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LIST OF ACRONYMS AND ABBREVIATIONS

ANSI	American National Standards Institute
ANS	American Nuclear Society
B-poly	Borated polyethylene
BWR	Boiling Water Reactor
B&W	Babcock & Wilcox
cc	cubic centimeter
CDB	Characteristics Data Base
CFR	Code of Federal Regulations
Ci	Curies
cm	centimeters
CPU	Central Processing Unit
DBF	Design Basis Fuel
DC	Document Control
DIRS	Document Input Reference Sheet
DOE	U.S Department of Energy
DTN	Data Tracking Number
EDA	Enhanced Design Alternative
fsd	fractional standard deviation
g	grams
hr	hours
kg	kilograms
LA	License Application
MCNP	Monte Carlo Neutral Particle transport code
MeV	Million Electron Volts
mrem	millirem
MTU	Metric Tons of Uranium
MWd	Megawatt Days
n	neutrons
N/A	Not Applicable
PWR	Pressurized Water Reactor
rem	Roentgen Equivalent in Man
s or sec	seconds
SNF	Spent Nuclear Fuel
STN	Software Tracking Number
TBV	To Be Verified
TDMS	Technical Data Management System
TIC	Technical Information Center
VA	Viability Assessment
WP	Waste Package
WPO	Waste Package Operations
YMP	Yucca Mountain Site Characterization Project
yr	years
γ	Gamma-rays or photons

1. PURPOSE

The purpose of this work is to provide supporting calculations for determination of the radiation source terms specific to subsurface shielding design and analysis. These calculations are not intended to provide the absolute values of the source terms, which are under the charter of the Waste Package Operations (WPO) Group. Rather, the calculations focus on evaluation of the various combinations of fuel enrichment, burnup and cooling time for a given decay heat output, consistent with the waste package (WP) thermal design basis. The objective is to determine the worst-case combination of the fuel characteristics (enrichment, burnup and cooling time) which would give the maximum radiation fields for subsurface shielding considerations.

The calculations are limited to PWR fuel only, since the WP design is currently evolving with thinner walls and a reduced heat load as compared to the viability assessment (VA) reference design. The results for PWR fuel will provide a comparable indication of the trend for BWR fuel, as their characteristics are similar. The source term development for defense high-level waste and other spent nuclear fuel (SNF) is the responsibility of the WPO Group, and therefore, is not included this work.

This work includes the following items responsive to the stated purpose and objective:

- Determine the possible fuel parameters (initial enrichment, burnup and cooling time), that give the same decay heat value as specified for the waste package thermal design
- Obtain the neutron and gamma source terms for the various combinations of the fuel parameters for use in radiation field calculations
- Calculate radiation fields on the surfaces of the waste package and its transporter to quantify the effects of the fuel parameters with the same decay heat value for use in identifying the worst-case combination of the fuel parameters.

The technical product development plan for this calculation is provided in *Development Plan for Subsurface Shielding-specific Source Term Evaluation*, Document Identifier No.: TDP-WER-NU-000002 REV 00 (MOL.19990806.0065).

2. METHOD

The radiation source terms required for this work are based on the Characteristics Data Base (CDB) (DIRS 1). The CDB is currently unqualified per TBV-455, and is suitable for scoping calculations only as for this work. The WPO Group has performed a series of source term calculations (DIRS 2) with the SCALE code package in response to a QAP-3-12 design input request from the Repository Subsurface Design Department. However, the SCALE calculations required revisions, owing to major unresolved technical issues. Without the source data from SCALE, the CDB represents the best-documented information available.

The waste package design is currently undergoing significant changes in terms of barrier materials and thicknesses. The Enhanced Design Alternative II (EDA II) is the recommended design as a result of the license application selection process (DIRS 3). Therefore, this source term evaluation is based on the EDA II waste package design. Section 5.1.7 provides the design parameters used in this calculation.

The WP thermal design basis of 11.8 kW/WP for EDA II (DIRS 3, p. 2-1) is used to obtain the decay heat output per fuel assembly for the 21 PWR waste package. This basis translates into 562 watts per fuel assembly. Various combinations of initial enrichment, burnup and cooling time are developed to correspond to this heat output, based on the CDB.

For the source term evaluation, the worst-case combination of the fuel characteristics is determined to correspond to the maximum radiation fields calculated outside the waste package. The radiation field calculations include two extreme cases: one on the surface of the waste package, and the other on the transporter external surface. The WP surface represents the interface between the WPO and Repository Subsurface Design Groups. The transporter surface constitutes a deep-penetration shielding problem. These two extreme cases will determine whether one set of the worst-case fuel specifications is applicable to all dose points outside the waste package and beyond.

Radiation field calculations are performed with the qualified MCNP4B code (DIRS 4). The quantity of interest is the dose rate on the surfaces of the waste package as well as the transporter in units of rem/hr or mrem/hr. The geometric model for the MCNP4B calculations is borrowed from the previous analyses documented in DIRS 5. Modifications made to the existing model are described in Section 5.4.

3. ASSUMPTIONS

The following assumptions are used:

- 3.1 The waste package contains 21 PWR fuel assemblies.

Rationale: This assumption is consistent with the previous shielding analyses as the bounding case. The EDA II design also focuses on the 21 PWR waste package only. Other waste package designs for different waste forms remain to be developed for the license application.

Usage: This assumption is used in Section 5.4 for WP modeling.

- 3.2 The EDA II WP design consists of an inner barrier of 50 mm SS316NG and an outer barrier of 20 mm alloy C-22.

Rationale: This assumption is based on the license application design selection described in DIRS 3 (p. 2-1) for EDA II.

Usage: This assumption is used in Section 5.4 for WP modeling.

- 3.3 The WP thermal design basis is 11.8 kW/WP, representing the maximum heat load per waste package, which is 20% over the average heat load of 9.8 kW/WP.

Rationale: This assumption is based on the license application design selection described in DIRS 3 (p. 2-1) for EDA II.

Usage: This assumption is used in Section 5.2 for determination of the combinations of the enrichment, burnup and cooling time to produce this heat output.

- 3.4 Each PWR fuel assembly in the waste package generates a heat output of 562 watts.

Rationale: This heat output is simply equal to the WP thermal load of 11.8 kW divided by 21 fuel assemblies, implying that uniform mixing is assumed with the same heat generation rate for each assembly. It needs to be recognized that some assemblies may be hotter or colder, as long as the thermal design basis of 11.8 kW/WP is met. The variation in the heat output from one assembly to another would cause an azimuthal dose rate profile around the waste package. However, the purpose of this analysis is to determine the worst-case combination of enrichment, burnup and cooling which results in the highest dose rate for a given decay heat value.

Usage: This assumption is used in Section 5.2 to obtain the various combinations of enrichment, burnup and cooling time corresponding to 562 watts per assembly.

- 3.5 The neutron and gamma source terms for radiation field calculations are based on the Characteristics Data Base (CDB).

Rationale: Per DIRS 1, the CDB is acceptable for planning and scoping analyses. The WPO Group is still in the process of generating the source terms for the various waste forms and waste packages with the SCALE code package. A documented comparison between the CDB and SCALE has shown that the CDB is generally comparable to SCALE, or slightly overpredicts the radiation fields (DIRS 8, p. 43).

Usage: This assumption is used in Section 5.3 to obtain the neutron and gamma source terms for a given combination of enrichment, burnup and cooling time.

- 3.6 Uniform fuel burnup is assumed.

Rationale: The axial fuel burnup profile is not considered here, as this profile is usually included after the uniform burnup value is determined.

Usage: This assumption applies to the source terms provided in Section 5.3.

- 3.7 The neutron source spectrum is assumed to be the Cm-244 spontaneous fission spectrum, regardless of initial enrichment, fuel burnup and cooling time.

Rationale: This assumption is consistent with NUREG-1536 (DIRS 9, p. 5-3) by selecting the nuclide with the largest contribution to spontaneous fission (Cm-244) and using that spectrum for all neutrons, since the contribution from the alpha-neutron reactions is generally small.

Usage: This assumption is used in Section 5.1.3.

- 3.8 The minimum cooling time for commercial spent nuclear fuel (SNF) is 5 years after discharge from reactor.

Rationale: This assumption is consistent with the DOE standard contract for receipt of commercial SNF (10CFR961).

Usage: This assumption is used Section 5.2.

- 3.9 The fresh fuel composition with 4.2% initial enrichment is assumed for all cases.

Rationale: The enrichment range considered in this calculation is 2 to 5%. The enrichment will only slightly affect the neutron contribution. The 4.2% fresh fuel composition is available in DIRS 12. The assumption of 4.2% is near the high end of the range. Coupled with the fresh fuel assumption, the use of 4.2% is conservative for the neutron contribution due to secondary fission over the range of enrichment used.

Usage: This assumption is used in Section 5.1.6 (Table 5-4).

- 3.10 The concrete lining in the main drift is retained from the VA reference design for modeling purposes.

Rationale: EDA II has replaced the concrete lining with other type of ground support such as steel sets. Concrete is retained in this calculation to simulate the backscattering effect from the host rock, which is neutronicallly similar to concrete in terms of material composition and density.

Usage: This assumption is used in Sections 5.1.7 and 5.3 for modeling.

4. USE OF COMPUTER SOFTWARE AND MODELS

4.1 SOFTWARE APPROVED FOR QUALITY ASSURANCE WORK

As stated in Section 2, this calculation uses the MCNP4B code (Version 4B2, Software Tracking Number: 30033 V4B2LV). MCNP4B has been fully qualified and baselined on the PC platform (DIRS 6). The computer hardware used in this calculation is a DELL PowerEdge 2200 desktop personal computer (CRWMS-M&O Tag Number 112106). This computer has been qualified to perform execution of the MCNP4B software used in this calculation.

The qualified version of the MCNP4B code was obtained from Software Configuration Management. The MCNP4B computer software used in this calculation was appropriate per the capabilities and applications of the code, and was used within the range of validation in the software qualification report.

The MCNP4B input and output file names from this calculation are listed in Attachment I and stored on an electronic medium (DIRS 7).

4.2 SOFTWARE ROUTINES

In addition to the software described in Section 4.1, computational support software was used for spreadsheets. The spreadsheets were performed, using Microsoft Excel 97. The user-defined formulas and algorithms for the spreadsheets are described in Section 5.5.

In addition, the Characteristics Data Base (CDB) software (DIRS 1) was used in this calculation to generate the radiation source terms. As stated in Section 2, the CDB is currently unqualified, and to be verified (TBV-455).

4.3 MODELS

No models validated as per AP-3.10Q were used in this calculation.

5. CALCULATION

5.1 INPUT PARAMETERS

This calculation uses primary controlled sources of inputs and references from Technical Data Management System (TDMS), Technical Information Center (TIC) and Document Control (DC) as provided on Document Input Reference Sheets (DIRS) in Attachment II. The calculation contains the TBV inputs including TBV-326 and TBV-455. Selected input parameters have been entered in the TDMS as DTN: MO98WASTAN00011.000 and MO9905WHODANQ1.000.

5.1.1 Fuel Type

Consistent with the previous subsurface shielding analyses for the 21 PWR waste package, the fuel type used in this calculation is the B&W 15x15 Mark B fuel assemblies with a heavy metal loading of 0.464 MTU per assembly (DIRS 8, p.12).

5.1.2 Decay Heat

Per Assumption 3.4, the decay heat value per assembly is 562 watts for a WP thermal design basis of 11.8 kW/WP for EDA II (DIRS 3, p. 2-1). With 0.464 MTU per assembly, this heat output equals 1211 watts/MTU.

5.1.3 Normalized Neutron Source Spectrum

According to Assumption 3.7, the neutron source spectrum is insensitive to the various combinations of enrichment, burnup and cooling time. The same neutron source spectrum as taken from DIRS 8 (p. 13) and reproduced in Table 5-1 applies to all cases of the fuel parameters.

5.1.4 Important Gamma Energy Groups

This calculation covers the important contributing gamma energy groups ranging from 0.4 to 2.5 MeV, encompassing the specification in NUREG-1536 (DIRS 9, p. 5-3). Photons outside this energy range are of little significance or consequence, as demonstrated in the PATH calculations documented in Appendix IX of DIRS 8.

5.1.5 Flux-to-Dose-Rate Conversion Factors

The flux-to-dose-rate conversion factors in Tables 5-2 and 5-3 are used to convert the calculated neutron and gamma fluxes to dose rates. These factors are based on the ANSI/ANS-6.1.1-1977 Standard (DIRS 10). These parameters are used in Section 5.5 for input to the MCNP4B code.

Table 5-1. Normalized Neutron Source Spectrum

Neutron Energy Range (MeV)	Normalized Spectrum^a
6.43 - 20.0	0.0183
3.00 - 6.43	0.2099
1.85 - 3.00	0.2341
1.40 - 1.85	0.1310
0.90 - 1.40	0.1767
0.40 - 0.90	0.1924
0.10 - 0.40	0.0376
0.00 - 0.10	0.0
Total	1.0

^a From DIRS 8, p. 13.

Table 5-2. Neutron Flux-to-Dose Rate Conversion Factors

Neutron Energy (MeV)	Conversion Factors (rem/hr)/(n/cm ² -s) ^a
2.50E-08	3.67E-06
1.00E-07	3.67E-06
1.00E-06	4.46E-06
1.00E-05	4.54E-06
1.00E-04	4.18E-06
1.00E-03	3.76E-06
1.00E-02	3.56E-06
1.00E-01	2.17E-05
5.00E-01	9.26E-05
1.0	1.32E-04
2.5	1.25E-04
5.0	1.56E-04
7.0	1.47E-04
10.0	1.47E-04
14.0	2.08E-04
20.0	2.27E-04

^a DIRS 10, p.4.

Table 5-3. Gamma Flux-To-Dose Rate Conversion Factors

Photon Energy (MeV)	Conversion Factors (rem/hr)/(photon/cm ² -s) ^a
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

^a DIRS 10, p.5.

5.1.6 Material Data

The material data required in this calculation include the density and elemental composition for each material. These data are based on the standard material handbooks, supplier's information, and published YMP documents. Table 5-4 lists the pertinent material data used in this calculation.

5.1.7 Physical Parameters

The physical parameters for the WP radial model are taken from DIRS 3 (p. 2-1) with the following specifications:

WP inner barrier thickness = 50 mm (SS316NG)

WP outer barrier thickness = 20 mm (alloy C-22)

The physical parameters for the transporter radial model are taken directly from DIRS 5 (p. 30) without any modifications. The following radial shielding dimensions (DIRS 5, p. 24) are used:

Inner clad = 5 mm (SS316L)

Gamma shield = 152.4 mm (A516 carbon steel)

Neutron shield = 101.6 mm (borated polyethylene)

Outer clad = 5 mm (SS316L)

Other physical parameters used in this calculation include the following:

Main drift diameter = 7.62 m (excavated) (DIRS 11, p. 62 or 71)

Concrete lining = 0.3 m thick (Assumption 3.10) (DIRS 11, p. 56)

5.2 DETERMINATION OF FUEL PARAMETERS

In order to obtain the source terms for radiation field calculations, the fuel parameters including initial enrichment, burnup and cooling time need to be specified. These fuel parameters were determined for a given decay heat value of 562 watts/assembly or 1211 watts/MTU for a thermal load of 11.8 kW/WP (21 PWR waste package).

A search for the corresponding burnup and cooling time to produce 1211 watts/MTU was made for each of the following initial enrichments: 2.0%, 2.5%, 3.0%, 3.5%, 4.0%, 4.5% and 5.0%. The search used the CDB decay heat data (DIRS 1). The minimum cooling time used in this determination was 5 years, per Assumption 3.8. For a given enrichment, the range of the burnup values covered corresponded to that used in the CDB. In general, the allowable burnup increases with the enrichment, as more enrichment is required to achieve a higher burnup.

Table 5-5 lists the various combinations of the fuel parameters, which give 1211 ± 3 watts/MTU. The ± 3 watts/MTU value was used for search purposes with the burnup value rounded to the nearest hundred.

Table 5-4. Material Data

Material	Density (g/cc)	Element	Weight Percent
Smeared Fuel Region (DIRS 12, p. 9) (Assumption 3.9)	3.0431	N	0.005
		O	10.54
		Cr	0.017
		Fe	0.035
		Zr	17.14
		Sn	0.244
		U-235	3.025
		U-238	68.995
Fuel Basket Liner (DIRS 8, p. II-6)	8.0038	B-10	0.077
		B-11	0.351
		C	0.043
		N	0.033
		Al	0.133
		Si	0.584
		P	0.015
		S	0.030
		Ti	0.600
		Cr	20.686
		Mn	1.336
		Fe	39.270
		Ni	32.505
		Cu	1.499
		Mo	2.836
WP Inner Barrier (SS316NG) Transporter Clad (SS316L) Same composition for SS316NG and SS316L (DIRS 13, p.14)	7.9497	C	0.030
		N	0.100
		Si	0.750
		P	0.045
		S	0.030
		Cr	17.00
		Mn	2.000
		Fe	65.545
		Ni	12.00
		Mo	2.500
WP Outer Barrier Alloy C-22 (DIRS 14)	8.69	C	0.010
		Si	0.08
		P	0.02
		S	0.010
		V	0.35
		Cr	22
		Mn	0.50
		Fe	3
		Co	2.50
		Ni	55.53
		Mo	13
		W	3
Transporter Gamma Shield A516 Carbon Steel (DIRS 13, pp. 5 & 6)	7.832	C	0.220
		Si	0.275
		S	0.035
		P	0.035
		Mn	0.900
		Fe	98.535
Transporter Neutron Shield 1.5% B-poly ^a (DIRS 15, p. 5.3-6)	0.92	H	12.56
		B-10	0.278
		B-11	1.223
		C	85.94

Material	Density (g/cc)	Element	Weight Percent
Type 04 Concrete (ANSI/ANS 6.4-1985) ^b (DIRS 16, p. 9)	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
Air (DIRS 13, p. 26)	0.001225	N	80.0
		O	20.0

^a The published value in DIRS 15 is 1%, which is a typographical error, based on the weight percents given for B-10 and B-11 which add up to 1.5%.

^b Obtained by dividing the partial density given in DIRS 16, p. 9 by the density of the cured concrete (2.35 g/cc) to lead to the weight percent for each element. Example: For H, wt. % = $(0.013/2.35) \times 100 = 0.55$. Partial densities (g/cc) for other elements in Type 04 concrete are: O = 0.103+1.068 = 1.171, Si = 0.742, Ca = 0.194, Na = 0.040, Mg = 0.006, Al = 0.107, S = 0.003, K = 0.045, and Fe = 0.029 (DIRS 16, p. 9).

Table 5-5. Fuel Parameters and Source Terms

Burnup (MWd/MTU)	Cooling (years)	Heat (w/MTU)	Neutrons (n/s-MTU)	Fuel Gammas (photons/s-MTU)				
				0.575 MeV	0.85 MeV	1.25 MeV	1.75 MeV	2.25 MeV
E = 2.0% ^a								
21600	5	1214	1.208E+08	3.280E+15	6.206E+14	4.233E+14	5.329E+12	3.816E+12
E = 2.5%								
22300	5	1210	8.081E+07	3.300E+15	5.993E+14	3.917E+14	4.914E+12	3.836E+12
26000	6	1212	1.551E+08	3.479E+15	5.785E+14	4.005E+14	4.200E+12	1.756E+12
28600	7	1210	2.311E+08	3.508E+15	5.127E+14	3.884E+14	3.767E+12	8.067E+11
30500	8	1213	3.045E+08	3.474E+15	4.359E+14	3.671E+14	3.516E+12	3.715E+11
E = 3.0%								
23000	5	1209	5.466E+07	3.324E+15	5.767E+14	3.628E+14	4.554E+12	3.859E+12
26700	6	1210	1.081E+08	3.511E+15	5.593E+14	3.731E+14	3.927E+12	1.755E+12
29400	7	1212	1.654E+08	3.559E+15	4.991E+14	3.637E+14	3.565E+12	8.022E+11
31200	8	1212	2.131E+08	3.517E+15	4.210E+14	3.414E+14	3.319E+12	3.658E+11
33700	10	1211	2.949E+08	3.365E+15	2.802E+14	2.906E+14	3.007E+12	7.030E+10
37900	15	1212	4.399E+08	3.119E+15	1.220E+14	1.869E+14	2.508E+12	2.226E+09
E = 3.5%								
27400	6	1210	7.464E+07	3.541E+15	5.372E+14	3.465E+14	3.669E+12	1.755E+12
30100	7	1211	1.145E+08	3.596E+15	4.792E+14	3.380E+14	3.346E+12	7.974E+11
31800	8	1209	1.511E+08	3.551E+15	4.048E+14	3.182E+14	3.135E+12	3.608E+11
34400	10	1210	2.179E+08	3.416E+15	2.721E+14	2.732E+14	2.885E+12	6.860E+10
38700	15	1210	3.377E+08	3.178E+15	1.195E+14	1.775E+14	2.439E+12	2.100E+09
42100	20	1211	4.113E+08	3.013E+15	7.084E+13	1.156E+14	1.955E+12	4.448E+08
45400	25	1211	4.722E+08	2.871E+15	5.059E+13	7.756E+13	1.541E+12	2.628E+08
E = 4.0%								
32400 ^b	8	1212	1.130E+08	3.587E+15	3.922E+14	2.998E+14	2.994E+12	3.568E+11
35000	10	1210	1.594E+08	3.475E+15	2.623E+14	2.563E+14	2.753E+12	6.702E+10
39500	15	1210	2.597E+08	3.237E+15	1.167E+14	1.686E+14	2.372E+12	1.988E+09
43000	20	1210	3.236E+08	3.074E+15	6.972E+13	1.112E+14	1.917E+12	4.118E+08
46500	25	1213	3.817E+08	2.937E+15	5.040E+13	7.541E+13	1.531E+12	2.398E+08
49900	30	1211	4.293E+08	2.795E+15	3.905E+13	5.129E+13	1.203E+12	1.970E+08
E = 4.5%								
41000 ^c	16	1212	2.169E+08	3.258E+15	1.015E+14	1.485E+14	2.225E+12	1.136E+09
43900	20	1211	2.531E+08	3.134E+15	6.850E+13	1.067E+14	1.876E+12	3.829E+08
47500	25	1212	3.018E+08	2.997E+15	4.991E+13	7.288E+13	1.511E+12	2.191E+08
51000	30	1212	3.481E+08	2.853E+15	3.900E+13	5.003E+13	1.197E+12	1.791E+08
E = 5.0%								
52000	30	1210	2.812E+08	2.906E+15	3.885E+13	4.872E+13	1.188E+12	1.636E+08

^a Initial U-235 enrichment^b Exact enrichment in the CDB for this case is 3.97% (rounded to 4.0% here)^c Exact enrichment in the CDB for this case is 4.48% (rounded to 4.5% here)

5.3 RADIATION SOURCE TERMS

Once the fuel parameters (enrichment, burnup and cooling time) were specified, the radiation source terms were obtained from the CDB (DIRS 1), per Assumption 3.5. Table 5-5 lists the neutron and gamma source terms for the fuel region for each combination of enrichment, burnup and cooling time with a decay heat value of 1211 ± 3 watts/MTU. The source terms are directly from the CDB in terms of particles/s per MTU. The neutron source represents the total number of neutrons per second per MTU for all energy groups with the spectrum as given in Table 5-1. The gamma source terms are provided only for the five contributing energy groups with the average energies of 0.575, 0.85, 1.25, 1.75 and 2.25 MeV, respectively. The gamma source terms given in Table 5-5 are in units of photons/s per MTU.

5.4 GEOMETRIC MODEL

The WP transporter shielding model was borrowed from DIRS 5 (p. 30) with minor modifications to reflect the EDA II waste package design. The modifications include the changes in the materials and thicknesses for the WP inner and outer barriers as given in Section 5.1.7. The model used a unit axial segment of 10 cm out of the full active fuel length of 360.18 cm with reflective boundaries on both ends to simulate the infinite length. Figure 5-1 illustrates the radial geometric model of the waste package and its transporter used for shielding calculations.

The model simulates the condition when the transporter travels in the main drift. The excavated diameter of the main drift is 7.62 m with a concrete lining thickness of 0.3 m (Section 5.1.7). The concrete lining is included in the model to account for drift wall scattering effects.

5.5 SHIELDING CALCULATIONS

The shielding analysis was performed with the MCNP4B code, including both neutron and gamma radiation fields. Two separate MCNP4B calculations were executed: one for the neutron and secondary gamma contributions, and the other for the primary gamma contributions from the active fuel region. The material compositions and densities for the various materials are taken from Table 5-4 for input to the MCNP4B code.

Instead of running MCNP4B for each set of enrichment, burnup and cooling time, the calculations were simplified by performing MCNP4B for a unit source. The result for the unit source was then used as a multiplier to obtain the absolute result. This calculational approach was implemented with the following five MCNP4B runs:

- Neutron and secondary gamma run per unit neutron source
- Gamma run per unit photon source for 0.575 MeV (average) photons
- Gamma run per unit photon source for 0.85 MeV (average) photons
- Gamma run per unit photon source for 1.25 MeV (average) photons
- Gamma run per unit photon source for 1.75 MeV (average) photons
- Gamma run per unit photon source for 2.25 MeV (average) photons

Figure 5-1. Waste Package (EDA II Design) Transporter Radial Shield Model

The results from these MCNP4B runs represent normalized results in units of rem/hr or mrem/hr per unit source (n/s or photons/s). The corresponding absolute result is simply the product of the normalized result and the number of source particles in the model.

The normalized results as determined with MCNP4B are listed in Table 5-6, along with the associated input and output MCNP4B files.

The absolute contribution to the dose rate from each source component is calculated as follows:

$$D_i = N_i \times S_i \times 0.464 \times 21 \times (10/360.18) = 0.2705N_iS_i, \text{ where}$$

D_i = dose rate contribution from source component i , rem/hr or mrem/hr

N_i = normalized result for source component i , rem/hr or mrem/hr per unit source

S_i = source strength for source component i , particles/s-MTU (Table 5-5)

0.464 = heavy metal loading, MTU/assembly (Section 5.1.1)

21 = number of fuel assemblies per waste package (Assumption 3.1)

(10/360.18) = fraction of the source height used in the model (Section 5.4)

The source components include neutrons, secondary gammas, and fuel gammas at 0.85, 1.25, 1.75 and 2.25 MeV as given in Table 5-6, along with the corresponding normalized results from the MCNP4B code. The one-sigma statistical error is about 1% or less in all cases. The input and output files associated with the MCNP4B calculations are listed in Attached I, and located in DIRS 7.

The total dose rate from all source components is then

$$D = \sum D_i$$

with summation over all contributing source components.

Table 5-7 provides the spreadsheet calculations of the dose rates on the WP surface, using Microsoft Excel 97.

Table 5-8 provides the spreadsheet calculations of the dose rates on the transporter surface, using Microsoft Excel 97.

Tables 5-7 and 5-8 include a special case at the end of each table for the VA design basis fuel (DBF) with 4.2% initial enrichment, 48086 MWd/MTU and 10 years cooling (DIRS 8, p. 12) for comparison purposes. For the DBF, the dose rate on the surface of the waste package increases from 38 rem/hr (DIRS 8, p. 96) for the VA reference design to 291 rem/hr (Table 5-7) for the EDA II design. On the transporter surface, the corresponding increase is from 31 mrem/hr (DIRS 5, p. 35) to 141 mrem/hr (Table 5-8). This increase is due to the effect of thinner WP walls for the EDA II design.

Table 5-6. Normalized Dose Rate Results

Source Component	Normalized Dose Rate		Input File	Output File
	WP Surface (rem/hr per unit source) ^a	Transporter Surface (mrem/hr per unit source) ^a		
Neutron	3.900E-08 (0.0028) ^b	8.430E-08 (0.0082) ^b	nsource	nsource.out
Secondary γ	2.153E-10 (0.0076)	4.574E-08 (0.0087)	nsource	nsource.out
0.575 MeV γ	7.346E-14 (0.0015)	3.422E-15 (0.0114)	g575kev	g575kev.out
0.85 MeV γ	3.590E-13 (0.0020)	6.831E-14 (0.0076)	gamma085	gamma085.out
1.25 MeV γ	1.330E-12 (0.0021)	9.338E-13 (0.0065)	gamma125	Gamma125.out
1.75 MeV γ	3.198E-12 (0.0025)	5.228E-12 (0.0056)	gamma175	Gamma175.out
2.25 MeV γ	5.417E-12 (0.0027)	1.554E-11 (0.0050)	gamma225	Gamma225.out

^a unit source = 1 n/s-MTU for neutron and secondary gamma source components, and 1 photons/s-MTU for 0.575, 0.85, 1.25, 1.75 and 2.25 MeV gamma source component.

^b The value in parentheses denotes the one-sigma statistical uncertainty in terms of fractional standard deviation (fsd).

Table 5-7. Waste Package Surface Dose Rate Calculations

Burnup (MWd/MTU)	Cooling (years)	Calculation ^b									
		Quantity	Neutrons	Second. Gammas	Fuel Gammas						TOTAL
					0.575 MeV	0.85 MeV	1.25 MeV	1.75 MeV	2.25 MeV	Subtotal	
E = 2.0% ^a											
21600	5	particles/s-MTU	1.208E+08	1.208E+08	3.280E+15	6.206E+14	4.233E+14	5.329E+12	3.816E+12		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	1.27	0.01	65.18	60.27	152.29	4.61	5.59	287.93	289.21
E = 2.5%											
22300	5	particles/s-MTU	8.081E+07	8.081E+07	3.300E+15	5.993E+14	3.917E+14	4.914E+12	3.836E+12		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	0.85	0.00	65.57	58.20	140.92	4.25	5.62	274.56	275.42
26000	6	particles/s-MTU	1.551E+08	1.551E+08	3.479E+15	5.785E+14	4.005E+14	4.200E+12	1.756E+12		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	1.64	0.01	69.13	56.18	144.09	3.63	2.57	275.60	277.25
28600	7	particles/s-MTU	2.311E+08	2.311E+08	3.508E+15	5.127E+14	3.884E+14	3.767E+12	8.067E+11		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	2.44	0.01	69.71	49.79	139.73	3.26	1.18	263.67	266.12
30500	8	particles/s-MTU	3.045E+08	3.045E+08	3.474E+15	4.359E+14	3.671E+14	3.516E+12	3.715E+11		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	3.21	0.02	69.03	42.33	132.07	3.04	0.54	247.02	250.25
E = 3.0%											
23000	5	particles/s-MTU	5.466E+07	5.466E+07	3.324E+15	5.767E+14	3.628E+14	4.554E+12	3.859E+12		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	0.58	0.00	66.05	56.00	130.52	3.94	5.65	262.17	262.75
26700	6	particles/s-MTU	1.081E+08	1.081E+08	3.511E+15	5.593E+14	3.731E+14	3.927E+12	1.755E+12		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	1.14	0.01	69.77	54.31	134.23	3.40	2.57	264.28	265.42
29400	7	particles/s-MTU	1.654E+08	1.654E+08	3.559E+15	4.991E+14	3.637E+14	3.565E+12	8.022E+11		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	1.74	0.01	70.72	48.47	130.85	3.08	1.18	254.29	256.05
31200	8	particles/s-MTU	2.131E+08	2.131E+08	3.517E+15	4.210E+14	3.414E+14	3.319E+12	3.658E+11		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	2.25	0.01	69.89	40.88	122.82	2.87	0.54	237.00	239.26
33700	10	particles/s-MTU	2.949E+08	2.949E+08	3.365E+15	2.802E+14	2.906E+14	3.007E+12	7.030E+10		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	3.11	0.02	66.87	27.21	104.55	2.60	0.10	201.33	204.46
37900	15	particles/s-MTU	4.399E+08	4.399E+08	3.119E+15	1.220E+14	1.869E+14	2.508E+12	2.226E+09		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	4.64	0.03	61.98	11.85	67.24	2.17	0.00	143.24	147.90

^a Initial U-235 enrichment.^b rem/hr = 0.2705 * (rem/hr per unit source) * (particles/s-MTU) (Section 5.5)

Table 5-7. Waste Package Surface Dose Rate Calculations (Continued)

Burnup (MWd/MTU)	Cooling (years)	Calculation									TOTAL
		Quantity	Neutrons	Second. Gammas	Fuel Gammas						
					0.575 MeV	0.85 MeV	1.25 MeV	1.75 MeV	2.25 MeV	Subtotal	
E = 3.5%											
27400	6	particles/s-MTU	7.464E+07	7.464E+07	3.541E+15	5.372E+14	3.465E+14	3.669E+12	1.755E+12		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	0.79	0.00	70.36	52.17	124.66	3.17	2.57	252.93	253.73
30100	7	particles/s-MTU	1.145E+08	1.145E+08	3.596E+15	4.792E+14	3.380E+14	3.346E+12	7.974E+11		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	1.21	0.01	71.46	46.53	121.60	2.89	1.17	243.65	244.87
31800	8	particles/s-MTU	1.511E+08	1.511E+08	3.551E+15	4.048E+14	3.182E+14	3.135E+12	3.608E+11		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	1.59	0.01	70.56	39.31	114.48	2.71	0.53	227.59	229.19
34400	10	particles/s-MTU	2.179E+08	2.179E+08	3.416E+15	2.721E+14	2.732E+14	2.885E+12	6.860E+10		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	2.30	0.01	67.88	26.42	98.29	2.50	0.10	195.19	197.50
38700	15	particles/s-MTU	3.377E+08	3.377E+08	3.178E+15	1.195E+14	1.775E+14	2.439E+12	2.100E+09		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	3.56	0.02	63.15	11.60	63.86	2.11	0.00	140.73	144.31
42100	20	particles/s-MTU	4.113E+08	4.113E+08	3.013E+15	7.084E+13	1.156E+14	1.955E+12	4.448E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	4.34	0.02	59.87	6.88	41.59	1.69	0.00	110.03	114.39
45400	25	particles/s-MTU	4.722E+08	4.722E+08	2.871E+15	5.059E+13	7.756E+13	1.541E+12	2.628E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	4.98	0.03	57.05	4.91	27.90	1.33	0.00	91.20	96.21
E = 4.0%											
32400	8	particles/s-MTU	1.130E+08	1.130E+08	3.587E+15	3.922E+14	2.998E+14	2.994E+12	3.568E+11		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	1.19	0.01	71.28	38.09	107.86	2.59	0.52	220.33	221.53
35000	10	particles/s-MTU	1.594E+08	1.594E+08	3.475E+15	2.623E+14	2.563E+14	2.753E+12	6.702E+10		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	1.68	0.01	69.05	25.47	92.21	2.38	0.10	189.21	190.90
39500	15	particles/s-MTU	2.597E+08	2.597E+08	3.237E+15	1.167E+14	1.686E+14	2.372E+12	1.988E+09		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	2.74	0.02	64.32	11.33	60.66	2.05	0.00	138.37	141.12
43000	20	particles/s-MTU	3.236E+08	3.236E+08	3.074E+15	6.972E+13	1.112E+14	1.917E+12	4.118E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	3.41	0.02	61.08	6.77	40.01	1.66	0.00	109.52	112.95
46500	25	particles/s-MTU	3.817E+08	3.817E+08	2.937E+15	5.040E+13	7.541E+13	1.531E+12	2.398E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	4.03	0.02	58.36	4.89	27.13	1.32	0.00	91.71	95.76
49900	30	particles/s-MTU	4.293E+08	4.293E+08	2.795E+15	3.905E+13	5.129E+13	1.203E+12	1.970E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	4.53	0.03	55.54	3.79	18.45	1.04	0.00	78.82	83.38

Table 5-7. Waste Package Surface Dose Rate Calculations (Continued)

Burnup (MWd/MTU)	Cooling (years)	Calculation									TOTAL
		Quantity	Neutrons	Second. Gammas	Fuel Gammas						
					0.575 MeV	0.85 MeV	1.25 MeV	1.75 MeV	2.25 MeV	Subtotal	
E = 4.5%											
41000	16	particles/s-MTU	2.169E+08	2.169E+08	3.258E+15	1.015E+14	1.485E+14	2.225E+12	1.136E+09		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	2.29	0.01	64.74	9.86	53.43	1.92	0.00	129.95	132.25
43900	20	particles/s-MTU	2.531E+08	2.531E+08	3.134E+15	6.850E+13	1.067E+14	1.876E+12	3.829E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	2.67	0.01	62.28	6.65	38.39	1.62	0.00	108.94	111.62
47500	25	particles/s-MTU	3.018E+08	3.018E+08	2.997E+15	4.991E+13	7.288E+13	1.511E+12	2.191E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	3.18	0.02	59.55	4.85	26.22	1.31	0.00	91.93	95.13
51000	30	particles/s-MTU	3.481E+08	3.481E+08	2.853E+15	3.900E+13	5.003E+13	1.197E+12	1.791E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	3.67	0.02	56.69	3.79	18.00	1.04	0.00	79.51	83.21
E = 5.0%											
52000	30	particles/s-MTU	2.812E+08	2.812E+08	2.906E+15	3.885E+13	4.872E+13	1.188E+12	1.636E+08		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	2.97	0.02	57.74	3.77	17.53	1.03	0.00	80.07	83.06
E = 4.2%											
48086	10	particles/s-MTU	6.740E+08	6.740E+08	4.798E+15	4.671E+14	3.849E+14	4.879E+12	7.693E+10		
		rem/hr per unit source	3.900E-08	2.153E-10	7.346E-14	3.590E-13	1.330E-12	3.198E-12	5.417E-12		
		rem/hr	7.11	0.04	95.34	45.36	138.47	4.22	0.11	283.51	290.66

Table 5-8. Transporter Surface Dose Rate Calculations

Burnup (MWd/MTU)	Cooling (years)	Calculation ^b									
		Quantity	Neutrons	Second. Gammas	Fuel Gammas						TOTAL
					0.575 MeV	0.85 MeV	1.25 MeV	1.75 MeV	2.25 MeV	Subtotal	
E = 2.0% ^a											
21600	5	particles/s-MTU	1.208E+08	1.208E+08	3.280E+15	6.206E+14	4.233E+14	5.329E+12	3.816E+12		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	2.75	1.49	3.04	11.47	106.92	7.54	16.04	145.00	149.25
E = 2.5%											
22300	5	particles/s-MTU	8.081E+07	8.081E+07	3.300E+15	5.993E+14	3.917E+14	4.914E+12	3.836E+12		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	1.84	1.00	3.05	11.07	98.94	6.95	16.12	136.14	138.99
26000	6	particles/s-MTU	1.551E+08	1.551E+08	3.479E+15	5.785E+14	4.005E+14	4.200E+12	1.756E+12		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	3.54	1.92	3.22	10.69	101.16	5.94	7.38	128.39	133.85
28600	7	particles/s-MTU	2.311E+08	2.311E+08	3.508E+15	5.127E+14	3.884E+14	3.767E+12	8.067E+11		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	5.27	2.86	3.25	9.47	98.11	5.33	3.39	119.55	127.68
30500	8	particles/s-MTU	3.045E+08	3.045E+08	3.474E+15	4.359E+14	3.671E+14	3.516E+12	3.715E+11		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	6.94	3.77	3.22	8.05	92.73	4.97	1.56	110.53	121.24
E = 3.0%											
23000	5	particles/s-MTU	5.466E+07	5.466E+07	3.324E+15	5.767E+14	3.628E+14	4.554E+12	3.859E+12		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	1.25	0.68	3.08	10.66	91.64	6.44	16.22	128.04	129.96
26700	6	particles/s-MTU	1.081E+08	1.081E+08	3.511E+15	5.593E+14	3.731E+14	3.927E+12	1.755E+12		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	2.47	1.34	3.25	10.33	94.24	5.55	7.38	120.76	124.56
29400	7	particles/s-MTU	1.654E+08	1.654E+08	3.559E+15	4.991E+14	3.637E+14	3.565E+12	8.022E+11		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	3.77	2.05	3.29	9.22	91.87	5.04	3.37	112.80	118.62
31200	8	particles/s-MTU	2.131E+08	2.131E+08	3.517E+15	4.210E+14	3.414E+14	3.319E+12	3.658E+11		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	4.86	2.64	3.26	7.78	86.24	4.69	1.54	103.50	111.00
33700	10	particles/s-MTU	2.949E+08	2.949E+08	3.365E+15	2.802E+14	2.906E+14	3.007E+12	7.030E+10		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	6.72	3.65	3.11	5.18	73.40	4.25	0.30	86.24	96.62
37900	15	particles/s-MTU	4.399E+08	4.399E+08	3.119E+15	1.220E+14	1.869E+14	2.508E+12	2.226E+09		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	10.03	5.44	2.89	2.25	47.21	3.55	0.01	55.91	71.38

^a Initial U-235 enrichment.^b mrem/hr = 0.2705 * (mrem/hr per unit source) * (particles/s-MTU) (Section 5.5)

Table 5-8. Transporter Surface Dose Rate Calculations (Continued)

Burnup (MWd/MTU)	Cooling (years)	Calculation									
		Quantity	Neutrons	Second. Gammas	Fuel Gammas						TOTAL
					0.575 MeV	0.85 MeV	1.25 MeV	1.75 MeV	2.25 MeV	Subtotal	
E = 3.5%											
27400	6	particles/s-MTU	7.464E+07	7.464E+07	3.541E+15	5.372E+14	3.465E+14	3.669E+12	1.755E+12		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	1.70	0.92	3.28	9.93	87.52	5.19	7.38	113.29	115.92
30100	7	particles/s-MTU	1.145E+08	1.145E+08	3.596E+15	4.792E+14	3.380E+14	3.346E+12	7.974E+11		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	2.61	1.42	3.33	8.85	85.38	4.73	3.35	105.64	109.67
31800	8	particles/s-MTU	1.511E+08	1.511E+08	3.551E+15	4.048E+14	3.182E+14	3.135E+12	3.608E+11		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	3.45	1.87	3.29	7.48	80.38	4.43	1.52	97.09	102.41
34400	10	particles/s-MTU	2.179E+08	2.179E+08	3.416E+15	2.721E+14	2.732E+14	2.885E+12	6.860E+10		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	4.97	2.70	3.16	5.03	69.01	4.08	0.29	81.57	89.23
38700	15	particles/s-MTU	3.377E+08	3.377E+08	3.178E+15	1.195E+14	1.775E+14	2.439E+12	2.100E+09		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	7.70	4.18	2.94	2.21	44.84	3.45	0.01	53.44	65.32
42100	20	particles/s-MTU	4.113E+08	4.113E+08	3.013E+15	7.084E+13	1.156E+14	1.955E+12	4.448E+08		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	9.38	5.09	2.79	1.31	29.20	2.76	0.00	36.06	50.53
45400	25	particles/s-MTU	4.722E+08	4.722E+08	2.871E+15	5.059E+13	7.756E+13	1.541E+12	2.628E+08		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	10.77	5.84	2.66	0.93	19.59	2.18	0.00	25.36	41.97
E = 4.0%											
32400	8	particles/s-MTU	1.130E+08	1.130E+08	3.587E+15	3.922E+14	2.998E+14	2.994E+12	3.568E+11		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	2.58	1.40	3.32	7.25	75.73	4.23	1.50	92.03	96.00
35000	10	particles/s-MTU	1.594E+08	1.594E+08	3.475E+15	2.623E+14	2.563E+14	2.753E+12	6.702E+10		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	3.63	1.97	3.22	4.85	64.74	3.89	0.28	76.98	82.58
39500	15	particles/s-MTU	2.597E+08	2.597E+08	3.237E+15	1.167E+14	1.686E+14	2.372E+12	1.988E+09		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	5.92	3.21	3.00	2.16	42.59	3.35	0.01	51.10	60.24
43000	20	particles/s-MTU	3.236E+08	3.236E+08	3.074E+15	6.972E+13	1.112E+14	1.917E+12	4.118E+08		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	7.38	4.00	2.85	1.29	28.09	2.71	0.00	34.93	46.32
46500	25	particles/s-MTU	3.817E+08	3.817E+08	2.937E+15	5.040E+13	7.541E+13	1.531E+12	2.398E+08		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	8.70	4.72	2.72	0.93	19.05	2.17	0.00	24.86	38.29
49900	30	particles/s-MTU	4.293E+08	4.293E+08	2.795E+15	3.905E+13	5.129E+13	1.203E+12	1.970E+08		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	9.79	5.31	2.59	0.72	12.96	1.70	0.00	17.97	33.07

Table 5-8. Transporter Surface Dose Rate Calculations (Continued)

Burnup (MWd/MTU)	Cooling (years)	Calculation									
		Quantity	Neutrons	Second. Gammas	Fuel Gammas						TOTAL
					0.575 MeV	0.85 MeV	1.25 MeV	1.75 MeV	2.25 MeV	Subtotal	
E = 4.5%											
41000	16	particles/s-MTU	2.169E+08	2.169E+08	3.258E+15	1.015E+14	1.485E+14	2.225E+12	1.136E+09		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	4.95	2.68	3.02	1.88	37.51	3.15	0.00	45.55	53.18
43900	20	particles/s-MTU	2.531E+08	2.531E+08	3.134E+15	6.850E+13	1.067E+14	1.876E+12	3.829E+08		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	5.77	3.13	2.90	1.27	26.95	2.65	0.00	33.77	42.68
47500	25	particles/s-MTU	3.018E+08	3.018E+08	2.997E+15	4.991E+13	7.288E+13	1.511E+12	2.191E+08		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	6.88	3.73	2.77	0.92	18.41	2.14	0.00	24.24	34.86
51000	30	particles/s-MTU	3.481E+08	3.481E+08	2.853E+15	3.900E+13	5.003E+13	1.197E+12	1.791E+08		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	7.94	4.31	2.64	0.72	12.64	1.69	0.00	17.69	29.94
E = 5.0%											
52000	30	particles/s-MTU	2.812E+08	2.812E+08	2.906E+15	3.885E+13	4.872E+13	1.188E+12	1.638E+08		
		rem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		rem/hr	6.41	3.48	2.69	0.72	12.31	1.68	0.00	17.39	27.29
E = 4.2%											
48086	10	particles/s-MTU	6.740E+08	6.740E+08	4.798E+15	4.671E+14	3.849E+14	4.879E+12	7.693E+10		
		mrem/hr per unit source	8.430E-08	4.574E-08	3.422E-15	6.831E-14	9.338E-13	5.228E-12	1.554E-11		
		mrem/hr	15.37	8.34	4.44	8.63	97.22	6.90	0.32	117.52	141.23

6. RESULTS

The results of this calculation are based on the TBV inputs including TBV-326 and TBV-455. Therefore, the calculation contains information and assumptions that must be confirmed prior to the use of the results of the calculation.

6.1 WP SURFACE DOSE RATE

As summarized from Table 5-7, Table 6-1 presents the WP surface dose rate result for each combination of fuel enrichment, burnup and cooling time, which gives 562 watts/assembly or 1211 watts/MTU for a thermal load of 11.8 kW/WP. The results are also plotted in Figure 6-1. Note that the WP surface dose rates are in units of rem/hr. Results indicate that the dose rate is sensitive to the fuel parameters.

6.2 TRANSPORTER SURFACE DOSE RATE

As summarized from Table 5-8, Table 6-2 presents the corresponding transporter surface dose rate results. The results are also plotted in Figure 6-2. Note that the transporter surface dose rates are in units of mrem/hr. Similar to the WP surface dose rate, the transporter surface dose rate varies significantly with the fuel parameters.

6.3 WORST-CASE COMBINATION OF FUEL PARAMETERS

It is apparent from Table 6-1 that for a given decay heat value, the highest WP surface dose rate is associated with the fuel parameters with the lowest enrichment, lowest burnup and shortest cooling time. The same trend is observed in Table 6-2 for the transporter surface dose rate. Therefore, one set of the fuel parameters is applicable for all dose points outside the waste package and beyond.

6.4 FUEL PARAMETERS FOR SUBSURFACE SHIELDING SOURCE TERMS

The worst-case fuel parameters described in Section 6.3 represent the upper bound source terms. However, the number of fuel assemblies, which fit in the category of the worst-case fuel parameters may be extremely limited. Selection of the appropriate fuel parameters for subsurface shielding source terms would require performance of a waste stream analysis. Such analysis would evaluate the distribution of the existing and projected fuel assembly inventory by enrichment, burnup and cooling time. The waste stream analysis will be the subject of a separate evaluation by the WPO Group.

6.5 MID-POINT DOSE RATE VALUES

Over the various combinations of the fuel parameters evaluated, the mid-point dose rate values are 186 rem/hr on the WP surface, and 88.3 mrem/hr on the transporter surface (average of the maximum and minimum values in the range considered). These values approximately correspond to 3.5% enrichment, 34400 MWd/MTU and 10 years cooling, or an equivalent combination. These fuel parameters are fairly comparable to those used in the GA-4 spent fuel transportation cask design (DIRS 15, p. 5.2-1).

Table 6-1. Waste Package Surface Dose Rates

Enrichment (% U-235)	Burnup (MWd/MTU)	Cooling (years)	Dose Rate (rem/hr)			
			Neutron	Second. γ	Fuel γ	Total
2.0	21600	5	1.27	0.01	287.93	289.21
2.5	22300	5	0.85	0.00	274.56	275.42
2.5	26000	6	1.64	0.01	275.60	277.25
2.5	28600	7	2.44	0.01	263.67	266.12
2.5	30500	8	3.21	0.02	247.02	250.25
3.0	23000	5	0.58	0.00	262.17	262.75
3.0	26700	6	1.14	0.01	264.28	265.42
3.0	29400	7	1.74	0.01	254.29	256.05
3.0	31200	8	2.25	0.01	237.00	239.26
3.0	33700	10	3.11	0.02	201.33	204.46
3.0	37900	15	4.64	0.03	143.24	147.90
3.5	27400	6	0.79	0.00	252.93	253.73
3.5	30100	7	1.21	0.01	243.65	244.87
3.5	31800	8	1.59	0.01	227.59	229.19
3.5	34400	10	2.30	0.01	195.19	197.50
3.5	38700	15	3.56	0.02	140.73	144.31
3.5	42100	20	4.34	0.02	110.03	114.39
3.5	45400	25	4.98	0.03	91.20	96.21
4.0	32400	8	1.19	0.01	220.33	221.53
4.0	35000	10	1.68	0.01	189.21	190.90
4.0	39500	15	2.74	0.02	138.37	141.12
4.0	43000	20	3.41	0.02	109.52	112.95
4.0	46500	25	4.03	0.02	91.71	95.76
4.0	49900	30	4.53	0.03	78.82	83.38
4.5	41000	16	2.29	0.01	129.95	132.25
4.5	43900	20	2.67	0.01	108.94	111.62
4.5	47500	25	3.18	0.02	91.93	95.13
4.5	51000	30	3.67	0.02	79.51	83.21
5.0	52000	30	2.97	0.02	80.07	83.06

PWR Heat = 562 w/assembly
 21 PWR Waste Package
 Total Heat Load = 11.8 kW/WP

Figure 6-1. EDA II Waste Package Surface Dose Rate

Table 6-2. Transporter Surface Dose Rates

Enrichment (% U-235)	Burnup (MWd/MTU)	Cooling (years)	Dose Rate (mrem/hr)			
			Neutron	Second. γ	Fuel γ	Total
2.0	21600	5	2.75	1.49	145.00	149.25
2.5	22300	5	1.84	1.00	136.14	138.99
2.5	26000	6	3.54	1.92	128.39	133.85
2.5	28600	7	5.27	2.86	119.55	127.68
2.5	30500	8	6.94	3.77	110.53	121.24
3.0	23000	5	1.25	0.68	128.04	129.96
3.0	26700	6	2.47	1.34	120.76	124.56
3.0	29400	7	3.77	2.05	112.80	118.62
3.0	31200	8	4.86	2.64	103.50	111.00
3.0	33700	10	6.72	3.65	86.24	96.62
3.0	37900	15	10.03	5.44	55.91	71.38
3.5	27400	6	1.70	0.92	113.29	115.92
3.5	30100	7	2.61	1.42	105.64	109.67
3.5	31800	8	3.45	1.87	97.09	102.41
3.5	34400	10	4.97	2.70	81.57	89.23
3.5	38700	15	7.70	4.18	53.44	65.32
3.5	42100	20	9.38	5.09	36.06	50.53
3.5	45400	25	10.77	5.84	25.36	41.97
4.0	32400	8	2.58	1.40	92.03	96.00
4.0	35000	10	3.63	1.97	76.98	82.58
4.0	39500	15	5.92	3.21	51.10	60.24
4.0	43000	20	7.38	4.00	34.93	46.32
4.0	46500	25	8.70	4.72	24.86	38.29
4.0	49900	30	9.79	5.31	17.97	33.07
4.5	41000	16	4.95	2.68	45.55	53.18
4.5	43900	20	5.77	3.13	33.77	42.68
4.5	47500	25	6.88	3.73	24.24	34.86
4.5	51000	30	7.94	4.31	17.69	29.94
5.0	52000	30	6.41	3.48	17.39	27.29

Figure 6-2. EDA II WP Transporter Surface Dose Rate

7. ATTACHMENTS

There are two attachments to this calculation as described below:

Attachment I List of MCNP4B input and output files (1 page)

Attachment II Document Input Reference Sheets (7 pages)

ATTACHMENT I

LIST OF MCNP4B INPUT AND OUTPUT FILES

This attachment lists the input and output file names for the MCNP4B calculations in support of the subsurface shielding-specific source term evaluation. Table 5-6 provides a brief description of these files. DIRS 7 stores all the input and output files on an electronic medium.

Volume in drive D is DELLWIN95-2
Volume Serial Number is 1BE3-2E60
Directory of D:\MCNP4B\source01

```

.                <DIR>                08-19-99  7:33a  .
..               <DIR>                08-19-99  7:33a  ..
NSOURCE          7,693  06-23-99  1:47p  nsource
NSOURCE  OUT      92,515  06-23-99  5:49p  nsource.out
G575KEV          7,096  08-16-99 10:44a  g575kev
G575KEV  OUT      83,873  08-17-99  4:55a  g575kev.out
GAMMA085         7,123  06-23-99  1:47p  gamma085
GAMMA085  OUT     71,091  06-23-99  9:49p  gamma085.out
GAMMA125         7,118  06-23-99  1:43p  gamma125
GAMMA125  OUT     73,787  06-24-99 12:20a  gamma125.out
GAMMA175         7,118  06-23-99  1:45p  gamma175
GAMMA175  OUT     65,090  06-24-99  2:01a  gamma175.out
GAMMA225         7,118  06-23-99  1:46p  gamma225
GAMMA225  OUT     65,288  06-24-99  3:42a  gamma225.out
FILE      LST          0  08-19-99  7:36a  file.lst
13 file(s)                494,910 bytes
 2 dir(s)   1,626,898,432 bytes free

```

ATTACHMENT II
DOCUMENT INPUT REFERENCE SHEETS (DIRS)

This attachment contains the Document Input Reference Sheets (DIRS) for the controlled sources of inputs and references including the following:

Technical Data Management System (TDMS): Data sets with DTN

Technical Information Center (TIC): References with TIC catalog numbers

Document Control (DC): Approved project documents with accession numbers.

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DOCUMENT INPUT REFERENCE SHEET

1. Document Identifier No./Rev.:		Change:	Title:						
CAL-WER-NU-000001/00		N/A	Subsurface Shielding-specific Source Term Evaluation Page II-2 of II-7						
Input Document		3. Section	4. Input Status	5. Section Used in	6. Input Description	7. TBV/TBD Priority	8. TBV Due To		
2. Technical Product Input Source Title and Identifier(s) with Version	Unqual.						From Uncontrolled Source	Un-confirmed	
2a	Oak Ridge National Laboratory 1992. <i>Characteristics of Potential Repository Wastes</i> . Vol. 1, 2, 3 and 4. DOE/RW-0184-R1. Oak Ridge, Tennessee: Martin Marietta Energy Systems, Inc. TIC: 204940.	2.4	TBV-455	5.2	Spent fuel characteristics (CDB)	2	X		
1		2.4	TBV-455	5.3	Spent fuel characteristics (CDB)	2			
2	CRWMS M&O 1999. <i>PWR Source Term Generation and Evaluation</i> . BBAC00000-01717-0210-00010 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990215.0395.	Entire	N/A (Controlled source)	2.	PWR source terms	N/A	N/A		
3	CRWMS M&O 1999. <i>Enhanced Design Alternative II Report</i> . B00000000-01717-5705-00131 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990712.0194.	2. 2.	N/A (Selected LA design)	5.1.2 5.1.7	WP thermal load WP barrier materials and thicknesses	N/A	N/A		

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2. Technical Product Input Source Title and Identifier(s) with Version	Unqual.						From Uncontrolled Source	Un-confirmed	
2a	Briesmeister, J. F., Editor 1997. <i>MCNP-A General Monte Carlo N-Particle Transport Code, Version 4B</i> . LA-12625-M. Los Alamos, New Mexico: Los Alamos National Laboratory. TIC: 241044.	Entire	N/A (Accepted practice)	2.	User's Manual	N/A	N/A		
4									
5	CRWMS M&O 1998. <i>Evaluation of WP Transporter Neutron Shielding Materials</i> . BCAE000000-01717-0210-00002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990119.0320.	5.1.6	TBV-326	5.1.7	Design inputs and outputs related to repository subsurface shielding design analysis (model & dose rate)	2	X		
	5.3	TBV-326	5.4						
	6.1	TBV-326	5.5						
6	CRWMS M&O 1998. <i>Software Qualification Report for MCNP Version 4B2, A General Monte Carlo N-Particle Transport Code</i> . CSCI: 30033 V4B2LV. DI: 30033-2003 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980622.0637.	Entire	N/A (Qualified software)	4.1	Qualified engineering software	N/A	N/A		

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2. Technical Product Input Source Title and Identifier(s) with Version							Unqual.	From Uncontrolled Source	Un-confirmed
2a	CRWMS M&O 1999. <i>Computer Files for Subsurface Shielding-specific Source Term Evaluation.</i> , DI: CAL-WER-NU-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990816.0239.	Entire	N/A (File locator)	5.5	Computer input and output files	N/A	N/A		
7									
8	CRWMS M&O 1997. <i>MGDS Subsurface Radiation Shielding Analysis.</i> BCAE00000-01717-0200-00001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971204.0497. Selected data entered in the TDMS as DTN: MO98WASTAN00011.000 DTN: MO9905WHODANQ1.000	4.1.1	N/A	5.1.1	Fuel type	N/A	N/A		
		4.1.2	(VA design	5.1.3	Neutron source spectrum				
		7.1.3	info. for	3.5	CDB & SCALE comparison				
		7.7.2.1	scoping	5.5	WP surface dose rate				
		Tab. II-3	use)	5.1.6	Fuel basket liner composition				
9	U. S. Nuclear Regulatory Commission 1997. <i>Standard Review Plan for Dry Cask Storage Systems.</i> NUREG-1536, Final Report. Washington, DC: U.S. Nuclear Regulatory Commission. TIC: 232373.	2.b	N/A	3.7	Neutron source energy spectrum	N/A	N/A		
		2.a	(Accepted practice)	5.1.4	Important gamma energy groups				

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2. Technical Product Input Source Title and Identifier(s) with Version							Unqual.	From Uncontrolled Source	Un-confirmed
2a	American Nuclear Society 1977. <i>Neutron and Gamma-Ray Flux-to-Dose-Rate Factors</i> . ANSI/ANS-6.1.1-1977. March 17, 1977. La Grange, Illinois: American Nuclear Society. TIC: 239401.	Table 1	N/A	5.1.5	Dose conversion factors	N/A	N/A		
10		Table 3	(Accepted practice per 10CFR20)						
11	CRWMS M&O 1997. <i>Repository Subsurface Layout Configuration Analysis</i> . BCA000000-01717-0200-00008 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971201.0879.	7.1.9.1	N/A	5.1.7	Main drift concrete lining	N/A	N/A		
		Fig. 7-17	(VA design info. for scoping use)	5.1.7	Main drift excavated diameter				
12	CRWMS M&O 1999. <i>Axial Source Profile Effect on Waste Package Transporter Shielding</i> . BCAE000000-01717-0210-00005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990811.0284.	Table 1	N/A	5.1.6	Smeared fuel region composition	N/A	N/A		
			(VA design info. for scoping use)						

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2. Technical Product Input Source Title and Identifier(s) with Version		3. Section					Unqual.	From Uncontrolled Source	Un-confirmed	
2a	CRWMS M&O 1996. <i>Material Compositions and Number Densities for Neutronics Calculations.</i> BBA000000-01717-0200-00002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: 19960624.0023.	4.1.1	N/A (Accepted material data)	5.1.6	Material composition data	N/A	N/A			
13										
14	Haynes International 1997. <i>Hastelloy Alloy C-22</i> . UCRL-ID-122526. Kokomo, Indiana: Haynes International, Inc. TIC: 238121.	Entire	N/A (Accepted material data)	5.1.6	Alloy C-22 composition	N/A	N/A			
15	General Atomics 1993. <i>GA-4 Legal Weight Truck From-Reactor Spent Fuel Shipping Cask Final Design Report.</i> Document No. 910353/0 (Dec. 1993). San Diego, California: General Atomics. TIC: 233567.	5.3.2	N/A	5.1.6	B-poly composition	N/A	N/A			
		5.2	(Approved controlled source)	6.5	Fuel specifications					

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2. Technical Product Input Source Title and Identifier(s) with Version	3. Section					Unqual.	From Uncontrolled Source	Un-confirmed	
2a	American Nuclear Society 1985. <i>American National Standard for Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants.</i> ANSI/ANS-6.4-1985. March 1, 1985. La Grange, Illinois: American Nuclear Society. TIC: 8768.	Table	N/A	5.1.6	Type 04 concrete composition	N/A	N/A		
16		5.1	(Accepted practice)						