis. These codes are described in Section 3.0.

2.1.5 Additional Considerations

The analyses in this study determine the variation of dose rates in bulk shielding from different types of waste sources. The following shielding details should be considered later in the definitive design:

- o Protective coatings applied to shielding,
- o Penetrations through shielding, and
- o Internal shielding voids or irregularities.

For this study, the effects of reinforcing steel on shielding characteristics are neglected. The steel bars may be regarded as an additional source of capture gammas in a neutron analysis, which may be counterbalanced by the improved primary gamma attenuating properties of the steel itself.

2.2 Description of Waste Types

Three different waste forms - pressurized water reactor (PWR) spent fuel, boiling water reactor (BWR) spent fuel, and high-level waste (HLW) - were considered in these analyses. Vitrified waste from the West Valley Demonstration Project (WVDP) and defense high-level waste (DHLW) are included in the HLW category.

Different waste type configurations were considered in the shielding calculations. For the gamma analysis, the configurations are as follows:

- o Single (consolidated) PWR fuel canister with rods from two assemblies in the volume of one;
- o Single (consolidated) BWR fuel canister with rods from two assemblies in the volume of one;
- o Single container of 10 intact BWR fuel assemblies (BWR-10);

- o Single container of 18 consolidated BWR fuel assemblies (BWR-18);
- o Single container of 4 intact BWR and 3 intact PWR fuel assemblies (hybrid);
- o Single container of 4 intact PWR fuel assemblies (PWR-4);
- o Single container of 6 consolidated PWR fuel assemblies (PWR-6);
- o Single container of Hanford DHLW;
- o Single container of Idaho National Engineering Laboratory (INEL) DHLW;
- o Single container of Savannah River Plant (SRP) DHLW;
- o Single container of WVDP HLW;
- o Single container of spent fuel in the borehole;
- o Arrays of spent fuel and HLW canisters in the surface storage vault; and
- o Array of spent fuel or HLW in a storage rack.

For the neutron analysis, the spent fuel source strengths and source configurations that would generate the highest total dose rates (BWR-18 and PWR-6) were considered.

General dimensions of single PWR and BWR fuel assemblies are assumed from Roddy (1986). HLW container dimensions are from DOE (1987). The dimensions of containers that hold multiple spent fuel assemblies are from SNL (1987).

2.3 Description of Reference Facility

The areas considered in this study include the shielded hot cells and vaults in the waste-handling buildings, boreholes, SSV, and other general areas of the facility.

Reference information was from the Site Characterization Plan - Conceptual Design Report (SCP-CDR) (SNL, 1987).

2.3.1 Waste-Handling Buildings

The major activities performed in the waste-handling buildings are waste unloading, preparation, packaging, and interim storage. The buildings are designed to receive and process up to 3,000 metric tons of uranium (MTU) as spent fuel and 400 MTU of vitrified HLW annually. All spent fuel and HLW are processed through the waste-handling buildings. There are two waste-handling buildings - one for spent fuel and one for HLW.

Incoming shipping casks are handled in the receiving and shipping bays, with an airlock at each bay for vehicle entrance and exit. This area is a high-bay, steel-framed structure with bridge cranes and auxiliary hoists to unload casks from transport vehicles. Next to this area is a truck bay for collecting and transferring site-generated waste to the waste treatment building.

A cask preparation area is located adjacent to the receiving and shipping bays. Motorized transfer cars are provided in the preparation area. Each car receives one waste cask from a bridge crane in the bay. The preparation area is equipped with work platforms, robots, long-handled tools, and other devices required to survey the inner lid surfaces, sample inner cavity gases, and install a cask seal adapter (for mating to the hot cell) before removing the spent fuel assemblies.

Each cask is moved by a transfer car through the transfer tunnel under an unloading hot cell, where the inner lid of the cask is removed. Bridge cranes in the unloading hot cells remove the cask lid, lift the fuel assemblies through the unloading port into the cell, and place the assemblies in storage racks. Separate hot cells handle spent fuel and vitrified HLW unloading. Equipment in each cell packages the received waste in containers and decontaminates the containers, if necessary. Empty casks are returned to the cask preparation area, where their interiors and exteriors are decontaminated, as necessary.

The waste containers are transferred to the SSV to await transportation underground. The SSV is adjacent to the cask-unloading hot cell and has a 1-1/2-month lag storage capacity. The vault is equipped with a container transfer machine which contains a shielded transfer cask into which containers are lifted during operations. Approximately 112 container positions are available in the vault, 56 in each of two subvaults separated by a partial-height concrete partition. At the loading port of the out-transfer bay adjacent to the SSV, containers are placed on a car to await transfer to the waste transporter. The transporter moves the waste containers underground for emplacement into the boreholes.

2.3.2 Performance Confirmation Building

Containers of spent fuel and HLW will be removed from emplacement for observation, testing, and evaluation as part of the repository performance confirmation program. Retrieved waste containers will be received, sampled, tested, and repackaged in the performance confirmation building for return and emplacement underground. The building is located in the geologic repository operations area near the portal of the waste-handling ramp. Container handling takes place in a reinforced-concrete, shielded hot cell equipped with remote-handling equipment.

2.3.3 Boreholes

For purposes of this study, vertical emplacement is assumed for the subsurface boreholes. However, shielding analysis results may also be applicable for a horizontal emplacement scheme. The borehole can be generalized as a 29-in.-diameter cylindrical hole surrounded by tuff rock. The borehole for spent fuel is approximately 25 ft deep and is equipped with a shield plug at the top for radiation protection purposes. The HLW borehole is approximately 20 ft deep and also has a 29-in. diameter.

2.4 Description of Materials Used in Analyses

It is important to accurately describe the shielding and source materials used in the gamma and neutron analyses. For the gamma analysis, the materials are described by the mass attenuation coefficients, the bulk densities, and the dose buildup factors. With the exception of the material density for all materials and the mass attenuation coefficients for tuff rock, the material data described above are contained in a data library as part of the CYLSEC computer program. Bulk material density is a code input parameter. The linear and mass attenuation coefficients as a function of mean gamma energy for tuff are also input to the code and are given in Table 2-1.

The attenuation coefficients for tuff were calculated by:

- o Determining the average composition and average dry bulk density from the Site Characterization Plan (DOE, 1988) (the use of dry tuff composition gives conservative results for neutron doses and required shielding thickness, because the low water content of tuff will increase its neutron attenuation);

TABLE 2-1

GAMMA ATTENUATION COEFFICIENTS OF TUFF ROCK

Mean Gamma Energy (MeV)	Linear Attenuation Coefficients (cm ⁻¹)	Mass Attenuation Coefficients* (cm ² /g)
1.00E-2	1.2944E+0	5.8836E-1
2.50E-2	1.0686E+0	4.8573E-1
3.75E-2	8.8042E-1	4.0019E-1
5.75E-2	5.7935E-1	2.6334E-1
8.50E-2	4.0025E-1	1.8193E-1
1.25E-1	3.3033E-1	1.5015E-1
2.25E-1	2.6270E-1	1.1941E-1
3.75E-1	2.1525E-1	9.7841E-2
5.75E-1	1.7970E-1	8.1682E-2
8.50E-1	1.5089E-1	6.8586E-2
1.25E+0	1.2622E-1	5.7373E-2
1.75E+0	1.0550E-1	4.7955E-2
2.25E+0	9.3142E-2	4.2337E-2
2.75E+0	8.4074E-2	3.8215E-2
3.50E+0	7.4421E-2	3.3828E-2
5.00E+0	6.2754E-2	2.8525E-2
7.00E+0	5.5441E-2	2.5200E-2
9.50E+0	5.0214E-2	2.2825E-2

* The designation for mass attenuation coefficient is μ_m/ρ , where ρ denotes the material density, which in the case of tuff is 2.2 g/cm³.

- o Calculating the elemental composition of the tuff and the weight percent values of each significant element;
- o Calculating the elemental partial densities from the average dry bulk density of the tuff and the elemental weight percents;
- o Importing these data and the desired energy structure into the QAD-CG computer program, which incorporates a routine to calculate the linear attenuation coefficients; and
- o Dividing the linear attenuation coefficients by the average dry bulk density of the tuff rock to obtain the mass attenuation coefficients.

In the neutron analysis, the ANISN computer code describes shielding and source materials primarily by generating coupled neutron and gamma cross section data for multiple energy groups. In the analysis discussed here, the DLC-23/CASK format for cross sectional data input was used.

ANISN/CASK requires that atomic densities for each element of each source or shield material be calculated and input to the code. These material data for spent fuel are given in Tables 2-2 and 2-3. Tables 2-4 through 2-8 provide material data for shielding used in the neutron analysis.

In the transfer cask calculations, a specific type of borated polyethelene was assumed: Type 259 produced by Reactor Experiments, Inc. (1985) and called POLY/CAST. POLY/CAST is a dry-mix material used for casting into closed containers. It is field-castable and has a 6% greater hydrogen content than water. There is 0.9% boron present to suppress radiation from capture gammas. Capture gammas resulting from the capture of neutrons by hydrogen nuclei have an energy of about 2.2 MeV. Gammas resulting from the capture of neutrons by boron have an energy of only 0.42

TABLE 2-2

MATERIAL DATA FOR BWR-18 CONSOLIDATED FUEL*

Significant Element or Isotope	Partial Density (g/cm ³)	Atom Density (atom/barn-cm)
O	0.46	1.7314E-02
Zr	1.91	1.2609E-02
U-235	0.09	2.2586E-04
U-238	3.30	8.3545E-03

Fuel region average density = 5.76 g/cm³.

Average enrichment = 2.6% (DOE, 1987).

* BWR-18 denotes a single container of 18 consolidated BWR fuel assemblies.

TABLE 2-3

MATERIAL DATA FOR PWR-6 CONSOLIDATED FUEL

<u>Significant Element or Isotope</u>	<u>Partial Density (g/cm³)</u>	<u>Atom Density (atom/barn-cm)</u>
O	0.39	1.4679E-02
Zr	0.69	4.5335E-03
U-235	0.08	2.0203E-04
U-238	2.84	7.1888E-03

Fuel region average density = 4.0 g/cm³.

Average enrichment = 2.7% (DOE, 1987).

* PWR-6 denotes a single container of 6 consolidated PWR fuel assemblies.

TABLE 2-4

MATERIAL DATA FOR LEAD AND WATER

<u>Element</u>	<u>Water</u>		<u>Lead</u>	
	<u>(wt %)</u>	<u>Atom Density (atom/barn-cm)</u>	<u>(wt %)</u>	<u>Atom Density (atom/barn-cm)</u>
H	0.111	6.6959E-02	--	--
O	0.889	3.3479E-02	--	--
Pb	--	--	100	3.2958E-02

Lead density = $11.34 \text{ g/cm}^3 = 710 \text{ lbm/ft}^3$.

Water density = $1.0 \text{ g/cm}^3 = 62.4 \text{ lbm/ft}^3$.

TABLE 2-5

MATERIAL DATA FOR CARBON STEEL AND STAINLESS STEEL

<u>Element</u>	<u>Carbon Steel (wt %)</u>	<u>Carbon Steel Atom Density (atom/barn-cm)</u>	<u>Stainless Steel (wt %)</u>	<u>Stainless Steel Atom Density (atom/barn-cm)</u>
Fe	98.155	8.40E-02	68.0	5.76E-02
Cr	--	---	19.0	1.73E-02
Ni	--	---	10.0	8.06E-03
C	0.270	1.10E-03	0.08	3.18E-04
Mn	1.000	8.59E-04	2.0	1.73E-03
Si	0.250	4.19E-04	1.0	1.69E-03
S	0.040	5.87E-05	0.03	4.44E-05
P	0.035	5.32E-05	0.045	6.90E-05

Carbon steel (SA-516) density = $7.83 \text{ g/cm}^3 = 490 \text{ lbm/ft}^3$.

Stainless steel (Type 304) density = $7.86 \text{ g/cm}^3 = 492 \text{ lbm/ft}^3$.

TABLE 2--6

MATERIAL DATA FOR BORATED POLYETHELENE*

<u>Element</u>	<u>Wt %</u>	<u>Atom density (atom/barn-cm)</u>
H	10.33	7.10E-02
B	0.90	5.79E-04
C	45.99	2.65E-02
O	32.52	1.41E-02
Al	0.03	6.70E-06
Si	0.43	1.05E-04
S	3.99	8.62E-04
Ca	5.72	9.89E-04
Mn	0.04	5.48E-06
Fe	0.05	5.39E-06

Density = 1.15 g/cm³ = 72 lbm/ft³.

* These material data are for a specific type of borated polyethelene:
Type 259, POLY/CAST provided by Reactor Experiments, Inc. (1985).

TABLE 2-7

MATERIAL DATA FOR TUFF ROCK

<u>Significant Element</u>	<u>Wt %</u>	<u>Atom density (atom/barn-cm)</u>
O	49.2	4.07E-02
Na	2.6	1.50E-03
Al	6.54	3.21E-03
Si	36.16	1.70E-02
K	4.65	1.58E-03
Ca	0.29	9.58E-05
Fe	0.56	1.33E-04

Dry bulk density = $2.2 \text{ g/cm}^3 = 138 \text{ lbm/ft}^3$.

Effective atomic number (effective-Z) = 11.2.

TABLE 2-8

MATERIAL DATA FOR ORDINARY CONCRETE

<u>Element</u>	<u>Wt %</u>	<u>Atom density (atom/barn-cm)</u>
H	0.56	7.47E-03
O	49.56	4.20E-02
Na	1.71	1.00E-03
Mg	0.24	1.36E-04
Al	4.56	2.28E-03
Si	31.35	1.50E-02
S	0.12	5.05E-05
K	1.92	6.59E-04
Ca	8.26	2.77E-03
Fe	1.22	2.94E-04

Density = $2.24 \text{ g/cm}^3 = 140 \text{ lbm/ft}^3$.

Effective-Z = 11.5.

Source: ANS (1985).

MeV, which significantly reduces the resulting overall dose rate. The manufacturer (Reactor Experiments, Inc., 1985) has stated that "POLY/CAST has been used to cast a variety of shielding wall barriers, and even an entire 4-ft-thick roof for a cyclotron facility." The importance of a free-flowing, field-castable neutron shielding material is that air pockets and voids are minimized around inner horizontal supports or spacers between the half-inch stainless steel container material.

2.5 Assumptions

The following assumptions were made in this study:

- o All gamma and neutron source strengths are taken from DOE (1987).
- o All fuel, HLW, and container dimensional information is from Roddy (1986) and DOE (1987).
- o For gamma shielding materials, the density of ordinary concrete is 140 lbm/ft^3 , the density of carbon steel is 490 lbm/ft^3 , the density of stainless steel is 492 lbm/ft^3 , and the density of tuff is 138 lbm/ft^3 .
- o For tuff rock gamma calculations, concrete dose buildup factors are based on the effective atomic number (effective-Z) of each material.
- o For gamma dose rates calculated for the waste container-to-shield interface location, uranium dose buildup factors are used for spent fuel and concrete dose buildup factors for HLW.

- o For surface storage vault gamma calculations, a 1-ft air gap exists between the top of the sources in the container and the bottom of the vault's roof slab. A 6-in. gap exists around all sides of the homogenized source region.
- o The storage rack has multiple source containers. Only one waste type needs to be considered. If the concrete gamma dose rates from multiple containers are divided by the gamma dose rate from a single container with the same source strength, the ratio should be relatively geometry-dependent, source-independent. (The ratios can then be multiplied by the gamma dose rates from any single source-type dose rate versus depth profile in concrete to obtain new multiple source gamma dose rates.)
- o For additional shielding materials used in the neutron dose rate analysis, the density of lead is 710 lbm/ft^3 , the density of water is 62.4 lbm/ft^3 , and the density of borated polyethelene is 72 lbm/ft^3 .

An S_8P_3 angular quadrature and order of scattering is used throughout all ANISN calculations.

3.0 COMPUTER CALCULATION METHODS

This section describes the computer codes used in gamma and neutron shielding calculations and discusses the documentation and verification of these computer programs to meet QA requirements. The CYLSEC computer program was used for the gamma analysis; for the neutron analysis, the ANISN computer program was employed in conjunction with DLC-23/CASK cross section libraries. The transfer cask calculations used ANISN/CASK for both gammas and neutrons.

3.1 Gamma Analysis Computer Code

CYLSEC is an interactive computer program designed to solve three-dimensional gamma ray transfer problems using the point kernel method. Any response function, such as dose rates, can be determined by specifying appropriate conversion factors and buildup factors. The program library contains attenuation coefficients and buildup factors for common shielding materials, and dose equivalent and absorbed dose conversion factors. However, the program does not contain library data for tuff rock; these data, described in Subsection 2.4, were input into the computer program. The geometries modeled for CYLSEC consist of a source boundary and individual source and shield bodies. The source and shield geometries must be described by a set of up to 20 orthogonal slabs, right cylinders, and/or right parallelepipeds. The radiation source geometry may be modeled by a set of up to the maximum of 20 bodies that are right cylinders and/or right parallelepipeds within the source boundary.

3.2 Neutron Analysis Computer Codes

3.2.1 ANISN

The ANISN code performs multigroup neutron/gamma shielding and eigenvalue calculations through discrete ordinates solutions of the one-dimensional Boltzmann transport equation in slab, cylindrical, and spherical geometries.

As with any numerical solution, the discrete ordinates solution reduces continuous variables to a discrete set of values. The choice of values for the independent variables is problem-dependent and is a function of the physical characteristics of the problem. With the proper choice of values, the discrete ordinates solution gives a good approximation and in the limit, approaches the correct solution.

The discrete ordinates method is basically an approach to making discrete the variables of the integro-differential form of the Boltzmann equation to solve for the angular flux. The three-dimensional form of the Boltzmann equation has generally been avoided as too cumbersome for the discrete ordinates method. Instead, the reduced forms, such as in the ANISN code, have been developed and applied to problems for which the one-dimensional representation is sufficient.

The discrete ordinates method is based on several important assumptions: the energy range is divided into energy groups; the angular flux is defined at specified spatial points and energy groups along certain discrete angular directions; the scalar flux is constructed from the angular fluxes using a set of weights matched to the discrete directions; and the complex scattering kernel describing the probability of scattering from one energy and directional state to another is expanded, generally in terms of Legendre polynomials.

For shielding calculations, a fixed source is input into the code, along with such quantities as the energy group structure, the spatial description, the set of discrete angles and weights, and the cross section data. The number of angles used in the calculations is referred to as the quadrature order, N , and is of such relevance that the method is referred to as S_N . All neutron calculations in this study use the S_8 quadrature order.

The cross section data are dependent on such factors as the type of particles involved, the energy group structure, the order of scattering (the number of Legendre polynomial terms used in the expansion), and the number of different materials.

A capability not available in predecessors of ANISN is the inclusion of general anisotropic scatter, which allows better representation of deep-penetration problems, where angle-dependent spectra must be calculated in detail.

The major limitation of ANISN is the one-dimensional nature of its solution. Figures 3-1 and 3-2 show the ANISN model for performing dose rate calculations in this study. In shield wall calculations, the neutron source is assumed to be an infinitely long cylinder with a finite radius, an assumption that tends to yield slightly overestimated shield wall dose rates. In shield slab calculations, the neutron source is assumed to be an infinite slab of finite thickness, which overestimates the shield slab dose rates.

3.2.2 CASK

The DLC-23/CASK code contains a 40-group coupled neutron and gamma ray cross section data library compiled by the Oak Ridge National Laboratory for the U.S. Department of Energy. The coupled (P_3) cross sections are for elements that are commonly used for shielding calculations and are given in the ANISN format. Since its release in 1972, the code has been employed industry-wide for various shielding design applications.

Figure 3-1. ANISN Model for Shield Wall Calculations

Figure 3-2. ANISN Model for Shield Slab Calculations

3.3 QA Considerations

Bechtel proprietary computer programs have been successfully used in the design of NRC-regulated facilities. The shielding codes employed in this study (CYLSEC, CASK, and ANISN) are Bechtel Standard Computer Programs which meet Yucca Mountain Site Characterization Project requirements for QA. The codes currently have full documentation and verification that satisfy requirements in 10CFR50, Appendix B, for nuclear power plant licensing, which are equivalent to the 10CFR60, Subpart G, requirements for repositories.

4.0 SOURCE STRENGTHS

Source strength data used in these analyses are presented in Appendix A. They were taken from the data base in DOE (1987) and converted to make them suitable for direct input into CYLSEC and ANISN.

Gamma and neutron source strengths for BWR spent fuel were provided for four levels of fuel burnup - 10, 20, 27.5, and 40 Gwd/MTU. Gamma and neutron source strengths for PWR spent fuel were provided for five levels of burnup - 10, 25, 33, 45, and 60 Gwd/MTU. For each burnup level, source strengths were given for cooling times of 5 yr, 10 yr, 20 yr, and 30 yr.

HLW gamma source terms were provided for Hanford Site DHLW, SRP DHLW, INEL DHLW, and WVDP HLW. HLW source strengths were given for decay times of 5 yr, 10 yr, 20 yr, and 30 yr.

Data conversion for the neutron source strengths required a redistribution of energy-dependent source strengths into a DLC-23/CASK format suitable for input into the ANISN computer program. Any dimensional information on the fuel or containers used in data conversion was taken from Roddy (1986) and DOE (1987).

In this study, the neutron energy spectrum is assumed to be similar to the Cf-252 fission spectrum, which is given in Table 4-1. The Cf-252 spectrum has been shown to be slightly more conservative than the U-235 fission spectrum in some shielding materials with a low effective atomic number (effective-Z). It is expected that the dose rates from the Cf-252 spectrum will tend to approach the dose rates from the U-235 spectrum in shields having appreciable quantities of moderate- to high-atomic-number elements, which have large inelastic scattering cross sections for high-energy neutrons (Jaeger et al., 1970).

TABLE 4-1

CASK NEUTRON GROUP STRUCTURE AND SOURCE SPECTRUM

Neutron Group	Upper Energy (eV)	Source Spectrum* (neutron/sec)
1	1.492E+7	0.4653E-3
2	1.220E+7	0.1883E-2
3	1.000E+7	0.5756E-2
4	8.180E+6	0.1924E-1
5	6.360E+6	0.4000E-1
6	4.960E+6	0.5174E-1
7	4.060E+6	0.1094E+0
8	3.010E+6	0.8804E-1
9	2.460E+6	0.2088E-1
10	2.350E+6	0.1156E+0
11	1.830E+6	0.2089E+0
12	1.110E+6	0.1920E+0
13	5.500E+5	0.1327E+0
14	1.110E+5	0.1345E-1
15	3.350E+3	0.0
16	5.830E+2	0.0
17	1.010E+2	0.0
18	2.900E+1	0.0
19	1.010E+1	0.0
20	3.060E+0	0.0
21	1.120E+0	0.0
22	4.140E-1	0.0

* The source spectrum is the ratio of the source strength in each energy group to the total source strength.

5.0 GAMMA DOSE RATE ANALYSIS

A primary gamma analysis was performed to determine the variation of dose rate with shielding thickness for selected source strengths and geometries that are representative of conditions that may be encountered at the repository.

5.1 Waste Types Considered

The following waste types were considered in the gamma analysis:

- o Single canister of 2 consolidated BWR fuel assemblies,
- o Single canister of 2 consolidated PWR fuel assemblies,
- o Single container of 10 intact BWR fuel assemblies (BWR-10),
- o Single container of 18 consolidated BWR fuel assemblies (BWR-18),
- o Single container of 4 intact BWR plus 3 intact PWR fuel assemblies (hybrid),
- o Single container of 4 intact PWR fuel assemblies (PWR-4),
- o Single container of 6 consolidated PWR fuel assemblies (PWR-6),
- o Single container of Hanford site DHLW,
- o Single container of INEL DHLW,
- o Single container of SRP DHLW,
- o Single container of West Valley HLW,
- o Single container of spent fuel in the borehole,

- o Arrays of spent fuel and HLW containers in the surface storage vault (SSV)
- o Storage rack containing multiple containers of spent fuel or HLW containers

5.2 Gamma Source Strengths

The gamma source strengths for various types of waste described above were obtained from the data base in DOE (1987). Four different ages were considered for each type of waste. Four different burnups were considered for BWR spent fuel, five different burnups for PWR spent fuel.

Source strengths for various waste types are presented in Appendix A, Tables A-1 through A-14, in a form compatible with the CYLSEC computer program, along with the source multiplication constants provided in Table 5-1. CYLSEC requires gamma source strengths in gammas/cm³-sec. The source multiplication constants in Table 5-1 convert the source strengths presented in Appendix A to data in the appropriate dimensional units.

5.3 Problem Geometries

A simplified model was selected to represent each problem geometry. For two consolidated spent fuel assemblies, a simple right parallelepiped was used to represent the radiation source; for the remainder of the single waste containers, a right circular cylinder was selected.

For a fuel assembly or a single waste disposal container next to a shield wall or slab, the source was assumed to be immediately adjacent to the shielding, as shown in Figure 5-1. This figure shows the simplified representation of the source and shielding used in the gamma analysis and modeled for use in the CYLSEC computer program.

TABLE 5-1

GAMMA CODE SOURCE MULTIPLICATION CONSTANTS*

<u>Source Description</u>	<u>Multiplication Constant (MC)</u>
Single Canister of 2 Consolidated PWR Fuel Assemblies	5.54E-6 MTU/cm ³
Single Canister of 2 Consolidated BWR Fuel Assemblies	5.05E-6 MTU/cm ³
Single Container of 6 Consolidated PWR Fuel Assemblies	2.92E-6 MTU/cm ³
Single Container of 18 Consolidated BWR Fuel Assemblies	3.39E-6 MTU/cm ³
Single Container of 4 Intact PWR Fuel Assemblies	1.34E-6 MTU/cm ³
Single Container of 10 Intact BWR Fuel Assemblies	1.30E-6 MTU/cm ³
Hybrid Container of 3 Intact PWR Plus 4 Intact BWR Fuel Assemblies	7.25E-7 (cm ³) ⁻¹
Single Container of Hanford Site DHLW	1.60E-6 (cm ³) ⁻¹
Single Container of INEL DHLW	1.75E-6 (cm ³) ⁻¹
Single Container of SRP DHLW	1.69E-6 (cm ³) ⁻¹
Single Container of WVDP HLW	1.42E-6 (cm ³) ⁻¹
Surface Storage Vault (Homogenized)	MC x 0.648 (void fraction)

* A PWR fuel assembly contains 0.4614 MTU; a BWR fuel assembly contains 0.1833 MTU (Roddy, 1986).

Figure 5-1. Geometric Model for Wall and Slab Dose Rates

The borehole contains only waste containers and may be surrounded by tuff rock, concrete, or steel. Figure 5-2 shows the problem geometry for the borehole. Basically, the source was assumed to be surrounded in the axial and radial directions by the shielding medium, intimate with the source container.

Problem geometries for the SSV and storage racks were homogenized into right parallelepipeds based upon their respective overall heights, lengths, and widths. The SSV geometric model is shown in Figure 5-3, the storage rack model in Figure 5-4. For SSV calculations, a top and side air gap is assumed to exist.

Source dimensions for each waste type considered in the gamma analysis are given in Tables 5-2 and 5-3.

5.4 Computer Program Input Assumptions

Source densities for each waste type are presented in Table 5-4 and were calculated on the basis of data from Roddy (1986).

Dose buildup factors selected for input to the code are representative of the shielding medium assumed. As an example, for concrete walls and slabs, concrete dose buildup factors internal to the computer program are specified. An exception to this are dose rates calculated at the interface between the source container and the shielding medium. For this case, the source material's buildup factors are specified because the scattering within the shielding material has not yet begun to dominate dose buildup at these locations.

Mass attenuation coefficients, with the exception of those for tuff rock, were taken from ANSI/ANS-6.4-1985 (ANS, 1985). The attenuation coefficients for tuff rock are presented in Subsection 2.4. Flux-to-dose rate conversion factors were also taken from ANSI/ANS-6.4-1985.

Figure 5-2. Geometric Model for the Borehole

Figure 5-3. Geometric Model for the Surface Storage Vault

- 1 Source region (fuel)
- 2 Container (steel)
- 3 Side wall (concrete)

Figure 5-4. Geometric Model for the Storage Rack

TABLE 5-2

GAMMA CODE GEOMETRIC SOURCE DIAMETERS

<u>Source Description</u>	<u>Source Diameter (in.)</u>
Single Canister of 2 Consolidated PWR Fuel Assemblies	8.4L x 8.4W
Single Canister of 2 Consolidated BWR Fuel Assemblies	5.47L x 5.47W
Single Container of 6 Consolidated PWR Fuel Assemblies	11.8 Internal Void 25.25 Outer Diameter
Single Container of 18 Consolidated BWR Fuel Assemblies	11.8 Internal Void 25.25 Outer Diameter
Single Container of 4 Intact PWR Fuel Assemblies	27.25
Single Container of 10 Intact BWR Fuel Assemblies	27.25
Hybrid Container of 3 Intact PWR Plus 4 Intact BWR Fuel Assemblies	27.25
Single Container of Hanford Site DHLW	23.25
Single Container of INEL DHLW	23.25
Single Container of SRP DHLW	23.25
Single Container of WVDP HLW	23.75
Surface Storage Vault (Homogenized)	240L x 210W
Storage Rack (Homogenized)	78L x 26W

TABLE 5-3

GAMMA CODE GEOMETRIC SOURCE HEIGHTS

<u>Source Description</u>	<u>Source Heights (in.)</u>
Single Canister of 2 Consolidated PWR Fuel Assemblies	144
Single Canister of 2 Consolidated BWR Fuel Assemblies	148
Single Container of 6 Consolidated PWR Fuel Assemblies	144
Single Container of 18 Consolidated BWR Fuel Assemblies	148
Single Container of 4 Intact PWR Fuel Assemblies	144
Single Container of 10 Intact BWR Fuel Assemblies	148
Hybrid Container of 3 Intact PWR Plus 4 Intact BWR Fuel Assemblies	148
Single Container of Hanford Site DHLW	90
Single Container of INEL DHLW	82
Single Container of SRP DHLW	85
Single Container of WVDP HLW	96
Surface Storage Vault (Homogenized)	148
Storage Rack (Homogenized)	144

TABLE 5-4

GAMMA CODE SOURCE DENSITIES

<u>Source Description</u>	<u>Source Density (g/cm³)</u>
Single Canister of 2 Consolidated PWR Fuel Assemblies	7.57
Single Canister of 2 Consolidated BWR Fuel Assemblies	8.58
Single Container of 6 Consolidated PWR Fuel Assemblies	4.00
Single Container of 18 Consolidated BWR Fuel Assemblies	5.76
Single Container of 4 Intact PWR Fuel Assemblies	1.83
Single Container of 10 Intact BWR Fuel Assemblies	2.20
Hybrid Container of 3 Intact PWR Plus 4 Intact BWR Fuel Assemblies	2.28
Single Container of Hanford Site DHLW	2.64
Single Container of INEL DHLW	3.20
Single Container of SRP DHLW	2.85
Single Container of WVDP HLW	2.70

5.5 Gamma Analysis Dose Rate Results

5.5.1 Single Canister of 2 Consolidated BWR/PWR Fuel Assembly Next to a Concrete Wall or Slab

Primary gamma dose rates for a single canister of 2 consolidated BWR or PWR fuel assemblies next to a concrete wall or slab are presented in Appendix B, Figures B-1 through B-18. Dose rates were calculated and are shown for each degree of fuel burnup and age provided by DOE (1987). This includes dose rate profiles in both shield walls and slabs.

The analysis shows that low age fuel results in higher dose rate profiles. Additionally, the greater the fuel burnup, the higher the dose rates in the shielding. For the same degree of fuel burnup and age, PWR spent fuel exhibits a higher dose rate profile in the concrete shielding than BWR spent fuel. Generally, the dose rates in concrete walls are greater than in concrete slabs, because the effect of self-shielding from the source is more predominant in the axial direction than in the radial direction.

Primary gamma dose rate ratios in concrete are presented in Appendix C, Tables C-1 through C-18, as a function of spent fuel age for the single canister of consolidated BWR and PWR assemblies. Both wall and slab dose rate ratios are tabulated for each degree of fuel burnup considered in this analysis. These ratios can be used to convert from one age to another for a specific fuel burnup.

5.5.2 Single Waste Container Next to a Concrete Wall or Slab

Dose rate profiles for a single waste container next to a concrete wall or slab are given in Appendix B, Figures B-19 through B-31. They include both intact and consolidated BWR and PWR spent fuel assembly combinations as well as HLW containers. For HLW containers, dose rates in concrete are provided as a function of waste age.

For the BWR spent fuel containers, the 10 intact BWR containers and the 18 consolidated BWR containers have similar dose rate profiles in concrete. This is due primarily to the effect of source geometry on the analysis. A BWR-10 container has a larger diameter than a BWR-18 container (28 in. versus 26 in.) and has no internal area reserved for fuel assembly hardware. Thus, the overall effect gives comparable dose rates in the concrete shielding.

The above analysis also holds true for the PWR spent fuel containers. The 4 intact PWR containers (PWR-4) and the 6 consolidated PWR containers (PWR-6) have similar dose rate profiles in concrete, for the same reasons as given above.

HLW container dose rates in concrete are shown in Appendix B, Figures B-24 through B-31, for Hanford, INEL, SRP, and West Valley sources. The highest dose rates result from Hanford Site DHLW sources, followed by SRP, West Valley, and finally, INEL DHLW. As with the single fuel element analysis, lower age sources result in higher dose rates. Concrete wall dose rates are generally slightly higher than concrete slab dose rates.

Primary gamma dose rate ratios in concrete for HLW containers are presented in Appendix C, Tables C-19 through C-26, as a function of decay age. Both wall and slab doses are tabulated from 6 in. to 7'-0" concrete thickness.

5.5.3 Single Waste Container in a Borehole

For the borehole analysis, primary gamma dose rates were calculated in concrete, tuff rock, and steel for a single container of 18 consolidated BWR fuel assemblies (BWR-18) and a single container of 6 consolidated PWR fuel assemblies (PWR-6). Maximum-burnup, minimum-age source strengths were used. Dose rates were calculated on the top (axial direction) and sides (radial direction) of the container. The resulting dose rate profiles in each shielding medium are presented in Appendix B, Figures B-32 through B-37.

Resulting dose rate profiles in tuff rock for the BWR-18 and PWR-6 are slightly higher than the concrete profiles. This difference is due primarily to the lower tuff dry density of 138 lbm/ft³, as compared with the concrete density of 140 lbm/ft³. A single container of PWR-6 will have higher dose rates than a BWR-18 container in all three shielding media.

5.5.4 Rack of Multiple Containers

In the storage rack analysis, only one source type was considered, the PWR-6 container at 60 GWD/MTU burnup and 5-yr decay. This case gives the highest gamma concrete dose rates of all the source types considered in this study. If the concrete dose rates from multiple containers are divided by the dose rates from a single container with the same source strength, the ratio should be relatively geometry-dependent, source-independent. These ratios can then be multiplied by the dose rates from any single source-type dose rate versus depth profile in concrete to obtain new multiple-source dose rates.

Figure 5-5 shows the results of the storage rack analysis from 2 to 6 ft of concrete for both walls and slabs. The ratio of multiple- to single-source gamma dose rates for both curves in Figures 5-5 increases with distance into the concrete shield. This fact is due primarily to the effect of source geometry. The multiple container dimensions are much larger than the single container dimensions. As the distance into the shield increases, the contribution from the sides of the multiple container sources becomes more significant.

5.5.5 Waste Containers in the Surface Storage Vault

Primary gamma dose rate graphs for waste containers of spent fuel or HLW in the concrete SSV are provided in Appendix B, Figures B-38 through B-41. The spent fuel calculations were performed using maximum fuel burnup and minimum decay age source strengths. The HLW calculations used 5-yr-old sources.

Figure 5-5. Rack of Multiple Containers - Dose Rate Ratios

For spent fuel, the PWR-6 case resulted in the highest dose rates, followed by PWR-4, the hybrid container, BWR-18, and finally, the BWR-10 waste container. For HLW, the Hanford Site DHLW container had the highest dose rate profile, followed by SRP, West Valley, and INEL DHLW. In both spent fuel and HLW calculations, wall and slab dose rates are virtually identical for this geometry.

5.5.6 Wall and Slab Surface Dose Rates

In Subsection 5.4, dose buildup factors for the interface between the waste containers and the concrete shielding were specified as representing the source material buildup and not the buildup associated with the shielding material. For spent fuel calculations, a uranium dose buildup factor was specified; for HLW calculations, a concrete dose buildup factor was specified based on the effective-Z of the waste (approximately 12.3, which is close to the effective-Z of ordinary concrete, 11.5).

Table 5-5 lists the primary gamma wall and slab surface dose rates for each waste type, fuel burnup, and age considered in the gamma analysis.

TABLE 5-5

WALL AND SLAB SURFACE GAMMA DOSE RATES

<u>Source Description</u>	<u>Burnup (GWd/MTU)</u>	<u>Age (yr)</u>	<u>Wall Dose Rate (rem/hr)</u>	<u>Slab Dose Rate (rem/hr)</u>
Single Canister of 2 Consolidated PWR Assemblies	10	5	1.02E+04	1.03E+04
	10	10	6.19E+03	6.27E+03
	10	20	4.02E+03	4.08E+03
	10	30	3.01E+03	3.06E+03
	25	5	3.06E+04	3.11E+04
	25	10	1.69E+04	1.71E+04
	25	20	1.02E+04	1.04E+04
	25	30	7.53E+03	7.64E+03
	33	5	4.54E+04	4.60E+04
	33	10	2.36E+04	2.39E+04
	33	20	1.38E+04	1.40E+04
	33	30	1.00E+04	1.02E+04
	45	5	6.18E+04	6.27E+04
	45	10	3.17E+04	3.21E+04
	45	20	1.84E+04	1.87E+04
	45	30	1.34E+04	1.36E+04
	60	5	9.03E+04	9.16E+04
	60	10	4.41E+04	4.47E+04
	60	20	2.47E+04	2.51E+04
	60	30	1.78E+04	1.81E+04
Single Canister of 2 Consolidated BWR Assemblies	10	5	6.87E+03	6.96E+03
	10	10	4.26E+03	4.32E+03
	10	20	2.91E+03	2.95E+03
	10	30	2.23E+03	2.26E+03
	20	5	1.65E+04	1.68E+04
	20	10	9.30E+03	9.43E+03
	20	20	5.96E+03	6.04E+03
	20	30	4.49E+03	4.55E+03
	27.5	5	2.50E+04	2.53E+04
	27.5	10	1.34E+04	1.36E+04
	27.5	20	8.30E+03	8.31E+03
	27.5	30	6.17E+03	6.26E+03
	40	5	3.77E+04	3.81E+04
	40	10	1.98E+04	2.00E+04
	40	20	1.20E+04	1.21E+04
	40	30	8.86E+03	8.98E+03

TABLE 5-5

WALL AND SLAB SURFACE GAMMA DOSE RATES (Cont'd)

<u>Source Description</u>	<u>Burnup (Gwd/MTU)</u>	<u>Age (yr)</u>	<u>Wall Dose Rate (rem/hr)</u>	<u>Slab Dose Rate (rem/hr)</u>
4 Intact PWR Assemblies	60	5	3.21E+04	3.32E+04
6 Consolidated PWR Assemblies	60	5	3.22E+04	3.33E+04
3 Intact PWR Plus 4 Intact BWR Assemblies	60/40	5/5	2.48E+04	2.57E+04
10 Intact BWR Assemblies	40	5	1.33E+04	1.38E+04
18 Consolidated BWR Assemblies	40	5	1.33E+04	1.37E+04
Hanford Site DHLW	--	5	1.87E+04	1.96E+04
	--	10	1.65E+04	1.72E+04
	--	20	1.30E+04	1.36E+04
	--	30	1.03E+04	1.08E+04
INEL DHLW	--	5	3.67E+03	3.89E+03
	--	10	2.93E+03	3.10E+03
	--	20	2.25E+03	2.38E+03
	--	30	1.78E+03	1.88E+03
SRP DHLW	---	5	9.43E+03	9.88E+03
	---	10	8.26E+03	8.65E+03
	---	20	6.47E+03	6.78E+03
	---	30	5.11E+03	5.35E+03
WVDP HLW	--	5	9.59E+03	1.00E+04
	--	10	8.50E+03	8.88E+03
	--	20	6.70E+03	7.00E+03
	--	30	5.29E+03	5.53E+03

6.0 NEUTRON DOSE RATE ANALYSIS

The contribution of neutrons and associated capture gammas to the total dose rate for selected cases is presented in this section.

The following waste types and locations were considered in the neutron dose rate analysis:

- o Single container of 18 consolidated BWR fuel assemblies (BWR-18) in a borehole,
- o Single container of 6 consolidated PWR fuel assemblies (PWR-6) in a borehole,
- o Containers of 6 consolidated PWR fuel assemblies (PWR-6) in the surface storage vault (SSV), and
- o Single container of 6 consolidated PWR fuel assemblies (PWR-6) in a transfer cask with varying combinations of steel, lead, and borated polyethylene, as well as varying combinations of steel, lead, and water.

6.1 Neutron Source Strengths

Neutron source strengths of the waste types described above were obtained from the data base in DOE (1987). These source strengths are presented in Appendix A, Tables A-15 through A-23, in a form compatible with the ANISN computer code (DLC-23/CASK format). Maximum fuel burnup and minimum age source strengths were used in this analysis. Because the data base does not contain HLW neutron source strengths, this study excludes HLW containers in the neutron dose rate analysis.

6.2 Neutron Cross Section Data

For a multigroup computer code like ANISN, coupled neutron and gamma ray cross section libraries such as DLC-23/CASK are included

in the code input to define a specific shielding material. For the neutron analysis, the following materials were defined by the CASK library as input in the ANISN program:

- o Ordinary concrete of density 140 lbm/ft³,
- o Tuff rock of density 138 lbm/ft³,
- o Water of density 62.4 lbm/ft³,
- o Lead of density 710 lbm/ft³,
- o Borated polyethylene of density 72 lbm/ft³,
- o Carbon steel (SA-516),
- o Stainless steel (Type 304),
- o BWR-18 fuel,
- o PWR-6 fuel, and
- o Dry air at standard temperature and pressure.

6.3 Computer Program Assumptions

All neutron and capture gamma calculations were performed with the ANISN computer program using DLC-23/CASK (P_3) cross section library data (22 neutron, 18 gamma ray groups) as input to define material regions.

The neutron calculations do not account for induced fissions. However, ANISN results indicate that the induced fission densities would be well below the neutron source strengths used in the calculations; therefore, they would not add significantly to the total number of neutrons emitted from the source.

6.4 Neutron Dose Rate Analysis Results for the Borehole and Surface Storage Vault

The problem geometry for the borehole analysis is the same as the geometry selected for the gamma analysis. The shielding medium (concrete, tuff, or steel) was assumed to be intimate with the waste containers in both radial and axial directions.

The resulting dose rate profiles for the borehole analysis is presented in Appendix D, Figures D-1 through D-12. The SSV profile is shown in Figure D-13. Data are presented as neutron dose rates and total dose rates. The total dose rate is the sum of the primary gamma, capture gamma, and neutron dose rates.

In the borehole analysis, concrete is more effective in attenuating neutrons than tuff rock or steel. Dry tuff rock and steel are not good neutron shielding materials, as evidenced by the closeness of the neutron dose profile to the total dose rate curve in the figures. Accounting for the small water content of tuff would increase the neutron attenuation; therefore, the results for dry tuff are conservative.

Capture gammas are produced in the shielding materials and are attenuated by the density effect of the medium in which they are produced. More capture gammas are produced in the tuff rock than in steel, primarily because of the abundance of more elements with a lower atomic number. Additionally, those capture gammas produced in steel are more easily attenuated as a result of the higher steel density. For these reasons, the neutron dose rate curve for steel dominates the total dose rate profile at approximately 12 in. into the shielding. The tuff profiles do not converge because capture gammas are produced throughout the shielding material and are not as effectively attenuated as they are in steel because of the lower bulk density of the tuff.

6.5 Effects of Fuel Burnup and Age

Tables 6-1 and 6-2 show the variation of total neutron source strength as a function of burnup and age for BWR and PWR spent fuel.

TABLE 6-1

TOTAL NEUTRON SOURCE STRENGTHS FOR BWR SPENT FUEL^a

Burnup (Gwd/MTU)	Age ^b			
	<u>5 Yr</u>	<u>10 Yr</u>	<u>20 Yr</u>	<u>30 Yr</u>
10	2.175E+06	2.147E+06	2.095E+06	2.039E+06
20	3.538E+07	3.021E+07	2.244E+07	1.707E+07
27.5	1.554E+08	1.299E+08	9.151E+07	6.524E+07
40	4.585E+08	3.811E+08	2.647E+08	1.852E+08

a. Source strengths were obtained from the data base in DOE (1987).

b. Source strength units are in neutrons/sec/MTU.

TABLE 6-2

TOTAL NEUTRON SOURCE STRENGTHS FOR PWR SPENT FUEL^a

Burnup (Gwd/MTU)	Age ^b			
	5 Yr	10 Yr	20 Yr	30 Yr
10	1.563E+06	1.599E+06	1.639E+06	1.649E+06
25	6.164E+07	5.212E+07	3.780E+07	2.794E+07
33	2.297E+08	1.915E+08	1.341E+08	9.490E+07
45	5.262E+08	4.373E+08	3.037E+08	2.123E+08
60	1.889E+09	1.565E+09	1.080E+09	7.492E+08

a. Source strengths were obtained from the data base in DOE (1987).

b. Source strength units are in neutrons/sec/MTU.

In general, the total neutron source strength increases with the degree of fuel burnup and decreases with age. An exception to this is the PWR fuel burnup source strengths at 10 GWd/MTU. At this burnup, there is a slight increase of source strength values with increasing age. For PWR fuel and this burnup, the (alpha, n) reactions generate far more neutrons than are generated from spontaneous fission, and the neutron population produced by the (alpha, n) reaction increases with age. At higher fuel burnup values, the spontaneous fission component dominates the (alpha, n) reaction and decreases with age.

The highest total neutron source strengths are from maximum-burnup, minimum-age spent fuel and should result in the highest neutron dose rates. A neutron sensitivity analysis was performed to confirm this and to demonstrate the significance of neutrons plus capture gammas on total dose rates for a specific problem geometry. The specific case selected was a single container of 6 consolidated PWR fuel assemblies (PWR-6) in a concrete borehole. Only radial (container side) doses were considered. Calculations were performed for various degrees of fuel burnup and age.

Results of the ANISN/CYLSEC calculations are shown in Appendix D, Figures D-14 through D-18. Primary gamma, neutron, and total dose rate values are plotted. The total dose rates also include the contribution due to capture gammas. To demonstrate the significance of neutrons and their associated capture gammas on the total dose rate, Figures D-19 through D-22 are provided. The percentage contribution of neutrons plus capture gammas to the total dose rate as a function of concrete thickness is plotted for each burnup/age case considered.

The effect of neutrons and capture gammas is generally not significant until their contribution to the total dose rate becomes about 10%. From the figures, low-burnup, low-age fuel does not have a significant neutron population to affect the total dose rate. PWR-6 fuel below 33 GWd/MTU burnup and at 5-yr decay

or less should have no significant contributions from neutrons and capture gammas. For these situations, a primary gamma analysis will give the required concrete shielding thicknesses. As the fuel ages, there is an increase in the contribution due to neutrons plus capture gammas to the total dose rate, and this should be considered in any formal shielding analyses.

As the degree of fuel burnup increases, the percentage of neutron and capture gamma rays in the total dose rate increases. In PWR-6 fuel at or above 45 Gwd/MTU burnup, their contribution should be considered, regardless of the fuel's age. This is illustrated in Figure 6-1, which shows the variation of concrete shielding with respect to burnup and age for the PWR-6 spent fuel cases considered in this sensitivity analysis.

6.6 Neutron and Gamma Dose Rate Analysis Results for the Transfer Cask

The transfer cask problem geometry is illustrated in Figures 6-2 and 6-3. In general, shielding materials are sandwiched between half-inch layers of stainless steel. A material with a high density is needed to attenuate gammas, and a material with a low atomic number is needed to attenuate neutrons. For this analysis, lead was selected for the high-density material. Either borated polyethylene (Poly) or water was selected to provide neutron shielding. Primary gamma, capture gamma, and neutron dose rates were calculated with the ANISN computer program.

Results of the transfer cask analysis are given in Tables 6-3 through 6-6. Surface dose rates were calculated as a function of thickness of lead and low-atomic-number material for both neutrons and total gammas. The higher axial dose rates for a given set of shielding thicknesses are due primarily to ANISN one-dimensional modeling assumptions, as described in Subsection 3.2.2.

Figure 6-1. Contribution of Gammas and Neutrons for Concrete Shielding

Figure 6-2. Geometric Model of Transfer Cask - Radial Direction

Figure 6-3. Geometric Model of Transfer Cask - Axial Direction

TABLE 6-3

TRANSFER CASK RADIAL SURFACE DOSE RATES USING LEAD AND POLY

Shielding Thickness (in.)		Cask Surface Dose Rates (mrem/hr)		
Lead	Poly	Neutron	Gamma	Total
6.00	4.00	68.3	39.7	108.0
6.50	4.00	61.6	23.7	85.4*
6.50	5.00	29.5	19.5	49.0
7.00	5.50	18.5	12.0	30.5
7.00	6.00	13.4	10.8	24.2
7.25	6.00	12.5	9.4	21.9
7.25	6.25	10.7	8.9	19.6
7.00	6.50	9.7	9.9	19.6

* Dose rate 1 m from cask surface is 25.6 mrem/hr.

TABLE 6-4

TRANSFER CASK AXIAL SURFACE DOSE RATES USING LEAD AND POLY

Shielding Thickness (in.)		Cask Surface Dose Rates (mrem/hr)		
<u>Lead</u>	<u>Poly</u>	<u>Neutron</u>	<u>Gamma</u>	<u>Total</u>
6.50	4.00	79.2	29.3	108.5
6.50	4.50	55.6	26.7	82.3
6.50	5.00	39.5	24.4	63.9
7.00	5.50	25.2	14.1	39.3
7.00	6.50	13.6	11.7	25.3

TABLE 6-5

TRANSFER CASK RADIAL SURFACE DOSE RATES USING LEAD AND WATER

<u>Shielding Thickness (in.)</u>		<u>Cask Surface Dose Rates (mrem/hr)</u>		
<u>Lead</u>	<u>Water</u>	<u>Neutron</u>	<u>Gamma</u>	<u>Total</u>
6.50	4.00	86.8	58.6	145.4
7.00	5.50	28.5	37.4	65.9
7.00	6.00	21.0	34.2	55.2
7.00	6.50	15.6	31.4	47.0
7.50	6.50	13.8	28.1	41.9
9.00	7.00	7.1	21.7	28.8
11.00	7.00	4.4	18.8	23.2
13.00	7.00	2.8	16.5	19.3

TABLE 6-6

TRANSFER CASK AXIAL SURFACE DOSE RATES USING LEAD AND WATER

Shielding Thickness (in.)		Cask Surface Dose Rates (mrem/hr)		
<u>Lead</u>	<u>Water</u>	<u>Neutron</u>	<u>Gamma</u>	<u>Total</u>
6.50	4.00	111.5	64.3	175.8
7.00	5.50	38.3	38.4	76.7
7.00	6.50	21.5	32.1	53.6
13.00	7.00	3.7	12.2	15.9

In one specific case (6-1/2 in. of lead with 4 in. Poly), the neutron, total gamma, and total dose rates were plotted through the transfer cask in the radial direction. These dose rate profiles are shown in Figure 6-4. Air-attenuated dose rates from the cask surface for this case are shown in Figure 6-5.

Gamma dose rates are greatly attenuated by the denser lead material in the cask, but not by the Poly. In fact, the generation of capture gammas in the Poly produces a slight increase in total gamma dose rates in the vicinity of the lead-Poly interface. Neutrons are not attenuated well in either lead or stainless steel. However, low-atomic-number materials such as Poly effectively reduce the neutron dose rates, as evidenced by the change in slope of the neutron curve in Figure 6-4 as the Poly material is penetrated. At a distance of 1 m from the cask surface, the total dose rate is reduced about a factor of 3 owing to geometry and air attenuation.

Figure 6-4. Transfer Cask Dose Rate Profiles

Figure 6-5. Transfer Cask Air-Attenuated Dose Rates

7.0 CONCLUSIONS

7.1 Summary

In both the gamma and neutron analysis, the highest dose rates result from high-burnup, low-age spent fuel. For HLW, the highest primary gamma dose rates come from Hanford Site DHLW with low age. PWR spent fuel produces a higher neutron and gamma dose rate than BWR spent fuel for the same degree of burnup and age. The highest gamma and neutron dose rates were from a single container of 6 consolidated PWR fuel assemblies at 60 GWd/MTU burnup and 5-yr decay. Generally, the neutron and gamma dose rates are expected to be slightly greater in the radial direction than in the axial.

Concrete is a relatively good material for shielding against both gammas and neutrons. Tuff, steel, and lead are good for attenuating gammas but poor against neutrons. Accounting for the small water content of tuff rock would increase the neutron attenuation; therefore, the results for dry tuff are conservative. Borated polyethylene captures neutrons effectively, but is a poor attenuator of gammas.

Neutron and capture gamma dose rates can be significant for (1) high-burnup spent fuel and concrete shielding and (2) older spent fuel. A neutron analysis should always be considered when using tuff, steel, or lead shielding materials.

7.2 Further Considerations and Recommendations

Conservative values of radiation shielding for the repository would result by using the waste source strengths with the highest gamma and neutron dose rates. If spent fuel consolidation were incorporated into the facility design, the highest source strengths would be containers of 6 consolidated PWR fuel assemblies at maximum burnup and minimum age. For this study, the

highest PWR-6 dose rates are for a 60 GWd/MTU burnup and a 5-yr age. If consolidation were not considered, the highest source strengths would be containers of 4 intact PWR fuel assemblies at the same burnup and decay age. There may be situations where only HLW containers are to be considered. If so, Hanford Site DHLW at 5-yr decay age should be used in the shielding analysis.

Except in special situations, concrete should be the primary shielding material for the repository structures. It is a relatively good attenuator of both gammas and neutrons.

Further shielding analyses for the repository should consider the effects of neutrons and capture gammas on the total dose rate.

Because of the presence of neutrons in the various waste forms, other special analyses may need to be performed. An example of this is a neutron activation analysis for facility areas, such as the surface storage vault, where large amounts of spent fuel may reside for a significant time period. Additional analyses may need to examine neutron shielding materials other than concrete, borated polyethylene, or water. Equipment qualification doses may need to take into account neutron as well as gamma ray doses. Radiation detection instrumentation requirements for the repository may have to take into account the effect of neutrons. Special analyses to evaluate penetration shielding incorporating neutron scattering may have to be considered.

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APPENDIX A

SOURCE STRENGTH TABLES

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TABLE A-1

GAMMA SOURCE STRENGTH OF BWR SPENT FUEL
AT 10 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	1.082E+13	6.504E+12	4.987E+12	3.933E+12
2.50E-02	6.518E+12	3.583E+12	2.568E+12	2.004E+12
3.75E-02	9.634E+12	5.719E+12	4.275E+12	3.331E+12
5.75E-02	1.224E+13	7.331E+12	5.825E+12	4.747E+12
8.50E-02	1.165E+13	6.388E+12	4.777E+12	3.718E+12
1.25E-01	1.548E+13	6.883E+12	4.881E+12	3.678E+12
2.25E-01	2.606E+13	1.402E+13	1.059E+13	8.271E+12
3.75E-01	2.672E+13	1.188E+13	7.639E+12	5.873E+12
5.75E-01	7.389E+14	5.541E+14	4.230E+14	3.348E+14
8.50E-01	1.098E+14	2.733E+13	6.388E+12	3.539E+12
1.25E+00	8.244E+13	3.924E+13	1.298E+13	5.054E+12
1.75E+00	2.670E+12	7.002E+11	3.533E+11	2.065E+11
2.25E+00	4.725E+12	6.602E+10	1.075E+08	3.335E+07
2.75E+00	9.845E+10	3.121E+09	6.047E+07	5.261E+07
3.50E+00	1.560E+10	5.019E+08	1.033E+06	4.631E+05
5.00E+00	3.943E+05	3.608E+05	3.120E+05	2.784E+05
7.00E+00	6.313E+04	5.762E+04	4.962E+04	4.411E+04
9.50E+00	9.804E+03	8.940E+03	7.680E+03	6.815E+03

TABLE A-2

GAMMA SOURCE STRENGTH OF BWR SPENT FUEL
AT 20 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	1.941E+13	1.200E+13	9.203E+12	7.274E+12
2.50E-02	1.176E+13	6.578E+12	4.690E+12	3.663E+12
3.75E-02	1.794E+13	1.121E+13	8.228E+12	6.349E+12
5.75E-02	2.222E+13	1.385E+13	1.118E+12	9.304E+12
8.50E-02	2.098E+13	1.186E+13	8.721E+12	6.752E+12
1.25E-01	2.940E+13	1.478E+13	9.829E+12	7.089E+12
2.25E-01	4.727E+13	2.612E+13	1.942E+13	1.505E+13
3.75E-01	4.856E+13	2.183E+13	1.381E+13	1.057E+13
5.75E-01	1.601E+15	1.122E+15	8.366E+14	6.618E+14
8.50E-01	3.818E+14	9.580E+13	2.016E+13	9.809E+12
1.25E+00	2.019E+14	1.039E+14	3.658E+13	1.484E+13
1.75E+00	6.893E+12	2.447E+12	1.159E+12	6.046E+11
2.25E+00	6.606E+12	1.008E+11	2.103E+08	6.692E+07
2.75E+00	2.102E+11	6.897E+09	2.918E+08	2.609E+08
3.50E+00	3.394E+10	1.100E+09	8.372E+06	5.243E+06
5.00E+00	7.425E+06	6.220E+06	4.420E+06	3.192E+06
7.00E+00	1.197E+06	1.002E+06	7.112E+05	5.129E+05
9.50E+00	1.866E+05	1.562E+05	1.108E+05	7.978E+04

TABLE A-3

GAMMA SOURCE STRENGTH OF BWR SPENT FUEL
AT 27.5 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	2.481E+13	1.563E+13	1.197E+13	9.462E+12
2.50E-02	1.500E+13	8.500E+12	6.045E+12	4.718E+12
3.75E-02	2.371E+13	1.523E+13	1.103E+13	8.445E+12
5.75E-02	2.849E+13	1.812E+13	1.463E+13	1.220E+13
8.50E-02	2.706E+13	1.561E+13	1.131E+13	8.704E+12
1.25E-01	3.988E+13	2.139E+13	1.364E+13	9.549E+12
2.25E-01	6.064E+13	3.422E+13	2.518E+13	1.941E+13
3.75E-01	6.188E+13	2.818E+13	1.769E+13	1.352E+13
5.75E-01	2.287E+15	1.547E+15	1.139E+15	8.993E+14
8.50E-01	6.564E+14	1.664E+14	3.447E+13	1.603E+13
1.25E+00	3.304E+14	1.661E+14	6.015E+13	2.480E+13
1.75E+00	1.070E+13	4.324E+13	2.013E+12	1.010E+12
2.25E+00	7.110E+12	1.149E+11	3.022E+08	1.057E+08
2.75E+00	2.816E+11	9.504E+09	6.284E+08	5.638E+08
3.50E+00	4.578E+10	1.514E+09	3.288E+07	2.189E+07
5.00E+00	3.330E+07	2.764E+07	1.915E+07	1.336E+07
7.00E+00	5.374E+06	4.460E+06	3.089E+06	2.153E+06
9.50E+00	8.378E+05	6.952E+05	4.814E+05	3.354E+05

TABLE A-4

GAMMA SOURCE STRENGTH OF BWR SPENT FUEL
AT 40 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	3.321E+13	2.220E+13	1.701E+13	1.343E+13
2.50E-02	1.956E+13	1.183E+13	8.510E+12	6.648E+12
3.75E-02	3.246E+13	2.193E+13	1.578E+13	1.203E+13
5.75E-02	3.794E+13	2.555E+13	2.046E+13	1.695E+13
8.50E-02	3.631E+13	2.232E+13	1.605E+13	1.232E+13
1.25E-01	5.555E+13	3.244E+13	2.025E+13	1.391E+13
2.25E-01	8.012E+13	4.880E+13	3.587E+13	2.752E+13
3.75E-01	7.793E+13	3.855E+13	2.490E+13	1.906E+13
5.75E-01	3.345E+15	2.222E+15	1.621E+15	1.279E+15
8.50E-01	1.084E+15	2.763E+14	5.680E+13	2.590E+13
1.25E+00	4.893E+14	2.504E+14	9.340E+13	3.938E+13
1.75E+00	1.558E+13	7.186E+12	3.339E+12	1.647E+12
2.25E+00	7.277E+12	1.221E+11	4.356E+08	1.806E+08
2.75E+00	3.251E+11	1.170E+10	1.441E+09	1.296E+09
3.50E+00	5.306E+10	1.833E+09	9.398E+07	6.377E+07
5.00E+00	9.880E+07	8.180E+07	5.635E+07	3.895E+07
7.00E+00	1.595E+07	1.321E+07	9.093E+06	6.284E+06
9.50E+00	2.486E+06	2.059E+06	1.417E+06	9.795E+05

TABLE A-5

GAMMA SOURCE STRENGTH OF PWR SPENT FUEL
AT 10 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	1.178E+13	6.686E+12	5.110E+12	4.025E+12
2.50E-02	6.968E+12	3.665E+12	2.633E+12	2.055E+12
3.75E-02	1.039E+13	5.801E+12	4.343E+12	3.388E+12
5.75E-02	1.332E+13	7.492E+12	5.911E+12	4.790E+12
8.50E-02	1.279E+13	6.560E+12	4.905E+12	3.819E+12
1.25E-01	1.743E+13	6.929E+12	4.946E+12	3.749E+12
2.25E-01	2.855E+13	1.438E+13	1.087E+13	8.501E+12
3.75E-01	2.840E+13	1.205E+13	7.838E+12	6.045E+12
5.75E-01	7.452E+14	5.575E+14	4.261E+14	3.374E+14
8.50E-01	1.078E+14	2.610E+13	6.015E+12	3.414E+12
1.25E+00	1.573E+14	7.699E+13	2.284E+13	7.581E+12
1.75E+00	2.833E+12	6.414E+11	3.264E+11	1.958E+11
2.25E+00	6.152E+12	8.399E+10	2.126E+08	6.075E+07
2.75E+00	1.089E+11	3.438E+09	7.175E+07	6.256E+07
3.50E+00	1.709E+10	5.495E+08	9.468E+05	3.542E+05
5.00E+00	2.694E+05	2.531E+05	2.291E+05	2.123E+05
7.00E+00	4.302E+04	4.031E+04	3.631E+04	3.354E+04
9.50E+00	6.676E+03	6.245E+03	5.612E+03	5.175E+03

TABLE A-6

GAMMA SOURCE STRENGTH OF PWR SPENT FUEL
AT 25 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	2.425E+13	1.501E+13	1.150E+13	9.082E+12
2.50E-02	1.445E+13	8.158E+12	5.848E+12	4.568E+12
3.75E-02	2.242E+13	1.407E+13	1.031E+13	7.943E+12
5.75E-02	2.773E+13	1.728E+13	1.389E+13	1.151E+13
8.50E-02	2.620E+13	1.485E+13	1.091E+13	8.433E+12
1.25E-01	3.725E+13	1.889E+13	1.246E+13	8.933E+12
2.25E-01	5.888E+13	3.269E+13	2.430E+13	1.881E+13
3.75E-01	5.876E+13	2.678E+13	1.720E+13	1.320E+13
5.75E-01	2.046E+15	1.410E+15	1.046E+15	8.263E+14
8.50E-01	5.336E+14	1.327E+14	2.703E+13	1.297E+13
1.25E+00	4.489E+14	2.254E+14	7.210E+13	2.604E+13
1.75E+00	8.908E+12	3.280E+12	1.548E+12	7.996E+11
2.25E+00	8.154E+12	1.253E+11	4.952E+08	1.479E+08
2.75E+00	2.600E+11	8.701E+09	5.506E+08	4.964E+08
3.50E+00	4.200E+10	1.365E+09	1.391E+07	8.904E+06
5.00E+00	1.305E+07	1.088E+07	7.640E+06	5.425E+06
7.00E+00	2.104E+06	1.754E+06	1.231E+06	8.729E+05
9.50E+00	3.279E+05	2.734E+05	1.917E+05	1.359E+05

TABLE A-7

GAMMA SOURCE STRENGTH OF PWR SPENT FUEL
AT 33 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	3.152E+13	1.908E+13	1.455E+13	1.148E+13
2.50E-02	1.881E+13	1.033E+13	7.340E+12	5.728E+12
3.75E-02	2.990E+13	1.858E+13	1.340E+13	1.025E+13
5.75E-02	3.616E+13	2.198E+13	1.758E+13	1.458E+13
8.50E-02	3.460E+13	1.907E+13	1.376E+13	1.058E+13
1.25E-01	5.165E+13	2.648E+13	1.676E+13	1.169E+13
2.25E-01	7.756E+13	4.183E+13	3.069E+13	2.363E+13
3.75E-01	7.740E+13	3.409E+13	2.151E+13	1.644E+13
5.75E-01	2.885E+15	1.890E+15	1.377E+15	1.087E+15
8.50E-01	9.104E+14	2.250E+14	4.395E+13	2.016E+13
1.25E+00	6.568E+14	3.303E+14	1.079E+14	3.984E+13
1.75E+00	1.405E+13	5.478E+12	2.538E+12	1.268E+12
2.25E+00	1.020E+13	1.636E+11	6.984E+08	2.189E+08
2.75E+00	3.845E+11	1.316E+10	1.075E+09	9.697E+08
3.50E+00	6.241E+10	2.070E+09	4.837E+07	3.218E+07
5.00E+00	4.935E+07	4.093E+07	2.828E+07	1.965E+07
7.00E+00	7.966E+06	6.605E+06	4.562E+06	3.167E+06
9.50E+00	1.242E+06	1.030E+06	7.110E+05	4.936E+05

TABLE A-8

GAMMA SOURCE STRENGTH OF PWR SPENT FUEL
AT 45 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	3.905E+13	2.552E+13	1.950E+13	1.540E+13
2.50E-02	2.283E+13	1.357E+13	9.765E+12	7.628E+12
3.75E-02	3.788E+13	2.511E+13	1.804E+13	1.374E+13
5.75E-02	4.467E+13	2.941E+13	2.352E+13	1.949E+13
8.50E-02	4.296E+13	2.576E+13	1.845E+13	1.414E+13
1.25E-01	6.556E+13	3.726E+13	2.321E+13	1.595E+13
2.25E-01	9.430E+13	5.605E+13	4.115E+13	3.159E+13
3.75E-01	9.068E+13	4.406E+13	2.858E+13	2.189E+13
5.75E-01	3.939E+15	2.549E+15	1.843E+15	1.454E+15
8.50E-01	1.366E+15	3.393E+14	6.576E+13	2.971E+13
1.25E+00	8.031E+14	4.075E+14	1.386E+14	5.354E+13
1.75E+00	1.824E+13	8.209E+12	3.812E+12	1.881E+12
2.25E+00	9.457E+12	1.568E+11	8.105E+08	2.889E+08
2.75E+00	3.979E+11	1.461E+10	2.124E+09	1.922E+09
3.50E+00	6.475E+10	2.228E+09	1.079E+08	7.312E+07
5.00E+00	1.134E+08	9.390E+07	6.460E+07	4.465E+07
7.00E+00	1.830E+07	1.516E+07	1.043E+07	7.203E+06
9.50E+00	2.853E+06	2.363E+06	1.625E+06	1.123E+06

TABLE A-9

GAMMA SOURCE STRENGTH OF PWR SPENT FUEL
AT 60 GWD/MTU BURNUP

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec MTU)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	4.956E+13	3.212E+13	2.441E+13	1.923E+13
2.50E-02	2.865E+13	1.678E+13	1.197E+13	9.338E+12
3.75E-02	4.950E+13	3.281E+13	2.320E+13	1.754E+13
5.75E-02	5.615E+13	3.639E+13	2.878E+13	2.372E+13
8.50E-02	5.522E+13	3.261E+13	2.284E+13	1.738E+13
1.25E-01	8.845E+13	5.108E+13	3.068E+13	2.050E+13
2.25E-01	1.195E+14	7.011E+13	5.083E+13	3.879E+13
3.75E-01	1.143E+14	5.449E+13	3.480E+13	2.655E+13
5.75E-01	5.530E+15	3.418E+15	2.426E+15	1.911E+15
8.50E-01	2.186E+15	5.375E+14	9.996E+13	4.376E+13
1.25E+00	1.160E+15	5.880E+14	2.025E+14	7.901E+13
1.75E+00	2.706E+13	1.261E+13	5.793E+12	2.802E+12
2.25E+00	1.071E+13	1.894E+11	1.362E+09	5.801E+08
2.75E+00	5.478E+11	2.135E+10	4.087E+09	3.669E+09
3.50E+00	8.988E+10	3.419E+09	3.832E+08	2.626E+08
5.00E+00	4.087E+08	3.382E+08	2.352E+08	1.604E+08
7.00E+00	6.599E+07	5.406E+07	3.753E+07	2.590E+07
9.50E+00	1.029E+07	8.514E+06	5.852E+06	4.038E+06

TABLE A-10

GAMMA SOURCE STRENGTH FOR A HYBRID CONTAINER

<u>Mean Gamma Energy (MeV)</u>	<u>Hybrid Gamma Source Strength (MeV/Sec)</u>
1.00E-2	9.295E+13
2.50E-2	5.400E+13
3.75E-2	9.232E+13
5.75E-2	1.055E+14
8.50E-2	1.031E+14
1.25E-1	1.631E+14
2.25E-1	2.241E+14
3.75E-1	2.153E+14
5.75E-1	1.011E+16
8.50E-1	3.821E+15
1.25E+0	1.965E+15
1.75E+0	4.888E+13
2.25E+0	2.016E+13
2.75E+0	9.967E+11
3.50E+0	1.633E+11
5.00E+0	6.381E+08
7.00E+0	1.030E+08
9.50E+0	1.606E+07

NOTE:

1. The hybrid source consists of a single container of 4 intact BWR fuel assemblies (40 Gwd/MTU burnup, 5 yr decay) plus 3 intact PWR fuel assemblies (60 Gwd/MTU burnup, 5 yr decay).

TABLE A-11

GAMMA SOURCE STRENGTH FOR HANFORD SITE DEFENSE
HIGH LEVEL WASTE

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec-canister)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	1.965E+13	1.724E+13	1.357E+13	1.072E+13
2.50E-02	1.052E+13	8.985E+12	6.990E+12	5.508E+12
3.75E-02	1.681E+13	1.468E+13	1.152E+13	9.086E+12
5.75E-02	2.220E+13	1.960E+13	1.558E+13	1.243E+13
8.50E-02	1.904E+13	1.663E+13	1.302E+13	1.024E+13
1.25E-01	1.854E+13	1.605E+13	1.245E+13	9.750E+12
2.25E-01	4.217E+13	3.690E+13	2.896E+13	2.282E+13
3.75E-01	3.333E+13	2.722E+13	2.073E+13	1.628E+13
5.75E-01	1.651E+15	1.460E+15	1.156E+15	9.177E+14
8.50E-01	2.219E+13	1.370E+13	8.849E+12	6.512E+12
1.25E+00	1.476E+13	1.025E+13	5.945E+12	3.856E+12
1.75E+00	9.235E+11	6.974E+11	4.827E+11	3.512E+11
2.25E+00	1.246E+11	2.009E+09	5.960E+07	4.507E+07
2.75E+00	4.298E+09	1.359E+08	1.946E+05	4.293E+04
3.50E+00	6.965E+08	2.244E+07	7.179E+04	4.127E+04
5.00E+00	3.636E+04	3.206E+04	2.553E+04	2.102E+04
7.00E+00	5.168E+03	4.478E+03	3.435E+03	2.717E+03
9.50E+00	7.422E+02	6.351E+02	4.734E+02	3.625E+02

TABLE A-12

GAMMA SOURCE STRENGTH FOR DEFENSE HIGH LEVEL WASTE FROM
IDAHO NATIONAL ENGINEERING LABORATORY (INEL)

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec-canister)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	4.397E+12	3.814E+12	2.994E+12	2.361E+12
2.50E-02	2.264E+12	1.967E+12	1.545E+12	1.219E+12
3.75E-02	3.542E+12	3.078E+12	2.415E+12	1.903E+12
5.75E-02	4.827E+12	4.203E+12	3.303E+12	2.604E+12
8.50E-02	4.237E+12	3.690E+12	2.904E+12	2.289E+12
1.25E-01	4.351E+12	3.685E+12	2.836E+12	2.206E+12
2.25E-01	9.576E+12	8.341E+12	6.548E+12	5.150E+12
3.75E-01	6.825E+12	5.948E+12	4.680E+12	3.685E+12
5.75E-01	3.338E+14	2.820E+14	2.207E+14	1.750E+14
8.50E-01	2.658E+13	7.763E+12	2.539E+12	1.661E+12
1.25E+00	7.134E+12	3.945E+12	1.956E+12	1.150E+12
1.75E+00	3.061E+11	2.200E+11	1.371E+11	9.203E+10
2.25E+00	8.213E+10	1.072E+09	1.311E+07	1.011E+07
2.75E+00	9.548E+08	2.896E+07	5.052E+04	1.689E+04
3.50E+00	1.448E+08	4.683E+06	2.905E+04	1.891E+04
5.00E+00	2.218E+04	1.914E+04	1.449E+04	1.124E+04
7.00E+00	3.516E+03	3.028E+03	2.281E+03	1.760E+03
9.50E+00	5.428E+02	4.667E+02	3.506E+02	2.696E+02

TABLE A-13

GAMMA SOURCE STRENGTH FOR DEFENSE HIGH LEVEL WASTE FROM
SAVANAH RIVER PLANT (SRP)

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec-canister)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	1.214E+13	1.068E+13	8.413E+12	6.644E+12
2.50E-02	6.378E+12	5.523E+12	4.318E+12	3.400E+12
3.75E-02	9.653E+12	8.430E+12	6.596E+12	5.194E+12
5.75E-02	1.336E+13	1.176E+13	9.258E+12	7.308E+12
8.50E-02	1.193E+13	1.043E+13	8.155E+12	6.414E+12
1.25E-01	1.211E+13	1.039E+13	7.961E+12	6.185E+12
2.25E-01	2.671E+13	2.342E+13	1.838E+13	1.446E+13
3.75E-01	2.022E+13	1.708E+13	1.320E+13	1.038E+13
5.75E-01	8.222E+14	7.291E+14	5.779E+14	4.583E+14
8.50E-01	1.494E+13	1.061E+13	6.696E+12	4.635E+12
1.25E+00	2.191E+13	1.400E+13	6.500E+12	3.500E+12
1.75E+00	8.166E+11	6.064E+11	3.805E+11	2.567E+11
2.25E+00	8.177E+10	1.223E+09	5.045E+07	3.355E+07
2.75E+00	2.115E+09	6.020E+08	5.401E+08	4.934E+08
3.50E+00	2.677E+08	1.222E+07	2.611E+06	1.809E+06
5.00E+00	2.763E+06	2.294E+06	1.585E+06	1.101E+06
7.00E+00	4.451E+05	3.693E+05	2.550E+05	1.768E+05
9.50E+00	6.930E+04	5.570E+04	3.968E+04	2.750E+04

TABLE A-14

GAMMA SOURCE STRENGTH FOR WEST VALLEY (WVDP)
HIGH LEVEL WASTE

Mean Gamma Energy (MeV)	Gamma Source Strength (MeV/sec-canister)			
	5 Yr	10 Yr	20 Yr	30 Yr
1.00E-02	7.845E+12	6.968E+12	5.502E+12	4.348E+12
2.50E-02	4.035E+12	3.583E+12	2.828E+12	2.231E+12
3.75E-02	6.465E+12	5.723E+12	4.500E+12	3.546E+12
5.75E-02	8.993E+12	8.016E+12	6.394E+12	5.113E+12
8.50E-02	7.608E+12	6.744E+12	5.314E+12	4.196E+12
1.25E-01	7.755E+12	6.755E+12	5.198E+12	4.046E+12
2.25E-01	1.712E+13	1.515E+13	1.191E+13	9.376E+12
3.75E-01	1.215E+13	1.077E+13	8.475E+12	6.675E+12
5.75E-01	6.015E+14	5.354E+14	4.248E+14	3.371E+14
8.50E-01	9.019E+12	6.960E+12	4.445E+12	3.043E+12
1.25E+00	9.671E+12	6.905E+12	3.718E+12	2.168E+12
1.75E+00	5.408E+11	4.125E+11	2.550E+11	1.696E+11
2.25E+00	4.127E+07	3.555E+07	2.714E+07	2.099E+07
2.75E+00	6.867E+07	1.463E+08	2.314E+08	2.620E+08
3.50E+00	5.198E+06	4.344E+06	3.060E+06	2.185E+06
5.00E+00	3.174E+06	2.653E+06	1.869E+06	1.333E+06
7.00E+00	5.121E+05	4.280E+05	3.014E+05	2.149E+05
9.50E+00	7.981E+04	6.671E+04	4.696E+04	3.349E+04

TABLE A-15

NEUTRON SOURCE STRENGTH OF A SINGLE PWR
ASSEMBLY (ANISN/CASK FORMAT)

<u>CASK Group</u>	<u>Upper Energy (MeV)</u>	<u>Source Strength (Neutron/ Sec-cm³)</u>
1	1.49E+01	2.4284E+00
2	1.22E+01	9.8274E+00
3	1.00E+01	3.0041E+01
4	8.18E+00	1.0041E+02
5	6.36E+00	2.0876E+02
6	4.96E+00	2.7003E+02
7	4.06E+00	5.7096E+02
8	3.01E+00	4.5948E+02
9	2.46E+00	1.0897E+02
10	2.35E+00	6.0332E+02
11	1.83E+00	1.0902E+03
12	1.11E+00	1.0020E+03
13	5.50E-01	6.9256E+02
14	1.10E-01	7.0196E+01

NOTE:

1. PWR fuel assembly at 60 Gwd/MTU burnup and 5 yr decay.

TABLE A-16

GAMMA SOURCE STRENGTH OF A SINGLE PWR
ASSEMBLY (ANISN/CASK FORMAT)

<u>CASK Group</u>	<u>Upper Energy (MeV)</u>	<u>Source Strength (Neutron/ Sec-cm²)</u>
23	1.00E+01	2.9923E+00
24	8.00E+01	2.6047E+01
25	6.50E+01	2.2585E+02
27	4.00E+00	7.0954E+04
28	3.00E+00	5.5039E+05
29	2.50E+00	1.3149E+07
30	2.00E+00	4.2716E+07
32	1.33E+00	2.5649E+09
33	1.00E+00	7.1064E+09
35	6.00E-01	2.6575E+10
36	4.00E-01	8.4189E+08
37	3.00E-01	1.4669E+09
38	2.00E-01	1.9551E+09
39	1.00E-01	4.4932E+09
40	5.00E-02	2.0507E+10

NOTE:

1. PWR fuel assembly at 60 Gwd/MTU burnup and 5 yr decay.

TABLE A-17

NEUTRON SOURCE STRENGTH OF A SINGLE BWR
ASSEMBLY (ANISN/CASK FORMAT)

<u>CASK Group</u>	<u>Upper Energy (MeV)</u>	<u>Source Strength (Neutron/ Sec-cm³)</u>
1	1.49E+01	5.3835E-01
2	1.22E+01	2.1786E+00
3	1.00E+01	6.6597E+00
4	8.18E+00	2.2261E+01
5	6.36E+00	4.6280E+01
6	4.96E+00	5.9863E+01
7	4.06E+00	1.2658E+02
8	3.01E+00	1.0186E+02
9	2.46E+00	2.4158E+01
10	2.35E+00	1.3375E+02
11	1.83E+00	2.4170E+02
12	1.11E+00	2.2214E+02
13	5.50E-01	1.5353E+02
14	1.10E-01	1.5562E+01

NOTE:

1. BWR fuel assembly at 40 Gwd/MTU burnup and 5 yr decay.

TABLE A-18

GAMMA SOURCE STRENGTH OF A SINGLE BWR
ASSEMBLY (ANISN/CASK FORMAT)

<u>CASK Group</u>	<u>Upper Energy (MeV)</u>	<u>Source Strength (Neutron/ Sec-cm³)</u>
23	1.00E+01	6.6053E-01
24	8.00E+00	5.7497E+00
25	6.50E+00	4.9874E+01
27	4.00E+00	3.8264E+04
28	3.00E+00	2.9834E+05
29	2.50E+00	8.1626E+06
30	2.00E+00	2.2464E+07
32	1.33E+00	9.8789E+08
33	1.00E+00	3.2181E+09
35	6.00E-01	1.4685E+10
36	4.00E-01	5.2449E+08
37	3.00E-01	8.9880E+08
38	2.00E-01	1.1217E+09
39	1.00E-01	2.7436E+09
40	5.00E-02	1.2541E+10

NOTE:

1. BWR fuel assembly at 40 Gwd/MTU burnup and 5 yr decay.

TABLE A-19

NEUTRON SOURCE STRENGTH OF A SINGLE CONTAINER
OF 6 CONSOLIDATED PWR FUEL ASSEMBLIES (ANISN/CASK FORMAT)

<u>CASK Group</u>	<u>Upper Energy (MeV)</u>	<u>Source Strength (Neutron/ Sec-cm³)</u>
1	1.49E+01	2.8407E+00
2	1.22E+01	1.1496E+01
3	1.00E+01	3.5140E+01
4	8.18E+00	1.1746E+02
5	6.36E+00	2.4420E+02
6	4.96E+00	3.1587E+02
7	4.06E+00	6.6789E+02
8	3.01E+00	5.3748E+02
9	2.46E+00	1.2747E+02
10	2.35E+00	7.0574E+02
11	1.83E+00	1.2753E+03
12	1.11E+00	1.1722E+03
13	5.50E-01	8.1013E+02
14	1.10E-01	8.2112E+01

NOTE:

1. PWR fuel assemblies at 60 Gwd/MTU burnup and 5 yr decay.

TABLE A-20

GAMMA SOURCE STRENGTH OF A SINGLE CONTAINER
OF 6 CONSOLIDATED PWR FUEL ASSEMBLIES (ANISN/CASK FORMAT)

<u>CASK Group</u>	<u>Upper Energy (MeV)</u>	<u>Source Strength (Neutron/ Sec-cm³)</u>
23	1.00E+01	3.5003E+00
24	8.00E+00	3.0468E+01
25	6.50E+00	2.6418E+02
27	4.00E+00	8.2998E+04
28	3.00E+00	6.4381E+05
29	2.50E+00	1.5381E+07
30	2.00E+00	4.9967E+07
32	1.33E+00	3.0003E+09
33	1.00E+00	8.3127E+09
35	6.00E-01	3.1085E+10
36	4.00E-01	9.8479E+08
37	3.00E-01	1.7159E+09
38	2.00E-01	2.2870E+09
39	1.00E-01	5.2559E+09
40	5.00E-02	2.3988E+10

NOTE:

1. PWR fuel assemblies at 60 Gwd/MTU burnup and 5 yr decay.

TABLE A-21

NEUTRON SOURCE STRENGTH FOR A SINGLE CONTAINER
OF 18 CONSOLIDATED BWR FUEL ASSEMBLIES (ANISN/CASK FORMAT)

<u>CASK Group</u>	<u>Upper Energy (MeV)</u>	<u>Source Strength (Neutron/ Sec-cm³)</u>
1	1.49E+01	7.9985E-01
2	1.22E+01	3.2369E+00
3	1.00E+01	9.8946E+00
4	8.18E+00	3.3074E+01
5	6.36E+00	6.8760E+01
6	4.96E+00	8.8941E+01
7	4.06E+00	1.8806E+02
8	3.01E+00	1.5134E+02
9	2.46E+00	3.5893E+01
10	2.35E+00	1.9872E+02
11	1.83E+00	3.5910E+02
12	1.11E+00	3.3005E+02
13	5.50E-01	2.2811E+02
14	1.10E-01	2.3121E+01

NOTE:

1. BWR fuel assemblies at 40 Gwd/MTU burnup and 5 yr decay.

TABLE A-22

GAMMA SOURCE STRENGTH OF A SINGLE CONTAINER
OF 18 CONSOLIDATED BWR FUEL ASSEMBLIES (ANISN/CASK FORMAT)

<u>CASK Group</u>	<u>Upper Energy (MeV)</u>	<u>Source Strength (Neutron/ Sec-cm³)</u>
23	1.00E+01	1.2450E+10
24	8.00E+00	2.9328E+09
25	6.50E+00	3.2448E+09
27	4.00E+00	2.4736E+09
28	3.00E+00	1.6016E+09
29	2.50E+00	1.6661E+09
30	2.00E+00	1.3350E+09
32	1.33E+00	7.7904E+08
33	1.00E+00	2.1812E+10
35	6.00E-01	4.7800E+09
36	4.00E-01	1.4674E+09
37	3.00E-01	3.3366E+07
38	2.00E-01	1.2124E+07
39	1.00E-01	4.9997E+05
40	5.00E-02	8.3602E+01

NOTE:

1. BWR fuel assemblies at 60 Gwd/MTU burnup and 5 yr decay.

TABLE A-23

NEUTRON SOURCE STRENGTH OF A SINGLE CONTAINER OF
6 CONSOLIDATED PWR FUEL ASSEMBLIES OF VARYING BURNUP AND AGE
(ANISN/CASK FORMAT)

CASK Group	Neutron Source Strength			
	33 Gwd/MTU, 5 Yr. Decay	45 Gwd/MTU, 5 Yr. Decay	33 Gwd/MTU, 20 Yr. Decay	45 Gwd/MTU, 20 Yr. Decay
	(n/cm ³ /sec)	(n/cm ³ /sec)	(n/cm ³ /sec)	(n/cm ³ /sec)
1	3.4543E-1	7.9134E-1	2.0166E-1	4.5672E-1
2	1.3979	3.2024	8.1611E-1	1.8483
3	4.2732	9.7892	2.4947	5.6498
4	1.4284E+1	3.2721E+1	8.3388	1.8885E+1
5	2.9696E+1	6.8028E+1	1.7336E+1	3.9262E+1
6	3.8411E+1	8.7994E+1	2.2425E+1	5.0786E+1
7	8.1217E+1	1.8606E+2	4.7415E+1	1.0738E+2
8	6.5360E+1	1.4973E+2	3.8157E+1	8.6416E+1
9	1.5501E+1	3.5511E+1	9.0496	2.0495E+1
10	8.5820E+1	1.9660E+2	5.0102E+1	1.1347E+2
11	1.5508E+2	2.5528E+2	9.0539E+1	2.0505E+2
12	1.4254E+2	3.2653E+2	8.3215E+1	1.8846E+2
13	9.8515E+1	2.2568E+2	5.7514E+1	1.3025E+2
14	9.9851	2.2874E+1	5.8294	1.3202E+1

APPENDIX B

GAMMA DOSE RATE FIGURES

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Figure B-1

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Consolidated
BWR Fuel Assembly (10 Gwd Burnup)

Figure B-2

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Bare Consolidated
BWR Fuel Assembly (10 Gwd Burnup)

Figure B-3

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Bare Consolidated
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Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Bare Consolidated
BWR Fuel Assembly (20 Gwd Burnup)

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Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Bare Consolidated
BWR Fuel Assembly (27.5 Gwd Burnup)

Figure B-6

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Bare Consolidated
BWR Fuel Assembly (27.5 Gwd Burnup)

Figure B-7

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Bare Consolidated
BWR Fuel Assembly (40 Gwd Burnup)

Figure B-8

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Bare Consolidated
BWR Fuel Assembly (40 Gwd Burnup)

Figure B-9

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Bare Consolidated
PWR Fuel Assembly (10 Gwd Burnup)

Figure B-10

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Bare Consolidated
PWR Fuel Assembly (10 Gwd Burnup)

Figure B-11

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Bare Consolidated
PWR Fuel Assembly (25 Gwd Burnup)

Figure B-12

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Bare Consolidated
PWR Fuel Assembly (25 Gwd Burnup)

Figure B-13

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Bare Consolidated
PWR Fuel Assembly (33 Gwd Burnup)

Figure B-14

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Bare Consolidated
PWR Fuel Assembly (33 Gwd Burnup)

Figure B-15

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Bare Consolidated
PWR Fuel Assembly (45 Gwd Burnup)

Figure B-16

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Bare Consolidated
PWR Fuel Assembly (45 Gwd Burnup)

Figure B-17

Gamma Dose Rates in a Concrete Wall as a
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Gamma Dose Rates in a Concrete Slab as a
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Figure B-19

Gamma Dose Rates in Concrete for a Single
Container of 3 Intact PWR + 4 intact
BWR Fuel Assemblies (Hybrid)

Figure B-20

Gamma Dose Rates in Concrete for a Single
Container of 18 Consolidated BWR Fuel
Assemblies at 40 Gwd Burnup, 5 Years Decay

Figure B-21

Gamma Dose Rates in Concrete for a Single
Container of 10 Intact BWR Fuel Assemblies
at 40 Gwd Burnup, 5 Years Decay

Figure B-22

Gamma Dose Rates in Concrete for a Single
Container of 4 Intact PWR Fuel Assemblies
at 60 Gwd Burnup, 5 Years Decay

Figure B-23

Gamma Dose Rates in Concrete for a Single
Container of 6 Consolidated PWR Fuel Assemblies
at 60 Gwd Burnup, 5 Years Decay

Figure B-24

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Container
of Hanford Site Defense HLW

Figure B-25

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Container
of Hanford Site Defense HLW

Figure B-26

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Container
of INEL Defense HLW

Figure B-27

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Container
of INEL Defense HLW

Figure B-28

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Container
of SRP Defense HLW

Figure B-29

Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Container
of SRP Defense HLW

Figure B-30

Gamma Dose Rates in a Concrete Wall as a
Function of Decay Age for a Single Container
of WVDP HLW

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Gamma Dose Rates in a Concrete Slab as a
Function of Decay Age for a Single Container
of WVDP HLW

Figure B-32

Gamma Dose Rates in a Concrete Borehole for a
Single Container of 18 Consolidated BWR Fuel Assemblies
at 40 Gwd Burnup, 5 Years Decay

Figure B-33

Gamma Dose Rates in a Tuff Borehole for a
Single Container of 18 Consolidated BWR Fuel Assemblies
at 40 Gwd Burnup, 5 Years Decay

Figure B-34

Gamma Dose Rates in a Steel Borehole for a
Single Container of 18 Consolidated BWR Fuel Assemblies
at 40 Gwd Burnup, 5 Years Decay

Figure B-35

Gamma Dose Rates in a Concrete Borehole for a
Single Container of 6 Consolidated PWR Fuel Assemblies
at 60 Gwd Burnup, 5 Years Decay

Figure B-36

Gamma Dose Rates in a Tuff Borehole for a
Single Container of 6 Consolidated PWR Fuel Assemblies
at 60 Gwd Burnup, 5 Years Decay

Figure B-37

Gamma Dose Rates in a Steel Borehole for a
Single Container of 6 Consolidated PWR Fuel Assemblies
at 60 Gwd Burnup, 5 Years Decay

Figure B-38

Gamma Dose Rates in a Concrete Surface Storage Vault for
Containers of Hybrid and other BWR Fuel Assembly Forms

Figure B-39

Gamma Dose Rates in a Concrete for
Containers of PWR Fuel Assembly Forms

Figure B-40

Gamma Dose Rates in a Concrete Surface Storage Vault for
Containers of Hanford Defense HLW or INEL Defense
HLW at 5 Years Decay

Figure B-41

Gamma Dose Rates in a Concrete Surface Storage Vault for
Containers of SRP Defense HLW or WVDP
HLW at 5 Years Decay

APPENDIX C

GAMMA DOSE RATE RATIO TABLES

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TABLE C-1

SINGLE CONSOLIDATED PWR ASSEMBLY, 10 GWD BURNUP - WALL RATIOS

<u>Wall Concrete Thickness</u>	<u>10yr/5yr Dose Rate Ratio</u>	<u>20yr/5yr Dose Rate Ratio</u>	<u>30yr/5yr Dose Rate Ratio</u>
0'-6"	5.95E-01	3.74E-01	2.76E-01
1'-0"	5.41E-01	3.01E-01	2.09E-01
1'-6"	4.85E-01	2.31E-01	1.45E-01
2'-0"	4.35E-01	1.75E-01	9.52E-02
2'-6"	3.85E-01	1.36E-01	6.39E-02
3'-0"	3.36E-01	1.11E-01	4.55E-02
3'-2"	3.18E-01	1.02E-01	4.11E-02
3'-4"	3.00E-01	9.46E-02	3.74E-02
3'-6"	2.82E-01	8.78E-02	3.40E-02
3'-8"	2.64E-01	8.10E-02	3.10E-02
3'-10"	2.47E-01	7.53E-02	2.85E-02
4'-0"	2.28E-01	6.95E-02	2.61E-02
4'-2"	2.10E-01	6.36E-02	2.38E-02
4'-4"	1.94E-01	5.82E-02	2.17E-02
4'-6"	1.77E-01	5.31E-02	1.99E-02
4'-8"	1.62E-01	4.83E-02	1.81E-02
4'-10"	1.48E-01	4.37E-02	1.64E-02
5'-0"	1.34E-01	3.93E-02	1.49E-02
5'-6"	9.91E-02	2.82E-02	1.09E-02
6'-0"	7.33E-02	1.99E-02	7.95E-03
6'-6"	5.50E-02	1.39E-02	5.76E-03
7'-0"	4.25E-02	9.70E-03	4.18E-03

TABLE C-2

SINGLE CONSOLIDATED PWR ASSEMBLY, 10 GWD BURNUP - SLAB RATIOS

<u>Slab Concrete Thickness</u>	<u>10yr/5yr Dose Rate Ratio</u>	<u>20yr/5yr Dose Rate Ratio</u>	<u>30yr/5yr Dose Rate Ratio</u>
0'-6"	6.00E-01	3.83E-01	2.85E-01
1'-0"	5.51E-01	3.14E-01	2.22E-01
1'-6"	4.97E-01	2.44E-01	1.56E-01
2'-0"	4.43E-01	1.85E-01	1.04E-01
2'-6"	3.99E-01	1.44E-01	6.96E-02
3'-0"	3.48E-01	1.15E-01	4.84E-02
3'-2"	3.28E-01	1.06E-01	4.36E-02
3'-4"	3.14E-01	1.00E-01	4.01E-02
3'-6"	2.97E-01	9.32E-02	3.65E-02
3'-8"	2.79E-01	8.69E-02	3.33E-02
3'-10"	2.61E-01	8.05E-02	3.05E-02
4'-0"	2.43E-01	7.43E-02	2.79E-02
4'-2"	2.26E-01	6.85E-02	2.56E-02
4'-4"	2.08E-01	6.30E-02	2.35E-02
4'-6"	1.92E-01	5.78E-02	2.15E-02
4'-8"	1.76E-01	5.26E-02	1.96E-02
4'-10"	1.59E-01	4.75E-02	1.78E-02
5'-0"	1.45E-01	4.29E-02	1.62E-02
5'-6"	1.08E-01	3.11E-02	1.20E-02
6'-0"	7.99E-02	2.21E-02	8.72E-03
6'-6"	5.98E-02	1.55E-02	6.36E-03
7'-0"	4.58E-02	1.08E-02	4.60E-03

TABLE C-3

SINGLE CONSOLIDATED PWR ASSEMBLY, 25 GWD BURNUP - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.38E-01	3.15E-01	2.28E-01
1'-0"	5.04E-01	2.62E-01	1.76E-01
1'-6"	4.74E-01	2.13E-01	1.29E-01
2'-0"	4.50E-01	1.78E-01	9.41E-02
2'-6"	4.30E-01	1.54E-01	7.15E-02
3'-0"	4.00E-01	1.37E-01	5.81E-02
3'-2"	3.92E-01	1.32E-01	5.47E-02
3'-4"	3.79E-01	1.27E-01	5.19E-02
3'-6"	3.66E-01	1.22E-01	4.91E-02
3'-8"	3.54E-01	1.17E-01	4.66E-02
3'-10"	3.39E-01	1.12E-01	4.45E-02
4'-0"	3.22E-01	1.07E-01	4.22E-02
4'-2"	3.06E-01	1.01E-01	3.99E-02
4'-4"	2.90E-01	9.55E-02	3.78E-02
4'-6"	2.73E-01	9.01E-02	3.55E-02
4'-8"	2.56E-01	8.42E-02	3.33E-02
4'-10"	2.38E-01	7.85E-02	3.12E-02
5'-0"	2.22E-01	7.31E-02	2.92E-02
5'-6"	1.74E-01	5.69E-02	2.31E-02
6'-0"	1.32E-01	4.25E-02	1.77E-02
6'-6"	1.01E-01	3.15E-02	1.34E-02
7'-0"	7.63E-02	2.30E-02	1.01E-02

TABLE C-4

SINGLE CONSOLIDATED PWR ASSEMBLY, 25 GWD BURNUP - SLAB RATIOS

Slab Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.42E-01	3.23E-01	2.35E-01
1'-0"	5.10E-01	2.71E-01	1.86E-01
1'-6"	4.80E-01	2.24E-01	1.38E-01
2'-0"	4.55E-01	1.83E-01	1.00E-01
2'-6"	4.33E-01	1.58E-01	7.53E-02
3'-0"	4.08E-01	1.40E-01	6.02E-02
3'-2"	3.99E-01	1.36E-01	5.70E-02
3'-4"	3.88E-01	1.30E-01	5.38E-02
3'-6"	3.77E-01	1.26E-01	5.12E-02
3'-8"	3.64E-01	1.21E-01	4.86E-02
3'-10"	3.50E-01	1.16E-01	4.62E-02
4'-0"	3.36E-01	1.11E-01	4.38E-02
4'-2"	3.22E-01	1.06E-01	4.18E-02
4'-4"	3.04E-01	1.00E-01	3.96E-02
4'-6"	2.87E-01	9.49E-02	3.74E-02
4'-8"	2.70E-01	8.89E-02	3.51E-02
4'-10"	2.54E-01	8.42E-02	3.32E-02
5'-0"	2.36E-01	7.78E-02	3.09E-02
5'-6"	1.87E-01	6.13E-02	2.48E-02
6'-0"	1.44E-01	4.64E-02	1.92E-02
6'-6"	1.09E-01	3.45E-02	1.46E-02
7'-0"	8.37E-02	2.54E-02	1.11E-02

TABLE C-5

SINGLE CONSOLIDATED PWR ASSEMBLY, 33 GWD BURNUP -- WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.01E-01	2.87E-01	2.05E-01
1'-0"	4.82E-01	2.41E-01	1.60E-01
1'-6"	4.59E-01	2.01E-01	1.19E-01
2'-0"	4.44E-01	1.72E-01	8.91E-02
2'-6"	4.28E-01	1.53E-01	7.04E-02
3'-0"	4.07E-01	1.40E-01	5.93E-02
3'-2"	3.99E-01	1.36E-01	5.63E-02
3'-4"	3.88E-01	1.32E-01	5.41E-02
3'-6"	3.77E-01	1.27E-01	5.18E-02
3'-8"	3.66E-01	1.23E-01	4.97E-02
3'-10"	3.53E-01	1.19E-01	4.77E-02
4'-0"	3.39E-01	1.14E-01	4.54E-02
4'-2"	3.23E-01	1.09E-01	4.35E-02
4'-4"	3.07E-01	1.03E-01	4.13E-02
4'-6"	2.92E-01	9.82E-02	3.92E-02
4'-8"	2.77E-01	9.22E-02	3.69E-02
4'-10"	2.58E-01	8.66E-02	3.48E-02
5'-0"	2.41E-01	8.14E-02	3.28E-02
5'-6"	1.91E-01	6.41E-02	2.63E-02
6'-0"	1.48E-01	4.92E-02	2.06E-02
6'-6"	1.14E-01	3.68E-02	1.58E-02
7'-0"	8.76E-02	2.74E-02	1.21E-02

TABLE C-6

SINGLE CONSOLIDATED PWR ASSEMBLY, 33 GWD BURNUP - SLAB RATIOS

<u>Slab Concrete Thickness</u>	<u>10yr/5yr Dose Rate Ratio</u>	<u>20yr/5yr Dose Rate Ratio</u>	<u>30yr/5yr Dose Rate Ratio</u>
0'-6"	5.15E-01	2.94E-01	2.11E-01
1'-0"	4.88E-01	2.49E-01	1.68E-01
1'-6"	4.64E-01	2.08E-01	1.26E-01
2'-0"	4.48E-01	1.77E-01	9.46E-02
2'-6"	4.31E-01	1.56E-01	7.32E-02
3'-0"	4.13E-01	1.42E-01	6.15E-02
3'-2"	4.06E-01	1.38E-01	5.84E-02
3'-4"	3.98E-01	1.35E-01	5.60E-02
3'-6"	3.88E-01	1.31E-01	5.34E-02
3'-8"	3.78E-01	1.27E-01	5.15E-02
3'-10"	3.64E-01	1.23E-01	4.92E-02
4'-0"	3.52E-01	1.18E-01	4.71E-02
4'-2"	3.36E-01	1.13E-01	4.50E-02
4'-4"	3.21E-01	1.08E-01	4.31E-02
4'-6"	3.05E-01	1.03E-01	4.10E-02
4'-8"	2.89E-01	9.73E-02	3.89E-02
4'-10"	2.72E-01	9.16E-02	3.68E-02
5'-0"	2.55E-01	8.60E-02	3.45E-02
5'-6"	2.05E-01	6.92E-02	2.81E-02
6'-0"	1.60E-01	5.34E-02	2.22E-02
6'-6"	1.23E-01	4.02E-02	1.72E-02
7'-0"	9.47E-01	3.00E-02	1.32E-02

TABLE C-7

SINGLE CONSOLIDATED PWR ASSEMBLY, 45 GWD BURNUP - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.02E-01	2.83E-01	2.02E-01
1'-0"	4.76E-01	2.40E-01	1.60E-01
1'-6"	4.59E-01	2.03E-01	1.22E-01
2'-0"	4.47E-01	1.77E-01	9.39E-02
2'-6"	4.38E-01	1.61E-01	7.64E-02
3'-0"	4.24E-01	1.51E-01	6.61E-02
3'-2"	4.17E-01	1.47E-01	6.37E-02
3'-4"	4.11E-01	1.45E-01	6.13E-02
3'-6"	4.02E-01	1.41E-01	5.94E-02
3'-8"	3.92E-01	1.38E-01	5.75E-02
3'-10"	3.83E-01	1.34E-01	5.57E-02
4'-0"	3.70E-01	1.30E-01	5.38E-02
4'-2"	3.57E-01	1.26E-01	5.20E-02
4'-4"	3.44E-01	1.21E-01	5.01E-02
4'-6"	3.30E-01	1.16E-01	4.80E-02
4'-8"	3.13E-01	1.10E-01	4.60E-02
4'-10"	2.98E-01	1.05E-01	4.38E-02
5'-0"	2.81E-01	9.98E-02	4.18E-02
5'-6"	2.29E-01	8.17E-02	3.46E-02
6'-0"	1.83E-01	6.51E-02	2.82E-02
6'-6"	1.43E-01	5.03E-02	2.23E-02
7'-0"	1.12E-01	3.86E-02	1.77E-02

TABLE C-8

SINGLE CONSOLIDATED PWR ASSEMBLY, 45 GWD BURNUP - SLAB RATIOS

<u>Slab Concrete Thickness</u>	<u>10yr/5yr Dose Rate Ratio</u>	<u>20yr/5yr Dose Rate Ratio</u>	<u>30yr/5yr Dose Rate Ratio</u>
0'-6"	5.07E-01	2.88E-01	2.08E-01
1'-0"	4.82E-01	2.47E-01	1.67E-01
1'-6"	4.61E-01	2.09E-01	1.29E-01
2'-0"	4.48E-01	1.81E-01	9.88E-02
2'-6"	4.40E-01	1.64E-01	7.92E-02
3'-0"	4.29E-01	1.53E-01	6.81E-02
3'-2"	4.24E-01	1.50E-01	6.56E-02
3'-4"	4.17E-01	1.47E-01	6.29E-02
3'-6"	4.09E-01	1.44E-01	6.08E-02
3'-8"	4.00E-01	1.40E-01	5.88E-02
3'-10"	3.92E-01	1.37E-01	5.72E-02
4'-0"	3.82E-01	1.34E-01	5.54E-02
4'-2"	3.70E-01	1.30E-01	5.35E-02
4'-4"	3.55E-01	1.24E-01	5.16E-02
4'-6"	3.42E-01	1.20E-01	4.99E-02
4'-8"	3.25E-01	1.15E-01	4.76E-02
4'-10"	3.11E-01	1.10E-01	4.56E-02
5'-0"	2.95E-01	1.05E-01	4.35E-02
5'-6"	2.44E-01	8.70E-02	3.68E-02
6'-0"	1.96E-01	6.96E-02	3.01E-02
6'-6"	1.54E-01	5.45E-02	2.39E-02
7'-0"	1.21E-01	4.20E-02	1.91E-02

TABLE C-9

SINGLE CONSOLIDATED PWR ASSEMBLY, 60 GWD BURNUP - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	4.79E-01	2.60E-01	1.84E-01
1'-0"	4.59E-01	2.23E-01	1.46E-01
1'-6"	4.46E-01	1.92E-01	1.13E-01
2'-0"	4.39E-01	1.71E-01	8.93E-02
2'-6"	4.36E-01	1.60E-01	7.48E-02
3'-0"	4.29E-01	1.52E-01	6.68E-02
3'-2"	4.26E-01	1.50E-01	6.47E-02
3'-4"	4.20E-01	1.49E-01	6.32E-02
3'-6"	4.15E-01	1.46E-01	6.16E-02
3'-8"	4.06E-01	1.44E-01	6.02E-02
3'-10"	3.98E-01	1.41E-01	5.86E-02
4'-0"	3.89E-01	1.38E-01	5.71E-02
4'-2"	3.78E-01	1.34E-01	5.57E-02
4'-4"	3.66E-01	1.30E-01	5.40E-02
4'-6"	3.53E-01	1.26E-01	5.23E-02
4'-8"	3.38E-01	1.21E-01	5.05E-02
4'-10"	3.24E-01	1.16E-01	4.86E-02
5'-0"	3.08E-01	1.11E-01	4.67E-02
5'-6"	2.57E-01	9.29E-02	3.96E-02
6'-0"	2.11E-01	7.60E-02	3.32E-02
6'-6"	1.67E-01	6.05E-02	2.72E-02
7'-0"	1.33E-01	4.84E-02	2.28E-02

TABLE C-10

SINGLE CONSOLIDATED PWR ASSEMBLY, 60 GWD BURNUP - SLAB RATIOS

Slab Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	4.83E-01	2.65E-01	1.89E-01
1'-0"	4.61E-01	2.28E-01	1.52E-01
1'-6"	4.47E-01	1.96E-01	1.18E-01
2'-0"	4.41E-01	1.74E-01	9.31E-02
2'-6"	4.38E-01	1.61E-01	7.73E-02
3'-0"	4.33E-01	1.54E-01	6.81E-02
3'-2"	4.29E-01	1.52E-01	6.62E-02
3'-4"	4.26E-01	1.50E-01	6.44E-02
3'-6"	4.18E-01	1.48E-01	6.26E-02
3'-8"	4.13E-01	1.46E-01	6.12E-02
3'-10"	4.06E-01	1.43E-01	5.98E-02
4'-0"	3.98E-01	1.40E-01	5.84E-02
4'-2"	3.88E-01	1.37E-01	5.71E-02
4'-4"	3.76E-01	1.34E-01	5.54E-02
4'-6"	3.64E-01	1.30E-01	5.38E-02
4'-8"	3.51E-01	1.25E-01	5.21E-02
4'-10"	3.36E-01	1.20E-01	5.02E-02
5'-0"	3.21E-01	1.15E-01	4.83E-02
5'-6"	2.72E-01	9.83E-02	4.17E-02
6'-0"	2.22E-01	8.10E-02	3.50E-02
6'-6"	1.78E-01	6.50E-02	2.90E-02
7'-0"	1.42E-01	5.17E-02	2.40E-02

TABLE C-11

SINGLE CONSOLIDATED BWR ASSEMBLY, 10 GWD BURNUP - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	6.06E-01	4.06E-01	3.09E-01
1'-0"	5.51E-01	3.40E-01	2.49E-01
1'-6"	4.88E-01	2.65E-01	1.82E-01
2'-0"	4.20E-01	1.97E-01	1.23E-01
2'-6"	3.58E-01	1.47E-01	8.11E-02
3'-0"	2.99E-01	1.11E-01	5.50E-02
3'-2"	2.77E-01	1.01E-01	4.86E-02
3'-4"	2.60E-01	9.28E-02	4.36E-02
3'-6"	2.41E-01	8.48E-02	3.90E-02
3'-8"	2.23E-01	7.75E-02	3.49E-02
3'-10"	2.05E-01	7.06E-02	3.16E-02
4'-0"	1.88E-01	6.42E-02	2.85E-02
4'-2"	1.73E-01	5.85E-02	2.59E-02
4'-4"	1.58E-01	5.30E-02	2.34E-02
4'-6"	1.44E-01	4.80E-02	2.12E-02
4'-8"	1.31E-01	4.35E-02	1.92E-02
4'-10"	1.19E-01	3.91E-02	1.74E-02
5'-0"	1.08E-01	3.52E-02	1.57E-02
5'-6"	8.13E-02	2.53E-02	1.15E-02
6'-0"	6.17E-02	1.82E-02	8.49E-03
6'-6"	4.82E-02	1.30E-02	6.25E-03
7'-0"	3.90E-02	9.37E-03	4.64E-03

TABLE C-12

SINGLE CONSOLIDATED BWR ASSEMBLY, 10 GWD BURNUP - SLAB RATIOS

<u>Slab Concrete Thickness</u>	<u>10yr/5yr Dose Rate Ratio</u>	<u>20yr/5yr Dose Rate Ratio</u>	<u>30yr/5yr Dose Rate Ratio</u>
0'-6"	6.16E-01	4.16E-01	3.18E-01
1'-0"	5.64E-01	3.54E-01	2.62E-01
1'-6"	5.01E-01	2.81E-01	1.95E-01
2'-0"	4.35E-01	2.12E-01	1.34E-01
2'-6"	3.73E-01	1.57E-01	8.89E-02
3'-0"	3.14E-01	1.19E-01	6.01E-02
3'-2"	2.94E-01	1.08E-01	5.30E-02
3'-4"	2.74E-01	9.93E-02	4.74E-02
3'-6"	2.56E-01	9.09E-02	4.23E-02
3'-8"	2.37E-01	8.32E-02	3.80E-02
3'-10"	2.19E-01	7.61E-02	3.43E-02
4'-0"	2.02E-01	6.95E-02	3.10E-02
4'-2"	1.85E-01	6.31E-02	2.80E-02
4'-4"	1.70E-01	5.75E-02	2.54E-02
4'-6"	1.56E-01	5.23E-02	2.31E-02
4'-8"	1.42E-01	4.74E-02	2.08E-02
4'-10"	1.29E-01	4.26E-02	1.88E-02
5'-0"	1.17E-01	3.85E-02	1.71E-02
5'-6"	8.79E-02	2.78E-02	1.26E-02
6'-0"	6.68E-02	2.00E-02	9.27E-03
6'-6"	5.16E-02	1.44E-02	6.84E-03
7'-0"	4.12E-02	1.03E-02	5.05E-03

TABLE C-13

SINGLE CONSOLIDATED BWR ASSEMBLY, 20 GWD BURNUP - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.52E-01	3.46E-01	2.58E-01
1'-0"	5.08E-01	2.89E-01	2.05E-01
1'-6"	4.68E-01	2.35E-01	1.54E-01
2'-0"	4.30E-01	1.91E-01	1.12E-01
2'-6"	3.95E-01	1.58E-01	8.32E-02
3'-0"	3.58E-01	1.35E-01	6.45E-02
3'-2"	3.43E-01	1.27E-01	5.96E-02
3'-4"	3.27E-01	1.21E-01	5.56E-02
3'-6"	3.12E-01	1.14E-01	5.16E-02
3'-8"	2.96E-01	1.08E-01	4.83E-02
3'-10"	2.80E-01	1.02E-01	4.51E-02
4'-0"	2.64E-01	9.53E-02	4.21E-02
4'-2"	2.46E-01	8.87E-02	3.90E-02
4'-4"	2.30E-01	8.27E-02	3.63E-02
4'-6"	2.15E-01	7.67E-02	3.38E-02
4'-8"	1.98E-01	7.08E-02	3.12E-02
4'-10"	1.83E-01	6.51E-02	2.88E-02
5'-0"	1.68E-01	5.98E-02	2.65E-02
5'-6"	1.30E-01	4.53E-02	2.03E-02
6'-0"	9.98E-02	3.37E-02	1.54E-03
6'-6"	7.73E-02	2.49E-02	1.16E-03
7'-0"	6.10E-02	1.81E-02	8.66E-03

TABLE C-14

SINGLE CONSOLIDATED BWR ASSEMBLY, 20 GWD BURNUP - SLAB RATIOS

Slab Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.57E-01	3.53E-01	2.65E-01
1'-0"	5.18E-01	3.00E-01	2.16E-01
1'-6"	4.76E-01	2.46E-01	1.64E-01
2'-0"	4.39E-01	1.99E-01	1.20E-01
2'-6"	4.03E-01	1.64E-01	8.85E-02
3'-0"	3.68E-01	1.40E-01	6.79E-02
3'-2"	3.53E-01	1.32E-01	6.29E-02
3'-4"	3.39E-01	1.26E-01	5.85E-02
3'-6"	3.25E-01	1.19E-01	5.46E-02
3'-8"	3.10E-01	1.13E-01	5.10E-02
3'-10"	2.94E-01	1.07E-01	4.76E-02
4'-0"	2.77E-01	1.00E-01	4.46E-02
4'-2"	2.61E-01	9.39E-02	4.15E-02
4'-4"	2.44E-01	8.82E-02	3.88E-02
4'-6"	2.28E-01	8.18E-02	3.60E-02
4'-8"	2.12E-01	7.57E-02	3.34E-02
4'-10"	1.96E-01	6.97E-02	3.08E-02
5'-0"	1.81E-01	6.47E-02	2.85E-02
5'-6"	1.40E-01	4.92E-02	2.20E-02
6'-0"	1.08E-01	3.68E-02	1.67E-02
6'-6"	8.30E-02	2.71E-02	1.25E-02
7'-0"	6.52E-02	1.99E-02	9.43E-03

TABLE C-15

SINGLE CONSOLIDATED BWR ASSEMBLY, 27.5 GWD BURNUP - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.26E-01	3.17E-01	2.33E-01
1'-0"	4.91E-01	2.68E-01	1.87E-01
1'-6"	4.62E-01	2.24E-01	1.43E-01
2'-0"	4.36E-01	1.88E-01	1.08E-01
2'-6"	4.14E-01	1.65E-01	8.48E-02
3'-0"	3.89E-01	1.48E-01	6.96E-02
3'-2"	3.77E-01	1.42E-01	6.57E-02
3'-4"	3.66E-01	1.37E-01	6.23E-02
3'-6"	3.53E-01	1.32E-01	5.92E-02
3'-8"	3.41E-01	1.26E-01	5.64E-02
3'-10"	3.25E-01	1.20E-01	5.34E-02
4'-0"	3.10E-01	1.15E-01	5.06E-02
4'-2"	2.95E-01	1.09E-01	4.82E-02
4'-4"	2.79E-01	1.03E-01	4.53E-02
4'-6"	2.62E-01	9.68E-02	4.26E-02
4'-8"	2.46E-01	9.05E-02	4.00E-02
4'-10"	2.30E-01	8.47E-02	3.74E-02
5'-0"	2.14E-01	7.87E-02	3.48E-02
5'-6"	1.69E-01	6.14E-02	2.76E-02
6'-0"	1.31E-01	4.69E-02	2.13E-02
6'-6"	1.02E-01	3.55E-02	1.64E-02
7'-0"	8.07E-02	2.65E-02	1.25E-02

TABLE C-16

SINGLE CONSOLIDATED BWR ASSEMBLY, 27.5 GWD BURNUP - SLAB RATIOS

<u>Slab Concrete Thickness</u>	<u>10yr/5yr Dose Rate Ratio</u>	<u>20yr/5yr Dose Rate Ratio</u>	<u>30yr/5yr Dose Rate Ratio</u>
0'-6"	5.33E-01	3.25E-01	2.41E-01
1'-0"	5.00E-01	2.79E-01	1.97E-01
1'-6"	4.67E-01	2.34E-01	1.52E-01
2'-0"	4.43E-01	1.95E-01	1.14E-01
2'-6"	4.21E-01	1.69E-01	8.90E-02
3'-0"	3.96E-01	1.51E-01	7.24E-02
3'-2"	3.86E-01	1.46E-01	6.88E-02
3'-4"	3.75E-01	1.40E-01	6.48E-02
3'-6"	3.65E-01	1.36E-01	6.16E-02
3'-8"	3.51E-01	1.30E-01	5.85E-02
3'-10"	3.39E-01	1.25E-01	5.59E-02
4'-0"	3.23E-01	1.20E-01	5.29E-02
4'-2"	3.10E-01	1.14E-01	5.03E-02
4'-4"	2.92E-01	1.08E-01	4.74E-02
4'-6"	2.75E-01	1.02E-01	4.48E-02
4'-8"	2.61E-01	9.59E-02	4.22E-02
4'-10"	2.43E-01	8.99E-02	3.97E-02
5'-0"	2.27E-01	8.36E-02	3.70E-02
5'-6"	1.81E-01	6.63E-02	2.95E-02
6'-0"	1.41E-01	5.09E-02	2.30E-02
6'-6"	1.10E-01	3.86E-02	1.78E-02
7'-0"	8.64E-02	2.88E-02	1.36E-02

TABLE C-17

SINGLE CONSOLIDATED BWR ASSEMBLY, 40 GWD BURNUP - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.14E-01	3.04E-01	2.22E-01
1'-0"	4.86E-01	2.60E-01	1.80E-01
1'-6"	4.61E-01	2.20E-01	1.39E-01
2'-0"	4.43E-01	1.90E-01	1.08E-01
2'-6"	4.31E-01	1.73E-01	8.85E-02
3'-0"	4.15E-01	1.60E-01	7.62E-02
3'-2"	4.09E-01	1.56E-01	7.31E-02
3'-4"	4.00E-01	1.53E-01	7.02E-02
3'-6"	3.90E-01	1.48E-01	6.76E-02
3'-8"	3.80E-01	1.44E-01	6.54E-02
3'-10"	3.69E-01	1.40E-01	6.29E-02
4'-0"	3.55E-01	1.35E-01	6.06E-02
4'-2"	3.42E-01	1.30E-01	5.80E-02
4'-4"	3.28E-01	1.25E-01	5.56E-02
4'-6"	3.13E-01	1.19E-01	5.31E-02
4'-8"	2.97E-01	1.13E-01	5.06E-02
4'-10"	2.81E-01	1.07E-01	4.80E-02
5'-0"	2.64E-01	1.01E-01	4.53E-02
5'-6"	2.16E-01	8.23E-02	3.72E-02
6'-0"	1.72E-01	6.52E-02	2.99E-02
6'-6"	1.36E-01	5.06E-02	2.36E-02
7'-0"	1.08E-01	3.90E-02	1.87E-02

TABLE C-18

SINGLE CONSOLIDATED BWR ASSEMBLY, 40 GWD BURNUP - SLAB RATIOS

Slab Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	5.22E-01	3.13E-01	2.30E-01
1'-0"	4.90E-01	2.69E-01	1.87E-01
1'-6"	4.64E-01	2.27E-01	1.47E-01
2'-0"	4.46E-01	1.96E-01	1.14E-01
2'-6"	4.34E-01	1.76E-01	9.18E-02
3'-0"	4.20E-01	1.63E-01	7.84E-02
3'-2"	4.14E-01	1.59E-01	7.49E-02
3'-4"	4.08E-01	1.56E-01	7.24E-02
3'-6"	3.99E-01	1.52E-01	6.96E-02
3'-8"	3.89E-01	1.48E-01	6.72E-02
3'-10"	3.78E-01	1.44E-01	6.46E-02
4'-0"	3.67E-01	1.39E-01	6.25E-02
4'-2"	3.54E-01	1.35E-01	6.02E-02
4'-4"	3.40E-01	1.30E-01	5.79E-02
4'-6"	3.25E-01	1.24E-01	5.51E-02
4'-8"	3.10E-01	1.18E-01	5.28E-02
4'-10"	2.94E-01	1.12E-01	5.01E-02
5'-0"	2.79E-01	1.07E-01	4.76E-02
5'-6"	2.30E-01	8.80E-02	3.96E-02
6'-0"	1.84E-01	7.01E-02	3.21E-02
6'-6"	1.46E-01	5.49E-02	2.54E-02
7'-0"	1.15E-01	4.22E-02	2.00E-02

TABLE C-19

HANFORD SITE DEFENSE HLW CONTAINER - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	8.74E-01	6.88E-01	5.44E-01
1'-0"	8.69E-01	6.79E-01	5.36E-01
1'-6"	8.55E-01	6.63E-01	5.21E-01
2'-0"	8.33E-01	6.33E-01	4.93E-01
2'-6"	7.98E-01	5.87E-01	4.49E-01
3'-0"	7.46E-01	5.25E-01	3.90E-01
3'-2"	7.28E-01	5.03E-01	3.70E-01
3'-4"	7.07E-01	4.80E-01	3.49E-01
3'-6"	6.88E-01	4.61E-01	3.32E-01
3'-8"	6.66E-01	4.40E-01	3.14E-01
3'-10"	6.46E-01	4.21E-01	2.99E-01
4'-0"	6.25E-01	4.05E-01	2.86E-01
4'-2"	6.07E-01	3.90E-01	2.74E-01
4'-4"	5.87E-01	3.75E-01	2.62E-01
4'-6"	5.69E-01	3.62E-01	2.53E-01
4'-8"	5.50E-01	3.50E-01	2.43E-01
4'-10"	5.33E-01	3.37E-01	2.34E-01
5'-0"	5.12E-01	3.24E-01	2.25E-01
5'-6"	4.56E-01	2.89E-01	2.01E-01
6'-0"	3.96E-01	2.51E-01	1.76E-01
6'-6"	3.38E-01	2.14E-01	1.51E-01
7'-0"	2.84E-01	1.80E-01	1.28E-01

TABLE C-20

HANFORD SITE DEFENSE HLW CONTAINER - SLAB RATIOS

Slab Concrete <u>Thickness</u>	10yr/5yr Dose Rate <u>Ratio</u>	20yr/5yr Dose Rate <u>Ratio</u>	30yr/5yr Dose Rate <u>Ratio</u>
0'-6"	8.73E-01	6.87E-01	5.44E-01
1'-0"	8.69E-01	6.80E-01	5.38E-01
1'-6"	8.58E-01	6.65E-01	5.24E-01
2'-0"	8.38E-01	6.39E-01	5.00E-01
2'-6"	8.10E-01	5.97E-01	4.60E-01
3'-0"	7.58E-01	5.40E-01	4.03E-01
3'-2"	7.41E-01	5.17E-01	3.83E-01
3'-4"	7.23E-01	4.96E-01	3.62E-01
3'-6"	7.03E-01	4.75E-01	3.45E-01
3'-8"	6.83E-01	4.55E-01	3.27E-01
3'-10"	6.62E-01	4.35E-01	3.10E-01
4'-0"	6.43E-01	4.17E-01	2.96E-01
4'-2"	6.25E-01	4.03E-01	2.83E-01
4'-4"	6.05E-01	3.87E-01	2.71E-01
4'-6"	5.86E-01	3.73E-01	2.61E-01
4'-8"	5.69E-01	3.61E-01	2.51E-01
4'-10"	5.50E-01	3.49E-01	2.43E-01
5'-0"	5.29E-01	3.34E-01	2.32E-01
5'-6"	4.72E-01	2.99E-01	2.08E-01
6'-0"	4.14E-01	2.63E-01	1.84E-01
6'-6"	3.55E-01	2.26E-01	1.59E-01
7'-0"	3.00E-01	1.90E-01	1.35E-01

TABLE C-21

INEL DEFENSE HLW CONTAINER - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	7.78E-01	5.87E-01	4.63E-01
1'-0"	7.46E-01	5.54E-01	4.33E-01
1'-6"	7.04E-01	5.05E-01	3.90E-01
2'-0"	6.51E-01	4.43E-01	3.34E-01
2'-6"	5.95E-01	3.78E-01	2.74E-01
3'-0"	5.45E-01	3.22E-01	2.22E-01
3'-2"	5.30E-01	3.05E-01	2.09E-01
3'-4"	5.14E-01	2.92E-01	1.96E-01
3'-6"	5.02E-01	2.80E-01	1.86E-01
3'-8"	4.90E-01	2.70E-01	1.76E-01
3'-10"	4.75E-01	2.60E-01	1.69E-01
4'-0"	4.64E-01	2.51E-01	1.62E-01
4'-2"	4.51E-01	2.44E-01	1.56E-01
4'-4"	4.36E-01	2.37E-01	1.50E-01
4'-6"	4.22E-01	2.28E-01	1.45E-01
4'-8"	4.10E-01	2.21E-01	1.40E-01
4'-10"	3.94E-01	2.13E-01	1.34E-01
5'-0"	3.80E-01	2.05E-01	1.30E-01
5'-6"	3.31E-01	1.82E-01	1.15E-01
6'-0"	2.85E-01	1.57E-01	1.00E-01
6'-6"	2.39E-01	1.32E-01	8.56E-02
7'-0"	2.00E-01	1.11E-01	7.19E-02

TABLE C-22

INEL DEFENSE HLW CONTAINER - SLAB RATIOS

Slab Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	7.78E-01	5.91E-01	4.65E-01
1'-0"	7.50E-01	5.56E-01	4.36E-01
1'-6"	7.13E-01	5.14E-01	3.98E-01
2'-0"	6.60E-01	4.54E-01	3.45E-01
2'-6"	6.06E-01	3.90E-01	2.86E-01
3'-0"	5.55E-01	3.32E-01	2.32E-01
3'-2"	5.38E-01	3.16E-01	2.17E-01
3'-4"	5.25E-01	3.01E-01	2.04E-01
3'-6"	5.13E-01	2.88E-01	1.93E-01
3'-8"	4.99E-01	2.78E-01	1.83E-01
3'-10"	4.86E-01	2.67E-01	1.74E-01
4'-0"	4.76E-01	2.58E-01	1.68E-01
4'-2"	4.62E-01	2.50E-01	1.60E-01
4'-4"	4.50E-01	2.42E-01	1.55E-01
4'-6"	4.36E-01	2.35E-01	1.50E-01
4'-8"	4.22E-01	2.28E-01	1.44E-01
4'-10"	4.08E-01	2.21E-01	1.39E-01
5'-0"	3.93E-01	2.12E-01	1.34E-01
5'-6"	3.46E-01	1.89E-01	1.20E-01
6'-0"	2.99E-01	1.64E-01	1.05E-01
6'-6"	2.52E-01	1.39E-01	9.00E-02
7'-0"	2.11E-01	1.17E-01	7.58E-02

TABLE C-23

SRP DEFENSE WASTE CONTAINER - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	8.69E-01	6.76E-01	5.30E-01
1'-0"	8.54E-01	6.51E-01	5.08E-01
1'-6"	8.28E-01	6.08E-01	4.66E-01
2'-0"	7.92E-01	5.47E-01	4.07E-01
2'-6"	7.38E-01	4.74E-01	3.35E-01
3'-0"	6.92E-01	4.07E-01	2.71E-01
3'-2"	6.75E-01	3.87E-01	2.53E-01
3'-4"	6.64E-01	3.72E-01	2.39E-01
3'-6"	6.54E-01	3.60E-01	2.26E-01
3'-8"	6.40E-01	3.47E-01	2.16E-01
3'-10"	6.30E-01	3.38E-01	2.08E-01
4'-0"	6.21E-01	3.31E-01	2.02E-01
4'-2"	6.13E-01	3.23E-01	1.96E-01
4'-4"	6.00E-01	3.17E-01	1.92E-01
4'-6"	5.90E-01	3.11E-01	1.89E-01
4'-8"	5.80E-01	3.05E-01	1.85E-01
4'-10"	5.71E-01	3.02E-01	1.83E-01
5'-0"	5.56E-01	2.95E-01	1.80E-01
5'-6"	5.20E-01	2.80E-01	1.73E-01
6'-0"	4.76E-01	2.62E-01	1.65E-01
6'-6"	4.29E-01	2.41E-01	1.54E-01
7'-0"	3.80E-01	2.18E-01	1.43E-01

TABLE C-24

SRP DEFENSE WASTE CONTAINER - SLAB RATIOS

<u>Slab Concrete Thickness</u>	<u>10yr/5yr Dose Rate Ratio</u>	<u>20yr/5yr Dose Rate Ratio</u>	<u>30yr/5yr Dose Rate Ratio</u>
0'-6"	8.69E-01	6.75E-01	5.30E-01
1'-0"	8.56E-01	6.55E-01	5.11E-01
1'-6"	8.33E-01	6.18E-01	4.74E-01
2'-0"	7.95E-01	5.58E-01	4.18E-01
2'-6"	7.48E-01	4.87E-01	3.47E-01
3'-0"	7.03E-01	4.20E-01	2.83E-01
3'-2"	6.86E-01	4.00E-01	2.64E-01
3'-4"	6.74E-01	3.82E-01	2.49E-01
3'-6"	6.59E-01	3.67E-01	2.34E-01
3'-8"	6.51E-01	3.56E-01	2.24E-01
3'-10"	6.38E-01	3.44E-01	2.14E-01
4'-0"	6.28E-01	3.36E-01	2.06E-01
4'-2"	6.18E-01	3.28E-01	2.00E-01
4'-4"	6.10E-01	3.22E-01	1.95E-01
4'-6"	6.00E-01	3.16E-01	1.91E-01
4'-8"	5.91E-01	3.11E-01	1.88E-01
4'-10"	5.78E-01	3.05E-01	1.85E-01
5'-0"	5.68E-01	3.01E-01	1.83E-01
5'-6"	5.30E-01	2.84E-01	1.75E-01
6'-0"	4.89E-01	2.68E-01	1.67E-01
6'-6"	4.44E-01	2.48E-01	1.58E-01
7'-0"	3.95E-01	2.26E-01	1.47E-01

TABLE C-25

WEST VALLEY WASTE CONTAINER - WALL RATIOS

Wall Concrete Thickness	10yr/5yr Dose Rate Ratio	20yr/5yr Dose Rate Ratio	30yr/5yr Dose Rate Ratio
0'-6"	8.82E-01	6.93E-01	5.47E-01
1'-0"	8.76E-01	6.82E-01	5.35E-01
1'-6"	8.67E-01	6.62E-01	5.14E-01
2'-0"	8.46E-01	6.23E-01	4.73E-01
2'-6"	8.16E-01	5.72E-01	4.20E-01
3'-0"	7.86E-01	5.14E-01	3.60E-01
3'-2"	7.76E-01	4.98E-01	3.42E-01
3'-4"	7.67E-01	4.83E-01	3.28E-01
3'-6"	7.60E-01	4.69E-01	3.12E-01
3'-8"	7.55E-01	4.58E-01	3.01E-01
3'-10"	7.49E-01	4.51E-01	2.92E-01
4'-0"	7.46E-01	4.42E-01	2.84E-01
4'-2"	7.44E-01	4.38E-01	2.80E-01
4'-4"	7.42E-01	4.34E-01	2.76E-01
4'-6"	7.40E-01	4.33E-01	2.74E-01
4'-8"	7.41E-01	4.30E-01	2.72E-01
4'-10"	7.41E-01	4.31E-01	2.73E-01
5'-0"	7.40E-01	4.31E-01	2.72E-01
5'-6"	7.42E-01	4.36E-01	2.78E-01
6'-0"	7.46E-01	4.44E-01	2.86E-01
6'-6"	7.50E-01	4.53E-01	2.96E-01
7'-0"	7.58E-01	4.65E-01	3.09E-01

TABLE C-26

WEST VALLEY WASTE CONTAINER - SLAB RATIOS

<u>Slab Concrete Thickness</u>	<u>10yr/5yr Dose Rate Ratio</u>	<u>20yr/5yr Dose Rate Ratio</u>	<u>30yr/5yr Dose Rate Ratio</u>
0'-6"	8.84E-01	6.93E-01	5.47E-01
1'-0"	8.76E-01	6.82E-01	5.33E-01
1'-6"	8.66E-01	6.62E-01	5.14E-01
2'-0"	8.43E-01	6.27E-01	4.78E-01
2'-6"	8.23E-01	5.82E-01	4.30E-01
3'-0"	7.88E-01	5.23E-01	3.70E-01
3'-2"	7.81E-01	5.08E-01	3.53E-01
3'-4"	7.72E-01	4.92E-01	3.35E-01
3'-6"	7.64E-01	4.76E-01	3.20E-01
3'-8"	7.58E-01	4.65E-01	3.08E-01
3'-10"	7.53E-01	4.54E-01	2.97E-01
4'-0"	7.50E-01	4.48E-01	2.90E-01
4'-2"	7.45E-01	4.40E-01	2.82E-01
4'-4"	7.43E-01	4.36E-01	2.78E-01
4'-6"	7.43E-01	4.33E-01	2.75E-01
4'-8"	7.39E-01	4.31E-01	2.73E-01
4'-10"	7.40E-01	4.30E-01	2.72E-01
5'-0"	7.38E-01	4.31E-01	2.72E-01
5'-6"	7.42E-01	4.34E-01	2.76E-01
6'-0"	7.45E-01	4.41E-01	2.83E-01
6'-6"	7.44E-01	4.49E-01	2.93E-01
7'-0"	7.57E-01	4.60E-01	3.06E-01

APPENDIX D

NEUTRON DOSE RATE FIGURES

ILLUSTRATIONS

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Dose Rates in a Concrete Surface Storage Vault Due to
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Contribution of Neutrons and Capture Gammas to Total Dose
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Figure D-22

Contribution of Neutrons and Capture Gammas to Total Dose
as a Function of Burnup, 20 Years Decay

APPENDIX E

Information from the Reference Information Base Used in this Report

This report contains no information from the Reference Information Base.

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