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LIST OF ACRONYMS AND ABBREVIATIONS

ACD	Advanced Conceptual Design
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
B-poly	Borated polyethylene
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
cc	cubic centimeter
CDA	Controlled Design Assumptions
CDB	Characteristics Data Base
CFR	Code of Federal Regulations
Ci	Curies
cm	centimeters
CSCI	Computer Software Configuration Index
DBF	Design Basis Fuel
DCSS	Design Concept Subsurface
EBDRD	Engineered Barrier Design Requirements Document
FA	Fuel Assembly
FSD or fsd	Fractional Standard Deviation
g	grams
GA	General Atomics
GROA	Geologic Repository Operations Area
GWd	Gigawatt Days
hr	hours
LEF	Lower End Fittings
m	meters
MeV	Millions of electron Volts
MGDS	Mined Geologic Disposal System
mm	millimeters
MTIHM	Metric Tons of Initial Heavy Metal
MTU	Metric Tons of Uranium (same as MTIHM)
MWd	Megawatt Days
NRC	Nuclear Regulatory Commission
n/s	neutrons/sec
ORNL	Oak Ridge National Laboratory
PC	Personal Computer
PHO or Pho	Photons
PWR	Pressurized Water Reactor
QARD	Quality Assurance Requirements and Description
RDRD	Repository Design Requirements Document

RG	Regulatory Guide
RSIC	Radiation Shielding Information Center
SFD	Spent Fuel Disassembly
SNF	Spent Nuclear Fuel
TBD	To Be Determined
TBV	To Be Verified
TEDE	Total Effective Dose Equivalent
TN	Transnuclear
UEF	Upper End Fittings
VF	Volume Fraction
wk	week(s)
WP	Waste Package
WPDD	Waste Package Development Department
wt. %	weight percent
YMP	Yucca Mountain Site Characterization Project
yr	year(s)
γ 's/s	gammas or photons per second

1. PURPOSE

The purpose of this analysis is to provide a preliminary Mined Geologic Disposal System (MGDS) subsurface shielding design and a radiological safety evaluation in support of the current emplacement concepts for the repository. The analysis will achieve the objective of providing radiation field data and shielding requirements for emplacement equipment design and personnel protection.

The analysis will also evaluate the radiation fields associated with radiation detection and control. The results will be used in a separate analysis to support identification of radiation instrumentation requirements (including types, ranges, and set points) and overall functional response characteristics of the repository to normal and abnormal conditions.

Initial focus for the shielding analysis is on commercial pressurized-water-reactor (PWR) spent nuclear fuel (SNF) assemblies as a representative waste form. Assumption 4.3.2.1 in Section 4.3.2 provides the rationale for this selection. Other waste forms including boiling-water-reactor (BWR) spent fuel and defense high-level waste (DHLW) are expected to be substantially bounded by this analysis, and will be specifically evaluated in follow-on analyses.

Specific scope of work to respond to the stated purpose and objective includes:

- Evaluation of bounding radiation source terms for PWR fuel assemblies for use in shielding design and radiation safety analysis, including non-fuel region activation source terms
- Determination of radiation hazards to repository personnel exposed to waste packages during emplacement, including a limited As-Low-As-Reasonably-Achievable (ALARA) assessment to minimize occupational radiation exposures
- Evaluation of the shadow shield and drift entrance door designs for personnel access and optimized use of drift space
- Analysis of radial and axial shielding configurations for the waste package transporter, and examination of the feasibility of reduced transporter bottom shielding for an appropriate balance between weight and personnel radiation safety requirements
- Evaluation of radiation fields associated with sensors, alarms, and control systems that may be subjected to long-term radiation conditions from intact waste packages and/or high radiation fields from accident conditions

2. QUALITY ASSURANCE

The Quality Assurance program applies to this design analysis. The work reported in this document is part of the preliminary design of the repository. This activity can affect the proper functioning and performance of the MGDS subsurface systems, structures, and components.

The items that are the subject of this design analysis have not been classified in accordance with QAP-2-3, Classification of Permanent Items. However, they have been placed on the MGDS Q-List (Ref. 5.30) by direct inclusion. The specific items on the Q-List important to safety and waste isolation that apply to this analysis include the waste packages (WP), WP transporter, emplacement drifts and access tunnels. This design analysis activity has been evaluated in accordance with QAP-2-0, Conduct of Activities. That evaluation determined that such activities are subject to Quality Assurance Requirements and Description (QARD) requirements (Ref. 5.1) for the following reasons:

- The activity affects items on the MGDS Q-List (Ref. 5.30)
- The activity provides data used to assess the potential dispersion of radioactive materials from a licensed facility

Therefore, this design analysis is prepared per QAP-3-9, Design Analysis.

All design inputs identified and used in this document are for preliminary design and shall be treated accordingly; these design inputs may require subsequent qualification (or superseding inputs) as the repository design proceeds. In addition, much of the input data used in this analysis are unqualified. Therefore, the outputs are also unqualified. This document will not directly support any construction, fabrication or procurement activity, and therefore the outputs are not required to be controlled as TBV (to be verified). However, use of any data or outputs from this analysis for input into documents supporting procurement, fabrication or construction is required to be controlled as TBV in accordance with NLP-3-15.

3. METHOD

3.1 RADIATION SOURCE TERMS

Radiation source terms are an essential and necessary element of shielding design and analysis. Ref. 5.2 provides the Advanced Conceptual Design (ACD) design basis fuel (DBF) for the design of waste packages. The same basis is used here for the MGDS subsurface shielding design.

The source terms required for shielding analysis include the neutron and gamma sources in the active fuel region, and Co-60 source due to neutron activation of non-fuel components in the end fittings of each PWR fuel assembly in the waste package. Co-60 is the bounding activation product. Other activation products such as Cr-51, Mn-54, Fe-55, Ni-59, and Ni-63 in the non-fuel components are unimportant for shielding considerations, because they have decayed to negligible levels after 10 years (Section 4.1.1) from reactor discharge, or emit weak or no gammas. The source terms are described in terms of the source strengths and energy spectra, which are obtained by the following method:

- Neutron source strength in active fuel: taken directly from Characteristics Data Base (CDB, Ref. 5.3) in terms of n/s per MTIHM (Metric Tons of Initial Heavy Metals) which is the same as MTU (Metric Tons of Uranium) for PWR and BWR SNF (Note that the unit of MTU is used in this analysis to be consistent with the convention used in the repository thermal loading analysis.)
- Neutron energy spectrum in active fuel: based on the spectral data in Ref. 5.4 for the GA-4 transportation cask
- Gamma source strength in active fuel: taken directly from CDB (Ref. 5.3) in terms of photons/s per MTU
- Gamma source spectrum in active fuel: taken directly from CDB (Ref. 5.3)
- Co-60 source activity (Ci) in non-fuel components: scaled from the CDB (Ref. 5.3) by the burnup ratio. Co-60 emits two significant gammas per disintegration having energies of 1.17 and 1.33 MeV (Ref. 5.22, p. 25), respectively, with a mean energy of 1.25 MeV

The radiation source terms are used in this analysis as inputs to the shielding codes for calculations of dose rates in the areas of interest.

3.2 SHIELDING ANALYSIS

This analysis uses two qualified computer codes, namely, MCNP4A (Refs. 5.5 and 5.15) and PATH

(Refs. 5.6 and 5.16). Both codes may be used for three-dimensional (3-D) shielding analysis. Both codes have been qualified for use in the Yucca Mountain Site Characterization Project (YMP), as per QAP-SI-0. MCNP4A is a Monte Carlo code capable of treating complex shielding problems involving scattering, streaming and secondary radiation generation. PATH is a point-kernel integration code suitable for calculations of direct gamma dose rates. Both codes are appropriate for this design analysis in view of their acceptance by the U. S. Nuclear Regulatory Commission (NRC) in previous licensing applications (References 5.4 and 5.25) for spent fuel storage and transportation casks.

MCNP4A is the principal code for shielding analyses such as scattering off MGDS drift walls where a significant scattered radiation component is present. This code can calculate contributions from primary gammas, neutrons, and secondary gammas in order to obtain the total radiation fields from all sources. Both direct and scattered radiation may be treated by MCNP4A.

For the cases where the direct gamma dose component is the principal contributor to the overall radiation field, the PATH code is available to calculate the direct gamma dose rate. PATH is based on the common point-kernel integration technique for the three-dimensional (3-D) source-shield configuration. In this design analysis, PATH is used to calculate the direct gamma dose rates from the waste packages for comparison with the MCNP4A calculations. In addition, PATH is used to determine the time-dependent gamma radiation conditions and to calculate gamma dose rates external to the waste package transporter.

Analytical models are developed and described in Section 7 to appropriately simulate the types of analyses performed. The models describe the sources, configurations, and materials required for use in the analyses with MCNP4A and PATH.

4. DESIGN INPUTS

4.1 DESIGN PARAMETERS

4.1.1 ACD Design Basis Fuel (DBF)

The ACD DBF parameters for PWR spent fuel assemblies are identified in References 5.2 and 5.7 to bound 97.85% of PWR SNF population (Ref. 5.7, p. 6) as follows:

Characteristics: 4.2% initial enrichment, 48086 MWd/MTU burnup, and 10 years decay from reactor discharge (Ref. 5.2, Vol III, Table 5-2; also Ref. 5.7, p. 6)

Fuel type: B&W 15x15 Mark B (Ref. 5.7, p. 7)

Heavy metal loadings: 0.464 MTU per fuel assembly (Ref. 5.7, p. 7)

These parameters are used in Section 7.1 to derive the source terms for the ACD DBF.

4.1.2 PWR DBF Neutron Source Data

For PWR SNF, the neutron source strength is taken directly from the CDB (Assumption 4.3.1.7). Since the neutron source spectrum is unavailable in the CDB, the neutron spectral data provided in the GA-4 spent fuel transportation cask design (Ref. 5.4) is used for this design analysis. These parameters are used in Section 7.1.1 to obtain the DBF neutron source term per waste package. The neutron source data for PWR SNF are presented below:

Neutron Source Strength (for DBF)

6.740E+8 neutrons/sec (n/s) per MTU (part of computer output from Ref. 5.3)

Neutron Source Spectrum (See Table 4.1-1 for GA-4 cask)

**TABLE 4.1-1
NEUTRON SOURCE SPECTRUM FOR GA-4 CASK**

Neutron Energy Range (MeV)	Unnormalized Spectrum ^a	Normalized Spectrum ^b
6.43-20.0	2.354+05 ^c	0.0183
3.00-6.43	2.691+06	0.2099
1.85-3.00	3.002+06	0.2341
1.40-1.85	1.680+06	0.1310
0.90-1.40	2.265+06	0.1767
0.40-0.90	2.466+06	0.1924
0.10-0.40	4.827+05	0.0376
0.00-0.10	0.0	0.0
Total	1.282+07	1.0

^a Unnormalized spectrum from Ref. 5.4, p. 5.2-6 for 35,000 MWd/MTU burnup and 10 year decay. At 10 year decay, the neutron spectrum is insensitive to the burnup.

^b Normalized to 1.0 (total) for input to the MCNP4A code. Example: The normalized spectrum for the 6.43 to 20.0 MeV group = $(2.354\text{E}+05)/(1.282\text{E}+07) = 0.0183$

^c Read as 2.354×10^5

4.1.3 PWR DBF Gamma Source Data

The gamma source strength for the active fuel regions of PWR DBF is obtained from the CDB (Assumption 4.3.1.7). The CDB also provides the gamma source energy spectrum required for this design analysis. Table 4.1-2 provides the PWR DBF gamma source data, which is part of the computer output from Ref. 5.3. These parameters are used in Section 7.1.2 to obtain the DBF gamma source term per waste package.

TABLE 4.1-2
LWR RADIOLOGICAL DATABASE
PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086
ENRICHMENT: 4.20%
DECAY TIME: 10 YEARS

The data are shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
=====	=====	=====
1.000E-02	2.728E+15	24.27%
2.500E-02	5.787E+14	5.15%
3.750E-02	7.125E+14	6.34%
5.750E-02	5.396E+14	4.80%
8.500E-02	3.229E+14	2.87%
1.250E-01	3.145E+14	2.80%
2.250E-01	2.640E+14	2.35%
3.750E-01	1.257E+14	1.12%
5.750E-01	4.798E+15	42.69%
8.500E-01	4.671E+14	4.16%
1.250E+00	3.849E+14	3.42%
1.750E+00	4.879E+12	0.04%
2.250E+00	7.693E+10	0.00%
2.750E+00	5.940E+09	0.00%
3.500E+00	7.264E+08	0.00%
5.000E+00	2.900E+07	0.00%
7.000E+00	3.343E+06	0.00%
9.500E+00	3.841E+05	0.00%
TOTAL	*1.124E+16	100.01%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/age combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases. (This paragraph is part of the computer output from Ref. 5.3.)

4.1.4 Time-dependent Neutron Source Data

The following time-dependent neutron source data in Table 4.1-3 are used to determine the neutron radiation field from the waste packages as a function of time for a maintainable preclosure service life of at least 150 years (Ref. 5.8, p. 4-11) after initiation of waste emplacement. The initial emplacement time corresponds to a decay time of 10 years as defined for the DBF. The end of the service life corresponds to a decay time of 160 years. The data are for PWR fuel assemblies with 4.2% initial enrichment and 48086 MWd/MTU burnup. These parameters are used in Section 7.7 for time-dependent radiation conditions.

TABLE 4.1-3
TIME-DEPENDENT NEUTRON SOURCE DATA

<u>Decay Time (yr)</u>	<u>Time After Emplacement (yr)</u>	<u>Neutron Source Strength (n/s-MTU)^a</u>
10	0	6.740+08 ^b
15	5	5.604+08
20	10	4.665+08
30	20	3.249+08
40	30	2.289+08
50	40	1.621+08
60	50	1.179+08
70	60	8.777+07
80	70	6.662+07
90	80	5.128+07
100	90	3.970+07
110	100	3.278+07
120	110	2.797+07
130	120	2.461+07
140	130	2.224+07
150	140	2.053+07
160	150	1.929+07

^a part of computer output from Ref. 5.3.

^b Read as 6.740x10⁸.

4.1.5 Time-dependent Gamma Source Data

Attachment I provides the time-dependent gamma source data used to determine the active fuel gamma radiation field from the waste packages as a function of time for the 150-year maintainable preclosure service life (Ref. 5.8, p. 4-11) after initial emplacement. The initial emplacement time corresponds to a decay time of 10 years as defined for the DBF. The end of the maintainable preclosure service life corresponds to a decay time of 160 years. The data are for PWR fuel assemblies with 4.2% initial enrichment and 48086 MWd/MTU burnup. The decay times include those as given in Section 4.1.4 for the time-dependent neutron source data. All the data are reproduced from the computer output of the CDB (Ref. 5.3).

These parameters are used in Section 7.7 for time-dependent radiation conditions.

4.1.6 Average PWR Spent Fuel Source Terms

The average spent fuel source data are used in this analysis for ALARA dose assessment. Consistent with Assumption 4.3.1.1, the following source data are obtained from the CDB for average PWR spent fuel [assumed to be 26 years old with 39750 MWd/MTU burnup and 3.72 wt. % initial enrichment (Assumption 4.3.1.1)]:

Neutron Source Data

2.241E+08 n/s per MTU (Ref. 5.3, computer output)

Gamma Source Data

Reproduced in Table 4.1-4 from Ref. 5.3, computer output.

These parameters are used in Section 7.6 for ALARA evaluation.

TABLE 4.1-4
LWR RADIOLOGICAL DATABASE
PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 39750

ENRICHMENT: 3.72%

DECAY TIME: 26 YEARS

The data are shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	1.503E+15	27.16%
2.500E-02	2.981E+14	5.39%
3.750E-02	3.599E+14	6.50%
5.750E-02	3.247E+14	5.87%
8.500E-02	1.633E+14	2.95%
1.250E-01	1.274E+14	2.30%
2.250E-01	1.375E+14	2.48%
3.750E-01	5.727E+13	1.03%
5.750E-01	2.469E+15	44.61%
8.500E-01	3.741E+13	0.68%
1.250E+00	5.668E+13	1.02%
1.750E+00	1.140E+12	0.02%
2.250E+00	1.763E+08	0.00%
2.750E+00	5.487E+08	0.00%
3.500E+00	2.209E+07	0.00%
5.000E+00	9.470E+06	0.00%
7.000E+00	1.091E+06	0.00%
9.500E+00	1.253E+05	0.00%
TOTAL	*5.535E+15	100.01%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases. (This paragraph is part of the computer output from Ref. 5.3)

4.1.7 Co-60 Sources in Non-fuel Components

The upper and lower end fittings of each fuel assembly contain Co-60 sources, which are due to neutron activation of non-fuel components during reactor operation. These sources also contribute to radiation fields. As discussed in Section 3.1, Co-60 is the only activation product considered in this design analysis. Table 4.1-5 provides the CDB (Assumption 4.3.1.7) Co-60 source data (from Ref. 5.3) for use in Section 7.2 to derive the Co-60 source terms for the DBF.

TABLE 4.1-5
CDB CO-60 SOURCE DATA

<u>Region</u>	<u>Co-60 Activity (Ci/assembly)^a</u>
Upper End Fittings	111.5
Gas Plenum	57.64
Lower End Fittings	151.0

^a From Ref. 5.3, computer output for 4.73% initial enrichment, 60000 MWd/MTU burnup and 10 years decay.

4.1.8 Material Data

The material data required in this design analysis include the density and elemental composition for each material. These data are based on the standard material handbooks and published YMP documents, and are consistent with those used by other design groups (e.g., the Waste Package Development Department). The materials considered include those for the waste form, waste package, WP transporter, emplacement drift, host tuff medium, etc. According to Assumption 4.3.2.4, Table 4.1-6 lists the pertinent material data used in this design analysis. These parameters are used throughout Section 7.

**TABLE 4.1-6
MATERIAL DATA FOR SHIELDING ANALYSIS**

<u>Material</u>	<u>Density (g/cc)</u>	<u>Element</u>	<u>Weight Percent</u>
ACD WP Inner Barrier Alloy 825 ^a (Ref. 5.9, p. 7)	8.14	C	0.05
		Al	0.20
		Si	0.50
		Ti	0.90
		Cr	21.50
		Mn	1.00
		Fe	28.57
		Ni	42.00
		Cu	2.25
		Mo	3.00
WP Outer Barrier Transporter γ Shield A516 Carbon Steel (Ref. 5.9, p. 5 & 6)	7.832	C	0.22
		Si	0.275
		S	0.035
		P	0.035
		Mn	0.90
		Fe	98.535
Transporter Clad SS316L (Ref. 5.9, p.14)	7.9497	Si	0.75
		Cr	17.00
		Mn	2.0
		Fe	65.545
		Ni	12.00
		Mo	2.50
		C	0.030
		P	0.045
		S	0.030
		N	0.100

<u>Material</u>	<u>Density (g/cc)</u>	<u>Element/Compound</u>	<u>Weight Percent</u>
WP Transporter	0.92	H	12.56
Neutron Shield		B-10	0.278
1.5% B-poly		B-11	1.223
(Ref. 5.4, p. 5.3-6)		C	85.94
Drift Wall	2.22	SiO ₂	78.900
Tsw2	(Dry)	Al ₂ O ₃	12.300
(Ref. 5.9, p. 27)		Fe ₂ O ₃	0.973
		CaO	0.451
		MgO	0.128
		TiO ₂	0.092
		Na ₂ O	3.920
		K ₂ O	3.180
		P ₂ O ₅	0.010
		MnO	0.046
Air	0.001225	N	80.0
(Ref. 5.9, p. 26)		O	20.0
Type 04 Concrete	2.35	H	0.55 ^b
(ANSI/ANS 6.4-1985)		O	49.83 ^b
(Ref. 5.10, p. 9)		Si	31.57 ^b
(Criteria 4.2.3)		Ca	8.26 ^b
		Na	1.70 ^b
		Mg	0.26 ^b
		Al	4.55 ^b
		S	0.13 ^b
		K	1.91 ^b
		Fe	1.23 ^b
Drift Entrance Door	7.9	C	0.0800
SS304		N	0.1000
(Ref. 5.9, p. 12)		Si	0.7500
		P	0.0450
		S	0.0300
		Cr	19.000
		Mn	2.0000
		Fe	68.745
		Ni	9.250

<u>Material</u>	<u>Density (g/cc)</u>	<u>Element/Compound</u>	<u>Weight Percent</u>
Fuel Basket	7.745	B-10	0.2311
SS316B6A		B-11	1.053
(Ref. 5.9, p. 17)		C	0.0301
		N	0.1003
		Si	0.7524
		P	0.0451
		S	0.0301
		Cr	19.061
		Mn	2.0064
		Fe	60.639
		Ni	13.5433
		Mo	2.508
Fuel Cladding	6.56	O	0.12
Zirc-4		Cr	0.10
(Ref. 5.9, p. 21)		Fe	0.20
		Sn	1.40
		Zr	98.18

^a Ref. 5.2 was used as the source of the WP design information. Current WP inner barrier material is Alloy 625. Alloy 825 and Alloy 625 are neutronically equivalent because of their similar densities and elemental compositions.

^b Obtained by dividing the partial density given in Ref. 5.10, p. 19 by 2.35 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 (Ref. 5.10, p. 9).

4.1.9 Flux-To-Dose-Rate Conversion Factors

The flux-to-dose-rate conversion factors in Table 4.1-7 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 (Ref. 5.11). These parameters are used throughout Section 7.

TABLE 4.1-7
FLUX-TO-DOSE-RATE CONVERSION FACTORS

Neutron Flux-To-Dose-Rate Conversion Factors (Ref. 5.11, p. 4)

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

^a Read as 2.5×10^{-8} .

Gamma-Ray-Flux-To-Dose-Rate Conversion Factors (Ref. 5.11, p. 5)

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

4.2 CRITERIA

The following design criteria were developed in response to the requirements found in the RDRD (Ref. 5.12) and EBD RD (Ref. 5.35) that specifically apply to this design analysis. RDRD and EBD RD requirements are cited for each criterion statement.

- 4.2.1** Occupational radiation exposures shall be maintained as low as reasonably achievable (ALARA), and within the exposure dose limits specified in 10CFR20.1201. ALARA principles shall be based on the applicable sections of NRC Regulatory Guides 8.8 (Rev. 3) and 8.10 (Rev. 1-R). [(RDRD: 3.2.1.1.D, 3.2.1.2.C, 3.2.1.3, 3.2.1.4.C, 3.2.2.1.A, 3.2.2.1.B, 3.2.2.1.C, and 3.2.3.2.2.B.2), and (EBDRD: 3.2.1.2.C, 3.2.1.3, 3.2.1.4.C, 3.2.2.1.A, 3.2.2.1.D, and 3.2.3.3.B.3)]

For this design analysis, the controlling limit is the total effective dose equivalent (TEDE) of 5 rem per year. To meet the 5 rem/yr limit, the dose rate shall be designed not to exceed 2.5 mrem/hr in normally occupied areas (based on a 2000 hr/yr occupational access). Higher dose rate limits shall be allowed in intermittently occupied areas depending on occupancy duration, and frequency of exposure, provided that annual exposures are within 5 rem.

This criterion statement is used in Sections 7.4 and 7.5 to provide shielding design for accessible areas (e.g., main drift and turnout areas), based on the design basis fuel.

- 4.2.2** For ALARA considerations, radiation shielding shall be designed with the goal of limiting the TEDE to less than 1 rem per year to workers, which is one-fifth of the 10CFR20.1201 annual occupational exposure limit. (RDRD 3.2.4.5.1.A and RDRD 3.2.4.5.1.B)

The ALARA limit of 1 rem/yr corresponds to 0.5 mrem/hr for normally occupied areas based on a 2000 hr/yr occupational access. Other areas requiring intermittent access can be designed with higher dose rate limits based on occupancy duration, and frequency of exposure, provided that the ALARA goal of 1 rem/yr is met.

This criterion statement is used in Section 7.6 for ALARA dose assessment, based on average SNF characteristics.

- 4.2.3** The concrete shadow shields provided near the entrances (east and west entrances) of each emplacement drift for radiation protection purposes shall comply with ANSI/ANS 6.4-1985 (Ref. 5.10) and ACI 349-90 (Ref. 5.32), and shall consider the material specifications of ANSI/ANS 6.4.2-1985 (Ref. 5.33) where it provides a critical confinement or structural function. For other shields, ACI 318-95 (Ref. 5.34) is appropriate and provides adequate strength for design earthquake loads. (RDRD 3.2.4.5.1.C)

This criterion statement is used in Section 4.1.8 for specification of concrete density and

material composition for radiation shielding analysis.

- 4.2.4** Additional suitable shielding shall be provided in the repository to maintain radiation doses in restricted areas (as defined in 10CFR20) of the GROA (Geologic Repository Operations Areas) within the limits specified for such areas in 10CFR20. [(RDRD: 3.2.3.2.2.A.6 and 3.7.7.A.3), and (EBDRD: 3.2.2.1.C, 3.2.3.3.A.4, 3.2.3.3.A.5, 3.2.4.5, and 3.7.1.4)].

This criterion statement applies to the waste package transporter shielding (Section 7.5), in-drift concrete shadow shields (Section 7.4), and drift entrance doors (Section 7.4).

- 4.2.5** Design of shield walls shall avoid straight-line penetrations to prevent radiation streaming. (RDRD 3.2.4.5.1.D)

The shielding design for the waste package transporter shielding (Section 7.5), in-drift concrete shadow shields (Section 7.4), and drift entrance doors (Section 7.4) meets this criterion by using no penetrations through the shields.

- 4.2.6** The minimum cooling time for standard SNF is 5 years. (EBDRD 3.2.3.4.C.1.d)

The ACD DBF specifications in Section 4.1.1 meet this criterion by using 10 years cooling.

- 4.2.7** Estimated SNF inventory of radionuclides will be (TBD). (EBDRD 3.2.3.4.C.4)

Assumption 4.3.1.7 provides the assumption for (TBD).

- 4.2.8** Engineered Barrier Segment structures, systems, and components shall be designed for a maintainable preclosure service life of at least (TBD) years. (EBDRD 3.2.5.4)

Assumption 4.3.1.8 provides the assumption for (TBD).

- 4.2.9** The waste package external dimensions shall be (TBD). (EBDRD 3.7.1.J.1)

Assumption 4.3.1.9 provides the assumption for (TBD).

4.3 ASSUMPTIONS

4.3.1 General Assumptions from the CDA

This section contains relevant excerpts from the Controlled Design Assumptions (CDA) Document (Ref. 5.8) that apply to this design analysis. Rationales for these assumptions are cited in Ref. 5.8. All controlled assumptions will require substantiation as discussed in Section 2.3 of the CDA.

- 4.3.1.1** The average commercial spent fuel characteristics upon receipt at the repository are: 26.4 years old with 39750 MWd/MTU burnup and 3.72 wt.% initial enrichment (PWR). (CDA Key 004)

For this analysis, the decay time is changed to 26 years. Rationale for this deviation: The CDB contains the source data only for the integer year values. Use of 26 years instead of 26.4 years produces a slightly higher radiation source strength for shielding analysis. This assumption is used in Section 4.1.6 to obtain the source data for average SNF.

- 4.3.1.2** Waste packages will be emplaced in-drift in a horizontal mode. (CDA Key 011)

This assumption is used in Section 7.3.1.1 for in-drift waste package modeling.

- 4.3.1.3** Waste packages will be placed center in-drift, on pedestals, using gantry emplacement. (CDA Key 066)

For shielding models, any shielding effect attributable to pedestals and gantry equipment is ignored for conservatism. Furthermore, the invert section is not modeled. Inclusion of the pedestals and the invert section would lower the amount of radiation streaming or scattering out of the emplacement drifts and decrease the radiation fields in the main drift and turnout areas. Therefore, this assumption is conservative. This assumption is used in Section 7.3.1.1 for in-drift waste package modeling.

- 4.3.1.4** No human entry is planned in emplacement drifts while WPs are present. The waste emplacement/retrieval equipment may use robotics and/or remote control features to perform operations and monitoring within the emplacement drifts to avoid personnel radiation exposures. Under off-normal conditions, human entry will be considered if protection to workers such as shielding can be provided. (CDA Key 013)

This assumption is used in Section 7.3.5 to evaluate radiation hazards to repository personnel and the need for personnel shielding for off-normal conditions.

- 4.3.1.5** WP containment barriers will provide sufficient shielding for protection of WP materials from radiation enhanced corrosion. Individual WPs will not provide any additional shielding for personnel protection. Additional shielding for personnel protection will be provided on the subsurface transporter and in subsurface facilities. (CDA Key 031 and CDA EBD RD 3.2.4.5)

This assumption is used in Sections 7.4 and 7.5 to provide shielding for personnel protection.

4.3.1.6 The following diameters are assumed for underground openings: (CDA Key 070)

Ramps	7.62 m
Access/Service Main	7.62 m
Emplacement Drift	TBD (See Assumption 4.3.2.2)

This assumption is used in Section 7.5.1.1 to account for drift wall scattering effects in the waste package transporter shielding analysis.

4.3.1.7 Inventories of radionuclides will be based on the Characteristics Data Base. (CDA EBDRD 3.2.3.4.C.4)

This assumption is used in Section 4.1 to obtain all radiation source data for use in shielding analyses.

4.3.1.8 Engineered Barrier Segment structures, systems, and components shall be designed for a maintainable preclosure service life of at least 150 years following first emplacement of waste. (CDA EBDRD 3.2.5.4)

This assumption is used in Section 7.7 for determination of time-dependent doses.

4.3.1.9 The external dimensions of the waste package for (a) commercial SNF that is repackaged and for (b) uncanistered commercial SNF shall not exceed: (CDA EBDRD 3.7.1.J.1)

Outer Diameter:	1.85 m
Outer length:	5.85 m

These dimensions along with Assumption 4.3.2.15 are used in Section 7.5.1.1 to establish the inside dimensions of the waste package transporter.

4.3.1.10 A 35 m thermal standoff distance is used to limit the surface rock temperature of the adjacent main drift. The thermal standoff is defined as the perpendicular distance from the center of the closest emplaced waste package to the nearest edge of the main drift. (CDA DCSS 033)

A 13 m radiological standoff distance is used for limiting the radiological dose in the adjacent main drift. The radiological standoff is defined as the distance from the center of the closest emplaced waste package to the door of the emplacement drift. (CDA DCSS 033)

This assumption is used in Sections 7.3.4.2, 7.4 and 7.6.2 to locate the dose points.

4.3.1.11 Doors are required at entrances to emplacement drifts. (CDA DCSS 036)

This assumption is used in Section 7.4 for the drift entrance door shielding analysis.

4.3.2 Specific Assumptions for This Analysis

4.3.2.1 The waste package contains 21 PWR fuel assemblies. Rationale: This assumption is consistent with the quantity in Table 3-10 of CDA Key 004 for the large waste package containing PWR fuel assemblies. The PWR SNF is selected for this analysis as a representative waste form. Other waste forms including BWR SNF and DHLW will be evaluated in follow-on analyses. Review of the information in Sections 6.3.6.3.1.1 (p. 6.3-167) and 6.3.6.3.2.1 (p. 6.3-171) of Ref. 5.2, Vol. III indicates that the characteristics of the BWR SNF are similar to those of the PWR SNF. The evaluation for the PWR SNF generally applies to the BWR SNF. Such applicability will be demonstrated in future analyses. The DHLW contains a gamma source with weak energy emission (essentially from Cs-137/Ba-137m), and a neutron source with strength considerably lower than that for the PWR or BWR SNF. The DHLW normally requires less shielding than the PWR and BWR SNF (Ref. 5.2, Vol III, Section 6.4.4.4, p. 6.4-56). This assumption is used throughout Section 7.

4.3.2.2 Current design assumptions for the subsurface repository are based on the emplacement drift bore diameter of 5.5 m. The emplacement drifts may be lined with pre-cast concrete linings up to 0.2 m thick. This lining will reduce the useful inner diameter to approximately 5.1 m (Ref. 5.13, p. 18). With the density and elemental composition of concrete similar to those of the tuff medium (Tsw2), the change in the concrete lining thickness would have little effect on the shielding results. These assumptions are used in Sections 7.3.1.1 and 7.4 for drift wall modeling.

4.3.2.3 The center-to-center spacing for the large PWR waste package based on 83 MTU/acre emplacement is 19.12 m (Ref. 5.2, Vol. II, p. 8-43). This spacing is rounded to 20 m to facilitate radiation dose calculations from multiple waste packages. This deviation has little effect on the results, since the last-emplaced waste package is the principal contributor to radiation fields in the main drift and turnout areas. This assumption is only used to calculate the scattered contributions from multiple waste packages. Rounding the WP spacing up to 20 m is less conservative than rounding down to 19 m. However, this less conservative nature is more than compensated by the conservative assumption that the intervening waste packages are transparent for scattered radiation (See Assumption 4.3.2.8). The WP spacing is subject to change, depending on the the repository thermal management strategy.

Once finalized, future shielding analysis will use the actual spacing. This assumption is used in Section 7.3.4 for multiple WP dose calculations.

4.3.2.4 The actual materials for the waste packages, transporter, drift entrance doors, and drift liners have not been finalized (TBD). For the purpose of this analysis, representative materials are used for shielding calculations. Ref. 5.2 is the source of design information on the WP materials. Type 04 concrete given in Table 4.1-6 is assumed for shadow shields and drift liners for the convenience of shielding analysis. This analysis will remain valid, provided that the actual materials selected have similar densities and compositions to those used in this analysis. This assumption is used in Section 4.1.8.

4.3.2.5 For shielding analysis, the uncanistered commercial PWR SNF waste package is assumed in order to obtain conservative results. This waste package has dimensions less than the maximum dimensions assumed in CDA EBD RD 3.7.1.J.1. Consistent with Ref. 5.14, p. 20, the following dimensions for the uncanistered PWR SNF waste package are assumed and used in the shielding analysis:

Active fuel length = 3601.8 mm
Fuel region diameter = 1220 mm
Fuel basket peripheral liner thickness = 16.8 mm
WP inner barrier thickness = 20 mm (radial), 25 mm (axial)
WP outer barrier thickness = 100 mm (radial), 110 mm (axial)
WP outer diameter = 1629 mm

This assumption is used throughout Section 7 for waste package modeling.

4.3.2.6 The material data provided in Table 4.3-1 were assumed for the radiation shielding analysis, based on the data used in Reference 5.14. Rationale: These data are derived from the materials contained in the respective region, and are representative for radiation shielding analysis. This assumption is used in Section 7.3.1.2 to obtain the smeared densities for the fuel region, fuel basket liner, lower end fittings and upper end fittings.

4.3.2.7 The dose rate limit for shielding analysis of the waste package transporter is assumed to be 50 mrem/hr on the general accessible surfaces of the transporter, which includes the sidewall and end surfaces. The dose rate limit for the bottom surface of the transporter where access is normally not required is assumed to be 100 mrem/hr. Rationale: Personnel access requirements in the vicinity of the transporter when loaded with a waste package have not been established. Typically, on-site transfer casks at nuclear power plants or other nuclear facilities were designed to a surface dose rate of 50 to 200 mrem/hr. For preliminary design of the waste package

TABLE 4.3-1
ASSUMED MATERIAL DATA

PWR Spent Fuel Assembly Constituents

(Data used for conversion to weight %, Ref. 5.14, p. 49)

<u>Element</u>	<u>atoms/barn-cm</u>
O	1.4127E-2
Cr	7.1994E-6
Fe	1.3406E-5
Zr	4.0290E-3
Sn	4.4155E-5
U-235	4.7717E-5
U-236	3.8027E-5
U-238	6.3510E-3
Pu-239	3.2181E-5
Pu-240	1.1558E-5
Pu-241	7.2740E-6

Volume Fractions (VF) for Homogenized Fuel Region (Ref. 5.14, p. 58)

<u>Component</u>	<u>VF</u>
Fuel	0.8547
Air	0.1453

Volume Fractions for Fuel Basket Liner (Ref. 5.14, p. 58)

<u>Component</u>	<u>VF</u>
SS316B6A	0.3448
A825	0.6552

Volume Fractions for PWR Fuel Lower End Fitting Region (Ref. 5.14, p. 58)

<u>Component</u>	<u>VF</u>
SS304	0.2908
Zirc-4	0.0369
Air	0.6723

Volume Fractions for PWR Fuel Upper End Fitting Region (Ref. 5.14, p. 58)

<u>Component</u>	<u>VF</u>
SS304	0.235
Inconel 718	0.072
Air	0.693

transporter, it is prudent to use the lower end of the 50 to 200 mrem/hr range, namely, 50 mrem/hr for the general accessible surfaces. No access is required to the bottom surface of the transporter for normal conditions. A higher dose rate limit is acceptable on the bottom surface. The limit of 100 mrem/hr is assumed to limit the scattering contributions off the floor, and avoid having to classify the area as a "high radiation area." This assumption is used in Section 7.5 for the waste package transporter shielding design.

- 4.3.2.8** For radiation analysis of multiple waste packages in the same emplacement drift, any intervening waste package is assumed to be an opaque object for direct radiation (i.e., direct contribution without scattering off the drift wall), but transparent for scattered or albedo radiation off the drift wall. Rationale: This assumption is conservative to estimate the radiation dose contributions from multiple waste packages. The closest waste package is the principal contributor to the radiation fields in the main drift and turnout areas. This waste package will block virtually all direct radiation from other waste packages in the same drift, leaving the contributions arising from scattering off the concrete liner or drift wall only. This assumption ignores the presence of the intervening waste packages for scattered radiation off the concrete liner or drift wall, which conservatively over-estimates the scattering contributions. This assumption is used in Section 7.3.4 to estimate the radiation fields for multiple waste packages.
- 4.3.2.9** The concrete shadow shield provided between the last-emplaced waste package and drift entrance door is assumed to be 2.15 m in diameter and 0.4572 m (18") thick placed at 4 m longitudinally from the WP center. The shadow shield is assumed to be circular in shape for modeling purposes. The shape could be circular, rectangular or other as appropriate, with an equivalent dimension. This analysis focuses on the thickness of the shadow shield. The shape will be determined in future analyses. Rationale: The diameter or equivalent dimension of the shadow shield needs to be larger than the maximum WP diameter to be effective for shielding direct radiation. On the other hand, the size of this shadow shield should not pose any interference with emplacement gantry operations. The assumption of the 2.15 m diameter is intended to meet these two conditions, and to facilitate modeling. The concrete thickness of 0.4572 m is to minimize direct radiation contributions to the main drift and turnout areas so that the direct contribution is small relative to the scattered contribution. The distance of 4 m provides a spacing of about 1 m between the waste package and shadow shield for clearance. This assumption is used in Section 7.4 for the shadow shield analysis.
- 4.3.2.10** The drift entrance door is assumed to be 25.4 mm (1") thick of SS304 (Type 304 stainless steel). Rationale: This door serves to control personnel access and ventilation flow. The door thickness depends on the structural requirements which have not been determined at this design stage (TBD). For shielding analysis, a

nominal thickness of 25.4 mm is used. This thickness is significant to personnel access in the main drift, when the door is closed. Any change in the door thickness would require a re-evaluation of its shielding effect. With respect to the door material, both stainless steel and carbon steel are acceptable for shielding, because of their practically equal shielding effectiveness. For ease of decontamination, stainless steel is preferred. However, carbon steel is less expensive. This assumption is used in Section 7.4 for the shadow shield and drift entrance door analysis.

- 4.3.2.11** The shielding materials for the waste package transporter are assumed to be carbon steel for gamma shielding and borated polyethylene (1.5% natural boron) for neutron shielding. Rationale: These materials are commonly used in the shielding design of fuel transfer casks, shipping casks, and storage casks. This assumption is used in Section 7.5 for the waste package transporter shielding design.
- 4.3.2.12** The waste package railcar loading dock at the drift entrance door is omitted in the emplacement drift model. Rationale: The loading dock is a concrete structure which would reduce contributions of radiation scattering off the bottom portion of the drift wall to the radiation fields in the main drift and turnout areas. Ignoring this dock in the model would result in a higher dose rate, which is conservative. This assumption is used in Sections 7.3.1.1 and 7.4.
- 4.3.2.13** For the waste package transporter shielding analysis, the transporter is assumed to be located inside one of the main drifts. Rationale: The transporter carries a waste package from the surface waste handling building through one of the main drifts to an emplacement drift. The transporter, when underground, can be located in the main drift or turnout areas. Their cross-sectional areas are practically equivalent from a shielding standpoint. In addition, personnel are more likely to occupy the main drifts than the turnout areas. This assumption is used in Section 7.5.1.1.
- 4.3.2.14** Uniform fuel burnup is assumed in this analysis. Rationale: This assumption is consistent with license applications submitted to the NRC for spent fuel transportation and storage casks which typically use uniform fuel burnup to generate source terms. This assumption is used throughout Section 7.
- 4.3.2.15** The spacing between the maximum dimensions of the waste package (Assumption 4.3.1.9) and the inside dimension of the WP transporter is assumed to be 0.3 m both radially and axially. Rationale: This spacing provides sufficient clearance between the waste package and transporter, based on the value used in Ref. 5.14, p. 20. Little shielding effect results from a change in this spacing. This assumption is used in Sections 7.5.1.1 and 7.5.2 for the transporter shielding models.
- 4.3.2.16** The WP design information in ACD (Ref. 5.2) is assumed, including the DBF,

physical dimensions and materials. Rationale: The ACD WP design information provides a documented basis for neutronics analysis until new or revised data are available. This assumption is used throughout Section 7.

4.4 CODES AND STANDARDS

4.4.1 American National Standards Institute (ANSI)

ANSI/ANS 6.4.2-1985 *Specification for Radiation Shielding Materials*, 5/31/85

ANSI/ANS 6.4-1985 *Guidelines on the Nuclear Analysis and Design of concrete Radiation Shielding for Nuclear Power Plants*, 3/1/85

4.4.2 Code of Federal Regulations (CFR)

10CFR20 *Standards for Protection Against Radiation*, Revised as of January 1, 1996

4.4.3 NRC Regulatory Guides (RGs)

RG 8.8, Rev. 3 *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will be As Low As is Reasonably Achievable (ALARA)*, Revision 3, June 1978 (Second Proposed Revision 4, Draft OP 618-4, May 1982)

RG 8.10, Rev. 1-R *Operating Philosophy for Maintaining Occupational Exposure ALARA*, Revision 1-R, September 1975

4.4.4 American Concrete Institute (ACI)

ACI 318-95 *Building Requirements for Structural Concrete (ACI 318-95) and Commentary (ACI 318R-95)*, March 1, 1995

ACI 349-90 *Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349-90)*, March 1, 1990.

5. REFERENCES

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- 5.2 *MGDS Advanced Conceptual Design Report*, DI: B000000000-01717-5705-00027, Rev. 00.
- 5.3 *Characteristics of Potential Repository Wastes*, Oak Ridge National Laboratory, DOE/RW-0184-R1, July 1992.
- 5.4 *GA-4 Legal Weight Truck From-Reactor Spent Fuel Shipping Cask Final Design Report*, General Atomics, Document No. 910353/0, December 1993.
- 5.5 RSIC Computer Code Collection, *MCNP 4A - Monte Carlo N-Particle Transport Code System*, CCC-200, Oak Ridge National Laboratory, January 1995.
- 5.6 S. Su et al, *PATH Gamma Shielding Code User's Manual*, GA-A16772 (Rev. 1), May 1987.
- 5.7 *Source Terms for Design Basis Event Analysis*, DI: BBA0000000-01717-0200-00019, Rev. 00.
- 5.8 *Controlled Design Assumptions Document*, B000000000-01717-4600-00032, Rev. 04.
- 5.9 *Material Compositions and Number Densities for Neutronics Calculations*, BBA0000000-01717-0200-00002, Rev. 00.
- 5.10 *Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants*, ANSI/ANS-6.4-1985, March 1, 1985.
- 5.11 *Neutron and Gamma-Ray Flux-to-Dose-Rate Factors*, ANSI/ANS-6.1.1-1977, March 17, 1977.
- 5.12 *Repository Design Requirements Document*, YMP/CM-0023, Rev. 0, ICN 1.
- 5.13 *Performance Confirmation Data Acquisition System*, BBAI000000-01717-0200-00002, Rev. 00.
- 5.14 *UCF Waste Package Shielding Analysis/2-D DORT*, BBAA000000-01717-0200-00006, Rev. 00.
- 5.15 *MCNP4A QAP-SI-1 Baseline Submission Package for QA (SCM) Approval*, CSCI: 30006 V4A, DI:30006-2001, -2002, -2003, and -2004, Rev. 01.

- 5.16 *Software Qualification Report for PATH/V88A - A Gamma Shielding Analysis Code*, DI: 30007-2003 Rev 00.
- 5.17 C. V. Parks, et al, *Assessment of Shielding Analysis Methods, Codes, and Data for Spent Fuel Transportation/Storage Applications*, Oak Ridge National Laboratory, ORNL/CSD/TM-246, July 1988.
- 5.18 *Tn-24 Dry Storage Cask Topical Report*, Transnuclear, Inc., E-7107, December 1989.
- 5.19 RSIC Computer Code Collection, *ORIGEN2 - Isotope Generation and Depletion Code, Matrix Exponential Method*, Oak Ridge National Laboratory, CCC-371, September 1989.
- 5.20 RSIC Computer Code Collection, *SCALE 4: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation*, Oak Ridge National Laboratory, CCC-545, February 1990.
- 5.21 *Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code*, Oak Ridge National Laboratory, ORNL/TM-6051, September 1978.
- 5.22 *Nuclides and Isotopes*, Fourteenth Edition, GE Nuclear Energy, 1989.
- 5.23 *Software Qualification Report for The SCALE Modular Code System (CSCI: 30004 V1.0)*, DI: 30004-2000 Rev 00.
- 5.24 *PWR Assembly Source Terms for Waste Package Design*, DI: BBA000000-01717-0200-00023 Rev 00.
- 5.25 *Standard Review Plan for Dry Cask Storage Systems*, U.S. Nuclear Regulatory Commission, NUREG-1536, Final Report, January 1997.
- 5.26 *Characteristics of Spent Fuel High-Level Waste, and Other Wastes Which May Require Long-Term Isolation*, DOE/RW-0184, December 1987.
- 5.27 *Software Qualification Report for the SCALE Modular Code System Version 4.3*, CSCI:30011 V4.3, DI: 30011-2002, Rev 00.
- 5.28 J. W. Roddy, et al., *Physical and Decay Characteristics of Commercial Spent Fuel*, Oak Ridge national laboratory, ORNL/TN-9591/V1&R1, January 1986.
- 5.29 *Computer Input and Output Files for MGDS Subsurface Radiation Shielding Analysis*, DI: BCAE00000-01717-0200-00001 REV 00, Batch Number MOY-970620-01.

- 5.30 *Q-List*, U.S. Department of Energy, YMP/90-55Q, Rev 04.
- 5.31 R. E. Walpole and R. H. Myers, *Probability and Statistics for Engineers and Scientists*, Fifth Edition, MacMillan Publishing Company, 1993.
- 5.32 *Code Requirements for Nuclear Safety Related Concrete Structures (ACI 349-90)*, ACI 349-90, March 1, 1990.
- 5.33 *Specification for Radiation Shielding Materials*, ANSI/ANS 6.4.2-1985, May 31, 1985.
- 5.34 *Building Requirements for Structural Concrete (ACI 318-95) and Commentary (ACI 318R-95)*, ACI 318-95, March 1, 1995.
- 5.35 *Engineered Barrier Design Requirements Document*, YMP/CM-0024, Rev. 0, ICN 1.

Reference 5.29 is submitted with this Analysis Records Package.

6. USE OF COMPUTER SOFTWARE

6.1 COMPUTER HARDWARE

The computer hardware used in this analysis is a GATEWAY 2000 P5-100 desktop personal computer (CRWMS-M&O 110519). This computer has been qualified to perform execution of the software used in this analysis.

6.2 COMPUTER SOFTWARE

This design analysis uses the two computer codes described in Section 3, namely, PATH (Version V88A, CSCI: 30007 V88A) and MCNP4A (Version 4A, CSCI: 30006 V4A). Both MCNP4A and PATH have been fully qualified on the PC platform (Ref. 5.15 and 5.16, respectively).

6.3 SOFTWARE APPLICATION

The computer software used in this analysis is appropriate per the capabilities and applications of the codes, and is used within the range of validation in the software qualification reports.

The MCNP4A and PATH input and output files from this design analysis are stored on electronic media as Ref. 5.29.

The MCNP4A code is selected for the following reasons:

- MCNP4A is a general-purpose industry standard code for shielding applications as will be encountered in the MGDS.
- MCNP4A accounts for detailed physics of radiation transport including primary and secondary radiation such as secondary gammas produced from neutron interactions with materials.
- MCNP4A is used widely for spent fuel transportation and storage cask licensing applications to the NRC (Ref. 5.4), and has been recommended by Oak Ridge National Laboratory (ORNL) to the DOE as a standard code for Monte Carlo shielding analysis (Ref. 5.17).

The basis for selecting the PATH code is as follows:

- PATH employs the common industry point-kernel integration techniques for efficient computation.

- PATH had been benchmarked, validated, and used previously in licensing applications.
- PATH contains an extensive data library, including attenuation coefficients, buildup factors, and dose conversion factors.
- PATH was accepted by the NRC as its in-house calculation tool.

6.4 SPREADSHEETS

In addition to the qualified software described in Section 6.2, computational support software is used for spreadsheets. The spreadsheets used in this analysis were performed, using Lotus 1-2-3, Release 5. The user-defined formulas and algorithms for the spreadsheets are provided in Attachments II, III and IV, including example hand calculations.

7. DESIGN ANALYSIS

The design analysis covers the following items, consistent with the scope of work and objectives specified in Section 1:

- Radiation source term evaluation
- Radiation hazard analysis
- Concrete shadow shield analysis
- Waste package transporter shielding analysis
- ALARA evaluation
- Time-dependent radiation conditions

7.1 ACTIVE FUEL REGION SOURCE TERMS FOR DBF

This section provides the neutron and gamma source terms per waste package including source strengths and spectra in the active fuel region for the design basis fuel. The waste package contains 21 PWR fuel assemblies with the characteristics of 4.2% initial enrichment, 48086 MWd/MTU burnup and 10 years decay (Section 4.1.1). These source terms are used in the radiation hazard analysis, concrete shadow shield analysis, and waste package transporter shielding analysis.

7.1.1 Neutron Source Term

Based on the data provided in Section 4.1.2, Table 7.1-1 lists the DBF neutron source data by neutron energy group. The source strength in n/s per MTU is converted to n/s per fuel assembly, using an initial uranium loading of 0.464 MTU per fuel assembly (Section 4.1.1). The source strength in n/s per WP is obtained by using 21 PWR fuel assemblies per WP.

7.1.2 Gamma Source Term

The DBF gamma source strength and energy spectrum provided in Section 4.1.3 in photons/s per MTU are converted to photons/s per WP in the same manner as for the neutron source data. Table 7.1-2 lists the gamma source term by photon energy.

In general, the important gamma energy groups for shielding considerations are in the range of 0.8 to 2.5 MeV (Ref. 5.25, p. 5-3). The groups with energies below 0.8 MeV make negligible contributions to the shielded dose rates, because of their high photon absorption cross sections. The relatively small magnitude of the source strength for energies above 2.5 MeV contributes insignificantly to the dose rates. Therefore, for the MCNP4A calculations, the analysis includes the energy range of 0.45 to 3.0 MeV only, which covers the range specified by the NRC in Ref. 5.25.

TABLE 7.1-1
PWR FUEL NEUTRON SOURCE TERM
 (4.2% Initial Enrichment, 48086 MWd/MTU, and 10 years Decay)

Neutron Energy Normalized		Neutron Source Strength		
Range (MeV)	Spectrum ^a	n/s-MTU ^b	n/s-assembly ^c	n/s-WP ^d
6.43 - 20.0	0.0183	1.233+7 ^e	5.723+6	1.202+8
3.00 - 6.43	0.2099	1.415+8	6.564+7	1.379+9
1.85 - 3.00	0.2341	1.578+8	7.321+7	1.537+9
1.40 - 1.85	0.1310	8.829+7	4.097+7	8.603+8
0.90 - 1.40	0.1767	1.191+8	5.526+7	1.160+9
0.40 - 0.90	0.1924	1.297+8	6.017+7	1.264+9
0.10 - 0.40	0.0376	2.534+7	1.176+7	2.469+8
TOTAL	1.0	6.740+8 ^a	3.127+8	6.567+9

^a From Section 4.1.2.

^b Equal to 6.740×10^8 x normalized spectrum for each group.

^c n/s-assembly = n/s-MTU x 0.464 MTU/assembly.

^d n/s-WP = n/s-assembly x 21 assemblies/WP.

^e Read as 1.233×10^7 .

TABLE 7.1-2
PWR FUEL GAMMA SOURCE TERM
 (4.2% initial enrichment, 48086 MWd/MTU, and 10 years decay)

Group	Photon Energy Range (MeV) ^a	Mean Energy (MeV)	Gamma Source Strength	
			(γ 's/sec per MTU) ^b	(γ 's/sec per WP) ^c
1	0.0 - 0.02	0.01	2.728+15 ^d	2.658+16
2	0.02 - 0.03	0.025	5.787+14	5.639+15
3	0.03 - 0.045	0.0375	7.125+14	6.943+15
4	0.045 - 0.07	0.0575	5.396+14	5.258+15
5	0.07 - 0.10	0.085	3.229+14	3.146+15
6	0.10 - 0.15	0.125	3.145+14	3.064+15
7	0.15 - 0.30	0.225	2.640+14	2.572+15
8	0.30 - 0.45	0.375	1.257+14	1.225+15
9	0.45 - 0.70	0.575	4.798+15	4.675+16
10	0.70 - 1.0	0.85	4.671+14	4.551+15
11	1.0 - 1.5	1.25	3.849+14	3.750+15
12	1.5 - 2.0	1.75	4.879+12	4.754+13
13	2.0 - 2.5	2.25	7.693+10	7.496+11
14	2.5 - 3.0	2.75	5.940+09	5.788+10
15	3.0 - 4.0	3.50	7.264+08	7.078+09
16	4.0 - 6.0	5.0	2.900+07	2.826+08
17	6.0 - 8.0	7.0	3.343+06	3.257+07
18	8.0 - 11.0	9.5	3.841+05	3.743+06
Total			1.124+16	1.095+17

^a Energy range per p. 3-46 of Ref. 5.28.^b Directly from Section 4.1.3.^c γ 's/sec per WP = γ 's/sec per MTU x 0.464 MTU/assembly x 21 assemblies/WP.^d Read as 2.728x10¹⁵.

7.1.3 Comparison with Independent SCALE Calculation

The CDB neutron and gamma source terms for the PWR SNF are compared with the data generated by the Waste Package Development Department (WPDD) and documented in Ref. 5.24. The WPDD source data were generated using the SAS2H module in the qualified SCALE4.2 code package (Ref. 5.23).

The comparison focuses on the design basis fuel with the characteristics of 4.2% initial enrichment, 48086 MWd/MTU burnup, 10 years decay, and 0.464 MTU per fuel assembly. The comparison is shown below:

Neutron Source Strength

CDB:	3.127E+08 n/s per fuel assembly (Section 7.1.1, Table 7.1-1)
WPDD:	2.736E+08 n/s per fuel assembly (Ref. 5.24, Attachment II, p. 28)

Neutron Energy Spectrum

The WPDD neutron energy group spectrum obtained with SCALE4.2 is identical to the spectrum used in this analysis as given in Table 7.1-1, verifying the spectral data provided in Ref. 5.4 for the GA-4 cask are accurate.

Gamma Source Strength

CDB:	5.215E+15 γ 's/s per fuel assembly (Section 7.1.1, Table 7.1-2)*
WPDD:	4.093E+15 γ 's/s per fuel assembly (Ref. 5.24, Attachment II, p. 32)

*Obtained by $1.124\text{E}+16 \text{ } \gamma\text{'s/s-MTU} \times 0.464 \text{ MTU/FA} = 5.215\text{E}+15 \text{ } \gamma\text{'s/s-FA}$

Gamma Energy Spectrum

The energy group structures are different between the CDB (Table 7.1-2) and WPDD (Ref. 24, Attachment II, p 31 & 32), though both have the same number of energy groups (18). The WPDD spectrum appears to be somewhat harder (more energetic) than the CDB spectrum. The harder spectrum tends to increase the gamma radiation fields outside the waste packages. However, this increase may be compensated by the lower gamma source strength from the WPDD.

The comparison indicates that the CDB source data are acceptable and conservative for preliminary design use.

7.2 NON-FUEL REGION CO-60 SOURCE EVALUATION

This activity evaluates the non-fuel region Co-60 source terms in PWR fuel assemblies for use in radiation shielding analysis. Included in this category are non-fuel items such as lower core plate, bottom nozzles, lower tie plate, lower end plug, gas plenum springs, upper end plug, upper tie plate, and assembly handle. The data available in the CDB are used to derive the necessary Co-60 source terms for this analysis. Comparisons of the CDB-derived source terms are made with the NRC licensing submittals for spent fuel storage and transportation casks, and an independent SCALE calculation (Ref. 5.24).

7.2.1 Co-60 Source Terms Based on CDB

Section 4.1.7 provides the Co-60 sources in PWR non-fuel components for the parameters of 4.73% initial enrichment, 60000 MWd/MTU burnup and 10 years decay. For the design basis fuel defined in Section 4.1.1 as 4.2% initial enrichment, 48086 MWd/MTU and 10 years decay, the CDB Co-60 source activity for 60000 MWd/MTU burnup is scaled down by the burnup ratio of (48086/60000), as the Co-60 activity is fairly proportional to fuel burnup. The DBF of 4.2% initial enrichment, which is different from the CDB 4.73% initial enrichment, poses only a minor effect on the Co-60 source activity, as confirmed by the data given in Ref. 5.4 for different enrichments.

Table 7.2-1 provides the Co-60 source data for the PWR fuel assembly end fittings for the DBF. The curie content per fuel assembly is simply a direct ratioing from Section 4.1.7 by (48086/60000). The photon source strength in photons/sec is equal to the curie content multiplied by 2 photons per disintegration for Co-60 and 3.7×10^{10} disintegrations/sec per curie. The energy source strength equals the photon source strength times a mean energy of 1.25 MeV per photon (Section 3.1).

TABLE 7.2-1
CO-60 SOURCE TERMS IN END FITTINGS
(4.2% initial enrichment, 48086 MWd/MTU, and 10 years decay)

<u>Region</u>	<u>Per Assembly</u>	<u>Per Waste Package^a</u>
Curie Content		
Upper End Fittings	89.36	1877
Gas Plenum	46.19	970
Lower End Fittings	121.0	2541
Photons/sec		
Upper End Fittings	6.61+12	1.39+14
Gas Plenum	3.42+12	7.18+13
Lower End Fittings	8.95+12	1.88+14
MeV/sec @ 1.25 MeV		
Upper End Fittings	8.26+12	1.74+14
Gas Plenum	4.28+12	8.98+13
Lower End Fittings	1.12+13	2.35+14

^a 21 PWR fuel assemblies per waste package.

7.2.2 Comparison with Previous NRC License Submittals

The CDB Co-60 source data for PWR fuel assemblies are compared with the previous NRC license submittals for the spent fuel transportation and storage casks. Selected for comparison here are the GA-4 transportation cask (Ref. 5.4), and the TN-24 storage cask (Ref. 5.18). The GA-4 Cask Safety Analysis Report for Packaging was recently submitted to the NRC for licensing review. The TN-24 cask has received NRC approval and license. Both casks are for PWR fuel assemblies. GA-4 can carry a maximum of 4 PWR fuel assemblies, and TN-24 has a capacity of 24 assemblies.

Table 7.2-2 compares the data sets from the CDB, GA-4 and TN-24. Description of each data set follows.

7.2.2.1 CDB Description

The CDB contains the physical and radiological data for the various fuel assembly types. Three typical fuel assembly types are selected here for comparison, including W15x15, W17x17, and B&W15x15. These assembly types are commonly used as design basis fuel for transportation and storage cask licensing applications.

For the PWR spent fuel disassembly (SFD) hardware, the CDB provides the radiological data only for standard burnup [3.11% initial U-235, and 30 Gigawatt Days (GWd)/MTU], and extended burnup (4.73% initial U-235, and 60 GWd/MTU) (Ref. 5.3, p. 2.7-2). The comparison here uses the extended burnup data.

Several factors affect the source activity in the fuel assembly end fittings. The major factors which may cause large uncertainties in the source data include cobalt contents in the materials, and spatial and spectral effects of the neutron fluxes for the non-fueled regions. The cobalt contents for the PWR hardware materials used to generate the CDB are given below, based on upper limits and ASTM specifications:

Inconel-718	4700 ppm (Ref. 5.3, p. 2.7-2)
Inconel X-750	6500 ppm (Ref. 5.3, p. 2.7-2)
Microbrazed 50	381 ppm (Ref. 5.3, p. 2.7-2)
SS302	800 ppm (Ref. 5.3, p. 2.7-2)
SS304	800 ppm (Ref. 5.3, p. 2.7-2)
Zr-4	10 ppm (Ref. 5.3, p. 2.7-2)

For the CDB, the ORIGEN2 code (Ref. 5.19) was first used to estimate the radioactivity of materials in the fueled region of the reactor. Scaling factors were then applied to the ORIGEN2 calculations to compensate for the neutron flux change outside the fueled region. The CDB PWR activation scaling factors relative to the fueled region are presented below:

Table 7.2-2 Comparison of Co-60 Source Terms for PWR Fuel Assembly End Fittings

Description	<u>GA-4</u>	<u>TN-24</u>	<u>GA-4</u>	<u>TN-24</u>	<u>CDB</u>	<u>CDB</u>	<u>CDB</u>
Fuel Assembly Type	W15x15	W17x17	W15x15	W17x17	W15x15	W17x17	BW15x15
Initial U-235, %	3.0	3.3	3.9	3.3	4.73	4.73	4.73
Burnup, GWd/MTU	35	35	60	60	60	60	60
Fuel Age, years	10	10	10	10	10	10	10
Flux Scaling Factor							
Upper End	N/A ^a	0.028	N/A	0.028	0.1	0.1	0.1
Gas Plenum	N/A	0.074	N/A	0.074	0.2	0.2	0.2
Lower End	N/A	0.028	N/A	0.028	0.2	0.2	0.2
Co-60, Ci/assembly							
Upper End	10.2	1.95	17.3	3.34	82.4	59.4	111.5
Gas Plenum	6.82	2.52	11.6	4.32	43.6	45.5	57.6
Lower End	3.73	1.63	6.33	2.80	51.8	56.1	151.0

^a Not available.

Upper end fittings	0.1 (Ref. 5.3, p. 2.7-3)
Gas plenum	0.2 (Ref. 5.3, p. 2.7-3)
Fueled region	1.0 (Ref. 5.3, p. 2.7-3)
Lower end fittings	0.2 (Ref. 5.3, p. 2.7-3)

7.2.2.2 GA-4 Description

Ref. 5.4 provides the Co-60 gamma source data for the W15x15 non-fueled regions in terms of MeV/s per fuel assembly. For comparison, the data were converted to Ci/assembly by dividing the MeV/s value by $(3.7 \times 10^{10} \text{ dis/s-Ci} \times 2.5 \text{ MeV/dis})$ as follows:

Upper end fittings	9.450E+11 MeV/s-FA (Ref. 5.4, p. 5.2-6)	10.2 Ci/FA
Gas plenum	6.310E+11 MeV/s-FA (Ref. 5.4, p. 5.2-6)	6.82 Ci/FA
Lower end fittings	3.450E+11 MeV/s-FA (Ref. 5.4, p. 5.2-6)	3.73 Ci/FA

The approach for the source data generation is similar to the CDB. However, the calculation used the SAS2H module in the SCALE computer code package (Ref. 5.20), which contains ORIGEN-S.

The SAS2H calculation was performed for incore activation. The non-fueled region source terms were obtained by using activation ratios relative to the fueled region (given in Ref. 5.21, developed by Croff). The cobalt contents in the SFD hardware materials are unavailable in Ref. 5.4.

7.2.2.3 TN-24 Description

The source data for the TN-24 storage cask are based on a single set of parameters: W17x17 fuel assembly type, 3.3% initial U-235 enrichment, 35 GWd/MTU burnup, and 5 years decay from discharge (Ref. 5.18, p. 7.2-1). The cobalt impurities used in the SFD hardware materials are:

Inconel	4700 ppm (Ref. 5.18, p. 7.2-1)
Nicrobraz	380 ppm (Ref. 5.18, p. 7.2-1)
Stainless steel	800 ppm (Ref. 5.18, p. 7.2-1)

These material data are consistent with those used in the CDB.

ORIGEN2 was used to perform incore activation, and generate neutron fluxes for the fueled region. The in-core fluxes were then adjusted using the methodology in Ref. 5.21 to irradiate the non-fueled regions. The correction factors included adjustment to the cobalt mass by a factor of 0.67 (Ref. 5.18, p. 7.2-2) for self shielding, and the following flux scaling factors for the non-fueled regions:

Upper end fittings	0.042 (Ref. 5.18, p. 7.2-2)
Gas plenum	0.11 (Ref. 5.18, p. 7.2-2)
Lower end fittings	0.042 (Ref. 5.18, p. 7.2-2)

The resulting overall correction factors are:

Upper end fittings	0.042*0.67	= 0.028
Gas plenum	0.11*0.67	= 0.074
Lower end fittings	0.042*0.67	= 0.028

Based on the above input data, Ref. 5.18 gives the Co-60 source strengths in units of photons/s per assembly for 35 GWd/MTU burnup and 5 years decay. The source data are converted here to Ci/assembly and decayed to 10 years from discharge for comparison with other data sets. This conversion is illustrated below:

$$\begin{aligned}\text{Upper end} &= [5.12 \text{ E}+11 \times 0.544 \text{ } \gamma\text{'s/s-FA (Ref. 5.18, p. 7.2-5)}](2)^{-5/5.27}/(7.4\text{E}+10 \text{ } \gamma\text{'s/s-Ci}) \\ &= 1.95 \text{ Ci/FA}\end{aligned}$$

$$\begin{aligned}\text{Gas plenum} &= [3.60 \text{ E}+11 \text{ } \gamma\text{'s/s-FA (Ref. 5.18, p. 7.2-5)}](2)^{-5/5.27}/(7.4\text{E}+10 \text{ } \gamma\text{'s/s-Ci}) \\ &= 2.52 \text{ Ci/FA}\end{aligned}$$

$$\begin{aligned}\text{Lower end} &= [5.12 \text{ E}+11 \times 0.456 \text{ } \gamma\text{'s/s-FA (Ref. 5.18, p. 7.2-5)}](2)^{-5/5.27}/(7.4\text{E}+10 \text{ } \gamma\text{'s/s-Ci}) \\ &= 1.63 \text{ Ci/FA}\end{aligned}$$

As demonstrated in Ref. 5.4, the Co-60 source strength is approximately proportional to fuel burnup. Scaling by burnup was used to obtain the Co-60 source data for 60 GWd/MTU burnup.

7.2.2.4 Discussion

For the extended burnup of 60 GWd/MTU, the CDB gives the highest Co-60 source strengths for the fuel assembly end fittings. The differences in the source strengths among the three data sets are significant. Two major factors contribute to the differences, including the cobalt mass per assembly, and flux scaling factors for the non-fueled regions. The cobalt mass estimates may account for a factor of 2 to 3 difference in the Co-60 activity. An additional factor of 3 to 7 may be attributed to the flux scaling factors. The combination of these two factors results in a difference of about one order of magnitude or more.

It is apparent that the evaluation points to a lower Co-60 source term for the fuel assembly end fittings. The use of the CDB for the MGDS shielding design is highly conservative. Re-evaluation of the end fitting source data is deemed necessary, as the MGDS design progresses.

7.2.3 Independent SCALE Analysis

The M&O Waste Package Development Department (WPDD) recently generated new source terms for the PWR DBF, using the SAS2H module in the SCALE4.2 code package (Ref. 5.23). The source terms are documented in Ref. 5.24, and converted to Ci/FA as follows:

Upper end = $[(6.6787\text{E}+12) + (1.8861\text{E}+12)] \gamma\text{'s/s-FA}$ (Ref. 5.24, Attachment V, p. 15 for Co-60 energy groups)/ $(7.4\text{E}+10) \gamma\text{'s/s-Ci} = 116 \text{ Ci/FA}$

Gas plenum = $[(7.5153\text{E}+12) + (2.1223\text{E}+12)] \gamma\text{'s/s-FA}$ (Ref. 5.24, Attachment IV, p. 16 for Co-60 energy groups)/ $(7.4\text{E}+10) \gamma\text{'s/s-Ci} = 130 \text{ Ci/FA}$

Lower end = $[(9.7011\text{E}+12) + (2.7396\text{E}+12)] \gamma\text{'s/s-FA}$ (Ref. 5.24, Attachment III, p. 15 for Co-60 energy groups)/ $(7.4\text{E}+10) \gamma\text{'s/s-Ci} = 168 \text{ Ci/FA}$

Table 7.2-3 compares the non-fuel region Co-60 source terms between the WPDD data and CDB. Again, the CDB values are scaled from the data for 60000 MWd/MTU burnup by the burnup ratio of (48086/60000). This scaling represents a good approximation, based on the analysis performed for the GA-4 transportation cask (Ref. 5.4).

The WPDD SCALE4.2 analysis increased the flux scaling factors by 50% to account for the uncertainty cited in the CDB. This increase makes the non-fuel region source terms even more conservative, as compared with the factors cited in Table 7.2-2 for the GA-4 and TN-24 casks.

The WPDD used a conservative value for Inconel-718, but omitted any cobalt in stainless steel. Such omission is compensated by the conservative value used for Inconel-718, resulting in good agreement between the WPDD and CDB in the cobalt mass for both upper and lower end fittings.

When the WPDD flux factors and cobalt contents are adjusted to be on the same basis as for the CDB, the resulting Co-60 source terms agree within 10%. Therefore, use of the scaled CDB Co-60 data is conservative, as compared to the adjusted WPDD data.

7.2.4 Summary of Evaluation

The end fitting source data in the CDB regarding the Co-60 content are conservative to use for preliminary analysis. B&W15x15, Mark B fuel assemblies represent the worst-case fuel assembly with respect to the end fittings, because of its largest cobalt content in SFD hardware. The large uncertainties in the source data should be recognized and resolved during the design process.

The Co-60 design values may be down-scaled if necessary to remove the conservative factors used in the CDB. However, as shown in Table 7.4-7 (Section 7.4.3), with the concrete shadow shield in place, the Co-60 contribution is about one-third of the total dose rate in the main drift and turnout areas. This down-scaling may not result in a significant change in the shielding thickness. The maximum savings in the shielding using steel is less than 12.7 mm (0.5") in the absence of the Co-60 source. Accordingly, use of the conservative Co-60 source data from the CDB poses only a minor impact on the shielding design.

TABLE 7.2-3**COMPARISON OF WPDD and CDB CO-60 SOURCE TERMS**

(4.2% Initial Enrichment, 48086 MWd/MTU, 10 years decay, B&W 15x15, Mark B)

<u>Description</u>	<u>Original WPDD^a</u>	<u>Adjusted WPDD^b</u>	<u>Scaled CDB</u>
Co-60 Activity, Ci/assembly			
Upper end fittings	116	81	89
Gas plenum	130	41	46
Lower end fittings	168	115	121
Flux Scaling Factor			
Upper end fittings	0.15	0.10	0.10
Gas plenum	0.30	0.20	0.20
Lower end fittings	0.30	0.20	0.20
Cobalt Impurity in Material, ppm			
Inconel-718	10000	4700	4700
SS-304	0	800	800
SS-304L	0	800	800
SS-CF3M	0	800	800
Cobalt Mass, g/assembly			
Upper end fittings	18.0	18.9	18.9
Gas plenum	10.4	4.90	4.90
Lower end fittings	13.0	13.3	13.3

^a From Ref. 5.24.^b Adjusted to have the same flux factors and cobalt impurities as the CDB.

7.3 RADIATION HAZARD ANALYSIS

The radiation hazard analysis was performed for the emplacement drifts full of waste packages, each containing 21 PWR fuel assemblies. This analysis calculated the radiation fields in the main drift and turnout areas to quantify the extent of radiation hazard to workers, and to determine if additional personnel shielding is required.

For this analysis, the in-drift radiation fields were first calculated for a single waste package in an emplacement drift to obtain direct and scattered dose components as a function of axial distance from the waste package. The calculation was then extended to consider a very large number of waste packages (i.e., multiple packages) that are located in the same emplacement drift. The analysis for the multiple packages was accomplished using a super-positioning technique, which entailed super-imposing the direct radiation effects from a single waste package at any field point upon the results of scattered radiation. Any intervening waste package was treated as an opaque object for direct radiation, but transparent for scattered or albedo radiation in order to calculate the scattered-only component. The scattered dose component was determined separately for each individual waste package in the emplacement drift for summation use.

7.3.1 Single Waste Package Model

7.3.1.1 Geometric Model

Figure 7.3-1 depicts the geometric model for the MCNP4A calculation. A waste package is placed and centered in the emplacement drift, based on Assumptions 4.3.1.2 and 4.3.1.3. The waste package is surrounded by the drift wall, which is lined with 0.2 m concrete (Assumption 4.3.2.2). The railcar loading dock is ignored (Assumption 4.3.2.12).

The waste package portion of the model was taken directly from Figure 7.3.1-1 (p. 20) of Ref. 5.14 with a minor modification to make the inner barrier thickness of the waste package on both ends identical to 25 mm. Figure 7.3.1-1 of Ref. 5.14 shows 55 mm inner barrier thickness for the upper end versus 25 mm for the lower end. Modification of the upper end thickness to 25 mm has no effect on this analysis, since the model places the lower end of the WP facing the drift entrance door to yield a higher radiation field due to the greater Co-60 source from the lower end fittings of the fuel assemblies.

In the model, the waste package contains B&W 15x15 Mark B fuel assemblies. The physical data for this assembly type are contained in References 5.3 and 5.26.

FIGURE 7.3-1
 MODEL OF A SINGLE WASTE
 PACKAGE IN EMPLACEMENT DRIFT

7.3.1.2 Homogenized Material Densities

As shown in Figure 7.3-1, the fuel assembly region is divided into five axial regions for the limiting B&W 15x15 Mark B fuel assembly type, including the upper end fittings, upper fission gas plenum, active fuel, lower fission gas plenum, and lower end fittings. Consistent with Ref. 5.14, homogenized material densities are used for these regions. Air is used for the upper and lower gas plenum regions, which ignore Zircoloy tubing for conservatism.

The liner used in the waste package model to represent the periphery of the fuel basket contains SS316B6A and A825 materials, which are also homogenized for shielding analysis.

Attachment II provides the homogenized material data for these regions. These data are derived using the values in Assumption 4.3.2.6.

7.3.2 Single Waste Package Radiation Analysis

For this analysis, the MCNP4A code was run for a single package, using the model in the Figure 7.3-1. The code produced the dose results as a function of axial distance from the WP center over a range of 3 to 115 m. The 3 m distance is near the axial end of the waste package which has a maximum length of 5.85 m (Assumption 4.3.1.9). The MCNP4A calculations included the following cases:

1. Neutron dose rates with drift wall
2. Active fuel gamma dose rates with drift wall
3. Lower end fitting Co-60 gamma dose rates with drift wall
4. Neutron dose rates without drift wall
5. Active fuel gamma dose rates without drift wall
6. Lower end fitting Co-60 gamma dose rates without drift wall

The cases with the drift wall refer to the condition that the waste package is emplaced in the drift. The cases without the drift wall represent the condition that the waste package is placed in space.

The lower end fittings of the fuel assemblies contain more Co-60 activity than the upper end fittings (121 Ci/assembly for the lower vs. 89 Ci/assembly for the upper end fittings). Therefore, it is conservative to place the waste package with the lower end fittings facing the drift entrance. In so doing, the contribution from the Co-60 gammas in the end fittings from the opposite end of the waste package becomes negligible, owing to shielding by the active fuel region and lower end fittings. The cases with and without the drift wall provide the results to quantify the effect of scattering from the drift wall, which is simply the difference between the two cases.

Point detectors were used to tally the dose result at each point of interest for a single waste package. All the dose points were located on the centerline of the emplacement drift at distances of 3, 4, 8,

13, 20, 35, 55, 75, 95 and 115 m from the center of the waste package. In view of the dose points at large distances from the waste package, point detectors are appropriate for this type of calculation.

The sum of the results for Cases 1, 2 and 3 gives the total dose rate at each axial point from the last-emplaced waste package, including both direct and scattered contributions. The direct contribution is referred to as the contribution directly from the waste package without drift wall scattering effects, and the scattered contribution is due solely to scattering off the drift wall. Cases 4, 5 and 6 provide the results of the direct contributions. The scattered contribution can be obtained by taking the difference between the cases with and without the drift wall.

All the calculations used the ANSI/ANS 6.1.1-1977 dose conversion factors (Ref. 5.11). This set of conversion factors are consistent with the current 10CFR20 regulations.

7.3.3 Single WP Axial Dose Profile Results

7.3.3.1 Dose Rate Results With Drift Wall

This section provides the axial dose rate result from a single waste package at each dose point for the cases with the drift wall (i.e., Cases 1, 2 and 3 in Section 7.3.2). The results are associated with the following MCNP4A input and output files.

Case 1:	Fuel neutrons	Input file DRIFTN1	Output file DRIFTN1.OUT
Case 2:	Fuel gammas	Input file DRIFTG1	Output file DRIFTG1.OUT
Case 3:	LEF Co-60 γ 's	Input file DRIFTCO1	Output file DRIFTCO1.OUT

Listing of the complete input files is given in Attachment V. Ref. 5.29 provides all the input and output files on electronic media.

Table 7.3-1 is a summary of the single WP dose rate results with the drift wall, along with the associated statistical fractional standard deviation (fsd). The contributing dose components and their fractional standard deviations are directly from the computer outputs as indicated in Table 7.3-1. The total result at each point is simply the sum of the three contributing components. The fractional standard deviation for the total result at each point was estimated by the square root of $(\sum f_i \sigma_i)^2$, where f_i is the fractional dose contribution from each source component, and σ_i is the associated fractional standard deviation.

According to Ref. 5.5, the results are reliable for point detector tallies if the associated fractional standard deviations are less than 0.05 (Ref. 5.5, p. 2-93). The total dose rate results given in Table 7.3-1 meet this acceptance criterion.

As observed from Table 7.3-1, the LEF Co-60 source is the principal contributor to the dose rate at

every axial dose point, representing over 70% of the total dose rate. The neutron contribution is relatively small.

7.3.3.2 Dose Rate Results Without Drift Wall

The dose rate results without the drift wall were obtained by modeling the drift wall and concrete lining as an air region. The MCNP4A cases executed for this model include the following:

Case 4:	Fuel neutrons	Input file DRIFTN2	Output file DRIFTN2.OUT
Case 5:	Fuel gammas	Input file DRIFTG2	Output file DRIFTG2.OUT
Case 6:	LEF Co-60 γ 's	Input file DRIFTCO2	Output file DRIFTCO2.OUT

The results from these cases represent the direct dose contribution from the waste package without drift wall scattering.

Table 7.3-2 summarizes the results from a single waste package for the cases without the drift wall, presented in the same format as Table 7.3-1. The total results are all acceptable, since the fractional standard deviations for the total calculated dose rates are less than 0.05.

Similar to the cases with the drift wall, major contribution arises from the LEF Co-60 source. The neutron dose component is less than 1% of the total.

TABLE 7.3-1
AXIAL DOSE RATE RESULTS WITH DRIFT WALL
 (Single Waste Package)

Axial Point	Distance from WP Center, M	Dose Rate with Drift Wall, mrem/hr				Associated Fractional Standard Deviation (fsd)			
		Co-60	Fuel Gamma	Fuel Neutron	Total	Co-60	Fuel Gamma	Fuel Neutron	Total
		driftco1.out	driftg1.out	driftn1.out	calculated	driftco1.out	driftg1.out	driftn1.out	calculated
1	3	2.808E+04	1.039E+04	5.475E+02	3.902E+04	0.0115	0.0656	0.0155	0.0193
2	4	9.716E+03	3.890E+03	2.753E+02	1.388E+04	0.0088	0.0235	0.0116	0.0090
3	8	1.325E+03	6.124E+02	6.989E+01	2.007E+03	0.0073	0.0274	0.0136	0.0097
4	13	4.411E+02	1.919E+02	2.263E+01	6.556E+02	0.0152	0.0308	0.0217	0.0137
5	20	1.644E+02	6.760E+01	6.614E+00	2.386E+02	0.0117	0.0318	0.0277	0.0121
6	35	4.153E+01	1.712E+01	1.101E+00	5.975E+01	0.0142	0.0642	0.0809	0.0209
7	55	1.262E+01	4.991E+00	1.879E-01	1.780E+01	0.0196	0.0384	0.0447	0.0176
8	75	5.437E+00	2.061E+00	7.589E-02	7.574E+00	0.0261	0.0634	0.0986	0.0255
9	95	2.815E+00	9.725E-01	2.596E-02	3.813E+00	0.0331	0.0316	0.0280	0.0257
10	115	1.579E+00	5.406E-01	1.345E-02	2.133E+00	0.0377	0.0290	0.0653	0.0289

TABLE 7.3-2
AXIAL DOSE RATE RESULTS WITHOUT DRIFT WALL
 (Single Waste Package)

Axial Point	Distance from WP Center, M	Dose Rate without Drift Wall, mrem/hr				Associated Fractional Standard Deviation (fsd)			
		Co-60	Fuel Gamma	Fuel Neutron	Total	Co-60	Fuel Gamma	Fuel Neutron	Total
		driftco2.out	driftg2.out	driftn2.out	calculated	driftco2.out	driftg2.out	driftn2.out	calculated
1	3	2.814E+04	9.037E+03	3.613E+02	3.754E+04	0.0159	0.0399	0.0407	0.0153
2	4	9.563E+03	3.294E+03	1.128E+02	1.297E+04	0.0117	0.0363	0.0382	0.0126
3	8	1.149E+03	4.242E+02	1.384E+01	1.587E+03	0.0082	0.0351	0.0355	0.0111
4	13	3.380E+02	1.268E+02	3.959E+00	4.688E+02	0.0085	0.0422	0.0350	0.0130
5	20	1.219E+02	4.602E+01	1.671E+00	1.696E+02	0.0129	0.0475	0.1385	0.0159
6	35	3.235E+01	1.222E+01	3.288E-01	4.490E+01	0.0170	0.0513	0.0345	0.0186
7	55	1.068E+01	4.058E+00	9.897E-02	1.484E+01	0.0247	0.0562	0.0388	0.0235
8	75	4.830E+00	1.803E+00	4.126E-02	6.674E+00	0.0367	0.0602	0.0545	0.0311
9	95	2.546E+00	9.337E-01	1.874E-02	3.498E+00	0.0453	0.0571	0.0364	0.0363
10	115	1.473E+00	5.260E-01	9.893E-03	2.009E+00	0.0522	0.0516	0.0371	0.0406

Note: At Axial Point 1, the Co-60 dose rate with the drift wall is numerically less than that without the drift wall, owing to statistical fluctuation. In practice, the two values should be equal, indicating negligible contribution due to scattering from the drift wall at this axial point. The scattered dose rate from Co-60 at this point is shown in Table 7.3-3 as zero rather than an unrealistic negative value.

7.3.3.3 Scattered Dose Rates

The scattered dose rate from the waste package at each axial dose point is the difference in the results between the cases with and without the drift wall. The scattered dose rate from each contributing source is calculated as follows:

Scattered neutron dose rate = DRIFTN1.OUT - DRIFTN2.OUT

Scattered fuel gamma dose rate = DRIFTG1.OUT - DRIFTG2.OUT

Scattered LEF Co-60 gamma dose rate = DRIFTCO1.OUT - DRIFTCO2.OUT

Table 7.3-3 provides the results of the scattered dose rates. Figure 7.3-2 plots the axial in-drift radiation fields from a single waste package including both direct and scattered components. The plot shows that the direct component is more significant than the scattered component in the unshielded case. Use of the shadow shield to block the direct component would be effective in reducing the overall radiation fields.

TABLE 7.3-3
SCATTERED DOSE RATES FROM A SINGLE WASTE PACKAGE

Axial Point	Distance from WP Center, M	Scattered Dose Rate , mrem/hr			
		Co-60	Fuel Gamma	Fuel Neutron	Total
1	3	0.00e+00	1.35e+03	1.86e+02	1.54e+03
2	4	1.53e+02	5.96e+02	1.63e+02	9.12e+02
3	8	1.75e+02	1.88e+02	5.61e+01	4.19e+02
4	13	1.03e+02	6.51e+01	1.87e+01	1.87e+02
5	20	4.25e+01	2.16e+01	4.94e+00	6.90e+01
6	35	9.18e+00	4.90e+00	7.72e-01	1.49e+01
7	55	1.94e+00	9.30e-01	8.90e-02	2.96e+00
8	75	6.07e-01	2.60e-01	3.46e-02	9.02e-01
9	95	2.69e-01	4.00e-02	7.22e-03	3.16e-01
10	115	1.06e-01	1.50e-02	3.56e-03	1.25e-01

Note: 1.54e+03 is the same as 1.54E+03 or 1.54x10³.

Fig. 7.3-2 Axial In-drift Radiation Fields from a Single Waste Package

7.3.4 Multiple Waste Package Radiation Analysis

7.3.4.1 Calculational Approach

This calculation determines the dose contributions from multiple waste packages in the same emplacement drift, which include the last-emplaced WP and other contributing WPs. The contribution from the last-emplaced WP is available in Table 7.3-1. The contributions from the remaining packages in the drift arise only from scattering off the drift wall. Their direct contributions are negligible, because of effective shielding by the intervening waste package(s) in the front, as indicated in Figure 7.3-2.

The scattering contributions from the remaining waste packages were determined using the difference in the results between the cases with and without the drift wall from the single WP calculation. For conservatism, the presence of the front intervening waste packages was ignored (Assumption 4.3.2.8). In other words, no attenuation of the scattering component by the front waste packages was assumed. This assumption resulted in an over-estimate of the scattering contribution from each waste package behind the last-emplaced waste package. The over-estimate is greater as more intervening waste packages are present. However, this over-estimate has only a small effect on the total dose results, since the last-emplaced WP is the largest contributor to the dose rates at points of interest.

For convenience, the last-emplaced waste package is referred to as WP1, and the waste package behind WP1 is WP2, followed sequentially by WP3, WP4, WP5, etc., as depicted in Figure 7.3-3. The scattering contribution from WP2 at a given dose point was read from the plot (Fig. 7.3-2) by adding 20 m (WP spacing in Assumption 4.3.2.3) to the axial distance. For instance, the scattered dose rate from WP2 at Point 6 (35 m distance from the last-emplaced WP center) was obtained from Table 7.3-3 or Figure 7.3-2 at a distance of 55 m. Another 20 m was added to determine the contribution from WP3. The same process was continued until the contribution from the waste package under consideration was less than 1% of that from WP1. The results from the contributing waste packages were then summed to produce the total result at each point. Recalling that this approach assumed no presence of any waste packages between the WP being considered and the dose points, the results of the scattered dose rates are undoubtedly conservative.

FIGURE 7.3-3
MULTIPLE WASTE
PACKAGE MODEL

7.3.4.2 Radiation Field at Main Drift

The multiple waste package radiation analysis was performed for a dose point located at the nearest edge of the main drift, which is 35 m (Assumption 4.3.1.10) from the center of the closest or last-emplaced waste package. The purpose of this calculation is to identify radiation hazard to workers in the main drift. The analysis includes the last five waste packages, designated as WP1, WP2, WP3, WP4 and WP5. WP1 is the closest waste package to the main drift. The contributions from these waste packages to the dose point of interest are calculated as follows:

Contribution from WP1 (direct + scattered) =	59.8 mrem/hr (Table 7.3-1 @ 35 m)
Contribution from WP2 (scattered only) =	2.96 mrem/hr (Table 7.3-3 @ 55 m)
Contribution from WP3 (scattered only) =	0.90 mrem/hr (Table 7.3-3 @ 75 m)
Contribution from WP4 (scattered only) =	0.32 mrem/hr (Table 7.3-3 @ 95 m)
Contribution from WP5 (scattered only) =	0.13 mrem/hr (Table 7.3-3 @ 115 m)

Total from all WP's =	64.1 mrem/hr

As can be observed from the above calculation, the closest waste package (i.e., WP1) is the principal contributor, accounting for more than 90% of the total dose rate. The contributions from the remaining waste packages are from scattering off the drift wall only, as the direct radiation is blocked by the intervening waste packages. The calculated scattered components are conservative, because of the assumption that the intervening waste packages are transparent for scattered radiation. The effect of this assumption on the total dose rate result is fairly small, since the collective contribution from the remaining waste packages is less than 10%. In addition, the analysis confirms that there is no need to consider the waste packages beyond WP5. The contributions from the waste packages beyond WP5 should be insignificant, based on the result for WP5 (only 0.2% of the total).

7.3.5 Radiation Hazard Evaluation

Without any personnel shielding placed between the emplaced waste packages and the main drift, it is apparent that the calculated dose rate of 64.1 mrem/hr in the main drift is excessive for normal occupational access (40 hr/wk). The turnout areas (between 13 m and 35 m from the center of the closest WP) would be in excess of 100 mrem/hr, based on the results given in Table 7.3-1. According to 10CFR20, these areas would be classified as "High Radiation Areas", requiring additional access control such as locks and alarms. Near the waste package, the dose rate is 39 rem/hr (Point 1 in Table 7.3-1), which is excessive for personnel entry to the emplacement drift. Consistent with Assumption 4.3.1.4, personnel shielding or other radiation protection measures would have to be provided if personnel entry is required during off-normal conditions.

The high dose rates present a radiation hazard to emplacement operation personnel. Shielding must be provided to reduce personnel radiation exposures to meet the criteria stated in Section 4.2. The following section describes the use of the concrete shadow shields for normal conditions. Further evaluation may be necessary to determine shielding requirements for off-normal conditions.

7.4 CONCRETE SHADOW SHIELD ANALYSIS

The purpose of the shadow shield is to provide radiation protection to permit personnel access in the main drift and turnout areas, per Assumption 4.3.1.5. For this analysis, the main drift is considered as “Normally Occupied Areas”, and the turnout areas as “Intermittently Occupied Areas”. The criteria statement in Section 4.2.1 applies to this analysis.

Using Assumption 4.3.1.10, radiation fields are calculated at the locations of the drift entrance door (13 m from the nearest WP center for radiological standoff) and nearest edge of the main drift (35 m from the nearest WP center for thermal standoff) for the following two cases:

- Drift entrance door is open for WP emplacement
- Drift entrance door is closed upon completion of WP emplacement

According to Assumption 4.3.2.9, the concrete shadow shield is assumed to be 2.15 m in diameter and 0.4572 m thick placed and centered at 4 m from the WP center. The drift entrance door (Assumption 4.3.1.11) is assumed to be 25.4 mm thick stainless steel (Assumption 4.3.2.10). These dimensions are used for modeling purposes and preliminary shielding evaluation only. Future analysis will use actual dimensions, configurations and material, when the design of the shadow shield and drift entrance door is available.

As shown in Figure 7.4-1, the model for this analysis is built by adding the shadow shield and drift entrance door to the unshielded model described in Section 7.3.1. The model considers the closest or last-emplaced waste package only, which is the principal contributor to the dose rates in the main drift and turnout areas. Additional contributions from other waste packages are small (<10%), based on the results for the unshielded cases in Section 7.3.4. The model includes the 0.2 m concrete lining (Assumption 4.3.2.2), but excludes the railcar loading dock (Assumption 4.3.2.12).

The analysis was performed with the MCNP4A code, consisting of three separate runs for each case to obtain the contributions from the fuel neutron, fuel gamma and LEF Co-60 sources, respectively. Surface tallies by radial segment were used at each tally location to evaluate the radial dose variation from the drift centerline to the drift wall.

7.4.1 Case with Drift Entrance Door Closed

The MCNP4A runs performed for this case include:

Fuel neutron source:	Input file DRIFTNA, output file DRIFTNA.OUT
Fuel gamma source:	Input file DRIFTGA, output file DRIFTGA.OUT
LEF Co-60 source:	Input file DRIFTCOA, output file DRIFTCOA.OUT

FIGURE 7.4-1
GEOMETRIC MODEL FOR
CONCRETE SHADOW SHIELD

Attachment VI lists the complete input files. All the input and output files are stored on electronic media as Ref. 5.29.

The results are presented in Tables 7.4-1 through 7.4-3 for the fuel neutron, fuel gamma, and LEF Co-60 sources, respectively. The tally for each radial segment is directly from the MCNP4A output. The overall area-averaged quantity is determined from $D_{ave} = \sum A_i D_i$, where A_i is the fractional area for Segment i , and D_i is the corresponding dose rate for Segment i . The fractional standard deviation associated with the average dose rate is estimated by the square root of $[\sum (A_i D_i f_i / D_{ave})^2]$, where f_i is the fractional standard deviation for Segment i , and D_{ave} is the average dose rate. This approximation is based on means and variances of linear combinations of discrete random variables given in Ref. 5.31, p. 100.

The calculated dose rates are generally statistically satisfactory, based on the fsd criterion of less than 0.1 for surface tallies cited in Ref. 5.5 (p. 2-93). The fuel gamma dose rate at 35 m has a statistical uncertainty of larger than 0.1. However, this large uncertainty would not cause the overall uncertainty in the total dose rate result to exceed 0.1 (See the total dose rate result in Table 7.4-7).

The radial variation in the radiation dose rate from the drift centerline to the concrete liner is less than a factor of 2 between the maximum and minimum values. The radial distribution is more uniform at the drift entrance door than at the main drift. The larger variation at the main drift may be caused by the larger statistical uncertainties at this location. Improvement in the statistical uncertainties for individual tally segments is recommended in future refined analysis.

7.4.2 Case with Drift Entrance Door Open

For this case, the entrance door region is modeled as an air region instead of steel to simulate the condition that the door is open for emplacement operations or inspection. The MCNP4A runs for this case include the following:

Fuel neutron source: Input file DRIFTNB, output file DRIFTNB.OUT
Fuel gamma source: Input file DRIFTGB, output file DRIFTGB.OUT
LEF Co-60 source: Input file DRIFTCOB, output file DRIFTCOB.OUT

The input files are listed in Attachment VI. Ref. 5.29 stores all the input and output files on electronic media.

Tables 7.4-4 through 7.4-6 provide the results for this case in the same manner as the case with the door closed.

The radial dose variation for this case is similar to the case with the door closed. The observations noted in Section 7.4.1 apply to this case.

TABLE 7.4-1
NEUTRON DOSE RATES WITH DOOR CLOSED

PWR ACTIVE FUEL NEUTRON CONTRIBUTION				
18" CONCRETE SHADOW SHIELD AND 1" STEEL DOOR				
DOSE RATE @ 13 M FROM NEAREST WP CENTER (DRIFT ENTRANCE DOOR)				
Surface Segment	Outer Radius, cm	Area Fraction	driftna.out,surface 1611	
			mrem/hr	FSD
1	107.5	0.1777	10.596	0.0221
2	130.0	0.0822	10.617	0.0326
3	150.0	0.0861	9.945	0.0281
4	170.0	0.0984	9.608	0.0276
5	190.0	0.1107	9.806	0.0284
6	210.0	0.1230	10.018	0.0254
7	230.0	0.1353	10.054	0.0255
8	245.0	0.1096	10.127	0.0280
9	255.0	0.0769	9.872	0.0351
Average			10.104	0.0092
DOSE RATE @ 35 M FROM NEAREST WP CENTER (MAIN DRIFT)				
Surface Segment	Outer Radius, cm	Area Fraction	driftna.out, surface 17	
			mrem/hr	FSD
1	107.5	0.1777	0.237	0.0602
2	130.0	0.0822	0.334	0.1568
3	150.0	0.0861	0.244	0.0960
4	170.0	0.0984	0.249	0.1211
5	190.0	0.1107	0.255	0.0846
6	210.0	0.1230	0.219	0.0954
7	230.0	0.1353	0.271	0.1035
8	245.0	0.1096	0.302	0.1144
9	255.0	0.0769	0.251	0.1032
Average			0.259	0.0351

TABLE 7.4-2
FUEL GAMMA DOSE RATES WITH DOOR CLOSED

PWR ACTIVE FUEL GAMMA CONTRIBUTION				
18" CONCRETE SHADOW SHIELD AND 1" STEEL DOOR				
DOSE RATE @ 13 M FROM NEAREST WP CENTER (DRIFT ENTRANCE DOOR)				
Surface Segment	Outer Radius, cm	Area Fraction	driftga.out,surface 1611	
			mrem/hr	FSD
1	107.5	0.1777	7.482	0.0713
2	130.0	0.0822	6.499	0.0783
3	150.0	0.0861	7.907	0.1716
4	170.0	0.0984	6.844	0.0935
5	190.0	0.1107	6.936	0.0907
6	210.0	0.1230	6.652	0.0942
7	230.0	0.1353	6.857	0.0970
8	245.0	0.1096	6.169	0.0953
9	255.0	0.0769	6.370	0.1096
Average			6.898	0.0338
DOSE RATE @ 35 M FROM NEAREST WP CENTER (MAIN DRIFT)				
Surface Segment	Outer Radius, cm	Area Fraction	driftga.out, surface 17	
			mrem/hr	FSD
1	107.5	0.1777	0.383	0.4387
2	130.0	0.0822	0.308	0.2411
3	150.0	0.0861	0.251	0.1999
4	170.0	0.0984	0.242	0.2195
5	190.0	0.1107	0.201	0.1172
6	210.0	0.1230	0.224	0.1178
7	230.0	0.1353	0.275	0.1618
8	245.0	0.1096	0.366	0.3279
9	255.0	0.0769	0.251	0.2081
Average			0.285	0.1223

TABLE 7.4-3
LEF CO-60 GAMMA DOSE RATES WITH DOOR CLOSED

PWR END FITTING CO-60 CONTRIBUTION				
18" CONCRETE SHADOW SHIELD AND 1" STEEL DOOR				
DOSE RATE @ 13 M FROM NEAREST WP CENTER (DRIFT ENTRANCE DOOR)				
Surface Segment	Outer Radius, cm	Area Fraction	driftcoa.out,surface 1611	
			mrem/hr	FSD
1	107.5	0.1777	7.808	0.0134
2	130.0	0.0822	7.659	0.0192
3	150.0	0.0861	7.378	0.0190
4	170.0	0.0984	7.380	0.0193
5	190.0	0.1107	7.797	0.0184
6	210.0	0.1230	6.573	0.0173
7	230.0	0.1353	6.256	0.0170
8	245.0	0.1096	6.233	0.0244
9	255.0	0.0769	5.808	0.0342
Average			7.026	0.0064
DOSE RATE @ 35 M FROM NEAREST WP CENTER (MAIN DRIFT)				
Surface Segment	Outer Radius, cm	Area Fraction	driftcoa.out, surface 17	
			mrem/hr	FSD
1	107.5	0.1777	0.243	0.0396
2	130.0	0.0822	0.224	0.0539
3	150.0	0.0861	0.255	0.0567
4	170.0	0.0984	0.241	0.0511
5	190.0	0.1107	0.248	0.0497
6	210.0	0.1230	0.266	0.0528
7	230.0	0.1353	0.239	0.0449
8	245.0	0.1096	0.248	0.0506
9	255.0	0.0769	0.269	0.0652
Average			0.248	0.0169

TABLE 7.4-4
NEUTRON DOSE RATES WITH DOOR OPEN

PWR ACTIVE FUEL NEUTRON CONTRIBUTION				
18" CONCRETE SHADOW SHIELD ONLY				
DOSE RATE @ 13 M FROM NEAREST WP CENTER (DRIFT ENTRANCE DOOR)				
Surface Segment	Outer Radius, cm	Area Fraction	driftnb.out,surface 1611	
			mrem/hr	FSD
1	107.5	0.1777	15.015	0.0270
2	130.0	0.0822	14.830	0.0325
3	150.0	0.0861	13.507	0.0317
4	170.0	0.0984	14.520	0.0354
5	190.0	0.1107	13.965	0.0332
6	210.0	0.1230	13.955	0.0279
7	230.0	0.1353	14.295	0.0289
8	245.0	0.1096	14.062	0.0326
9	255.0	0.0769	14.152	0.0387
Average			14.305	0.0106
DOSE RATE @ 35 M FROM NEAREST WP CENTER (MAIN DRIFT)				
Surface Segment	Outer Radius, cm	Area Fraction	driftnb.out, surface 17	
			mrem/hr	FSD
1	107.5	0.1777	0.424	0.0809
2	130.0	0.0822	0.585	0.1534
3	150.0	0.0861	0.426	0.1256
4	170.0	0.0984	0.579	0.1184
5	190.0	0.1107	0.521	0.1568
6	210.0	0.1230	0.513	0.1050
7	230.0	0.1353	0.521	0.1425
8	245.0	0.1096	0.650	0.1480
9	255.0	0.0769	0.578	0.2238
Average			0.524	0.0465

**TABLE 7.4-5
FUEL GAMMA DOSE RATES WITH DOOR OPEN**

PWR ACTIVE FUEL GAMMA CONTRIBUTION				
18" CONCRETE SHADOW SHIELD ONLY				
DOSE RATE @ 13 M FROM NEAREST WP CENTER (DRIFT ENTRANCE DOOR)				
Surface Segment	Outer Radius, cm	Area Fraction	driftgb.out,surface 1611	
			mrem/hr	FSD
1	107.5	0.1777	44.978	0.0517
2	130.0	0.0822	36.627	0.0581
3	150.0	0.0861	46.145	0.0958
4	170.0	0.0984	38.122	0.0670
5	190.0	0.1107	40.579	0.0582
6	210.0	0.1230	38.491	0.0578
7	230.0	0.1353	41.610	0.0559
8	245.0	0.1096	36.712	0.0569
9	255.0	0.0769	40.734	0.1128
Average			40.740	0.0223
DOSE RATE @ 35 M FROM NEAREST WP CENTER (MAIN DRIFT)				
Surface Segment	Outer Radius, cm	Area Fraction	driftgb.out, surface 17	
			mrem/hr	FSD
1	107.5	0.1777	2.087	0.3845
2	130.0	0.0822	2.505	0.2507
3	150.0	0.0861	2.129	0.2112
4	170.0	0.0984	1.554	0.1601
5	190.0	0.1107	1.270	0.1140
6	210.0	0.1230	1.729	0.1438
7	230.0	0.1353	1.918	0.2163
8	245.0	0.1096	2.392	0.3064
9	255.0	0.0769	1.462	0.1630
Average			1.900	0.1002

TABLE 7.4-6
LEF CO-60 GAMMA DOSE RATES WITH DOOR OPEN

PWR END FITTING CO-60 CONTRIBUTION				
18" CONCRETE SHADOW SHIELD ONLY				
DOSE RATE @ 13 M FROM NEAREST WP CENTER (DRIFT ENTRANCE DOOR)				
Surface Segment	Outer Radius, cm	Area Fraction	driftcob.out,surface 1611	
			mrem/hr	FSD
1	107.5	0.1777	31.896	0.0172
2	130.0	0.0822	31.309	0.0219
3	150.0	0.0861	29.520	0.0209
4	170.0	0.0984	30.160	0.0210
5	190.0	0.1107	29.253	0.0235
6	210.0	0.1230	27.843	0.0212
7	230.0	0.1353	27.445	0.0192
8	245.0	0.1096	27.441	0.0291
9	255.0	0.0769	27.812	0.0371
Average			29.274	0.0076
DOSE RATE @ 35 M FROM NEAREST WP CENTER (MAIN DRIFT)				
Surface Segment	Outer Radius, cm	Area Fraction	driftcob.out, surface 17	
			mrem/hr	FSD
1	107.5	0.1777	1.056	0.0493
2	130.0	0.0822	1.220	0.0915
3	150.0	0.0861	0.997	0.0756
4	170.0	0.0984	1.151	0.0925
5	190.0	0.1107	1.161	0.0711
6	210.0	0.1230	1.052	0.0668
7	230.0	0.1353	1.097	0.0578
8	245.0	0.1096	1.061	0.0682
9	255.0	0.0769	1.296	0.0916
Average			1.109	0.0238

7.4.3 Evaluation of Results

Table 7.4-7 summarizes the results of this shadow shield analysis, along with the unshielded results from Section 7.3 for comparison. It should be noted that the results are for the contribution from the closest waste package only. Additional contributions from other waste packages in the same emplacement drift, expected to be small, should be quantified in future analyses.

It is clear from Table 7.4-7 that the combination of the shadow shield and drift entrance door is effective to reduce the dose rates to 24 mrem/hr at the door entrance, and 0.8 mrem/hr at the nearest edge of the main drift. Since the dose rate in the main drift is less than 2.5 mrem/hr specified in Section 4.2.1, the main drift can be classified as a “normally occupied area” for 40 hr/wk occupational access.

When the drift entrance door is open for emplacement operations or drift inspection, the dose rates increase to 84 mrem/hr at the door, and 3.5 mrem/hr in the main drift. The dose rate at the door is less than 100 mrem/hr defined for a high radiation area. Therefore, intermittent access is permissible in the turnout areas, provided that the annual limit is not exceeded. The dose rate in the main drift is higher than the limit of 2.5 mrem/hr for “normally occupied areas”, some administrative control will be required to ensure that personnel radiation exposures are maintained ALARA with the drift door open.

It needs to be emphasized that this evaluation is limited to the material and thickness of the shadow shields. The design specification of the shadow shields should be sufficiently large in width and height to preclude a direct line of sight from the sides of any waste packages in the emplacement drifts to a potential worker located somewhere on the inside radius beyond the 35 m thermal standoff. Further shielding analysis is required to provide the detailed design specification of the shadow shields.

TABLE 7.4-7
SUMMARY RESULTS OF SHADOW SHIELD ANALYSIS
 (Contributions from the Nearest Waste Package Only)

<u>Contribution</u>	<u>Dose Rate (mrem/hr)</u>		
	<u>Unshielded Case</u>	<u>Shadow Shield With Drift Door Open</u>	<u>Shadow Shield with Drift Door Closed</u>
Dose Rate at Drift Entrance Door (13 m from nearest WP center)			
Fuel Neutrons	22.6 (0.0217) ^a	14.3 (0.0106)	10.1 (0.0092)
Fuel Gammas	192 (0.0308)	40.7 (0.0223)	6.90 (0.0338)
LEF Co-60	441 (0.0152)	29.3 (0.0076)	7.03 (0.0064)
Total	656 (0.0137)	84.3 (0.0112)	24.0 (0.0106)
Dose Rate at Main Drift (35 m from nearest WP center)			
Fuel Neutrons	1.10 (0.0809)	0.52 (0.0465)	0.26 (0.0351)
Fuel Gammas	17.1 (0.0642)	1.90 (0.1002)	0.29 (0.1223)
LEF Co-60	41.5 (0.0142)	1.11 (0.0238)	0.25 (0.0169)
Total	59.7 (0.0209)	3.53 (0.0549)	0.80 (0.0461)

^a The value in the parentheses is the associated statistical fractional standard deviation obtained as follows:

Unshielded case: Obtained from Table 7.3-1

Shadow shield with drift door open: Obtained from Tables 7.4-4 through 7.4-6 for the individual dose components; and calculated by the square root of $\Sigma(f_i\sigma_i)^2$ for the total, where f_i is the fractional contribution for dose component i , and σ_i is the associated fsd.

Shadow shield with drift door closed: Obtained from Tables 7.4-1 through 7.4-3 for the individual dose components; and calculated by the square root of $\Sigma(f_i\sigma_i)^2$ for the total, where f_i is the fractional contribution for dose component i , and σ_i is the associated fsd.

7.5 WASTE PACKAGE TRANSPORTER SHIELDING ANALYSIS

The waste package transporter shield design approach follows the common practice used in the design of spent fuel storage and transportation casks, which requires an optimized shield configuration to minimize the weight. The shield design provides both neutron and gamma shielding for personnel protection (per Assumption 4.3.1.5). According to Assumption 4.3.2.11, carbon steel is assumed for gamma shielding to balance the cost and shielding effectiveness; and borated polyethylene is assumed for neutron shielding to take advantage of its high hydrogen content for slowing down neutrons. The borated polyethylene contains 1.5% natural boron to minimize secondary gamma dose contributions. Both materials are commonly used for storage and transportation casks.

For weight minimization, the shielding layout for the transporter is arranged such that the heavy gamma shield material (carbon steel) is on the inside of the transporter, followed by the light neutron shield material (borated polyethylene). A layer of 5 mm SS316L is used to encase the shield materials to provide a smooth surface for ease of decontamination.

With dose contributions from both neutrons and gammas, there is an optimum neutron-to-gamma dose ratio where the transporter weight is at a minimum. Based on the design experience with the GA-4 and GA-9 transportation casks, the weight is near optimum when the neutron dose contribution on the outside of the shield is maintained at 20 to 30% of the total dose (Ref. 5.4). This ratio is used in this analysis to optimize the thicknesses of carbon steel and borated polyethylene.

The transporter shielding analysis includes radial shielding, axial shielding and bottom shielding. The radial shielding requirements are determined with the MCNP4A code using a cylindrical shell model with an infinite axial height. The gamma analysis is checked with the PATH code to verify the applicability of PATH for this analysis. With such verification, PATH is then used to efficiently perform the axial shielding evaluation for the end portions of the transporter without running the more sophisticated MCNP4A code. The bottom shielding requirements are evaluated using the results from the radial and axial shielding analyses.

7.5.1 Radial Shielding Analysis

7.5.1.1 Physical Model

For determination of the bulk radial shielding requirements, a simplified conservative cylindrical model was used, with an infinite axial length, in view of the large fuel length relative to the distances to the dose locations. This model simulates a one-dimensional cylindrical calculation. For simplicity, the model used a unit axial segment of 10 cm long with reflective boundaries on both ends to simulate the infinite length. Figure 7.5-1 illustrates the geometric model of the waste package and its transporter used for shielding analysis. Table 7.5-1 lists the radial intervals and material thicknesses for the regions considered in the model.

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FIGURE 7.5-1
WASTE PACKAGE TRANSPORTER
RADIAL SHIELD MODEL

TABLE 7.5-1
WASTE PACKAGE TRANSPORTER RADIAL GEOMETRIC MODEL

<u>Description</u>	<u>Radial Interval (cm)</u>	<u>Thickness (cm)</u>	<u>Reference or Basis</u>
Fuel Region	0.00 - 61.0	61.0	Fig. 7.3-1
Fuel Basket	61.00 - 62.68	1.68	Fig. 7.3-1
Radial Gap	62.68 - 69.45	6.77	Fig. 7.3-1
Waste Package			
Inner Barrier	69.45 - 71.45	2.0	Fig. 7.3-1
Outer Barrier	71.45 - 81.45	10.0	Fig. 7.3-1
Air Gap			
Before Max WP OD	81.45 - 92.50	11.05	Assumption 4.3.1.9
After Max WP OD	92.50 - 122.5	30.0	Assumption 4.3.2.15
Transporter			
SS316L Clad	122.5 - 123.0	0.5	Nominal cladding
Gamma Shield	123.0 - 138.24	15.24	Estimated thickness
Neutron Shield	138.24-148.4	10.16	Estimated thickness
SS316L Clad	148.4 - 148.9	0.5	Nominal cladding
Air Space	148.9 - 361.0	103.6	Assumption 4.3.1.6 less lining
Concrete Liner	361.0 - 381.0	20.0	Assumption 4.3.2.2
Drift Wall	381.0 - 395.0	14.0	Assumption 4.3.1.6 for inner radius; outer radius sufficient for back-scattering effects

The transporter inside diameter is based on an air gap of 0.3 m (Assumption 4.3.2.15) from the maximum envelope of the waste package (1.85 m diameter in Assumption 4.3.1.9). The change in the air gap dimension would have little impact on the shielding requirements, but would affect the transporter weight noticeably, which varies approximately with the radius. Note that all the dimensions used here are for modeling purposes only.

The model simulates the condition when the transporter travels in the main drift (Assumption 4.3.2.13). The bore diameter of the main drift is 7.62 m (Assumption 4.3.1.6). Per Assumption 4.3.2.2, the drift is lined with 0.2 m concrete. This lining is included in the model to account for drift wall scattering effects.

7.5.1.2 Shielding Evaluation

The shielding analysis was performed with the MCNP4A code, including both neutron and gamma radiation fields. Two separate MCNP4A calculations were executed: one for the neutron and secondary gamma contributions, and the other for the primary gamma contributions from the active fuel region.

For shielding analysis, the energy groups which do not contribute to the dose rates of interest can be ignored, especially in the 3-D Monte Carlo analysis. For gammas, the energy groups in the range of 0.45 to 3.0 MeV make virtually all contributions to the dose rates outside the waste package and transporter (Section 7.1.2). Therefore, only these contributing groups are considered in the present analysis.

To improve statistics, selected biasing schemes were used in the MCNP4A calculations, including source biasing, cell importance specification, and energy splitting biasing. The source biasing consisted of position and energy biasing by placing more particles in outer source cells and in high energy bins. The source energy biasing was more essential for the gamma source than for the neutron source, as the 1.0 - 1.5 MeV gamma energy group generally contributes >50% of the gamma dose, despite its small source fraction (see Section 4.1.3).

The cell importance was specified by subdividing each major material interval into subcells. For instance, the carbon steel gamma shield for the transporter was divided into 3 subcells at 5.08 cm interval each, and the neutron shield region was composed of 4 subcells at 2.54 cm interval each. The cell importance was specified separately for neutrons and gammas, because of their different attenuation characteristics. A general rule of thumb for the cell importance specification is to increase the cell importance by a factor of 2, when the quantity of interest (such as dose rate) drops by a factor of 2 to 4. This rule provides a good balance of the particle's weight without over-biasing or under-biasing.

Energy splitting was used in the random walk process to play Russian roulette as the particle energy got lower and lower. This biasing improved calculational efficiency by killing the less important particles sooner.

To obtain the absolute results, appropriate normalization factors were included in the MCNP4A input for applying to the calculated results "per source particle". The normalization factors were 6.567×10^9 (Table 7.1-1) $\times (10/360.18) = 1.823 \times 10^9$ n/s for the neutron and secondary gamma calculations, and 5.510×10^{16} (Table 7.1-2 for gamma energies from 0.45 to 3.0 MeV only) $\times (10/360.18) = 1.530 \times 10^{15}$ γ/s for the primary gamma calculations. These factors corresponded to the total number of source particles in the model determined by adjusting the total number of source particles in the waste package from Sections 7.1.1 and 7.1.2 by the height ratio (10 cm in the model to the active fuel length of 360.18 cm).

With the infinite axial model, surface detector tallies were employed to obtain the results at various radial distances from the waste package through the transporter and to the drift wall. The surface detector tallies are more efficient than the point detector tallies. The two types of the tallies should give approximately equal results at the mid-plane.

7.5.1.3 Radial Shielding Results

Radial shielding results are obtained from the following MCNP4A runs:

Neutron & secondary gamma analysis:	Input file TSPRT01N, output file TSPRT01N.OUT
Primary gamma analysis:	Input file TSPRT01G, output file TSPRT01G.OUT

Attachment VII provides listing of the input files. Ref. 5.29 stores all the input and output files on electronic media.

Table 7.5-2 provides the radial dose rate results of the MCNP4A calculations. The results are also plotted in Figure 7.5-2 as a function of radial distance from the WP centerline. The results include all the contributing components: neutrons, primary gammas, and secondary gammas, as well as all scattering from drift walls. The one-sigma statistical uncertainty for each contributing component is indicated in parentheses. These results are based on the ANSI/ANS-6.1.1-1977 dose conversion factor standard. All the results are statistically satisfactory ($fsd < 0.1$ for surface tallies), per guidelines in Ref. 5.5.

The total dose rate at the surface of the transporter is 33 mrem/hr, comprising 8.3 mrem/hr from neutrons, 17.9 mrem/hr from primary gammas, and 6.8 mrem/hr from secondary gammas. This dose rate meets the limit of 50 mrem/hr at the surface specified in Assumption 4.3.2.7, confirming the shielding adequacy for the transporter with 152.4 mm carbon steel and 101.6 mm B-poly clad in 5 mm SS316L. The neutron dose is about 25% of the total dose, indicating that the shielding design is near optimum for a minimum transporter weight. Note that these results are for the PWR DBF. The dose rate values for the average fuel characteristics are much less (See Section 7.6).

TABLE 7.5-2
RADIAL DOSE RATE PROFILE FOR WASTE PACKAGE TRANSPORTER

RADIAL DOSE RATE RESULTS FOR WASTE PACKAGE TRANSPORTER									
Region	Radial Dist. (cm)	Dose Rate (mrem/hr)				Fractional Standard Deviation			
		Neutron	Primary G	Second. G	Total	Neutron	Primary G	Second. G	Total
		tsprt01n.out	tsprt01g.out	tsprt01n.out	calculated	tsprt01n.out	tsprt01g.out	tsprt01n.out	calculated
Fuel	20.00	3.298E+04	7.588E+07	1.031E+02	7.591E+07	0.0071	0.0242	0.0168	0.0242
	35.00	3.002E+04	7.490E+07	9.794E+01	7.493E+07	0.0054	0.0156	0.0140	0.0156
	45.00	2.652E+04	7.399E+07	8.795E+01	7.402E+07	0.0048	0.0117	0.0127	0.0117
	54.00	2.254E+04	7.027E+07	7.771E+01	7.029E+07	0.0048	0.0089	0.0120	0.0089
Liner	61.00	1.820E+04	4.019E+07	7.126E+01	4.021E+07	0.0050	0.0103	0.0132	0.0103
Gap	62.68	1.443E+04	1.251E+07	6.252E+01	1.252E+07	0.0051	0.0095	0.0166	0.0095
	69.45	1.400E+04	1.102E+07	6.136E+01	1.103E+07	0.0058	0.0106	0.0178	0.0106
WP	71.45	1.160E+04	4.054E+06	4.624E+01	4.066E+06	0.0054	0.0125	0.0139	0.0125
	76.45	7.228E+03	3.798E+05	2.439E+01	3.871E+05	0.0059	0.0197	0.0177	0.0193
	81.45	4.307E+03	3.551E+04	1.527E+01	3.983E+04	0.0061	0.0220	0.0246	0.0196
Transporter	122.50	3.716E+03	2.328E+04	1.496E+01	2.701E+04	0.0067	0.0216	0.0275	0.0186
	123.00	3.614E+03	1.955E+04	1.468E+01	2.318E+04	0.0065	0.0222	0.0208	0.0188
	128.08	2.499E+03	2.439E+03	9.972E+00	4.948E+03	0.0066	0.0276	0.0218	0.0140
	133.16	1.594E+03	3.129E+02	9.087E+00	1.916E+03	0.0067	0.0367	0.0246	0.0082
	138.24	8.070E+02	4.460E+01	1.921E+01	8.708E+02	0.0065	0.0393	0.0240	0.0064
	140.78	2.141E+02	3.966E+01	2.079E+01	2.746E+02	0.0083	0.0403	0.0180	0.0088
	143.32	6.607E+01	3.522E+01	1.725E+01	1.185E+02	0.0107	0.0412	0.0173	0.0138
	145.86	2.188E+01	2.988E+01	1.332E+01	6.508E+01	0.0137	0.0372	0.0176	0.0181
	148.40	9.289E+00	2.455E+01	1.011E+01	4.395E+01	0.0192	0.0386	0.0189	0.0224
	148.90	8.302E+00	1.791E+01	6.781E+00	3.299E+01	0.0204	0.0338	0.0211	0.0195
Air Space	250.00	6.024E+00	1.083E+01	3.633E+00	2.049E+01	0.0225	0.0460	0.0206	0.0255
Drift Wall	361.00	5.412E+00	8.305E+00	2.789E+00	1.651E+01	0.0234	0.0433	0.0205	0.0234

Fig. 7.5-2 Radial Dose Rate Profile for Waste Package Transporter

7.5.2 Axial Shielding Analysis

The axial shielding analysis for the waste package transporter was performed with the PATH code. The geometric model for the PATH calculations is depicted in Figure 7.5-3, which includes the half of the waste package containing the PWR fuel assemblies with lower end fittings. The axial shielding dimensions are changed from the radial shielding dimensions to the following:

SS316L	5 mm	(0.2")
Carbon Steel	177.8 mm	(7")
B-poly	76.2 mm	(3")
SS316L	5 mm	(0.2")

The rationale for this change is as follows: More gamma shielding (carbon steel) is required to account for increased gamma radiation field from the end fittings; and neutron shielding (B-poly) can be reduced because of lower-than-average burnup at the fuel ends.

The PATH model includes both radial and axial directions. The radial model, basically identical to the MCNP4A model described in Section 7.5.1.1, is used to verify the applicability of the PATH code to this analysis.

As PATH is a gamma shielding code, only the gammas from the active fuel and end fittings are treated with PATH. The neutron and secondary gamma contributions in the axial direction are conservatively taken to be the same as those in the radial direction. The lower burnup at the fuel ends generally provides more than sufficient compensation for the reduced axial neutron shielding thickness, resulting in a lower neutron dose and accompanying secondary gamma dose axially than those radially.

The PATH calculations including the following cases. Attachment VII lists the input files. Ref. 5.29 stores all the input and output files on electronic media.

a. Radial fuel gamma analysis:	Input file TSPRT01.INP, output file TSPRT01.OUT
b. Axial fuel gamma analysis:	Input file TSPRT03.INP, output file TSPRT03.OUT
c. Axial LEF Co-60 analysis:	Input file TSPRT03A.INP, output file TSPRT03A.OUT

Case a is intended to verify the applicability of PATH by comparing with the MCNP4A result in Section 7.5.1.3. The comparison is made for the gamma dose rate from the active fuel region on the transporter surface at the WP mid-plane. The PATH result for this case is 17.6 mrem/hr versus 17.9 mrem/hr from MCNP4A. The agreement is obviously excellent between PATH and MCNP4A. Therefore, it is satisfactory to use the PATH code for the transporter gamma shielding analysis. Accordingly, the axial gamma shielding analysis is performed with PATH rather than the more sophisticated MCNP4A code.

DOSE POINTS

ALL DIMENSIONS ARE IN CENTIMETERS

FIGURE 7.5-3
PATH MODEL OF
WASTE PACKAGE TRANSPORTER

Cases b and c provide the results of the gamma dose rates from the active fuel region and LEF Co-60 sources, respectively. The dose point is located on the end surface of the transporter on the WP centerline. The results are presented below:

Active fuel gamma contribution:	4.88 mrem/hr (TSPRT03.OUT)
LEF Co-60 gamma contribution:	18.5 mrem/hr (TSPRT03A.OUT)
Active fuel neutron contribution:	8.30 mrem/hr (taken from Table 7.5-2 @ 148.9 cm)
Secondary gamma contribution:	6.78 mrem/hr (taken from Table 7.5-2 @ 148.9 cm)

Total dose rate:	38.5 mrem/hr

This axial dose rate of 38.5 mrem/hr is similar to the radial dose rate of 33 mrem/hr on the transporter surface, and is below the limit of 50 mrem/hr given in Assumption 4.3.2.7. The similar radial and axial dose rates indicate that the radial and axial shielding configurations have about the same shielding effectiveness.

7.5.3 Bottom Shielding Analysis

The objective of this evaluation is to examine the feasibility of reduced transporter bottom shielding in order to reduce the transporter overall weight. Assumption 4.3.2.7 specifies that the dose rate on the bottom surface of the transporter be limited to less than 100 mrem/hr to facilitate repair when the transporter is still loaded with a waste package. This limit is higher than the limit of 50 mrem/hr used for the sidewall and ends, resulting in less bottom shielding for the transporter.

From the results in Table 7.5-2, the carbon steel is effective to reduce the gamma dose rate by a factor of 2.8 per inch of steel, and the B-poly material provides a factor of 3 reduction in the neutron dose rate per inch. Since the B-poly material is much lighter than steel, the thickness of B-poly for radial shielding is retained for bottom shielding. In other words, the B-poly thickness for the bottom shielding remains at 101.6 mm (4 in.). The carbon steel thickness is reduced to 127 mm (5 in.), resulting in an increase in the primary gamma contribution from the active fuel from 17.9 mrem/hr to 50 mrem/hr (17.9×2.8). The reduction in the carbon steel thickness will also increase the neutron and secondary gamma contributions from 15 mrem/hr (8.3 for neutrons + 6.78 for secondary gammas) to about 19 mrem/hr (15×1.25). The resulting total dose rate on the bottom surface of the transporter is 69 mrem/hr ($50 + 19$). This dose rate is less than the desired limit of 100 mrem/hr, leaving some margin for calculational uncertainties and modeling approximation. Future analysis may be required to optimize the bottom shielding thickness.

7.5.4 Summary of Transporter Shielding Analysis

Table 7.5-3 provides a summary of the shielding thicknesses for the waste package transporter, including radial (sidewall), axial (end), and bottom shielding. The dose rates on the various surfaces of the transporter are also indicated.

The transporter external dose rate exceeds the limit of 2.5 mrem/hr for 40 hr/wk access in normally occupied areas (Criteria 4.2.1). Therefore, the vicinity of the transporter, when loaded, should be controlled as an "intermittently occupied area" to control personnel exposures.

It should be noted that the design of the transporter is still evolving. The shielding results presented here are for preliminary design of the transporter to establish the required shielding thicknesses which may be converted into weight requirements in follow-on analysis. More detailed shielding analysis may be required to confirm the design of the transporter, with considerations given to the actual geometry and axial source distribution. Furthermore, additional shielding optimization may be warranted to achieve the minimum transporter weight without compromising radiation safety requirements.

TABLE 7.5-3
SUMMARY OF TRANSPORTER SHIELDING ANALYSIS

<u>Description</u>	<u>Radial or Sidewall</u>	<u>Axial or End</u>	<u>Bottom</u>
Shield Thickness, mm (in.)			
SS316L	5.0 (0.2)	5.0 (0.2)	5.0 (0.2)
Carbon Steel	152 (6.0)	178 (7.0)	127 (5.0)
B-poly	102 (4.0)	76 (3.0)	102 (4.0)
SS316L	5.0 (0.2)	5.0 (0.2)	5.0 (0.2)
Surface Dose Rate, mrem/hr	33	38.5	69

7.6 ALARA EVALUATION

The time and motion data for emplacement operations need to be developed as the repository design progresses. Pending the availability of such data, the ALARA evaluation is performed on a limited basis by considering average SNF characteristics. Use of the average SNF is a logical choice, as the radiation exposure limit is on an annual basis rather than at a particular time. Over the year, the radiation exposures from various waste packages should be averaged out to correspond to the average SNF.

Since the last-emplaced waste package contributes over 90% of the dose rates in the main drift and turnout areas, the ALARA objective can be achieved by placing the waste package with average fuel characteristics in the last location of the emplacement drift.

In this evaluation, the radiation fields in the main drift and turnout areas are determined using the average SNF source terms given in Section 4.1.6. The evaluation is to address the criteria statement in Section 4.2.2 regarding the ALARA goal of 1 rem/yr or 0.5 mrem/hr in normally occupied areas. In addition, the radiation field outside the waste package transporter is evaluated to quantify the effect of using the average SNF.

7.6.1 Average SNF Source Term Evaluation

The average SNF source terms are evaluated for PWR fuel assemblies against the design basis fuel. Per Section 1, other waste forms (BWR SNF and DHLW) are not considered. The evaluation includes the neutron, fuel gamma, and LEF Co-60 sources, which contribute to the radiation fields of interest. Table 7.6-1 provides a comparison of the average and design basis PWR fuel, along with the source term ratios for use in this ALARA evaluation.

To obtain the dose rate for the average fuel, the neutron source term ratio can be used directly for scaling purposes, as the neutron energy spectrum remains fairly constant with time. The contribution from Co-60 can also be directly scaled, since it is a single isotope contribution.

The contribution from active fuel gammas requires special attention. Only the important energy groups which contribute to the radiation fields need to be considered. Based on the results from Sections 7.3 through 7.5, the 1.25 MeV group is the principal contributing group, generally accounting for about 70% of the dose rate due to active fuel gammas. Therefore, the fuel gamma source term ratio for the 1.25 MeV group is used as a representative ratio for scaling the active fuel gamma dose contribution. This approximation is verified in Section 7.6.3 to be satisfactory.

7.6.2 Radiation Fields in Main Drift and Turnout Areas

Using the source term ratios between the average and design basis fuel, the dose rate results in Table 7.4-7 are scaled down to obtain the corresponding results for the average PWR fuel. The results are presented in Table 7.6-2.

TABLE 7.6-1
AVERAGE SNF VERSUS DESIGN BASIS FUEL

<u>Description</u>	<u>Average Fuel</u>	<u>Design Basis Fuel</u>	<u>Ratio^a</u>
Characteristics			
Enrichment (% U-235)	3.72	4.2	N/A ^d
Burnup (MWd/MTU)	39750	48086	N/A
Decay Time (year)	26 ^b	10	N/A
Neutron Source (n/s-MTU)	2.241+08	6.740+08	0.332
Gamma Source (γ's/s-MTU)			
Total	5.535+15	1.124+16	0.492
1.25 MeV Group	5.668+13	3.849+14	0.147
LEF Co-60 (Ci/assembly)	12.1 ^c	121	0.100

^a Ratio of average to design basis fuel

^b Assumption 4.3.1.1

^c Obtained by $(121 \text{ Ci for DBF})(39750/48086)/2^{(16/5.27)}$, where 16 is the decay time (yr) from 10 to 26 years, and 5.27 is the Co-60 half life in years.

^d Not applicable

TABLE 7.6-2
RADIATION FIELDS FOR AVERAGE FUEL
 (Contributions from the Nearest Waste Package Only)

<u>Contribution</u>	<u>Dose Rate (mrem/hr)</u>		
	<u>No Shadow Shield</u>	<u>Shadow Shield With</u>	<u>Shadow Shield with</u>
	<u>Drift Door Open</u>	<u>Drift Door Open</u>	<u>Drift Door Closed</u>
Dose Rate at Drift Entrance Door (13 m from nearest WP center)			
Fuel Neutrons	7.50	4.75	3.35
Fuel Gammas	28.2	5.98	1.01
LEF Co-60	44.1	2.93	0.70
Total	79.8	13.7	5.06
Dose Rate at Main Drift (35 m from nearest WP center)			
Fuel Neutrons	0.37	0.17	0.09
Fuel Gammas	2.51	0.28	0.04
LEF Co-60	4.15	0.11	0.03
Total	7.03	0.56	0.16

With the average fuel, the dose rate at the main drift (35 m thermal standoff, per Assumption 4.3.1.10) is 0.16 mrem/hr with the concrete shadow shield in place and drift entrance door closed. This dose rate is far below the ALARA goal of 0.5 mrem/hr in normally occupied areas (Criteria 4.2.2). When the door is open for drift inspection, the dose rate increases to 0.56 mrem/hr at the main drift. This dose rate is only slightly above the 0.5 mrem/hr goal. Consideration of the credit for shielding by the railcar loading dock would decrease this radiation field below 0.5 mrem/hr required to meet the ALARA goal.

7.6.3 Radiation Field External to Transporter

The radiation field external to the waste package transporter is determined using the average fuel for comparison with the design basis fuel. The determination includes the radial and axial dose rates on the transporter surface.

The source term scaling factors are used to scale the dose rate results in Sections 7.5.1 and 7.5.2 for the DBF to yield the results for the average fuel. The scaling factor for the secondary gamma contribution is taken to be the same as the neutron scaling factor, since secondary gamma production is proportional to the neutron source.

Use of the scaling factor for the fuel gamma contribution is verified with the PATH code. This PATH model is identical to Case a in Section 7.5.2 for the radial gamma analysis, with the exception that the fuel gamma source terms are for the average fuel. The input and output files for this PATH analysis are TSPRT01A.INP and TSPRT01A.OUT, respectively. The PATH output shows that the dose rate on the surface of the transporter is 2.8 mrem/hr due to the active fuel gammas. The value is in good agreement with the value of 2.6 mrem/hr ($17.9 \text{ mrem/hr for DBF} \times 0.147 \text{ source ratio for the } 1.25 \text{ MeV group}$) based on the scaling method. Therefore, the scaling approach is acceptable.

Table 7.6-3 presents the dose rate results for the average fuel. The radial dose rate is reduced from 33 mrem/hr for the DBF to 7.64 mrem/hr for the average fuel; and the axial dose rate from 38.5 mrem/hr to 7.58 mrem/hr. The difference in the dose rates between the DBF and average fuel provides a large margin (about a factor of 5) to meet the ALARA requirements.

Attachment VIII lists the TSPRT01A.INP input file. Ref. 5.29 stores the TSPRT01A.INP and TSPRT01A.OUT on electronic media.

TABLE 7.6-3
WASTE PACKAGE TRANSPORTER SURFACE DOSE RATES FOR AVERAGE FUEL

<u>Contribution</u>	Average/DBF <u>Source Ratio</u>	<u>Surface Dose Rate (mrem/hr)</u>	
		<u>Radial</u>	<u>Axial</u>
Fuel Neutrons	0.332	2.76	2.76
Fuel Gammas	0.147	2.63	0.72
Secondary Gammas	0.332	2.25	2.25
LEF Co-60 Gammas	0.100	-	1.85
Total		7.64	7.58

7.7 TIME-DEPENDENT RADIATION CONDITIONS

This analysis determines the time-dependent radiation fields from waste packages in emplacement drifts for normal and off-normal conditions. This activity is in support of the repository design to respond to these conditions. For normal conditions, both dose rates and integrated doses are calculated as a function of time over 150 years after emplacement of waste packages (Assumption 4.3.1.8). The off-normal condition is evaluated for a breached waste package containing the design basis fuel with the characteristics of 4.2% initial enrichment, 48086 MWd/MTU burnup and 10 years decay. These radiation field data are important for evaluation of sensors, alarms, and control systems in the overall repository monitoring system that may be subjected to long-term radiation conditions from intact waste packages and/or high radiation fields from accident conditions.

7.7.1 General Approach

The analysis requires the time-dependent source data for neutrons and gammas. These data are obtained from the CDB for the design basis fuel (4.2% initial enrichment, 48086 MWd/MTU burnup and 10 years decay). This analysis considers the neutron and gamma sources from the active fuel region only. These sources are representative of the general radiation fields in the emplacement drifts. The end-fitting Co-60 sources are not considered, since these sources affect the radiation fields on the ends only with the resulting radiation field similar to that from the active fuel gammas.

The radiation fields from waste packages include all significant neutron and gamma contributions. The neutron and gamma dose rates are calculated with the MCNP4A code at the time of emplacement (e.g., 10 years decay for commercial SNF). The neutron dose rate is then proportionally adjusted by the total source strength (for all energy groups) to obtain the respective values at other decay times, since the neutron energy spectrum does not vary significantly with time and the neutron dose rate is fairly insensitive to the neutron energy spectrum. The time-dependent gamma dose rates are determined with the PATH code for the various decay times from 10 years to 160 years (10 years decay plus 150 year maintainable service life). The PATH results are then normalized to the MCNP4A result at the time of emplacement (10 years decay). This approach simplifies the calculational process without running the MCNP4A code at every decay time.

A fitting equation is used to express the dose rate as a function of time. For accuracy and ease of computation, the time span of 150 years is divided into a number of intervals. Each time interval has a separate fitting equation, in exponential form similar to the decay equation. The fitting equations are used to calculate the integrated dose at a given time after emplacement.

7.7.2 Normal Conditions

For normal conditions, the waste packages in the emplacement drifts remain intact with no loss of shielding effectiveness by the inner and outer barriers of the waste packages. Figure 7.7-1 depicts the geometric model of the waste packages in an emplacement drift, represented by a cylindrical

configuration with an infinite axial length to simulate the large number of waste packages in the drift. The model includes radiation scattering off the concrete lining.

7.7.2.1 Radiation Fields

The radiation fields are calculated on the surface of the waste package, and on the surface of the concrete lining. The radiation fields are expressed in units of rem/hr for human exposure. For evaluation of radiation effects on the materials and equipment used in the drifts, the dose rates are generally expressed in units of rad/hr for absorbed energy deposition in a given material. For gammas which have a quality factor of 1 for biological effects, 1 rad/hr is close to 1 rem/hr. However, 1 rad/hr neutron absorbed dose rate is 5-20 rem/hr dose equivalent rate for neutrons, since the quality factor for neutrons is 5 or greater depending on the energy. With the neutron dose contribution being small relative to the gamma contribution, the units of rem/hr and rad/hr are considered equivalent here.

The MCNP4A code is used to calculate the neutron and gamma dose rates for the design basis fuel. The MCNP4A runs include the following:

Neutron calculation:	Input file INFWP01N	Output file INFWP01N.OUT
Gamma calculation:	Input file INFWP01G	Output file INFWP01G.OUT

Table 7.7-1 provides the results of the MCNP4A calculations at the two dose locations of interest. Note that these results are for the DBF, which are used later to normalize the PATH results.

For time dependence, the PATH code is used to calculate the gamma dose rates for the following decay times: 10, 15, 20, 30, 40, 50, 60, 70, 80, 90, 100, 110, 120, 130, 140, 150 and 160 years. The gamma source data in Section 4.1.5 are converted to photons/s-cc as provided in Attachment III for input to PATH. The PATH calculations take advantage of the geometric symmetry by modeling the half length of the fuel source. The PATH output is multiplied by 2 to account for the full length. The PATH results are then adjusted by normalizing the gamma dose rate at 10 years decay to the MCNP4A gamma result given in Table 7.7-1. For example, the PATH results are 26.25 rem/hr at 10 years, and 15.13 rem/hr at 15 years. The normalized value at 15 years is then 35.20 rem/hr (from MCNP4A at 10 years) $\times (15.13/26.25) = 20.29 \text{ rem/hr}$. Table 7.7-2 lists the unnormalized and normalized PATH results for the various decay times, along with the input and output file names.

The unnormalized and normalized results can also be interpreted as the results from the PATH and MCNP4A calculations, respectively. The MCNP4A results are used here as the reference results, because of better modeling and physics with MCNP4A. The difference between PATH and MCNP4A is less than a factor of 2, which is within calculational uncertainties in shielding analysis between the simple and detailed models used.

Attachment IX lists the MCNP4A and PATH input files. Ref. 5.29 stores all the input and output files on electronic media.

As mentioned in Section 7.7.1, the time-dependent neutron dose rates are obtained by scaling from the MCNP4A neutron result at 10 years decay by the neutron source strength ratios (using the neutron source data in Section 4.1.4). Table 7.7-3 summarizes the time-dependent radiation fields for in-drift waste packages. The results are also plotted in Figure 7.7-2. These results are used for determination of time-integrated doses over the 150-year preclosure maintainable service life.

As shown in Table 7.7-3 and Figure 7.7-2, the radiation field on the surface of the waste package is 38 rem/hr at the time of emplacement, based on the design basis fuel. The radiation field decreases with time, dropping to less than 1 rem/hr on the WP surface at 90 years after emplacement. The radiation field at the concrete liner starts at about 14 rem/hr, and decreases to 1 rem/hr at 50 years after emplacement.

The radiation fields presented in this section are at the mid-length of the waste package. The radial radiation field of 38 rem/hr (Table 7.7-3 at 10 years decay) on the WP surface is similar to the field of 39 rem/hr (Table 7.3-1, Axial Point 1) near the WP end. Note that the main contributor to the radial radiation field is the gamma source in the active fuel region, while the end fitting Co-60 source accounts for over 70% of the axial radiation field. These radial and axial results indicate that the conservatism used in the end fitting Co-60 source term (Section 7.2) would not cause excessive contributions to the in-drift radiation fields.

TABLE 7.7-1

MCNP4A RESULTS OF IN-DRIFT RADIATION FIELDS

(PWR DBF: 4.2% Initial Enrichment, 48086 MWd/MTU Burnup and 10 Years Decay)

<u>Contribution</u>	<u>Dose Rate (rem/hr or rad/hr)</u>		<u>Output</u>
	<u>WP Surface</u>	<u>Concrete Lining</u>	<u>File</u>
Neutrons	2.793 (0.004)	1.388 (0.005) ^a	infwp01n.out
Gammas	35.20 (0.014)	12.33 (0.016)	infwp01g.out
Total	37.99 (0.013)	13.72 (0.014)	

^a Fractional standard deviation in parentheses.

TABLE 7.7-2
PATH RESULTS OF TIME-DEPENDENT IN-DRIFT GAMMA RADIATION FIELDS

Decay Time (yr)	Time After Emplacement (yr)	<u>WP Surface Dose Rate (rem/hr)</u>		Input File	Output File
		<u>Unnormalized</u>	<u>Normalized</u>		
10	0	26.25	35.20	pwrwp10.inp	pwrwp10.out
15	5	15.13	20.29	Pwrwp15.inp	pwrwp15.out
20	10	9.938	13.33	pwrwp20.inp	pwrwp20.out
30	20	5.215	6.994	pwrwp30.inp	pwrwp30.out
40	30	3.245	4.351	pwrwp40.inp	pwrwp40.out
50	40	2.213	2.967	pwrwp50.inp	pwrwp50.out
60	50	1.615	2.165	pwrwp60.inp	pwrwp60.out
70	60	1.225	1.643	pwrwp70.inp	pwrwp70.out
80	70	0.949	1.271	pwrwp80.inp	pwrwp80.out
90	80	0.743	0.996	pwrwp90.inp	pwrwp90.out
100	90	0.584	0.785	pwrwp100.inp	pwrwp100.out
110	100	0.460	0.616	pwrwp110.inp	pwrwp110.out
120	110	0.364	0.489	pwrwp120.inp	pwrwp120.out
130	120	0.288	0.387	pwrwp130.inp	pwrwp130.out
140	130	0.229	0.306	pwrwp140.inp	pwrwp140.out
150	140	0.182	0.243	pwrwp150.inp	pwrwp150.out
160	150	0.144	0.194	pwrwp160.inp	pwrwp160.out

Decay Time (yr)	Time After Emplacement (yr)	<u>Concrete Lining Dose Rate (rem/hr)</u>		Input File	Output File
		<u>Unnormalized</u>	<u>Normalized</u>		
10	0	7.468	12.33	pwrwp10.inp	pwrwp10.out
15	5	4.296	7.107	Pwrwp15.inp	pwrwp15.out
20	10	2.813	4.668	pwrwp20.inp	pwrwp20.out
30	20	1.468	2.450	pwrwp30.inp	pwrwp30.out
40	30	0.909	1.524	pwrwp40.inp	pwrwp40.out
50	40	0.617	1.039	pwrwp50.inp	pwrwp50.out
60	50	0.449	0.758	pwrwp60.inp	pwrwp60.out
70	60	0.340	0.576	pwrwp70.inp	pwrwp70.out
80	70	0.264	0.445	pwrwp80.inp	pwrwp80.out
90	80	0.206	0.349	pwrwp90.inp	pwrwp90.out
100	90	0.162	0.275	pwrwp100.inp	pwrwp100.out
110	100	0.128	0.216	pwrwp110.inp	pwrwp110.out
120	110	0.101	0.171	pwrwp120.inp	pwrwp120.out
130	120	0.080	0.136	pwrwp130.inp	pwrwp130.out
140	130	0.063	0.107	pwrwp140.inp	pwrwp140.out
150	140	0.050	0.085	pwrwp150.inp	pwrwp150.out
160	150	0.040	0.068	pwrwp160.inp	pwrwp160.out

TABLE 7.7-3
TIME-DEPENDENT IN-DRIFT RADIATION FIELDS
(PWR Fuel, 4.2% Initial Enrichment, and 48086 MWd/MTU Burnup)

Decay Time (yr)	Years After Emplacement	WP Surface Dose Rate (rem/hr)			Concrete Lining D. R. (rem/hr)		
		Neutron	Gamma	Total	Neutron	Gamma	Total
10	0	2.793	35.20	37.99	1.388	12.33	13.72
15	5	2.322	20.29	22.61	1.154	7.107	8.261
20	10	1.933	13.33	15.26	0.961	4.668	5.629
30	20	1.346	6.994	8.340	0.669	2.450	3.119
40	30	0.949	4.351	5.300	0.471	1.524	1.995
50	40	0.672	2.967	3.639	0.334	1.039	1.373
60	50	0.488	2.165	2.653	0.243	0.758	1.001
70	60	0.364	1.643	2.007	0.181	0.576	0.757
80	70	0.276	1.271	1.547	0.137	0.445	0.582
90	80	0.213	0.996	1.209	0.106	0.349	0.455
100	90	0.165	0.785	0.950	0.082	0.275	0.357
110	100	0.136	0.616	0.752	0.067	0.216	0.283
120	110	0.116	0.489	0.605	0.058	0.171	0.229
130	120	0.102	0.387	0.489	0.051	0.136	0.187
140	130	0.092	0.306	0.398	0.046	0.107	0.153
150	140	0.085	0.243	0.328	0.042	0.085	0.127
160	150	0.080	0.194	0.274	0.040	0.068	0.108

Fig. 7.7-2 Time-dependent In-drift Radiation Fields

7.7.2.2 Integrated Radiation Doses

To calculate the integrated dose at each time interval, the following approximation is used:

$$D(t) = D1 * e^{-\lambda t} \quad (\text{Eq. 7.7-1})$$

where $D(t)$ = Dose rate at time t (rem/hr)

$D1$ = Dose rate at the beginning of the time interval (rem/hr)

λ = Decay factor (1/yr)

t = Time (yr)

The decay factor λ is determined by:

$$\lambda = [1/(T2-T1)] \text{Log}_e (D1/D2) \quad (\text{Eq. 7.7-2})$$

where $T1$ = Time at the beginning of the time interval (yr)

$T2$ = Time at the end of the time interval (yr)

$D1$ = Dose rate at $T1$ (rem/hr)

$D2$ = Dose rate at $T2$ (rem/hr)

The integrated dose for time interval $T1$ to $T2$ is then the integral of $D(t)dt$ over $T1$ to $T2$, which is equal to:

$$\text{Dose (rem)} = (D1 * 8766 / \lambda) [1 - e^{-\lambda(T2-T1)}] \quad (\text{Eq. 7.7-3})$$

where 8766 is the number of hours per year ($24 * 365.25$) to convert the dose rate from rem/hr to rem/yr.

Attachment IV shows the calculations of the integrated doses using Eq. 7.7-3. Separate calculations are performed for neutrons and gammas at the waste package surface and concrete liner. Table 7.7-4 summarizes the time-integrated in-drift radiation doses. Figure 7.7-3 plots the integrated dose at the waste package surface and concrete liner surface as a function of time after waste emplacement.

The integrated dose results presented in Table 7.7-4 and Figure 7.7-3 are useful for evaluation of radiation effects on the materials. As can be observed from Figure 7.7-3, the materials used in the emplacement drift receive most of the doses in the early preclosure period of time. The cumulative dose reaches about 40% at 10 years after emplacement, and 84% at 50 years after emplacement. Radiation effects on the materials, if any, should appear in the early time. Therefore, performance confirmation with respect to the radiation effects should focus on the first 50 years after initiation of waste emplacement.

TABLE 7.7-4
TIME-INTEGRATED RADIATION DOSES

Years After Emplacement	Cumulative WP Surface Dose (rem)			Cumulative Concrete Lining Dose (rem)		
	Neutron	Gamma	Total	Neutron	Gamma	Total
0	0	0	0	0	0	0
5	1.118e+05	1.186e+06	1.298e+06	5.555e+04	4.155e+05	4.711e+05
10	2.048e+05	1.912e+06	2.117e+06	1.018e+05	6.698e+05	7.716e+05
20	3.470e+05	2.773e+06	3.120e+06	1.724e+05	9.714e+05	1.144e+06
30	4.465e+05	3.261e+06	3.708e+06	2.219e+05	1.142e+06	1.364e+06
40	5.169e+05	3.578e+06	4.095e+06	2.568e+05	1.253e+06	1.510e+06
50	5.673e+05	3.801e+06	4.368e+06	2.819e+05	1.332e+06	1.614e+06
60	6.044e+05	3.967e+06	4.571e+06	3.004e+05	1.390e+06	1.690e+06
70	6.323e+05	4.094e+06	4.726e+06	3.142e+05	1.434e+06	1.748e+06
80	6.536e+05	4.193e+06	4.847e+06	3.248e+05	1.469e+06	1.794e+06
90	6.700e+05	4.271e+06	4.941e+06	3.330e+05	1.496e+06	1.829e+06
100	6.832e+05	4.332e+06	5.015e+06	3.395e+05	1.517e+06	1.857e+06
110	6.942e+05	4.380e+06	5.074e+06	3.450e+05	1.534e+06	1.879e+06
120	7.038e+05	4.418e+06	5.122e+06	3.498e+05	1.548e+06	1.898e+06
130	7.123e+05	4.449e+06	5.161e+06	3.540e+05	1.558e+06	1.912e+06
140	7.200e+05	4.473e+06	5.193e+06	3.579e+05	1.567e+06	1.925e+06
150	7.272e+05	4.492e+06	5.219e+06	3.615e+05	1.573e+06	1.935e+06

Figure 7.7-3 Integrated Radiation Doses

7.7.3 Off-normal Conditions

In a worst-case scenario where both the inner and outer waste packages have been breached (an extremely unlikely situation), the radiation field near the breach could be much higher than the 38 rem/hr (or rad/hr) for an intact waste package. Preliminary estimates with the qualified MCNP4A code show that the breached waste package with loss or rupture of the inner and outer barriers would result in a maximum radiation level on the order of 10,000 rad/hr (Reference 5.13, p. IV-4) for the DBF in the vicinity of the waste packages. This radiation level is based on the condition of the breached waste package for which the inner and outer barriers have been removed; and the source radioactivity in spent fuel assemblies has not been redistributed. In this scenario, radiation sensors on the Remote Inspection Gantry used for performance confirmation data acquisition (Ref. 5.13) would provide adequate warning of increased radiation levels to the operator. Appropriate measures such as retrieval may be taken to mitigate the effects of a breached waste package. The retrieval consideration is beyond the scope of this analysis. A separate radiation analysis will be performed to support the retrieval operations.

8. CONCLUSIONS

This design analysis has met the objectives outlined in Section 1 by performing shielding evaluations in accord with the criteria specified in Section 4.2, and providing the preliminary radiation field data to support emplacement operations and radiation instrumentation specification. The evaluations lead to the following conclusions:

- 8.1 The radiation source terms presented in Sections 7.1 and 7.2 for the design basis fuel (PWR fuel assemblies with 4.2% initial enrichment, 48086 MWd/MTU burnup and 10 years decay), based on the CDB, are acceptable and conservative for preliminary design use. This conclusion is based on comparison with previous license applications submitted to the NRC for spent fuel storage and transportation casks, and an independent calculation with the SCALE code package.

Uniform fuel burnup is assumed in the current analysis. It is recommended that future analyses be refined by considering the axial variation in burnup for the active fuel source terms. This refinement would help reduce the axial dose rates from the active fuel region, and provide a better estimate of the radial dose rates without applying the peaking factor to the uniform burnup case. The latest version of SCALE, SCALE4.3 (Ref. 5.27), is proposed to generate the axial dependent source terms for use in refined radiation analysis.

The CDB Co-60 source terms provided in Section 7.2 for the end fittings are conservative to use for preliminary analysis. Re-evaluation of the end fitting source data is recommended to better quantify the Co-60 contributions, as the MGDS design progresses.

- 8.2 Radiation hazards for emplacement personnel are evaluated in Section 7.3. The emplaced waste packages, each containing 21 PWR fuel assemblies, give a maximum in-drift radiation field of about 40 rem/hr for the design basis fuel, which is prohibitive for human entry to emplacement drifts. Without any additional shielding, the radiation field in the main drift is about 64 mrem/hr, exceeding the limit of 2.5 mrem/hr required for normal occupational access (40 hr/wk or 2000 hr/yr). The high dose rates present a radiation hazard to emplacement personnel. Shielding must be provided to reduce personnel radiation exposures to meet the criterion stated in Section 4.2.
- 8.3 As demonstrated in Section 7.4, use of a concrete shadow shield installed between the last-emplaced waste package and drift entrance door is effective in reducing the radiation field to 0.8 mrem/hr (for DBF) at the nearest edge of the main drift when the door is closed. Since this dose rate is less than the 2.5 mrem/hr limit specified in Section 4.2.1, the main drift can be classified as "normally occupied areas" for 40 hr/wk (or 2000 hr/yr) occupational access. However, when the door is opened, the main-drift radiation field increases to 3.5 mrem/hr (for DBF), requiring some administrative control to ensure that personnel radiation exposures are maintained ALARA.

The evaluation in Section 7.4 considers the last-emplaced waste package only, and focuses on the material and thickness of the shadow shields. Additional shielding analysis may be necessary to address the effect of other waste packages in the same emplacement drift, and to establish the width and height of the shadow shields as part of the design specification of the shadow shields.

- 8.4 Shielding evaluation of the waste package transporter in Section 7.5 indicates that a near-optimized cost-effective shielding configuration is achieved with appropriate shielding materials (carbon steel for gamma shielding and 1.5% B-poly for neutron shielding), and an optimum neutron-to-gamma dose ratio. This shielding configuration results in a transporter surface radiation field of 33 mrem/hr on the sidewall, and 38.5 mrem/hr on the ends for the DBF. The shield design meets the limit of 50 mrem/hr specified in Assumption 4.3.2.7. The bottom of the transporter requires less shielding than the sidewall or the ends, as access to the bottom surface is normally not required. Table 7.5-3 summarizes the transporter shielding requirements and corresponding surface dose rates.

The transporter external dose rate exceeds the limit of 2.5 mrem/hr for 40 hr/wk access in normally occupied areas. Therefore, the vicinity of the transporter, when loaded, will be required to be controlled as an "intermittently occupied area" with radiation protection measures to control personnel exposures.

Additional shielding analysis will be required to optimize the full configuration of the transporter. Other candidate shielding materials (particularly for neutron shielding) for the transporter need to be evaluated to optimize both weight and size of the transporter, and to address the organic nature and flammability issue associated with the neutron shield material. Furthermore, a detailed shielding analysis needs to be performed to confirm the bottom shielding thicknesses estimated in Table 7.5-3.

- 8.5 Section 7.6 describes the limited ALARA evaluation. To meet the ALARA goal of 1 rem/yr or an average of 0.5 mrem/hr for 40 hr/wk (or 2000 hr/yr) normal occupational access, consideration may be given to the use of average SNF characteristics for ALARA assessment, since radiation exposures should be averaged out to correspond to the average fuel characteristics on a yearly basis. The difference in the radiation fields between the DBF and average SNF is about a factor of 5 as presented in Section 7.6. This factor provides the necessary margin to meet the ALARA goal (i.e., from the 10CFR20 limit of 5 rem/hr to the ALARA goal of 1 rem/yr). Shielding design based on the DBF radiation source terms and 10CFR20 dose limit can achieve the ALARA goal, when the occupational dose assessment is performed using the average SNF characteristics.

For occupational personnel access in the main drift and turnout areas, the ALARA objective can be achieved by placing waste packages with average SNF characteristics at both east and west ends of each emplacement drift.

- 8.6** For normal conditions, the in-drift radiation field on the WP surface is about 40 rem/hr (for DBF), dropping to less than 1 rem/hr at 90 years after emplacement, as presented in Section 7.7. As a result, the materials used in the emplacement drifts will receive most of radiation doses in the early preclosure period of time. Over the preclosure maintainable service life of 150 years, the cumulative dose reaches 40% at 10 years after emplacement, and 84% at 50 years after emplacement. Accordingly, performance confirmation with respect to radiation effects on the in-drift materials should focus on the first 50 years after initiation of waste emplacement.

For an off-normal condition, a breached waste package with loss or rupture of the inner and outer barriers (a worst-case scenario) would result in a maximum radiation level on the order of about 10,000 rem/hr near the breach, based on the DBF. Retrieval of the breached waste package would require significant additional personnel shielding for off-normal conditions. This requirement will be evaluated in a separate radiation analysis related to retrieval operations.

As identified in Section 2, much of the input data used in this analysis are preliminary and unqualified. Therefore, the outputs are also unqualified. Use of any data or outputs from this analysis for input into documents supporting construction, fabrication, or procurement is required to be controlled as TBV in accordance with NLP-3-15.

9. ATTACHMENTS

There are nine attachments to this design analysis document as described below:

Attachment I	Time-dependent Gamma Source Data (17 pages)
Attachment II	Homogenized Material Data (8 pages)
Attachment III	Conversion of Gamma Source Data (18 pages)
Attachment IV	Integrated Radiation Dose Calculation (5 pages)
Attachment V	Input and Output Files for Radiation Hazard Analysis (31 pages)
Attachment VI	Input and Output Files for Concrete Shadow Shield Analysis (31 pages)
Attachment VII	Input and Output Files for WP Transporter Shielding Analysis (16 pages)
Attachment VIII	Input and Output Files for ALARA Evaluation (3 pages)
Attachment IX	Input and Output Files for Time-dependent Radiation Conditions (25 pages)

**ATTACHMENT I
TIME-DEPENDENT GAMMA SOURCE DATA**

Per Section 4.1.5, this attachment provides the time-dependent gamma source data as a function time of time for the 150-year maintainable preclosure service life after initial waste emplacement. The initial emplacement corresponds to a decay time of 10 years as defined for the design basis fuel. The end of the maintainable preclosure service life corresponds to a decay time of 160 years. This attachment includes the decay times of 15, 20, 30, 40, 50, 60, 70, 80, 90, 100, 110, 120, 130, 140, 150 and 160 years. The 10-yr-decay data are given in Section 4.1.3 for the design basis fuel. The data in this attachment are for PWR fuel assemblies of 4.2% initial enrichment and 48086 MWd/MTU burnup. All the data are part of the computer output from Ref. 5.3.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 15 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	2.357E+15	26.09%
2.500E-02	4.771E+14	5.28%
3.750E-02	5.945E+14	6.58%
5.750E-02	4.758E+14	5.27%
8.500E-02	2.679E+14	2.97%
1.250E-01	2.437E+14	2.70%
2.250E-01	2.232E+14	2.47%
3.750E-01	9.597E+13	1.06%
5.750E-01	3.919E+15	43.38%
8.500E-01	1.614E+14	1.79%
1.250E+00	2.159E+14	2.39%
1.750E+00	3.274E+12	0.04%
2.250E+00	2.316E+09	0.00%
2.750E+00	1.073E+09	0.00%
3.500E+00	7.767E+07	0.00%
5.000E+00	2.403E+07	0.00%
7.000E+00	2.772E+06	0.00%
9.500E+00	3.183E+05	0.00%

TOTAL *9.035E+15 100.00%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 20 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	2.082E+15	26.80%
2.500E-02	4.140E+14	5.33%
3.750E-02	5.110E+14	6.58%
5.750E-02	4.286E+14	5.52%
8.500E-02	2.303E+14	2.96%
1.250E-01	1.960E+14	2.52%
2.250E-01	1.936E+14	2.49%
3.750E-01	8.080E+13	1.04%
5.750E-01	3.422E+15	44.05%
8.500E-01	8.227E+13	1.06%
1.250E+00	1.268E+14	1.63%
1.750E+00	2.264E+12	0.03%
2.250E+00	4.689E+08	0.00%
2.750E+00	8.947E+08	0.00%
3.500E+00	4.733E+07	0.00%
5.000E+00	1.994E+07	0.00%
7.000E+00	2.298E+06	0.00%
9.500E+00	2.641E+05	0.00%

TOTAL *7.770E+15 100.01%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 30 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	1.642E+15	27.31%
2.500E-02	3.230E+14	5.37%
3.750E-02	3.894E+14	6.48%
5.750E-02	3.534E+14	5.88%
8.500E-02	1.765E+14	2.94%
1.250E-01	1.349E+14	2.24%
2.250E-01	1.488E+14	2.47%
3.750E-01	6.183E+13	1.03%
5.750E-01	2.698E+15	44.87%
8.500E-01	3.650E+13	0.61%
1.250E+00	4.757E+13	0.79%
1.750E+00	1.120E+12	0.02%
2.250E+00	1.712E+08	0.00%
2.750E+00	8.075E+08	0.00%
3.500E+00	3.219E+07	0.00%
5.000E+00	1.376E+07	0.00%
7.000E+00	1.587E+06	0.00%
9.500E+00	1.822E+05	0.00%

TOTAL *6.013E+15 100.00%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 40 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	1.304E+15	27.55%
2.500E-02	2.547E+14	5.38%
3.750E-02	3.025E+14	6.39%
5.750E-02	2.947E+14	6.23%
8.500E-02	1.379E+14	2.91%
1.250E-01	9.848E+13	2.08%
2.250E-01	1.157E+14	2.44%
3.750E-01	4.834E+13	1.02%
5.750E-01	2.139E+15	45.20%
8.500E-01	1.936E+13	0.41%
1.250E+00	2.011E+13	0.42%
1.750E+00	5.978E+11	0.01%
2.250E+00	8.640E+07	0.00%
2.750E+00	7.274E+08	0.00%
3.500E+00	2.232E+07	0.00%
5.000E+00	9.571E+06	0.00%
7.000E+00	1.103E+06	0.00%
9.500E+00	1.267E+05	0.00%

TOTAL *4.735E+15 100.06%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 50 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	1.040E+15	27.70%
2.500E-02	2.017E+14	5.37%
3.750E-02	2.369E+14	6.31%
5.750E-02	2.481E+14	6.61%
8.500E-02	1.085E+14	2.89%
1.250E-01	7.395E+13	1.97%
2.250E-01	9.052E+13	2.41%
3.750E-01	3.802E+13	1.01%
5.750E-01	1.697E+15	45.21%
8.500E-01	1.141E+13	0.30%
1.250E+00	8.895E+12	0.24%
1.750E+00	3.289E+11	0.01%
2.250E+00	5.325E+07	0.00%
2.750E+00	6.552E+08	0.00%
3.500E+00	1.562E+07	0.00%
5.000E+00	6.676E+06	0.00%
7.000E+00	7.690E+05	0.00%
9.500E+00	8.828E+04	0.00%

TOTAL *3.755E+15 100.04%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 60 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	8.351E+14	27.89%
2.500E-02	1.603E+14	5.35%
3.750E-02	1.863E+14	6.22%
5.750E-02	2.112E+14	7.05%
8.500E-02	8.562E+13	2.86%
1.250E-01	5.665E+13	1.89%
2.250E-01	7.109E+13	2.37%
3.750E-01	2.992E+13	1.00%
5.750E-01	1.346E+15	44.96%
8.500E-01	7.194E+12	0.24%
1.250E+00	4.388E+12	0.15%
1.750E+00	2.018E+11	0.01%
2.250E+00	3.392E+07	0.00%
2.750E+00	5.918E+08	0.00%
3.500E+00	1.115E+07	0.00%
5.000E+00	4.765E+06	0.00%
7.000E+00	5.484E+05	0.00%
9.500E+00	6.295E+04	0.00%

TOTAL *2.994E+15 100.00%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 70 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	6.746E+14	28.09%
2.500E-02	1.277E+14	5.32%
3.750E-02	1.470E+14	6.12%
5.750E-02	1.820E+14	7.58%
8.500E-02	6.779E+13	2.82%
1.250E-01	4.406E+13	1.83%
2.250E-01	5.601E+13	2.33%
3.750E-01	2.359E+13	0.98%
5.750E-01	1.068E+15	44.48%
8.500E-01	4.895E+12	0.20%
1.250E+00	2.434E+12	0.10%
1.750E+00	1.335E+11	0.01%
2.250E+00	2.373E+07	0.00%
2.750E+00	5.352E+08	0.00%
3.500E+00	8.113E+06	0.00%
5.000E+00	3.465E+06	0.00%
7.000E+00	3.985E+05	0.00%
9.500E+00	4.573E+04	0.00%

TOTAL *2.398E+15 99.88%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 80 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	5.469E+14	28.31%
2.500E-02	1.020E+14	5.28%
3.750E-02	1.162E+14	6.01%
5.750E-02	1.583E+14	8.19%
8.500E-02	5.377E+13	2.78%
1.250E-01	3.464E+13	1.79%
2.250E-01	4.422E+13	2.29%
3.750E-01	1.862E+13	0.96%
5.750E-01	8.477E+14	43.88%
8.500E-01	3.536E+12	0.18%
1.250E+00	1.507E+12	0.08%
1.750E+00	9.346E+10	0.00%
2.250E+00	1.794E+07	0.00%
2.750E+00	4.844E+08	0.00%
3.500E+00	5.992E+06	0.00%
5.000E+00	2.557E+06	0.00%
7.000E+00	2.938E+05	0.00%
9.500E+00	3.371E+04	0.00%

TOTAL *1.927E+15 99.77%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 90 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	4.436E+14	28.52%
2.500E-02	8.135E+13	5.23%
3.750E-02	9.198E+13	5.91%
5.750E-02	1.387E+14	8.92%
8.500E-02	4.267E+13	2.74%
1.250E-01	2.740E+13	1.76%
2.250E-01	3.495E+13	2.25%
3.750E-01	1.470E+13	0.94%
5.750E-01	6.727E+14	43.24%
8.500E-01	2.659E+12	0.17%
1.250E+00	1.012E+12	0.07%
1.750E+00	6.791E+10	0.00%
2.250E+00	1.430E+07	0.00%
2.750E+00	4.387E+08	0.00%
3.500E+00	4.460E+06	0.00%
5.000E+00	1.902E+06	0.00%
7.000E+00	2.183E+05	0.00%
9.500E+00	2.503E+04	0.00%
TOTAL	*1.552E+15	99.76%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 100 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	3.589E+14	28.71%
2.500E-02	6.470E+13	5.18%
3.750E-02	7.285E+13	5.83%
5.750E-02	1.221E+14	9.77%
8.500E-02	3.381E+13	2.70%
1.250E-01	2.175E+13	1.74%
2.250E-01	2.761E+13	2.21%
3.750E-01	1.160E+13	0.93%
5.750E-01	5.338E+14	42.70%
8.500E-01	2.046E+12	0.16%
1.250E+00	7.125E+11	0.06%
1.750E+00	5.035E+10	0.00%
2.250E+00	1.178E+07	0.00%
2.750E+00	3.975E+08	0.00%
3.500E+00	3.313E+06	0.00%
5.000E+00	1.411E+06	0.00%
7.000E+00	1.617E+05	0.00%
9.500E+00	1.853E+04	0.00%

TOTAL *1.250E+15 99.99%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 110 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	2.972E+14	29.03%
2.500E-02	5.245E+13	5.12%
3.750E-02	5.797E+13	5.66%
5.750E-02	1.101E+14	10.75%
8.500E-02	2.716E+13	2.65%
1.250E-01	1.734E+13	1.69%
2.250E-01	2.204E+13	2.15%
3.750E-01	9.232E+12	0.90%
5.750E-01	4.237E+14	41.39%
8.500E-01	1.527E+12	0.15%
1.250E+00	5.061E+11	0.05%
1.750E+00	3.832E+10	0.00%
2.250E+00	9.206E+06	0.00%
2.750E+00	3.605E+08	0.00%
3.500E+00	2.639E+06	0.00%
5.000E+00	1.123E+06	0.00%
7.000E+00	1.285E+05	0.00%
9.500E+00	1.472E+04	0.00%

TOTAL *1.019E+15 99.56%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 120 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	2.484E+14	29.40%
2.500E-02	4.279E+13	5.06%
3.750E-02	4.621E+13	5.47%
5.750E-02	1.004E+14	11.88%
8.500E-02	2.193E+13	2.60%
1.250E-01	1.391E+13	1.65%
2.250E-01	1.765E+13	2.09%
3.750E-01	7.366E+12	0.87%
5.750E-01	3.364E+14	39.82%
8.500E-01	1.166E+12	0.14%
1.250E+00	3.743E+11	0.04%
1.750E+00	2.959E+10	0.00%
2.250E+00	7.392E+06	0.00%
2.750E+00	3.272E+08	0.00%
3.500E+00	2.178E+06	0.00%
5.000E+00	9.258E+05	0.00%
7.000E+00	1.058E+05	0.00%
9.500E+00	1.212E+04	0.00%
TOTAL	*8.366E+14	99.03%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 130 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	2.095E+14	29.83%
2.500E-02	3.512E+13	5.00%
3.750E-02	3.689E+13	5.25%
5.750E-02	9.245E+13	13.16%
8.500E-02	1.778E+13	2.53%
1.250E-01	1.123E+13	1.60%
2.250E-01	1.418E+13	2.02%
3.750E-01	5.888E+12	0.84%
5.750E-01	2.671E+14	38.03%
8.500E-01	9.117E+11	0.13%
1.250E+00	2.853E+11	0.04%
1.750E+00	2.306E+10	0.00%
2.250E+00	6.113E+06	0.00%
2.750E+00	2.970E+08	0.00%
3.500E+00	1.861E+06	0.00%
5.000E+00	7.904E+05	0.00%
7.000E+00	9.027E+04	0.00%
9.500E+00	1.033E+04	0.00%
TOTAL	*6.914E+14	98.44%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 140 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	1.782E+14	30.32%
2.500E-02	2.898E+13	4.93%
3.750E-02	2.948E+13	5.02%
5.750E-02	8.592E+13	14.62%
8.500E-02	1.447E+13	2.46%
1.250E-01	9.108E+12	1.55%
2.250E-01	1.142E+13	1.94%
3.750E-01	4.712E+12	0.80%
5.750E-01	2.121E+14	36.08%
8.500E-01	7.297E+11	0.12%
1.250E+00	2.219E+11	0.04%
1.750E+00	1.809E+10	0.00%
2.250E+00	5.209E+06	0.00%
2.750E+00	2.697E+08	0.00%
3.500E+00	1.642E+06	0.00%
5.000E+00	6.968E+05	0.00%
7.000E+00	7.952E+04	0.00%
9.500E+00	9.093E+03	0.00%

TOTAL *5.754E+14 97.88%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 150 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	1.528E+14	30.87%
2.500E-02	2.402E+13	4.85%
3.750E-02	2.356E+13	4.76%
5.750E-02	8.047E+13	16.26%
8.500E-02	1.181E+13	2.39%
1.250E-01	7.422E+12	1.50%
2.250E-01	9.202E+12	1.86%
3.750E-01	3.772E+12	0.76%
5.750E-01	1.684E+14	34.02%
8.500E-01	5.970E+11	0.12%
1.250E+00	1.750E+11	0.04%
1.750E+00	1.424E+10	0.00%
2.250E+00	4.571E+06	0.00%
2.750E+00	2.449E+08	0.00%
3.500E+00	1.490E+06	0.00%
5.000E+00	6.316E+05	0.00%
7.000E+00	7.203E+04	0.00%
9.500E+00	8.234E+03	0.00%

TOTAL *4.822E+14 97.41%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 160 YEARS

The data is shown in Photons per second/MTU

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	1.319E+14	31.47%
2.500E-02	1.999E+13	4.77%
3.750E-02	1.881E+13	4.49%
5.750E-02	7.586E+13	18.10%
8.500E-02	9.645E+12	2.30%
1.250E-01	6.067E+12	1.45%
2.250E-01	7.415E+12	1.77%
3.750E-01	3.016E+12	0.72%
5.750E-01	1.337E+14	31.90%
8.500E-01	4.982E+11	0.12%
1.250E+00	1.392E+11	0.03%
1.750E+00	1.124E+10	0.00%
2.250E+00	4.118E+06	0.00%
2.750E+00	2.225E+08	0.00%
3.500E+00	1.382E+06	0.00%
5.000E+00	5.857E+05	0.00%
7.000E+00	6.676E+04	0.00%
9.500E+00	7.630E+03	0.00%
TOTAL	*4.071E+14	97.13%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

ATTACHMENT II HOMOGENIZED MATERIAL DATA

This attachment provides the material data by element for the following homogenized regions used in the waste package model:

- Active fuel region (PWR)
- Fuel basket (liner)
- Upper end fittings (PWR)
- Lower end fittings (PWR)

The smeared density and elemental composition for each region are calculated with Lotus 1-2-3 Spreadsheets (Release 5), using the material data in Section 4.1.8 and Assumption 4.3.2.6. The calculation requires the input of density, volume fraction and elemental composition (in weight %) for each material in the region of interest.

II.1 CONVERSION OF FUEL COMPOSITION TO WEIGHT PERCENTS

The fuel composition in Assumption 4.3.2.6 is in units of atoms/b-cm, which is converted to weight % for use in obtaining the smeared material data in the fuel region as follows:

$$N_i = (D_i/M_i) \times 0.602252$$

$$W_i = (D_i/\Sigma D_i) \times 100$$

where N_i = atom density for element i in atoms/b-cm given in Assumption 4.3.2.6

D_i = partial density in g/cc for element i

M_i = atomic weight for element i

0.602252 = Avogadro's number $\times 10^{-24}$ (Ref. 5.9, p. 31)

An example calculation is provided below for U-238.

U-238 atom density $N_i = 6.3510\text{E-}3$ atoms/b-cm (Assumption 4.3.2.6)

U-238 atomic weight $M_i = 238.05077$ (Ref. 5.9, p. 30)

U-238 partial density $D_i = (N_i \times M_i)/0.602252 = 2.5103$ g/cc

The same exercise is repeated, using Lotus 1-2-3 spreadsheets for all elements in the fuel to obtain the total density of 3.5602 g/cc as given in Table II-1. The weight percent of U-238 in the fuel is then:

$$\text{U-238 weight \%} = 2.5103/3.5602 \times 100 = 70.51\%$$

II.2 CALCULATION OF HOMOGENIZED MATERIAL DATA

The following steps are used to obtain the homogenized material data for each region:

1. Obtain the material density and elemental composition for each material in the region from Sections 4.1.8 and II.1.
2. Obtain the volume fraction for each material from Assumption 4.3.2.6.
3. Calculate the total partial density for each element with the equation:

$$D_i = \sum \rho_j \times W_{ij} \times VF_j / 100 \quad (\text{Summation over all materials in a given region})$$

where D_i = partial density for element i, g/cc

ρ_j = density for material j, g/cc

VF_j = volume fraction for material j

W_{ij} = weight percent of element i in material j, %

Repeat this step for all elements in the region

4. Calculate the total density, D , of the homogenized region as follows:

$$D = \sum D_i \quad (\text{Summation over all elements in the region})$$

5. Calculate the weight percent, W_i , of element i in the region:

$$W_i = (D_i/D) \times 100$$

The above calculational procedure is performed with Lotus 1-2-3 spreadsheets for all elements in each region as shown in Tables II-2 through II-5. An example is given here for illustration purposes.

Example: Oxygen in Active Fuel Region

1. Oxygen weight % in fuel: 10.539% in fuel @ 3.5602 g/cc (Table II-1)
Oxygen weight % in air: 20% in air @ 0.001225 g/cc (Table 4.1-1)
2. VF of fuel in active fuel region: 0.8547 (Assumption 4.3.2.6)
VF of air in active fuel region: 0.1453 (Assumption 4.3.2.6)
3. D_i for oxygen = $(3.5602 \times 10.539 \times 0.8547 + 0.001225 \times 20 \times 0.1453) / 100 = 0.3207$ g/cc
4. Step 3 calculation is repeated for other elements in the active fuel region to obtain the total

smear density of 3.0431 g/cc as indicated in Table II-2.

5. Oxygen weight % in the homogenized active fuel region = $(0.3207/3.0431) \times 100 = 10.54\%$

TABLE II-1
CONVERSION OF FUEL COMPOSITION TO WEIGHT PERCENTS

Element	Fuel	Fuel	Fuel
	atoms/b-cm	g/cc	wt %
O	1.4127e-02	0.3752	10.539
Cr	7.1994e-06	0.0006	0.017
Fe	1.3406e-05	0.0012	0.035
Zr	4.0290e-03	0.6103	17.141
Sn	4.4155e-05	0.0087	0.244
U235	4.7717e-05	0.0186	0.523
U236	3.8027e-05	0.0149	0.419
U238	6.3510e-03	2.5103	70.512
Pu239	3.2181e-05	0.0128	0.359
Pu240	1.1558e-05	0.0046	0.129
Pu241	7.2740e-06	0.0029	0.082
Sum	2.4709e-02	3.5602	100.000

TABLE II-2
HOMOGENIZED DENSITY FOR ACTIVE FUEL REGION (PWR)

Homogenized Density for Active Fuel Region (PWR)						
Material	g/cc	VF				
Fuel	3.5602	0.8547				
Air	0.001225	0.1453				
Element	Fuel wt %	Air wt %	Total g/cc	Total wt %	Atomic Wt.	atoms/b-cm
N		80	0.0001	0.005	14.00307	6.1242e-06
O	10.539	20	0.3207	10.540	15.99492	1.2076e-02
Cr	0.017		0.0005	0.017	51.996	5.9916e-06
Fe	0.035		0.0011	0.035	55.847	1.1485e-05
Zr	17.141		0.5216	17.140	91.22	3.4436e-03
Sn	0.244		0.0074	0.244	118.69	3.7674e-05
U235	0.523		0.0159	0.523	235.04392	4.0777e-05
U236	0.419		0.0127	0.419	236.04564	3.2530e-05
U238	70.512		2.1456	70.508	238.0508	5.4282e-03
Pu239	0.359		0.0109	0.359	239.0521	2.7521e-05
Pu240	0.129		0.0039	0.129	240.0539	9.8480e-06
Pu241	0.082		0.0025	0.082	241.0567	6.2339e-06
Sum	100.000	100.000	3.0431	100.000		2.1126e-02

TABLE II-3
HOMOGENIZED DENSITY FOR FUEL BASKET (LINER)

Homogenized Density for PWR Fuel Basket (Liner)					
Material	g/cc	VF			
SS316B6A	7.745	0.3448			
A825	8.14	0.6552			
Element	SS316 wt%	A825 wt%	Air wt%	Total g/cc	Total wt%
B10	0.231			0.0062	0.077
B11	1.053			0.0281	0.351
C	0.03	0.05		0.0035	0.043
N	0.1			0.0027	0.033
Al		0.2		0.0107	0.133
Si	0.753	0.5		0.0468	0.584
P	0.045			0.0012	0.015
S	0.03	0.03		0.0024	0.030
Ti		0.9		0.0480	0.600
Cr	19.061	21.5		1.6557	20.686
Mn	2.007	1		0.1069	1.336
Fe	60.639	28.57		3.1431	39.270
Ni	13.543	42		2.6017	32.505
Cu		2.25		0.1200	1.499
Mo	2.508	3		0.2270	2.836
Sum	100.000	100.000	0.000	8.0038	100.000

TABLE II-4
HOMOGENIZED DENSITY FOR UPPER END FITTINGS (PWR)

Homogenized Density for PWR Upper End Fittings					
Material	g/cc	VF			
SS304	7.9	0.235			
Inc718	8.51	0.072			
Air	0.001225	0.693			
Element	SS304 wt%	Inc718 wt%	Air wt%	Total g/cc	Total wt%
C	0.08	0.04		0.0017	0.070
N	0.4		80	0.0081	0.328
O			20	0.0002	0.007
Al		0.5		0.0031	0.124
Si	1	0.18		0.0197	0.796
P	0.045			0.0008	0.034
S	0.03	0.008		0.0006	0.025
Ti		0.9		0.0055	0.223
Cr	19	19		0.4692	18.994
Mn	2	0.18		0.0382	1.548
Fe	67.245	18.5		1.3618	55.131
Co	0.2			0.0037	0.150
Ni	10	52.36		0.5065	20.504
Cu		0.15		0.0009	0.037
Nb		2.565		0.0157	0.636
Mo		3.05		0.0187	0.757
Ta		2.565		0.0157	0.636
Sum	100.000	99.998	100.000	2.4701	100.000

TABLE II-5
HOMOGENIZED DENSITY FOR LOWER END FITTINGS (PWR)

Homogenized Density for PWR Lower End Fittings					
Material	g/cc	VF			
SS304	7.9	0.2908			
Zirc-4	6.56	0.0369			
Air	0.001225	0.6723			
Element	SS304 wt%	Zirc-4 wt%	Air wt%	Total g/cc	Total wt%
C	0.08			0.0018	0.072
N	0.4		80	0.0098	0.388
O		0.12	20	0.0005	0.018
Si	1			0.0230	0.904
P	0.045			0.0010	0.041
S	0.03			0.0007	0.027
Cr	19	0.1		0.4367	17.193
Mn	2			0.0459	1.809
Fe	67.245	0.2		1.5453	60.834
Co	0.2			0.0046	0.181
Ni	10			0.2297	9.044
Zr		98.18		0.2377	9.356
Sn		1.4		0.0034	0.133
Sum	100.000	100.000	100.000	2.5402	100.0000

ATTACHMENT III CONVERSION OF GAMMA SOURCE DATA

The gamma source data in the active fuel region are converted to the forms in suitable units for input to the computer codes (MCNP4A and PATH). The MCNP4A code uses the data in units of photons/s per waste package (WP). The PATH code requires the source input in terms of photons/s-cc. The gamma source data in Sections 4.1.3, 4.1.5 and 4.1.6, and Attachment I are given in photons/s per MTU. These data are converted to other units in this Attachment, including photons/s per fuel assembly (FA), photons/s per WP, and photons/s-cc. The conversion uses the following data:

Heavy metal loadings = 0.464 MTU/FA	(Section 4.1.1)
No. of PWR fuel assemblies per WP = 21	(Assumption 4.3.2.1)
Volume of active fuel region in WP = $\pi(61)^2(360.18) \text{ cm}^3$ or $4.21 \times 10^6 \text{ cm}^3$	(Assumption 4.3.2.5)

The following pages provide the converted gamma source data from Lotus 1-2-3 spreadsheets, which are obtained as follows:

$$\begin{aligned}\text{Photons/s-FA} &= \text{Photons/s-MTU} \times 0.464 \text{ MTU/FA} \\ \text{Photons/s-WP} &= \text{Photons/s-FA} \times 21 \text{ FAs/WP} \\ \text{Photons/s-cc} &= (\text{Photons/s-WP}) / (4.21 \times 10^6 \text{ cc/WP})\end{aligned}$$

Example

$$\begin{aligned}\text{Page III-2 for 4.2\% enrichment, 48086 MWd/MTU, and 10 years decay} \\ \text{Gamma energy} &= 1.25 \text{ MeV} \\ \text{Photons/s-FA} &= (3.849\text{E}+14)(0.464) = 1.786\text{E}+14 \\ \text{Photons/s-WP} &= (1.786\text{E}+14)(21) = 3.750\text{E}+15 \\ \text{Photons/s-cc} &= (3.750\text{E}+15) / (4.21 \times 10^6) = 8.908\text{E}+08\end{aligned}$$

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 10 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	2.728e+15	24.27%	1.266e+15	2.658e+16	6.314e+09
2.500e-02	5.787e+14	5.15%	2.685e+14	5.639e+15	1.339e+09
3.750e-02	7.125e+14	6.34%	3.306e+14	6.943e+15	1.649e+09
5.750e-02	5.396e+14	4.80%	2.504e+14	5.258e+15	1.249e+09
8.500e-02	3.229e+14	2.87%	1.498e+14	3.146e+15	7.473e+08
1.250e-01	3.145e+14	2.80%	1.459e+14	3.064e+15	7.279e+08
2.250e-01	2.640e+14	2.35%	1.225e+14	2.572e+15	6.110e+08
3.750e-01	1.257e+14	1.12%	5.832e+13	1.225e+15	2.909e+08
5.750e-01	4.798e+15	42.69%	2.226e+15	4.675e+16	1.110e+10
8.500e-01	4.671e+14	4.16%	2.167e+14	4.551e+15	1.081e+09
1.250e+00	3.849e+14	3.42%	1.786e+14	3.750e+15	8.908e+08
1.750e+00	4.879e+12	0.04%	2.264e+12	4.754e+13	1.129e+07
2.250e+00	7.693e+10	0.00%	3.570e+10	7.496e+11	1.781e+05
2.750e+00	5.940e+09	0.00%	2.756e+09	5.788e+10	1.375e+04
3.500e+00	7.264e+08	0.00%	3.370e+08	7.078e+09	1.681e+03
5.000e+00	2.900e+07	0.00%	1.346e+07	2.826e+08	6.712e+01
7.000e+00	3.343e+06	0.00%	1.551e+06	3.257e+07	7.737e+00
9.500e+00	3.841e+05	0.00%	1.782e+05	3.743e+06	8.890e-01

TOTAL *1.124E+16

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 15 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	2.357e+15	26.09%	1.094e+15	2.297e+16	5.455e+09
2.500e-02	4.771e+14	5.28%	2.214e+14	4.649e+15	1.104e+09
3.750e-02	5.945e+14	6.58%	2.758e+14	5.793e+15	1.376e+09
5.750e-02	4.758e+14	5.27%	2.208e+14	4.636e+15	1.101e+09
8.500e-02	2.679e+14	2.97%	1.243e+14	2.610e+15	6.201e+08
1.250e-01	2.437e+14	2.70%	1.131e+14	2.375e+15	5.640e+08
2.250e-01	2.232e+14	2.47%	1.036e+14	2.175e+15	5.166e+08
3.750e-01	9.597e+13	1.06%	4.453e+13	9.351e+14	2.221e+08
5.750e-01	3.919e+15	43.38%	1.818e+15	3.819e+16	9.070e+09
8.500e-01	1.614e+14	1.79%	7.489e+13	1.573e+15	3.736e+08
1.250e+00	2.159e+14	2.39%	1.002e+14	2.104e+15	4.997e+08
1.750e+00	3.274e+12	0.04%	1.519e+12	3.190e+13	7.578e+06
2.250e+00	2.316e+09	0.00%	1.075e+09	2.257e+10	5.360e+03
2.750e+00	1.073e+09	0.00%	4.979e+08	1.046e+10	2.483e+03
3.500e+00	7.767e+07	0.00%	3.604e+07	7.568e+08	1.798e+02
5.000e+00	2.403e+07	0.00%	1.115e+07	2.341e+08	5.562e+01
7.000e+00	2.772e+06	0.00%	1.286e+06	2.701e+07	6.416e+00
9.500e+00	3.183e+05	0.00%	1.477e+05	3.102e+06	7.367e-01

TOTAL *9.035E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 20 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	2.082e+15	26.80%	9.660e+14	2.029e+16	4.819e+09
2.500e-02	4.140e+14	5.33%	1.921e+14	4.034e+15	9.582e+08
3.750e-02	5.110e+14	6.58%	2.371e+14	4.979e+15	1.183e+09
5.750e-02	4.286e+14	5.52%	1.989e+14	4.176e+15	9.920e+08
8.500e-02	2.303e+14	2.96%	1.069e+14	2.244e+15	5.330e+08
1.250e-01	1.960e+14	2.52%	9.094e+13	1.910e+15	4.536e+08
2.250e-01	1.936e+14	2.49%	8.983e+13	1.886e+15	4.481e+08
3.750e-01	8.080e+13	1.04%	3.749e+13	7.873e+14	1.870e+08
5.750e-01	3.422e+15	44.05%	1.588e+15	3.334e+16	7.920e+09
8.500e-01	8.227e+13	1.63%	5.884e+13	1.236e+15	2.935e+08
1.750e+00	2.264e+12	0.03%	1.050e+12	2.206e+13	5.240e+06
2.250e+00	4.689e+08	0.00%	2.176e+08	4.569e+09	1.085e+03
2.750e+00	8.947e+08	0.00%	4.151e+08	8.718e+09	2.071e+03
3.500e+00	4.733e+07	0.00%	2.196e+07	4.612e+08	1.095e+02
5.000e+00	1.994e+07	0.00%	9.252e+06	1.943e+08	4.615e+01
7.000e+00	2.298e+06	0.00%	1.066e+06	2.239e+07	5.319e+00
9.500e+00	2.641e+05	0.00%	1.225e+05	2.573e+06	6.113e-01

TOTAL *7.770E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 30 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	1.642e+15	27.31%	7.619e+14	1.600e+16	3.800e+09
2.500e-02	3.230e+14	5.37%	1.499e+14	3.147e+15	7.476e+08
3.750e-02	3.894e+14	6.48%	1.807e+14	3.794e+15	9.013e+08
5.750e-02	3.534e+14	5.88%	1.640e+14	3.444e+15	8.179e+08
8.500e-02	1.765e+14	2.94%	8.190e+13	1.720e+15	4.085e+08
1.250e-01	1.349e+14	2.24%	6.259e+13	1.314e+15	3.122e+08
2.250e-01	1.488e+14	2.47%	6.904e+13	1.450e+15	3.444e+08
3.750e-01	6.183e+13	1.03%	2.869e+13	6.025e+14	1.431e+08
5.750e-01	2.698e+15	44.87%	1.252e+15	2.629e+16	6.244e+09
8.500e-01	3.650e+13	0.61%	1.694e+13	3.557e+14	8.448e+07
1.250e+00	4.757e+13	0.79%	2.207e+13	4.635e+14	1.101e+08
1.750e+00	1.120e+12	0.02%	5.197e+11	1.091e+13	2.592e+06
2.250e+00	1.712e+08	0.00%	7.944e+07	1.668e+09	3.962e+02
2.750e+00	8.075e+08	0.00%	3.747e+08	7.868e+09	1.869e+03
3.500e+00	3.219e+07	0.00%	1.494e+07	3.137e+08	7.450e+01
5.000e+00	1.376e+07	0.00%	6.385e+06	1.341e+08	3.185e+01
7.000e+00	1.587e+06	0.00%	7.364e+05	1.546e+07	3.673e+00
9.500e+00	1.822e+05	0.00%	8.454e+04	1.775e+06	4.217e-01

TOTAL *6.013E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 40 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	1.304e+15	27.55%	6.051e+14	1.271e+16	3.018e+09
2.500e-02	2.547e+14	5.38%	1.182e+14	2.482e+15	5.895e+08
3.750e-02	3.025e+14	6.39%	1.404e+14	2.948e+15	7.001e+08
5.750e-02	2.947e+14	6.23%	1.367e+14	2.872e+15	6.821e+08
8.500e-02	1.379e+14	2.91%	6.399e+13	1.344e+15	3.192e+08
1.250e-01	9.848e+13	2.08%	4.569e+13	9.596e+14	2.279e+08
2.250e-01	1.157e+14	2.44%	5.368e+13	1.127e+15	2.678e+08
3.750e-01	4.834e+13	1.02%	2.243e+13	4.710e+14	1.119e+08
5.750e-01	2.139e+15	45.20%	9.925e+14	2.084e+16	4.951e+09
8.500e-01	1.936e+13	0.41%	8.983e+12	1.886e+14	4.481e+07
1.250e+00	2.011e+13	0.42%	9.331e+12	1.960e+14	4.654e+07
1.750e+00	5.978e+11	0.01%	2.774e+11	5.825e+12	1.384e+06
2.250e+00	8.640e+07	0.00%	4.009e+07	8.419e+08	2.000e+02
2.750e+00	7.274e+08	0.00%	3.375e+08	7.088e+09	1.684e+03
3.500e+00	2.232e+07	0.00%	1.036e+07	2.175e+08	5.166e+01
5.000e+00	9.571e+06	0.00%	4.441e+06	9.326e+07	2.215e+01
7.000e+00	1.103e+06	0.00%	5.118e+05	1.075e+07	2.553e+00
9.500e+00	1.267e+05	0.00%	5.879e+04	1.235e+06	2.932e-01

TOTAL *4.735E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 50 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	1.040e+15	27.70%	4.826e+14	1.013e+16	2.407e+09
2.500e-02	2.017e+14	5.37%	9.359e+13	1.965e+15	4.668e+08
3.750e-02	2.369e+14	6.31%	1.099e+14	2.308e+15	5.483e+08
5.750e-02	2.481e+14	6.61%	1.151e+14	2.417e+15	5.742e+08
8.500e-02	1.085e+14	2.89%	5.034e+13	1.057e+15	2.511e+08
1.250e-01	7.395e+13	1.97%	3.431e+13	7.206e+14	1.712e+08
2.250e-01	9.052e+13	2.41%	4.200e+13	8.820e+14	2.095e+08
3.750e-01	3.802e+13	1.01%	1.764e+13	3.705e+14	8.800e+07
5.750e-01	1.697e+15	45.21%	7.874e+14	1.654e+16	3.928e+09
8.500e-01	1.141e+13	0.30%	5.294e+12	1.112e+14	2.641e+07
1.250e+00	8.895e+12	0.24%	4.127e+12	8.667e+13	2.059e+07
1.750e+00	3.289e+11	0.01%	1.526e+11	3.205e+12	7.612e+05
2.250e+00	5.325e+07	0.00%	2.471e+07	5.189e+08	1.232e+02
2.750e+00	6.552e+08	0.00%	3.040e+08	6.384e+09	1.516e+03
3.500e+00	1.562e+07	0.00%	7.248e+06	1.522e+08	3.615e+01
5.000e+00	6.676e+06	0.00%	3.098e+06	6.505e+07	1.545e+01
7.000e+00	7.690e+05	0.00%	3.568e+05	7.493e+06	1.780e+00
9.500e+00	8.828e+04	0.00%	4.096e+04	8.602e+05	2.043e-01

TOTAL *3.755E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 60 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	8.351e+14	27.89%	3.875e+14	8.137e+15	1.933e+09
2.500e-02	1.603e+14	5.35%	7.438e+13	1.562e+15	3.710e+08
3.750e-02	1.863e+14	6.22%	8.644e+13	1.815e+15	4.312e+08
5.750e-02	2.112e+14	7.05%	9.800e+13	2.058e+15	4.888e+08
8.500e-02	8.562e+13	2.86%	3.973e+13	8.343e+14	1.982e+08
1.250e-01	5.665e+13	1.89%	2.629e+13	5.520e+14	1.311e+08
2.250e-01	7.109e+13	2.37%	3.299e+13	6.927e+14	1.645e+08
3.750e-01	2.992e+13	1.00%	1.388e+13	2.915e+14	6.925e+07
5.750e-01	1.346e+15	44.96%	6.245e+14	1.312e+16	3.115e+09
8.500e-01	7.194e+12	0.24%	3.338e+12	7.010e+13	1.665e+07
1.250e+00	4.388e+12	0.15%	2.036e+12	4.276e+13	1.016e+07
1.750e+00	2.018e+11	0.01%	9.364e+10	1.966e+12	4.671e+05
2.250e+00	3.392e+07	0.00%	1.574e+07	3.305e+08	7.851e+01
2.750e+00	5.918e+08	0.00%	2.746e+08	5.766e+09	1.370e+03
3.500e+00	1.115e+07	0.00%	5.174e+06	1.086e+08	2.581e+01
5.000e+00	4.765e+06	0.00%	2.211e+06	4.643e+07	1.103e+01
7.000e+00	5.484e+05	0.00%	2.545e+05	5.344e+06	1.269e+00
9.500e+00	6.295e+04	0.00%	2.921e+04	6.134e+05	1.457e-01

TOTAL *2.994E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 70 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	6.746e+14	28.09%	3.130e+14	6.573e+15	1.561e+09
2.500e-02	1.277e+14	5.32%	5.925e+13	1.244e+15	2.956e+08
3.750e-02	1.470e+14	6.12%	6.821e+13	1.432e+15	3.402e+08
5.750e-02	1.820e+14	7.58%	8.445e+13	1.773e+15	4.212e+08
8.500e-02	6.779e+13	2.82%	3.145e+13	6.605e+14	1.569e+08
1.250e-01	4.406e+13	1.83%	2.044e+13	4.293e+14	1.020e+08
2.250e-01	5.601e+13	2.33%	2.599e+13	5.458e+14	1.296e+08
3.750e-01	2.359e+13	0.98%	1.095e+13	2.299e+14	5.460e+07
5.750e-01	1.068e+15	44.48%	4.956e+14	1.041e+16	2.472e+09
8.500e-01	4.895e+12	0.20%	2.271e+12	4.770e+13	1.133e+07
1.250e+00	2.434e+12	0.10%	1.129e+12	2.372e+13	5.633e+06
1.750e+00	1.335e+11	0.01%	6.194e+10	1.301e+12	3.090e+05
2.250e+00	2.373e+07	0.00%	1.101e+07	2.312e+08	5.492e+01
2.750e+00	5.352e+08	0.00%	2.483e+08	5.215e+09	1.239e+03
3.500e+00	8.113e+06	0.00%	3.764e+06	7.905e+07	1.878e+01
5.000e+00	3.465e+06	0.00%	1.608e+06	3.376e+07	8.020e+00
7.000e+00	3.985e+05	0.00%	1.849e+05	3.883e+06	9.223e-01
9.500e+00	4.573e+04	0.00%	2.122e+04	4.456e+05	1.058e-01

TOTAL *2.398E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 80 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	5.469e+14	28.31%	2.538e+14	5.329e+15	1.266e+09
2.500e-02	1.020e+14	5.28%	4.733e+13	9.939e+14	2.361e+08
3.750e-02	1.162e+14	6.01%	5.392e+13	1.132e+15	2.689e+08
5.750e-02	1.583e+14	8.19%	7.345e+13	1.542e+15	3.664e+08
8.500e-02	5.377e+13	2.78%	2.495e+13	5.239e+14	1.245e+08
1.250e-01	3.464e+13	1.79%	1.607e+13	3.375e+14	8.017e+07
2.250e-01	4.422e+13	2.29%	2.052e+13	4.309e+14	1.023e+08
3.750e-01	1.862e+13	0.96%	8.640e+12	1.814e+14	4.310e+07
5.750e-01	8.477e+14	43.88%	3.933e+14	8.260e+15	1.962e+09
8.500e-01	3.536e+12	0.18%	1.641e+12	3.445e+13	8.184e+06
1.250e+00	1.507e+12	0.08%	6.992e+11	1.468e+13	3.488e+06
1.750e+00	9.346e+10	0.00%	4.337e+10	9.107e+11	2.163e+05
2.250e+00	1.794e+07	0.00%	8.324e+06	1.748e+08	4.152e+01
2.750e+00	4.844e+08	0.00%	2.248e+08	4.720e+09	1.121e+03
3.500e+00	5.992e+06	0.00%	2.780e+06	5.839e+07	1.387e+01
5.000e+00	2.557e+06	0.00%	1.186e+06	2.492e+07	5.918e+00
7.000e+00	2.938e+05	0.00%	1.363e+05	2.863e+06	6.800e-01
9.500e+00	3.371e+04	0.00%	1.564e+04	3.285e+05	7.802e-02

TOTAL *1.927E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 90 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	4.436e+14	28.52%	2.058e+14	4.322e+15	1.027e+09
2.500e-02	8.135e+13	5.23%	3.775e+13	7.927e+14	1.883e+08
3.750e-02	9.198e+13	5.91%	4.268e+13	8.963e+14	2.129e+08
5.750e-02	1.387e+14	8.92%	6.436e+13	1.351e+15	3.210e+08
8.500e-02	4.267e+13	2.74%	1.980e+13	4.158e+14	9.876e+07
1.250e-01	2.740e+13	1.76%	1.271e+13	2.670e+14	6.342e+07
2.250e-01	3.495e+13	2.25%	1.622e+13	3.406e+14	8.089e+07
3.750e-01	1.470e+13	0.94%	6.821e+12	1.432e+14	3.402e+07
5.750e-01	6.727e+14	43.24%	3.121e+14	6.555e+15	1.557e+09
8.500e-01	2.659e+12	0.17%	1.234e+12	2.591e+13	6.154e+06
1.250e+00	1.012e+12	0.07%	4.696e+11	9.861e+12	2.342e+06
1.750e+00	6.791e+10	0.00%	3.151e+10	6.617e+11	1.572e+05
2.250e+00	1.430e+07	0.00%	6.635e+06	1.393e+08	3.310e+01
2.750e+00	4.387e+08	0.00%	2.036e+08	4.275e+09	1.015e+03
3.500e+00	4.460e+06	0.00%	2.069e+06	4.346e+07	1.032e+01
5.000e+00	1.902e+06	0.00%	8.825e+05	1.853e+07	4.402e+00
7.000e+00	2.183e+05	0.00%	1.013e+05	2.127e+06	5.053e-01
9.500e+00	2.503e+04	0.00%	1.161e+04	2.439e+05	5.793e-02

TOTAL *1.552E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 100 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	3.589e+14	28.71%	1.665e+14	3.497e+15	8.307e+08
2.500e-02	6.470e+13	5.18%	3.002e+13	6.304e+14	1.497e+08
3.750e-02	7.285e+13	5.83%	3.380e+13	7.099e+14	1.686e+08
5.750e-02	1.221e+14	9.77%	5.665e+13	1.190e+15	2.826e+08
8.500e-02	3.381e+13	2.70%	1.569e+13	3.294e+14	7.825e+07
1.250e-01	2.175e+13	1.74%	1.009e+13	2.119e+14	5.034e+07
2.250e-01	2.761e+13	2.21%	1.281e+13	2.690e+14	6.390e+07
3.750e-01	1.160e+13	0.93%	5.382e+12	1.130e+14	2.685e+07
5.750e-01	5.338e+14	42.70%	2.477e+14	5.201e+15	1.235e+09
8.500e-01	2.046e+12	0.16%	9.493e+11	1.994e+13	4.735e+06
1.250e+00	7.125e+11	0.06%	3.306e+11	6.943e+12	1.649e+06
1.750e+00	5.035e+10	0.00%	2.336e+10	4.906e+11	1.165e+05
2.250e+00	1.178e+07	0.00%	5.466e+06	1.148e+08	2.726e+01
2.750e+00	3.975e+08	0.00%	1.844e+08	3.873e+09	9.200e+02
3.500e+00	3.313e+06	0.00%	1.537e+06	3.228e+07	7.668e+00
5.000e+00	1.411e+06	0.00%	6.547e+05	1.375e+07	3.266e+00
7.000e+00	1.617e+05	0.00%	7.503e+04	1.576e+06	3.743e-01
9.500e+00	1.853e+04	0.00%	8.598e+03	1.806e+05	4.289e-02

TOTAL *1.250E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 110 YEARS

ENERGY (MeV)	Pho/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	2.972e+14	29.03%	1.379e+14	2.896e+15	6.879e+08
2.500e-02	5.245e+13	5.12%	2.434e+13	5.111e+14	1.214e+08
3.750e-02	5.797e+13	5.66%	2.690e+13	5.649e+14	1.342e+08
5.750e-02	1.101e+14	10.75%	5.109e+13	1.073e+15	2.548e+08
8.500e-02	2.716e+13	2.65%	1.260e+13	2.646e+14	6.286e+07
1.250e-01	1.734e+13	1.69%	8.046e+12	1.690e+14	4.013e+07
2.250e-01	2.204e+13	2.15%	1.023e+13	2.148e+14	5.101e+07
3.750e-01	9.232e+12	0.90%	4.284e+12	8.996e+13	2.137e+07
5.750e-01	4.237e+14	41.39%	1.966e+14	4.129e+15	9.806e+08
8.500e-01	1.527e+12	0.15%	7.085e+11	1.488e+13	3.534e+06
1.250e+00	5.061e+11	0.05%	2.348e+11	4.931e+12	1.171e+06
1.750e+00	3.832e+10	0.00%	1.778e+10	3.734e+11	8.869e+04
2.250e+00	9.206e+06	0.00%	4.272e+06	8.970e+07	2.131e+01
2.750e+00	3.605e+08	0.00%	1.673e+08	3.513e+09	8.344e+02
3.500e+00	2.639e+06	0.00%	1.224e+06	2.571e+07	6.108e+00
5.000e+00	1.123e+06	0.00%	5.211e+05	1.094e+07	2.599e+00
7.000e+00	1.285e+05	0.00%	5.962e+04	1.252e+06	2.974e-01
9.500e+00	1.472e+04	0.00%	6.830e+03	1.434e+05	3.407e-02

TOTAL *1.019E+15

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 120 YEARS

ENERGY (MeV)	PHO/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	2.484e+14	29.40%	1.153e+14	2.420e+15	5.749e+08
2.500e-02	4.279e+13	5.06%	1.985e+13	4.169e+14	9.904e+07
3.750e-02	4.621e+13	5.47%	2.144e+13	4.503e+14	1.070e+08
5.750e-02	1.004e+14	11.88%	4.659e+13	9.783e+14	2.324e+08
8.500e-02	2.193e+13	2.60%	1.018e+13	2.137e+14	5.076e+07
1.250e-01	1.391e+13	1.65%	6.454e+12	1.355e+14	3.219e+07
2.250e-01	1.765e+13	2.09%	8.190e+12	1.720e+14	4.085e+07
3.750e-01	7.366e+12	0.87%	3.418e+12	7.177e+13	1.705e+07
5.750e-01	3.364e+14	39.82%	1.561e+14	3.278e+15	7.786e+08
8.500e-01	1.166e+12	0.14%	5.410e+11	1.136e+13	2.699e+06
1.250e+00	3.743e+11	0.04%	1.737e+11	3.647e+12	8.663e+05
1.750e+00	2.959e+10	0.00%	1.373e+10	2.883e+11	6.849e+04
2.250e+00	7.392e+06	0.00%	3.430e+06	7.203e+07	1.711e+01
2.750e+00	3.272e+08	0.00%	1.518e+08	3.188e+09	7.573e+02
3.500e+00	2.178e+06	0.00%	1.011e+06	2.122e+07	5.041e+00
5.000e+00	9.258e+05	0.00%	4.296e+05	9.021e+06	2.143e+00
7.000e+00	1.058e+05	0.00%	4.909e+04	1.031e+06	2.449e-01
9.500e+00	1.212e+04	0.00%	5.624e+03	1.181e+05	2.805e-02

TOTAL *8.366E+14

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 130 YEARS

ENERGY (MeV)	PHO/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	2.095e+14	29.83%	9.721e+13	2.041e+15	4.849e+08
2.500e-02	3.512e+13	5.00%	1.630e+13	3.422e+14	8.128e+07
3.750e-02	3.689e+13	5.25%	1.712e+13	3.595e+14	8.538e+07
5.750e-02	9.245e+13	13.16%	4.290e+13	9.008e+14	2.140e+08
8.500e-02	1.778e+13	2.53%	8.250e+12	1.732e+14	4.115e+07
1.250e-01	1.123e+13	1.60%	5.211e+12	1.094e+14	2.599e+07
2.250e-01	1.418e+13	2.02%	6.580e+12	1.382e+14	3.282e+07
3.750e-01	5.888e+12	0.84%	2.732e+12	5.737e+13	1.363e+07
5.750e-01	2.671e+14	38.03%	1.239e+14	2.603e+15	6.182e+08
8.500e-01	9.117e+11	0.13%	4.230e+11	8.884e+12	2.110e+06
1.250e+00	2.853e+11	0.04%	1.324e+11	2.780e+12	6.603e+05
1.750e+00	2.306e+10	0.00%	1.070e+10	2.247e+11	5.337e+04
2.250e+00	6.113e+06	0.00%	2.836e+06	5.957e+07	1.415e+01
2.750e+00	2.970e+08	0.00%	1.378e+08	2.894e+09	6.874e+02
3.500e+00	1.861e+06	0.00%	8.635e+05	1.813e+07	4.307e+00
5.000e+00	7.904e+05	0.00%	3.667e+05	7.702e+06	1.829e+00
7.000e+00	9.027e+04	0.00%	4.189e+04	8.796e+05	2.089e-01
9.500e+00	1.033e+04	0.00%	4.793e+03	1.007e+05	2.391e-02

TOTAL *6.914E+14

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 140 YEARS

ENERGY (MeV)	PHO/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	1.782e+14	30.32%	8.268e+13	1.736e+15	4.124e+08
2.500e-02	2.898e+13	4.93%	1.345e+13	2.824e+14	6.707e+07
3.750e-02	2.948e+13	5.02%	1.368e+13	2.873e+14	6.823e+07
5.750e-02	8.592e+13	14.62%	3.987e+13	8.372e+14	1.989e+08
8.500e-02	1.447e+13	2.46%	6.714e+12	1.410e+14	3.349e+07
1.250e-01	9.108e+12	1.55%	4.226e+12	8.875e+13	2.108e+07
2.250e-01	1.142e+13	1.94%	5.299e+12	1.113e+14	2.643e+07
3.750e-01	4.712e+12	0.80%	2.186e+12	4.591e+13	1.091e+07
5.750e-01	2.121e+14	36.08%	9.841e+13	2.067e+15	4.909e+08
8.500e-01	7.297e+11	0.12%	3.386e+11	7.110e+12	1.689e+06
1.250e+00	2.219e+11	0.04%	1.030e+11	2.162e+12	5.136e+05
1.750e+00	1.809e+10	0.00%	8.394e+09	1.763e+11	4.187e+04
2.250e+00	5.209e+06	0.00%	2.417e+06	5.076e+07	1.206e+01
2.750e+00	2.697e+08	0.00%	1.251e+08	2.628e+09	6.242e+02
3.500e+00	1.642e+06	0.00%	7.619e+05	1.600e+07	3.800e+00
5.000e+00	6.968e+05	0.00%	3.233e+05	6.790e+06	1.613e+00
7.000e+00	7.952e+04	0.00%	3.690e+04	7.748e+05	1.840e-01
9.500e+00	9.093e+03	0.00%	4.219e+03	8.860e+04	2.105e-02

TOTAL *5.754E+14

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 150 YEARS

ENERGY (MeV)	PHO/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	1.528e+14	30.87%	7.090e+13	1.489e+15	3.537e+08
2.500e-02	2.402e+13	4.85%	1.115e+13	2.341e+14	5.559e+07
3.750e-02	2.356e+13	4.76%	1.093e+13	2.296e+14	5.453e+07
5.750e-02	8.047e+13	16.26%	3.734e+13	7.841e+14	1.862e+08
8.500e-02	1.181e+13	2.39%	5.480e+12	1.151e+14	2.733e+07
1.250e-01	7.422e+12	1.50%	3.444e+12	7.232e+13	1.718e+07
2.250e-01	9.202e+12	1.86%	4.270e+12	8.966e+13	2.130e+07
3.750e-01	3.772e+12	0.76%	1.750e+12	3.675e+13	8.730e+06
5.750e-01	1.684e+14	34.02%	7.814e+13	1.641e+15	3.898e+08
8.500e-01	5.970e+11	0.12%	2.770e+11	5.817e+12	1.382e+06
1.250e+00	1.750e+11	0.04%	8.120e+10	1.705e+12	4.050e+05
1.750e+00	1.424e+10	0.00%	6.607e+09	1.388e+11	3.296e+04
2.250e+00	4.571e+06	0.00%	2.121e+06	4.454e+07	1.058e+01
2.750e+00	2.449e+08	0.00%	1.136e+08	2.386e+09	5.668e+02
3.500e+00	1.490e+06	0.00%	6.914e+05	1.452e+07	3.449e+00
5.000e+00	6.316e+05	0.00%	2.931e+05	6.154e+06	1.462e+00
7.000e+00	7.203e+04	0.00%	3.342e+04	7.019e+05	1.667e-01
9.500e+00	8.234e+03	0.00%	3.821e+03	8.023e+04	1.906e-02

TOTAL *4.822E+14

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

LWR Radiological DATABASE

PHOTONS REPORT

REACTOR TYPE & BURNUP: PWR 48086

ENRICHMENT: 4.20%

DECAY TIME: 160 YEARS

ENERGY (MeV)	PHO/s-MTU	% TOTAL	Pho/s-FA	Pho/s-WP	Pho/s-cc
1.000e-02	1.319e+14	31.47%	6.120e+13	1.285e+15	3.053e+08
2.500e-02	1.999e+13	4.77%	9.275e+12	1.948e+14	4.627e+07
3.750e-02	1.881e+13	4.49%	8.728e+12	1.833e+14	4.354e+07
5.750e-02	7.586e+13	18.10%	3.520e+13	7.392e+14	1.756e+08
8.500e-02	9.645e+12	2.30%	4.475e+12	9.398e+13	2.232e+07
1.250e-01	6.067e+12	1.45%	2.815e+12	5.912e+13	1.404e+07
2.250e-01	7.415e+12	1.77%	3.441e+12	7.225e+13	1.716e+07
3.750e-01	3.016e+12	0.72%	1.399e+12	2.939e+13	6.980e+06
5.750e-01	1.337e+14	31.90%	6.204e+13	1.303e+15	3.094e+08
8.500e-01	4.982e+11	0.12%	2.312e+11	4.854e+12	1.153e+06
1.250e+00	1.392e+11	0.03%	6.459e+10	1.356e+12	3.222e+05
1.750e+00	1.124e+10	0.00%	5.215e+09	1.095e+11	2.601e+04
2.250e+00	4.118e+06	0.00%	1.911e+06	4.013e+07	9.531e+00
2.750e+00	2.225e+08	0.00%	1.032e+08	2.168e+09	5.150e+02
3.500e+00	1.382e+06	0.00%	6.412e+05	1.347e+07	3.199e+00
5.000e+00	5.857e+05	0.00%	2.718e+05	5.707e+06	1.356e+00
7.000e+00	6.676e+04	0.00%	3.098e+04	6.505e+05	1.545e-01
9.500e+00	7.630e+03	0.00%	3.540e+03	7.435e+04	1.766e-02

TOTAL *4.071E+14

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination.

ATTACHMENT IV INTEGRATED RADIATION DOSE CALCULATION

The time-integrated radiation doses at the waste package surface and concrete lining are calculated with Lotus 1-2-3 spreadsheets, using the equations in Section 7.7.2.2. All the symbols used here are defined in Section 7.7.2.2, and repeated here for convenience.

T1 = Time at the beginning of the time interval (yr)

T2 = Time at the end of the time interval (yr)

D1 = Dose rate at T1 (rem/hr)

D2 = Dose rate at T2 (rem/hr)

λ = Decay factor (1/yr) calculated with Equation 7.7-2

Dose = Integrated dose for time interval from T1 to T2 (rem) calculated with Equation 7.7-3

Cumulative Dose = sum of the doses as of T2

Separate calculations are performed for neutrons and gammas using the results in Table 7.7-3. Table IV-1 through IV-4 provides the calculated doses from the spreadsheets.

Example

Table IV-1

T1 = 15 yr

T2 = 20 yr

D1 = 2.322 rem/hr

D2 = 1.933 rem/hr

$\lambda = [1/(20-15)]\text{Log}_e(2.322/1.933) = 0.0367 \text{ yr}^{-1}$

Dose = $(2.322 * 8766 / 0.0367) [1 - e^{-0.0367 * (20-15)}] = 9.299\text{E}+04 \text{ rem}$

Cumulative dose = $(1.118\text{E}+05) + (9.299\text{E}+04) = 2.048\text{E}+05 \text{ rem}$

TABLE IV-1
NEUTRON DOSE AT WASTE PACKAGE SURFACE

T1 (yr)	T2 (yr)	D1 (rem/hr)	D2 (rem/hr)	Decay Factor (1/yr)	Dose (rem) T1 to T2	Cumulative Dose (rem)
10	15	2.793	2.322	0.0369	1.118e+05	1.118e+05
15	20	2.322	1.933	0.0367	9.299e+04	2.048e+05
20	30	1.933	1.346	0.0362	1.422e+05	3.470e+05
30	40	1.346	0.949	0.0349	9.958e+04	4.465e+05
40	50	0.949	0.672	0.0345	7.035e+04	5.169e+05
50	60	0.672	0.488	0.0320	5.041e+04	5.673e+05
60	70	0.488	0.364	0.0293	3.708e+04	6.044e+05
70	80	0.364	0.276	0.0277	2.787e+04	6.323e+05
80	90	0.276	0.213	0.0259	2.131e+04	6.536e+05
90	100	0.213	0.165	0.0255	1.648e+04	6.700e+05
100	110	0.165	0.136	0.0193	1.315e+04	6.832e+05
110	120	0.136	0.116	0.0159	1.102e+04	6.942e+05
120	130	0.116	0.102	0.0129	9.542e+03	7.038e+05
130	140	0.102	0.092	0.0103	8.495e+03	7.123e+05
140	150	0.092	0.085	0.0079	7.754e+03	7.200e+05
150	160	0.085	0.080	0.0061	7.230e+03	7.272e+05

TABLE IV-2
GAMMA DOSE AT WASTE PACKAGE SURFACE

T1 (yr)	T2 (yr)	D1 (rem/hr)	D2 (rem/hr)	Decay Factor (1/yr)	Dose (rem) T1 to T2	Cumulative Dose (rem)
10	15	35.200	20.290	0.1102	1.186e+06	1.186e+06
15	20	20.290	13.330	0.0840	7.261e+05	1.912e+06
20	30	13.330	6.994	0.0645	8.612e+05	2.773e+06
30	40	6.994	4.351	0.0475	4.881e+05	3.261e+06
40	50	4.351	2.967	0.0383	3.169e+05	3.578e+06
50	60	2.967	2.165	0.0315	2.231e+05	3.801e+06
60	70	2.165	1.643	0.0276	1.659e+05	3.967e+06
70	80	1.643	1.271	0.0257	1.270e+05	4.094e+06
80	90	1.271	0.996	0.0244	9.887e+04	4.193e+06
90	100	0.996	0.785	0.0238	7.769e+04	4.271e+06
100	110	0.785	0.616	0.0242	6.111e+04	4.332e+06
110	120	0.616	0.489	0.0231	4.822e+04	4.380e+06
120	130	0.489	0.387	0.0234	3.822e+04	4.418e+06
130	140	0.387	0.306	0.0235	3.024e+04	4.449e+06
140	150	0.306	0.243	0.0231	2.396e+04	4.473e+06
150	160	0.243	0.194	0.0225	1.907e+04	4.492e+06

TABLE IV-3
NEUTRON DOSE AT DRIFT CONCRETE LINING

T1 (yr)	T2 (yr)	D1 (rem/hr)	D2 (rem/hr)	Decay Factor (1/yr)	Dose (rem) T1 to T2	Cumulative Dose (rem)
10	15	1.388	1.154	0.0369	5.555e+04	5.555e+04
15	20	1.154	0.961	0.0366	4.622e+04	1.018e+05
20	30	0.961	0.669	0.0362	7.067e+04	1.724e+05
30	40	0.669	0.471	0.0351	4.946e+04	2.219e+05
40	50	0.471	0.334	0.0344	3.494e+04	2.568e+05
50	60	0.334	0.243	0.0318	2.508e+04	2.819e+05
60	70	0.243	0.181	0.0295	1.845e+04	3.004e+05
70	80	0.181	0.137	0.0279	1.385e+04	3.142e+05
80	90	0.137	0.106	0.0257	1.059e+04	3.248e+05
90	100	0.106	0.082	0.0257	8.195e+03	3.330e+05
100	110	0.082	0.067	0.0202	6.509e+03	3.395e+05
110	120	0.067	0.058	0.0144	5.469e+03	3.450e+05
120	130	0.058	0.051	0.0129	4.771e+03	3.498e+05
130	140	0.051	0.046	0.0103	4.248e+03	3.540e+05
140	150	0.046	0.042	0.0091	3.854e+03	3.579e+05
150	160	0.042	0.040	0.0049	3.593e+03	3.615e+05

TABLE IV-4
GAMMA DOSE AT DRIFT CONCRETE LINING

T1 (yr)	T2 (yr)	D1 (rem/hr)	D2 (rem/hr)	Decay Factor (1/yr)	Dose (rem) T1 to T2	Cumulative Dose (rem)
10	15	12.330	7.107	0.1102	4.155e+05	4.155e+05
15	20	7.107	4.668	0.0841	2.543e+05	6.698e+05
20	30	4.668	2.450	0.0645	3.016e+05	9.714e+05
30	40	2.450	1.524	0.0475	1.710e+05	1.142e+06
40	50	1.524	1.039	0.0383	1.110e+05	1.253e+06
50	60	1.039	0.758	0.0315	7.812e+04	1.332e+06
60	70	0.758	0.576	0.0275	5.810e+04	1.390e+06
70	80	0.576	0.445	0.0258	4.450e+04	1.434e+06
80	90	0.445	0.349	0.0243	3.463e+04	1.469e+06
90	100	0.349	0.275	0.0238	2.722e+04	1.496e+06
100	110	0.275	0.216	0.0241	2.142e+04	1.517e+06
110	120	0.216	0.171	0.0234	1.689e+04	1.534e+06
120	130	0.171	0.136	0.0229	1.340e+04	1.548e+06
130	140	0.136	0.107	0.0240	1.060e+04	1.558e+06
140	150	0.107	0.085	0.0230	8.378e+03	1.567e+06
150	160	0.085	0.068	0.0223	6.678e+03	1.573e+06

ATTACHMENT V
INPUT AND OUTPUT FILES FOR RADIATION HAZARD ANALYSIS

This attachment gives the computer input and output file names of the MCNP4A runs associated with the radiation hazard analysis (Section 7.3), and provides a complete listing of each input file. All the input and output files are stored on electronic media as Ref. 5.29. This reference is submitted with this document as part of the Design Analysis Records Package. This attachment covers the following MCNP4A cases:

<u>Case Description</u>	<u>Input file</u>	<u>Output File</u>
Single WP, axial fuel neutron dose rates with drift wall	DRIFTN1	DRIFTN1.OUT
Single WP, axial fuel gamma dose rates with drift wall	DRIFTG1	DRIFTG1.OUT
Single WP, axial LEF Co-60 dose rates with drift wall	DRIFTCO1	DRIFTCO1.OUT
Single WP, axial fuel neutron dose rates without drift wall	DRIFTN2	DRIFTN2.OUT
Single WP, axial fuel gamma dose rates without drift wall	DRIFTG2	DRIFTG2.OUT
Single WP, axial LEF Co-60 dose rates without drift wall	DRIFTCO2	DRIFTCO2.OUT

V.1 LISTING OF INPUT FILE DRIFTN1

```

MGDS in-drift dose calculation--neutrons from active fuel region
c   neutrons from 21 PWR fuel assemblies in waste package
c   48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c   drift inside diameter=5.1 m, 0.2 m concrete liner
c   ****
1   1 -3.0431  -1 20 -7          $active fuel (homogenized)
2   5 -0.001225 -1 7 -8          $lower plenum (air assumed)
3   5 -0.001225 -1 21 -20        $upper plenum (air assumed)
4   8 -2.5402   -1 8 -9          $lower end fittings
5   7 -2.4701   -1 22 -21        $upper end fittings
51  4 -8.0038    1 -111 221 -91   $radial WP liner
52  5 -0.001225 111 -112 221 -91  $radial wp gap
53  5 -0.001225 -1 9 -91         $bottom wp gap
54  5 -0.001225 -1 221 -22       $top wp gap
6   2 -8.14      112 -2 221 -91   $WP inner barrier, side
7   2 -8.14      -2 91 -10        $WP inner barrier, bottom
8   2 -8.14      -2 23 -221       $WP inner barrier, top
9   3 -7.832     2 -211 23 -10    $WP outer barrier, side
91  3 -7.832     211 -3 23 -10    $WP outer barrier, side
10  3 -7.832     -3 10 -101       $WP outer barrier, bottom
101 3 -7.832     -3 101 -11       $WP outer barrier, bottom
11  3 -7.832     -3 24 -23       $WP outer barrier, top
12  5 -0.001225 3 -4 24 -11      $radial space between WP & drift wall
13  5 -0.001225 -4 11 -12        $1m tally cell (from WP bottom end)
14  5 -0.001225 -4 12 -13        $2m tally cell
15  5 -0.001225 -4 13 -14        $5m tally cell
16  5 -0.001225 -4 14 -15        $10m tally cell
17  5 -0.001225 -4 15 -16        $20m tally cell
18  5 -0.001225 -4 16 -17        $50m tally cell
19  5 -0.001225 -4 17 -18        $100m tally cell
20  5 -0.001225 -4 18 -19        $150m boundary cell
21  5 -0.001225 -4 25 -24        $rear end air cell
22  9 -2.35      4 -5 25 -11      $first 20 cm drift wall (concrete)
23  6 -2.22      5 -6 25 -11      $second 15 cm drift wall (tuff)
24  0             6:-25:19        $zero importance cell
131 9 -2.35      4 -5 11 -12      $20 cm conc. liner, next to cell 13
141 9 -2.35      4 -5 12 -13      $20 cm conc. liner, next to cell 14
151 9 -2.35      4 -5 13 -14      $20 cm conc. liner, next to cell 15
161 9 -2.35      4 -5 14 -15      $20 cm conc. liner, next to cell 16
171 9 -2.35      4 -5 15 -16      $20 cm conc. liner, next to cell 17
181 9 -2.35      4 -5 16 -17      $20 cm conc. liner, next to cell 18
191 9 -2.35      4 -5 17 -18      $20 cm conc. liner, next to cell 19
201 9 -2.35      4 -5 18 -19      $20 cm conc. liner, next to cell 20
132 6 -2.22      5 -6 11 -12      $15 cm into tuff, next to cell 13
142 6 -2.22      5 -6 12 -13      $15 cm into tuff, next to cell 14
152 6 -2.22      5 -6 13 -14      $15 cm into tuff, next to cell 15
162 6 -2.22      5 -6 14 -15      $15 cm into tuff, next to cell 16
172 6 -2.22      5 -6 15 -16      $15 cm into tuff, next to cell 17
182 6 -2.22      5 -6 16 -17      $15 cm into tuff, next to cell 18
192 6 -2.22      5 -6 17 -18      $15 cm into tuff, next to cell 19
202 6 -2.22      5 -6 18 -19      $15 cm into tuff, next to cell 20

1   cx 61.0          $fuel radial surface
111 cx 62.68        $radial wp liner surface

```

112	cx 69.45	\$radial gap surface
2	cx 71.45	\$WP inner barrier radial surf.
211	cx 76.45	\$WP outer barrier radial, midpoint
3	cx 81.45	\$WP outer barrier radial surf.
4	cx 255.0	\$drift inside radius
5	cx 275.0	\$20 cm concrete liner
6	cx 290.0	\$15 cm into tuff
7	px 180.09	\$bottom of active fuel
8	px 191.69	\$top of lower plenum
9	px 195.50	\$top of lower end fittings
91	px 202.49	\$top of lower WP gap
10	px 204.99	\$bottom of inner barrier
101	px 210.99	\$midpoint, WP outer barrier, bottom
11	px 215.99	\$bottom of outer barrier
12	px 316.0	\$1m from WP end
13	px 416.0	\$2m
14	px 716.0	\$5m
15	px 1216.0	\$10m
16	px 2216.0	\$20m
17	px 5216.0	\$50m
18	px 10216.0	\$100m
19	px 15216.0	\$150m
20	px -180.09	\$top of active fuel
21	px -202.80	\$bottom of upper plenum
22	px -220.11	\$top of fuel assembly
221	px -256.01	\$top of upper WP gap
23	px -258.51	\$top of inner barrier
24	px -269.51	\$top of outer barrier
25	px -500.0	\$rear boundary surface

c active fuel region (smear density 3.0431 g/cc)

m1	7014	-0.00005
	8016	-0.10540
	24000	-0.00017
	26000	-0.00035
	40000	-0.17140
	50000	-0.00244
	92235	-0.00523
	92236	-0.00419
	92238	-0.70508
	94239	-0.00359
	94240	-0.00129
	94241	-0.00082

c alloy 825 (8.14 g/cc)

m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300

c A516 carbon steel (7.832 g/cc)

m3	6000	-0.0022
----	------	---------

	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007
	13027	-0.00124
	14000	-0.00796
	15031	-0.00034
	16032	-0.00025
	22000	-0.00223
	24000	-0.18994
	25055	-0.01548
	26000	-0.55131
	27059	-0.00150
	28000	-0.20504
	29000	-0.00037
	41093	-0.00636
	42000	-0.00757
	73181	-0.00636
c	lower end fittings (2.5402 g/cc)	
m8	6000	-0.00072
	7014	-0.00388

```

      8016  -0.00018
      14000 -0.00904
      15031 -0.00041
      16032 -0.00027
      24000 -0.17193
      25055 -0.01809
      26000 -0.60834
      27059 -0.00181
      28000 -0.09044
      40000 -0.09356
      50000 -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9      8016  -0.4983
        1001  -0.0055
        11023 -0.0170
        12000 -0.0026
        13027 -0.0455
        14000 -0.3157
        16032 -0.0013
        19000 -0.0191
        20000 -0.0826
        26000 -0.0123
mode n
c      cell importance increases as particles move closer to detectors
imp:n 1 1 1 1 0.5 1 1 1 0.5 1 1 0.25 2 4 2 4
      0.25 4 6 6 6 8 8 12 24 24
      0.25 4 1 0
          6 6 6 8 8 12 24 24
          2 2 2 3 3 4 8 8
esplt:n 0.5 1.0-3 0.5 1.0-4 0.5 1.0-5 0.25 1.0-6
phys:n 25.0 1.0-7
sdef cel=1 pos=0 0 0 ext=d1 erg=d2 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c      d1=uniform axial source distribution with biasing towards detectors
si1 -180.09 -90.045 0.0 90.045 180.09
sp1 0.0 0.25 0.25 0.25 0.25
sb1 0.0 0.10 0.15 0.25 0.50
c      d2=neutron energy spectrum
si2 0.10 0.40 0.9 1.4 1.85 3.0 6.43 20.0
sp2 0.0 0.0376 0.1924 0.1767 0.1310 0.2341 0.2099 0.0183
c      d3=uniform radial source distribution (unbiased)
si3 0.0 61.0
c      d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc15 point detectors, ANSI/ANS 6.1.1-1977 (mrem/hr)
f15:n 300.00 0.0 0.0 0.0
      400.00 0.0 0.0 0.0
      800.00 0.0 0.0 0.0
      1300.0 0.0 0.0 0.0
      2000.0 0.0 0.0 0.0
      3500.00 0.0 0.0 0.0
      5500.00 0.0 0.0 0.0
      7500.00 0.0 0.0 0.0
      9500.00 0.0 0.0 0.0
      11500.00 0.0 0.0 0.0
e15 6.25-7 0.1 1.0 25.0

```

```
de15  2.5-8  1.0-7  1.0-6  1.0-5  1.0-4
      1.0-3  1.0-2  0.1    0.5    1.0
      2.5    5.0    7.0    10.0   14.0
      20.0
df15  3.67-3  3.67-3  4.46-3  4.54-3  4.18-3
      3.76-3  3.56-3  2.17-2  9.26-2  1.32-1
      1.25-1  1.56-1  1.47-1  1.47-1  2.08-1
      2.27-1
c      multiplier=6.567+9 neutrons/s per WP for 21 PWR fuel assemblies
fm15  6.567+9
dd15  0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6
      0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6
fq0   e f
ctme  900.0
```

V.2 LISTING OF INPUT FILE DRIFTG1

```

MGDS in-drift dose calculation--primary gammas from active fuel region
c    primary gammas from 21 PWR fuel assemblies in waste package
c    48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c    drift inside diameter=5.1 m, 0.2 m concrete liner
c    ****
1    1 -3.0431  -1 20 -7      $active fuel (homogenized)
2    5 -0.001225 -1 7 -8      $lower plenum (air assumed)
3    5 -0.001225 -1 21 -20    $upper plenum (air assumed)
4    8 -2.5402   -1 8 -9      $lower end fittings
5    7 -2.4701   -1 22 -21    $upper end fittings
51   4 -8.0038    1 -111 221 -91 $radial WP liner
52   5 -0.001225 111 -112 221 -91 $radial wp gap
53   5 -0.001225 -1 9 -91     $bottom wp gap
54   5 -0.001225 -1 221 -22    $top wp gap
6    2 -8.14      112 -2 221 -91 $WP inner barrier, side
7    2 -8.14      -2 91 -10     $WP inner barrier, bottom
8    2 -8.14      -2 23 -221    $WP inner barrier, top
9    3 -7.832     2 -211 23 -10  $WP outer barrier, side
91   3 -7.832     211 -3 23 -10  $WP outer barrier, side
10   3 -7.832     -3 10 -101     $WP outer barrier, bottom
101  3 -7.832     -3 101 -11     $WP outer barrier, bottom
11   3 -7.832     -3 24 -23     $WP outer barrier, top
12   5 -0.001225 3 -4 24 -11     $radial space between WP & drift wall
13   5 -0.001225 -4 11 -12     $1m tally cell (from WP bottom end)
14   5 -0.001225 -4 12 -13     $2m tally cell
15   5 -0.001225 -4 13 -14     $5m tally cell
16   5 -0.001225 -4 14 -15     $10m tally cell
17   5 -0.001225 -4 15 -16     $20m tally cell
18   5 -0.001225 -4 16 -17     $50m tally cell
19   5 -0.001225 -4 17 -18     $100m tally cell
20   5 -0.001225 -4 18 -19     $150m boundary cell
21   5 -0.001225 -4 25 -24     $rear end air cell
22   9 -2.35      4 -5 25 -11    $first 20 cm drift wall (concrete)
23   6 -2.22      5 -6 25 -11    $second 15 cm drift wall (tuff)
24   0             6:-25:19     $zero importance cell
131  9 -2.35      4 -5 11 -12    $20 cm conc. liner, next to cell 13
141  9 -2.35      4 -5 12 -13    $20 cm conc. liner, next to cell 14
151  9 -2.35      4 -5 13 -14    $20 cm conc. liner, next to cell 15
161  9 -2.35      4 -5 14 -15    $20 cm conc. liner, next to cell 16
171  9 -2.35      4 -5 15 -16    $20 cm conc. liner, next to cell 17
181  9 -2.35      4 -5 16 -17    $20 cm conc. liner, next to cell 18
191  9 -2.35      4 -5 17 -18    $20 cm conc. liner, next to cell 19
201  9 -2.35      4 -5 18 -19    $20 cm conc. liner, next to cell 20
132  6 -2.22      5 -6 11 -12    $15 cm into tuff, next to cell 13
142  6 -2.22      5 -6 12 -13    $15 cm into tuff, next to cell 14
152  6 -2.22      5 -6 13 -14    $15 cm into tuff, next to cell 15
162  6 -2.22      5 -6 14 -15    $15 cm into tuff, next to cell 16
172  6 -2.22      5 -6 15 -16    $15 cm into tuff, next to cell 17
182  6 -2.22      5 -6 16 -17    $15 cm into tuff, next to cell 18
192  6 -2.22      5 -6 17 -18    $15 cm into tuff, next to cell 19
202  6 -2.22      5 -6 18 -19    $15 cm into tuff, next to cell 20

1    cx 61.0      $fuel radial surface
111  cx 62.68    $radial wp liner surface

```


112	cx	69.45	\$radial gap surface
2	cx	71.45	\$WP inner barrier radial surf.
211	cx	76.45	\$WP outer barrier radial, midpoint
3	cx	81.45	\$WP outer barrier radial surf.
4	cx	255.0	\$drift inside radius
5	cx	275.0	\$20 cm concrete liner
6	cx	290.0	\$15 cm into tuff
7	px	180.09	\$bottom of active fuel
8	px	191.69	\$top of lower plenum
9	px	195.50	\$top of lower end fittings
91	px	202.49	\$top of lower WP gap
10	px	204.99	\$bottom of inner barrier
101	px	210.99	\$midpoint, WP outer barrier, bottom
11	px	215.99	\$bottom of outer barrier
12	px	316.0	\$1m from WP end
13	px	416.0	\$2m
14	px	716.0	\$5m
15	px	1216.0	\$10m
16	px	2216.0	\$20m
17	px	5216.0	\$50m
18	px	10216.0	\$100m
19	px	15216.0	\$150m
20	px	-180.09	\$top of active fuel
21	px	-202.80	\$bottom of upper plenum
22	px	-220.11	\$top of fuel assembly
221	px	-256.01	\$top of upper WP gap
23	px	-258.51	\$top of inner barrier
24	px	-269.51	\$top of outer barrier
25	px	-500.0	\$rear boundary surface

c active fuel region (smear density 3.0431 g/cc)

m1	7014	-0.00005
	8016	-0.10540
	24000	-0.00017
	26000	-0.00035
	40000	-0.17140
	50000	-0.00244
	92235	-0.00523
	92236	-0.00419
	92238	-0.70508
	94239	-0.00359
	94240	-0.00129
	94241	-0.00082

c alloy 825 (8.14 g/cc)

m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300

c A516 carbon steel (7.832 g/cc)

m3	6000	-0.0022
----	------	---------

	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007
	13027	-0.00124
	14000	-0.00796
	15031	-0.00034
	16032	-0.00025
	22000	-0.00223
	24000	-0.18994
	25055	-0.01548
	26000	-0.55131
	27059	-0.00150
	28000	-0.20504
	29000	-0.00037
	41093	-0.00636
	42000	-0.00757
	73181	-0.00636
c	lower end fittings (2.5402 g/cc)	
m8	6000	-0.00072
	7014	-0.00388

```

      8016  -0.00018
      14000 -0.00904
      15031 -0.00041
      16032 -0.00027
      24000 -0.17193
      25055 -0.01809
      26000 -0.60834
      27059 -0.00181
      28000 -0.09044
      40000 -0.09356
      50000 -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9      8016  -0.4983
      1001  -0.0055
      11023 -0.0170
      12000 -0.0026
      13027 -0.0455
      14000 -0.3157
      16032 -0.0013
      19000 -0.0191
      20000 -0.0826
      26000 -0.0123
mode p
c      cell importance increases as particles move closer to detectors
imp:p 1 1 1 1 0.5 1 1 1 1 1 0.25 2 8 4 16
      0.25 16 64 70 100 200 400 1000 4000 8000
      0.25 16 5 0
      64 70 100 200 400 1000 4000 8000
      16 18 25 50 100 250 1000 2000
esplt:p 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=1 pos=0 0 0 ext=d1 erg=d2 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c      d1=uniform axial source distribution with biasing towards detectors
si1 -180.09 -90.045 0.0 90.045 180.09
sp1 0.0 0.25 0.25 0.25 0.25
sb1 0.0 0.05 0.10 0.15 0.70
c      d2=photon energy spectrum for contributing groups only
c      with biasing for high energy photons
si2 0.45 0.70 1.0 1.5 2.0 2.5 3.0
sp2 0.0 0.8485 0.08260 6.806-02 8.628-04 1.360-05 1.050-06
sb2 0.0 0.17 0.15 0.65 0.03 0.001 0.0002
c      d3=uniform radial source distribution (unbiased)
si3 0.0 61.0
c      d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc15 point detectors, ANSI/ANS 6.1.1-1977 (mrem/hr)
f15:p 300.00 0.0 0.0 0.0
      400.00 0.0 0.0 0.0
      800.00 0.0 0.0 0.0
      1300.0 0.0 0.0 0.0
      2000.0 0.0 0.0 0.0
      3500.00 0.0 0.0 0.0
      5500.00 0.0 0.0 0.0
      7500.00 0.0 0.0 0.0
      9500.00 0.0 0.0 0.0

```

```

      11500.00 0.0 0.0 0.0
e15   0.045 0.1 0.45 0.70 1.0 1.5 2.0 2.5 3.0
de15  0.010 0.030 0.050 0.070 0.100 0.150
      0.200 0.250 0.300 0.350 0.400 0.450
      0.500 0.550 0.600 0.650 0.700 0.800
      1.000 1.400 1.800 2.200 2.600 2.800
      3.25  3.75  4.25  4.75  5.0  5.25
      5.75  6.25  6.75  7.5  9.0  11.0
      13.0  15.0
df15  3.96-3 5.82-4 2.90-4 2.58-4 2.83-4 3.79-4
      5.01-4 6.31-4 7.59-4 8.78-4 9.85-4 1.08-3
      1.17-3 1.27-3 1.36-3 1.44-3 1.52-3 1.68-3
      1.98-3 2.51-3 2.99-3 3.42-3 3.82-3 4.01-3
      4.41-3 4.83-3 5.23-3 5.60-3 5.80-3 6.01-3
      6.37-3 6.74-3 7.11-3 7.66-3 8.77-3 1.03-2
      1.18-2 1.33-2
c      multiplier=5.51+16 photons/s per WP for 21 PWR fuel assemblies
c      and for 0.45 - 3.0 gammas, which contribute to dose results
fm15  5.51+16
dd15  0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6
      0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6
fq0   e f
ctme  900.0

```

V.3 LISTING OF INPUT FILE DRIFTCO1

```

MGDS in-drift dose calculation--Co-60 gammas from lower end fittings
c    Co-60 gammas from 21 PWR fuel assemblies in waste package
c    48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c    drift inside diameter=5.1 m, 0.2 m concrete liner
c    *****
1    1 -3.0431  -1 20 -7          $active fuel (homogenized)
2    5 -0.001225 -1 7 -8          $lower plenum (air assumed)
3    5 -0.001225 -1 21 -20        $upper plenum (air assumed)
4    8 -2.5402  -1 8 -9          $lower end fittings
5    7 -2.4701  -1 22 -21        $upper end fittings
51   4 -8.0038  1 -111 221 -91    $radial WP liner
52   5 -0.001225 111 -112 221 -91 $radial wp gap
53   5 -0.001225 -1 9 -91        $bottom wp gap
54   5 -0.001225 -1 221 -22      $top wp gap
6    2 -8.14    112 -2 221 -91   $WP inner barrier, side
7    2 -8.14    -2 91 -10        $WP inner barrier, bottom
8    2 -8.14    -2 23 -221       $WP inner barrier, top
9    3 -7.832   2 -211 23 -10     $WP outer barrier, side
91   3 -7.832   211 -3 23 -10     $WP outer barrier, side
10   3 -7.832   -3 10 -101       $WP outer barrier, bottom
101  3 -7.832   -3 101 -11       $WP outer barrier, bottom
11   3 -7.832   -3 24 -23       $WP outer barrier, top
12   5 -0.001225 3 -4 24 -11     $radial space between WP & drift wall
13   5 -0.001225 -4 11 -12      $1m tally cell (from WP bottom end)
14   5 -0.001225 -4 12 -13      $2m tally cell
15   5 -0.001225 -4 13 -14      $5m tally cell
16   5 -0.001225 -4 14 -15      $10m tally cell
17   5 -0.001225 -4 15 -16      $20m tally cell
18   5 -0.001225 -4 16 -17      $50m tally cell
19   5 -0.001225 -4 17 -18      $100m tally cell
20   5 -0.001225 -4 18 -19      $150m boundary cell
21   5 -0.001225 -4 25 -24      $rear end air cell
22   9 -2.35    4 -5 25 -11      $first 20 cm drift wall (concrete)
23   6 -2.22    5 -6 25 -11      $second 15 cm drift wall (tuff)
24   0          6:-25:19        $zero importance cell
131  9 -2.35    4 -5 11 -12      $20 cm conc. liner, next to cell 13
141  9 -2.35    4 -5 12 -13      $20 cm conc. liner, next to cell 14
151  9 -2.35    4 -5 13 -14      $20 cm conc. liner, next to cell 15
161  9 -2.35    4 -5 14 -15      $20 cm conc. liner, next to cell 16
171  9 -2.35    4 -5 15 -16      $20 cm conc. liner, next to cell 17
181  9 -2.35    4 -5 16 -17      $20 cm conc. liner, next to cell 18
191  9 -2.35    4 -5 17 -18      $20 cm conc. liner, next to cell 19
201  9 -2.35    4 -5 18 -19      $20 cm conc. liner, next to cell 20
132  6 -2.22    5 -6 11 -12      $15 cm into tuff, next to cell 13
142  6 -2.22    5 -6 12 -13      $15 cm into tuff, next to cell 14
152  6 -2.22    5 -6 13 -14      $15 cm into tuff, next to cell 15
162  6 -2.22    5 -6 14 -15      $15 cm into tuff, next to cell 16
172  6 -2.22    5 -6 15 -16      $15 cm into tuff, next to cell 17
182  6 -2.22    5 -6 16 -17      $15 cm into tuff, next to cell 18
192  6 -2.22    5 -6 17 -18      $15 cm into tuff, next to cell 19
202  6 -2.22    5 -6 18 -19      $15 cm into tuff, next to cell 20

1    cx 61.0          $fuel radial surface
111  cx 62.68        $radial wp liner surface

```

112	cx 69.45	\$radial gap surface
2	cx 71.45	\$WP inner barrier radial surf.
211	cx 76.45	\$WP outer barrier radial,midpoint
3	cx 81.45	\$WP outer barrier radial surf.
4	cx 255.0	\$drift inside radius
5	cx 275.0	\$20 cm concrete liner
6	cx 290.0	\$15 cm into tuff
7	px 180.09	\$bottom of active fuel
8	px 191.69	\$top of lower plenum
9	px 195.50	\$top of lower end fittings
91	px 202.49	\$top of lower WP gap
10	px 204.99	\$bottom of inner barrier
101	px 210.99	\$midpoint, WP outer barrier, bottom
11	px 215.99	\$bottom of outer barrier
12	px 316.0	\$1m from WP end
13	px 416.0	\$2m
14	px 716.0	\$5m
15	px 1216.0	\$10m
16	px 2216.0	\$20m
17	px 5216.0	\$50m
18	px 10216.0	\$100m
19	px 15216.0	\$150m
20	px -180.09	\$top of active fuel
21	px -202.80	\$bottom of upper plenum
22	px -220.11	\$top of fuel assembly
221	px -256.01	\$top of upper WP gap
23	px -258.51	\$top of inner barrier
24	px -269.51	\$top of outer barrier
25	px -500.0	\$rear boundary surface

c active fuel region (smear density 3.0431 g/cc)

m1	7014	-0.00005
	8016	-0.10540
	24000	-0.00017
	26000	-0.00035
	40000	-0.17140
	50000	-0.00244
	92235	-0.00523
	92236	-0.00419
	92238	-0.70508
	94239	-0.00359
	94240	-0.00129
	94241	-0.00082

c alloy 825 (8.14 g/cc)

m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300

c A516 carbon steel (7.832 g/cc)

m3	6000	-0.0022
----	------	---------

	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007
	13027	-0.00124
	14000	-0.00796
	15031	-0.00034
	16032	-0.00025
	22000	-0.00223
	24000	-0.18994
	25055	-0.01548
	26000	-0.55131
	27059	-0.00150
	28000	-0.20504
	29000	-0.00037
	41093	-0.00636
	42000	-0.00757
	73181	-0.00636
c	lower end fittings (2.5402 g/cc)	
m8	6000	-0.00072
	7014	-0.00388

```

      8016  -0.00018
      14000 -0.00904
      15031 -0.00041
      16032 -0.00027
      24000 -0.17193
      25055 -0.01809
      26000 -0.60834
      27059 -0.00181
      28000 -0.09044
      40000 -0.09356
      50000 -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9      8016  -0.4983
        1001  -0.0055
        11023 -0.0170
        12000 -0.0026
        13027 -0.0455
        14000 -0.3157
        16032 -0.0013
        19000 -0.0191
        20000 -0.0826
        26000 -0.0123
mode p
c      importance set by cell to move particle tracks towards detectors
c      (by increasing impotance as particles move closer to detectors)
imp:p 1 1 1 1 0.5 1 1 1 1 1 1 0.25 2 8 4 16
      0.25 16 64 70 100 200 400 1000 4000 16000
      0.25 16 5 0
          64 70 100 200 400 1000 4000 16000
          16 18 25 50 100 250 1000 4000
esplt:p 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=4 pos=0 0 0 ext=d1 erg=1.25 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c      d1=uniform axial source distribution p(x)=c
sil 191.69 195.50
spl 0.0 1.0
c      d3=unifrom radial source distribution p(x)=cx
si3 0.0 61.0
c      d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc15 point detectors, ANSI/ANS 6.1.1-1977 (mrem/hr)
f15:p 300.00 0.0 0.0 0.0
      400.00 0.0 0.0 0.0
      800.00 0.0 0.0 0.0
      1300.00 0.0 0.0 0.0
      2000.00 0.0 0.0 0.0
      3500.00 0.0 0.0 0.0
      5500.00 0.0 0.0 0.0
      7500.00 0.0 0.0 0.0
      9500.00 0.0 0.0 0.0
      11500.00 0.0 0.0 0.0
e15 0.045 0.1 0.45 0.70 1.0 1.5 2.0
de15 0.010 0.030 0.050 0.070 0.100 0.150
      0.200 0.250 0.300 0.350 0.400 0.450
      0.500 0.550 0.600 0.650 0.700 0.800

```


	1.000	1.400	1.800	2.200	2.600	2.800
	3.25	3.75	4.25	4.75	5.0	5.25
	5.75	6.25	6.75	7.5	9.0	11.0
	13.0	15.0				
df15	3.96-3	5.82-4	2.90-4	2.58-4	2.83-4	3.79-4
	5.01-4	6.31-4	7.59-4	8.78-4	9.85-4	1.08-3
	1.17-3	1.27-3	1.36-3	1.44-3	1.52-3	1.68-3
	1.98-3	2.51-3	2.99-3	3.42-3	3.82-3	4.01-3
	4.41-3	4.83-3	5.23-3	5.60-3	5.80-3	6.01-3
	6.37-3	6.74-3	7.11-3	7.66-3	8.77-3	1.03-2
	1.18-2	1.33-2				
c	multiplier=(121 Ci Co60/fuel assembly) (21 assemblies per WP)					
c	(3.7E10 dis/s per Ci) (2 photons/dis)					
c	=1.88+14 photons/s per WP					
fm15	1.88+14					
ddl5	0	1.0+6	0	1.0+6	0	1.0+6
	0	1.0+6	0	1.0+6	0	1.0+6
fq0	e f					
ctme	400.0					

V.4 LISTING OF INPUT FILE DRIFTN2

```

MGDS in-drift dose calculation--neutrons from active fuel region
c   neutrons from 21 PWR fuel assemblies in waste package
c   48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c   no drift wall
c   ****
1   1 -3.0431   -1 20 -7           $active fuel (homogenized)
2   5 -0.001225 -1 7 -8           $lower plenum (air assumed)
3   5 -0.001225 -1 21 -20        $upper plenum (air assumed)
4   8 -2.5402   -1 8 -9           $lower end fittings
5   7 -2.4701   -1 22 -21        $upper end fittings
51  4 -8.0038    1 -111 221 -91    $radial WP liner
52  5 -0.001225 111 -112 221 -91  $radial wp gap
53  5 -0.001225 -1 9 -91         $bottom wp gap
54  5 -0.001225 -1 221 -22        $top wp gap
6   2 -8.14      112 -2 221 -91    $WP inner barrier, side
7   2 -8.14      -2 91 -10        $WP inner barrier, bottom
8   2 -8.14      -2 23 -221       $WP inner barrier, top
9   3 -7.832     2 -211 23 -10     $WP outer barrier, side
91  3 -7.832     211 -3 23 -10     $WP outer barrier, side
10  3 -7.832      -3 10 -101       $WP outer barrier, bottom
101 3 -7.832      -3 101 -11       $WP outer barrier, bottom
11  3 -7.832      -3 24 -23       $WP outer barrier, top
12  5 -0.001225 3 -4 24 -11       $radial space between WP & drift wall
13  5 -0.001225 -4 11 -12        $1m tally cell (from WP bottom end)
14  5 -0.001225 -4 12 -13        $2m tally cell
15  5 -0.001225 -4 13 -14        $5m tally cell
16  5 -0.001225 -4 14 -15        $10m tally cell
17  5 -0.001225 -4 15 -16        $20m tally cell
18  5 -0.001225 -4 16 -17        $50m tally cell
19  5 -0.001225 -4 17 -18        $100m tally cell
20  5 -0.001225 -4 18 -19        $150m boundary cell
21  5 -0.001225 -4 25 -24        $rear end air cell
22  5 -0.001225 4 -5 25 -11       $first 20 cm drift wall (concrete)
23  5 -0.001225 5 -6 25 -11       $second 15 cm drift wall (tuff)
24  0             6:-25:19        $zero importance cell
131 5 -0.001225 4 -5 11 -12       $20 cm conc. liner, next to cell 13
141 5 -0.001225 4 -5 12 -13       $20 cm conc. liner, next to cell 14
151 5 -0.001225 4 -5 13 -14       $20 cm conc. liner, next to cell 15
161 5 -0.001225 4 -5 14 -15       $20 cm conc. liner, next to cell 16
171 5 -0.001225 4 -5 15 -16       $20 cm conc. liner, next to cell 17
181 5 -0.001225 4 -5 16 -17       $20 cm conc. liner, next to cell 18
191 5 -0.001225 4 -5 17 -18       $20 cm conc. liner, next to cell 19
201 5 -0.001225 4 -5 18 -19       $20 cm conc. liner, next to cell 20
132 5 -0.001225 5 -6 11 -12       $15 cm into tuff, next to cell 13
142 5 -0.001225 5 -6 12 -13       $15 cm into tuff, next to cell 14
152 5 -0.001225 5 -6 13 -14       $15 cm into tuff, next to cell 15
162 5 -0.001225 5 -6 14 -15       $15 cm into tuff, next to cell 16
172 5 -0.001225 5 -6 15 -16       $15 cm into tuff, next to cell 17
182 5 -0.001225 5 -6 16 -17       $15 cm into tuff, next to cell 18
192 5 -0.001225 5 -6 17 -18       $15 cm into tuff, next to cell 19
202 5 -0.001225 5 -6 18 -19       $15 cm into tuff, next to cell 20

1   cx 61.0           $fuel radial surface
111 cx 62.68         $radial wp liner surface

```

112	cx 69.45	\$radial gap surface
2	cx 71.45	\$WP inner barrier radial surf.
211	cx 76.45	\$WP outer barrier radial,midpoint
3	cx 81.45	\$WP outer barrier radial surf.
4	cx 255.0	\$drift inside radius
5	cx 275.0	\$20 cm concrete liner
6	cx 290.0	\$15 cm into tuff
7	px 180.09	\$bottom of active fuel
8	px 191.69	\$top of lower plenum
9	px 195.50	\$top of lower end fittings
91	px 202.49	\$top of lower WP gap
10	px 204.99	\$bottom of inner barrier
101	px 210.99	\$midpoint, WP outer barrier, bottom
11	px 215.99	\$bottom of outer barrier
12	px 316.0	\$1m from WP end
13	px 416.0	\$2m
14	px 716.0	\$5m
15	px 1216.0	\$10m
16	px 2216.0	\$20m
17	px 5216.0	\$50m
18	px 10216.0	\$100m
19	px 15216.0	\$150m
20	px -180.09	\$top of active fuel
21	px -202.80	\$bottom of upper plenum
22	px -220.11	\$top of fuel assembly
221	px -256.01	\$top of upper WP gap
23	px -258.51	\$top of inner barrier
24	px -269.51	\$top of outer barrier
25	px -500.0	\$rear boundary surface

c active fuel region (smear density 3.0431 g/cc)

m1	7014	-0.00005
	8016	-0.10540
	24000	-0.00017
	26000	-0.00035
	40000	-0.17140
	50000	-0.00244
	92235	-0.00523
	92236	-0.00419
	92238	-0.70508
	94239	-0.00359
	94240	-0.00129
	94241	-0.00082

c alloy 825 (8.14 g/cc)

m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300

c A516 carbon steel (7.832 g/cc)

m3	6000	-0.0022
----	------	---------

	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007
	13027	-0.00124
	14000	-0.00796
	15031	-0.00034
	16032	-0.00025
	22000	-0.00223
	24000	-0.18994
	25055	-0.01548
	26000	-0.55131
	27059	-0.00150
	28000	-0.20504
	29000	-0.00037
	41093	-0.00636
	42000	-0.00757
	73181	-0.00636
c	lower end fittings (2.5402 g/cc)	
m8	6000	-0.00072
	7014	-0.00388

```

      8016  -0.00018
      14000 -0.00904
      15031 -0.00041
      16032 -0.00027
      24000 -0.17193
      25055 -0.01809
      26000 -0.60834
      27059 -0.00181
      28000 -0.09044
      40000 -0.09356
      50000 -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9      8016  -0.4983
      1001  -0.0055
      11023 -0.0170
      12000 -0.0026
      13027 -0.0455
      14000 -0.3157
      16032 -0.0013
      19000 -0.0191
      20000 -0.0826
      26000 -0.0123
mode n
c      cell importance increases as particles move closer to detectors
imp:n 1 1 1 1 0.5 1 1 1 0.5 1 1 0.25 2 4 2 4
      0.25 4 6 6 6 8 8 12 24 24
      0.25 4 1 0
           6 6 6 8 8 12 24 24
           2 2 2 3 3 4 8 8
esplt:n 0.5 1.0-3 0.5 1.0-4 0.5 1.0-5 0.25 1.0-6
phys:n 25.0 1.0-7
sdef cel=1 pos=0 0 0 ext=d1 erg=d2 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c      d1=uniform axial source distribution with biasing towards detectors
si1 -180.09 -90.045 0.0 90.045 180.09
sp1 0.0 0.25 0.25 0.25 0.25
sb1 0.0 0.10 0.15 0.25 0.50
c      d2=neutron energy spectrum
si2 0.10 0.40 0.9 1.4 1.85 3.0 6.43 20.0
sp2 0.0 0.0376 0.1924 0.1767 0.1310 0.2341 0.2099 0.0183
c      d3=uniform radial source distribution (unbiased)
si3 0.0 61.0
c      d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc15 point detectors, ANSI/ANS 6.1.1-1977 (mrem/hr)
f15:n 300.00 0.0 0.0 0.0
      400.00 0.0 0.0 0.0
      800.00 0.0 0.0 0.0
      1300.0 0.0 0.0 0.0
      2000.0 0.0 0.0 0.0
      3500.00 0.0 0.0 0.0
      5500.00 0.0 0.0 0.0
      7500.00 0.0 0.0 0.0
      9500.00 0.0 0.0 0.0
      11500.00 0.0 0.0 0.0
e15 6.25-7 0.1 1.0 25.0

```

```
de15  2.5-8  1.0-7  1.0-6  1.0-5  1.0-4
      1.0-3  1.0-2  0.1    0.5    1.0
      2.5    5.0    7.0    10.0   14.0
      20.0
df15  3.67-3  3.67-3  4.46-3  4.54-3  4.18-3
      3.76-3  3.56-3  2.17-2  9.26-2  1.32-1
      1.25-1  1.56-1  1.47-1  1.47-1  2.08-1
      2.27-1
c      multiplier=6.567+9 neutrons/s per WP for 21 PWR fuel assemblies
fm15  6.567+9
dd15  0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6
      0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6
fq0   e f
ctme  200.0
```

V.5 LISTING OF INPUT FILE DRIFTG1

```

MGDS in-drift dose calculation--primary gammas from active fuel region
c    primary gammas from 21 PWR fuel assemblies in waste package
c    48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c    no drift wall
c    *****
1    1 -3.0431  -1 20 -7      $active fuel (homogenized)
2    5 -0.001225 -1 7 -8      $lower plenum (air assumed)
3    5 -0.001225 -1 21 -20    $upper plenum (air assumed)
4    8 -2.5402  -1 8 -9      $lower end fittings
5    7 -2.4701  -1 22 -21    $upper end fittings
51   4 -8.0038   1 -111 221 -91 $radial WP liner
52   5 -0.001225 111 -112 221 -91 $radial wp gap
53   5 -0.001225 -1 9 -91     $bottom wp gap
54   5 -0.001225 -1 221 -22    $top wp gap
6    2 -8.14     112 -2 221 -91 $WP inner barrier, side
7    2 -8.14     -2 91 -10     $WP inner barrier, bottom
8    2 -8.14     -2 23 -221    $WP inner barrier, top
9    3 -7.832    2 -211 23 -10  $WP outer barrier, side
91   3 -7.832    211 -3 23 -10  $WP outer barrier, side
10   3 -7.832    -3 10 -101     $WP outer barrier, bottom
101  3 -7.832    -3 101 -11     $WP outer barrier, bottom
11   3 -7.832    -3 24 -23     $WP outer barrier, top
12   5 -0.001225 3 -4 24 -11    $radial space between WP & drift wall
13   5 -0.001225 -4 11 -12     $1m tally cell (from WP bottom end)
14   5 -0.001225 -4 12 -13     $2m tally cell
15   5 -0.001225 -4 13 -14     $5m tally cell
16   5 -0.001225 -4 14 -15     $10m tally cell
17   5 -0.001225 -4 15 -16     $20m tally cell
18   5 -0.001225 -4 16 -17     $50m tally cell
19   5 -0.001225 -4 17 -18     $100m tally cell
20   5 -0.001225 -4 18 -19     $150m boundary cell
21   5 -0.001225 -4 25 -24     $rear end air cell
22   5 -0.001225 4 -5 25 -11    $first 20 cm drift wall (concrete)
23   5 -0.001225 5 -6 25 -11    $second 15 cm drift wall (tuff)
24   0          6:-25:19       $zero importance cell
131  5 -0.001225 4 -5 11 -12    $20 cm conc. liner, next to cell 13
141  5 -0.001225 4 -5 12 -13    $20 cm conc. liner, next to cell 14
151  5 -0.001225 4 -5 13 -14    $20 cm conc. liner, next to cell 15
161  5 -0.001225 4 -5 14 -15    $20 cm conc. liner, next to cell 16
171  5 -0.001225 4 -5 15 -16    $20 cm conc. liner, next to cell 17
181  5 -0.001225 4 -5 16 -17    $20 cm conc. liner, next to cell 18
191  5 -0.001225 4 -5 17 -18    $20 cm conc. liner, next to cell 19
201  5 -0.001225 4 -5 18 -19    $20 cm conc. liner, next to cell 20
132  5 -0.001225 5 -6 11 -12    $15 cm into tuff, next to cell 13
142  5 -0.001225 5 -6 12 -13    $15 cm into tuff, next to cell 14
152  5 -0.001225 5 -6 13 -14    $15 cm into tuff, next to cell 15
162  5 -0.001225 5 -6 14 -15    $15 cm into tuff, next to cell 16
172  5 -0.001225 5 -6 15 -16    $15 cm into tuff, next to cell 17
182  5 -0.001225 5 -6 16 -17    $15 cm into tuff, next to cell 18
192  5 -0.001225 5 -6 17 -18    $15 cm into tuff, next to cell 19
202  5 -0.001225 5 -6 18 -19    $15 cm into tuff, next to cell 20

1    cx 61.0                  $fuel radial surface

```

111	cx 62.68	\$radial wp liner surface
112	cx 69.45	\$radial gap surface
2	cx 71.45	\$WP inner barrier radial surf.
211	cx 76.45	\$WP outer barrier radial, midpoint
3	cx 81.45	\$WP outer barrier radial surf.
4	cx 255.0	\$drift inside radius
5	cx 275.0	\$20 cm concrete liner
6	cx 290.0	\$15 cm into tuff
7	px 180.09	\$bottom of active fuel
8	px 191.69	\$top of lower plenum
9	px 195.50	\$top of lower end fittings
91	px 202.49	\$top of lower WP gap
10	px 204.99	\$bottom of inner barrier
101	px 210.99	\$midpoint, WP outer barrier, bottom
11	px 215.99	\$bottom of outer barrier
12	px 316.0	\$1m from WP end
13	px 416.0	\$2m
14	px 716.0	\$5m
15	px 1216.0	\$10m
16	px 2216.0	\$20m
17	px 5216.0	\$50m
18	px 10216.0	\$100m
19	px 15216.0	\$150m
20	px -180.09	\$top of active fuel
21	px -202.80	\$bottom of upper plenum
22	px -220.11	\$top of fuel assembly
221	px -256.01	\$top of upper WP gap
23	px -258.51	\$top of inner barrier
24	px -269.51	\$top of outer barrier
25	px -500.0	\$rear boundary surface

c active fuel region (smear density 3.0431 g/cc)

m1	7014	-0.00005
	8016	-0.10540
	24000	-0.00017
	26000	-0.00035
	40000	-0.17140
	50000	-0.00244
	92235	-0.00523
	92236	-0.00419
	92238	-0.70508
	94239	-0.00359
	94240	-0.00129
	94241	-0.00082

c alloy 825 (8.14 g/cc)

m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300

c A516 carbon steel (7.832 g/cc)

m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner	(SS316B6A + A825, 8.0038 g/cc)
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air	(0.001225 g/cc)
m5	7014	-0.8
	8016	-0.2
c	tuff material	(2.22 g/cc)
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings	(2.4701 g/cc)
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007
	13027	-0.00124
	14000	-0.00796
	15031	-0.00034
	16032	-0.00025
	22000	-0.00223
	24000	-0.18994
	25055	-0.01548
	26000	-0.55131
	27059	-0.00150
	28000	-0.20504
	29000	-0.00037
	41093	-0.00636
	42000	-0.00757
	73181	-0.00636
c	lower end fittings	(2.5402 g/cc)
m8	6000	-0.00072

```

7014 -0.00388
8016 -0.00018
14000 -0.00904
15031 -0.00041
16032 -0.00027
24000 -0.17193
25055 -0.01809
26000 -0.60834
27059 -0.00181
28000 -0.09044
40000 -0.09356
50000 -0.00133
c ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9 8016 -0.4983
1001 -0.0055
11023 -0.0170
12000 -0.0026
13027 -0.0455
14000 -0.3157
16032 -0.0013
19000 -0.0191
20000 -0.0826
26000 -0.0123
mode p
c cell importance increases as particles move closer to detectors
imp:p 1 1 1 1 0.5 1 1 1 1 1 0.25 2 8 4 16
0.25 16 64 70 100 200 400 1000 4000 8000
0.25 16 5 0
64 70 100 200 400 1000 4000 8000
16 18 25 50 100 250 1000 2000
esplt:p 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=1 pos=0 0 0 ext=d1 erg=d2 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c d1=uniform axial source distribution with biasing towards detectors
si1 -180.09 -90.045 0.0 90.045 180.09
sp1 0.0 0.25 0.25 0.25 0.25
sb1 0.0 0.05 0.10 0.15 0.70
c d2=photon energy spectrum for contributing groups only
c with biasing for high energy photons
si2 0.45 0.70 1.0 1.5 2.0 2.5 3.0
sp2 0.0 0.8485 0.08260 6.806-02 8.628-04 1.360-05 1.050-06
sb2 0.0 0.17 0.15 0.65 0.03 0.001 0.0002
c d3=uniform radial source distribution (unbiased)
si3 0.0 61.0
c d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc15 point detectors, ANSI/ANS 6.1.1-1977 (mrem/hr)
f15:p 300.00 0.0 0.0 0.0
400.00 0.0 0.0 0.0
800.00 0.0 0.0 0.0
1300.0 0.0 0.0 0.0
2000.0 0.0 0.0 0.0
3500.00 0.0 0.0 0.0
5500.00 0.0 0.0 0.0
7500.00 0.0 0.0 0.0

```

```

          9500.00  0.0 0.0 0.0
          11500.00 0.0 0.0 0.0
e15      0.045 0.1 0.45 0.70 1.0 1.5 2.0 2.5 3.0
de15     0.010  0.030  0.050  0.070  0.100  0.150
          0.200  0.250  0.300  0.350  0.400  0.450
          0.500  0.550  0.600  0.650  0.700  0.800
          1.000  1.400  1.800  2.200  2.600  2.800
          3.25  3.75  4.25  4.75  5.0  5.25
          5.75  6.25  6.75  7.5  9.0  11.0
          13.0  15.0
df15     3.96-3  5.82-4  2.90-4  2.58-4  2.83-4  3.79-4
          5.01-4  6.31-4  7.59-4  8.78-4  9.85-4  1.08-3
          1.17-3  1.27-3  1.36-3  1.44-3  1.52-3  1.68-3
          1.98-3  2.51-3  2.99-3  3.42-3  3.82-3  4.01-3
          4.41-3  4.83-3  5.23-3  5.60-3  5.80-3  6.01-3
          6.37-3  6.74-3  7.11-3  7.66-3  8.77-3  1.03-2
          1.18-2  1.33-2
c        multiplier=5.51+16 photons/s per WP for 21 PWR fuel assemblies
c        and for 0.45 - 3.0 gammas, which contribute to dose results
fm15     5.51+16
dd15     0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6
          0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6 0 1.0+6
fq0      e f
ctme     400.0

```

V.6 LISTING OF INPUT FILE DRIFTCO2

MGDS in-drift dose calculation--Co-60 gammas from lower end fittings

```

c      Co-60 gammas from 21 PWR fuel assemblies in waste package
c      48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c      no drift wall
c      *****
1      1 -3.0431   -1 20 -7          $active fuel (homogenized)
2      5 -0.001225 -1 7 -8          $lower plenum (air assumed)
3      5 -0.001225 -1 21 -20        $upper plenum (air assumed)
4      8 -2.5402   -1 8 -9          $lower end fittings
5      7 -2.4701   -1 22 -21        $upper end fittings
51     4 -8.0038    1 -111 221 -91   $radial WP liner
52     5 -0.001225 111 -112 221 -91  $radial wp gap
53     5 -0.001225 -1 9 -91         $bottom wp gap
54     5 -0.001225 -1 221 -22        $top wp gap
6      2 -8.14      112 -2 221 -91   $WP inner barrier, side
7      2 -8.14      -2 91 -10        $WP inner barrier, bottom
8      2 -8.14      -2 23 -221       $WP inner barrier, top
9      3 -7.832     2 -211 23 -10     $WP outer barrier, side
91     3 -7.832     211 -3 23 -10     $WP outer barrier, side
10     3 -7.832     -3 10 -101        $WP outer barrier, bottom
101    3 -7.832     -3 101 -11        $WP outer barrier, bottom
11     3 -7.832     -3 24 -23        $WP outer barrier, top
12     5 -0.001225 3 -4 24 -11       $radial space between WP & drift wall
13     5 -0.001225 -4 11 -12        $1m tally cell (from WP bottom end)
14     5 -0.001225 -4 12 -13        $2m tally cell
15     5 -0.001225 -4 13 -14        $5m tally cell
16     5 -0.001225 -4 14 -15        $10m tally cell
17     5 -0.001225 -4 15 -16        $20m tally cell
18     5 -0.001225 -4 16 -17        $50m tally cell
19     5 -0.001225 -4 17 -18        $100m tally cell
20     5 -0.001225 -4 18 -19        $150m boundary cell
21     5 -0.001225 -4 25 -24        $rear end air cell
22     5 -0.001225 4 -5 25 -11       $first 20 cm drift wall (concrete)
23     5 -0.001225 5 -6 25 -11       $second 15 cm drift wall (tuff)
24     0              6:-25:19       $zero importance cell
131    5 -0.001225 4 -5 11 -12       $20 cm conc. liner, next to cell 13
141    5 -0.001225 4 -5 12 -13       $20 cm conc. liner, next to cell 14
151    5 -0.001225 4 -5 13 -14       $20 cm conc. liner, next to cell 15
161    5 -0.001225 4 -5 14 -15       $20 cm conc. liner, next to cell 16
171    5 -0.001225 4 -5 15 -16       $20 cm conc. liner, next to cell 17
181    5 -0.001225 4 -5 16 -17       $20 cm conc. liner, next to cell 18
191    5 -0.001225 4 -5 17 -18       $20 cm conc. liner, next to cell 19
201    5 -0.001225 4 -5 18 -19       $20 cm conc. liner, next to cell 20
132    5 -0.001225 5 -6 11 -12       $15 cm into tuff, next to cell 13
142    5 -0.001225 5 -6 12 -13       $15 cm into tuff, next to cell 14
152    5 -0.001225 5 -6 13 -14       $15 cm into tuff, next to cell 15
162    5 -0.001225 5 -6 14 -15       $15 cm into tuff, next to cell 16
172    5 -0.001225 5 -6 15 -16       $15 cm into tuff, next to cell 17
182    5 -0.001225 5 -6 16 -17       $15 cm into tuff, next to cell 18
192    5 -0.001225 5 -6 17 -18       $15 cm into tuff, next to cell 19
202    5 -0.001225 5 -6 18 -19       $15 cm into tuff, next to cell 20

1      cx 61.0          $fuel radial surface
111    cx 62.68        $radial wp liner surface

```

112	cx 69.45	\$radial gap surface
2	cx 71.45	\$WP inner barrier radial surf.
211	cx 76.45	\$WP outer barrier radial,midpoint
3	cx 81.45	\$WP outer barrier radial surf.
4	cx 255.0	\$drift inside radius
5	cx 275.0	\$20 cm concrete liner
6	cx 290.0	\$15 cm into tuff
7	px 180.09	\$bottom of active fuel
8	px 191.69	\$stop of lower plenum
9	px 195.50	\$stop of lower end fittings
91	px 202.49	\$stop of lower WP gap
10	px 204.99	\$bottom of inner barrier
101	px 210.99	\$midpoint, WP outer barrier, bottom
11	px 215.99	\$bottom of outer barrier
12	px 316.0	\$1m from WP end
13	px 416.0	\$2m
14	px 716.0	\$5m
15	px 1216.0	\$10m
16	px 2216.0	\$20m
17	px 5216.0	\$50m
18	px 10216.0	\$100m
19	px 15216.0	\$150m
20	px -180.09	\$stop of active fuel
21	px -202.80	\$bottom of upper plenum
22	px -220.11	\$stop of fuel assembly
221	px -256.01	\$stop of upper WP gap
23	px -258.51	\$stop of inner barrier
24	px -269.51	\$stop of outer barrier
25	px -500.0	\$rear boundary surface

c active fuel region (smear density 3.0431 g/cc)

m1

7014	-0.00005
8016	-0.10540
24000	-0.00017
26000	-0.00035
40000	-0.17140
50000	-0.00244
92235	-0.00523
92236	-0.00419
92238	-0.70508
94239	-0.00359
94240	-0.00129
94241	-0.00082

c alloy 825 (8.14 g/cc)

m2

6000	-0.0005
13027	-0.0020
14000	-0.0050
16032	-0.0003
22000	-0.0090
24000	-0.2150
25055	-0.0100
26000	-0.2857
28000	-0.4200
29000	-0.0225
42000	-0.0300

c A516 carbon steel (7.832 g/cc)

m3

6000	-0.0022
------	---------

	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007
	13027	-0.00124
	14000	-0.00796
	15031	-0.00034
	16032	-0.00025
	22000	-0.00223
	24000	-0.18994
	25055	-0.01548
	26000	-0.55131
	27059	-0.00150
	28000	-0.20504
	29000	-0.00037
	41093	-0.00636
	42000	-0.00757
	73181	-0.00636
c	lower end fittings (2.5402 g/cc)	
m8	6000	-0.00072
	7014	-0.00388

```

      8016  -0.00018
      14000 -0.00904
      15031 -0.00041
      16032 -0.00027
      24000 -0.17193
      25055 -0.01809
      26000 -0.60834
      27059 -0.00181
      28000 -0.09044
      40000 -0.09356
      50000 -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9      8016  -0.4983
        1001  -0.0055
        11023 -0.0170
        12000 -0.0026
        13027 -0.0455
        14000 -0.3157
        16032 -0.0013
        19000 -0.0191
        20000 -0.0826
        26000 -0.0123
mode p
c      importance set by cell to move particle tracks towards detectors
c      (by increasing impotance as particles move closer to detectors)
imp:p 1 1 1 1 0.5 1 1 1 1 1 1 0.25 2 8 4 16
      0.25 16 64 70 100 200 400 1000 4000 16000
      0.25 16 5 0
          64 70 100 200 400 1000 4000 16000
          16 18 25 50 100 250 1000 4000
esplt:p 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=4 pos=0 0 0 ext=d1 erg=1.25 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c      d1=uniform axial source distribution p(x)=c
si1 191.69 195.50
sp1 0.0 1.0
c      d3=unifrom radial source distribution p(x)=cx
si3 0.0 61.0
c      d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc15 point detectors, ANSI/ANS 6.1.1-1977 (mrem/hr)
f15:p 300.00 0.0 0.0 0.0
      400.00 0.0 0.0 0.0
      800.00 0.0 0.0 0.0
      1300.00 0.0 0.0 0.0
      2000.00 0.0 0.0 0.0
      3500.00 0.0 0.0 0.0
      5500.00 0.0 0.0 0.0
      7500.00 0.0 0.0 0.0
      9500.00 0.0 0.0 0.0
      11500.00 0.0 0.0 0.0
e15 0.045 0.1 0.45 0.70 1.0 1.5 2.0
de15 0.010 0.030 0.050 0.070 0.100 0.150
      0.200 0.250 0.300 0.350 0.400 0.450
      0.500 0.550 0.600 0.650 0.700 0.800

```

	1.000	1.400	1.800	2.200	2.600	2.800
	3.25	3.75	4.25	4.75	5.0	5.25
	5.75	6.25	6.75	7.5	9.0	11.0
	13.0	15.0				
df15	3.96-3	5.82-4	2.90-4	2.58-4	2.83-4	3.79-4
	5.01-4	6.31-4	7.59-4	8.78-4	9.85-4	1.08-3
	1.17-3	1.27-3	1.36-3	1.44-3	1.52-3	1.68-3
	1.98-3	2.51-3	2.99-3	3.42-3	3.82-3	4.01-3
	4.41-3	4.83-3	5.23-3	5.60-3	5.80-3	6.01-3
	6.37-3	6.74-3	7.11-3	7.66-3	8.77-3	1.03-2
	1.18-2	1.33-2				
c	multiplier=(121 Ci Co60/fuel assembly) (21 assemblies per WP)					
c	(3.7E10 dis/s per Ci) (2 photons/dis)					
c	=1.88+14 photons/s per WP					
fm15	1.88+14					
dd15	0	1.0+6	0	1.0+6	0	1.0+6
	0	1.0+6	0	1.0+6	0	1.0+6
fq0	e f					
ctme	200.0					

ATTACHMENT VI
INPUT AND OUTPUT FILES FOR CONCRETE SHADOW SHIELD ANALYSIS

This attachment gives the computer input and output file names of the MCNP4A runs associated with the concrete shadow shield analysis (Section 7.4), and provides a complete listing of each input file. All the input and output files are stored on electronic media as Ref. 5.29. This reference is submitted with this document as part of the Design Analysis Records Package. This attachment covers the following MCNP4A cases:

<u>Case Description</u>	<u>Input file</u>	<u>Output File</u>
Fuel neutron dose rates with drift entrance door closed	DRIFTNA	DRIFTNA.OUT
Fuel gamma dose rates with drift entrance door closed	DRIFTGA	DRIFTGA.OUT
LEF Co-60 dose rates with drift entrance door closed	DRIFTCOA	DRIFTCOA.OUT
Fuel neutron dose rates with drift entrance door open	DRIFTNB	DRIFTNB.OUT
Fuel gamma dose rates with drift entrance door open	DRIFTGB	DRIFTGB.OUT
LEF Co-60 dose rates with drift entrance door open	DRIFTCOB	DRIFTCOB.OUT

VI.1 LISTING OF INPUT FILE DRIFTNA

```

MGDS in-drift dose calculation--neutrons from active fuel
c   neutron from 21 PWR fuel assemblies in waste package
c   48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c   drift inside diameter=5.1 m, 0.2 m concrete liner
c   18" concrete shadow shield and 1" steel isolation door modeled
c   *****
1   1 -3.0431   -1 20 -7           $active fuel (homogenized)
2   5 -0.001225 -1 7 -8           $lower plenum (air assumed)
3   5 -0.001225 -1 21 -20        $upper plenum (air assumed)
4   8 -2.5402   -1 8 -9           $lower end fittings
5   7 -2.4701   -1 22 -21        $upper end fittings
51  4 -8.0038    1 -111 221 -91   $radial WP liner
52  5 -0.001225 111 -112 221 -91  $radial wp gap
53  5 -0.001225 -1 9 -91         $bottom wp gap
54  5 -0.001225 -1 221 -22       $top wp gap
6   2 -8.14      112 -2 221 -91  $WP inner barrier, side
7   2 -8.14      -2 91 -10       $WP inner barrier, bottom
8   2 -8.14      -2 23 -221      $WP inner barrier, top
9   3 -7.832     2 -211 23 -10    $WP outer barrier, side
91  3 -7.832     211 -3 23 -10    $WP outer barrier, side
10  3 -7.832      -3 10 -101      $WP outer barrier, bottom
101 3 -7.832      -3 101 -11      $WP outer barrier, bottom
11  3 -7.832      -3 24 -23      $WP outer barrier, top
12  5 -0.001225 3 -4 24 -11      $radial space between WP & drift wall
13  5 -0.001225 -4 11 -12       $4m from WP center
1311 9 -2.35      -31 12 -1311    $1st 0.75 ft concrete shield
1312 5 -0.001225 31 -4 12 -1311  $air adjacent to cell 1311
1315 9 -2.35      -31 1311 -1312  $2nd 0.75 ft concrete shield
1316 5 -0.001225 31 -4 1311 -1312  $air adjacent to cell 1315
14  5 -0.001225 -4 1312 -13      $5m from WP center
15  5 -0.001225 -4 13 -14        $7m from WP center
16  5 -0.001225 -4 14 -15        $13m from WP center
1611 10 -7.90     -4 15 -1611     $1" drift isolation door
17  5 -0.001225 -4 1611 -16      $24m from WP center
18  5 -0.001225 -4 16 -17        $35m from WP center
19  5 -0.001225 -4 17 -18        $50m from WP center
21  5 -0.001225 -4 25 -24        $rear end air cell
22  9 -2.35       4 -5 25 -11     $first 20 cm drift wall (concrete)
23  6 -2.22       5 -6 25 -11     $second 15 cm drift wall (tuff)
24  0             6:-25:18        $zero importance cell
131 9 -2.35       4 -5 11 -12     $20 cm conc. liner, next to cell 13
1313 9 -2.35      4 -5 12 -1311    $20 cm liner next to #1311 & #1312
1317 9 -2.35      4 -5 1311 -1312  $20 cm liner next to #1315 & #1316
141 9 -2.35       4 -5 1312 -13    $20 cm conc. liner, next to cell 14
151 9 -2.35       4 -5 13 -14      $20 cm conc. liner, next to cell 15
161 9 -2.35       4 -5 14 -15      $20 cm conc. liner, next to cell 16
171 9 -2.35       4 -5 15 -16      $20 cm conc. liner, next to cell 17
181 9 -2.35       4 -5 16 -17      $20 cm conc. liner, next to cell 18
191 9 -2.35       4 -5 17 -18      $20 cm conc. liner, next to cell 19
132 6 -2.22       5 -6 11 -12     $15 cm into tuff, next to cell 13
1314 6 -2.22      5 -6 12 -1311    $15 cm into tuff next to #1311 & #1312
1318 6 -2.22      5 -6 1311 -1312  $15 cm into tuff next to #1315 & #1316
142 6 -2.22       5 -6 1312 -13    $15 cm into tuff, next to cell 14
152 6 -2.22       5 -6 13 -14      $15 cm into tuff, next to cell 15

```

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162	6	-2.22	5	-6	14	-15	\$15 cm into tuff, next to cell 16
172	6	-2.22	5	-6	15	-16	\$15 cm into tuff, next to cell 17
182	6	-2.22	5	-6	16	-17	\$15 cm into tuff, next to cell 18
192	6	-2.22	5	-6	17	-18	\$15 cm into tuff, next to cell 19
1	cx	61.0					\$fuel radial surface
111	cx	62.68					\$radial wp liner surface
112	cx	69.45					\$radial gap surface
2	cx	71.45					\$WP inner barrier radial surf.
211	cx	76.45					\$WP outer barrier radial, midpoint
3	cx	81.45					\$WP outer barrier radial surf.
31	cx	107.5					\$concrete shadow shield radius
4	cx	255.0					\$drift inside radius
5	cx	275.0					\$20 cm concrete liner
6	cx	290.0					\$15 cm into tuff
7	px	180.09					\$bottom of active fuel
8	px	191.69					\$stop of lower plenum
9	px	195.50					\$stop of lower end fittings
91	px	202.49					\$stop of lower WP gap
10	px	204.99					\$bottom of inner barrier
101	px	210.99					\$midpoint, WP outer barrier, bottom
11	px	215.99					\$bottom of outer barrier
12	px	400.0					\$4m from WP center(2m from end)
1311	px	422.86					\$1st 0.75 ft concrete shield
1312	px	445.72					\$2nd 0.75 ft concrete shield
13	px	500.0					\$5m from wp center
14	px	700.0					\$7m from wp center
15	px	1300.0					\$13m from ***wp center***
1611	px	1302.54					\$1" isolation door
16	px	2400.0					\$24m from WP center
17	px	3500.0					\$35m from wp center
18	px	5000.0					\$50m from wp center
20	px	-180.09					\$stop of active fuel
21	px	-202.80					\$bottom of upper plenum
22	px	-220.11					\$stop of fuel assembly
221	px	-256.01					\$stop of upper WP gap
23	px	-258.51					\$stop of inner barrier
24	px	-269.51					\$stop of outer barrier
25	px	-500.0					\$rear boundary surface
311	cx	130.0					
312	cx	150.0					
313	cx	170.0					
314	cx	190.0					
315	cx	210.0					
316	cx	230.0					
317	cx	245.0					
c	active fuel region (smear density 3.0431 g/cc)						
m1	7014	-0.00005					
	8016	-0.10540					
	24000	-0.00017					
	26000	-0.00035					
	40000	-0.17140					
	50000	-0.00244					
	92235	-0.00523					
	92236	-0.00419					
	92238	-0.70508					

	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007

```

13027 -0.00124
14000 -0.00796
15031 -0.00034
16032 -0.00025
22000 -0.00223
24000 -0.18994
25055 -0.01548
26000 -0.55131
27059 -0.00150
28000 -0.20504
29000 -0.00037
41093 -0.00636
42000 -0.00757
73181 -0.00636
c lower end fittings (2.5402 g/cc)
m8 6000 -0.00072
7014 -0.00388
8016 -0.00018
14000 -0.00904
15031 -0.00041
16032 -0.00027
24000 -0.17193
25055 -0.01809
26000 -0.60834
27059 -0.00181
28000 -0.09044
40000 -0.09356
50000 -0.00133
c ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9 8016 -0.4983
1001 -0.0055
11023 -0.0170
12000 -0.0026
13027 -0.0455
14000 -0.3157
16032 -0.0013
19000 -0.0191
20000 -0.0826
26000 -0.0123
c SS304 for drift isolation door
m10 6000 -0.00080
7014 -0.00100
14000 -0.00750
15031 -0.00045
16032 -0.00030
24000 -0.19000
25055 -0.02000
26000 -0.68745
28000 -0.09250
mode n
c importance set by cell to move particle tracks towards detectors
c (by increasing impotance as particles move closer to detectors)
imp:n 1 1 1 1 0.5 1 1 1 0.5 1 1 0.25 2 4 2 4
0.25 8 8 10 1r 20 1r 20 20 30 100 200 200 200
0.25 8 2 0
8 10 20 20 20 30 200 200 200
2 2.5 5 5 5 7.5 50 50 50

```

```

esplt:n 0.5 1.0-3 0.5 1.0-4 0.5 1.0-5 0.25 1.0-6
phys:n 25.0 1.0-7
sdef cel=1 pos=0 0 0 ext=d1 erg=d2 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c d1=uniform axial source distribution with biasing towards detectors
si1 -180.09 -90.045 0.0 90.045 180.09
sp1 0.0 0.25 0.25 0.25 0.25
sb1 0.0 0.10 0.15 0.25 0.50
c d2=neutron energy spectrum
si2 0.10 0.40 0.9 1.4 1.85 3.0 6.43 20.0
sp2 0.0 0.0376 0.1924 0.1767 0.1310 0.2341 0.2099 0.0183
c d3=uniform radial source distribution (unbiased)
si3 0.0 61.0
c d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc12 surface tallies, ANSI/ANS 6.1.1-1977 (mrem/hr)
fl2:n 15 1611 16 17
fs12 -31 -311 -312 -313 -314 -315 -316 -317 -4
sd12 3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
e12 6.25-7 0.1 1.0 25.0
de12 2.5-8 1.0-7 1.0-6 1.0-5 1.0-4
1.0-3 1.0-2 0.1 0.5 1.0
2.5 5.0 7.0 10.0 14.0
20.0
df12 3.67-3 3.67-3 4.46-3 4.54-3 4.18-3
3.76-3 3.56-3 2.17-2 9.26-2 1.32-1
1.25-1 1.56-1 1.47-1 1.47-1 2.08-1
2.27-1
c multiplier=6.567+9 neutrons/s per WP for 21 PWR fuel assemblies
fm12 6.567+9
fq0 e s
ctme 720.0

```

VI.2 LISTING OF INPUT FILE DRIFTGA

```

MGDS in-drift dose calculation--primary gammas from active fuel
c    primary gammas from 21 PWR fuel assemblies in waste package
c    48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c    drift inside diameter=5.1 m, 0.2 m concrete liner
c    18" concrete shadow shield and 1" steel isolation door modeled
c    *****
1    1 -3.0431  -1 20 -7      $active fuel (homogenized)
2    5 -0.001225 -1 7 -8      $lower plenum (air assumed)
3    5 -0.001225 -1 21 -20    $upper plenum (air assumed)
4    8 -2.5402  -1 8 -9      $lower end fittings
5    7 -2.4701  -1 22 -21    $upper end fittings
51   4 -8.0038   1 -111 221 -91 $radial WP liner
52   5 -0.001225 111 -112 221 -91 $radial wp gap
53   5 -0.001225 -1 9 -91     $bottom wp gap
54   5 -0.001225 -1 221 -22   $top wp gap
6    2 -8.14     112 -2 221 -91 $WP inner barrier, side
7    2 -8.14     -2 91 -10     $WP inner barrier, bottom
8    2 -8.14     -2 23 -221    $WP inner barrier, top
9    3 -7.832    2 -211 23 -10  $WP outer barrier, side
91   3 -7.832    211 -3 23 -10  $WP outer barrier, side
10   3 -7.832     -3 10 -101    $WP outer barrier, bottom
101  3 -7.832     -3 101 -11    $WP outer barrier, bottom
11   3 -7.832     -3 24 -23     $WP outer barrier, top
12   5 -0.001225 3 -4 24 -11    $radial space between WP & drift wall
13   5 -0.001225 -4 11 -12     $4m from WP center
1311 9 -2.35     -31 12 -1311    $1st 0.75 ft concrete shield
1312 5 -0.001225 31 -4 12 -1311 $air adjacent to cell 1311
1315 9 -2.35     -31 1311 -1312 $2nd 0.75 ft concrete shield
1316 5 -0.001225 31 -4 1311 -1312 $air adjacent to cell 1315
14   5 -0.001225 -4 1312 -13   $5m from WP center
15   5 -0.001225 -4 13 -14     $7m from WP center
16   5 -0.001225 -4 14 -15     $13m from WP center
1611 10 -7.90    -4 15 -1611    $1" drift isolation door
17   5 -0.001225 -4 1611 -16    $24m from WP center
18   5 -0.001225 -4 16 -17     $35m from WP center
19   5 -0.001225 -4 17 -18     $50m from WP center
21   5 -0.001225 -4 25 -24     $rear end air cell
22   9 -2.35     4 -5 25 -11    $first 20 cm drift wall (concrete)
23   6 -2.22     5 -6 25 -11    $second 15 cm drift wall (tuff)
24   0           6:-25:18      $zero importance cell
131  9 -2.35     4 -5 11 -12    $20 cm conc. liner, next to cell 13
1313 9 -2.35     4 -5 12 -1311  $20 cm liner next to #1311 & #1312
1317 9 -2.35     4 -5 1311 -1312 $20 cm liner next to #1315 & #1316
141  9 -2.35     4 -5 1312 -13   $20 cm conc. liner, next to cell 14
151  9 -2.35     4 -5 13 -14     $20 cm conc. liner, next to cell 15
161  9 -2.35     4 -5 14 -15     $20 cm conc. liner, next to cell 16
171  9 -2.35     4 -5 15 -16     $20 cm conc. liner, next to cell 17
181  9 -2.35     4 -5 16 -17     $20 cm conc. liner, next to cell 18
191  9 -2.35     4 -5 17 -18     $20 cm conc. liner, next to cell 19
132  6 -2.22     5 -6 11 -12    $15 cm into tuff, next to cell 13
1314 6 -2.22     5 -6 12 -1311  $15 cm into tuff next to #1311 & #1312
1318 6 -2.22     5 -6 1311 -1312 $15 cm into tuff next to #1315 & #1316
142  6 -2.22     5 -6 1312 -13   $15 cm into tuff, next to cell 14
152  6 -2.22     5 -6 13 -14     $15 cm into tuff, next to cell 15

```

162	6 -2.22	5 -6 14 -15	\$15 cm into tuff, next to cell 16
172	6 -2.22	5 -6 15 -16	\$15 cm into tuff, next to cell 17
182	6 -2.22	5 -6 16 -17	\$15 cm into tuff, next to cell 18
192	6 -2.22	5 -6 17 -18	\$15 cm into tuff, next to cell 19
1	cx 61.0		\$fuel radial surface
111	cx 62.68		\$radial wp liner surface
112	cx 69.45		\$radial gap surface
2	cx 71.45		\$WP inner barrier radial surf.
211	cx 76.45		\$WP outer barrier radial, midpoint
3	cx 81.45		\$WP outer barrier radial surf.
31	cx 107.5		\$concrete shadow shield radius
4	cx 255.0		\$drift inside radius
5	cx 275.0		\$20 cm concrete liner
6	cx 290.0		\$15 cm into tuff
7	px 180.09		\$bottom of active fuel
8	px 191.69		\$top of lower plenum
9	px 195.50		\$top of lower end fittings
91	px 202.49		\$top of lower WP gap
10	px 204.99		\$bottom of inner barrier
101	px 210.99		\$midpoint, WP outer barrier, bottom
11	px 215.99		\$bottom of outer barrier
12	px 400.0		\$4m from WP center(2m from end)
1311	px 422.86		\$1st 0.75 ft concrete shield
1312	px 445.72		\$2nd 0.75 ft concrete shield
13	px 500.0		\$5m from wp center
14	px 700.0		\$7m from wp center
15	px 1300.0		\$13m from ***wp center***
1611	px 1302.54		\$1" isolation door
16	px 2400.0		\$24m from WP center
17	px 3500.0		\$35m from wp center
18	px 5000.0		\$50m from wp center
20	px -180.09		\$top of active fuel
21	px -202.80		\$bottom of upper plenum
22	px -220.11		\$top of fuel assembly
221	px -256.01		\$top of upper WP gap
23	px -258.51		\$top of inner barrier
24	px -269.51		\$top of outer barrier
25	px -500.0		\$rear boundary surface
311	cx 130.0		
312	cx 150.0		
313	cx 170.0		
314	cx 190.0		
315	cx 210.0		
316	cx 230.0		
317	cx 245.0		
c	active fuel region (smear density 3.0431 g/cc)		
ml	7014	-0.00005	
	8016	-0.10540	
	24000	-0.00017	
	26000	-0.00035	
	40000	-0.17140	
	50000	-0.00244	
	92235	-0.00523	
	92236	-0.00419	
	92238	-0.70508	

	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007

```

13027  -0.00124
14000  -0.00796
15031  -0.00034
16032  -0.00025
22000  -0.00223
24000  -0.18994
25055  -0.01548
26000  -0.55131
27059  -0.00150
28000  -0.20504
29000  -0.00037
41093  -0.00636
42000  -0.00757
73181  -0.00636
c      lower end fittings (2.5402 g/cc)
m8     6000  -0.00072
       7014  -0.00388
       8016  -0.00018
14000  -0.00904
15031  -0.00041
16032  -0.00027
24000  -0.17193
25055  -0.01809
26000  -0.60834
27059  -0.00181
28000  -0.09044
40000  -0.09356
50000  -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9     8016  -0.4983
       1001  -0.0055
11023  -0.0170
12000  -0.0026
13027  -0.0455
14000  -0.3157
16032  -0.0013
19000  -0.0191
20000  -0.0826
26000  -0.0123
c      SS304 for drift isolation door
m10    6000  -0.00080
       7014  -0.00100
14000  -0.00750
15031  -0.00045
16032  -0.00030
24000  -0.19000
25055  -0.02000
26000  -0.68745
28000  -0.09250
mode   p
c      importance set by cell to move particle tracks towards detectors
c      (by increasing impotance as particles move closer to detectors)
imp:p  1 1 1 1 0.5 1 1 1 0.5 1 1 0.25 2 8 3 12
       0.25 16 48 100 1r 200 1r 200 250 750 3000 12000 48000 48000
       0.25 16 5 0
           48 100      200      200 250 750      12000 48000 48000
           12 25      50      50 65 200      3000 12000 12000

```

```

esplt:p 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=1 pos=0 0 0 ext=d1 erg=d2 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c d1=uniform axial source distribution with biasing towards detectors
si1 -180.09 -90.045 0.0 90.045 180.09
sp1 0.0 0.25 0.25 0.25 0.25
sb1 0.0 0.10 0.15 0.20 0.55
c d2=photon energy spectrum for contributing groups only
c with biasing for high energy photons
si2 0.45 0.70 1.0 1.5 2.0 2.5 3.0
sp2 0.0 0.8485 0.08260 6.806-02 8.628-04 1.360-05 1.050-06
sb2 0.0 0.17 0.15 0.65 0.03 0.001 0.0002
c d3=uniform radial source distribution (unbiased)
si3 0.0 61.0
c d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc12 surface tallies, ANSI/ANS 6.1.1-1977 (mrem/hr)
fl2:p 15 1611 16 17
fs12 -31 -311 -312 -313 -314 -315 -316 -317 -4
sd12 3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
e12 0.045 0.1 0.45 0.70 1.0 1.5 2.0 2.5 3.0
de12 0.010 0.030 0.050 0.070 0.100 0.150
0.200 0.250 0.300 0.350 0.400 0.450
0.500 0.550 0.600 0.650 0.700 0.800
1.000 1.400 1.800 2.200 2.600 2.800
3.25 3.75 4.25 4.75 5.0 5.25
5.75 6.25 6.75 7.5 9.0 11.0
13.0 15.0
df12 3.96-3 5.82-4 2.90-4 2.58-4 2.83-4 3.79-4
5.01-4 6.31-4 7.59-4 8.78-4 9.85-4 1.08-3
1.17-3 1.27-3 1.36-3 1.44-3 1.52-3 1.68-3
1.98-3 2.51-3 2.99-3 3.42-3 3.82-3 4.01-3
4.41-3 4.83-3 5.23-3 5.60-3 5.80-3 6.01-3
6.37-3 6.74-3 7.11-3 7.66-3 8.77-3 1.03-2
1.18-2 1.33-2
c multiplier=5.51+16 photons/s per WP for 21 PWR fuel assemblies
c and for 0.45 - 3.0 gammas, which contribute to dose results
fm12 5.51+16
fq0 e s
ctme 2600.0

```

VI.3 LISTING OF INPUT FILE DRIFTCOA

```

MGDS in-drift dose calculation--Co-60 gammas from lower end fittings
c   Co-60 gammas from 21 PWR fuel assemblies in waste package
c   48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c   drift inside diameter=5.1 m, 0.2 m concrete liner
c   18" concrete shadow shield and 1" steel isolation door modeled
c   ****
1   1 -3.0431 -1 20 -7      $active fuel (homogenized)
2   5 -0.001225 -1 7 -8     $lower plenum (air assumed)
3   5 -0.001225 -1 21 -20   $upper plenum (air assumed)
4   8 -2.5402 -1 8 -9       $lower end fittings
5   7 -2.4701 -1 22 -21     $upper end fittings
51  4 -8.0038 1 -111 221 -91 $radial WP liner
52  5 -0.001225 111 -112 221 -91 $radial wp gap
53  5 -0.001225 -1 9 -91    $bottom wp gap
54  5 -0.001225 -1 221 -22   $top wp gap
6   2 -8.14 112 -2 221 -91  $WP inner barrier, side
7   2 -8.14 -2 91 -10       $WP inner barrier, bottom
8   2 -8.14 -2 23 -221      $WP inner barrier, top
9   3 -7.832 2 -211 23 -10   $WP outer barrier, side
91  3 -7.832 211 -3 23 -10   $WP outer barrier, side
10  3 -7.832 -3 10 -101      $WP outer barrier, bottom
101 3 -7.832 -3 101 -11      $WP outer barrier, bottom
11  3 -7.832 -3 24 -23       $WP outer barrier, top
12  5 -0.001225 3 -4 24 -11  $radial space between WP & drift wall
13  5 -0.001225 -4 11 -12    $4m from WP center
1311 9 -2.35 -31 12 -1311    $1st 0.75 ft concrete shield
1312 5 -0.001225 31 -4 12 -1311 $air adjacent to cell 1311
1315 9 -2.35 -31 1311 -1312  $2nd 0.75 ft concrete shield
1316 5 -0.001225 31 -4 1311 -1312 $air adjacent to cell 1315
14  5 -0.001225 -4 1312 -13  $5m from WP center
15  5 -0.001225 -4 13 -14     $7m from WP center
16  5 -0.001225 -4 14 -15     $13m from WP center
1611 10 -7.90 -4 15 -1611    $1" drift isolation door
17  5 -0.001225 -4 1611 -16   $24m from WP center
18  5 -0.001225 -4 16 -17     $35m from WP center
19  5 -0.001225 -4 17 -18     $50m from WP center
21  5 -0.001225 -4 25 -24     $rear end air cell
22  9 -2.35 4 -5 25 -11      $first 20 cm drift wall (concrete)
23  6 -2.22 5 -6 25 -11      $second 15 cm drift wall (tuff)
24  0 6:-25:18               $zero importance cell
131 9 -2.35 4 -5 11 -12      $20 cm conc. liner, next to cell 13
1313 9 -2.35 4 -5 12 -1311    $20 cm liner next to #1311 & #1312
1317 9 -2.35 4 -5 1311 -1312  $20 cm liner next to #1315 & #1316
141 9 -2.35 4 -5 1312 -13     $20 cm conc. liner, next to cell 14
151 9 -2.35 4 -5 13 -14       $20 cm conc. liner, next to cell 15
161 9 -2.35 4 -5 14 -15       $20 cm conc. liner, next to cell 16
171 9 -2.35 4 -5 15 -16       $20 cm conc. liner, next to cell 17
181 9 -2.35 4 -5 16 -17       $20 cm conc. liner, next to cell 18
191 9 -2.35 4 -5 17 -18       $20 cm conc. liner, next to cell 19
132 6 -2.22 5 -6 11 -12      $15 cm into tuff, next to cell 13
1314 6 -2.22 5 -6 12 -1311    $15 cm into tuff next to #1311 & #1312
1318 6 -2.22 5 -6 1311 -1312  $15 cm into tuff next to #1315 & #1316
142 6 -2.22 5 -6 1312 -13     $15 cm into tuff, next to cell 14
152 6 -2.22 5 -6 13 -14       $15 cm into tuff, next to cell 15

```

162	6 -2.22	5 -6 14 -15	\$15 cm into tuff, next to cell 16
172	6 -2.22	5 -6 15 -16	\$15 cm into tuff, next to cell 17
182	6 -2.22	5 -6 16 -17	\$15 cm into tuff, next to cell 18
192	6 -2.22	5 -6 17 -18	\$15 cm into tuff, next to cell 19
1	cx 61.0		\$fuel radial surface
111	cx 62.68		\$radial wp liner surface
112	cx 69.45		\$radial gap surface
2	cx 71.45		\$WP inner barrier radial surf.
211	cx 76.45		\$WP outer barrier radial, midpoint
3	cx 81.45		\$WP outer barrier radial surf.
31	cx 107.5		\$concrete shadow shield radius
4	cx 255.0		\$drift inside radius
5	cx 275.0		\$20 cm concrete liner
6	cx 290.0		\$15 cm into tuff
7	px 180.09		\$bottom of active fuel
8	px 191.69		\$top of lower plenum
9	px 195.50		\$top of lower end fittings
91	px 202.49		\$top of lower WP gap
10	px 204.99		\$bottom of inner barrier
101	px 210.99		\$midpoint, WP outer barrier, bottom
11	px 215.99		\$bottom of outer barrier
12	px 400.0		\$4m from WP center(2m from end)
1311	px 422.86		\$1st 0.75 ft concrete shield
1312	px 445.72		\$2nd 0.75 ft concrete shield
13	px 500.0		\$5m from wp center
14	px 700.0		\$7m from wp center
15	px 1300.0		\$13m from ***wp center***
1611	px 1302.54		\$1" isolation door
16	px 2400.0		\$24m from WP center
17	px 3500.0		\$35m from wp center
18	px 5000.0		\$50m from wp center
20	px -180.09		\$top of active fuel
21	px -202.80		\$bottom of upper plenum
22	px -220.11		\$top of fuel assembly
221	px -256.01		\$top of upper WP gap
23	px -258.51		\$top of inner barrier
24	px -269.51		\$top of outer barrier
25	px -500.0		\$rear boundary surface
311	cx 130.0		
312	cx 150.0		
313	cx 170.0		
314	cx 190.0		
315	cx 210.0		
316	cx 230.0		
317	cx 245.0		
c	active fuel region (smear density 3.0431 g/cc)		
m1	7014	-0.00005	
	8016	-0.10540	
	24000	-0.00017	
	26000	-0.00035	
	40000	-0.17140	
	50000	-0.00244	
	92235	-0.00523	
	92236	-0.00419	
	92238	-0.70508	

	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007

```

13027  -0.00124
14000  -0.00796
15031  -0.00034
16032  -0.00025
22000  -0.00223
24000  -0.18994
25055  -0.01548
26000  -0.55131
27059  -0.00150
28000  -0.20504
29000  -0.00037
41093  -0.00636
42000  -0.00757
73181  -0.00636
c      lower end fittings (2.5402 g/cc)
m8     6000  -0.00072
       7014  -0.00388
       8016  -0.00018
       14000 -0.00904
       15031 -0.00041
       16032 -0.00027
       24000 -0.17193
       25055 -0.01809
       26000 -0.60834
       27059 -0.00181
       28000 -0.09044
       40000 -0.09356
       50000 -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9     8016  -0.4983
       1001  -0.0055
       11023 -0.0170
       12000 -0.0026
       13027 -0.0455
       14000 -0.3157
       16032 -0.0013
       19000 -0.0191
       20000 -0.0826
       26000 -0.0123
c      SS304 for drift isolation door
m10    6000  -0.00080
       7014  -0.00100
       14000 -0.00750
       15031 -0.00045
       16032 -0.00030
       24000 -0.19000
       25055 -0.02000
       26000 -0.68745
       28000 -0.09250
mode   p
c      importance set by cell to move particle tracks towards detectors
c      (by increasing impotance as particles move closer to detectors)
imp:p  1 1 1 1 0.5 1 1 1 0.5 1 1 0.25 2 8 3 12
       0.25 16 48 100 1r 200 1r 200 300 900 3000 9000 30000 30000
       0.25 16 5 0
           48 100      200      200 300 900      9000 30000 30000
           12 25      50      50 75 250      2500 7500 7500

```

```

esplt:p 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=4 pos=0 0 0 ext=d1 erg=1.25 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c d1=uniform axial source distribution p(x)=c
si1 191.69 195.50
spl 0.0 1.0
c d3=uniform radial source distribution p(x)=cx
si3 0.0 61.0
c d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc12 surface tallies, ANSI/ANS 6.1.1-1977 (mrem/hr)
fl2:p 15 1611 16 17
fs12 -31 -311 -312 -313 -314 -315 -316 -317 -4
sd12 3.631+4 1.679+4 1.759+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
e12 0.045 0.1 0.45 0.70 1.0 1.5
de12 0.010 0.030 0.050 0.070 0.100 0.150
0.200 0.250 0.300 0.350 0.400 0.450
0.500 0.550 0.600 0.650 0.700 0.800
1.000 1.400 1.800 2.200 2.600 2.800
3.25 3.75 4.25 4.75 5.0 5.25
5.75 6.25 6.75 7.5 9.0 11.0
13.0 15.0
df12 3.96-3 5.82-4 2.90-4 2.58-4 2.83-4 3.79-4
5.01-4 6.31-4 7.59-4 8.78-4 9.85-4 1.08-3
1.17-3 1.27-3 1.36-3 1.44-3 1.52-3 1.68-3
1.98-3 2.51-3 2.99-3 3.42-3 3.82-3 4.01-3
4.41-3 4.83-3 5.23-3 5.60-3 5.80-3 6.01-3
6.37-3 6.74-3 7.11-3 7.66-3 8.77-3 1.03-2
1.18-2 1.33-2
c multiplier=(121 Ci Co60/fuel assembly) (21 assemblies per WP)
c (3.7E10 dis/s per Ci) (2 photons/dis)
c =1.88+14 photons/s per WP
fm12 1.88+14
fq0 e s
ctme 720.0

```


VI.4 LISTING OF INPUT FILE DRIFTNB

```

MGDS in-drift dose calculation--neutrons from active fuel
c   neutron from 21 PWR fuel assemblies in waste package
c   48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c   drift inside diameter=5.1 m, 0.2 m concrete liner
c   18" concrete shadow shield only (no steel door)
c   *****
1   1 -3.0431  -1 20 -7      $active fuel (homogenized)
2   5 -0.001225 -1 7 -8      $lower plenum (air assumed)
3   5 -0.001225 -1 21 -20    $upper plenum (air assumed)
4   8 -2.5402  -1 8 -9      $lower end fittings
5   7 -2.4701  -1 22 -21    $upper end fittings
51  4 -8.0038   1 -111 221 -91 $radial WP liner
52  5 -0.001225 111 -112 221 -91 $radial wp gap
53  5 -0.001225 -1 9 -91     $bottom wp gap
54  5 -0.001225 -1 221 -22   $top wp gap
6   2 -8.14     112 -2 221 -91 $WP inner barrier, side
7   2 -8.14     -2 91 -10     $WP inner barrier, bottom
8   2 -8.14     -2 23 -221    $WP inner barrier, top
9   3 -7.832    2 -211 23 -10  $WP outer barrier, side
91  3 -7.832    211 -3 23 -10  $WP outer barrier, side
10  3 -7.832    -3 10 -101     $WP outer barrier, bottom
101 3 -7.832    -3 101 -11     $WP outer barrier, bottom
11  3 -7.832    -3 24 -23     $WP outer barrier, top
12  5 -0.001225 3 -4 24 -11    $radial space between WP & drift wall
13  5 -0.001225 -4 11 -12     $4m from WP center
1311 9 -2.35    -31 12 -1311   $1st 0.75 ft concrete shield
1312 5 -0.001225 31 -4 12 -1311 $air adjacent to cell 1311
1315 9 -2.35    -31 1311 -1312 $2nd 0.75 ft concrete shield
1316 5 -0.001225 31 -4 1311 -1312 $air adjacent to cell 1315
14  5 -0.001225 -4 1312 -13   $5m from WP center
15  5 -0.001225 -4 13 -14     $7m from WP center
16  5 -0.001225 -4 14 -15     $13m from WP center
1611 5 -0.001225 -4 15 -1611   $1" drift isolation door (air)
17  5 -0.001225 -4 1611 -16   $24m from WP center
18  5 -0.001225 -4 16 -17     $35m from WP center
19  5 -0.001225 -4 17 -18     $50m from WP center
21  5 -0.001225 -4 25 -24     $rear end air cell
22  9 -2.35     4 -5 25 -11    $first 20 cm drift wall (concrete)
23  6 -2.22     5 -6 25 -11    $second 15 cm drift wall (tuff)
24  0           6:-25:18      $zero importance cell
131 9 -2.35     4 -5 11 -12    $20 cm conc. liner, next to cell 13
1313 9 -2.35     4 -5 12 -1311  $20 cm liner next to #1311 & #1312
1317 9 -2.35     4 -5 1311 -1312 $20 cm liner next to #1315 & #1316
141 9 -2.35     4 -5 1312 -13   $20 cm conc. liner, next to cell 14
151 9 -2.35     4 -5 13 -14     $20 cm conc. liner, next to cell 15
161 9 -2.35     4 -5 14 -15     $20 cm conc. liner, next to cell 16
171 9 -2.35     4 -5 15 -16     $20 cm conc. liner, next to cell 17
181 9 -2.35     4 -5 16 -17     $20 cm conc. liner, next to cell 18
191 9 -2.35     4 -5 17 -18     $20 cm conc. liner, next to cell 19
132 6 -2.22     5 -6 11 -12    $15 cm into tuff, next to cell 13
1314 6 -2.22     5 -6 12 -1311  $15 cm into tuff next to #1311 & #1312
1318 6 -2.22     5 -6 1311 -1312 $15 cm into tuff next to #1315 & #1316
142 6 -2.22     5 -6 1312 -13   $15 cm into tuff, next to cell 14
152 6 -2.22     5 -6 13 -14     $15 cm into tuff, next to cell 15

```

162	6 -2.22	5 -6 14 -15	\$15 cm into tuff, next to cell 16
172	6 -2.22	5 -6 15 -16	\$15 cm into tuff, next to cell 17
182	6 -2.22	5 -6 16 -17	\$15 cm into tuff, next to cell 18
192	6 -2.22	5 -6 17 -18	\$15 cm into tuff, next to cell 19
1	cx 61.0		\$fuel radial surface
111	cx 62.68		\$radial wp liner surface
112	cx 69.45		\$radial gap surface
2	cx 71.45		\$WP inner barrier radial surf.
211	cx 76.45		\$WP outer barrier radial, midpoint
3	cx 81.45		\$WP outer barrier radial surf.
31	cx 107.5		\$concrete shadow shield radius
4	cx 255.0		\$drift inside radius
5	cx 275.0		\$20 cm concrete liner
6	cx 290.0		\$15 cm into tuff
7	px 180.09		\$bottom of active fuel
8	px 191.69		\$top of lower plenum
9	px 195.50		\$top of lower end fittings
91	px 202.49		\$top of lower WP gap
10	px 204.99		\$bottom of inner barrier
101	px 210.99		\$midpoint, WP outer barrier, bottom
11	px 215.99		\$bottom of outer barrier
12	px 400.0		\$4m from WP center (2m from end)
1311	px 422.86		\$1st 0.75 ft concrete shield
1312	px 445.72		\$2nd 0.75 ft concrete shield
13	px 500.0		\$5m from wp center
14	px 700.0		\$7m from wp center
15	px 1300.0		\$13m from ***wp center***
1611	px 1302.54		\$1" isolation door
16	px 2400.0		\$24m from WP center
17	px 3500.0		\$35m from wp center
18	px 5000.0		\$50m from wp center
20	px -180.09		\$top of active fuel
21	px -202.80		\$bottom of upper plenum
22	px -220.11		\$top of fuel assembly
221	px -256.01		\$top of upper WP gap
23	px -258.51		\$top of inner barrier
24	px -269.51		\$top of outer barrier
25	px -500.0		\$rear boundary surface
311	cx 130.0		
312	cx 150.0		
313	cx 170.0		
314	cx 190.0		
315	cx 210.0		
316	cx 230.0		
317	cx 245.0		
c	active fuel region (smear density 3.0431 g/cc)		
m1	7014	-0.00005	
	8016	-0.10540	
	24000	-0.00017	
	26000	-0.00035	
	40000	-0.17140	
	50000	-0.00244	
	92235	-0.00523	
	92236	-0.00419	
	92238	-0.70508	

	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007

```

13027  -0.00124
14000  -0.00796
15031  -0.00034
16032  -0.00025
22000  -0.00223
24000  -0.18994
25055  -0.01548
26000  -0.55131
27059  -0.00150
28000  -0.20504
29000  -0.00037
41093  -0.00636
42000  -0.00757
73181  -0.00636
c      lower end fittings (2.5402 g/cc)
m8     6000  -0.00072
       7014  -0.00388
       8016  -0.00018
       14000 -0.00904
       15031 -0.00041
       16032 -0.00027
       24000 -0.17193
       25055 -0.01809
       26000 -0.60834
       27059 -0.00181
       28000 -0.09044
       40000 -0.09356
       50000 -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9     8016  -0.4983
       1001  -0.0055
       11023 -0.0170
       12000 -0.0026
       13027 -0.0455
       14000 -0.3157
       16032 -0.0013
       19000 -0.0191
       20000 -0.0826
       26000 -0.0123
c      SS304 for drift isolation door
m10    6000  -0.00080
       7014  -0.00100
       14000 -0.00750
       15031 -0.00045
       16032 -0.00030
       24000 -0.19000
       25055 -0.02000
       26000 -0.68745
       28000 -0.09250
mode n
c      importance set by cell to move particle tracks towards detectors
c      (by increasing impotance as particles move closer to detectors)
imp:n  1 1 1 1 0.5 1 1 1 0.5 1 1 0.25 2 4 2 4
       0.25 8 8 10 1r 20 1r 20 20 30 30 60 60 60
       0.25 8 2 0
           8 10    20    20 20 30    60 60 60
           2 2.5    5     5 5 7.5 15 15 15

```

```

esplt:n 0.5 1.0-3 0.5 1.0-4 0.5 1.0-5 0.25 1.0-6
phys:n 25.0 1.0-7
sdef cel=1 pos=0 0 0 ext=d1 erg=d2 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c d1=uniform axial source distribution with biasing towards detectors
si1 -180.09 -90.045 0.0 90.045 180.09
sp1 0.0 0.25 0.25 0.25 0.25
sb1 0.0 0.10 0.15 0.25 0.50
c d2=neutron energy spectrum
si2 0.10 0.40 0.9 1.4 1.85 3.0 6.43 20.0
sp2 0.0 0.0376 0.1924 0.1767 0.1310 0.2341 0.2099 0.0183
c d3=uniform radial source distribution (unbiased)
si3 0.0 61.0
c d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc12 surface tallies, ANSI/ANS 6.1.1-1977 (mrem/hr)
f12:n 15 1611 16 17
fs12 -31 -311 -312 -313 -314 -315 -316 -317 -4
sdl2 3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
e12 6.25-7 0.1 1.0 25.0
de12 2.5-8 1.0-7 1.0-6 1.0-5 1.0-4
1.0-3 1.0-2 0.1 0.5 1.0
2.5 5.0 7.0 10.0 14.0
20.0
df12 3.67-3 3.67-3 4.46-3 4.54-3 4.18-3
3.76-3 3.56-3 2.17-2 9.26-2 1.32-1
1.25-1 1.56-1 1.47-1 1.47-1 2.08-1
2.27-1
c multiplier=6.567+9 neutrons/s per WP for 21 PWR fuel assemblies
fm12 6.567+9
fq0 e s
ctme 400.0

```

VI.5 LISTING OF INPUT FILE DRIFTGB

```

MGDS in-drift dose calculation--primary gammas from active fuel
c   primary gammas from 21 PWR fuel assemblies in waste package
c   48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c   drift inside diameter=5.1 m, 0.2 m concrete liner
c   18" concrete shadow shield only
c   *****
1   1 -3.0431   -1 20 -7          $active fuel (homogenized)
2   5 -0.001225 -1 7 -8          $lower plenum (air assumed)
3   5 -0.001225 -1 21 -20        $upper plenum (air assumed)
4   8 -2.5402   -1 8 -9          $lower end fittings
5   7 -2.4701   -1 22 -21        $upper end fittings
51  4 -8.0038    1 -111 221 -91   $radial WP liner
52  5 -0.001225 111 -112 221 -91  $radial wp gap
53  5 -0.001225 -1 9 -91         $bottom wp gap
54  5 -0.001225 -1 221 -22       $top wp gap
6   2 -8.14      112 -2 221 -91  $WP inner barrier, side
7   2 -8.14      -2 91 -10       $WP inner barrier, bottom
8   2 -8.14      -2 23 -221      $WP inner barrier, top
9   3 -7.832     2 -211 23 -10    $WP outer barrier, side
91  3 -7.832     211 -3 23 -10    $WP outer barrier, side
10  3 -7.832      -3 10 -101      $WP outer barrier, bottom
101 3 -7.832      -3 101 -11      $WP outer barrier, bottom
11  3 -7.832      -3 24 -23       $WP outer barrier, top
12  5 -0.001225 3 -4 24 -11      $radial space between WP & drift wall
13  5 -0.001225 -4 11 -12        $4m from WP center
1311 9 -2.35      -31 12 -1311   $1st 0.75 ft concrete shield
1312 5 -0.001225 31 -4 12 -1311  $air adjacent to cell 1311
1315 9 -2.35      -31 1311 -1312 $2nd 0.75 ft concrete shield
1316 5 -0.001225 31 -4 1311 -1312 $air adjacent to cell 1315
14  5 -0.001225 -4 1312 -13      $5m from WP center
15  5 -0.001225 -4 13 -14        $7m from WP center
16  5 -0.001225 -4 14 -15        $13m from WP center
1611 5 -0.001225 -4 15 -1611     $1" drift isolation door (air)
17  5 -0.001225 -4 1611 -16      $24m from WP center
18  5 -0.001225 -4 16 -17        $35m from WP center
19  5 -0.001225 -4 17 -18        $50m from WP center
21  5 -0.001225 -4 25 -24        $rear end air cell
22  9 -2.35      4 -5 25 -11     $first 20 cm drift wall (concrete)
23  6 -2.22      5 -6 25 -11     $second 15 cm drift wall (tuff)
24  0             6:-25:18       $zero importance cell
131 9 -2.35      4 -5 11 -12     $20 cm conc. liner, next to cell 13
1313 9 -2.35      4 -5 12 -1311  $20 cm liner next to #1311 & #1312
1317 9 -2.35      4 -5 1311 -1312 $20 cm liner next to #1315 & #1316
141 9 -2.35      4 -5 1312 -13   $20 cm conc. liner, next to cell 14
151 9 -2.35      4 -5 13 -14     $20 cm conc. liner, next to cell 15
161 9 -2.35      4 -5 14 -15     $20 cm conc. liner, next to cell 16
171 9 -2.35      4 -5 15 -16     $20 cm conc. liner, next to cell 17
181 9 -2.35      4 -5 16 -17     $20 cm conc. liner, next to cell 18
191 9 -2.35      4 -5 17 -18     $20 cm conc. liner, next to cell 19
132 6 -2.22      5 -6 11 -12     $15 cm into tuff, next to cell 13
1314 6 -2.22      5 -6 12 -1311  $15 cm into tuff next to #1311 & #1312
1318 6 -2.22      5 -6 1311 -1312 $15 cm into tuff next to #1315 & #1316
142 6 -2.22      5 -6 1312 -13   $15 cm into tuff, next to cell 14
152 6 -2.22      5 -6 13 -14     $15 cm into tuff, next to cell 15

```

162	6 -2.22	5 -6 14 -15	\$15 cm into tuff, next to cell 16
172	6 -2.22	5 -6 15 -16	\$15 cm into tuff, next to cell 17
182	6 -2.22	5 -6 16 -17	\$15 cm into tuff, next to cell 18
192	6 -2.22	5 -6 17 -18	\$15 cm into tuff, next to cell 19
1	cx 61.0		\$fuel radial surface
111	cx 62.68		\$radial wp liner surface
112	cx 69.45		\$radial gap surface
2	cx 71.45		\$WP inner barrier radial surf.
211	cx 76.45		\$WP outer barrier radial, midpoint
3	cx 81.45		\$WP outer barrier radial surf.
31	cx 107.5		\$concrete shadow shield radius
4	cx 255.0		\$drift inside radius
5	cx 275.0		\$20 cm concrete liner
6	cx 290.0		\$15 cm into tuff
7	px 180.09		\$bottom of active fuel
8	px 191.69		\$top of lower plenum
9	px 195.50		\$top of lower end fittings
91	px 202.49		\$top of lower WP gap
10	px 204.99		\$bottom of inner barrier
101	px 210.99		\$midpoint, WP outer barrier, bottom
11	px 215.99		\$bottom of outer barrier
12	px 400.0		\$4m from WP center(2m from end)
1311	px 422.86		\$1st 0.75 ft concrete shield
1312	px 445.72		\$2nd 0.75 ft concrete shield
13	px 500.0		\$5m from wp center
14	px 700.0		\$7m from wp center
15	px 1300.0		\$13m from ***wp center***
1611	px 1302.54		\$1" isolation door
16	px 2400.0		\$24m from WP center
17	px 3500.0		\$35m from wp center
18	px 5000.0		\$50m from wp center
20	px -180.09		\$top of active fuel
21	px -202.80		\$bottom of upper plenum
22	px -220.11		\$top of fuel assembly
221	px -256.01		\$top of upper WP gap
23	px -258.51		\$top of inner barrier
24	px -269.51		\$top of outer barrier
25	px -500.0		\$rear boundary surface
311	cx 130.0		
312	cx 150.0		
313	cx 170.0		
314	cx 190.0		
315	cx 210.0		
316	cx 230.0		
317	cx 245.0		
c	active fuel region (smear density 3.0431 g/cc)		
m1	7014	-0.00005	
	8016	-0.10540	
	24000	-0.00017	
	26000	-0.00035	
	40000	-0.17140	
	50000	-0.00244	
	92235	-0.00523	
	92236	-0.00419	
	92238	-0.70508	

	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007


```

13027  -0.00124
14000  -0.00796
15031  -0.00034
16032  -0.00025
22000  -0.00223
24000  -0.18994
25055  -0.01548
26000  -0.55131
27059  -0.00150
28000  -0.20504
29000  -0.00037
41093  -0.00636
42000  -0.00757
73181  -0.00636
c      lower end fittings (2.5402 g/cc)
m8     6000  -0.00072
       7014  -0.00388
       8016  -0.00018
14000  -0.00904
15031  -0.00041
16032  -0.00027
24000  -0.17193
25055  -0.01809
26000  -0.60834
27059  -0.00181
28000  -0.09044
40000  -0.09356
50000  -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9     8016  -0.4983
       1001  -0.0055
11023  -0.0170
12000  -0.0026
13027  -0.0455
14000  -0.3157
16032  -0.0013
19000  -0.0191
20000  -0.0826
26000  -0.0123
c      SS304 for drift isolation door
m10    6000  -0.00080
       7014  -0.00100
14000  -0.00750
15031  -0.00045
16032  -0.00030
24000  -0.19000
25055  -0.02000
26000  -0.68745
28000  -0.09250
mode p
c      importance set by cell to move particle tracks towards detectors
c      (by increasing impotance as particles move closer to detectors)
imp:p  1 1 1 1 0.5 1 1 1 0.5 1 1 0.25 2 8 3 12
       0.25 16 48 100 1r 200 1r 200 250 750 750 3000 12000 12000
       0.25 16 5 0
           48 100      200      200 250 750      3000 12000 12000
           12 25      50      50 65 200      750 3000 3000

```

```

esplt:p 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=1 pos=0 0 0 ext=d1 erg=d2 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c d1=uniform axial source distribution with biasing towards detectors
si1 -180.09 -90.045 0.0 90.045 180.09
sp1 0.0 0.25 0.25 0.25 0.25
sb1 0.0 0.10 0.15 0.20 0.55
c d2=photon energy spectrum for contributing groups only
c with biasing for high energy photons
si2 0.45 0.70 1.0 1.5 2.0 2.5 3.0
sp2 0.0 0.8485 0.08260 6.806-02 8.628-04 1.360-05 1.050-06
sb2 0.0 0.17 0.15 0.65 0.03 0.001 0.0002
c d3=uniform radial source distribution (unbiased)
si3 0.0 61.0
c d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fc12 surface tallies, ANSI/ANS 6.1.1-1977 (mrem/hr)
f12:p 15 1611 16 17
fs12 -31 -311 -312 -313 -314 -315 -316 -317 -4
sd12 3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
e12 0.045 0.1 0.45 0.70 1.0 1.5 2.0 2.5 3.0
de12 0.010 0.030 0.050 0.070 0.100 0.150
0.200 0.250 0.300 0.350 0.400 0.450
0.500 0.550 0.600 0.650 0.700 0.800
1.000 1.400 1.800 2.200 2.600 2.800
3.25 3.75 4.25 4.75 5.0 5.25
5.75 6.25 6.75 7.5 9.0 11.0
13.0 15.0
df12 3.96-3 5.82-4 2.90-4 2.58-4 2.83-4 3.79-4
5.01-4 6.31-4 7.59-4 8.78-4 9.85-4 1.08-3
1.17-3 1.27-3 1.36-3 1.44-3 1.52-3 1.68-3
1.98-3 2.51-3 2.99-3 3.42-3 3.82-3 4.01-3
4.41-3 4.83-3 5.23-3 5.60-3 5.80-3 6.01-3
6.37-3 6.74-3 7.11-3 7.66-3 8.77-3 1.03-2
1.18-2 1.33-2
c multiplier=5.51+16 photons/s per WP for 21 PWR fuel assemblies
c and for 0.45 - 3.0 gammas, which contribute to dose results
fm12 5.51+16
fq0 e s
ctme 3600.0

```

VI.6 LISTING OF INPUT FILE DRIFTCOB

MGDS in-drift dose calculation--Co-60 gammas from lower end fittings

```

c      Co-60 gammas from 21 PWR fuel assemblies in waste package
c      48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c      drift inside diameter=5.1 m, 0.2 m concrete liner
c      18" concrete shadow shield only (no steel door)
c      *****
1      1 -3.0431   -1 20 -7          $active fuel (homogenized)
2      5 -0.001225 -1 7 -8          $lower plenum (air assumed)
3      5 -0.001225 -1 21 -20        $upper plenum (air assumed)
4      8 -2.5402   -1 8 -9          $lower end fittings
5      7 -2.4701   -1 22 -21        $upper end fittings
51     4 -8.0038    1 -111 221 -91   $radial WP liner
52     5 -0.001225 111 -112 221 -91  $radial wp gap
53     5 -0.001225 -1 9 -91         $bottom wp gap
54     5 -0.001225 -1 221 -22       $top wp gap
6      2 -8.14      112 -2 221 -91  $WP inner barrier, side
7      2 -8.14      -2 91 -10       $WP inner barrier, bottom
8      2 -8.14      -2 23 -221      $WP inner barrier, top
9      3 -7.832     2 -211 23 -10   $WP outer barrier, side
91     3 -7.832     211 -3 23 -10   $WP outer barrier, side
10     3 -7.832     -3 10 -101      $WP outer barrier, bottom
101    3 -7.832     -3 101 -11      $WP outer barrier, bottom
11     3 -7.832     -3 24 -23       $WP outer barrier, top
12     5 -0.001225 3 -4 24 -11      $radial space between WP & drift wall
13     5 -0.001225 -4 11 -12       $4m from WP center
1311   9 -2.35      -31 12 -1311    $1st 0.75 ft concrete shield
1312   5 -0.001225 31 -4 12 -1311  $air adjacent to cell 1311
1315   9 -2.35      -31 1311 -1312  $2nd 0.75 ft concrete shield
1316   5 -0.001225 31 -4 1311 -1312 $air adjacent to cell 1315
14     5 -0.001225 -4 1312 -13     $5m from WP center
15     5 -0.001225 -4 13 -14       $7m from WP center
16     5 -0.001225 -4 14 -15       $13m from WP center
1611   5 -0.001225 -4 15 -1611     $1" drift isolation door (air)
17     5 -0.001225 -4 1611 -16     $24m from WP center
18     5 -0.001225 -4 16 -17       $35m from WP center
19     5 -0.001225 -4 17 -18       $50m from WP center
21     5 -0.001225 -4 25 -24       $rear end air cell
22     9 -2.35      4 -5 25 -11     $first 20 cm drift wall (concrete)
23     6 -2.22      5 -6 25 -11     $second 15 cm drift wall (tuff)
24     0             6:-25:18       $zero importance cell
131    9 -2.35      4 -5 11 -12     $20 cm conc. liner, next to cell 13
1313   9 -2.35      4 -5 12 -1311   $20 cm liner next to #1311 & #1312
1317   9 -2.35      4 -5 1311 -1312 $20 cm liner next to #1315 & #1316
141    9 -2.35      4 -5 1312 -13   $20 cm conc. liner, next to cell 14
151    9 -2.35      4 -5 13 -14     $20 cm conc. liner, next to cell 15
161    9 -2.35      4 -5 14 -15     $20 cm conc. liner, next to cell 16
171    9 -2.35      4 -5 15 -16     $20 cm conc. liner, next to cell 17
181    9 -2.35      4 -5 16 -17     $20 cm conc. liner, next to cell 18
191    9 -2.35      4 -5 17 -18     $20 cm conc. liner, next to cell 19
132    6 -2.22      5 -6 11 -12     $15 cm into tuff, next to cell 13
1314   6 -2.22      5 -6 12 -1311   $15 cm into tuff next to #1311 & #1312
1318   6 -2.22      5 -6 1311 -1312 $15 cm into tuff next to #1315 & #1316
142    6 -2.22      5 -6 1312 -13   $15 cm into tuff, next to cell 14
152    6 -2.22      5 -6 13 -14     $15 cm into tuff, next to cell 15

```

Title: MGDS Subsurface Radiation Shielding Analysis

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162	6 -2.22	5 -6 14 -15	\$15 cm into tuff, next to cell 16
172	6 -2.22	5 -6 15 -16	\$15 cm into tuff, next to cell 17
182	6 -2.22	5 -6 16 -17	\$15 cm into tuff, next to cell 18
192	6 -2.22	5 -6 17 -18	\$15 cm into tuff, next to cell 19
1	cx 61.0		\$fuel radial surface
111	cx 62.68		\$radial wp liner surface
112	cx 69.45		\$radial gap surface
2	cx 71.45		\$WP inner barrier radial surf.
211	cx 76.45		\$WP outer barrier radial, midpoint
3	cx 81.45		\$WP outer barrier radial surf.
31	cx 107.5		\$concrete shadow shield radius
4	cx 255.0		\$drift inside radius
5	cx 275.0		\$20 cm concrete liner
6	cx 290.0		\$15 cm into tuff
7	px 180.09		\$bottom of active fuel
8	px 191.69		\$top of lower plenum
9	px 195.50		\$top of lower end fittings
91	px 202.49		\$top of lower WP gap
10	px 204.99		\$bottom of inner barrier
101	px 210.99		\$midpoint, WP outer barrier, bottom
11	px 215.99		\$bottom of outer barrier
12	px 400.0		\$4m from WP center (2m from end)
1311	px 422.86		\$1st 0.75 ft concrete shield
1312	px 445.72		\$2nd 0.75 ft concrete shield
13	px 500.0		\$5m from wp center
14	px 700.0		\$7m from wp center
15	px 1300.0		\$13m from ***wp center***
1611	px 1302.54		\$1" isolation door
16	px 2400.0		\$24m from WP center
17	px 3500.0		\$35m from wp center
18	px 5000.0		\$50m from wp center
20	px -180.09		\$top of active fuel
21	px -202.80		\$bottom of upper plenum
22	px -220.11		\$top of fuel assembly
221	px -256.01		\$top of upper WP gap
23	px -258.51		\$top of inner barrier
24	px -269.51		\$top of outer barrier
25	px -500.0		\$rear boundary surface
311	cx 130.0		
312	cx 150.0		
313	cx 170.0		
314	cx 190.0		
315	cx 210.0		
316	cx 230.0		
317	cx 245.0		
c	active fuel region (smear density 3.0431 g/cc)		
m1	7014	-0.00005	
	8016	-0.10540	
	24000	-0.00017	
	26000	-0.00035	
	40000	-0.17140	
	50000	-0.00244	
	92235	-0.00523	
	92236	-0.00419	
	92238	-0.70508	

	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner (SS316B6A + A825, 8.0038 g/cc)	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	upper end fittings (2.4701 g/cc)	
m7	6000	-0.00070
	7014	-0.00328
	8016	-0.00007

```

13027  -0.00124
14000  -0.00796
15031  -0.00034
16032  -0.00025
22000  -0.00223
24000  -0.18994
25055  -0.01548
26000  -0.55131
27059  -0.00150
28000  -0.20504
29000  -0.00037
41093  -0.00636
42000  -0.00757
73181  -0.00636
c      lower end fittings (2.5402 g/cc)
m8     6000  -0.00072
       7014  -0.00388
       8016  -0.00018
14000  -0.00904
15031  -0.00041
16032  -0.00027
24000  -0.17193
25055  -0.01809
26000  -0.60834
27059  -0.00181
28000  -0.09044
40000  -0.09356
50000  -0.00133
c      ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)
m9     8016  -0.4983
       1001  -0.0055
11023  -0.0170
12000  -0.0026
13027  -0.0455
14000  -0.3157
16032  -0.0013
19000  -0.0191
20000  -0.0826
26000  -0.0123
c      SS304 for drift isolation door
m10    6000  -0.00080
       7014  -0.00100
14000  -0.00750
15031  -0.00045
16032  -0.00030
24000  -0.19000
25055  -0.02000
26000  -0.68745
28000  -0.09250
mode   p
c      importance set by cell to move particle tracks towards detectors
c      (by increasing impotance as particles move closer to detectors)
imp:p  1 1 1 1 0.5 1 1 1 0.5 1 1 0.25 2 8 3 12
       0.25 16 48 100 1r 200 1r 200 300 900 900 3000 12000 12000
       0.25 16 5 0
           48 100      200      200 300 900      3000 12000 12000
           12 25      50      50 75 250      750 3000 3000

```

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```
esplt:p 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=4 pos=0 0 0 ext=d1 erg=1.25 rad=d3 axs=1 0 0 dir=d4 vec=1 0 0
c d1=uniform axial source distribution p(x)=c
si1 191.69 195.50
sp1 0.0 1.0
c d3=uniform radial source distribution p(x)=cx
si3 0.0 61.0
c d4=isotropic source with biasing towards detectors
si4 -1.0 0.0 1.0
sp4 0.0 0.5 0.5
sb4 0.0 0.25 0.75
fcl2 surface tallies, ANSI/ANS 6.1.1-1977 (mrem/hr)
fl2:p 15 1611 16 17
fs12 -31 -311 -312 -313 -314 -315 -316 -317 -4
sd12 3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
      2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
      3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
      2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
      3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
      2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
      3.631+4 1.679+4 1.759+4 2.011+4 2.262+4
      2.513+4 2.765+4 2.238+4 1.571+4 5.993+4
el2 0.045 0.1 0.45 0.70 1.0 1.5
del2 0.010 0.030 0.050 0.070 0.100 0.150
      0.200 0.250 0.300 0.350 0.400 0.450
      0.500 0.550 0.600 0.650 0.700 0.800
      1.000 1.400 1.800 2.200 2.600 2.800
      3.25 3.75 4.25 4.75 5.0 5.25
      5.75 6.25 6.75 7.5 9.0 11.0
      13.0 15.0
df12 3.96-3 5.82-4 2.90-4 2.58-4 2.83-4 3.79-4
      5.01-4 6.31-4 7.59-4 8.78-4 9.85-4 1.08-3
      1.17-3 1.27-3 1.36-3 1.44-3 1.52-3 1.68-3
      1.98-3 2.51-3 2.99-3 3.42-3 3.82-3 4.01-3
      4.41-3 4.83-3 5.23-3 5.60-3 5.80-3 6.01-3
      6.37-3 6.74-3 7.11-3 7.66-3 8.77-3 1.03-2
      1.18-2 1.33-2
c multiplier=(121 Ci Co60/fuel assembly) (21 assemblies per WP)
c (3.7E10 dis/s per Ci) (2 photons/dis)
c =1.88+14 photons/s per WP
fml2 1.88+14
fq0 e s
ctme 400.0
```

ATTACHMENT VII
INPUT AND OUTPUT FILES FOR WP TRANSPORTER SHIELDING ANALYSIS

This attachment gives the computer input and output file names of the MCNP4A and PATH runs associated with the waste package transporter shielding analysis (Section 7.5), and provides a complete listing of each input file. All the input and output files are stored on electronic media as Ref. 5.29. This reference is submitted with this document as part of the Design Analysis Records Package. This attachment covers the following MCNP4A and PATH cases:

<u>Case Description</u>	<u>Input file</u>	<u>Output File</u>
MCNP4A, radial analysis, neutron & secondary gammas	TSPRT01N	TSPRT01N.OUT
MCNP4A, radial analysis, primary fuel gammas	TSPRT01G	TSPRT01G.OUT
PATH, radial analysis, primary fuel gammas	TSPRT01.INP	TSPRT01.OUT
PATH, axial analysis, primary fuel gammas	TSPRT02A.INP	TSPRT02A.OUT
PATH, axial analysis, LEF Co-60 gammas	TSPRT03A.INP	TSPRT03A.OUT

VII.1 LISTING OF INPUT FILE TSPRT01N (MCNP4A)

```

MGDS waste package transporter radial shielding analysis--n + second. gamma
c   neutrons from 21 PWR fuel assemblies in waste package
c   48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c   infinite axial height model
c   maximum waste package od=1.85 m
c   transporter id=2.45 m, 5 mm ss316L+6" carbon steel+4" N-S+5 mm ss316L
c   main access drift diameter=7.62 m
c   concrete lining=0.2 m
c   *****
c   fuel region (homogenized, divided in 5 cells for source biasing)
101  1 -3.0431  -101  12 -13
102  1 -3.0431  101 -102 12 -13
103  1 -3.0431  102 -103 12 -13
104  1 -3.0431  103 -104 12 -13
105  1 -3.0431  104 -105 12 -13
c   liner for fuel basket on the outside
1   4 -8.0038  105 -1  12 -13
c   WP radial gap
111  5 -0.001225 1  -111 12 -13
c   WP inner barrier (2 cm)
2   2 -8.14    111 -2  12 -13
c   WP outer barriers (10 cm total)
3   3 -7.832   2  -3  12 -13
4   3 -7.832   3  -4  12 -13
c   air gap between WP and transporter
401  5 -0.001225 4 -401  12 -13
c   ss316L inner clad for transporter (0.5 cm)
402  8 -7.9497  401 -402 12 -13
c   transporter shield (6"=15.24 cm total)
5   3 -7.832   402 -5  12 -13
6   3 -7.832   5   -6  12 -13
7   3 -7.832   6   -7  12 -13
c   B-POLY neutron shield (4"=10.16 cm total, 4 cells)
701  7 -0.92    7  -701 12 -13
702  7 -0.92    701 -702 12 -13
703  7 -0.92    702 -703 12 -13
704  7 -0.92    703 -704 12 -13
c   SS316L outer clad for transporter (0.5 cm)
705  8 -7.9497  704 -705 12 -13
c   air space between transporter and drift
8   5 -0.001225 705 -8  12 -13
9   5 -0.001225 8   -9  12 -13
c   concrete lining (20 cm)
10  9 -2.235    9  -10 12 -13
c   tuff medium (15 cm modeled)
11  6 -2.22     10 -11 12 -13
c   zero importance cell
12  0          11:-12:13

101  cz 20.0      $1st fuel cell
102  cz 35.0      $2nd fuel cell
103  cz 45.0      $3rd fuel cell
104  cz 54.0      $4th fuel cell
105  cz 61.0      $5th fuel cell

```

1	cz 62.68	\$WP liner
111	cz 69.45	\$radial gap
2	cz 71.45	\$inner barrier
3	cz 76.45	\$1st 5 cm outer barrier
4	cz 81.45	\$2nd 5 cm outer barrier
401	cz 122.5	\$transporter ID
402	cz 123.0	\$0.5 cm ss316L inner clad
5	cz 128.08	\$1st 2" carbon steel
6	cz 133.16	\$2nd 2" carbon steel
7	cz 138.24	\$3rd 2" carbon steel
701	cz 140.78	\$1st 1" b-poly
702	cz 143.32	\$2nd 1" b-poly
703	cz 145.86	\$3rd 1" b-poly
704	cz 148.40	\$4th 1" b-poly
705	cz 148.90	\$0.5 cm ss316L outer clad
8	cz 250.0	\$~1 m from transporter surface
9	cz 361.0	\$concrete lining inner surface
10	cz 381.0	\$drift wall surface
11	cz 395.0	\$15 cm into tuff
*12	pz 0.0	\$model lower axial boundary (reflective)
*13	pz 10.0	\$model upper axial boundary (reflective)
c	active fuel region (smear density 3.0431 g/cc)	
m1	7014	-0.00005
	8016	-0.10540
	24000	-0.00017
	26000	-0.00035
	40000	-0.17140
	50000	-0.00244
	92235	-0.00523
	92236	-0.00419
	92238	-0.70508
	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner [SS316B6A (vf=0.3448)+A825 (vf=0.6552), 8.0038 g/cc]	
m4	5010	-0.00077
	5011	-0.00351

	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	dry tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	neutron shield - 1.5% B-Polypropylene (0.92 g/cc per GA-4 cask)	
m7	1001	-0.1256
	5010	-0.00278
	5011	-0.01223
	6000	-0.8594
c	ss316L for transporter clad (7.9497 g/cc)	
m8	6000	-0.00030
	7014	-0.00100
	14000	-0.00750
	15031	-0.00045
	16032	-0.00030
	24000	-0.17000
	25055	-0.02000
	26000	-0.65545
	28000	-0.12000
	42000	-0.02500
c	ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)	
m9	8016	-0.4983
	1001	-0.0055
	11023	-0.0170
	12000	-0.0026
	13027	-0.0455
	14000	-0.3157
	16032	-0.0013
	19000	-0.0191
	20000	-0.0826
	26000	-0.0123
mode	n	p
imp:n	1	7r 1.5 2 2 2r 3 4.5 7 14 40 120 3r 60 30 0

```

imp:p  0.1 0.2 0.4 0.6 1 2r
      2 4 16 2r 48 144 423 500 600 750 1000 3r 500 250 0
esplt:n 0.5 1.0-3 0.5 1.0-4 0.25 1.0-5 0.25 1.0-6
esplt:p 0.5 0.5 0.5 0.25 0.5 0.1
phys:n  25.0 1.0-7
phys:p  3.0
sdef    cel=d1 pos=0 0 5.0 rad=fcel d4 ext=d2 erg=d3 axs=0 0 1
c        uniform radial source distribution
si1      1 101      102      103      104      105
spl      0.1075 0.2217 0.2150 0.2395 0.2163
c        uniform axial distribution from 0 to 10 cm p(x)=c
si2      5.0
c        unbiased neutron source spectrum
si3      0.10 0.40 0.9 1.4 1.85 3.0 6.43 20.0
sp3      0.0 0.0376 0.1924 0.1767 0.1310 0.2341 0.2099 0.0183
c        Radii for 5 fuel cells
ds4      s 5 6 7 8 9
si5      0.0 20.0
si6      20.0 35.0
si7      35.0 45.0
si8      45.0 54.0
si9      54.0 61.0
fc2      surface neutron flux (n/cm**2-s)
f2:n     101 102 103 104 105 1 111 2 3 4 401 402 5 6 7
      701 702 703 704 705 8 9 10 11
e2       6.25-7 0.1 1.0 25.0
c        multiplier=(6.567+9 n/s for 21 PWR FA) (10/360.18 for 10 cm)
fm2      1.823+8
fc12     surface neutron dose rate--ANSI 6.1.1-1977 (mrem/hr)
f12:n    101 102 103 104 105 1 111 2 3 4 401 402 5 6 7
      701 702 703 704 705 8 9 10 11
e12      6.25-7 0.1 1.0 25.0
de12     2.5-8 1.0-7 1.0-6 1.0-5 1.0-4
      1.0-3 1.0-2 1.0-1 5.0-1 1.0
      2.5 5.0 7.0 10.0 14.0 20.0
df12     3.67-3 3.67-3 4.46-3 4.54-3 4.18-3
      3.76-3 3.56-3 2.17-2 9.26-2 1.32-1
      1.25-1 1.56-1 1.47-1 1.47-1 2.08-1 2.27-1
fm12     1.823+8
fc32     surface gamma dose rates--ANSI 6.1.1-1977 (mrem/hr)
e32      0.045 0.1 0.45 0.70 1.0 1.5 2.0 3.0 5.0 7.5 10.0 15.0
f32:p    101 102 103 104 105 1 111 2 3 4 401 402 5 6 7
      701 702 703 704 705 8 9 10 11
de32     0.010 0.030 0.050 0.070 0.100 0.150
      0.200 0.250 0.300 0.350 0.400 0.450
      0.500 0.550 0.600 0.650 0.700 0.800
      1.000 1.400 1.800 2.200 2.600 2.800
      3.25 3.75 4.25 4.75 5.0 5.25
      5.75 6.25 6.75 7.5 9.0 11.0
      13.0 15.0
df32     3.96-3 5.82-4 2.90-4 2.58-4 2.83-4 3.79-4
      5.01-4 6.31-4 7.59-4 8.78-4 9.85-4 1.08-3
      1.17-3 1.27-3 1.36-3 1.44-3 1.52-3 1.68-3
      1.98-3 2.51-3 2.99-3 3.42-3 3.82-3 4.01-3
      4.41-3 4.83-3 5.23-3 5.60-3 5.80-3 6.01-3
      6.37-3 6.74-3 7.11-3 7.66-3 8.77-3 1.03-2
      1.18-2 1.33-2

```

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fm32 1.823+8
fq0 e f
ctme 200.0

VII.2 LISTING OF INPUT FILE TSPRT01G (MCNP4A)

```

MGDS waste package transporter radial shielding analysis--fuel primary gamma
c primary gammas from 21 PWR fuel assemblies in waste package
c 48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c infinite axial height model
c maximum waste package od=1.85 m
c transporter id=2.45 m, 5 mm ss316L+6" carbon steel+4" N-S+5 mm ss316L
c main access drift diameter=7.62 m
c concrete lining=0.2 m
c *****
c fuel region (homogenized, divided in 5 cells for source biasing)
101 1 -3.0431 -101 12 -13
102 1 -3.0431 101 -102 12 -13
103 1 -3.0431 102 -103 12 -13
104 1 -3.0431 103 -104 12 -13
105 1 -3.0431 104 -105 12 -13
c liner for fuel basket on the outside
1 4 -8.0038 105 -1 12 -13
c WP radial gap
111 5 -0.001225 1 -111 12 -13
c WP inner barrier (2 cm)
2 2 -8.14 111 -2 12 -13
c WP outer barriers (10 cm total)
3 3 -7.832 2 -3 12 -13
4 3 -7.832 3 -4 12 -13
c air gap between WP and transporter
401 5 -0.001225 4 -401 12 -13
c ss316L inner clad for transporter (0.5 cm)
402 8 -7.9497 401 -402 12 -13
c transporter shield (6"=15.24 cm total)
5 3 -7.832 402 -5 12 -13
6 3 -7.832 5 -6 12 -13
7 3 -7.832 6 -7 12 -13
c B-POLY neutron shield (4"=10.16 cm total, 4 cells)
701 7 -0.92 7 -701 12 -13
702 7 -0.92 701 -702 12 -13
703 7 -0.92 702 -703 12 -13
704 7 -0.92 703 -704 12 -13
c SS316L outer clad for transporter (0.5 cm)
705 8 -7.9497 704 -705 12 -13
c air space between transporter and drift
8 5 -0.001225 705 -8 12 -13
9 5 -0.001225 8 -9 12 -13
c concrete lining (20 cm)
10 9 -2.235 9 -10 12 -13
c tuff medium (15 cm modeled)
11 6 -2.22 10 -11 12 -13
c zero importance cell
12 0 11:-12:13

101 cz 20.0 $1st fuel cell
102 cz 35.0 $2nd fuel cell
103 cz 45.0 $3rd fuel cell
104 cz 54.0 $4th fuel cell
105 cz 61.0 $5th fuel cell

```

1	cz 62.68	\$WP liner
111	cz 69.45	\$radial gap
2	cz 71.45	\$inner barrier
3	cz 76.45	\$1st 5 cm outer barrier
4	cz 81.45	\$2nd 5 cm outer barrier
401	cz 122.5	\$transporter ID
402	cz 123.0	\$0.5 cm ss316L inner clad
5	cz 128.08	\$1st 2" carbon steel
6	cz 133.16	\$2nd 2" carbon steel
7	cz 138.24	\$3rd 2" carbon steel
701	cz 140.78	\$1st 1" b-poly
702	cz 143.32	\$2nd 1" b-poly
703	cz 145.86	\$3rd 1" b-poly
704	cz 148.40	\$4th 1" b-poly
705	cz 148.90	\$0.5 cm ss316L outer clad
8	cz 250.0	\$~1 m from transporter surface
9	cz 361.0	\$concrete lining inner surface
10	cz 381.0	\$drift wall surface
11	cz 395.0	\$15 cm into tuff
*12	pz 0.0	\$model lower axial boundary (reflective)
*13	pz 10.0	\$model upper axial boundary (reflective)
c	active fuel region (smear density 3.0431 g/cc)	
m1	7014	-0.00005
	8016	-0.10540
	24000	-0.00017
	26000	-0.00035
	40000	-0.17140
	50000	-0.00244
	92235	-0.00523
	92236	-0.00419
	92238	-0.70508
	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner [SS316B6A (vf=0.3448)+A825 (vf=0.6552), 8.0038 g/cc]	
m4	5010	-0.00077
	5011	-0.00351

	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	dry tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898
	15031	-0.00004
	19000	-0.02641
	20000	-0.00322
	22000	-0.00056
	25055	-0.00035
	26000	-0.00682
c	neutron shield - 1.5% B-Polypropylene (0.92 g/cc per GA-4 cask)	
m7	1001	-0.1256
	5010	-0.00278
	5011	-0.01223
	6000	-0.8594
c	ss316L for transporter clad (7.9497 g/cc)	
m8	6000	-0.00030
	7014	-0.00100
	14000	-0.00750
	15031	-0.00045
	16032	-0.00030
	24000	-0.17000
	25055	-0.02000
	26000	-0.65545
	28000	-0.12000
	42000	-0.02500
c	ordinary concrete (2.35 g/cc--ANSI/ANS 6.4-1985)	
m9	8016	-0.4983
	1001	-0.0055
	11023	-0.0170
	12000	-0.0026
	13027	-0.0455
	14000	-0.3157
	16032	-0.0013
	19000	-0.0191
	20000	-0.0826
	26000	-0.0123
mode	p	
imp:p	0.1 0.2 0.4 0.6 1 3 5	


```

5 10 40 120 120 150 600 2400 8000 9500 11000 13000 3r 6000 3000 0
esplt:p 0.5 0.5 0.5 0.25 0.5 0.1
phys:p 3.0
sdef cel=d1 pos=0 0 5.0 rad=fcel d4 ext=d2 erg=d3 axs=0 0 1
c uniform radial source distribution
si1 1 101 102 103 104 105
spl 0.1075 0.2217 0.2150 0.2395 0.2163
sb1 0.5 1.0 1.5 2.0 3.0
c uniform axial distribution from 0 to 10 cm p(x)=c
si2 5.0
c biased gamma source spectrum
si3 0.45 0.70 1.0 1.5 2.00 2.5 3.0
sp3 0.0 0.8485 0.0826 6.806-02 8.628-04 1.360-05 1.050-06
sb3 0.0 0.17 0.15 0.65 0.03 0.001 0.0002
c Radii for 5 fuel cells
ds4 s 5 6 7 8 9
si5 0.0 20.0
si6 20.0 35.0
si7 35.0 45.0
si8 45.0 54.0
si9 54.0 61.0
fc2 surface gamma flux (photons/cm**2-s)
f2:p 101 102 103 104 105 1 111 2 3 4 401 402 5 6 7
701 702 703 704 705 8 9 10 11
e2 0.045 0.1 0.45 0.70 1.0 1.5 2.0 3.0
c multiplier=(5.51+16 phtons/s for 21 PWR FA)(10/360.18 for 10 cm)
c photons for 0.45 to 3 MeV only, which contribute to doses
fm2 1.53+15
fc32 surface gamma dose rates--ANSI 6.1.1-1977 (mrem/hr)
e32 0.045 0.1 0.45 0.70 1.0 1.5 2.0 3.0
f32:p 101 102 103 104 105 1 111 2 3 4 401 402 5 6 7
701 702 703 704 705 8 9 10 11
de32 0.010 0.030 0.050 0.070 0.100 0.150
0.200 0.250 0.300 0.350 0.400 0.450
0.500 0.550 0.600 0.650 0.700 0.800
1.000 1.400 1.800 2.200 2.600 2.800
3.25 3.75 4.25 4.75 5.0 5.25
5.75 6.25 6.75 7.5 9.0 11.0
13.0 15.0
df32 3.96-3 5.82-4 2.90-4 2.58-4 2.83-4 3.79-4
5.01-4 6.31-4 7.59-4 8.78-4 9.85-4 1.08-3
1.17-3 1.27-3 1.36-3 1.44-3 1.52-3 1.68-3
1.98-3 2.51-3 2.99-3 3.42-3 3.82-3 4.01-3
4.41-3 4.83-3 5.23-3 5.60-3 5.80-3 6.01-3
6.37-3 6.74-3 7.11-3 7.66-3 8.77-3 1.03-2
1.18-2 1.33-2
fm32 1.53+15
fq0 e f
ctme 60.0

```

VII.3 LISTING OF INPUT FILE TSPRT01.INP (PATH)

MGDS WASTE PACKAGE TRANSPORTER ANALYSIS--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 10 YR
 RADIAL DOSES AT WASTE PACKAGE MIDPLANE
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O		3.0431 0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
10.0		1.0		FE	7.9497	1.0
11.0		2.0		H	0.92	0.1256
C		0.92		0.8744		
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
202.49						
4.0	CYL	5.0		69.45	0.0	
202.49						
5.0	CYL	2.0		71.45	0.0	
204.99						
6.0	CYL	3.0		81.45	0.0	
215.99						
7.0	CYL	5.0		122.5	0.0	
322.5						
8.0	CYL	10.0		123.0	0.0	
323.0						
9.0	CYL	3.0		138.24	0.0	
338.24						
10.0	CYL	11.0		148.40	0.0	
348.40						
11.0	CYL	10.0		148.90	0.0	
348.90						
12.0	CYL	5.0		361.0	0.0	
1000.0						
13.0	CYL	9.0		381.0	0.0	
1000.0						
14.0	CYL	5.0		61.0	180.09	
191.69						
15.0	CYL	8.0		61.0	191.69	
195.5						
16.0	CYL	5.0		61.0	195.5	
202.49						

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1.0	13.0	12.0	11.0	10.0	9.0	8.0	7.0	6.0	5.0	4.0	3.0
1.0	2.0	14.0	15.0	16.0							

1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			

180.09

1.0	PHO	0.125		7.279+08	0.225			6.110+08
1.0	PHO	0.375		2.909+08	0.575			1.110+10
1.0	PHO	0.850		1.081+09	1.25			8.908+08
1.0	PHO	1.75		1.129+07	2.25			1.781+05
1.0	PHO	2.75		1.375+04	3.50			1.681+03

1.0		0.0		81.45		0.0
1.0		26.0	1.0	ALL		
2.0		0.0		148.9		0.0
2.0		6.0	1.0	ALL		
3.0		0.0		250.0		
3.0		6.0	1.0	ALL		
4.0		0.0		361.0		
4.0		6.0	1.0	ALL		

VII.4 LISTING OF INPUT FILE TSPRT02A.INP (PATH)

MGDS WASTE PACKAGE TRANSPORTER ANALYSIS--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 10 YR
 AXIAL DOSES ON WASTE PACKAGE CENTERLINE
 NO DOSE CORRECTION REQUIRED

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O		3.0431 0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
10.0		1.0		FE	7.9497	1.0
11.0		2.0		H	0.92	0.1256
C		0.92		0.8744		
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
202.49						
4.0	CYL	5.0		69.45	0.0	
202.49						
5.0	CYL	2.0		71.45	0.0	
204.99						
6.0	CYL	3.0		81.45	0.0	
215.99						
7.0	CYL	5.0		122.5	0.0	
322.5						
8.0	CYL	10.0		123.0	0.0	
323.0						
9.0	CYL	3.0		138.24	0.0	
340.78						
10.0	CYL	11.0		148.40	0.0	
348.40						
11.0	CYL	10.0		148.90	0.0	
348.90						
12.0	CYL	5.0		361.0	0.0	
1000.0						
13.0	CYL	9.0		381.0	0.0	
1000.0						
14.0	CYL	5.0		61.0	180.09	
191.69						
15.0	CYL	8.0		61.0	191.69	
195.5						
16.0	CYL	5.0		61.0	195.5	
202.49						
1.0	13.0	12.0	11.0	10.0	9.0	8.0 7.0 6.0 5.0 4.0 3.0

1.0 2.0 14.0 15.0 16.0

1.0 RAZ 3999.0 RAD 40.0 HT 99.0 ANG 1.0
2.0 CYL 1.0 61.0 0.0

180.09

1.0 PHO 0.125 7.279+08 0.225 6.110+08
1.0 PHO 0.375 2.909+08 0.575 1.110+10
1.0 PHO 0.850 1.081+09 1.25 8.908+08
1.0 PHO 1.75 1.129+07 2.25 1.781+05
1.0 PHO 2.75 1.375+04 3.50 1.681+03

1.0 215.99 0.000 0.0
1.0 26.0 1.0 ALL
2.0 348.9 0.000 0.0
2.0 6.0 1.0 ALL
3.0 450.0 0.000
3.0 6.0 1.0 ALL
4.0 550.0 0.000
4.0 6.0 1.0 ALL

VII.5 LISTING OF INPUT FILE TSPRT03A.INP (PATH)

MGDS WASTE PACKAGE TRANSPORTER ANALYSIS
 CO-60 GAMMAS FROM LOWER END FITTINGS
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 10 YR
 AXIAL DOSES ON WASTE PACKAGE CENTERLINE
 NO DOSE CORRECTION REQUIRED

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O	3.0431	0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
10.0		1.0		FE	7.9497	1.0
11.0		2.0		H	0.92	0.1256
C		0.92		0.8744		
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
202.49						
4.0	CYL	5.0		69.45	0.0	
202.49						
5.0	CYL	2.0		71.45	0.0	
204.99						
6.0	CYL	3.0		81.45	0.0	
215.99						
7.0	CYL	5.0		122.5	0.0	
322.5						
8.0	CYL	10.0		123.0	0.0	
323.0						
9.0	CYL	3.0		138.24	0.0	
340.78						
10.0	CYL	11.0		148.40	0.0	
348.40						
11.0	CYL	10.0		148.90	0.0	
348.90						
12.0	CYL	5.0		361.0	0.0	
1000.0						
13.0	CYL	9.0		381.0	0.0	
1000.0						
14.0	CYL	5.0		61.0	180.09	
191.69						
15.0	CYL	8.0		61.0	191.69	
195.5						
16.0	CYL	5.0		61.0	195.5	
202.49						

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1.0	13.0	12.0	11.0	10.0	9.0	8.0	7.0	6.0	5.0	4.0	3.0
1.0	2.0	14.0	15.0	16.0							

1.0	RAZ	3999.0		RAD	80.0	HT	5.0	ANG	1.0
15.0	CYL	8.0		61.0		191.69			
195.50									
1.0	ISO	CO60			5.702-02				

1.0		215.99		0.000		0.0
1.0		26.0	1.0	ALL		
2.0		348.9		0.000		0.0
2.0		6.0	1.0	ALL		
3.0		450.0		0.000		
3.0		6.0	1.0	ALL		
4.0		550.0		0.000		
4.0		6.0	1.0	ALL		

ATTACHMENT VIII
INPUT AND OUTPUT FILES FOR ALARA EVALUATION

This attachment gives the computer input and output file names of the PATH run associated with the ALARA evaluation (Section 7.6), and provides a complete listing of the input file. The input and output files are stored on electronic media as Ref. 5.29. This reference is submitted with this document as part of the Design Analysis Records Package. This attachment covers the following PATH case:

<u>Case Description</u>	<u>Input file</u>	<u>Output File</u>
Radial transporter analysis, average SNF, fuel gammas	TSPRT01A.INP	TSPRT01A.OUT

VIII.1 LISTING OF INPUT FILE TSPRT01A.INP

MGDS WASTE PACKAGE TRANSPORTER ANALYSIS--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--3.72%, 39750 MWD/MTU, 26 YR
 RADIAL DOSES AT WASTE PACKAGE MIDPLANE
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O		3.0431 0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
10.0		1.0		FE	7.9497	1.0
11.0		2.0		H	0.92	0.1256
C		0.92		0.8744		
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
202.49						
4.0	CYL	5.0		69.45	0.0	
202.49						
5.0	CYL	2.0		71.45	0.0	
204.99						
6.0	CYL	3.0		81.45	0.0	
215.99						
7.0	CYL	5.0		122.5	0.0	
322.5						
8.0	CYL	10.0		123.0	0.0	
323.0						
9.0	CYL	3.0		138.24	0.0	
338.24						
10.0	CYL	11.0		148.40	0.0	
348.40						
11.0	CYL	10.0		148.90	0.0	
348.90						
12.0	CYL	5.0		361.0	0.0	
1000.0						
13.0	CYL	9.0		381.0	0.0	
1000.0						
14.0	CYL	5.0		61.0	180.09	
191.69						
15.0	CYL	8.0		61.0	191.69	
195.5						
16.0	CYL	5.0		61.0	195.5	
202.49						

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1.0	13.0	12.0	11.0	10.0	9.0	8.0	7.0	6.0	5.0	4.0	3.0
1.0	2.0	14.0	15.0	16.0							

1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			

180.09

1.0	PHO	0.125		2.949+08	0.225		3.182+08
1.0	PHO	0.375		1.326+08	0.575		5.714+09
1.0	PHO	0.850		8.659+07	1.25		1.312+08
1.0	PHO	1.75		2.639+06	2.25		4.080+02
1.0	PHO	2.75		1.270+03	3.50		5.113+01

1.0		0.0		81.45		0.0
1.0		26.0	1.0	ALL		
2.0		0.0		148.9		0.0
2.0		6.0	1.0	ALL		
3.0		0.0		250.0		
3.0		6.0	1.0	ALL		
4.0		0.0		361.0		
4.0		6.0	1.0	ALL		

ATTACHMENT IX

INPUT AND OUTPUT FILES FOR TIME-DEPENDENT RADIATION CONDITIONS

This attachment gives the computer input and output file names of the MCNP4A and PATH runs associated with the time-dependent radiation condition analysis (Section 7.7), and provides a complete listing of each input file. All the input and output files are stored on electronic media as Ref. 5.29. This reference is submitted with this document as part of the Design Analysis Records Package. This attachment covers the following MCNP4A and PATH cases:

<u>Case Description</u>	<u>Input file</u>	<u>Output File</u>
MCNP4A		
Radial analysis, infinite WP axial length, DBF neutrons	INFWP01N	INFWP01N.OUT
Radial analysis, infinite WP axial length, DBF gammas	INFWP01G	INFWP01G.OUT
PATH		
Radial analysis, DBF gammas (10 yr decay)	PWRWP10.INP	PWRWP10.OUT
Radial analysis, DBF gammas (15 yr decay)	PWRWP15.INP	PWRWP15.OUT
Radial analysis, DBF gammas (20 yr decay)	PWRWP20.INP	PWRWP20.OUT
Radial analysis, DBF gammas (30 yr decay)	PWRWP30.INP	PWRWP30.OUT
Radial analysis, DBF gammas (40 yr decay)	PWRWP40.INP	PWRWP40.OUT
Radial analysis, DBF gammas (50 yr decay)	PWRWP50.INP	PWRWP50.OUT
Radial analysis, DBF gammas (60 yr decay)	PWRWP60.INP	PWRWP60.OUT
Radial analysis, DBF gammas (70 yr decay)	PWRWP70.INP	PWRWP70.OUT
Radial analysis, DBF gammas (80 yr decay)	PWRWP80.INP	PWRWP80.OUT
Radial analysis, DBF gammas (90 yr decay)	PWRWP90.INP	PWRWP90.OUT
Radial analysis, DBF gammas (100 yr decay)	PWRWP100.INP	PWRWP100.OUT
Radial analysis, DBF gammas (110 yr decay)	PWRWP110.INP	PWRWP110.OUT
Radial analysis, DBF gammas (120 yr decay)	PWRWP120.INP	PWRWP120.OUT
Radial analysis, DBF gammas (130 yr decay)	PWRWP130.INP	PWRWP130.OUT
Radial analysis, DBF gammas (140 yr decay)	PWRWP140.INP	PWRWP140.OUT
Radial analysis, DBF gammas (150 yr decay)	PWRWP150.INP	PWRWP150.OUT
Radial analysis, DBF gammas (160 yr decay)	PWRWP160.INP	PWRWP160.OUT

IX.1 LISTING OF INPUT FILE INFWP01N (MCNP4A)

```

MGDS waste package and drift wall neutron + second. gamma analysis
c   neutrons from 21 PWR fuel assemblies in waste package
c   48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c   infinite axial height model
c   drift inside diameter=5.1 m
c   concrete lining=0.2 m
c   evaluation of radial doses
c   intact waste package
c   *****
c   fuel region (homogenized, divided in 5 cells for source biasing)
101  1 -3.0431  -101  12 -13
102  1 -3.0431  101 -102 12 -13
103  1 -3.0431  102 -103 12 -13
104  1 -3.0431  103 -104 12 -13
105  1 -3.0431  104 -105 12 -13
c   liner for fuel basket on the outside
1    4 -8.0038  105 -1  12 -13
c   WP radial gap
111  5 -0.001225 1  -111 12 -13
c   WP inner barrier (2 cm)
2    2 -8.14    111 -2  12 -13
c   WP outer barriers (10 cm total)
3    3 -7.832   2  -3  12 -13
4    3 -7.832   3  -4  12 -13
c   air space between waste package and drift
8    5 -0.001225 4  -8  12 -13
81   5 -0.001225 8  -81 12 -13
82   5 -0.001225 81 -82 12 -13
9    5 -0.001225 82 -9  12 -13
c   concrete lining (20 cm thick)
10   7 -2.35    9  -10 12 -13
c   dry tuff medium (15 cm modeled)
11   6 -2.22    10 -11 12 -13
c   zero importance cell
12   0          11:-12:13

101  cz 20.0
102  cz 35.0
103  cz 45.0
104  cz 54.0
105  cz 61.0
1    cz 62.68
111  cz 69.45
2    cz 71.45
3    cz 76.45
4    cz 81.45
8    cz 128.0    $~11 in. from max. WP envelope
81   cz 176.0    $~30 in. from max. WP envelope
82   cz 245.0
9    cz 255.0
10   cz 275.0
11   cz 290.0
*12  pz 0.0
*13  pz 10.0

```

c	active fuel region (smear density 3.0431 g/cc)
m1	7014 -0.00005
	8016 -0.10540
	24000 -0.00017
	26000 -0.00035
	40000 -0.17140
	50000 -0.00244
	92235 -0.00523
	92236 -0.00419
	92238 -0.70508
	94239 -0.00359
	94240 -0.00129
	94241 -0.00082
c	alloy 825 (8.14 g/cc)
m2	6000 -0.0005
	13027 -0.0020
	14000 -0.0050
	16032 -0.0003
	22000 -0.0090
	24000 -0.2150
	25055 -0.0100
	26000 -0.2857
	28000 -0.4200
	29000 -0.0225
	42000 -0.0300
c	A516 carbon steel (7.832 g/cc)
m3	6000 -0.0022
	14000 -0.00275
	15031 -0.00035
	16032 -0.00035
	25055 -0.0090
	26000 -0.98535
c	liner [SS316B6A (vf=0.3448)+A825 (vf=0.6552), 8.0038 g/cc]
m4	5010 -0.00077
	5011 -0.00351
	6000 -0.00043
	7014 -0.00033
	13027 -0.00133
	14000 -0.00584
	15031 -0.00015
	16032 -0.00030
	22000 -0.00600
	24000 -0.20686
	25055 -0.01336
	26000 -0.39270
	28000 -0.32505
	29000 -0.01499
	42000 -0.02836
c	air (0.001225 g/cc)
m5	7014 -0.8
	8016 -0.2
c	dry tuff material (2.22 g/cc)
m6	8016 -0.49863
	11023 -0.02909
	12000 -0.00077
	13027 -0.06513
	14000 -0.36898

```

15031  -0.00004
19000  -0.02641
20000  -0.00322
22000  -0.00056
25055  -0.00035
26000  -0.00682
c      ordinary concrete (2.35 g/cc, per ANSI/ANS 6.4-1985)
m7     8016  -0.4983
       1001  -0.0055
       11023 -0.0170
       12000 -0.0026
       13027 -0.0455
       14000 -0.3157
       16032 -0.0013
       19000 -0.0191
       20000 -0.0826
       26000 -0.0123
mode    n p
imp:n    1 7r 1.5 2 4r 2 1 0
imp:p    0.25 0.35 0.5 0.7 1 1 1.5 2 4 15 4r 15 4 0
esplt:n  0.5 1.0-3 0.5 1.0-4 0.25 1.0-5 0.25 1.0-6
esplt:p  0.5 0.25 0.5 0.1
phys:n   25.0 1.0-7
phys:p   3.0
sdef     cel=d1 pos=0 0 5.0 rad=fcel d4 ext=d2 erg=d3 axs=0 0 1
si1      1 101      102      103      104      105
spl      0.1075 0.2217 0.2150 0.2395 0.2164
si2      5.0
c        unbiased neutron source spectrum
si3      0.10      0.40      0.9      1.4      1.85      3.0      6.43      20.0
sp3      0.0      0.0376 0.1924 0.1767 0.1310 0.2341 0.2099 0.0183
ds4      s 5 6 7 8 9
si5      0.0      20.0
si6      20.0 35.0
si7      35.0 45.0
si8      45.0 54.0
si9      54.0 61.0
fc12     surface neutron dose rate--ANSI 6.1.1-1977 (mrem/hr)
fl12:n   101 102 103 104 105 2 4 8 81 82 9
el2      6.25-7 0.1 1.0 25.0
de12     2.5-8 1.0-7 1.0-6 1.0-5 1.0-4
          1.0-3 1.0-2 1.0-1 5.0-1 1.0
          2.5 5.0 7.0 10.0 14.0 20.0
df12     3.67-3 3.67-3 4.46-3 4.54-3 4.18-3
          3.76-3 3.56-3 2.17-2 9.26-2 1.32-1
          1.25-1 1.56-1 1.47-1 1.47-1 2.08-1 2.27-1
c        multiplier=(6.567+9 n/s for 21 PWR FA) (10/360.18 for 10 cm)
fm12     1.823+8
fc42     surface gamma dose rates--mrem/hr--ANSI 6.1.1-1977
e42      0.045 0.1 0.45 0.70 1.0 1.5 2.0 3.0 5.0 7.5 10.0 15.0
f42:p    101 102 103 104 105 2 4 8 81 82 9
de42     0.010 0.030 0.050 0.070 0.100 0.150
          0.200 0.250 0.300 0.350 0.400 0.450
          0.500 0.550 0.600 0.650 0.700 0.800
          1.000 1.400 1.800 2.200 2.600 2.800
          3.25 3.75 4.25 4.75 5.0 5.25
          5.75 6.25 6.75 7.5 9.0 11.0

```

	13.0	15.0				
df42	3.96-3	5.82-4	2.90-4	2.58-4	2.83-4	3.79-4
	5.01-4	6.31-4	7.59-4	8.78-4	9.85-4	1.08-3
	1.17-3	1.27-3	1.36-3	1.44-3	1.52-3	1.68-3
	1.98-3	2.51-3	2.99-3	3.42-3	3.82-3	4.01-3
	4.41-3	4.83-3	5.23-3	5.60-3	5.80-3	6.01-3
	6.37-3	6.74-3	7.11-3	7.66-3	8.77-3	1.03-2
	1.18-2	1.33-2				
c	multiplier=(6.567+9 n/s for 21 PWR FA) (10/360.18 for 10 cm)					
fm42	1.823+8					
fq0	e f					
ctme	300.0					
nps	6000000					

IX.2 LISTING OF INPUT FILE INFWP01G (MCNP4A)

MGDS waste package and drift wall primary gamma analysis

```

c    primary gammas from 21 PWR fuel assemblies in waste package
c    48086 MWD/MTU, 4.2% initial enrichment and 10 years old
c    infinite axial height model
c    drift inside diameter=5.1 m
c    concrete lining=0.2 m
c    evaluation of radial doses
c    intact waste package
c    *****
c    fuel region (homogenized, divided in 5 cells for source biasing)
101  1 -3.0431  -101  12 -13
102  1 -3.0431  101 -102 12 -13
103  1 -3.0431  102 -103 12 -13
104  1 -3.0431  103 -104 12 -13
105  1 -3.0431  104 -105 12 -13
c    liner for fuel basket on the outside
1    4 -8.0038  105 -1  12 -13
c    WP radial gap
111  5 -0.001225 1  -111 12 -13
c    WP inner barrier (2 cm)
2    2 -8.14    111 -2  12 -13
c    WP outer barriers (10 cm total)
3    3 -7.832   2  -3  12 -13
4    3 -7.832   3  -4  12 -13
c    air space between waste package and drift
8    5 -0.001225 4  -8  12 -13
81   5 -0.001225 8  -81 12 -13
82   5 -0.001225 81 -82 12 -13
9    5 -0.001225 82 -9  12 -13
c    concrete lining (20 cm thick)
10   7 -2.35    9  -10 12 -13
c    dry tuff medium (15 cm modeled)
11   6 -2.22    10 -11 12 -13
c    zero importance cell
12   0          11:-12:13

101  cz 20.0
102  cz 35.0
103  cz 45.0
104  cz 54.0
105  cz 61.0
1    cz 62.68
111  cz 69.45
2    cz 71.45
3    cz 76.45
4    cz 81.45
8    cz 128.0    $~11 in. from max. WP envelope
81   cz 176.0    $~30 in. from max. WP envelope
82   cz 245.0
9    cz 255.0
10   cz 275.0
11   cz 290.0
*12  pz 0.0
*13  pz 10.0

```


c	active fuel region (smear density 3.0431 g/cc)	
m1	7014	-0.00005
	8016	-0.10540
	24000	-0.00017
	26000	-0.00035
	40000	-0.17140
	50000	-0.00244
	92235	-0.00523
	92236	-0.00419
	92238	-0.70508
	94239	-0.00359
	94240	-0.00129
	94241	-0.00082
c	alloy 825 (8.14 g/cc)	
m2	6000	-0.0005
	13027	-0.0020
	14000	-0.0050
	16032	-0.0003
	22000	-0.0090
	24000	-0.2150
	25055	-0.0100
	26000	-0.2857
	28000	-0.4200
	29000	-0.0225
	42000	-0.0300
c	A516 carbon steel (7.832 g/cc)	
m3	6000	-0.0022
	14000	-0.00275
	15031	-0.00035
	16032	-0.00035
	25055	-0.0090
	26000	-0.98535
c	liner [SS316B6A (vf=0.3448)+A825 (vf=0.6552), 8.0038 g/cc]	
m4	5010	-0.00077
	5011	-0.00351
	6000	-0.00043
	7014	-0.00033
	13027	-0.00133
	14000	-0.00584
	15031	-0.00015
	16032	-0.00030
	22000	-0.00600
	24000	-0.20686
	25055	-0.01336
	26000	-0.39270
	28000	-0.32505
	29000	-0.01499
	42000	-0.02836
c	air (0.001225 g/cc)	
m5	7014	-0.8
	8016	-0.2
c	dry tuff material (2.22 g/cc)	
m6	8016	-0.49863
	11023	-0.02909
	12000	-0.00077
	13027	-0.06513
	14000	-0.36898

```

15031  -0.00004
19000  -0.02641
20000  -0.00322
22000  -0.00056
25055  -0.00035
26000  -0.00682
c      ordinary concrete (2.35 g/cc, per ANSI/ANS 6.4-1985)
m7     8016  -0.4983
       1001  -0.0055
       11023 -0.0170
       12000 -0.0026
       13027 -0.0455
       14000 -0.3157
       16032 -0.0013
       19000 -0.0191
       20000 -0.0826
       26000 -0.0123
mode    p
imp:p   0.25 0.35 0.5 0.7 1 1 1.5 2 4 15 4r 15 4 0
esplt:p 0.5 0.25 0.5 0.1
phys:p  3.0
sdef    cel=d1 pos=0 0 5.0 rad=fcel d4 ext=d2 erg=d3 axs=0 0 1
si1     1 101 102 103 104 105
spl     0.1075 0.2217 0.2150 0.2395 0.2164
sbl     0.5 1.0 1.5 2.0 3.0
si2     5.0
si3     0.45 0.70 1.0 1.5 2.0 2.5 3.0
sp3     0.0 0.8485 0.0826 6.806-02 8.628-04 1.360-05 1.050-06
sb3     0.0 0.17 0.15 0.65 0.03 0.001 0.0002
ds4     s 5 6 7 8 9
si5     0.0 20.0
si6     20.0 35.0
si7     35.0 45.0
si8     45.0 54.0
si9     54.0 61.0
fc42    surface gamma dose rates--mrem/hr--ANSI 6.1.1-1977
e42     0.045 0.1 0.45 0.70 1.0 1.5 2.0 3.0
f42:p   101 102 103 104 105 2 4 8 81 82 9
de42    0.010 0.030 0.050 0.070 0.100 0.150
        0.200 0.250 0.300 0.350 0.400 0.450
        0.500 0.550 0.600 0.650 0.700 0.800
        1.000 1.400 1.800 2.200 2.600 2.800
        3.25 3.75 4.25 4.75 5.0 5.25
        5.75 6.25 6.75 7.5 9.0 11.0
        13.0 15.0
df42    3.96-3 5.82-4 2.90-4 2.58-4 2.83-4 3.79-4
        5.01-4 6.31-4 7.59-4 8.78-4 9.85-4 1.08-3
        1.17-3 1.27-3 1.36-3 1.44-3 1.52-3 1.68-3
        1.98-3 2.51-3 2.99-3 3.42-3 3.82-3 4.01-3
        4.41-3 4.83-3 5.23-3 5.60-3 5.80-3 6.01-3
        6.37-3 6.74-3 7.11-3 7.66-3 8.77-3 1.03-2
        1.18-2 1.33-2
c      multiplier=(5.51+16 photons/s for 0.45 - 3.0 MeV) (10/360.18 for 10 cm)
fm42    1.53+15
fq0     e f
ctme    300.0
nps     6000000

```

IX.3 LISTING OF INPUT FILE PWRWP10.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL

21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 10 YR

RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL

HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005		MO	3.0431	0.1714	
SN		3.0431		0.0024		U	3.0431	0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0		0.0			
180.09									
3.0	CYL	4.0		62.68		0.0			
180.09									
4.0	CYL	5.0		69.45		0.0			
180.09									
5.0	CYL	2.0		71.45		0.0			
180.09									
6.0	CYL	3.0		81.45		0.0			
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125		7.279+08	0.225			6.110+08	
1.0	PHO	0.375		2.909+08	0.575			1.110+10	
1.0	PHO	0.850		1.081+09	1.25			8.908+08	
1.0	PHO	1.75		1.129+07	2.25			1.781+05	
1.0	PHO	2.75		1.375+04	3.50			1.681+03	
1.0		0.0		81.45		0.0			
1.0		26.0	1.0	ALL					
2.0		0.0		255.0		0.0			
2.0		26.0	1.0	ALL					

IX.4 LISTING OF INPUT FILE PWRWP15.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL

21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 15 YR

RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL

HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005		MO	3.0431	0.1714	
SN		3.0431		0.0024		U	3.0431	0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0		0.0			
180.09									
3.0	CYL	4.0		62.68		0.0			
180.09									
4.0	CYL	5.0		69.45		0.0			
180.09									
5.0	CYL	2.0		71.45		0.0			
180.09									
6.0	CYL	3.0		81.45		0.0			
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125			5.640+08	0.225		5.166+08	
1.0	PHO	0.375			2.221+08	0.575		9.070+09	
1.0	PHO	0.850			3.736+08	1.25		4.997+08	
1.0	PHO	1.75			7.578+06	2.25		5.360+03	
1.0	PHO	2.75			2.483+03	3.50		1.798+02	
1.0		0.0		81.45		0.0			
1.0		26.0	1.0	ALL					
2.0		0.0		255.0		0.0			
2.0		26.0	1.0	ALL					

IX.5 LISTING OF INPUT FILE PWRWP20.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL

21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 20 YR

RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL

HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O	3.0431	0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
180.09						
4.0	CYL	5.0		69.45	0.0	
180.09						
5.0	CYL	2.0		71.45	0.0	
180.09						
6.0	CYL	3.0		81.45	0.0	
180.09						
1.0	6.0	5.0	4.0	3.0	2.0	
1.0	RAZ	3999.0		RAD	10.0	HT 20.0 ANG 19.0
2.0	CYL	1.0		61.0	0.0	
180.09						
1.0	PHO	0.125		4.536+08	0.225	4.481+08
1.0	PHO	0.375		1.870+08	0.575	7.920+09
1.0	PHO	0.850		1.904+08	1.25	2.935+08
1.0	PHO	1.75		5.240+06	2.25	1.085+03
1.0	PHO	2.75		2.071+03	3.50	1.095+02
1.0		0.0		81.45	0.0	
1.0		26.0	1.0	ALL		
2.0		0.0		255.0	0.0	
2.0		26.0	1.0	ALL		

IX.6 LISTING OF INPUT FILE PWRWP30.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL

21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 30 YR

RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL

HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005		MO	3.0431	0.1714	
SN		3.0431		0.0024		U	3.0431	0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0		0.0			
180.09									
3.0	CYL	4.0		62.68		0.0			
180.09									
4.0	CYL	5.0		69.45		0.0			
180.09									
5.0	CYL	2.0		71.45		0.0			
180.09									
6.0	CYL	3.0		81.45		0.0			
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125		3.122+08	0.225			3.444+08	
1.0	PHO	0.375		1.431+08	0.575			6.244+09	
1.0	PHO	0.850		8.448+07	1.25			1.101+08	
1.0	PHO	1.75		2.592+06	2.25			3.962+02	
1.0	PHO	2.75		1.869+03	3.50			7.450+01	
1.0		0.0		81.45		0.0			
1.0		26.0	1.0	ALL					
2.0		0.0		255.0		0.0			
2.0		26.0	1.0	ALL					

IX.7 LISTING OF INPUT FILE PWRWP40.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 40 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005		MO	3.0431	0.1714	
SN		3.0431		0.0024		U	3.0431	0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0		0.0			
180.09									
3.0	CYL	4.0		62.68		0.0			
180.09									
4.0	CYL	5.0		69.45		0.0			
180.09									
5.0	CYL	2.0		71.45		0.0			
180.09									
6.0	CYL	3.0		81.45		0.0			
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125		2.279+08	0.225			2.678+08	
1.0	PHO	0.375		1.119+08	0.575			4.951+09	
1.0	PHO	0.850		4.481+07	1.25			4.654+07	
1.0	PHO	1.75		1.384+06	2.25			2.000+02	
1.0	PHO	2.75		1.684+03	3.50			5.166+01	
1.0		0.0		81.45		0.0			
1.0		26.0	1.0	ALL					
2.0		0.0		255.0		0.0			
2.0		26.0	1.0	ALL					

IX.8 LISTING OF INPUT FILE PWRWP50.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL

21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 50 YR

RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL

HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O		3.0431 0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE		8.14 1.0
3.0		1.0		FE		7.832 1.0
4.0		1.0		FE		8.0038 1.0
5.0		1.0		AIR		0.001225 1.0
6.0		1.0		CONC		2.22 1.0
7.0		1.0		FE		2.4701 1.0
8.0		1.0		FE		2.5402 1.0
9.0		1.0		CONC		2.35 1.0
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
180.09						
4.0	CYL	5.0		69.45	0.0	
180.09						
5.0	CYL	2.0		71.45	0.0	
180.09						
6.0	CYL	3.0		81.45	0.0	
180.09						
1.0	6.0	5.0	4.0	3.0	2.0	
1.0	RAZ	3999.0		RAD	10.0 HT 20.0 ANG 19.0	
2.0	CYL	1.0		61.0	0.0	
180.09						
1.0	PHO	0.125		1.712+08	0.225	2.095+08
1.0	PHO	0.375		8.800+07	0.575	3.928+09
1.0	PHO	0.850		2.641+07	1.25	2.059+07
1.0	PHO	1.75		7.612+05	2.25	1.232+02
1.0	PHO	2.75		1.516+03	3.50	3.615+01
1.0		0.0		81.45	0.0	
1.0		26.0	1.0	ALL		
2.0		0.0		255.0	0.0	
2.0		26.0	1.0	ALL		

IX.9 LISTING OF INPUT FILE PWRWP60.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL

21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 60 YR

RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL

HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O	3.0431	0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
180.09						
4.0	CYL	5.0		69.45	0.0	
180.09						
5.0	CYL	2.0		71.45	0.0	
180.09						
6.0	CYL	3.0		81.45	0.0	
180.09						
1.0	6.0	5.0	4.0	3.0	2.0	
1.0	RAZ	3999.0		RAD	10.0 HT	20.0 ANG 19.0
2.0	CYL	1.0		61.0	0.0	
180.09						
1.0	PHO	0.125		1.311+08	0.225	1.645+08
1.0	PHO	0.375		6.925+07	0.575	3.115+09
1.0	PHO	0.850		1.665+07	1.25	1.016+07
1.0	PHO	1.75		4.671+05	2.25	7.851+01
1.0	PHO	2.75		1.370+03	3.50	2.581+01
1.0		0.0		81.45	0.0	
1.0		26.0	1.0	ALL		
2.0		0.0		255.0	0.0	
2.0		26.0	1.0	ALL		

IX.10 LISTING OF INPUT FILE PWRWP70.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 70 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005		MO	3.0431	0.1714	
SN		3.0431		0.0024		U	3.0431	0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0	0.0				
180.09									
3.0	CYL	4.0		62.68	0.0				
180.09									
4.0	CYL	5.0		69.45	0.0				
180.09									
5.0	CYL	2.0		71.45	0.0				
180.09									
6.0	CYL	3.0		81.45	0.0				
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125		1.020+08	0.225			1.296+08	
1.0	PHO	0.375		5.460+07	0.575			2.472+09	
1.0	PHO	0.850		1.133+07	1.25			5.633+06	
1.0	PHO	1.75		3.090+05	2.25			5.492+01	
1.0	PHO	2.75		1.239+03	3.50			1.878+01	
1.0		0.0		81.45	0.0				
1.0		26.0	1.0	ALL					
2.0		0.0		255.0	0.0				
2.0		26.0	1.0	ALL					

IX.11 LISTING OF INPUT FILE PWRWP80.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 80 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O	3.0431	0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
180.09						
4.0	CYL	5.0		69.45	0.0	
180.09						
5.0	CYL	2.0		71.45	0.0	
180.09						
6.0	CYL	3.0		81.45	0.0	
180.09						
1.0	6.0	5.0	4.0	3.0	2.0	
1.0	RAZ	3999.0		RAD	10.0	HT 20.0 ANG 19.0
2.0	CYL	1.0		61.0	0.0	
180.09						
1.0	PHO	0.125		8.017+07	0.225	1.023+08
1.0	PHO	0.375		4.310+07	0.575	1.962+09
1.0	PHO	0.850		8.184+06	1.25	3.488+06
1.0	PHO	1.75		2.163+05	2.25	4.152+01
1.0	PHO	2.75		1.121+03	3.50	1.387+01
1.0		0.0		81.45	0.0	
1.0		26.0	1.0	ALL		
2.0		0.0		255.0	0.0	
2.0		26.0	1.0	ALL		

IX.12 LISTING OF INPUT FILE PWRWP90.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 90 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005	MO	3.0431		0.1714	
SN		3.0431		0.0024	U	3.0431		0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0	0.0				
180.09									
3.0	CYL	4.0		62.68	0.0				
180.09									
4.0	CYL	5.0		69.45	0.0				
180.09									
5.0	CYL	2.0		71.45	0.0				
180.09									
6.0	CYL	3.0		81.45	0.0				
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125		6.342+07	0.225			8.089+07	
1.0	PHO	0.375		3.402+07	0.575			1.557+09	
1.0	PHO	0.850		6.154+06	1.25			2.342+06	
1.0	PHO	1.75		1.572+05	2.25			3.310+01	
1.0	PHO	2.75		1.015+03	3.50			1.032+01	
1.0		0.0		81.45	0.0				
1.0		26.0	1.0	ALL					
2.0		0.0		255.0	0.0				
2.0		26.0	1.0	ALL					

IX.13 LISTING OF INPUT FILE PWRWP100.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 100 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005		MO	3.0431	0.1714	
SN		3.0431		0.0024		U	3.0431	0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0		0.0			
180.09									
3.0	CYL	4.0		62.68		0.0			
180.09									
4.0	CYL	5.0		69.45		0.0			
180.09									
5.0	CYL	2.0		71.45		0.0			
180.09									
6.0	CYL	3.0		81.45		0.0			
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125		5.034+07	0.225			6.390+07	
1.0	PHO	0.375		2.685+07	0.575			1.235+09	
1.0	PHO	0.850		4.735+06	1.25			1.649+06	
1.0	PHO	1.75		1.165+05	2.25			2.726+01	
1.0	PHO	2.75		9.200+02	3.50			7.668+00	
1.0		0.0		81.45		0.0			
1.0		26.0	1.0	ALL					
2.0		0.0		255.0		0.0			
2.0		26.0	1.0	ALL					

IX.14 LISTING OF INPUT FILE PWRWP110.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 110 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005		MO	3.0431	0.1714	
SN		3.0431		0.0024		U	3.0431	0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0		0.0			
180.09									
3.0	CYL	4.0		62.68		0.0			
180.09									
4.0	CYL	5.0		69.45		0.0			
180.09									
5.0	CYL	2.0		71.45		0.0			
180.09									
6.0	CYL	3.0		81.45		0.0			
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125			4.013+07	0.225		5.101+07	
1.0	PHO	0.375			2.137+07	0.575		9.806+08	
1.0	PHO	0.850			3.534+06	1.25		1.171+06	
1.0	PHO	1.75			8.869+04	2.25		2.131+01	
1.0	PHO	2.75			8.344+02	3.50		6.108+00	
1.0		0.0		81.45		0.0			
1.0		26.0	1.0	ALL					
2.0		0.0		255.0		0.0			
2.0		26.0	1.0	ALL					

IX.15 LISTING OF INPUT FILE PWRWP120.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 120 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005	MO	3.0431		0.1714	
SN		3.0431		0.0024	U	3.0431		0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0		0.0			
180.09									
3.0	CYL	4.0		62.68		0.0			
180.09									
4.0	CYL	5.0		69.45		0.0			
180.09									
5.0	CYL	2.0		71.45		0.0			
180.09									
6.0	CYL	3.0		81.45		0.0			
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125			3.219+07	0.225			4.085+07
1.0	PHO	0.375			1.705+07	0.575			7.786+08
1.0	PHO	0.850			2.699+06	1.25			8.663+05
1.0	PHO	1.75			6.849+04	2.25			1.711+01
1.0	PHO	2.75			7.573+02	3.50			5.041+00
1.0		0.0		81.45		0.0			
1.0		26.0	1.0	ALL					
2.0		0.0		255.0		0.0			
2.0		26.0	1.0	ALL					

IX.16 LISTING OF INPUT FILE PWRWP130.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 130 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP			
1.0		5.0		O		3.0431	0.1055		
FE		3.0431		0.0005		MO	3.0431	0.1714	
SN		3.0431		0.0024		U	3.0431	0.7202	
2.0		1.0		FE		8.14	1.0		
3.0		1.0		FE		7.832	1.0		
4.0		1.0		FE		8.0038	1.0		
5.0		1.0		AIR		0.001225	1.0		
6.0		1.0		CONC		2.22	1.0		
7.0		1.0		FE		2.4701	1.0		
8.0		1.0		FE		2.5402	1.0		
9.0		1.0		CONC		2.35	1.0		
1.0	SPH	5.0		50000.0					
2.0	CYL	1.0		61.0	0.0				
180.09									
3.0	CYL	4.0		62.68	0.0				
180.09									
4.0	CYL	5.0		69.45	0.0				
180.09									
5.0	CYL	2.0		71.45	0.0				
180.09									
6.0	CYL	3.0		81.45	0.0				
180.09									
1.0	6.0	5.0	4.0	3.0	2.0				
1.0	RAZ	3999.0		RAD	10.0	HT	20.0	ANG	19.0
2.0	CYL	1.0		61.0		0.0			
180.09									
1.0	PHO	0.125		2.599+07	0.225			3.282+07	
1.0	PHO	0.375		1.363+07	0.575			6.182+08	
1.0	PHO	0.850		2.110+06	1.25			6.603+05	
1.0	PHO	1.75		5.337+04	2.25			1.415+01	
1.0	PHO	2.75		6.874+02	3.50			4.307+00	
1.0		0.0		81.45	0.0				
1.0		26.0	1.0	ALL					
2.0		0.0		255.0	0.0				
2.0		26.0	1.0	ALL					

IX.17 LISTING OF INPUT FILE PWRWP140.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL

21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 140 YR

RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL

HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O		3.0431 0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE		8.14 1.0
3.0		1.0		FE		7.832 1.0
4.0		1.0		FE		8.0038 1.0
5.0		1.0		AIR		0.001225 1.0
6.0		1.0		CONC		2.22 1.0
7.0		1.0		FE		2.4701 1.0
8.0		1.0		FE		2.5402 1.0
9.0		1.0		CONC		2.35 1.0
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0		0.0
180.09						
3.0	CYL	4.0		62.68		0.0
180.09						
4.0	CYL	5.0		69.45		0.0
180.09						
5.0	CYL	2.0		71.45		0.0
180.09						
6.0	CYL	3.0		81.45		0.0
180.09						
1.0	6.0	5.0	4.0	3.0	2.0	
1.0	RAZ	3999.0		RAD	10.0 HT	20.0 ANG 19.0
2.0	CYL	1.0		61.0		0.0
180.09						
1.0	PHO	0.125		2.108+07	0.225	2.643+07
1.0	PHO	0.375		1.091+07	0.575	4.909+08
1.0	PHO	0.850		1.689+06	1.25	5.136+05
1.0	PHO	1.75		4.187+04	2.25	1.206+01
1.0	PHO	2.75		6.242+02	3.50	3.800+00
1.0		0.0		81.45		0.0
1.0		26.0	1.0	ALL		
2.0		0.0		255.0		0.0
2.0		26.0	1.0	ALL		

IX.18 LISTING OF INPUT FILE PWRWP150.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL
 21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 150 YR
 RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL
 HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O	3.0431	0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
180.09						
4.0	CYL	5.0		69.45	0.0	
180.09						
5.0	CYL	2.0		71.45	0.0	
180.09						
6.0	CYL	3.0		81.45	0.0	
180.09						
1.0	6.0	5.0	4.0	3.0	2.0	
1.0	RAZ	3999.0		RAD	10.0 HT	20.0 ANG 19.0
2.0	CYL	1.0		61.0	0.0	
180.09						
1.0	PHO	0.125		1.718+07	0.225	2.130+07
1.0	PHO	0.375		8.730+06	0.575	3.898+08
1.0	PHO	0.850		1.382+06	1.25	4.050+05
1.0	PHO	1.75		3.296+04	2.25	1.058+01
1.0	PHO	2.75		5.668+02	3.50	3.449+00
1.0		0.0		81.45	0.0	
1.0		26.0	1.0	ALL		
2.0		0.0		255.0	0.0	
2.0		26.0	1.0	ALL		

IX.19 LISTING OF INPUT FILE PWRWP160.INP (PATH)

MGDS IN-DRIFT DOSE CALCULATION--PRIMARY GAMMAS FROM FUEL

21 PWR WASTE PACKAGE--4.2%, 48086 MWD/MTU, 160 YR

RADIAL DOSES @ WASTE PACKAGE SURFACE (MIDPLANE) AND DRIFT WALL

HALF SOURCE HEIGHT MODELED, MULTIPLY DOSE RESULTS BY 2.0

CM	ATTN	BLD	MEV	MFP	PATH	NOSP
1.0		5.0		O	3.0431	0.1055
FE		3.0431		0.0005	MO	3.0431 0.1714
SN		3.0431		0.0024	U	3.0431 0.7202
2.0		1.0		FE	8.14	1.0
3.0		1.0		FE	7.832	1.0
4.0		1.0		FE	8.0038	1.0
5.0		1.0		AIR	0.001225	1.0
6.0		1.0		CONC	2.22	1.0
7.0		1.0		FE	2.4701	1.0
8.0		1.0		FE	2.5402	1.0
9.0		1.0		CONC	2.35	1.0
1.0	SPH	5.0		50000.0		
2.0	CYL	1.0		61.0	0.0	
180.09						
3.0	CYL	4.0		62.68	0.0	
180.09						
4.0	CYL	5.0		69.45	0.0	
180.09						
5.0	CYL	2.0		71.45	0.0	
180.09						
6.0	CYL	3.0		81.45	0.0	
180.09						
1.0	6.0	5.0	4.0	3.0	2.0	
1.0	RAZ	3999.0		RAD	10.0	HT 20.0 ANG 19.0
2.0	CYL	1.0		61.0	0.0	
180.09						
1.0	PHO	0.125		1.404+07	0.225	1.716+07
1.0	PHO	0.375		6.980+06	0.575	3.094+08
1.0	PHO	0.850		1.153+06	1.25	3.222+05
1.0	PHO	1.75		2.601+04	2.25	9.531+00
1.0	PHO	2.75		5.150+02	3.50	3.199+00
1.0		0.0		81.45	0.0	
1.0		26.0	1.0	ALL		
2.0		0.0		255.0	0.0	
2.0		26.0	1.0	ALL		