OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
CALCULATION COVER SHEET

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Dose Rate Calculation for the 2-MCO/2-DHLW Waste Package	
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9. Remarks

Attachment VI (compact disk) moved to REV 00

Revision History

10. Revision No.	11. Description of Revision
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FIGURES

TABLES

1. PURPOSE

The objective of this calculation is to determine the dose rates on the external surfaces of the waste package (WP) containing two Hanford defense high-level waste (DHLW) glass canisters and two Hanford multi-canister overpacks (MCO). Each MCO is loaded with the N Reactor spent nuclear fuel (SNF). The information provided by the sketches attached to this calculation is that of the potential design for the WP type considered in this calculation. The scope of this calculation is limited to reporting dose rates averaged over segments of the WP radial and axial surfaces and of surfaces 1 m and 2 m from the WP. The results of this calculation will be used to assess the shielding performance of the 2-MCO/2-DHLW WP engineering design.

The planning requirements that apply to the generation of this calculation have been identified in the development plan *DOE SNF Analysis Plan for FY 2000* (CRWMS M&O [Civilian Radioactive Waste Management and Operator Contractor] 2000a). This calculation is performed and documented according to AP-3.12Q, *Calculations*.

2. METHOD

The Monte Carlo radiation transport method, which is implemented in the MCNP computer code (Briesmeister 1997), is used to calculate surface dose rates of WPs. MCNP uses continuous-energy cross sections processed from the evaluated nuclear data files (ENDF) (Briesmeister 1997, pp. 2-16 through 2-22 and Appendix G).

The control of the electronic management of data is accomplished in accordance with the process control evaluation for the development plan of this calculation, CRWMS M&O (2000b).

3. ASSUMPTIONS

3.1 The geometry of the 4.5-m-long Hanford canister is represented as a cylinder of nominal length, wall thickness, and outer diameter. The rationale for this assumption is that radiation transport in the upper part of the canister is not affected by the canister shape because this portion of the canister is empty and the wall thickness is maintained. This assumption is used throughout Section 5.

3.2 It is supposed that Savannah River Site Design-Basis Glass fills the Hanford 15-ft-long canisters. The rationale for this assumption is that this glass form provides conservative (higher) dose rates. This assumption is used throughout Section 5.

3.3 The axial peaking factor for the N Reactor SNF is not available. An axial peaking factor of two is assumed to bound the axial source distribution of the N Reactor SNF. The rationale for this assumption is that this value provides conservative (higher) dose rates. However, the calculations show that the surface dose rates next to the MCOs are lower that those next to the DHLW glass canisters. Therefore, the maximum dose rate on the WP external surfaces is not sensitive to the axial peaking factor value of the N Reactor

SNF. Therefore, this assumption does not require further confirmation. This assumption is used in Section 5 and Attachment I.

3.4 The neutron spectrum of the N Reactor SNF is not available. A Watt fission neutron spectrum, which characterizes the pre-irradiated fuel, is assumed for the neutron spectrum of the N Reactor SNF. The rationale for this assumption is that the dose rate evaluation is not sensitive to the neutron spectrum because the neutron dose rates have a negligible contribution to the total dose rates. Therefore, this assumption does not require further confirmation. This assumption is used throughout Section 5.

3.5 The material compositions used in this calculation have elements with allowable ranges of weight percentages. For elements with weight percent range, the midpoint value is used, and the weight percent of the most abundant element is adjusted. The rationale for this assumption is that small weight variations for the affected elements do not affect the accuracy of dose results, as long as the total weight is maintained. This assumption is used throughout Section 5.

4. USE OF COMPUTER SOFTWARE AND MODELS

4.1 SOFTWARE

The MCNP V4B2LV computer code is used to calculate neutron and gamma fluxes on the WP surfaces for dose rate evaluations.

- Program Name: MCNP.
- Version/Revision number: Version 4B2.
- Computer Software Configuration Item (CSCI) Number: 30033 V4B2LV (CRWMS M&O 1998).
- Computer Type: Hewlett Packard (HP) 9000 Series workstation "Bloom" (CRWMS M&O Tag number 700887).
- The MCNP V4B2LV computer code is an appropriate tool to determine the dose rates on the surfaces of a 2-MCO/2-DHLW WP.
- This software has been validated over the range it was used.
- This software was previously obtained from the Software Control Management in accordance with appropriate procedures.

The output files for the various MCNP calculations are documented in Section 8 and Attachment VI. The input files used in this calculation are echoed in the output files.

4.2 SOFTWARE ROUTINES

None used.

4.3 MODELS

None used.

5. CALCULATION

This section presents the input information and describes the calculation performed to evaluate the surface dose rates of a 2-MCO/2-DHLW WP.

The physical parameters of the N Reactor SNF are from Taylor (2000). The technical information was prepared in support of criticality and related calculations, and is used in this document only to determine the dose rate bounding values for this fuel type (N Reactor). The burden is placed on the custodian of the SNF to demonstrate, before acceptance of SNF by the CRWMS, that SNF characteristics important to shielding performances of the WP are not exceeded. Therefore, Taylor (2000) is reference only for this calculation.

The glass density provided by Volume 8 of the SRS Waste Form Qualification Report (WQR) (Marra et al. 1995) is reference only. This WQR demonstrates that the requirements for HLW glass canisters are satisfied through procurement specifications and process controls.

Compositions for structural and other non-fuel-related materials are from standard handbooks, and due to the nature of these sources, these data are established facts and are therefore considered as accepted data.

The Savannah River Site HLW glass composition and source terms are from CRWMS M&O (1999a) and CRWMS M&O (1999b), respectively. These data are unqualified.

The dimensions of a 4.5-m-long Hanford glass pour canister are from Taylor (1997). These data are unqualified.

5.1 CALCULATION INPUTS

The following sections present the inputs used in WP surface dose rate calculations. Each MCNP calculation requires the following inputs: geometry, material, radiation source, and tally specifications. The WP consists of the disposal container, two MCOs loaded with N Reactor SNF, and two 4.5-m-long DHLW glass canisters. Attachment IV presents sketch SK-0198 REV 03 that shows the design for the WP configuration. There are two N Reactor fuel types, Mark IV and Mark IA, that differ in initial uranium enrichments and element geometrical dimensions (Taylor 2000, pp. 10 through 13). Due to the different geometrical dimensions of these two fuel types, two baskets have been designed to accommodate the two fuels inside an MCO. An MCO can accommodate either six baskets containing Mark IA fuels (intact and/or scrap) or five baskets containing Mark IV fuels (intact and/or scrap). An MCO loaded with Mark IA fuels weights up to 7310 kg and contains up to 48 intact rods per basket, while an MCO loaded with Mark IV fuels weights up to 8746.4 kg and contains up to 54 intact rods per basket (Taylor 2000, p. 33). Therefore, an MCO loaded with Mark IA fuels is less self-shielded than an MCO loaded with Mark IV fuels. This calculation uses the N Reactor fuel Mark IA for conservative (higher) dose rate evaluations. Only the intact fuel rods are considered in this calculation.

The number of digits for the numerical values cited herein may be the result of a calculation or may reflect the input from another source; consequently, the number of digits should not be interpreted as an indication of accuracy.

5.1.1 WP Configuration

The WP configuration consists of the DHLW disposal container, fuel support cylinders and plates, and plates, and is shown in sketch SK-0198 REV 03 (presented in Attachment IV). The disposal container is designed to allow the emplacement of two 15-ft- (4.5 m) long Hanford DHLW glass canisters and two Hanford MCOs loaded with N Reactor SNF. Two carbon steel plates, centrally and perpendicularly positioned inside the disposal container cavity, separate the four canisters. A support carbon steel cylinder and plate assembly has been design to elevate each MCO, thus reducing the deformation of the MCO during a drop event. Table 1 presents the geometry and material specifications of the DHLW disposal container, the plates, and the fuel support cylinders and plates. Tables 2 through 4 present the chemical compositions and densities for the structural materials.

Table 1. Geometry and Material Specifications for the WP Configuration

Component	Material Specification[a]	Characteristic	Dimension (mm)
Inner shell	SA-240 S31600	Thickness	50
Outer shell	SB-575 N06022	Thickness	25
Inner shell lids	SA-240 S31600	Thickness	105
Extended outer shell lid base	SB-575 N06022	Thickness	25
Outer shell flat closure lid	SB-575 N06022	Thickness	10
Outer shell flat bottom lid	SB-575 N06022	Thickness	25
A-plate	SA-516 K02700	Thickness	10
Cavity	N/A	Length	4,617
		Diameter	1,584
Closure lid to outer lid gap	N/A	Thickness	30
Inner lid to closure lid gap	N/A	Thickness	30
Fuel support plate	SA-516 K02700	Thickness	5
Fuel support cylinder	SA-516 K02700	Inner diameter	590
		Thickness	5
		Length	270
Gap between the bottom inner and outer shell lids	N/A	Thickness	70

SOURCE: Attachment IV.
NOTE: [a] American Society of Mechanical Engineers (ASME) specification identifier.

Table 2. Chemical Composition of SA-240 S31600

Element[a]	Weight Percent Range[a]	Value Used
Carbon	0.08 (max)	0.08
Manganese	2.00 (max)	2.00
Phosphorus	0.045 (max)	0.045
Sulfur	0.030 (max)	0.03
Silicon	0.75 (max)	0.75
Chromium	16.00-18.00	17.00
Nickel	10.00-14.00	12.00
Molybdenum	2.00-3.00	2.50
Nitrogen	0.10 (max)	0.10
Iron	Balance	65.495
Density[b] = 7.98 g/cm^3		

SOURCE: [a] ASTM (American Society for Testing and Materials) A 240/A240M-97a, *Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels*, p. 2.
[b] ASTM G 1-90, *Standard Practice for Preparing, Cleaning, and Evaluating Corrosion Test Specimens*, p. 7.
NOTE: SA-240 identical with ASTM Specification A 240-97a (ASME 1998, p. 363).

Table 3. Chemical Composition of SB-575 N06022

Element	Weight Percent Range	Value Used
Carbon	0.015 (max)	0.015
Manganese	0.50 (max)	0.50
Silicon	0.08 (max)	0.08
Chromium	20.00-22.50	21.25
Molybdenum	12.5-14.5	13.50
Cobalt	2.50 (max)	2.50
Tungsten	2.5-3.5	3.00
Vanadium	0.35 (max)	0.35
Iron	2.0-6.0	4.00
Sulfur	0.02 (max)	0.02
Phosphorus	0.02 (max)	0.02
Nickel	Balance	54.765
Density = 8.69 g/cm^3		

SOURCE: ASTM B 575-97, *Standard Specification for Low-Carbon Nickel-Molybdenum-Chromium, Low-Carbon Nickel-Chromium-Molybdenum, Low-Carbon Nickel-Chromium-Molybdenum-Copper and Low-Carbon Nickel-Chromium-Molybdenum-Tungsten Alloy Plate, Sheet, and Strip*, p. 2.
NOTE: SA-575 identical with ASTM Specification B 575-97 (ASME 1998, p. 759).

Table 4. Chemical Composition of SA-516 K02700

Element[a]	Weight Percent Range[a]	Value Used
Carbon	0.27	0.27
Manganese	0.79-1.30[c]	1.045
Phosphorus	0.035 (max)	0.035
Sulfur	0.035 (max)	0.035
Silicon	0.13-0.45[c]	0.29
Iron	Balance	98.325
Density[b] = 7.85 g/cm^3		

SOURCE: [a] ASTM A 516/A516M-90, *Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service*, p. 2.
[b] ASTM A 20/A 20M-95, *Standard Specification for General Requirements for Steel Plates for Pressure Vessels*, p. 8.
[c] Product analysis specifications.
NOTE: SA-516 identical with ASTM Specification A 516/A 516M-90 (ASME 1998, p. 923).

5.1.2 DHLW Glass Canisters

Tables 5 through 8 provide geometry, material, and radiation source specifications for the 15-ft- (4.5-m) long Hanford DHLW glass canisters.

Table 5. Geometry and Material Specifications for DHLW Glass Canisters

Component	Material	Characteristic	Value
Canister wall	304L stainless steel (SS)	Outer diameter	610 mm
		Wall thickness	10.5 mm
		Length	4,572 mm
DHLW glass	(Provided in Table 7)	Glass height	3,670 mm

SOURCE: Taylor 1997, pp. 1 and 2.

Table 6. Chemical Composition of 304L SS

Element[a]	Weight Percent Range[a]	Value Used
Carbon	0.030 (max)	0.03
Manganese	2.00 (max)	2.00
Phosphorus	0.045 (max)	0.045
Sulfur	0.03 (max)	0.03
Silicon	0.75 (max)	0.75
Chromium	18.00-20.00	19.00
Nickel	8.00-12.00	10.00
Nitrogen	0.10	0.10
Iron	Balance	68.045
Density[b] = 7.94 g/cm^3		

SOURCE: [a] ASTM A 240/A 240M-97a, p. 2.
[b] ASTM G 1-90, p. 7.

Table 7. Chemical Composition of SRS DHLW Glass

Element/Isotope[a]	Weight Percent[a]	Element/Isotope[a]	Weight Percent[a]
^{6}Li	9.5955E-02	^{7}Li	1.3804E+00
^{10}B	5.9176E-01	^{11}B	2.6189E+00
O	4.4770E+01	F	3.1852E-02
Na	8.6284E+00	Mg	8.2475E-01
Al	2.3318E+00	Si	2.1888E+01
S	1.2945E-01	K	2.9887E+00
Ca	6.6188E-01	Ti	5.9676E-01
Mn	1.5577E+00	Fe	7.3907E+00
Ni	7.3490E-01	P	1.4059E-02
Cr	8.2567E-02	Cu	1.5264E-01
Ag	5.0282E-02	^{137}Ba[c]	1.1267E-01
Pb	6.0961E-02	Cl	1.1591E-01
^{232}Th	1.8559E-01	^{133}Cs	4.0948E-02
^{135}Cs	5.1615E-03	^{234}U	3.2794E-04
^{236}U	1.0415E-03	Zn[d]	6.4636E-02
^{235}U	4.3514E-03	^{238}U	1.8666E+00
^{238}Pu	5.1819E-03	^{239}Pu	1.2412E-02
^{240}Pu	2.2773E-03	^{241}Pu	9.6857E-04
^{242}Pu	1.9168E-04		
Density[b] = 2.65 g/cm^3			

SOURCE: [a] CRWMS M&O 1999a, p. 7.
[b] Marra et al. 1995, p. 39.

NOTES: [c] The neutron and photon cross sections for ^{138}Ba were used in calculations because the cross-section tables for ^{137}Ba were not provided with MCNP code.
[d] Neutron cross-section tables for Zn were not provided with MCNP code; therefore, the Al neutron cross-section tables are used instead because of the similarities between neutron cross sections of these two elements.

Table 8. Gamma and Neutron Sources per 3-m-Long DHLW Glass Canister

Gamma Source		**Neutron Source**	
Upper Energy Boundary (MeV)	**Intensity (photons/s)**	**Upper Energy Boundary (MeV)**	**Intensity (neutrons/s)**
0.05	1.32E+15	0.10	1.97E+05
0.10	3.96E+14	0.40	1.89E+06
0.20	3.10E+14	0.90	6.34E+06
0.30	8.74E+13	1.40	6.92E+06
0.40	6.39E+13	1.85	6.12E+06
0.60	8.83E+13	3.00	2.61E+07
0.80	1.35E+15	6.43	3.42E+07
1.00	2.13E+13	20.00	3.07E+05
1.33	2.96E+13		
1.66	6.42E+12		
2.00	5.14E+11		
2.50	2.94E+12		
3.00	2.04E+10		
4.00	2.28E+09		
5.00	5.25E+05		
6.50	2.11E+05		
8.00	4.13E+04		
10.00	8.75E+03		
Total	3.68E+15		8.21E+07

SOURCE: CRWMS M&O 1999b, Attachment VIII, p. 1, and Attachment IX, p. 1.

5.1.3 N Reactor Mark IA Fuel

Mark IA fuel rods consist of two concentric tubes of uranium metal co-extruded into Zircaloy cladding. The uranium enrichments for the inner and outer fuel elements are 0.947 wt% ^{235}U and 1.25 wt% ^{235}U, respectively (Taylor 2000, p. 13). Mark IA fuel assemblies functioned as the seed or driver fuel in the N Reactor because of their higher enrichment. The mass density of uranium used in fabrication of the N Reactor fuels was 18.82 g/cm^3 (Taylor 2000, p. 20). Table 9 presents the characteristics of Mark IA fuel elements. The chemical composition of Zircaloy 2, which composes the fuel cladding, is presented in Table 10.

Table 9. Characteristics of Mark IA Fuel Elements

Component	Characteristic	Value
Outer element	Pre-irradiation ^{235}U content	1.2500 wt%
	Pre-irradiation ^{236}U content	0.0392 wt%
	Pre-irradiation ^{238}U content	98.7108 wt%
	Outer diameter	6.096 cm
	Inner diameter	4.4958 cm
	Inner Zr tube thickness	0.0555 cm
	Outer Zr tube thickness	0.0635 cm
	End-cap thickness	0.4830 cm
	Length[a]	53.086 cm
Inner element	Pre-irradiation ^{235}U content	0.9470 wt%
	Pre-irradiation ^{236}U content	0.0392 wt%
	Pre-irradiation ^{238}U content	99.0138 wt%
	Outer diameter	3.175 cm
	Inner diameter	1.1176 cm
	Inner Zr tube thickness	0.0635 cm
	Outer Zr tube thickness	0.1015 cm
	End-cap thickness	0.4830 cm
	Length[a]	53.086 cm

SOURCE: Taylor 2000, pp. 11 and 13.
NOTE: [a] This calculation uses the M-type Mark IA element, which is the longest Mark IA element.

Table 10. Chemical Composition of Zircaloy 2

Element[a]	Weight Percent Range[a]	Value Used
Tin	1.20-1.70	1.45
Iron	0.07-0.20	0.135
Chromium	0.05-0.15	0.1
Nickel	0.03-0.08	0.055
Oxygen	0.09-0.16	0.125
Iron + Chromium + Nickel	0.18-0.38	-
Zirconium	Balance	98.135
Density[b] = 6.56 g/cm^3		

SOURCE: [a] ASTM B 811-90, *Standard Specification for Wrought Zirconium Alloy Seamless Tubes for Nuclear Reactor Fuel Cladding*, p. 2.
[b] American Society for Metals 1990, p. 666.
NOTE: The Universal Numbering System for Zircaloy 2 is R60802 (CRWMS M&O 1999c, p. 41).

Table 11 presents the maximum gamma source term per MCO, which is provided in Taylor (2000, p. 39). The maximum neutron source intensity per MCO is 1.17×10^{7} neutrons/s (Taylor 2000, p. 41).

Table 11. Maximum Gamma Source Term per MCO

Upper Energy Boundary (MeV)	Average Group Energy (MeV)	Gamma Intensity (photons/s/MCO)
0.02	1.50E-02	1.75E+15
0.03	2.50E-02	3.87E+14
0.05	3.75E-02	4.21E+14
0.07	5.75E-02	3.46E+14
0.10	8.50E-02	1.95E+14
0.15	1.25E-01	1.48E+14
0.30	2.25E-01	1.66E+14
0.45	3.75E-01	8.64E+13
0.70	6.62E-01	2.81E+15
1.00	8.50E-01	1.04E+14
1.50	1.25E+00	4.33E+13
2.00	1.75E+00	1.29E+12
2.50	2.25E+00	9.42E+10
3.00	2.75E+00	4.67E+09
4.00	3.50E+00	6.04E+08
6.00	5.00E+00	3.71E+05
8.00	7.00E+00	4.23E+04
14.00	1.10E+01	4.84E+03
Total		6.45E+15

SOURCE: Taylor 2000, p. 39.

5.1.4 MCO Container and Basket for Mark IA Fuel

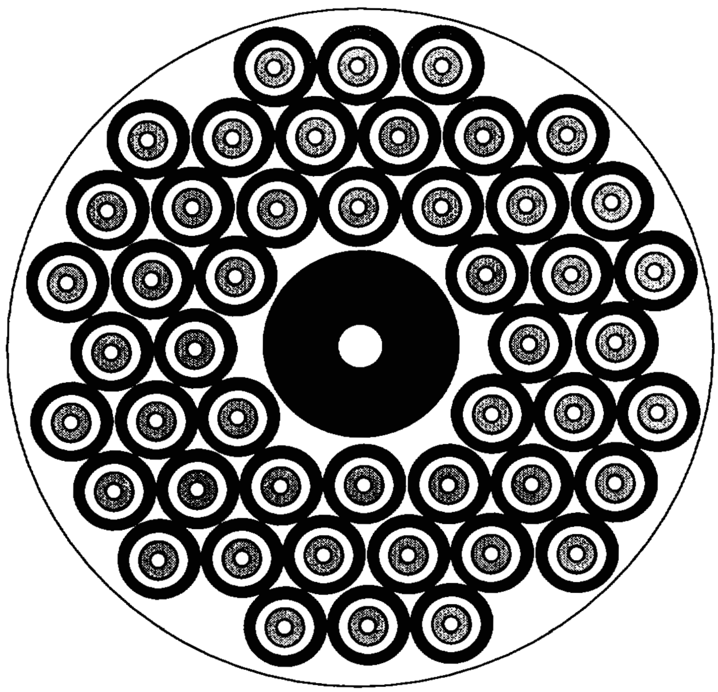
The MCO container is a cylindrical tube with a plate welded at the bottom and a stainless-steel-shield plug at the top. Inside the MCO, six baskets loaded with Mark IA SNF intact rods are stacked. The MCO container and fuel baskets are constructed out of 304L stainless steel (Taylor 2000, pp. 23 and 27). The nominal outside diameter and the overall length of an MCO are 60.922 cm and 422.707 cm, respectively (Taylor 2000, p. 24). The geometry specifications for the MCO container and baskets for the Mark IA intact fuel are presented in Table 12.

Table 12. Geometry Specifications for an MCO

Component	Characteristic	Dimension (cm)
Cylindrical tube	Outer diameter	60.9219
	Wall thickness	1.27
	Inner cavity height	356.545
Bottom plate	Thickness	5.11
Center post	Outer diameter	16.8275
	Nominal inner diameter	4.45135
Fuel baskets	Number per MCO	6
	Maximum intact fuel rods per basket[a]	48
	Nominal height	58.85688
	Inner diameter of the outer shell	57.4675
	Bottom plate thickness	3.048
	Outer shell thickness[b]	0.122

SOURCE: Taylor 2000, pp. 23, 27, 30, and 33.
NOTES: [a] Figure 1 shows the arrangement of Mark IA fuel rods in a triangular-pitch configuration inside MCO.
[b] The outer shell is constructed of 18-gauge (0.048-in.) 304L stainless steel (Taylor 2000, p. 26).

The loading pattern of Mark IA fuel rods in an MCO container is shown in Figure 1.



SOURCE: Taylor 2000, p. 25.
NOTE: Drawing not to scale.

Figure 1. Loading Arrangement for Mark IA Fuel Elements in an MCO Container

5.2 DESCRIPTION OF MCNP SHIELDING CALCULATIONS

The MCNP features employed in the shielding calculations for WPs are the general source and the surface flux tally. MCNP estimates the gamma or the neutron flux averaged over a surface, and then calculates the surface dose rates in rem/h. The dose rate for a certain energy group is the product of group flux and the flux-to-dose rate conversion factor for the energy group. The flux-to-dose rate conversion factors are taken from the American National Standard Institute/American Nuclear Society (ANSI/ANS) Standard 6.1.1-1977 (Briesmeister 1997, App. H).

Since MCNP reports the relative error associated with a tally, the estimated standard deviation is derived from the reported tally and relative error. The estimated relative error (Briesmeister 1997, pp. 2-89 through 2-93) is given by:

$$R = \frac{S_{\bar{x}}}{\bar{x}}, \text{ where}$$

$\bar{x}$ = estimated tally

$S_{\bar{x}}$ = estimated standard deviation

Two calculations, one for the gamma and one for the neutron transport, are required. Therefore, the total dose rate (in rem/h) is the sum of the gamma and neutron dose rates (in rem/h). The estimated standard deviation of the total dose rate (in rem/h) is derived using the following equation:

$$S_{\text{total dose rate}} = \sqrt{(R \cdot \bar{x})^2_{\text{gamma dose rate}} + (R \cdot \bar{x})^2_{\text{neutron dose rate}}}$$

The total dose rates and their standard deviations are calculated in Attachment III.

For the 2-MCO/2-DHLW WP, the gamma dose rates dominate the total dose rates. Therefore, neither a neutron dose rate calculation in rad/h nor a coupled neutron-photon calculation is necessary. The gamma dose rates in rem/h and rad/h are identical because the quality factor, which shows the biological effectiveness of the radiation, is unity for photons.

The radiation sources of this WP, gamma and neutron, originate from the DHLW glass and the N Reactor fuel rods. The sources in the WP are spatially sampled according to the source intensities of each source region. Then, within each source region, the locations of source particles are sampled uniformly. The gamma and neutron sources of the fuel regions have axial spatial distributions. To account for these distributions, constant axial peaking factors are directly multiplied to the source intensities. This method yields conservative dose results for the WP (see Assumption 3.3). Attachment I provides the fractions of gamma and neutron source samplings in each source region, required by the source probability (sp) card, and the total source intensity, required by the tally multiplier (fm) card in the MCNP input.

The calculation needs the source terms to be scaled up to the 4.5-m-long glass canister. The scaling factor is 3.67/2.17=1.69 (Taylor 1997), which gives a gamma and a neutron intensity per 4.5-m-long canister of 6.22E+15 photons/s and 1.39E+08 neutrons/s, respectively.

In radiation shielding analysis, particle population diminishes considerably due to attenuation in shield materials. Therefore, variance reduction techniques are used to increase sampling in regions of the tally. Geometric splitting/Russian roulette across cell boundaries and energy biasing for gamma radiation are the variance reduction techniques employed in this calculation.

Geometric splitting with Russian roulette is a reliable variance reduction technique when the importances of adjacent cells are different by a factor not greater than 4 (Briesmeister 1997, page 2-121). Each particle is split when entering a cell of higher importance. The number of split particles is proportional to the relative importance of the adjacent cells. The weight of each particle is adjusted by the same factor such that a fair game is preserved. On the other hand, if particles enter a cell of lower importance, Russian roulette is played so as to improve calculational efficiency by not tracking unimportant particles. In this calculation, the relative importances assigned to adjacent cells do not exceed a factor of 4. An extra-cell has been created in the waste package inner shell and lids. Hence, the gamma and neutron importances in the source regions are unity, and increase by a factor of 4 for each geometrical cell in the disposal container materials. By increasing particle importance in these materials, the particle population is maintained approximately constant.

The gamma spectra of the SNF and DHLW have a large fraction of low-energy radiation. However, because the mean-free path of low-energy gamma radiation in stainless steel has values in the range of tenths of millimeters, the low-energy radiation that escapes from source regions is entirely absorbed in the inner shell and lids. Therefore, biased gamma source spectra are specified in the MCNP input that increase the importance of the medium and high-energy gamma radiation relative to that of the low-energy gamma radiation.

Figure 2 shows a horizontal cross section of the WP and the angular segments of the radial surfaces, each 90 degrees wide. Higher dose rates are obtained on surface segments adjacent to the DHLW glass canisters. Although the source terms of an MCO are higher than those of a DHLW glass canister, they generate lower dose rates on WP surfaces because of the fuel self-shielding.

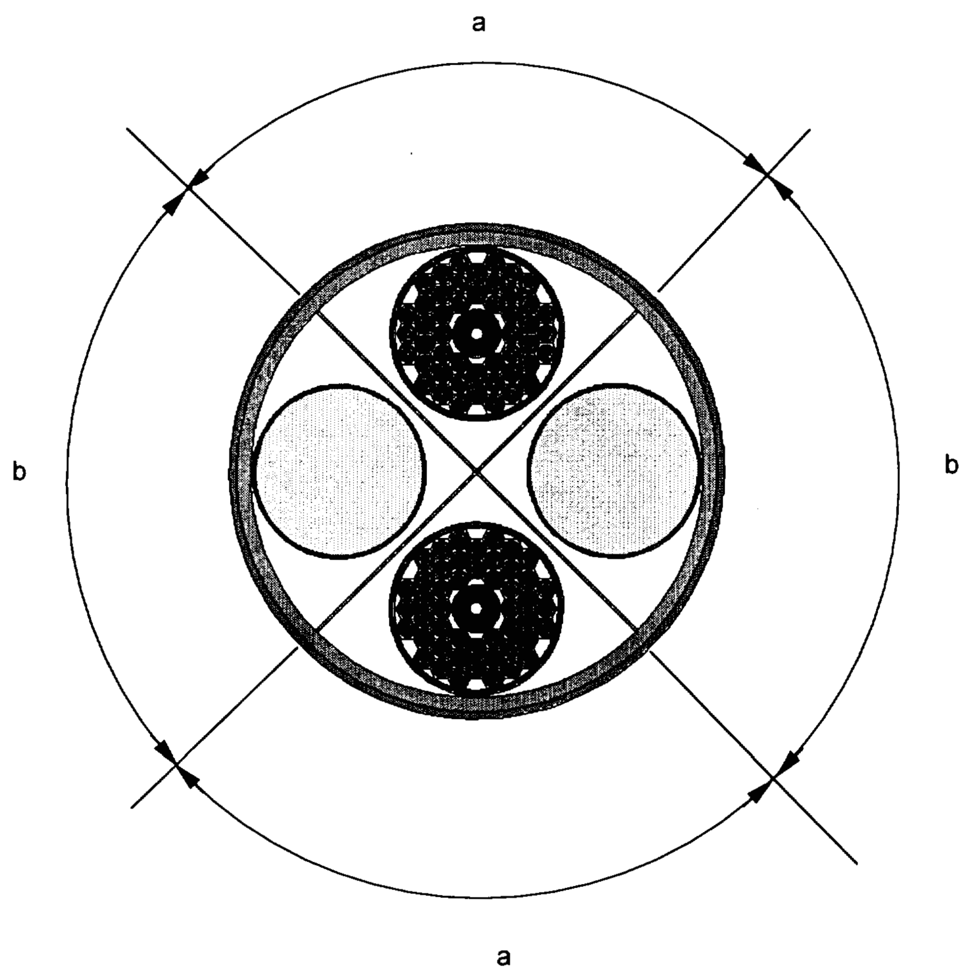
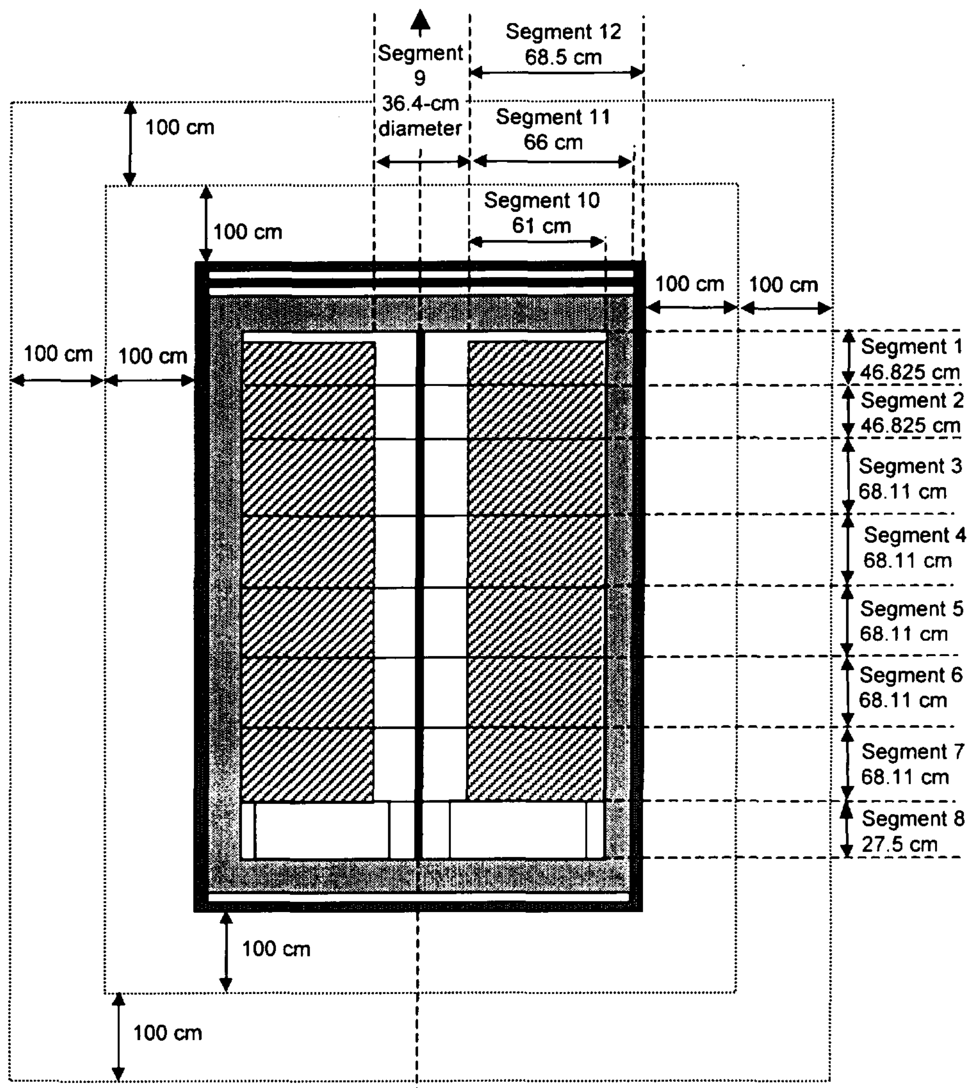


Figure 2. Angular Segments Used in Dose Rate Calculations

Figure 3 shows radial and axial segments used in dose rate tallies. Segments 1 and 2, each 46.825-cm long, are segments of the radial surfaces between the top plane of glass and the top plane of WP cavity. Segments 3 through 7, each 68.11-cm long, are radial surface segments between the bottom plane of the MCO and the top plane of DHLW glass. Segment 8 is between the bottom of the WP cavity and the bottom of MCOs. Each WP axial surface is divided into two segments by a circle of 36.4-cm radius. The axial dose rates for surfaces 1 m and 2 m from the WP outer surfaces are averaged on axial surfaces 1 m and 2 m from the top and bottom WP surfaces, shown in Figure 3. Attachment II contains area calculations for the asymmetrical segments used in MCNP tally specifications.



NOTES: Drawing not to scale.
The figure shows a vertical cross section through the MCO canisters. The contents of the MCO canisters (fuel elements and storage baskets) are not shown in this figure.

Figure 3. Radial and Axial Surfaces and Segments of the WP Used in Dose Rate Calculations

6. RESULTS

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

Tables 13 through 18 present the total radial and axial dose rates (and the corresponding standard deviation) averaged over WP surface segments and segments of surfaces located 1 m and 2 m from the WP outer surfaces (see Figures 2 and 3 for segment locations). The gamma and neutron contributions to these dose rates are shown in Attachment III, where the total dose rates are calculated.

Table 13. Dose Rates on the Radial Surface of the WP Cavity

Axial Location[a]	Segment a[b]		Segment b[c]	
	Dose Rate (rem/h)	Standard Deviation (rem/h)	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	182.6275	11.2835	546.7934	19.2988
Segment 2	1159.6793	24.1188	1956.6811	29.9348
Segment 3	2716.0123	28.7872	6302.3179	47.2646
Segment 4	2950.7801	30.0950	6858.4783	74.7527
Segment 5	2929.6867	29.8798	6846.0839	53.3961
Segment 6	2911.1047	29.9813	6845.5924	47.9161
Segment 7	2603.0683	27.8501	6673.0654	52.0467
Segment 8	1110.3961	26.9779	5296.2527	59.8441

NOTE: [a] See Figure 3 for segment locations.
[b] Segment a is adjacent to an MCO (see Figure 2 for segment location).
[c] Segment b is adjacent to a HLW glass canister (see Figure 2 for segment location).

Table 14. Dose Rates on the Inner Surface of the Outer Shell

Axial Location[a]	Segment a[b]		Segment b[c]	
	Dose Rate (rem/h)	Standard Deviation (rem/h)	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	4.0953	0.4196	12.7302	0.7322
Segment 2	61.9279	1.7818	97.9572	2.2707
Segment 3	156.3524	2.3590	363.3386	3.7405
Segment 4	169.8129	2.5110	387.7186	3.9138
Segment 5	165.8848	2.4031	397.3652	3.9715
Segment 6	163.3804	2.4321	382.7911	3.8257
Segment 7	150.9012	2.3369	379.2422	3.8662
Segment 8	63.5526	2.3223	312.1482	5.5223

NOTE: [a] See Figure 3 for segment locations.
[b] Segment a is adjacent to an MCO (see Figure 2 for segment location).
[c] Segment b is adjacent to a HLW glass canister (see Figure 2 for segment location).

Table 15. Dose Rates on the WP Outer Radial Surface

Axial Location[a]	Segment a[b]		Segment b[c]	
	Dose Rate (rem/h)	Standard Deviation (rem/h)	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	0.5920	0.0679	1.9578	0.1290
Segment 2	11.0123	0.3582	17.5959	0.4953
Segment 3	28.3893	0.4761	67.3416	0.8206
Segment 4	30.7318	0.5122	72.2033	0.8509
Segment 5	30.3483	0.4997	75.0814	0.8924
Segment 6	28.8816	0.4698	72.0221	0.8488
Segment 7	27.2182	0.4699	70.9662	0.8506
Segment 8	12.0523	0.5201	58.4442	1.2260

NOTE: [a] See Figure 3 for segment locations.
[b] Segment a is adjacent to an MCO (see Figure 2 for segment location).
[c] Segment b is adjacent to a HLW glass canister (see Figure 2 for segment location).

Table 16. Dose Rates on a Radial Surface 1 m from the WP Outer Radial Surface

	Segment a[b]		Segment b[c]	
Axial Location[a]	Dose Rate (rem/h)	Standard Deviation (rem/h)	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	2.8758	0.0709	5.0572	0.1045
Segment 2	6.3081	0.1140	10.6980	0.1613
Segment 3	10.3900	0.1297	18.9510	0.1988
Segment 4	13.2583	0.1496	24.2418	0.2204
Segment 5	13.6909	0.1490	25.5179	0.2269
Segment 6	13.2183	0.1478	24.7873	0.2229
Segment 7	10.5763	0.1331	20.8818	0.2086
Segment 8	6.6144	0.1413	14.4221	0.2348

NOTE: [a] See Figure 3 for segment locations.
[b] Segment a is adjacent to an MCO (see Figure 2 for segment location).
[c] Segment b is adjacent to a HLW glass canister. (see Figure 2 for segment location)

Table 17. Dose Rates on a Radial Surface 2 m from the WP Outer Radial Surface

	Segment a[b]		Segment b[c]	
Axial Location[a]	Dose Rate (rem/h)	Standard Deviation (rem/h)	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	2.8915	0.0551	4.8107	0.0793
Segment 2	4.3333	0.0701	7.1151	0.0967
Segment 3	5.9874	0.0735	10.3266	0.1063
Segment 4	7.6541	0.0833	12.8317	0.1179
Segment 5	8.1572	0.0855	13.9213	0.1210
Segment 6	7.5171	0.0818	13.3757	0.1203
Segment 7	6.1288	0.0741	10.8463	0.1084
Segment 8	4.7135	0.0951	8.3895	0.1324

NOTE: [a] See Figure 3 for segment locations.
[b] Segment a is adjacent to an MCO (see Figure 2 for segment location).
[c] Segment b is adjacent to a HLW glass canister (see Figure 2 for segment location).

Table 18. Dose Rates on Segments of the Axial Surfaces

Axial Location	Segment	Dose Rate (rem/h)	Standard Deviation (rem/h)
Top surface of the lower inner shell lid	Segment 9	5624.5040	146.7804
	Segment 10	3619.7954	30.4040
Top surface of the lower outer shell lid	Segment 9	5.4476	0.6185
	Segment 11	9.1295	0.2472
Bottom surface of the WP	Segment 9	1.1907	0.1657
	Segment 12	1.7729	0.0544
Bottom surface 1 m from the WP	Bottom surface 1 m from the WP	0.6120	0.0113
Bottom surface 2 m from the WP	Bottom surface 2 m from the WP	0.3791	0.0053
Bottom surface of the upper inner shell lid	Segment 9	265.0654	22.4706
	Segment 10	326.0561	5.2811
Bottom surface of the extended shell lid base	Segment 9	2.2693	0.6287
	Segment 11	2.3192	0.1420
Top surface of the extended shell lid base	Segment 9	1.0304	0.2531
	Segment 11	1.1470	0.0750
Top surface of the WP	Segment 9	0.2513	0.0680
	Segment 12	0.2709	0.0208
Top surface 1 m from the WP	Top surface 1 m from the WP	0.1665	0.0050
Top surface 2 m from the WP	Top surface 2 m from the WP	0.1606	0.0029

NOTE: See Figure 3 for segment locations.

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7.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

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8. ATTACHMENTS

The hardcopy and electronic attachments are listed below in Table 19. Electronic output files are provided on a compact disk (CD) (Attachment VI), and are listed in Table 20 below. The input files used in this calculation are echoed in the output files. Each output file is identified by its name, size (in bytes), and the date and time of last access. It should be noted that for files transferred from the HP to the personal computer, the date and time reflect the time of transfer. The actual date and time of run completion can be found in the file.

Table 19. List of Attachments

Description	Attachment Number	No. of Pages
Total source intensity and source intensity fractions	I	3
Segment areas used in MCNP tally specifications	II	1
Gamma, neutron, and total surface dose rates	III	4
SK-0198 REV 03[a] *2-MCO/2-DHLW WP Configuration for Site Recommendation*	IV	3
SK-0199 REV 01[b] *2-MCO/2-DHLW Waste Package Weld Configuration*	V	1
MCNP electronic output files (CD)	VI	N/A

NOTE: [a] The documents referenced in this sketch, *Waste Acceptance System Requirements Document* and *Multi-Canister Overpack Design Report*, are identified in this document as DOE (1999) and Smith (1997), respectively.
[b] This sketch is cited in Attachment IV.

Table 20. File Attributes for the Contents of Attachment VI

File Name	Content	File Size (bytes)	File Date	File Time
gamma_dose.io	MCNP output file for gamma surface dose rate calculation	365,980	07/18/2000	8:54 a.m.
neutron_dose.io	MCNP output file for neutron surface dose rate calculation	328,112	07/17/2000	8:32 a.m.

Total Source Intensity and Source Intensity Fractions

Calculation of Gamma Source Intensity Fractions

This calculation reflects the source definition setup in the mcnp input.
It should be noted that the paths for the U inner and outer elements in each of the 12 baskets are specified in sdef (source definition) deck. Therefore, the source fractions in each of the 12 baskets for the U inner and outer elements and in each glass canister are calculated.

Gamma intensity per MCO (photons/s) = 6.45E+15 (see Table 11)
N-Reactor peaking factor (see Assumption 3.3) = 2
Gamma intensity per MCO * peaking factor (photons/s) = 1.29E+16
Gamma intensity per 4.5-m (DHLW SRS glass) canister (photons/s) = 6.22E+15 (see Table 8)
Gamma intensity per waste package (photons/s) = 2*gamma intensity per MCO canister + 2*gamma intensity per glass canister = 2*1.29E+16 + 2*6.22E+15 = **3.82E+16**

Source fraction in each glass canister = gamma intensity per glass canister/gamma intensity per waste package = **0.1627**
Source fraction in each MCO canister = gamma intensity per MCO canister/gamma intensity per waste package = 0.3373
Source fraction in each of the 6 baskets per MCO = fraction in each MCO/6 = 0.3373/6 = 0.0562

U volume fractions in the inner and outer elements are calculated first.
Source intensities of the inner and outer U elements are proportional to U mass of the elements, i.e., the volumes of U inner and outer elements.

Using the dimensions provided in Table 9, derive the inner and outer radii of the inner and outer U elements.
Inner U element dimensions

inner radius (cm) = inner diameter/2 + thickness of the inner Zircaloy layer, for the inner element = 1.1176/2+0.0635 = 0.6223
outer radius (cm) = outer diameter/2 - thickness of the outer Zircaloy layer, for the inner element = 3.175/2-0.1015 = 1.486
length (cm) = fuel length-2*end-cap length = 53.086-2*0.483 = 52.12
volume (cm^3) = π*U length*(outer radius^2-inner radius^2) = $\pi*52.12*(1.486^2-0.6223^2)$ = 298.1602

Outer U element dimensions

inner radius (cm) = inner diameter/2 + thickness of the inner Zircaloy layer, for the outer element = 4.4958/2+0.0555 = 2.3034

outer radius (cm) = outer diameter/2 - thickness of the outer Zircaloy layer, for the outer element = 6.096/2-0.0635 = 2.9845

length (cm) = fuel length-2*end-cap length = 53.086-2*0.483 = 52.12

volume (cm^3) = π*U length*(outer radius^2-inner radius^2) = $\pi*52.12*(2.9845^2-2.3034^2)$ = 589.723

Total uranium volume per fuel elements (cm^3) = 298.1602+589.7234 = 887.884

U volume fraction in the inner element: U volume of the inner element/total U volume per elements = 298.1602/887.8836 = 0.3358

U volume fraction in the outer element: U volume of the outer element/total U volume per elements = 589.7234/887.8836 = 0.6642

Source fraction in the inner U element per each of the 6 baskets of the MCO canister = 0.0562*0.3358 = **0.0189**

Source fraction in the outer U element per each of the 6 baskets of the MCO canister = 0.0562*0.6642 = **0.0373**

Calculation of Neutron Source Intensity Fractions

This calculation reflects the source definition setup in the mcnp input.
It should be noted that the paths for the U inner and outer elements in each of the 12 baskets are specified in sdef deck.
Therefore, the source fractions in each of the 12 baskets for the U inner and outer elements and in each glass canister are calculated.

Neutron intensity per MCO (neutrons/s) = 1.17E+07 (see p. 14)
N-Reactor peaking factor (see Assumption 3.3) = 2
Neutron intensity per MCO * peaking factor (neutrons/s) = 2.34E+07

Neutron intensity per 4.5-m (DHLW SRS glass) canister (neutrons/s) = 1.39E+08 (see Table 8)

Neutron intensity per waste package (neutrons/s) = 2*neutron intensity per MCO canister + 2*neutron intensity per glass canister = 2*2.34E+07 + 2*1.39E+08 = **3.25E+08**
Source fraction in each glass canister = neutron intensity per glass canister/neutron intensity per waste package = **0.4280**
Source fraction in each MCO canister = neutron intensity per MCO canister/neutron intensity per waste package = 0.0720
Source fraction in each of the 6 baskets per MCO = fraction in each MCO/6 = 0.0720/6 = 0.0120
Source fraction in the inner U element per each of the 6 baskets of the MCO canister = 0.012*0.3358 = **0.0040**
Source fraction in the outer U element per each of the 6 baskets of the MCO canister = 0.012*0.6642 = **0.0080**

Segment Areas Used in MCNP Tally Specifications

MCNP is unable to calculate the areas of azimuthal segments because these segments have asymmetric shapes.

The following formula was used to calculate segment areas (see Figures 2 and 3 for segment identifiers and sizes):

$$\text{Segment Area} = 2 * \pi * \text{Surface Radius} * \text{Segment Height}/4$$

Table II-1. Surface Radii Used in MCNP Tally Specifications

Surface	Radius (cm)
Waste package cavity radial surface	79.2
Outer radial surface of the inner shell	84.2
Outer radial surface of the outer shell	86.7
Radial surface 1 m from the waste package	186.7
Radial surface 2 m from the waste package	286.7

Table II-2. Segment Areas Used in MCNP Tally Specifications

Radius (cm)		79.2	84.2	86.7	186.7	286.7
Segment	Height (cm)	Area (cm^2)				
1	46.825	5825.3610	6193.1237	6377.0050	13732.2588	21087.5126
2	46.825	5825.3610	6193.1237	6377.0050	13732.2588	21087.5126
3	68.11	8473.3655	9008.3002	9275.7675	19974.4613	30673.1551
4	68.11	8473.3655	9008.3002	9275.7675	19974.4613	30673.1551
5	68.11	8473.3655	9008.3002	9275.7675	19974.4613	30673.1551
6	68.11	8473.3655	9008.3002	9275.7675	19974.4613	30673.1551
7	68.11	8473.3655	9008.3002	9275.7675	19974.4613	30673.1551
8	27.5	3421.1944	3637.1789	3745.1711	8064.8610	12384.5509

Gamma, Neutron, and Total Surface Dose Rates

Table III-1. Dose Rates on the Radial Surface of the WP Cavity: Angular Segment a

	Gamma		Neutron		Total	
Axial Location	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	182.5810	0.0618	0.0465	0.0106	182.6275	11.2835
Segment 2	1159.5600	0.0208	0.1193	0.0067	1159.6793	24.1188
Segment 3	2715.7700	0.0106	0.2423	0.0040	2716.0123	28.7872
Segment 4	2950.4900	0.0102	0.2901	0.0037	2950.7801	30.0950
Segment 5	2929.3900	0.0102	0.2967	0.0036	2929.6867	29.8798
Segment 6	2910.8100	0.0103	0.2947	0.0037	2911.1047	29.9813
Segment 7	2602.8100	0.0107	0.2583	0.0040	2603.0683	27.8501
Segment 8	1110.2000	0.0243	0.1961	0.0064	1110.3961	26.9779

NOTE: See Figures 2 and 3 for segment locations.

Table III-2. Dose Rates on the Radial Surface of the WP Cavity: Angular Segment b

	Gamma		Neutron		Total	
Axial Location	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	546.7070	0.0353	0.0864	0.0079	546.7934	19.2988
Segment 2	1956.5200	0.0153	0.1611	0.0057	1956.6811	29.9348
Segment 3	6301.9500	0.0075	0.3679	0.0034	6302.3179	47.2646
Segment 4	6858.0500	0.0109	0.4283	0.0031	6858.4783	74.7527
Segment 5	6845.6500	0.0078	0.4339	0.0031	6846.0839	53.3961
Segment 6	6845.1600	0.0070	0.4324	0.0031	6845.5924	47.9161
Segment 7	6672.6600	0.0078	0.4054	0.0032	6673.0654	52.0467
Segment 8	5295.9400	0.0113	0.3127	0.0053	5296.2527	59.8441

NOTE: See Figures 2 and 3 for segment locations.

Table III-3. Dose Rates on the Inner Surface of the Outer Shell: Angular Segment a

	Gamma		Neutron		Total	
Axial Location	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	4.0739	0.1030	0.0214	0.0091	4.0953	0.4196
Segment 2	61.8678	0.0288	0.0601	0.0059	61.9279	1.7818
Segment 3	156.2260	0.0151	0.1264	0.0036	156.3524	2.3590
Segment 4	169.6630	0.0148	0.1499	0.0033	169.8129	2.5110
Segment 5	165.7320	0.0145	0.1528	0.0032	165.8848	2.4031
Segment 6	163.2280	0.0149	0.1524	0.0032	163.3804	2.4321
Segment 7	150.7670	0.0155	0.1342	0.0035	150.9012	2.3369
Segment 8	63.4503	0.0366	0.1023	0.0055	63.5526	2.3223

NOTE: See Figures 2 and 3 for segment locations.

Table III-4. Dose Rates on the Inner Surface of the Outer Shell: Angular Segment b

Axial Location	Gamma		Neutron		Total	
	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	12.6900	0.0577	0.0402	0.0069	12.7302	0.7322
Segment 2	97.8763	0.0232	0.0809	0.0050	97.9572	2.2707
Segment 3	363.1510	0.0103	0.1876	0.0030	363.3386	3.7405
Segment 4	387.5000	0.0101	0.2186	0.0027	387.7186	3.9138
Segment 5	397.1450	0.0100	0.2202	0.0027	397.3652	3.9715
Segment 6	382.5710	0.0100	0.2201	0.0027	382.7911	3.8257
Segment 7	379.0350	0.0102	0.2072	0.0028	379.2422	3.8662
Segment 8	311.9920	0.0177	0.1562	0.0047	312.1482	5.5223

NOTE: See Figures 2 and 3 for segment locations.

Table III-5. Dose Rates on the WP Outer Radial Surface: Angular Segment a

Axial Location	Gamma		Neutron		Total	
	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	0.5840	0.1163	0.0080	0.0095	0.5920	0.0679
Segment 2	10.9886	0.0326	0.0237	0.0060	11.0123	0.3582
Segment 3	28.3386	0.0168	0.0507	0.0037	28.3893	0.4761
Segment 4	30.6718	0.0167	0.0600	0.0033	30.7318	0.5122
Segment 5	30.2872	0.0165	0.0611	0.0033	30.3483	0.4997
Segment 6	28.8208	0.0163	0.0608	0.0033	28.8816	0.4698
Segment 7	27.1645	0.0173	0.0537	0.0036	27.2182	0.4699
Segment 8	12.0115	0.0433	0.0408	0.0056	12.0523	0.5201

NOTE: See Figures 2 and 3 for segment locations.

Table III-6. Dose Rates on the WP Outer Radial Surface: Angular Segment b

Axial Location	Gamma		Neutron		Total	
	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	1.9423	0.0664	0.0155	0.0071	1.9578	0.1290
Segment 2	17.5632	0.0282	0.0327	0.0051	17.5959	0.4953
Segment 3	67.2640	0.0122	0.0776	0.0030	67.3416	0.8206
Segment 4	72.1133	0.0118	0.0900	0.0028	72.2033	0.8509
Segment 5	74.9907	0.0119	0.0907	0.0028	75.0814	0.8924
Segment 6	71.9315	0.0118	0.0906	0.0028	72.0221	0.8488
Segment 7	70.8807	0.0120	0.0855	0.0029	70.9662	0.8506
Segment 8	58.3799	0.0210	0.0643	0.0048	58.4442	1.2260

NOTE: See Figures 2 and 3 for segment locations.

Table III-7. Dose Rates on a Radial Surface 1 m from the WP Outer Radial Surface: Angular Segment a

Axial Location	Gamma		Neutron		Total	
	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	2.8694	0.0247	0.0064	0.0028	2.8758	0.0709
Segment 2	6.2979	0.0181	0.0102	0.0025	6.3081	0.1140
Segment 3	10.3750	0.0125	0.0150	0.0020	10.3900	0.1297
Segment 4	13.2395	0.0113	0.0188	0.0018	13.2583	0.1496
Segment 5	13.6709	0.0109	0.0200	0.0017	13.6909	0.1490
Segment 6	13.1992	0.0112	0.0191	0.0018	13.2183	0.1478
Segment 7	10.5606	0.0126	0.0157	0.0020	10.5763	0.1331
Segment 8	6.6028	0.0214	0.0116	0.0027	6.6144	0.1413

NOTE: See Figures 2 and 3 for segment locations.

Table III-8. Dose Rates on a Radial Surface 1 m from the WP Outer Radial Surface: Angular Segment b

Axial Location	Gamma		Neutron		Total	
	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	5.0486	0.0207	0.0086	0.0026	5.0572	0.1045
Segment 2	10.6846	0.0151	0.0134	0.0022	10.6980	0.1613
Segment 3	18.9312	0.0105	0.0198	0.0018	18.9510	0.1988
Segment 4	24.2169	0.0091	0.0249	0.0016	24.2418	0.2204
Segment 5	25.4915	0.0089	0.0264	0.0015	25.5179	0.2269
Segment 6	24.7618	0.0090	0.0255	0.0016	24.7873	0.2229
Segment 7	20.8607	0.0100	0.0211	0.0018	20.8818	0.2086
Segment 8	14.4066	0.0163	0.0155	0.0024	14.4221	0.2348

NOTE: See Figures 2 and 3 for segment locations.

Table III-9. Dose Rates on a Radial Surface 2 m from the WP Outer Radial Surface: Angular Segment a

Axial Location	Gamma		Neutron		Total	
	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	2.8865	0.0191	0.0050	0.0023	2.8915	0.0551
Segment 2	4.3269	0.0162	0.0064	0.0021	4.3333	0.0701
Segment 3	5.9791	0.0123	0.0082	0.0018	5.9874	0.0735
Segment 4	7.6443	0.0109	0.0098	0.0016	7.6541	0.0833
Segment 5	8.1468	0.0105	0.0104	0.0016	8.1572	0.0855
Segment 6	7.5072	0.0109	0.0099	0.0016	7.5171	0.0818
Segment 7	6.1204	0.0121	0.0084	0.0017	6.1288	0.0741
Segment 8	4.7065	0.0202	0.0070	0.0024	4.7135	0.0951

NOTE: See Figures 2 and 3 for segment locations.

Table III-10. Dose Rates on a Radial Surface 2 m from the WP Outer Radial Surface: Angular Segment b

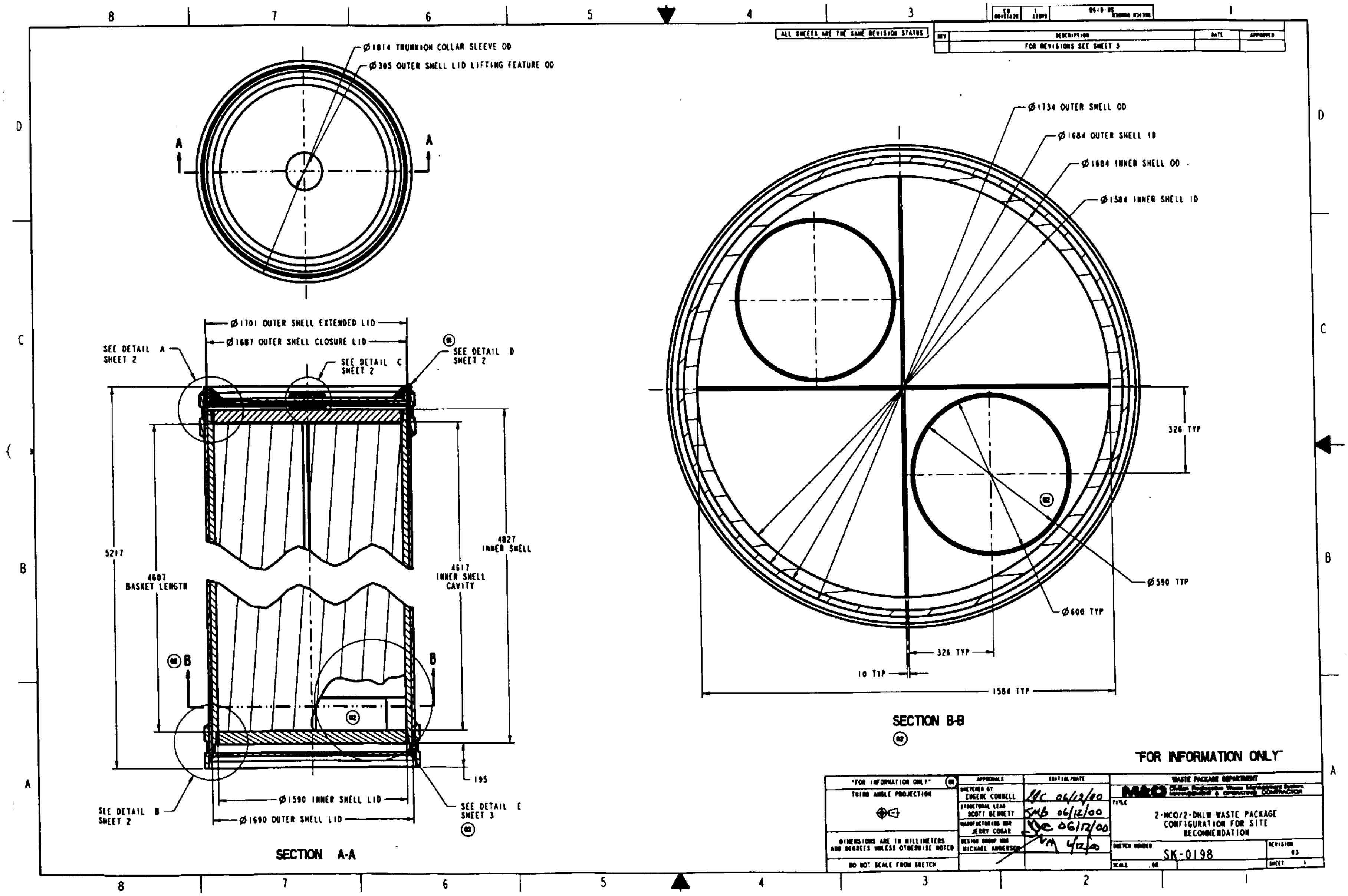
Axial Location	Gamma		Neutron		Total	
	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Segment 1	4.8044	0.0165	0.0063	0.0021	4.8107	0.0793
Segment 2	7.1070	0.0136	0.0081	0.0019	7.1151	0.0967
Segment 3	10.3162	0.0103	0.0104	0.0016	10.3266	0.1063
Segment 4	12.8193	0.0092	0.0124	0.0015	12.8317	0.1179
Segment 5	13.9082	0.0087	0.0131	0.0015	13.9213	0.1210
Segment 6	13.3631	0.0090	0.0126	0.0015	13.3757	0.1203
Segment 7	10.8356	0.0100	0.0107	0.0016	10.8463	0.1084
Segment 8	8.3807	0.0158	0.0088	0.0022	8.3895	0.1324

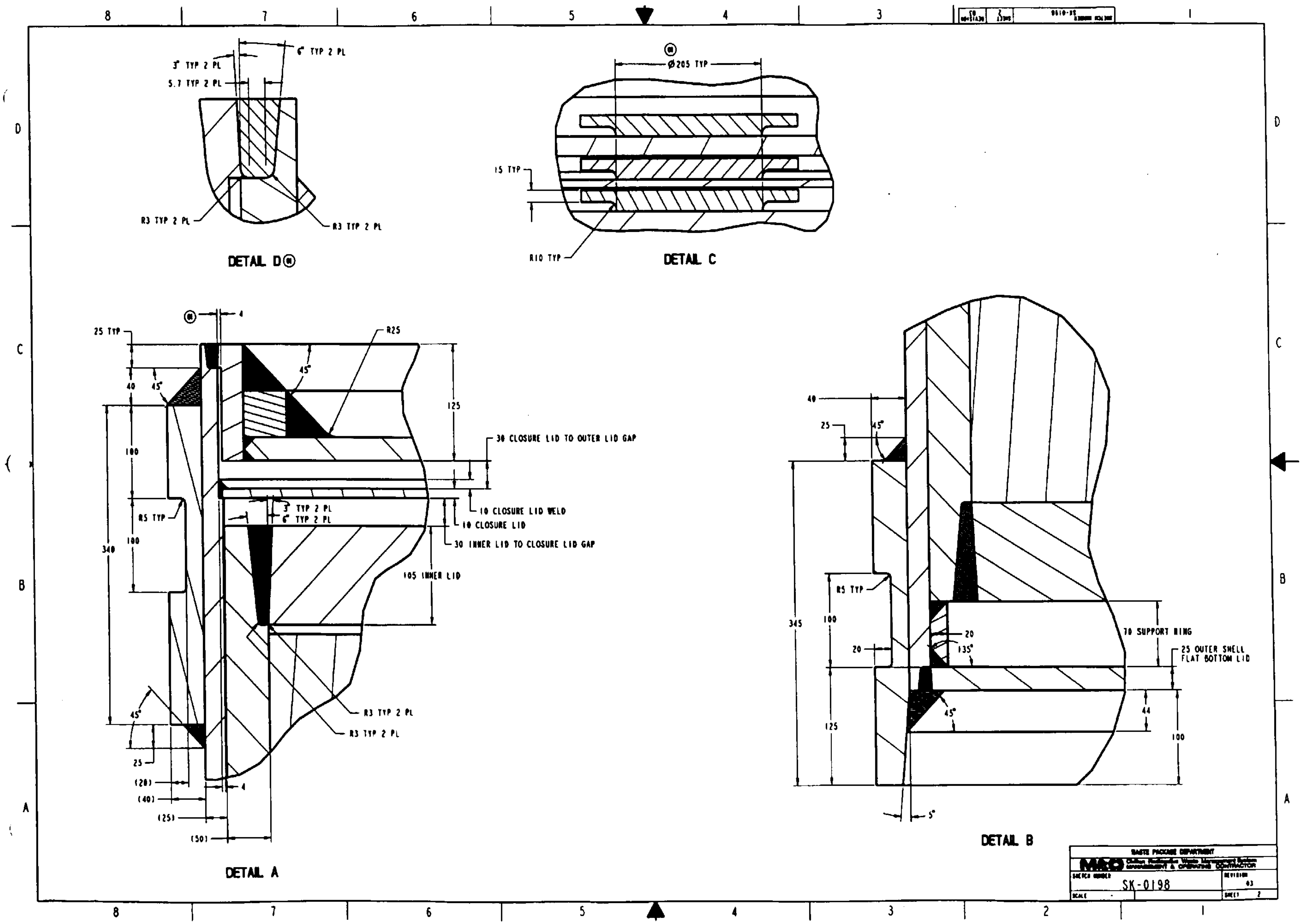
NOTE: See Figures 2 and 3 for segment locations.

Table III-11. Dose Rates on Segments of the Axial Surfaces

Axial Surface	Segment	Gamma		Neutron		Total	
		Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Relative Error	Dose Rate (rem/h)	Standard Deviation (rem/h)
Top surface of the lower inner shell lid	Segment 9	5623.7700	0.0261	0.7340	0.0054	5624.5040	146.7804
	Segment 10	3619.5200	0.0084	0.2754	0.0028	3619.7954	30.4040
Top surface of the lower outer shell lid	Segment 9	5.3455	0.1157	0.1022	0.0062	5.4476	0.6185
	Segment 11	9.0552	0.0273	0.0744	0.0026	9.1295	0.2472
Bottom surface of WP	Segment 9	1.1475	0.1444	0.0431	0.0064	1.1907	0.1657
	Segment 12	1.7436	0.0312	0.0293	0.0026	1.7729	0.0544
Bottom surface 1 m from the WP	Bottom surface 1 m from the WP	0.6069	0.0186	0.0051	0.0022	0.6120	0.0113
Bottom surface 2 m from the WP	Bottom surface 2 m from the WP	0.3768	0.0141	0.0023	0.0018	0.3791	0.0053
Bottom surface of the upper inner shell lid	Segment 9	264.9840	0.0848	0.0814	0.0146	265.0654	22.4706
	Segment 10	325.9950	0.0162	0.0611	0.0050	326.0561	5.2811
Bottom surface of the extended shell lid base	Segment 9	2.2382	0.2809	0.0311	0.0122	2.2693	0.6287
	Segment 11	2.2985	0.0618	0.0208	0.0050	2.3192	0.1420
Top surface of the extended shell lid base	Segment 9	1.0050	0.2518	0.0254	0.0119	1.0304	0.2531
	Segment 11	1.1303	0.0664	0.0167	0.0050	1.1470	0.0750
Top surface of the WP	Segment 9	0.2411	0.2819	0.0102	0.0130	0.2513	0.0680
	Segment 12	0.2644	0.0785	0.0064	0.0052	0.2709	0.0208
Top surface 1 m from the WP	Top surface 1 m from the WP	0.1651	0.0304	0.0014	0.0035	0.1665	0.0050
Top surface 2 m from the WP	Top surface 2 m from the WP	0.1597	0.0184	0.0009	0.0023	0.1606	0.0029

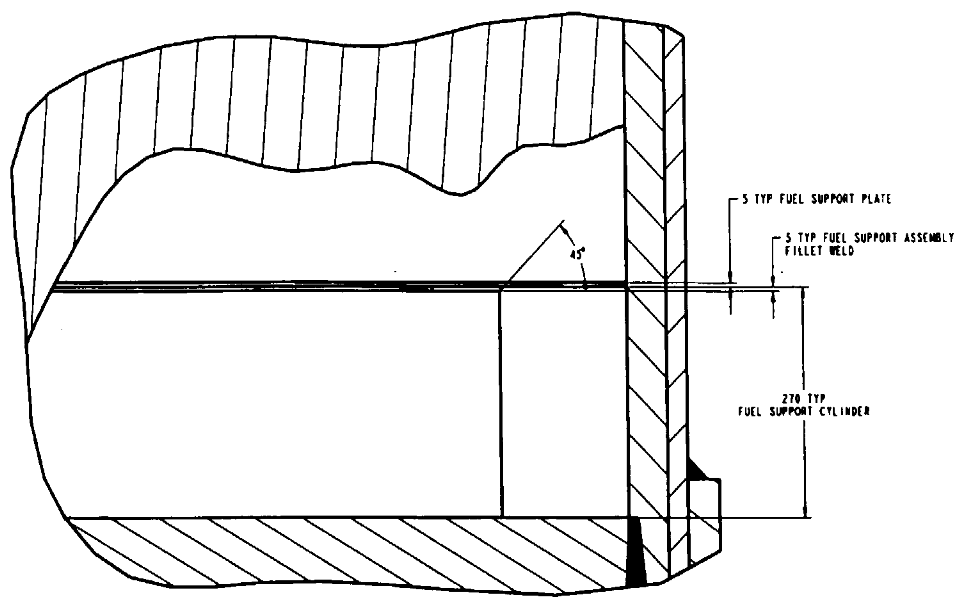
NOTE: See Figure 3 for segment locations.


ALL SHEETS ARE THE SAME REVISION STATUS
FOR REVISIONS SEE SHEET 3
Ø1814 TRUNNION COLLAR SLEEVE OD
Ø305 OUTER SHELL LID LIFTING FEATURE OD
A
A
Ø1701 OUTER SHELL EXTENDED LID
Ø1687 OUTER SHELL CLOSURE LID
SEE DETAIL A SHEET 2
SEE DETAIL C SHEET 2
SEE DETAIL D SHEET 2
5217
4607 BASKET LENGTH
4617 INNER SHELL CAVITY
4827 INNER SHELL
B
B
195
Ø1590 INNER SHELL LID
Ø1690 OUTER SHELL LID
SEE DETAIL B SHEET 2
SEE DETAIL E SHEET 3
SECTION A-A
Ø1734 OUTER SHELL OD
Ø1684 OUTER SHELL ID
Ø1604 INNER SHELL OD
Ø1584 INNER SHELL ID
326 TYP
Ø590 TYP
Ø600 TYP
326 TYP
10 TYP
1584 TYP
SECTION B-B
"FOR INFORMATION ONLY"
"FOR INFORMATION ONLY"
THIRD ANGLE PROJECTION
DIMENSIONS ARE IN MILLIMETERS AND DEGREES UNLESS OTHERWISE NOTED
DO NOT SCALE FROM SKETCH
APPROVALS
INITIAL/DATE
SKETCHED BY EUGENE CONNELL
EJC 06/12/00
STRUCTURAL LEAD SCOTT BENNETT
SMB 06/12/00
MANUFACTURING ENR JERRY COGAR
JJC 06/12/00
DESIGN GROUP MGR MICHAEL ANDERSON
WM 6/12/00
WASTE PACKAGE DEPARTMENT
M&O
TITLE
2-MCO/2-DHLW WASTE PACKAGE CONFIGURATION FOR SITE RECOMMENDATION
SKETCH NUMBER
SK-0198
REVISION
03
SCALE .04
SHEET 1


6° TYP 2 PL
3° TYP 2 PL
5.7 TYP 2 PL
R3 TYP 2 PL
R3 TYP 2 PL
DETAIL D
Ø205 TYP
15 TYP
R10 TYP
DETAIL C
25 TYP
4
R25
40
45°
45°
125
100
30 CLOSURE LID TO OUTER LID GAP
R5 TYP
100
340
3° TYP 2 PL
6° TYP 2 PL
10 CLOSURE LID WELD
10 CLOSURE LID
30 INNER LID TO CLOSURE LID GAP
105 INNER LID
45°
R3 TYP 2 PL
R3 TYP 2 PL
25
(20)
(40)
(25)
(50)
4
DETAIL A
40
25
45°
R5 TYP
345
100
20
20
135°
70 SUPPORT RING
25 OUTER SHELL FLAT BOTTOM LID
44
45°
125
100
5°
DETAIL B
WASTE PACKAGE DEPARTMENT
SK-0198

REVISION TABLE

REV	DESCRIPTION	DRW BY	DATE
00	ISSUED APPROVED	DM	01/25/00
01	MASS 9743 "WAS" 9730, MASS 1641 "WAS" 1647, MASS 5897 "WAS" 4670, MASS 561 "WAS" 558, MASS 550 "WAS" 547, MASS 276 "WAS" 274, MASS 23801 "WAS" 21846, MASS 49300 "WAS" 48066, THICKNESS 25 "WAS" 20, SKETCHED BY "WAS" ORIGINATOR, "CREATED" DETAIL D, "ADDED" Ø TO DETAIL C, "DELETED" WELD DIMENSIONS FROM DETAIL A	BH	03/09/00
02	MASS 23159 "WAS" 23081, MASS 49378 "WAS" 49300, "ADDED" FUEL SUPPORT ASSEMBLY, "MODIFIED COMPONENTS LIST, "CREATED" DETAIL E, MOVED COMPONENTS LIST TO SHEET 3, "MOVED" REVISION TABLE TO SHEET 3, "MODIFIED REVISION TABLE FORMAT, "ADDED" DIMENSIONS TO SECTION B-B.	BH	05/18/00
03	DOE/RW-0351 "WAS" DOE/RW-0315P, "ADDED" NEW FORMAT, "MODIFIED" REVISION TABLE	EJC	06/09/00



DETAIL E

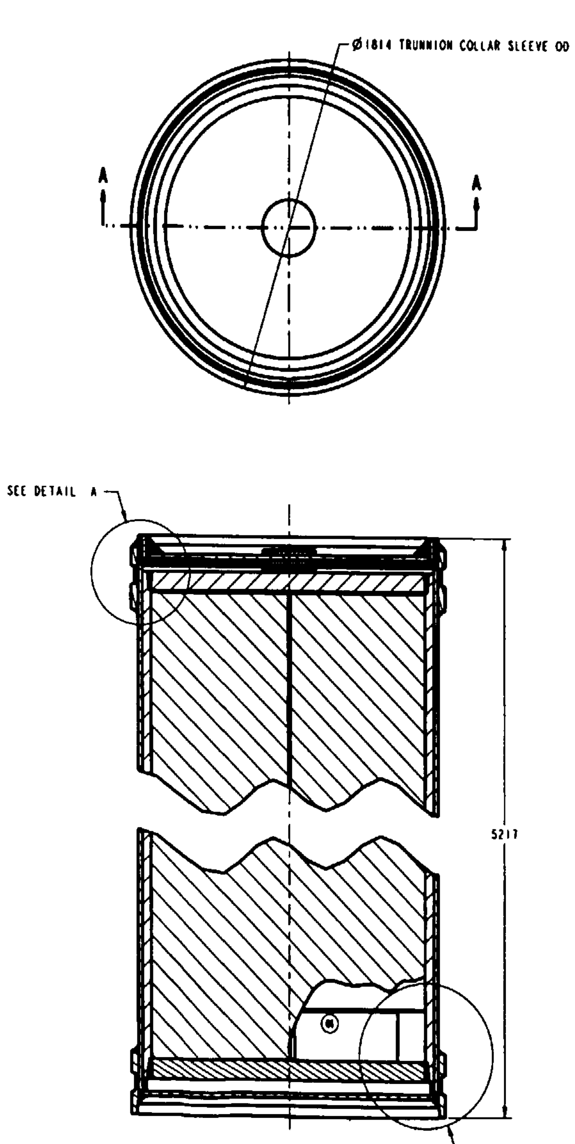
COMPONENT NAME	MATERIAL	THICKNESS	MASS (KG)	QTY RQD
A-PLATE	SA-516 K02700	10	571	2
02 FUEL SUPPORT PLATE	SA-516 K02700	5	19	2
02 FUEL SUPPORT CYLINDER	SA-516 K02700	5	20	2
INNER SHELL	SA-240 S31600	50	01 9743	1
INNER SHELL LID	SA-240 S31600	105	01 1641	2
INNER LID LIFTING FEATURE	SA-240 S31600	27	12	1
OUTER SHELL	SB-575 N06022	01 25	01 5897	1
EXTENDED OUTER SHELL LID	SB-575 N06022	25	146	1
EXTENDED OUTER SHELL LID BASE	SB-575 N06022	25	450	1
EXTENDED OUTER LID REINFORCING RING	SB-575 N06022	50	108	1
OUTER LID LIFTING FEATURE	SB-575 N06022	27	13	2
OUTER SHELL FLAT CLOSURE LID	SB-575 N06022	10	194	1
OUTER SHELL FLAT BOTTOM LID	SB-575 N06022	25	404	1
UPPER TRUNNION COLLAR SLEEVE	SB-575 N06022	40	01 561	1
LOWER TRUNNION COLLAR SLEEVE	SB-575 N06022	40	01 550	1
INNER SHELL SUPPORT RING	SB-575 N06022	20	45	1
TOTAL ALLOY 22 WELDS	SFA-5.14 N06022	-	01 276	***
TOTAL 316 WELDS	SFA-5.9 S31680	-	164	***
02 TOTAL CARBON STEEL WELDS	SFA-5.18 K10726	-	0.37	***
WASTE PACKAGE ASSEMBLY	-	-	01 02 23159	1
HLW GLASS ASSEMBLY	-	-	4200*	2
MCO	-	-	8909.6**	2
WP ASSEMBLY WITH SNF	-	-	01 02 49378	1

* WASTE ACCEPTANCE SYSTEM REQUIREMENTS DOCUMENT. E00000000-00811-1708-00001 REV 03, 03 DOE/RW-0351. ACC: MOO.19990226.0001, PAGE 18, SECTION 4.2.3.1.A.4.

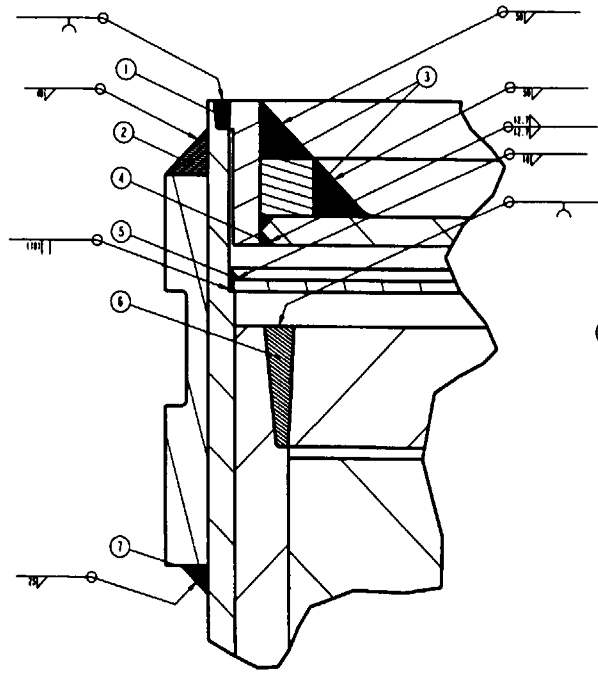
** DES HANFORD 1997. MULTI-CANISTER OVERPACK DESIGN REPORT. HNF-SD-SNF-DR-003 REV 0. JUNE 9, 1997. RICHLAND, WASHINGTON: U.S. DEPARTMENT OF ENERGY, RICHLAND OPERATIONS OFFICE, DUKE ENGINEERING SERVICES HANFORD, INC. ACC: MOL.19980625.0219.

*** SEE SK-0199 FOR WELD CONFIGURATION AND MASSES.

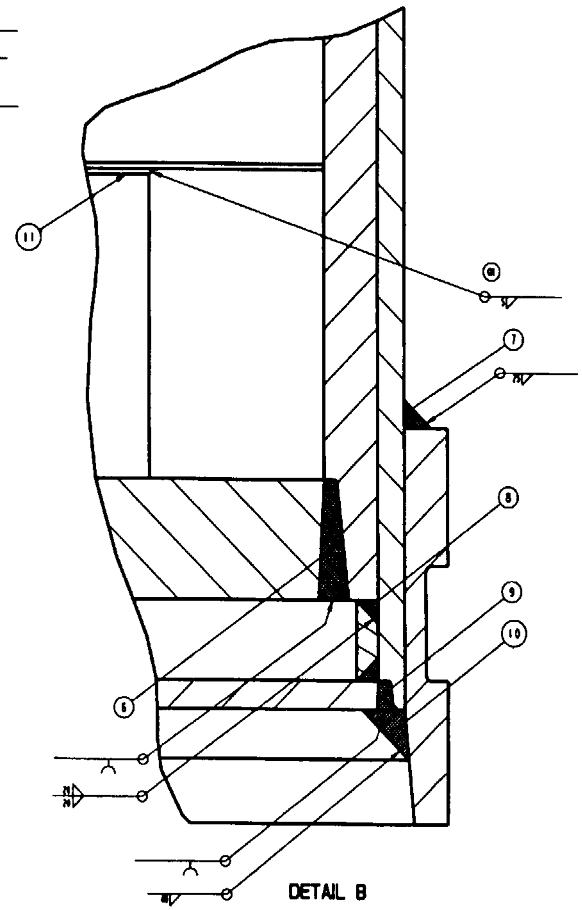
WASTE PACKAGE DEPARTMENT
M&O Civilian Radioactive Waste Management System MANAGEMENT & OPERATING CONTRACTOR
SKETCH NUMBER SK-0198 | REVISION 03
SCALE 40 | SHEET 3



SECTION A-A



DETAIL A



DETAIL B

REVISION TABLE			
REV	DESCRIPTION	REV BY	DATE
00	ISSUED APPROVED	BH	03/09/00
01	FUEL SUPPORT ASSEMBLY WELD ADDED TO WP ASSEMBLY	BH	05/09/00
01	WELD SYMBOL FOR FUEL SUPPORT ASSEMBLY WELD ADDED TO DETAIL B	BH	05/09/00
01	FUEL SUPPORT ASSEMBLY WELD ADDED TO WELD TABLE AS WELD 11	BH	05/09/00
01	TOTAL CARBON STEEL WELDS ADDED TO WELD TABLE	BH	05/09/00
01	LOCATION OF DETAIL B FROM SECTION A-A WAS MODIFIED	BH	05/09/00

WELD	MATERIAL	MASS (KG)	QTY RQD
1	SFA-5.14 N06022	15	1
2	SFA-5.14 N06022	38	1
3	SFA-5.14 N06022	107	1
4	SFA-5.14 N06022	3.5	2
5	SFA-5.14 N06022	4.2	1
6	SFA-5.9 S31680	52	2
7	SFA-5.14 N06022	15	2
8	SFA-5.14 N06022	9.1	2
9	SFA-5.14 N06022	15	1
10	SFA-5.14 N06022	41	1
01 11	SFA-5.18 K10726	0.19	2
TOTAL ALLOY 22 WELDS	SFA-5.14 N06022	276	-
TOTAL 316 WELDS	SFA-5.9 S31680	164	-
01 TOTAL CARBON STEEL WELDS	SFA-5.18 K10726	0.38	-

UNITS: mm

DO NOT SCALE FROM SKETCH

"FOR INFORMATION ONLY"

2-MCO / 2-DHLW WASTE PACKAGE WELD CONFIGURATION

SKETCH NUMBER: SK-0199 REV 01
SKETCHED BY: BRYAN HARKINS BH 24 May 00 SMB 05/24/00 5/24/00
DATE: 05/09/00
FILE: /home/pro_library/checkout/sketches/2mco_2dhlw/sk-0199rev01.dwg

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
SPECIAL INSTRUCTION SHEET
Complete Only Applicable Items

1. QA: QA MB 10/14/2

Page: 1 of: ~~1~~ 2

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

2. Record Date 10/03/2000	3. Accession Number ATT-TO MOL.20001016.0006
4. Author Name(s) GEORGETA RADULESCU	5. Author Organization N/A
6. Title DOSE RATE CALCULATION FOR THE 2MCO/2-DHLW WASTE PACKAGE	
7. Document Number(s) CAL-DDC-NU-000002	8. Version REV. 00
9. Document Type DATA	10. Medium CD-ROM
11. Access Control Code PUB	
12. Traceability Designator DC #25516	

13. Comments

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OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
ELECTRONIC SOURCE FILE VERIFICATION

QA: N/A

2 of 2

1. DOCUMENT TITLE: Dose Rate Calculation for the 2-MCO/2-DHLW Waste Package	
2. DOCUMENT IDENTIFIER: CAL-DDC-NU-000002	3. REVISION DESIGNATOR: 00

ELECTRONIC SOURCE FILE INFORMATION

4. ELECTRONIC SOURCE FILE NAME WITH FILE EXTENSION PROVIDED BY THE SOFTWARE: CAL-DDC-NU-000002-REV00.doc		
5. DATE LAST MODIFIED: 10/03/2000	6. ELECTRONIC SOURCE FILE APPLICATION: (I.E., EXCEL, WORD, CORELDRAW) Word 97	7. FILE SIZE IN KILOBYTES: 895
8. FILE LINKAGE INSTRUCTIONS/INFORMATION: N/A		
9. FILE CUSTODIAN: (I.E., DC, OR DC APPROVED CUSTODIAN) DC	10. FILE LOCATION FOR DC APPROVED CUSTODIAN (I.E., SERVER, DIRECTORY) DC GR 10/03/00	
11. PRINTER SPECIFICATION (i. e., HP4SI) INCLUDING POSTSCRIPT INFORMATION (I.E., PRINTER DRIVER) AND PRINTING PAGE SETUP (I.E., LANDSCAPE, 11 X 17 PAPER) HP Laser Jet 5SI; Portrait, 8.5x11 in. paper		
12. COMPUTING PLATFORM USED: (I.E., SUN) PC	13. OPERATING EQUIPMENT USED: (I.E., UNIX, SOLARIS) DOS, WINDOWS	
14. ADDITIONAL HARDWARE/SOFTWARE REQUIREMENT USED TO CREATE FILE(S): N/A	15. ACCESS RESTRICTIONS: (IF ANY) N/A	

COMMENTS/SPECIAL INSTRUCTIONS

16.
N/A

CERTIFICATION

17. NAME (Print and Sign) Georgeta Radulescu Georgeta Radulescu			18. DATE: 10/03/2000
19. ORGANIZATION FCF	20. DEPARTMENT WPD	21. LOCATION/MAIN STOP 1026E/MS 423	22. PHONE 5-4546

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23. DATE RECEIVED: 10/04/2000	24. DATE REVIEWED: 10/13/2000	25. DATE FILES TRANSFERRED: 10/13/2000
26. NAME (Print and Sign): Teri McCoy		27. DATE: 10/13/2000