OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
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Revision History

10. Revision No.	11. Description of Revision
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FIGURES

TABLES

ACRONYMS AND ABBREVIATIONS

ANSI	American National Standards Institute
ANS	American Nuclear Society
BWR	Boiling Water Reactor
B&W	Babcock & Wilcox
B-poly	Borated polyethylene
cc	cubic centimeters
CFR	Code of Federal Regulations
cm	centimeters
CPU	Central Processing Unit
CRM	Corrosion Resistant Material
CRWMS	Civilian Radioactive Waste Management System
CSCI	Computer Software Configuration Item
DHLW	Defense High-Level Waste
DIRS	Document Input Reference System
DOE	U.S Department of Energy
FFTF	Fast Flux Test Facility
FY	Fiscal Year
g	grams
GWd	Gigawatt Days
hr	hours
ID	Identification
kg	kilograms
kW	kilowatts
MCNP	Monte Carlo N-Particle transport code
MeV	Million Electron Volts
MOX	Mixed Oxide
MTU	Metric Tons of Uranium
M&O	Management and Operating Contractor
n	neutrons
NRC	U.S. Nuclear Regulatory Commission
PC	Personal Computer
PWR	Pressurized Water Reactor

rad	A radiation dose unit with 1 rad = 100 ergs/g
rem	Roentgen Equivalent in Man
s or sec	seconds
SNF	Spent Nuclear Fuel
SRP	Savannah River Plant
SS	Stainless Steel
TBV	To Be Verified
UCF	Uncanistered Fuel
VA	Viability Assessment
W	Watts
WP	Waste Package
WPD	Waste Package Department
yr	year
γ	Gammas or photons

1. PURPOSE

The purpose of this calculation is to establish appropriate and defensible waste-package radiation source terms for use in repository subsurface shielding design. This calculation supports the shielding design for the waste emplacement and retrieval system, and subsurface facility system. The objective is to identify the limiting waste package and specify its associated source terms including source strengths and energy spectra. Consistent with the *Technical Work Plan for Subsurface Design Section FY 01 Work Activities* (CRWMS M&O 2001, p. 15), the scope of work includes the following:

- Review source terms generated by the Waste Package Department (WPD) for various waste forms and waste package types, and compile them for shielding-specific applications.
- Determine acceptable waste package specific source terms for use in subsurface shielding design, using a reasonable and defensible methodology that is not unduly conservative.

This calculation is associated with the engineering and design activity for the waste emplacement and retrieval system, and subsurface facility system. The technical work plan for this calculation is provided in CRWMS M&O 2001. Development and performance of this calculation conforms to the procedure, AP-3.12Q, *Calculations*.

2. METHOD

The following step-by-step approach is used to determine the waste package specific source terms for subsurface shielding applications:

a. Review and compile the source terms generated by the WPD for the various waste forms including pressurized water reactor (PWR) spent nuclear fuel (SNF), boiling water reactor (BWR) SNF, defense high-level waste (DHLW), naval reactor SNF, and U.S. Department of Energy (DOE) SNF.

b. Identify the limiting waste package by comparing the dose rates on the waste package (WP) surfaces calculated by the WPD for the different types of waste packages.

c. Review the *1999 Design Basis Waste Input Report for Commercial Spent Nuclear Fuel* (CRWMS M&O 1999a) on the projected waste streams and fuel assembly heat distributions for selection of the design basis assembly heat and fuel characteristics (initial enrichment, burnup and cooling time).

d. Review the WPD calculation document on *Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams* (CRWMS M&O 2000h) to identify the peak heat fuel assembly in the waste packages received by year. This information is used to determine reasonableness of the design basis fuel selected from Step c on the basis of the assembly heat distribution.

e. Determine the relative contribution from each individual fuel assembly to the dose rate on the WP surface and transporter surface for evaluation and selection of the fuel loading pattern. The determination considers the gamma contribution only, which is much more significant than the neutron contribution (CRWMS M&O 2000c, p.28). The PATH gamma shielding code (Su et al. 1987) is used for this step of the calculation.

f. Develop and select the geometric model for the design basis fuel, considering the explicit and homogeneous models for the selected fuel loading pattern. Again, this step considers the principal dose component only (i.e., from gamma radiation), and uses the PATH code.

g. Perform the dose rate calculations for the design basis fuel with the MCNP code (Briesmeister 1997), using the geometric model developed from Step f. Because of the ability of treating detailed particle transport physics, use of the MCNP code produces more accurate results. PATH is useful mainly for scoping or prototypic calculations.

No method of controlling electronic management of data for this calculation was specified in the technical work plan (CRWMS M&O 2001). The computer input and output files generated from this calculation are stored on an electronic medium, and submitted as an attachment to this document in accordance with AP-17.1Q.

3. ASSUMPTIONS

The following assumptions are used in this calculation.

3.1 For the smeared material calculation, the active fuel region of each SNF assembly contains fresh, unirradiated fuel. This assumption applies to the material composition. By no mean does it apply to the source term, which is based on spent fuel.

Rationale: This assumption is conservative since it leads to slightly higher dose rates. The higher percentage of fissile ^{235}U in each assembly results in a greater number of induced fission neutrons, and consequently a stronger radiation field. No further confirmation of this assumption is required.

Usage: This assumption is used in Section 5.6.1.

3.2 Uniform fuel burnup is assumed.

Rationale: The axial fuel burnup profile is not considered here, as the purpose of this calculation is to determine waste package specific source terms rather than shielding requirements. This profile needs to be incorporated into subsequent shielding design in the future. No further confirmation of this assumption is required insofar as this calculation is concerned.

Usage: This assumption applies to the source terms provided in Section 5.5.2.

3.3 Internal fuel basket materials are ignored for purposes of smeared material composition.

Rationale: Inclusion of fuel basket materials in the smeared material composition would increase the overall density and, consequently, increase the attenuation of the radiation. Therefore, this assumption is conservative. No further confirmation of this assumption is required.

Usage: This assumption is used in Section 5.6.2.2.

3.4 For the point-kernel integration approximation with the PATH code, iron (Fe) is assumed to simulate the following materials: Inconel-718, A516 carbon steel, SS316L stainless steel and Alloy 22. In addition, the chemical element, Mo is assumed to represent Zr-4 in the fuel region.

Rationale: PATH performs gamma calculations only. Gamma attenuation through a medium is mainly affected by the material density (g/cm^3), and its mass attenuation coefficients (cm^2/g). PATH calculations use the actual density for each material. However, the mass attenuation coefficients for iron are used for Inconel-718, A516 carbon steel, SS316L stainless steel and Alloy 22, as the attenuation coefficients are fairly insensitive to elements with similar or close atomic numbers (ANSI/ANS-6.4.3-1991, p. 7). The same rationale applies to replacement of Zr-4 with Mo (ANSI/ANS-6.4.3-1991, p. 7).

These replacements are necessary to facilitate the calculation, because of the lack of the attenuation coefficients for certain elements in the data library. Since PATH is used in this calculation for scoping purposes only, no further confirmation of this assumption is required.

Usage: This assumption is used in Section 5.6.2.1.

3.5 For shielding modeling purposes, the waste package and its transporter are both centered in the main drift.

Rationale: This assumption simplifies the waste package and transporter model, and facilitates tallies of dose rate results on the surfaces for improved statistics in the Monte Carlo calculation. This assumption poses no significant effect on the shielding results. Therefore, no further confirmation of this assumption is required.

Usage: This assumption is used in Sections 5.6.2 and 5.7.

3.6 For MCNP calculations, the axial length of the shielding model is assumed to be infinite.

Rationale: The axial length of the source region (i.e., fuel assembly) is sufficiently long, relative to the distance from the source to the dose points of interest. Furthermore, an infinite model results in slightly higher dose rates than a finite model. Therefore, this assumption is conservative, and no further confirmation of this assumption is required.

Usage: This assumption is used in Section 5.7.

3.7 For dose rate calculations, all the fuel assemblies in the waste package are assumed to be of the same characteristics with identical source terms.

Rationale: This assumption simplifies the analytical model and enables homogenization of the fuel region. Because of the self-shielding effects by fuel assemblies, only the outer row of fuel assemblies in the fuel basket make significant contributions to the dose rates external to the waste package (demonstrated in Table 16). This assumption represents a common practice used in this type of shielding calculation. Therefore, no further confirmation of this assumption is required.

Usage: This assumption is used in Section 5.6.

4. USE OF COMPUTER SOFTWARE AND MODELS

4.1 BASELINE SOFTWARE

The following baseline software items were obtained from Software Configuration Management in accordance with appropriate procedures. These items were appropriate for use in this calculation, according to the applications and capabilities of these codes. They were used within the range of validation. The input and output files are listed in Attachment I, and stored in ASCII format on an electronic medium in accordance with AP-17.1Q.

4.1.1 MCNP

The MCNP code as identified below was used to calculate both neutron and gamma dose rates on the WP surface and transporter surface.

Program name: MCNP (CRWMS M&O 1998b)
Version/revision number: Version 4B2LV
CSCI number: 30033, 4B2LV
Computer type: Desktop Pentium PC (CPU #112111)

4.1.2 PATH

The PATH code as identified below was used to calculate the gamma dose rates only on the WP surface and transporter surface.

Program name: PATH (CRWMS M&O 1996a)
Version/revision number: Version 88A
CSCI number: 30007, 88A
Computer type: Desktop Pentium PC (CPU #112111)

4.2 SOFTWARE ROUTINES

The following Excel spreadsheet program was used to perform simple arithmetic calculations, as indicated in Section 6.2.1 of this document. The user-defined formulas, inputs, and results are documented in sufficient detail to allow independent repetition of the various calculations.

Title: Excel
Version/revision number: Microsoft Excel 97

4.3 MODELS

No models of natural systems or processes were used in this calculation.

5. CALCULATION

5.1 REVIEW AND COMPILATION OF SOURCE TERMS

This section reviews, compiles and discusses the source terms generated by the WPD for the various waste forms. The focus here is on the source terms for shielding-specific applications.

5.1.1 PWR SNF

The source terms for PWR SNF are provided in *PWR Source Term Generation and Evaluation* (CRWMS M&O 1999d) for the various combinations of initial enrichment, burnup and cooling time. Of particular interest to shielding applications are the source terms for the maximum and average PWR fuel assemblies, respectively, as specified below:

Maximum PWR assembly:	5.0%, 75 GWd/MTU and 5 years old (CRWMS M&O 1999d, p.24)
Average PWR assembly:	4.0%, 48 GWd/MTU and 25 years old (CRWMS M&O 1999d, p. 24)

The corresponding heat generation rates are given below:

Maximum PWR assembly:	2.266 kW/assembly (CRWMS M&O 1999d, Attachment IV)
Average PWR assembly:	0.601 kW/assembly (CRWMS M&O 1999d, Attachment IV)

Table 1 lists the neutron source terms by energy group for the maximum and average PWR fuel assembly, obtained from CRWMS M&O 1999d. Table 2 provides the same type of information for the gamma source terms. Note that the neutron source terms are in the fuel region only, and the gamma source terms include the four different regions of the fuel assembly: fuel, bottom, plenum and top.

Note that the maximum and average fuel assemblies described in this section represent the specifications defined by the WPD in CRWMS M&O 1999d for use in waste package radiation analysis. These assemblies may not correspond to the expected maximum and average fuel assemblies based on the projected waste stream scenarios (see Section 5.5.1).

Table 1. Maximum and Average PWR SNF Neutron Source Terms

Neutron Energy Range (MeV)		Neutron Source (n/s per assembly)	
Upper Bound	Lower Bound	Maximum[a]	Average[b]
2.00E+01	6.43E+00	3.06E+07	3.65E+06
6.43E+00	3.00E+00	3.45E+08	4.18E+07
3.00E+00	1.85E+00	3.78E+08	4.69E+07
1.85E+00	1.40E+00	2.15E+08	2.61E+07
1.40E+00	9.00E-01	2.92E+08	3.52E+07
9.00E-01	4.00E-01	3.19E+08	3.82E+07
4.00E-01	1.00E-01	6.25E+07	7.48E+06
1.00E-01	1.70E-02	0.00E+00	0.00E+00
1.70E-02	3.00E-03	0.00E+00	0.00E+00
3.00E-03	5.50E-04	0.00E+00	0.00E+00
5.50E-04	1.00E-04	0.00E+00	0.00E+00
1.00E-04	3.00E-05	0.00E+00	0.00E+00
3.00E-05	1.00E-05	0.00E+00	0.00E+00
1.00E-05	3.05E-06	0.00E+00	0.00E+00
3.05E-06	1.77E-06	0.00E+00	0.00E+00
1.77E-06	1.30E-06	0.00E+00	0.00E+00
1.30E-06	1.13E-06	0.00E+00	0.00E+00
1.13E-06	1.00E-06	0.00E+00	0.00E+00
1.00E-06	8.00E-07	0.00E+00	0.00E+00
8.00E-07	4.00E-07	0.00E+00	0.00E+00
4.00E-07	3.25E-07	0.00E+00	0.00E+00
3.25E-07	2.25E-07	0.00E+00	0.00E+00
2.25E-07	1.00E-07	0.00E+00	0.00E+00
1.00E-07	5.00E-08	0.00E+00	0.00E+00
5.00E-08	3.00E-08	0.00E+00	0.00E+00
3.00E-08	1.00E-08	0.00E+00	0.00E+00
1.00E-08	1.00E-11	0.00E+00	0.00E+00

[a] Source: CRWMS M&O 1999d, Attachment IV, File *PWR.neutron.source* for PWR fuel with 5% enrichment, 75 GWd/MTU burnup and 5 yr cooling.

[b] Source: CRWMS M&O 1999d, Attachment IV, File *PWR.neutron.source* for PWR fuel with 4% enrichment, 48 GWd/MTU burnup and 25 yr cooling.

Table 2. Maximum and Average PWR SNF Gamma Source Terms

Gamma Energy Range (MeV)		Maximum Source (γ's/s per assembly)[a]				Average Source (γ's/s per assembly)[b]			
		Fuel	Bottom	Plenum	Top	Fuel	Bottom	Plenum	Top
5.00E-02	1.00E-02	2.23E+15	5.60E+11	5.01E+11	3.58E+11	6.70E+14	3.36E+10	1.86E+10	2.17E+10
1.00E-01	5.00E-02	6.20E+14	1.09E+11	5.76E+10	7.01E+10	1.99E+14	6.02E+09	3.14E+09	3.87E+09
2.00E-01	1.00E-01	5.03E+14	2.67E+10	3.35E+10	1.69E+10	1.26E+14	1.46E+09	8.53E+08	9.36E+08
3.00E-01	2.00E-01	1.43E+14	1.33E+09	1.86E+09	8.41E+08	3.89E+13	7.30E+07	4.39E+07	4.69E+07
4.00E-01	3.00E-01	9.52E+13	1.79E+09	5.58E+09	1.10E+09	2.63E+13	9.47E+07	7.16E+07	6.07E+07
6.00E-01	4.00E-01	1.42E+15	1.82E+09	1.05E+11	6.99E+07	2.05E+13	1.41E+07	5.04E+08	3.83E+06
8.00E-01	6.00E-01	4.40E+15	4.07E+09	5.67E+10	2.20E+09	1.24E+15	2.08E+09	1.92E+09	1.44E+09
1.00E+00	8.00E-01	6.53E+14	1.33E+11	7.67E+09	7.44E+10	1.10E+13	2.08E+09	1.64E+09	1.44E+09
1.33E+00	1.00E+00	4.29E+14	3.19E+13	1.64E+13	2.05E+13	2.95E+13	1.75E+12	9.09E+11	1.12E+12
1.66E+00	1.33E+00	1.22E+14	9.00E+12	4.64E+12	5.78E+12	5.13E+12	4.94E+11	2.57E+11	3.18E+11
2.00E+00	1.66E+00	1.39E+12	1.85E+03	8.72E+02	1.13E+03	6.75E+10	9.73E-01	6.14E+01	8.72E-03
2.50E+00	2.00E+00	2.48E+12	2.14E+08	1.10E+08	1.37E+08	3.53E+09	1.17E+07	6.09E+06	7.54E+06
3.00E+00	2.50E+00	1.05E+11	3.31E+05	1.71E+05	2.13E+05	2.88E+08	1.82E+04	9.44E+03	1.17E+04
4.00E+00	3.00E+00	1.32E+10	5.35E-08	7.00E-09	2.91E-08	1.98E+07	1.86E-11	1.49E-11	1.27E-11
5.00E+00	4.00E+00	5.54E+07	0.00E+00	0.00E+00	0.00E+00	6.69E+06	0.00E+00	0.00E+00	0.00E+00
6.50E+00	5.00E+00	2.22E+07	0.00E+00	0.00E+00	0.00E+00	2.69E+06	0.00E+00	0.00E+00	0.00E+00
8.00E+00	6.50E+00	4.36E+06	0.00E+00	0.00E+00	0.00E+00	5.27E+05	0.00E+00	0.00E+00	0.00E+00
1.00E+01	8.00E+00	9.26E+05	0.00E+00	0.00E+00	0.00E+00	1.12E+05	0.00E+00	0.00E+00	0.00E+00

[a] Source: CRWMS M&O 1999d, Attachment IV, File *PWR.gamma.source* for PWR fuel with 5% enrichment, 75 GWd/MTU burnup and 5 yr cooling.

[b] Source: CRWMS M&O 1999d, Attachment IV, File *PWR.gamma.source* for PWR fuel with 4% enrichment, 48 GWd/MTU burnup and 25 yr cooling.

5.1.2 BWR SNF

The source terms for BWR SNF are provided in *BWR Source Term Generation and Evaluation* (CRWMS M&O 1999b) for the various combinations of initial enrichment, burnup and cooling time. Of particular interest to shielding applications are the source terms for the maximum and average BWR fuel assemblies, respectively, as specified below:

Maximum BWR assembly:	5.0%, 75 GWd/MTU and 5 years old (CRWMS M&O 1999b, p.46)
Average BWR assembly:	3.5%, 40 GWd/MTU and 25 years old (CRWMS M&O 1999b, p.46)

The corresponding heat generation rates are given below:

Maximum BWR assembly:	0.78 kW/assembly (CRWMS M&O 1999b, Attachment VII)
Average BWR assembly:	0.19 kW/assembly (CRWMS M&O 1999b, Attachment VII)

Table 3 lists the neutron source terms by energy group for the maximum and average BWR fuel assembly, obtained from CRWMS M&O 1999b. Table 4 provides the same type of information for the gamma source terms. Note that the neutron source terms are in the fuel region only, and the gamma source terms include the four different regions of the fuel assembly: fuel, bottom, plenum and top.

Table 3. Maximum and Average BWR SNF Neutron Source Terms

Neutron Energy Range (MeV)		Neutron Source (n/s per assembly)	
Upper Bound	Lower Bound	Maximum[a]	Average[b]
2.00E+01	6.43E+00	1.07E+07	6.88E+05
6.43E+00	3.00E+00	1.21E+08	7.95E+06
3.00E+00	1.85E+00	1.32E+08	9.02E+06
1.85E+00	1.40E+00	7.53E+07	4.97E+06
1.40E+00	9.00E-01	1.02E+08	6.65E+06
9.00E-01	4.00E-01	1.12E+08	7.21E+06
4.00E-01	1.00E-01	2.19E+07	1.41E+06
1.00E-01	1.70E-02	0.00E+00	0.00E+00
1.70E-02	3.00E-03	0.00E+00	0.00E+00
3.00E-03	5.50E-04	0.00E+00	0.00E+00
5.50E-04	1.00E-04	0.00E+00	0.00E+00
1.00E-04	3.00E-05	0.00E+00	0.00E+00
3.00E-05	1.00E-05	0.00E+00	0.00E+00
1.00E-05	3.05E-06	0.00E+00	0.00E+00
3.05E-06	1.77E-06	0.00E+00	0.00E+00
1.77E-06	1.30E-06	0.00E+00	0.00E+00
1.30E-06	1.13E-06	0.00E+00	0.00E+00
1.13E-06	1.00E-06	0.00E+00	0.00E+00
1.00E-06	8.00E-07	0.00E+00	0.00E+00
8.00E-07	4.00E-07	0.00E+00	0.00E+00
4.00E-07	3.25E-07	0.00E+00	0.00E+00
3.25E-07	2.25E-07	0.00E+00	0.00E+00
2.25E-07	1.00E-07	0.00E+00	0.00E+00
1.00E-07	5.00E-08	0.00E+00	0.00E+00
5.00E-08	3.00E-08	0.00E+00	0.00E+00
3.00E-08	1.00E-08	0.00E+00	0.00E+00
1.00E-08	1.00E-11	0.00E+00	0.00E+00

[a] Source: CRWMS M&O 1999b, Attachment VII, File *BWR.neutron.source* for BWR fuel with 5% enrichment, 75 GWd/MTU burnup and 5 yr cooling.

[b] Source: CRWMS M&O 1999b, Attachment VII, File *BWR.neutron.source* for BWR fuel with 3.5% enrichment, 40 GWd/MTU burnup and 25 yr cooling.

Table 4. Maximum and Average BWR SNF Gamma Source Terms

Gamma Energy Range (MeV)		Maximum Source (γ/s per assembly)[a]				Average Source (γ/s per assembly)[b]			
		Fuel	Bottom	Plenum	Top	Fuel	Bottom	Plenum	Top
5.00E-02	1.00E-02	8.01E+14	3.63E+11	7.92E+11	3.80E+11	2.30E+14	3.32E+10	5.26E+10	3.41E+10
1.00E-01	5.00E-02	2.21E+14	8.51E+10	1.55E+11	8.86E+10	6.79E+13	4.02E+09	7.64E+09	4.21E+09
2.00E-01	1.00E-01	1.73E+14	1.44E+11	1.68E+11	1.45E+11	4.33E+13	2.20E+10	2.29E+10	2.20E+10
3.00E-01	2.00E-01	4.96E+13	2.37E+10	2.49E+10	2.37E+10	1.34E+13	4.24E+09	4.28E+09	4.24E+09
4.00E-01	3.00E-01	3.31E+13	2.22E+09	5.05E+09	2.27E+09	9.21E+12	3.44E+08	4.09E+08	3.47E+08
6.00E-01	4.00E-01	4.38E+14	3.06E+10	6.93E+10	3.06E+10	6.92E+12	4.95E+09	5.13E+09	4.95E+09
8.00E-01	6.00E-01	1.60E+15	9.41E+10	1.15E+11	9.41E+10	4.16E+14	1.67E+10	1.72E+10	1.68E+10
1.00E+00	8.00E-01	2.03E+14	1.28E+11	1.06E+11	1.06E+11	3.15E+12	1.76E+10	1.80E+10	1.77E+10
1.33E+00	1.00E+00	6.77E+13	4.32E+12	2.47E+13	5.34E+12	4.91E+12	2.42E+11	1.30E+12	2.97E+11
1.66E+00	1.33E+00	1.94E+13	1.18E+12	6.93E+12	1.47E+12	5.40E+11	6.09E+10	3.58E+11	7.66E+10
2.00E+00	1.66E+00	3.77E+11	1.48E+07	1.48E+07	1.48E+07	2.36E+10	2.69E+06	2.69E+06	2.69E+06
2.50E+00	2.00E+00	5.99E+11	2.78E+07	1.64E+08	3.47E+07	1.20E+09	1.41E+06	8.47E+06	1.78E+06
3.00E+00	2.50E+00	2.72E+10	4.31E+04	2.55E+05	5.37E+04	6.44E+07	2.19E+03	1.31E+04	2.76E+03
4.00E+00	3.00E+00	3.43E+09	1.47E-10	3.07E-10	1.74E-10	3.75E+06	2.00E-10	2.25E-10	2.04E-10
5.00E+00	4.00E+00	1.95E+07	3.71E-11	3.70E-11	3.70E-11	1.27E+06	5.06E-11	5.06E-11	5.06E-11
6.50E+00	5.00E+00	7.81E+06	1.07E-11	1.07E-11	1.07E-11	5.07E+05	1.46E-11	1.46E-11	1.46E-11
8.00E+00	6.50E+00	1.53E+06	1.36E-12	1.36E-12	1.36E-12	9.95E+04	1.85E-12	1.85E-12	1.85E-12
1.00E+01	8.00E+00	3.25E+05	1.81E-13	1.81E-13	1.81E-13	2.11E+04	2.47E-13	2.47E-13	2.47E-13

[a] Source: CRWMS M&O 1999b, Attachment VII, File *BWR.gamma.source* for BWR fuel with 5% enrichment, 75 GWd/MTU burnup and 5 yr cooling.

[b] Source: CRWMS M&O 1999b, Attachment VII, File *BWR.gamma.source* for BWR fuel with 3.5% enrichment, 40 GWd/MTU burnup and 25 yr cooling.

5.1.3 DHLW

The source terms from DHLW canisters are documented in CRWMS M&O 1999f, including those representative of each of four facilities: Savannah River Plant (SRP), Hanford, Idaho Engineering and Environmental Laboratory, and West Valley Project. Comparison of these source terms for different DHLW canisters indicates that the canister from the SRP represents the worst-case for shielding considerations. The SRP canister has been selected as a basis in the dose rate calculations for the DHLW and DOE SNF co-disposal waste package (CRWMS M&O 2000e).

Consistent with the basis used in CRWMS M&O 2000e, Table 5 provides the neutron and gamma sources (CRWMS M&O 1999f, pp. VIII-1 and IX-1, respectively) for the SRP DHLW canister at 1 day after pouring.

5.1.4 DOE SNF

There is a wide variety of DOE SNF such as Fast Flux Test Facility (FFTF) fuel, Fermi fuel, TRIGA (Training, Research, Isotopes, General Atomic) fuel, etc. The report on *DOE SNF Source Term Sensitivity Studies* (CRWMS M&O 1999c, p. 9) has determined that the bounding source term for the DOE SNF arises from the FFTF. The neutron and gamma source terms for the FFTF SNF are available in CRWMS M&O 1999c (p. 11).

Since fuel burnup for DOE SNF is much lower than that for commercial SNF, the DOE SNF should contain less radioactivity and lower source terms (CRWMS M&O 2000e, pp. 13 - 15). As commercial SNF is more bounding than DOE SNF, use of commercial SNF for shielding design should be conservative. Furthermore, the DOE SNF canisters will be co-disposed of with the DHLW canisters. The DHLW/DOE SNF co-disposal waste package contains a DOE SNF canister in the center surrounded by five DHLW canisters (CRWMS M&O 2000e). The DHLW canisters provide shielding for the DOE SNF canister, resulting in a relatively insignificant dose contribution from the DOE SNF canister (CRWMS M&O 2000e).

5.1.5 Naval SNF

Unlike other waste forms such as commercial and DOE SNF, the source terms for naval SNF are unavailable in units of neutrons and photons per second in the fuel region. Instead, *Thermal, Shielding, and Structural Information on the Naval Spent Nuclear Fuel (SNF) Canister* (Naples 1999, Enclosure 2, p. 2) provides the neutron and photon currents on various surfaces of the naval SNF canister at 2 years decay.

Since the minimum cooling time required for standard SNF is 5 years per 10CFR961.11 (Appendix E), CRWMS M&O 2000a (Attachment II, p. II-2) adjusted the photon currents by the group-dependent decay factors provided in Naples 1999, Enclosure 2 (p.7) to reflect the decay time of 5 years. However, no adjustment was made to the neutron current, as the neutron contribution is relatively insignificant in comparison to the photon current. Table 6 reproduces the source terms used in CRWMS M&O 2000a for dose rate calculations.

Table 5. Neutron and Gamma Sources per SRP DHLW Glass Canister at 1 Day after Pouring

Neutron Source[a]			Gamma Source[b]		
Energy Range (MeV)		n /s per canister	Energy Range (MeV)		γ/s per canister
2.00E+01	6.43E+00	3.07E+05	5.00E-02	1.00E-02	1.32E+15
6.43E+00	3.00E+00	3.42E+07	1.00E-01	5.00E-02	3.96E+14
3.00E+00	1.85E+00	2.61E+07	2.00E-01	1.00E-01	3.10E+14
1.85E+00	1.40E+00	6.12E+06	3.00E-01	2.00E-01	8.74E+13
1.40E+00	9.00E-01	6.92E+06	4.00E-01	3.00E-01	6.39E+13
9.00E-01	4.00E-01	6.34E+06	6.00E-01	4.00E-01	8.83E+13
4.00E-01	1.00E-01	1.89E+06	8.00E-01	6.00E-01	1.35E+15
1.00E-01	1.70E-02	1.97E+05	1.00E+00	8.00E-01	2.13E+13
1.70E-02	3.00E-03	0.00E+00	1.33E+00	1.00E+00	2.96E+13
3.00E-03	5.50E-04	0.00E+00	1.66E+00	1.33E+00	6.42E+12
5.50E-04	1.00E-04	0.00E+00	2.00E+00	1.66E+00	5.14E+11
1.00E-04	3.00E-05	0.00E+00	2.50E+00	2.00E+00	2.94E+12
3.00E-05	1.00E-05	0.00E+00	3.00E+00	2.50E+00	2.04E+10
1.00E-05	3.05E-06	0.00E+00	4.00E+00	3.00E+00	2.28E+09
3.05E-06	1.77E-06	0.00E+00	5.00E+00	4.00E+00	5.25E+05
1.77E-06	1.30E-06	0.00E+00	6.50E+00	5.00E+00	2.11E+05
1.30E-06	1.13E-06	0.00E+00	8.00E+00	6.50E+00	4.13E+04
1.13E-06	1.00E-06	0.00E+00	1.00E+01	8.00E+00	8.75E+03
1.00E-06	8.00E-07	0.00E+00			
8.00E-07	4.00E-07	0.00E+00			
4.00E-07	3.25E-07	0.00E+00			
3.25E-07	2.25E-07	0.00E+00			
2.25E-07	1.00E-07	0.00E+00			
1.00E-07	5.00E-08	0.00E+00			
5.00E-08	3.00E-08	0.00E+00			
3.00E-08	1.00E-08	0.00E+00			
1.00E-08	1.00E-11	0.00E+00			
Total		8.21E+07			3.68E+15

[a] Source: CRWMS M&O 1999f, Attachment VIII, p. VIII-1 for the neutron source.

[b] Source: CRWMS M&O 1999f, Attachment IX, p. IX-1 for the gamma source.

Table 6. Neutron and Gamma Current Sources Exiting Naval Spent Fuel Canister Side Surface

Neutron Current[a]		Gamma Current[b]	
Upper Energy (MeV)	2-yr Decay (n /cm^2-s)	Upper Energy (MeV)	5-yr Decay (γ/cm^2-s)
2.117E+01	3.644E-02	4.000E-01	9.928E+06
1.284E+01	2.242E-01	9.000E-01	8.120E+09
1.000E+01	9.292E-01	1.350E+00	6.486E+07
7.790E+00	2.600E+00	1.800E+00	7.720E+07
6.070E+00	5.663E+00	2.200E+00	7.403E+06
4.720E+00	2.475E+01	2.600E+00	1.091E+05
2.860E+00	5.713E+01	3.000E+00	1.269E+04
1.740E+00	1.937E+02	4.000E+00	3.490E+02
8.210E-01	2.875E+02	5.000E+00	5.492E+00
3.897E-01	2.426E+02	6.000E+00	4.631E-08
1.830E-01	1.479E+02	1.000E+01	2.296E-03
6.740E-02	9.030E+01		
5.530E-03	2.405E+01		
2.260E-05	2.327E+00		
6.250E-07	2.357E-02		
Total	1.080E+03	Total	8.280E+09

[a] Source: Naples 1999, Enclosure 2, p. 4 at 2 yr after reactor shutdown. For conservatism, no correction to 5 yr was made, because of a small neutron contribution relative to gamma.

[b] Source: CRWMS M&O 2000a, Attachment II, p II-2 at 5 yr after reactor shutdown, corrected from Naples 1999, Enclosure 2, p. 4 at 2 yr after reactor shutdown.

5.2 REVIEW OF WASTE PACKAGE DOSE RATE CALCULATIONS

The Waste Package Department has performed several dose rate calculations for the different waste package types including the following:

21-PWR Uncanistered Fuel (UCF) Waste Package
44-BWR UCF Waste Package
DHLW/DOE SNF Co-disposal Waste Package
Single-Corrosion-Resistant-Material (CRM) Naval SNF Waste Package

Other WP types such as the 12-PWR and 24-BWR small waste packages are less radiation limiting than the corresponding large WPs, and thus pose no impact to determination of the limiting waste package. It has been demonstrated for the viability assessment (VA) WP design that with the same radiation source term, the resulting dose rate is less from the 12-PWR WP than from the 21-PWR WP (CRWMS M&O 1999e, p. 23 & Att. I-57, and p. 25 & Att. I-25).

Table 7 lists the maximum dose rate on the WP surface for each WP type considered, based on the bounding source terms. Table 7 also includes the average dose rates for the 21-PWR and 44-BWR waste packages for comparison. Comparison of the dose rates indicates that the 21-PWR waste package has the highest dose rate among the various WP types, and thus represents the limiting waste package for shielding considerations. Therefore, the 21-PWR waste package is selected here for specification of shielding-specific source terms.

5.3 WASTE STREAM AND FUEL ASSEMBLY HEAT DISTRIBUTION

The projected waste stream for commercial PWR and BWR spent fuel to be received at the repository is described in CRWMS M&O 1999a. For this calculation, the focus is on PWR fuel only, which is more radiation limiting than BWR fuel, as discussed in Section 5.2.

The document cited above also provides the heat distribution for PWR fuel at arrival as reproduced in Table 8 and plotted in Figure 1. The distribution of fuel assemblies in each heat range depends on the waste receipt scenarios defined on p. 3 of CRWMS M&O 1999a for the likely scenarios of Cases A, B and C as follows:

Case A – Fuel selection begins with 10-year-old fuel and progresses to older fuel

Case B - Fuel selection begins with 10-year-old fuel and progresses to older fuel in strict order of fuel age

Case C - Fuel selection begins with oldest fuel still in pool and progresses to younger fuel

The heat distribution in Table 8 shows that 97 % of PWR fuel assemblies have heat generation rates at less than 1.2 kW/assembly. Use of ~1.2 kW/assembly as a basis for selection of the design basis fuel for subsurface shielding design will cover 97% of the historical and projected PWR SNF population (see discussion in Section 5.5). The remaining 3% will require special considerations as discussed in Section 6.3.

Table 7. WP Surface Dose Rate Comparison

Waste Package Type[a]	Dose Rate (rem/hr)		Reference
	Maximum	Average	
21-PWR Waste Package	1039.2	171.4	CRWMS M&O 2000c, pp. 28 & 40
44-BWR Waste Package	900.5	116.4	CRWMS M&O 2000d, pp. 24 & 36
DHLW/DOE SNF Co-disposal WP	103.7	N/A	CRWMS M&O 2000e, p. 21
Naval SNF Waste Package	188.5	N/A	CRWMS M&O 2000a, p. 19

[a] No dose rate calculations performed for the current 12-PWR and 24-BWR waste packages.

Table 8. Summary Heat Distribution for PWR Fuel upon Receipt at Repository

Heat Range (watts/assembly)	**Range of Percent of Assemblies**[a]	**Average Percent**	**Average Cumulative Percent**
0 - 99	0.9 – 1.0	0.95	0.95
100 - 199	7.0 – 8.3	7.65	8.60
200 - 299	13.9 – 17.9	15.90	24.50
300 - 399	13.9 – 16.3	15.10	39.60
400 - 499	8.8 – 13.8	11.30	50.90
500 - 599	7.9 – 13.4	10.65	61.55
600 - 699	7.6 – 14.4	11.00	72.55
700 - 799	8.3 – 9.4	8.85	81.40
800 - 999	8.1 – 14.3	11.20	92.60
1000 - 1199	2.3 – 6.5	4.40	97.00
1200 - 1399	0.9 – 2.0	1.45	98.45
1400 - 1599	0.5 – 0.6	0.55	99.00
1600 - 1799	0.2 – 0.3	0.25	99.25
1800 - 1999	0.1 – 0.1	0.10	99.35
MOX Fuel[b]	1.4 – 1.4	1.40	100.75[c]

[a] Source: CRWMS M&O 1999a, pp. 15 & 16.

[b] Heat not calculated for mixed oxide (MOX) fuel.

[c] Greater than 100%, owing to numerical averaging used.

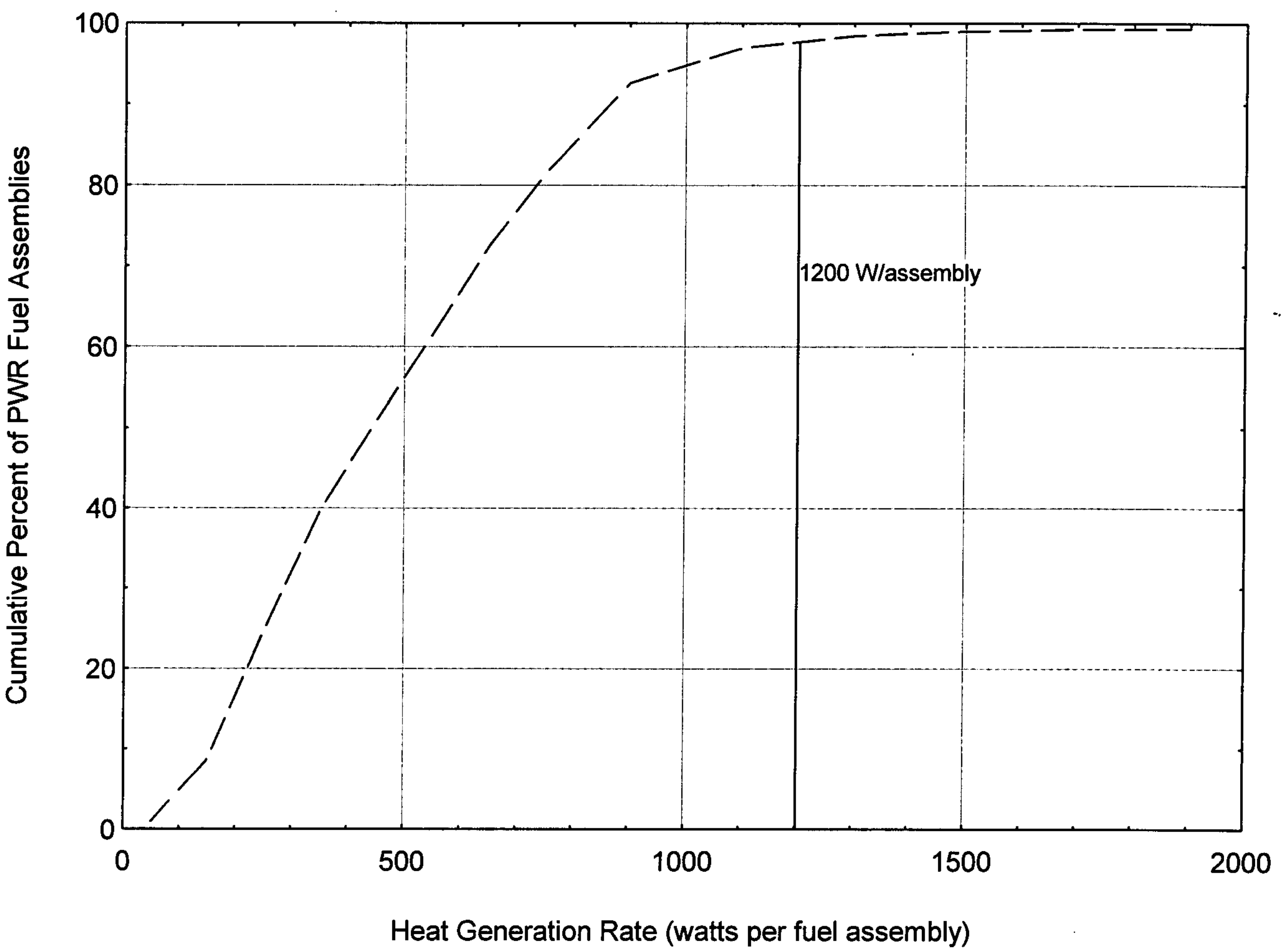


Figure 1. Heat Distribution of PWR SNF Inventory upon Receipt at Repository

5.4 WASTE PACKAGE THERMAL SOURCE TERMS

The calculation document: *Waste Packages and Source Terms for the Commercial 1999 Design Basis Waste Streams* (CRWMS M&O 2000h) provides characterization of the heat generation rates of waste packages loaded with uncanistered commercial SNF assemblies. It contains the information on the fuel assembly contents and heat generation rate for each waste package at the time of emplacement in the repository. Loading of fuel assemblies in the waste packages is based on the waste stream scenarios described in CRWMS M&O 1999a, and a WP thermal output limit of 11.8 kW/WP (CRWMS M&O 2000g, Section 1.2.4.2, p. 17). In addition to the WP heat generation rates, this calculation document also provides the maximum assembly heat generation rate as a function of year from the initial emplacement (i.e., 2010) to 2032 for the 63,000 MTU inventory case, and to 2040 for the 84,000 MTU commercial SNF inventory case.

For all fuel receipt scenarios, the maximum assembly heat generation rate is about 1.8 kW/assembly (CRWMS M&O 2000h, Table 17, File *preblend.dat*). For Cases A and B defined in Section 5.3, the typical maximum assembly heat generation rate is less than 1 kW/assembly until the later year of emplacement.

5.5 SELECTION OF DESIGN BASIS FUEL

5.5.1 SELECTION METHOD

The waste package specific source terms for shielding applications depend on the waste forms and waste package types. As shown in Section 5.2, the limiting waste package is the 21 PWR waste package. Hence, this waste package type is selected here for specification of the shielding source terms.

Consistent with the specification of contents required for the SNF transportation and storage casks (10CFR71.33 and 10CFR72.236, respectively), the waste package source term is specified by selecting a design basis fuel. Description of the design basis fuel needs to include the relevant fuel parameters such as fuel type, uranium loading, heat generation rate, initial enrichment, burnup and cooling time.

Selection of the design basis fuel requires the knowledge of the WP thermal output limit and maximum heat fuel assembly. For the 21 PWR waste package, the thermal output limit is 11.8 kW/WP as specified in Section 1.2.4.2 of CRWMS M&O 2000g, corresponding to an average of 0.562 kW per fuel assembly (11.8 kW/21). The maximum heat fuel assembly is provided in Table 17, File *preblend.dat* of CRWMS M&O 2000h as ~1.8 kW per assembly, based on the projected waste stream scenarios. This maximum fuel assembly is 3.2 times the average fuel assembly (based on the 11.8 kW/WP limit for 21 PWR fuel assemblies) in terms of the thermal output. Use of the maximum fuel assembly as a design basis fuel would result in a total thermal output of 37.8 kW/WP for the 21 PWR waste package, which is unduly conservative. On the other hand, it is inappropriate to use the average fuel assembly, because a small number of fuel assemblies in the waste package can dictate the shielding source terms and resulting dose rates.

Because of the importance of a small number of fuel assemblies in the waste package for shielding considerations, the selected design basis fuel must provide sufficient coverage of the historical and projected SNF inventory to be received at the repository. For the VA design, a design basis fuel was used, bounding 97.85% of PWR SNF population (CRWMS M&O 1996b, p. 7) based on the oldest-fuel-first receipt scenario. To be comparable to the VA design, a similar coverage is selected in this calculation.

5.5.2 PWR DESIGN BASIS FUEL

The design basis fuel for the 21 PWR waste package is selected and justified below:

Fuel assembly class: Babcock & Wilcox (B&W) Mark B 15x15 PWR fuel assemblies as used in *PWR Source Term Generation and Evaluation* (CRWMS M&O 1999d, p. 10).

Initial mass of uranium per assembly for source terms: 0.475 MTU (CRWMS M&O 1999d, p. 7).

Cooling time: 10 years after reactor shutdown, corresponding to the initial fuel age for Cases A and B (CRWMS M&O 1999a, p. 3). See Section 6.3 for shorter cooling times.

Heat generation rate: ~1.2 kW/assembly, which is more than 2 times the average (Section 5.5.1), and covers 97% of PWR SNF population (Table 8), similar to the VA design. With 1.2 kW/assembly, the WP thermal output would be 25.2 kW, exceeding the limit of 11.8 kW. However, only a limited number of fuel assemblies in the waste package could contribute to the WP external dose rate. A waste package loaded with only a few assemblies with 1.2 kW/assembly while meeting the limit of 11.8 kW/WP could produce the same dose rate as if the waste package were loaded with all 21 fuel assemblies containing 1.2 kW each. See Section 6.3 for loading of fuel assemblies with heat generation rates of more than 1.2 kW.

Initial fuel enrichment: 4%, which is the low end of the enrichment range for high burnup fuel based on the DOE Characteristics Data Base (DOE 1992, p. 2.4-3). A lower enrichment tends to produce a higher source term, as demonstrated in *PWR Source Term Generation and Evaluation* (CRWMS M&O 1999d), and indicated in *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 2000, p. 7-10).

Fuel burnup: 60 GWd/MTU, which is higher than the maximum burnup (58 GWd/MTU) for the historical PWR SNF inventory as of 1995 (CRWMS M&O 1999a, p. B-14). The burnup value of 60 GWd/MTU in conjunction with the cooling time of 10 years and initial enrichment of 4% results in a heat generation rate of approximately 1.2 kW/assembly (CRWMS M&O 1999d, Attachment IV). See Section 6.3 for fuel assemblies with burnup in excess of 60 GWd/MTU expected in the projected inventory.

Tables 9 and 10 provide the neutron and gamma source terms for the selected design basis fuel specification. Uniform fuel burnup within the fuel assembly is assumed (Assumption 3.2). These source terms are used in all subsequent dose rate calculations for this evaluation.

Table 9. Design Basis PWR SNF Neutron Source Terms

Neutron Energy Range (MeV)		Neutron Source (n/s per assembly)
Upper Bound	Lower Bound	Design Basis[a]
2.00E+01	6.43E+00	1.54E+07
6.43E+00	3.00E+00	1.74E+08
3.00E+00	1.85E+00	1.91E+08
1.85E+00	1.40E+00	1.09E+08
1.40E+00	9.00E-01	1.48E+08
9.00E-01	4.00E-01	1.61E+08
4.00E-01	1.00E-01	3.16E+07
1.00E-01	1.70E-02	0.00E+00
1.70E-02	3.00E-03	0.00E+00
3.00E-03	5.50E-04	0.00E+00
5.50E-04	1.00E-04	0.00E+00
1.00E-04	3.00E-05	0.00E+00
3.00E-05	1.00E-05	0.00E+00
1.00E-05	3.05E-06	0.00E+00
3.05E-06	1.77E-06	0.00E+00
1.77E-06	1.30E-06	0.00E+00
1.30E-06	1.13E-06	0.00E+00
1.13E-06	1.00E-06	0.00E+00
1.00E-06	8.00E-07	0.00E+00
8.00E-07	4.00E-07	0.00E+00
4.00E-07	3.25E-07	0.00E+00
3.25E-07	2.25E-07	0.00E+00
2.25E-07	1.00E-07	0.00E+00
1.00E-07	5.00E-08	0.00E+00
5.00E-08	3.00E-08	0.00E+00
3.00E-08	1.00E-08	0.00E+00
1.00E-08	1.00E-11	0.00E+00

[a] Source: CRWMS M&O 1999d, Attachment IV, File *PWR.neutron.source* for PWR fuel with 4% enrichment, 60 GWd/MTU burnup and 10 yr cooling.

Table 10. Design Basis PWR SNF Gamma Source Terms

Gamma Energy Range (MeV)		Gamma Source (γ's/s per assembly)[a]			
		Fuel	Bottom	Plenum	Top
5.00E-02	1.00E-02	1.21E+15	2.73E+11	1.88E+11	1.75E+11
1.00E-01	5.00E-02	3.29E+14	5.28E+10	2.77E+10	3.39E+10
2.00E-01	1.00E-01	2.45E+14	1.28E+10	1.17E+10	8.19E+09
3.00E-01	2.00E-01	7.13E+13	6.39E+08	6.33E+08	4.07E+08
4.00E-01	3.00E-01	4.55E+13	8.50E+08	1.64E+09	5.33E+08
6.00E-01	4.00E-01	2.26E+14	4.92E+08	2.69E+10	3.37E+07
8.00E-01	6.00E-01	2.37E+15	2.91E+09	1.60E+10	1.86E+09
1.00E+00	8.00E-01	1.22E+14	5.40E+09	2.48E+09	3.41E+09
1.33E+00	1.00E+00	1.95E+14	1.54E+13	7.97E+12	9.90E+12
1.66E+00	1.33E+00	4.50E+13	4.35E+12	2.25E+12	2.80E+12
2.00E+00	1.66E+00	1.52E+11	2.35E+00	1.49E+02	2.15E-02
2.50E+00	2.00E+00	5.17E+10	1.03E+08	5.34E+07	6.64E+07
3.00E+00	2.50E+00	3.79E+09	1.60E+05	8.29E+04	1.03E+05
4.00E+00	3.00E+00	4.97E+08	9.43E-10	1.55E-10	5.19E-10
5.00E+00	4.00E+00	2.82E+07	0.00E+00	0.00E+00	0.00E+00
6.50E+00	5.00E+00	1.13E+07	0.00E+00	0.00E+00	0.00E+00
8.00E+00	6.50E+00	2.22E+06	0.00E+00	0.00E+00	0.00E+00
1.00E+01	8.00E+00	4.71E+05	0.00E+00	0.00E+00	0.00E+00

[a] Source: CRWMS M&O 1999d, Attachment IV, File *PWR.gamma.source* for PWR fuel with 4% enrichment, 60 GWd/MTU burnup and 10 yr cooling.

5.6 FUEL ASSEMBLIES CONTRIBUTING TO DOSE RATES

In the following shielding calculations, all the fuel assemblies in the waste package are assumed to be of the same characteristics with identical source term in order to simplify the analytical model (Assumption 3.7). This approach is appropriate, as the fuel assemblies provide self-shielding effects. Only the fuel assemblies close to the dose points of interest make contributions to the dose rates external to the waste package. This section provides a calculation to determine the contributing fuel assemblies to the WP external dose rates. The calculation covers the gamma contribution only, which is more significant than the neutron contribution, and controls the shielding mass of the WP transporter.

Since the radial dose rate is more sensitive to the self-shielding effects of the fuel assemblies, the calculation considers the radial configuration only.

5.6.1 Calculation Inputs

Fuel cavity cross section: square (CRWMS M&O 2000b, Attachment I)

Fuel cavity dimension: 22.64 cm (CRWMS M&O 2000b, Attachment I)

WP and fuel basket dimensions and materials: (CRWMS M&O 2000b, Attachment I)

Fuel type: B&W Mark B 15x15 PWR fuel assembly (CRWMS M&O 1999d, p. 10)

Mass in active fuel region (per assembly)
U: 463.63 kg (CRWMS M&O 1999d, p. 10)
O: 62.83 kg (CRWMS M&O 1999d, p. 10)
Zr-4: 115.12 kg (CRWMS M&O 1999d, p. 10)
Inconel-718: 4.90 kg (CRWMS M&O 1999d, p. 10)
Total: 646.48 kg

Design basis fuel neutron source term: See Table 9.

Design basis fuel gamma source term: See Table 10.

Material densities and compositions: See Table 11. Material data for the SNF and WP components are consistent with those used by the Waste Package Department. Fresh, unirradiated fuel is assumed (Assumption 3.1).

Flux-to-dose rate conversion factors for neutrons: ANSI/ANS-6.1.1-1977, p. 4.

Flux-to-dose rate conversion factors for photons: ANSI/ANS-6.1.1-1977, p. 5.

Table 11. Material Compositions of SNF, WP Components, and Shielding Materials

Region	Material	Density (g/cm³)	Wt. % Used	Reference
Fuel	4% Enriched Uranium	18.9 (Reference only, not used)	U234: 0.0347 U235: 4.0 U236: 0.0184 U238: 95.9469	Bowman et al. 1995 & CRWMS M&O 2000c, p. II-3
Fuel	Zircaloy-4	6.56 (Reference only, not used)	Sn: 1.45 Fe: 0.21 Cr: 0.425 O: 0.125 Zr: 97.79	CRWMS M&O 1999g, Section 5.11 & CRWMS M&O 2000c, p.14
Fuel	Inconel 718	8.19 (Reference only, not used)	Ni: 51.50 Cr: 19.00 Nb: 5.125 Ta: 0 Mo: 3.05 Ti: 0.90 Al: 0.50 Co: 1.00 C: 0.08 Mn: 0.35 Si: 0.35 P: 0.015 S: 0.015 B: 0.006 Cu: 0.30 Fe: 17.809	Inco Alloys International 1988, p. 11 & CRWMS M&O 2000c, p. 14 (TBV-4059)
WP Outer Barrier	Alloy C-22 (or Alloy 22)	8.69	Co: 2.5 Cr: 21.25 Mo: 13.5 W: 3.0 Fe: 4.0 Si: 0.08 Mn: 0.50 C: 0.015 V: 0.35 P: 0.02 O: 0.02 Ni: 54.765	CRWMS M&O 1999g, p.30 & CRWMS M&O 2000c, p.11 (TBV-3885)
WP Inner Barrier & Transporter skin	SS316L (or SS316NG)	7.98	C: 0.03 Mn: 2.00 Si: 1.00 Cr: 17.00 Ni: 12.0 P: 0.045 S: 0.03 Mo: 2.5 N: 0.1 Fe: 65.295	CRWMS M&O 2000c, p.10 (TBV-3885)
Transporter Gamma Shielding	A516 carbon steel	7.85	C: 0.27 Mn: 1.025 P: 0.035 S: 0.035 Si: 0.275 Fe: 98.36	CRWMS M&O 1999g, Section 5.2, & CRWMS M&O 2000c, p.10 (TBV-4046)
Transporter Neutron Shielding	Boron-Polyethylene	0.95	C: 61.20 O: 22.20 H: 11.60 B: 5.00	Reactor Experiments 1991 (TBV-4891)
Main Drift Lining	Concrete (cured)	2.35	H: 0.55 O: 49.83 Si: 31.57 Ca: 8.26 Na: 1.70 Mg: 0.26 Al: 4.55 S: 0.13 K: 1.91 Fe: 1.23	ANSI/ANS-6.4-1997, Table 5.2, p. 9
Air Space	Dry Air	0.001204	N: 75.52 O: 23.18 C: 0.01 Ar: 1.29 note: these values are *wt.%* converted from volume %	Weast 1985, pp. F-10, F-156
Host Rock	Dry Tuff	2.21 (TBV-4386)	Si: 36.801 Al: 6.441 Fe: 0.697 Ca: 0.339 Mg: 0.074 Ti: 0.061 Na: 3.027 K: 2.723 P: 0.009 Mn: 0.040 O: 49.815	CRWMS M&O 2000f, pp. 10 & 32

5.6.2 PATH Calculation

The gamma dose rates external to the waste package were calculated with the PATH code (CRWMS M&O 1996a). This code is appropriate and suitable to quickly identify the important contributing fuel assemblies and to compare the results between different geometric models used. However, because of the approximations used in the code, PATH results lack the accuracy produced from other sophisticated shielding codes such as MCNP.

The PATH calculation considers two different geometric models; one with each individual fuel assembly explicitly modeled, and the other with all 21 fuel assemblies homogenized as a single region. This comparison serves to validate the model adopted for subsurface shielding applications. In both models, the waste package and transporter are both centered in the main drift (Assumption 3.5).

5.6.2.1 Explicit Fuel Assembly Model

For this model, each fuel assembly in the waste package is individually represented as shown in Figure 2. The fuel basket plates are included in the model, based on the dimensional and material inputs from CRWMS M&O 2000b (Attachment I). The waste package is contained in the transporter. The transporter model is identical to that used in *Evaluation of WP Transporter Neutron Shielding Materials* (CRWMS M&O 1998a, p. 30).

The contribution to the gamma dose rate from each fuel assembly is individually calculated with the PATH code at the selected dose points indicated in Figure 2 for identification of the peak position. The gamma source for each fuel assembly is represented by a volumetric source, obtained by dividing the gamma source strength in Table 9 for each energy group by the assembly cavity volume for the active fuel region. The respective cavity volume is given as:

$$(22.64 \text{ cm})^2 \times (360.172 \text{ cm}) = 1.846 \times 10^5 \text{ cm}^3$$

where 22.64 cm represents the side dimension of the square cavity, and 360.172 cm is the active fuel length (see Section 5.6.1). The materials in the fuel region are homogenized in a smeared region as shown in Table 12.

For simplicity in the PATH calculation, which is based on the point-kernel integration method, the following materials are simulated by iron: Inconel-718, A516 carbon steel, SS316L stainless steel, and Alloy 22, according to Assumption 3.4. This simulation is reasonable as iron with a similar atomic number is representative of the mass attenuation coefficients (cm^2/g) for these materials. Zr-4 in the fuel region is replaced by Mo, since PATH contains the attenuation data library for Mo, but not for Zr. The mass attenuation coefficients are similar for Mo and Zr (ANSI/ANS-6.4.3-1991, p.7).

Section 6.2.1 presents the results of the PATH calculation for the explicit individual fuel assembly model. The associated PATH input and output files are listed and described in Table 13. The fuel assemblies, which make negligible contributions to the dose rates at the points of interest, are omitted from the calculation and indicated in Table 13.

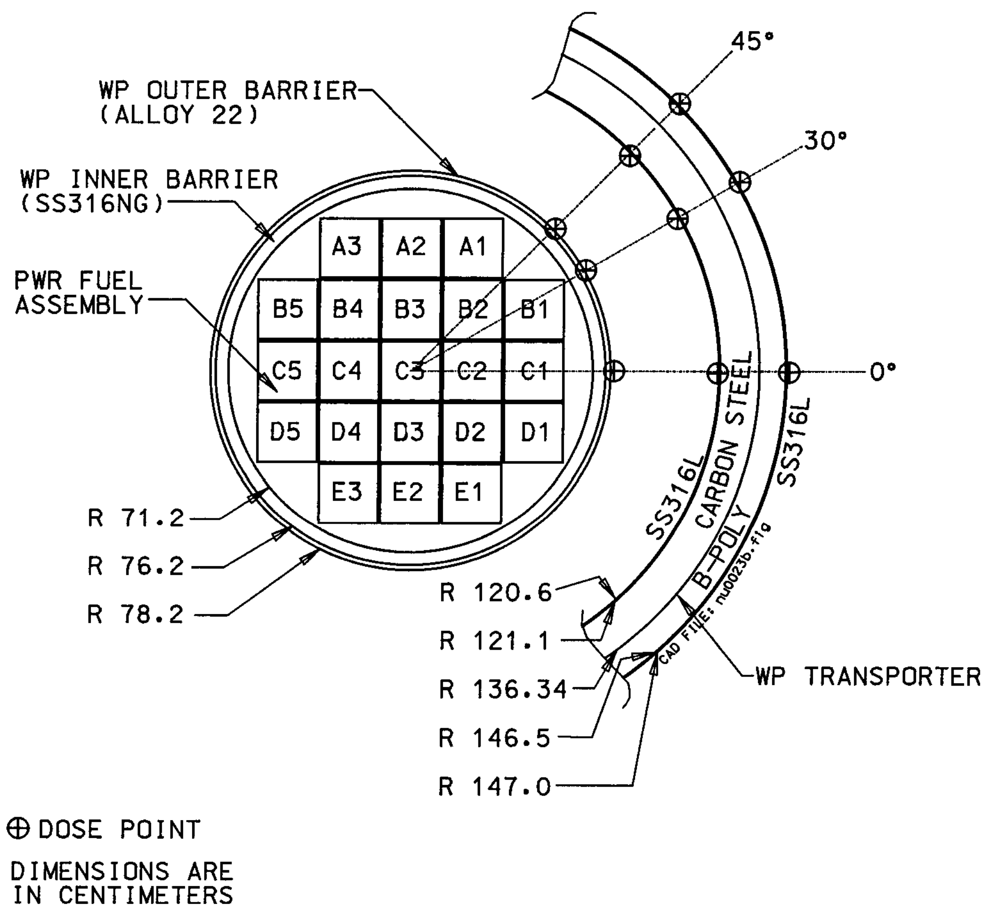


Figure 2. Explicit Fuel Assembly Model for PATH Calculation

Table 12. Smeared PWR Fuel Region

Material	**Mass (kg)**[a]	**Mass Fraction**[b]	**Partial Density (g/cc)**[c]
U	463.63	0.7172	2.512
O	62.83	0.0972	0.340
Zr-4[d]	115.12	0.1780	0.624
Inconel-718[e]	4.90	0.0076	0.026
Total	646.48	1.0	3.502

[a] Mass inputs from Section 5.6.1.

[b] Mass fraction = mass for a given material divided by 646.48 kg total.

[c] Partial density = mass for a given material divided by the fuel cavity volume of 1.846×10^5 cm^3

[d] Simulated by Mo in PATH (Assumption 3.4).

[e] Simulated by Fe in PATH (Assumption 3.4).

Table 13. PATH Input and Output Files for Explicit Fuel Assembly Model

Fuel ID[a]	Input File	Output File	Remark
A1	pwrwp05b.inp	pwrwp05b.out	For all dose points
A2	pwrwp08b.inp	pwrwp08b.out	For all dose points
A3	pwrwp10b.inp	pwrwp10b.out	For dose points @ 45° only
B1	pwrwp02b.inp	pwrwp02b.out	For all dose points
B2	pwrwp03b.inp	pwrwp03b.out	For all dose points
B3	pwrwp07b.inp	pwrwp07b.out	For all dose points
B4	Not calculated[b]		Negligible contribution[b]
B5	Not calculated		Negligible contribution
C1	pwrwp01b.inp	pwrwp01b.out	For all dose points
C2	pwrwp04b.inp	pwrwp04b.out	For all dose points
C3	pwrwp06b.inp	pwrwp06b.out	For all dose points
C4	Not calculated		Negligible contribution
C5	Not calculated		Negligible contribution
D1	pwrwp10b.inp	pwrwp10b.out	For all dose points
D2	pwrwp09b.inp	pwrwp09b.out	For all dose points
D3	pwrwp07b.inp	pwrwp07b.out	For dose points @ 0° only
D4	Not calculated		Negligible contribution
D5	Not calculated		Negligible contribution
E1	pwrwp05b.inp	pwrwp05b.out	For dose points @ 0° only
E2	pwrwp08b.inp	pwrwp08b.out	For dose points @ 0° only
E3	Not calculated		Negligible contribution

[a] See Figure 2 for identification (ID) and location of each fuel assembly.

[b] The contribution from this fuel assembly is not calculated, owing to substantial shielding by the front intervening fuel assemblies. This shielding effect results in a negligible contribution as estimated from the result for the immediately adjacent assembly. This footnote applies to all fuel assemblies marked with "Not Calculated".

5.6.2.2 Homogenized Model

The homogenized model for the fuel region is the same as the model used in *Shielding Calculation for Emplacement Operations and Subsurface Layout* (CRWMS M&O 2000f). All 21 fuel assemblies are homogenized as a single region with an equivalent radius of 58.53 cm (CRWMS M&O 2000f, p. 22). The model includes the thicknesses of the fuel basket corner guide and outer fuel assembly tube described as a ring around the homogenized fuel region. However, the model conservatively omits the internal fuel basket plates between the fuel assemblies (Assumption 3.3), ignoring the shielding effects of these plates.

With the homogenized model, there is no azimuthal or angular variation in the dose rate around the waste package or transporter. Hence, the dose rate is calculated only at a point on each of the following surfaces: WP outer surface, transporter inner surface, and transporter outer surface.

The results of the PATH calculation for the homogenized model are provided in Section 6.2.2. The associated input and output files are *PWRWP00.INP* and *PWRWP00.OUT*, respectively, as listed in Attachment I.

5.7 MCNP CALCULATION FOR HOMOGENIZED MODEL

The MCNP code is capable of treating detailed particle transport process for both neutrons and photons with accuracy. The MCNP code is used here to calculate both neutron and gamma dose rates. The calculation uses the homogenized fuel region model for comparison with the PATH calculation on the gamma contribution. The homogenized model is more conservative (i.e., higher dose rate results) for subsurface shielding applications than the explicit fuel assembly model, as demonstrated in the PATH calculation (Section 5.6) and its results (Section 6.2.1).

For the MCNP calculation, the model includes the main drift concrete lining and surrounding tuff medium to account for scattering effects from the drift wall as depicted in Figure 3, and uses Assumptions 3.5 and 3.6. Note that the PATH model in Figure 2 does not include the drift wall or tuff medium, as these regions are not in the path from the source to dose point for the point kernel integration.

The neutron and gamma source terms are from Tables 9 and 10, respectively. The neutron source input covers the full energy spectrum. However, only the contributing energy groups between 0.4 and 4.0 MeV are included in the gamma source input, consistent with the calculation in *Shielding Calculation for Emplacement Operations and Subsurface Layout* (CRWMS M&O 2000f, p. 10).

The material property inputs including densities and compositions are taken from Table 11. Table 14 lists the composition of smeared fuel for input into MCNP. Unlike the PATH calculations, the MCNP calculations use the actual material compositions for accurate representation.

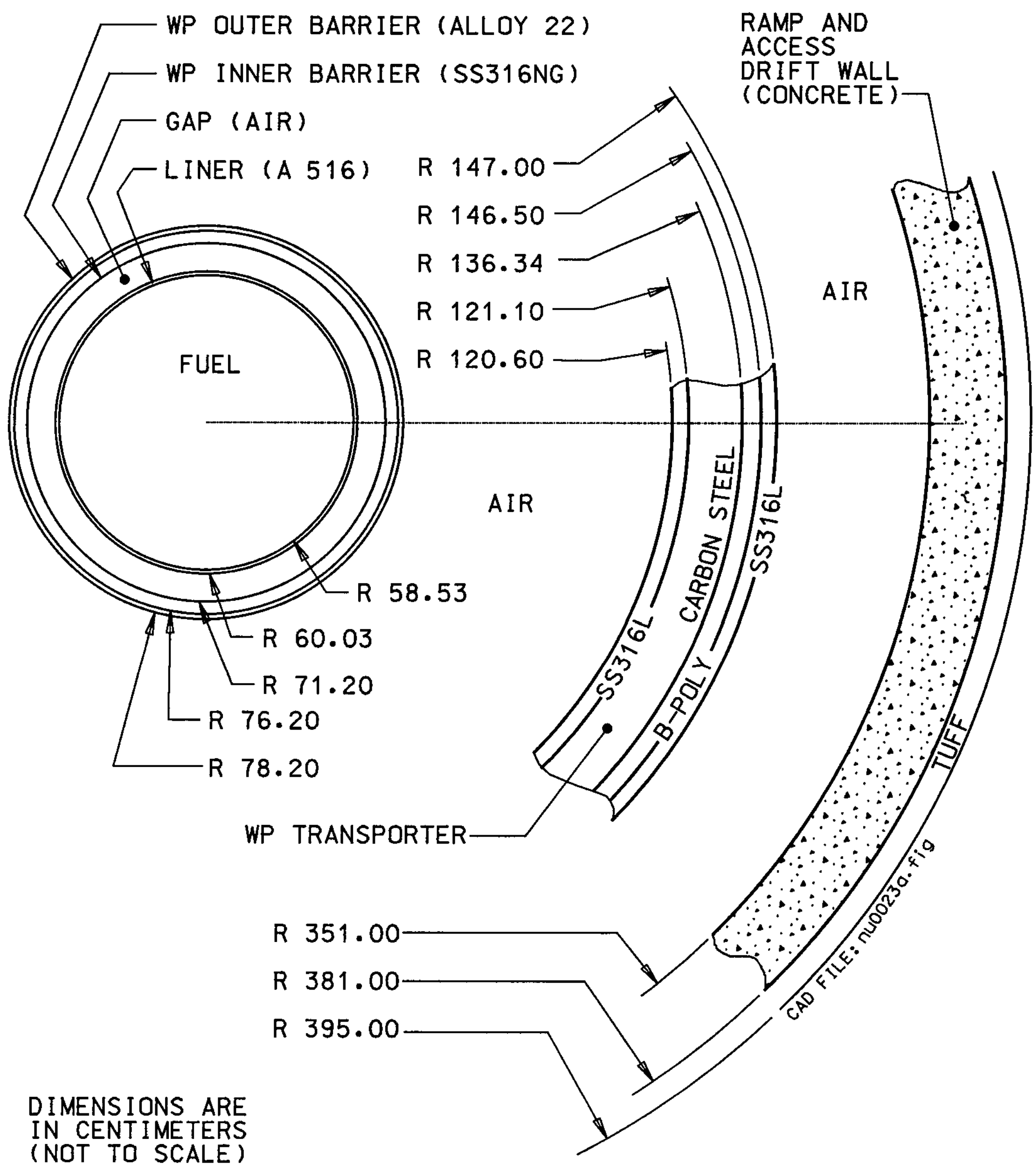


Figure 3. Radial WP and Transporter Model for MCNP Calculation

Table 14. Fuel Region Smeared Composition for MCNP

Element	Mass per Fuel Assembly (g)[a]			Mass (g)	Weight Fraction
	Inconel - 718	Zircaloy – 4	UO_2 (4% U-235)		
U - 238	0	0	4.4484E+05	4.4484E+05	6.881E-01
U - 235	0	0	1.8545E+04	1.8545E+04	2.869E-02
U - 236	0	0	8.5308E+01	8.5308E+01	1.320E-04
U - 234	0	0	1.6101E+02	1.6101E+02	2.491E-04
O	0	1.4390E+02	6.2830E+04	6.2974E+04	9.741E-02
Fe	8.7264E+02	2.4175E+02	0	1.1144E+03	1.724E-03
Cr	9.3100E+02	4.8926E+02	0	1.4203E+03	2.197E-03
Ni	2.5235E+03	0	0	2.5235E+03	3.903E-03
Nb	2.5223E+02	0	0	2.5223E+02	3.885E-04
Mo	1.4945E+02	0	0	1.4945E+02	2.312E-04
Ti	4.4100E+01	0	0	4.4100E+01	6.822E-05
Al	2.4500E+01	0	0	2.4500E+01	3.790E-05
Co	4.9000E+01	0	0	4.9000E+01	7.580E-05
C	3.9200E+00	0	0	3.9200E+00	6.064E-06
Mn	1.7150E+01	0	0	1.7150E+01	2.653E-05
Si	1.7150E+01	0	0	1.7150E+01	2.653E-05
P	7.3500E-01	0	0	7.3500E-01	1.137E-06
S	7.3500E-01	0	0	7.3500E-01	1.137E-06
B	2.9400E-01	0	0	2.9400E-01	4.548E-07
Cu	1.4700E+01	0	0	1.4700E+01	2.274E-05
Sn	0	1.6692E+03	0	1.6692E+03	2.582E-03
Zr	0	1.1258E+05	0	1.1258E+05	1.741E-01
Total	4.9000E+03	1.1512E+05	5.2646E+05	6.4648E+05	1

[a] Elemental mass in each material obtained by multiplying the material mass (Section 5.6.1) by the weight percent (Table 11).

The MCNP flux results are converted to dose rates in rem/hr, using the ANSI/ANS-6.1.1-1977 conversion factors. This unit is more appropriate for personnel shielding considerations than the unit of rad/hr used for radiation effects on materials.

As for the PATH homogenized model, the dose rates are calculated on the WP outer surface, transporter inner surface, and transporter outer surface. To obtain the contributions from all radiation source components, two MCNP runs were executed; one for the neutron and secondary gamma contributions, and the other for the fuel gamma contribution. The results of the MCNP calculations are presented in Section 6.2.3. The associated input and output files are as follows:

Neutron and secondary gamma: *wpneut1* (input) and *wpneut1.out* (output)
Fuel gamma: *wpgamma1* (input) and *wpgamma1.out* (output)

The MCNP input and output files are listed in Attachment I.

6. RESULTS

This document may be affected by technical product input information that requires confirmation. Any change to the document that may occur as a result of completing the confirmation activities will be reflected in subsequent revisions. The status of the technical product input information quality may be confirmed by review of the Document Input Reference System (DIRS) database.

6.1 WP SPECIFIC SOURCE TERMS

The waste package specific source terms are provided and discussed in Section 5.5. The specification is for the limiting waste package, which is the 21 PWR waste package containing the selected design basis fuel. The characteristics of the design basis fuel are described in Section 5.5.2. The shielding source terms for this design basis fuel are provided in Table 9 for the neutron source and Table 10 for the gamma source.

6.2 DOSE RATE RESULTS

6.2.1 PATH CALCULATION – EXPLICIT FUEL ASSEMBLY MODEL

Table 15 presents the results of the PATH calculation for the explicit fuel assembly model. The individual contributions are directly from the PATH outputs. Summation of the individual contributions is made with the Excel spreadsheet. The PATH calculation covers the gamma dose rates only. The results include the contribution from each fuel assembly for the design basis fuel specification with the characteristics of 4% initial enrichment, 60 GWd/MTU burnup and 10 years cooling. It needs to be pointed out that the PATH results are only approximate, because of the nature of the point-kernel integration method used in the PATH code.

The principal fuel assemblies contributing to the gamma dose rates are identified in Table 16 along with their percent contributions. Clearly, at every dose point, the closest three or four fuel assemblies contribute about 99% of the dose rate. The remaining fuel assemblies make little or no contributions, because of the shielding effects by the contributing assemblies. These results demonstrate that it takes only three or four fuel assemblies in the outer positions to produce the same dose rate as for the waste package containing 21 fuel assemblies with the same source term. Therefore, the dose rate outside the waste package is not dictated by the number of fuel assemblies contained in the waste package, but is rather affected by the characteristics of the fuel assemblies in the outer positions.

With the same source term for all fuel assemblies, the position of the peak dose rate depends on the surface location. On the waste package outer surface, the peak position occurs at the angle of 45^0 (see Table 15), because of the proximity of the contributing fuel assemblies to this dose point. As the dose point moves away from the WP surface, the peak position changes to the angle of 0^0, as noted in Table 15 for the results on the inner and outer surfaces of the transporter.

Table 15. PATH Gamma Dose Rate Results for Explicit Fuel Assembly Model

Fuel ID	Gamma Dose Rate (rem/hr)								
	WP Outer Surface			Transporter Inner Surface			Transporter Outer Surface		
	0^0	30^0	45^0	0^0	30^0	45^0	0^0	30^0	45^0
A1	3.611E-03	6.069E+01	1.772E+02	7.057E-01	5.820E+01	9.305E+01	1.199E-03	4.328E-02	7.535E-02
A2	3.342E-05	1.426E-01	1.799E-01	2.077E-04	9.479E-02	1.239E+00	1.222E-06	9.583E-05	2.269E-03
A3	nil	nil	6.982E-04	nil	nil	2.805E-02	nil	nil	8.073E-05
B1	9.709E+00	2.793E+02	1.772E+02	4.505E+01	1.200E+02	9.305E+01	3.885E-02	1.024E-01	7.535E-02
B2	6.410E-02	3.783E-01	3.783E+01	8.234E-02	3.767E+00	1.554E+01	1.661E-04	4.874E-03	1.504E-02
B3	2.810E-04	1.204E-03	6.575E-02	2.277E-04	2.693E-02	2.561E-02	6.979E-07	4.903E-05	3.469E-05
B4	nil	nil	nil	nil	nil	nil	nil	nil	nil
B5	nil	nil	nil	nil	nil	nil	nil	nil	nil
C1	3.527E+02	1.475E+00	1.799E-01	1.420E+02	9.943E+00	8.709E-01	1.170E-01	1.097E-02	1.386E-03
C2	5.733E-01	2.120E-02	6.575E-02	3.175E-01	2.421E-02	2.561E-02	5.057E-04	4.493E-05	3.469E-05
C3	2.412E-03	2.046E-04	5.982E-03	1.557E-03	1.339E-04	3.296E-03	2.778E-06	2.541E-07	5.608E-06
C4	nil	nil	nil	nil	nil	nil	nil	nil	nil
C5	nil	nil	nil	nil	nil	nil	nil	nil	nil
D1	9.709E+00	1.006E-02	6.982E-04	4.505E+01	1.049E+00	2.805E-02	3.885E-02	1.041E-03	8.073E-05
D2	6.410E-02	2.906E-04	2.561E-04	8.234E-02	1.806E-03	1.007E-04	1.661E-04	3.201E-06	1.835E-07
D3	2.810E-04	nil	nil	2.277E-04	nil	nil	6.979E-07	nil	nil
D4	nil	nil	nil	nil	nil	nil	nil	nil	nil
D5	nil	nil	nil	nil	nil	nil	nil	nil	nil
E1	3.611E-03	nil	nil	7.057E-01	nil	nil	1.199E-03	nil	nil
E2	3.342E-05	nil	nil	2.077E-04	nil	nil	1.222E-06	nil	nil
E3	nil	nil	nil	nil	nil	nil	nil	nil	nil
total	3.728E+02	3.420E+02	3.927E+02	2.340E+02	1.931E+02	2.039E+02	1.979E-01	1.628E-01	1.696E-01

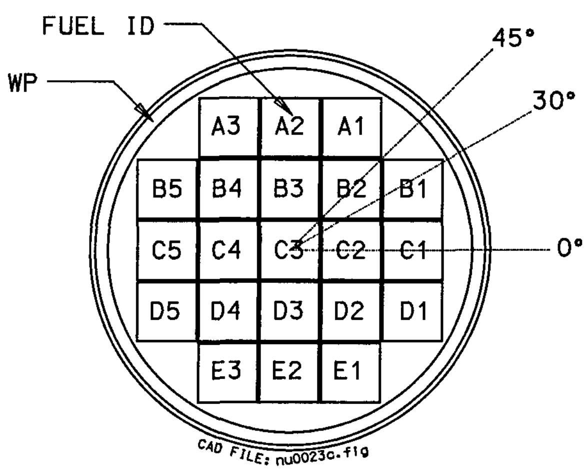
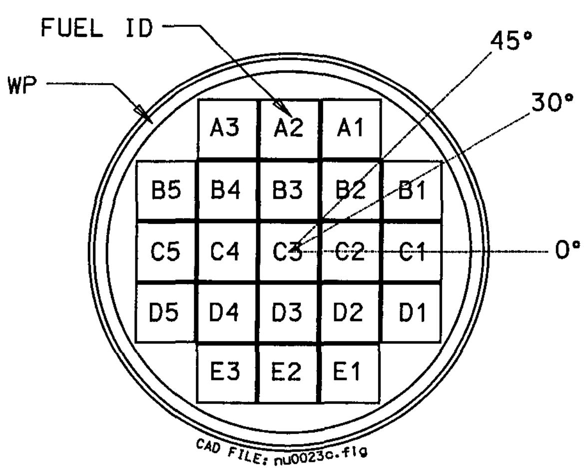


Table 16. Identification of Contributing Fuel Assemblies

Location	Dose Rate (rem/hr)	Fuel ID (% Contribution)				Subtotal
WP Outer Surface						
0^0	372.8	B1 (2.60%)	C1 (94.61%)	D1 (2.60%)		99.81%
30^0	342.0	A1 (17.75%)	B1 (81.67%)	C1 (0.43%)		99.85%
45^0	392.7	A1 (45.12%)	B1 (45.12%)	B2 (9.63%)		99.87%
Transporter Inner Surface						
0^0	234.0	B1 (19.25%)	C1 (60.68%)	D1 (19.25%)		99.18%
30^0	193.1	A1 (30.14%)	B1 (62.14%)	B2 (1.95%)	C1 (5.15%)	99.38%
45^0	203.9	A1 (45.64%)	A2 (0.61%)	B1 (45.64%)	B2 (7.62%)	99.51%
Transporter Outer Surface						
0^0	0.198	B1 (19.63%)	C1 (59.12%)	D1 (19.63%)		98.38%
30^0	0.163	A1 (26.58%)	B1 (62.90%)	B2 (3.00%)	C1 (6.74%)	99.22%
45^0	0.170	A1 (44.43%)	A2 (1.34%)	B1 (44.43%)	B2 (8.87%)	99.07%



6.2.2 PATH CALCULATION – HOMOGENIZED MODEL

Using the same design basis fuel specification as for the explicit fuel assembly model, the PATH calculation for the homogenized model produces the following results of the gamma dose rates:

WP Outer Surface:	339.20 rem/hr
Transporter Inner Surface:	257.14 rem/hr
Transporter Outer Surface:	0.22 rem/hr

As compared with the results in Table 15 for the explicit fuel assembly model, the homogenized model produces higher results except on the WP outer surface. The lower dose rate on the WP outer surface with the homogenized model is due to the fact that the fuel assemblies are pushed inward for smearing purposes, increasing the distance from the source to the WP surface. The effect of this distance increase becomes less and less significant, as the dose point moves farther and farther away from the source.

For subsurface shielding applications, the interest is typically in the dose locations away from the waste package. The PATH results indicate that the homogenized model is conservative to use for subsurface shielding design, since it produces higher dose rates than the explicit fuel assembly modes at the points of interest to subsurface shielding considerations.

6.2.3 MCNP CALCULATION – HOMOGENIZED MODEL

Table 17 presents the MCNP results for the homogenized model, which is verified in the PATH calculation to be conservative and appropriate to use. The MCNP calculation includes the contributions from all pertinent radiation components: fuel gammas, fuel neutrons, and secondary gammas.

In comparison with the PATH gamma results for the homogenized model (see Table 18), there is no clear trend as to the corresponding MCNP results being higher or lower. On the WP outer surface and transporter surface, the MCNP results are higher; however, the trend reverses on the transporter outer surface. The difference can be attributed to the single-medium treatment of buildup factors in the PATH code, which lacks the capability of correcting for multiple media. Over-prediction or under-prediction could occur in the PATH calculation, depending on the selection of the medium used for the buildup factors.

The total dose rate on the WP outer surface from the MCNP calculation is 465.5 ± 1.9 rem/hr. This total includes 446.9 rem/hr from fuel gammas, 18.5 rem/hr from fuel neutrons, and 0.1 rem/hr from secondary gammas.

From the results presented in Table 17, the contribution from fuel gammas is a predominant component of the total dose rate on the WP outer surface, constituting 96% of the total. This contribution decreases to 71.4% on the transporter outer surface, because of neutron and gamma shielding provided for the transporter to achieve an appropriate gamma-to-neutron dose ratio for weight optimization purposes.

Table 17. MCNP Dose Rate Results for Homogenized Fuel Assembly Model

Radiation Component	Dose Rate (rem/hr)		
	Waste Package Outer Surface	Transporter Inner Surface	Transporter Outer Surface
Fuel Gamma	4.469E+02 (0.0042)[a]	2.898E+02 (0.0046)	1.497E-01 (0.0091)
Fuel Neutron	1.848E+01(0.0101)	1.634E+01(0.0108)	4.304E-02 (0.0256)
Secondary Gamma	1.073E-01(0.0242)	7.987E-02 (0.0266)	1.704E-02 (0.0215)
Total	4.655E+02 (0.0041)	3.062E+02 (0.0044)	2.097E-01(0.0085)

[a] The value in parentheses denotes the fractional standard deviation for one-sigma uncertainty.

Table 18. Comparison of Gamma Dose Rate Results for Different Models

Model	Gamma Dose Rate (rem/hr)		
	Waste Package Outer Surface	Transporter Inner Surface	Transporter Outer Surface
PATH - Explicit Model[a]	3.728E+02	2.340E+02	1.979E-01
PATH - Homogenized Model	3.392E+02	2.571E+02	2.200E-01
MCNP- Homogenized Model	4.469E+02	2.898E+02	1.497E-01

[a] Results at typical locations of 0^0.

6.3 SPECIAL FUEL LOADING CONSIDERATIONS

The results presented in Tables 15, 16 and 17 are for the design basis fuel specification with the characteristics of 4% initial enrichment, 60 GWd/MTU burnup and 10 years cooling. This specification results in a heat generation rate of about 1.2 kW per fuel assembly, and covers 97% of the historical and projected PWR SNF inventory. The remaining 3% of the inventory include 1.6% of assemblies with thermal output in excess of 1.2 kW/assembly, and 1.4% of MOX fuel (see Table 8).

To accommodate fuel assemblies outside the design basis fuel specification, special fuel loading considerations are required. The options available include the following:

- Additional cooling time
- Selective fuel loading
- Administrative operational and access control

Additional cooling will reduce the thermal output with time until the assembly heat generation rate is below 1.2 kW/assembly. However, the additional cooling time would require more storage of hot assemblies on surface. The surface storage requirement will depend on the number of hot assemblies received at the repository.

To avoid the additional storage requirement, selective fuel loading may be used. There are four corner positions in the fuel assembly basket available for loading hot assemblies (i.e., assemblies with more than 1.2 kW/assembly) with minimal thermal performance impact. These positions include B2, B4, D2 and D4 (see Figure 2). Loading of the hot assemblies in these positions will mainly affect the dose rates at locations directly facing these assemblies such as the points at 45^0. According to Table 17, File *preblend.dat* of CRWMS M&O 2000h, the maximum assembly heat generation rate loaded in the waste packages is ~1.8 kW/assembly. If a maximum heat assembly is loaded in one of the four designated positions (say, B2), the dose rate from this particular assembly will increase by approximately 50% as compared to the design basis fuel with 1.2 kW/assembly. Based on the results in Table 13, this increase poses no effect on the transporter shielding, as the dose rate at 45^0 is still less than that at 0^0. Therefore, this selective loading option is acceptable from a shielding standpoint.

Administrative operational and access control provides a means to deal with special or rare operational occurrences. In the event that three hot assemblies are loaded in the outer fuel basket positions and adjacent to each other, the resulting dose rate will exceed the design goal for the design basis fuel specification. Personnel access control may be imposed to restrict access to the operational areas such as the vicinity of the transporter where the design goal is exceeded. Other control measures such as reduction in occupancy time in an accessible area will also serve to bring down personnel exposures to be in line with the design goal. These administrative control practices are commonly used in the nuclear industry, provided that the occupational dose limits in 10CFR20.1201 such as 5 rem per year are met.

In summary, the design basis fuel specification for the waste packages provides coverage of 97% of the historical and projected PWR SNF population. With special considerations as discussed in this section, the remaining 3% can be accommodated without imposing undue restrictions on the fuel loading scheme.

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8. ATTACHMENTS

This calculation document includes one attachment:

ATTACHMENT I	Listing of Computer Files (1 page)

ATTACHMENT I
LISTING OF COMPUTER FILES

This attachment lists the input and output file names for the PATH and MCNP calculations. All input and output files are stored on an electronic medium (compact disk) in ASCII format as part of this attachment.

Date	Time	File Size	File Name
11/09/00	10:46a	2,275	pwrwp00.inp
11/09/00	10:47a	19,364	pwrwp00.out
11/09/00	09:36a	5,977	pwrwp01b.inp
11/09/00	09:37a	45,678	pwrwp01b.out
11/09/00	09:56a	5,978	pwrwp02b.inp
11/09/00	09:56a	45,342	pwrwp02b.out
11/09/00	10:01a	5,978	pwrwp03b.inp
11/09/00	10:02a	46,372	pwrwp03b.out
01/12/01	08:23a	5,977	pwrwp04b.inp
01/12/01	08:24a	46,835	pwrwp04b.out
11/09/00	10:04a	5,978	pwrwp05b.inp
11/09/00	10:05a	45,615	pwrwp05b.out
01/12/01	08:27a	5,977	pwrwp06b.inp
01/12/01	08:28a	47,280	pwrwp06b.out
01/12/01	08:34a	5,977	pwrwp07b.inp
01/12/01	08:35a	47,410	pwrwp07b.out
01/12/01	08:38a	5,978	pwrwp08b.inp
01/12/01	08:39a	46,759	pwrwp08b.out
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01/12/01	11:20a	49,469	pwrwp09b.out
01/12/01	11:21a	6,756	pwrwp10b.inp
01/12/01	11:22a	48,285	pwrwp10b.out
03/13/01	08:34a	7,477	wpgamma1
03/13/01	02:08p	82,708	wpgamma1.out
03/13/01	07:54a	8,216	wpneut1
03/13/01	08:29a	108,552	wpneut1.out

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
SPECIAL INSTRUCTION SHEET ok 4-23-01 ntc
Complete Only Applicable Items

1. QA: QA
Page: 1 of: 1

This is a placeholder page for records that cannot be scanned or microfilmed

2. Record Date 04/18/2001	3. Accession Number ATT TO MOL.20010423.0323
4. Author Name(s) S. SU	5. Author Organization N/A

6. Title
SUBSURFACE SHIELDING SOURCE TERM SPECIFICATION CALCULATION

7. Document Number(s) CAL-WER-NU-000003	8. Version REV. 00
9. Document Type DATA	10. Medium CD-ROM

11. Access Control Code
PUB

12. Traceability Designator
DC # 27458

13. Comments
THIS IS A SPECIAL PROCESS CD-ROM ENCLOSED AS PART OF ATTACHMENT 1, AND CAN BE LOCATED THROUGH THE RPC

NOTE: SEE ATTACHMENT OF ELECTRONIC SOURCE FILE VERIFICATION FORM PER AP-17.1Q/ICN 3, SECTION 5.1 (C), ELECTRONIC RECORDS

DC# 27458

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
ELECTRONIC SOURCE FILE VERIFICATION

QA: N/A

1. DOCUMENT TITLE: Subsurface Shielding Source Term Specification Calculation	
2. DOCUMENT IDENTIFIER: CAL-WER-NU-000003	3. REVISION DESIGNATOR: 00

ELECTRONIC SOURCE FILE INFORMATION

4. ELECTRONIC SOURCE FILE NAME WITH FILE EXTENSION PROVIDED BY THE SOFTWARE: source-term.doc		
5. DATE LAST MODIFIED: April 16, 2001	6. ELECTRONIC SOURCE FILE APPLICATION: (I.E., EXCEL, WORD, CORELDRAW) WORD	7. FILE SIZE IN KILOBYTES: 353 KB
8. FILE LINKAGE INSTRUCTIONS/INFORMATION: None		
9. FILE CUSTODIAN: (I.E., DC, OR DC APPROVED CUSTODIAN) DC	10. FILE LOCATION FOR DC APPROVED CUSTODIAN: (I.E., SERVER, DIRECTORY) N/A	
11. PRINTER SPECIFICATION (I.E., HP4SI) INCLUDING POSTSCRIPT INFORMATION (I.E., PRINTER DRIVER) AND PRINTING PAGE SETUP: (I.E., LANDSCAPE, 11 X 17 PAPER) Standard printer and page setup, Hewlett Packard LaserJet 5SiMX, T1005HP5SI		
12. COMPUTING PLATFORM USED: (I.E., SUN) Desktop personal computer	13. OPERATING EQUIPMENT USED: (I.E., UNIX, SOLARIS) Microsoft Windows NT 4.00	
14. ADDITIONAL HARDWARE/SOFTWARE REQUIREMENT USED TO CREATE FILE(S): None	15. ACCESS RESTRICTIONS: (IF ANY) None	

COMMENTS/SPECIAL INSTRUCTIONS

16.

None 3.5 for attachment #1 provided to DC

CERTIFICATION

17. NAME (Print and Sign) S. Su S. Su			18. DATE: 04/16/2001
19. ORGANIZATION: Subsurface Facilities	20. DEPARTMENT: Repository Design	21. LOCATION/MAILSTOP: 1031L/MS423	22. PHONE: 702-295-4555

DC USE ONLY

23. DATE RECEIVED: 04/18/01	24. DATE REVIEWED: 04-19-2001	25. DATE FILES TRANSFERRED: 04-19-2001
26. NAME (Print and Sign): Marina Blackwell Marina Blackwell		27. DATE: 04-19-2001