CRWMS/M&O

# Calculation Cover Sheet

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1. QA: L



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|---|---|---|---|
| 2. Calculation Title<br>Dose Calculations for the Codisposal WP of HLW Glass and the TRIGA SNF | | | |
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**Revision History**

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|---|---|
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# TABLES

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## 1. PURPOSE

This calculation is prepared by the Monitored Geologic Repository (MGR) Waste Package Operations (WPO). The purpose of this calculation is to determine the surface dose rates of a codisposal waste package (WP) containing a centrally located Department of Energy (DOE) standardized 18-in. spent nuclear fuel (SNF) canister, loaded with the TRIGA (Training, Research, Isotopes, General Atomics) SNF. This canister is surrounded by five 3-m long canisters, loaded with Savannah River Site (SRS) high-level waste (HLW) glass. The results are to support the WP design and radiological analyses.

## 2. METHOD

The Monte Carlo radiation transport code, MCNP V4B2LV (Ref. 7.1), is used to calculate average dose rates on the surfaces and segments of the WP.

## 3. ASSUMPTIONS

3.1 The 3-m long SRS canisters are approximated by cylinders of nominal length, wall thickness, and outer diameter. The head and neck of the canisters are neglected. The basis for this is that radiation transport in the upper part of the canister is not affected, because this portion of the canister is empty and the wall thickness is maintained. This assumption is used throughout Section 5.

3.2 The inner and outer brackets and the divider plates supporting the SRS HLW glass canisters in the WP are neglected. The basis for this is that the calculated dose rates on the surfaces of the WP will be conservative, since these structural components that would attenuate neutrons and gamma rays are not modeled. This assumption is used throughout Section 5.

3.3 The contents of the DOE SNF canister are homogenized inside the cavity of canister. The basis for this is that the calculated surface dose rates will be conservative, because the homogenization process essentially decreases the self-shielding of the fuel and moves the radiation source closer to the outer surfaces of the WP allowing more particles to reach the outer surface (Ref. 7.2, p. 23). This assumption is used throughout Section 5.

3.4 The attenuation characteristics of the fresh and spent fuels are identical for gamma radiation. Therefore, the uranium weight fraction for the fresh TRIGA fuel is used in calculations. The basis for this is the engineering judgement, which is discussed further. The shielding properties of a material against gamma radiation depend only on atomic mass and gamma radiation characteristics. The difference between atomic masses of fresh and spent fuels is negligible, because a negligible amount of initial fissile mass is lost due to capture in reactor hardware and moderator, and leakage of the neutrons produced by fission. Fission events in fuel produce fission products and neutrons, conserving fissionable nucleus mass. This assumption is used throughout Section 5.

3.5 Erbium is neglected in the composition of the fresh fuel, because photon and neutron cross sections for erbium are not provided by MCNP-4B2. The basis for this is that the calculated dose rates on the surfaces of the WP will be conservative, since this element that would attenuate neutrons and gamma rays is not represented. This assumption is used throughout Section 5.

3.6 A peaking factor of 1.25 is used for the TRIGA SNF radiation source to bound the axial source distribution. This value is based on the predicted heat profile shown in Ref. 7.3, Fig. 3-19. The basis for this is that the source term, provided by a depletion calculation, was uniformly created within the fuel volume and the axial source profile must be considered in shielding calculations. This assumption is used throughout Section 5.

3.7 The material compositions of alloys ASTM (American Society for Testing and Materials) B 575, ASTM A 516 Grade 70, and stainless steel (SS) 304L, SS 316L, and SS 304 are assumed to be representatives of materials used in the manufacture of nuclear fuel assemblies. Some of these alloys have elements with allowable range of weight percent. For elements with weight percent range, the midpoint value is used, and the weight percent of the most abundant element is adjusted. The basis for this 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.

3.8 The neutron absorber is neglected. The basis for this is that the calculated dose rates on the surfaces of the WP will be conservative, since these components that would attenuate neutrons and gamma rays are not represented. This assumption is used throughout Section 5.

# 4. USE OF COMPUTER SOFTWARE AND MODELS

## 4.1 SOFTWARE APPROVED FOR QA WORK

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 (Ref. 7.1)
- Computer Type: Hewlett Packard (HP) 9000 Series: "Bloom" (Tag: CRWMS-M&O 700887)
- The MCNP V4B2LV computer code is an appropriate tool to determine the dose rate on and near the surface of a WP containing HLW glass and TRIGA SNF.
- 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 input files used are echoed in the output files, which are listed in Table 8-2.

## 4.2 SOFTWARE ROUTINES

- Title: Excel
- Version/Revision Number: Microsoft Excel 97

The Excel spreadsheet program was used to perform simple numeric calculations, as documented in Section 5 of this calculation. The user-defined formulas, input data, and results are documented in sufficient detail in Section 5 to allow independent repetition of the various computations.

## 4.3 MODELS

None.

# 5. CALCULATION

## 5.1 CALCULATION INPUTS

The following sections outline the data used in the calculation of the dose rates on the WP surfaces. Each MCNP calculation requires the following input data: geometry, material, and source parameters. The WP consists of the codisposal waste container, five canisters containing SRS HLW glass, a support tube, and the 18-in. standardized DOE SNF canister containing the TRIGA SNF and its basket assembly. The sketch for the 5-Defense High-Level Waste (DHLW)/DOE spent fuel disposal container, SK-0069 REV 01, is shown in Att. I, and the sketch for the TRIGA DOE SNF basket assembly, SK-0124 REV 00, is shown in Att. II.

The number of digits cited for values converted from English to metric units does not indicate the accuracy; it is an artifact of the conversion process.

Existing data were used in the development of the results; therefore, the use of any result from this calculation for input into documents supporting procurement, fabrication, or construction is required to be identified and tracked as TBV (to be verified) in accordance with appropriate procedures.

### 5.1.1 Disposal Waste Container

The material compositions and dimensions of the codisposal waste container are presented in Tables 5.1.1-1 through 5.1.1-3.

Table 5.1.1-1. Geometry and Material Specifications for the Disposal Waste Container[1]

| Component | Material | Characteristic | Dimension (mm) |
|---|---|---|---|
| Cavity | Air | Diameter | 1,880 |
| | | Height | 3,040 |
| Corrosion allowance shell | ASTM A 516 Grade 70 (K02700) (given in Table 5.1.1-2) | Thickness | 100 |
| Corrosion resistant shell | ASTM B 575 (Alloy 22) (N06022) (given in Table 5.1.1-3) | Thickness | 20 |
| Top and bottom corrosion allowance shell lids | ASTM A 516 Grade 70 | Thickness | 110 |
| Top and bottom corrosion resistant shell lids | ASTM B 575 | Thickness | 25 |
| Closure lid gap | Air | Thickness | 30 |
| Support tube | ASTM A 516 Grade 70 | Outer diameter | 565 |
| | | Inner diameter | 501.5 |
| | | Length | 3,030 |

[1] Attachment I (SK-0069 REV 01).

Table 5.1.1-2. Chemical Composition of ASTM A 516 Grade 70[1]

| Element | Weight Percent Range | Value Used |
|---|---|---|
| Carbon | 0.30 | 0.30 |
| Manganese | 0.85 - 1.20 | 1.025 |
| Phosphorus | 0.035 (max) | 0.035 |
| Sulfur | 0.035 (max) | 0.035 |
| Silicon | 0.15 - 0.40 | 0.275 |
| Iron | Balance | 98.33 |
| Density = 7.85 g/cm$^3$ | | |

[1] Ref. 7.5, p. 10.

Table 5.1.1-3. Chemical Composition of ASTM B 575[1]

| 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 | 20.0 - 22.5 | 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 | 0.020 (max) | 0.02 |
| Phosphorus | 0.02 (max) | 0.02 |
| Nickel | Balance | 54.765 |
| Density = 8.69 g/cm$^3$ | | |

[1] Ref. 7.5, p. 30.

### 5.1.2 HLW Glass Canisters

Tables 5.1.2-1 through 5.1.2-4 provide geometry, material, and source specifications for the HLW glass canisters.

Table 5.1.2-1. Geometry and Material Specifications for HLW Glass Canisters[1]

| Component | Material | Characteristic | Value |
|---|---|---|---|
| Canister wall | SS 304L (given in Table 5.1.2-2) | Outer diameter (mm) | 610 |
| | | Wall thickness (mm) | 9.525 |
| | | Length (mm) | 3,000 |
| Canister content | SRS HLW glass (given in Table 5.1.2-3) | Weight (kg) | 1,682 |
| | | Density[2] ($g/cm^3$) | 2.85 |

[1] Ref. 7.6, pp. 3.3-4 through 3.3-6.
[2] The average fill temperature (i.e., the average temperature of the glass upon completion of filling to 85% of canister volume) is 825 °C. The glass volume per canister when cooled to 25 °C is about 0.59 $m^3$. The density of the glass is about 2.69 $g/cm^3$ at 825 °C and 2.85 $g/cm^3$ at 25 °C (Ref. 7.7, p. 2.2.1.1-4).

Table 5.1.2-2. Chemical Composition of Type 304L Stainless Steel

| Element | Weight Percent Range[1] | Value Used |
|---|---|---|
| Carbon | 0.03 (max) | 0.03 |
| Manganese | 2.0 (max) | 2.00 |
| Phosphorus | 0.045 (max) | 0.045 |
| Sulfur | 0.03 (max) | 0.03 |
| Silicon | 0.75 (max) | 0.75 |
| Chromium | 18 – 20 | 19.00 |
| Nickel | 8 – 12 | 10.00 |
| Nitrogen | 0.10 (max) | 0.10 |
| Iron | Balance | 68.045 |
| Density = 7.94 $g/cm^3$ | | |

[1] Ref. 7.5, p. 17.

Table 5.1.2-3. Chemical Composition of SRS HLW Glass[1, 2]

| Element/Isotope | Weight Percent | Element/Isotope | Weight Percent |
|---|---|---|---|
| $^{6}Li$ | 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 | $^{137}Ba$[4] | 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[3] | 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 at 25 °C = 2.85 g/cm$^3$ | | | |

[1] Ref. 7.8, p. 7.
[2] TBV
[3] Neutron cross-section tables for Zn are missing from the ENDF/B-V library. The Al neutron cross-section tables are used instead because of the similarities between neutron cross sections of these two elements.
[4] The neutron and photon cross sections for $^{138}Ba$ were used in calculations, because the neutron cross-section tables for $^{137}Ba$ are missing from the ENDF/B-V library.

Table 5.1.2-4. Gamma and Neutron Sources of SRS HLW Glass at Day 1 after Pouring[1]

| 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 |

[1] Ref. 7.9, Att. VIII, p. VIII-1, and Att. IX, p. IX-1.

### 5.1.3 DOE SNF Canister

Tables 5.1.3-1 through 5.1.3-5 present the geometry, material, and radiation source specifications for the components of the DOE SNF canister. Three basket assemblies, each containing 37 fuel elements, are stacked in the DOE SNF canister to provide a total of 111 elements per canister (see Att. II).

According to Ref. 7.11, p. B-7, the bounding source term was obtained for the TRIGA FLIP (Fuel Improvement Program) element with the following characteristics: SS clad, 51.09 wt% average U-235 burnup, 8.5 wt% U-235 with 70% enrichment, and 66.52 MWd average burnup. Therefore, this calculation uses the geometry, material, and radiation source specifications for this TRIGA fuel type.

Table 5.1.3-1. Geometry and Material Specifications for the DOE SNF Canister[1]

| Component | Material | Characteristic | Dimension (mm) |
|---|---|---|---|
| Circular cylinder | SS 316L (given in Table 5.1.3-2) | Outer diameter | 457.2 |
| | | Wall thickness | 9.525 |
| | | Internal length | 2,547 |
| Impact disk | ASTM A 516 Grade 70 | Thickness | 50.8 |
| Top and bottom curved plates | SS 316L | Thickness | 9.525 |

[1] Ref. 7.10, p. 5 and Appendix A.

Table 5.1.3-2. Chemical Composition of Type 316L Stainless Steel

| Element | Weight Percent Range[1] | Value Used |
|---|---|---|
| Carbon | 0.03 (max) | 0.03 |
| Manganese | 2.00 (max) | 2.00 |
| Phosphorus | 0.045 (max) | 0.045 |
| Sulfur | 0.03 (max) | 0.03 |
| Silicon | 1.00 (max) | 1.00 |
| 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 | 65.295 |
| Density = 7.98 g/cm$^3$ | | |

[1] Ref. 7.5, p. 13.

Table 5.1.3-3. Material Specifications for the Contents of the DOE SNF Canister

| Component | Sub-Component | Material | Characteristic | Value | Reference |
|---|---|---|---|---|---|
| Fuel (TRIGA-FLIP) | Fuel section | U-238 | Mass/element (g) | 59.0 | Ref. 7.11, p. 25 |
| | | U-235 | Mass/element (g) | 137.0 | |
| | | Zr (as hydrate) | Mass/element (g) | 2,060.0 | |
| | | Erbium[1] | Mass/element (g) | 36.0 | |
| | Zirconium rod | Zr | Mass/element (g) | 63.7 | |
| | Reflectors | Graphite | Mass/element (g) | 450.0 | |
| | Cladding | SS 304 | Mass/element (g) | 270.0 | |
| | End fittings | SS 304 | Mass/element (g) | 530.0 | |
| Basket Assembly | | | Number/canister | 3 | Attachment II |
| | SS tubes | SS 316L | Mass/basket (g) | 235,600.0 | Ref. 7.12, p. 8 |
| | Base plates | SS 316L | Mass/basket (g) | 10,800.0 | |
| | Basket support brackets | SS 316L | Mass/basket (g) | 10,600.0 | |

[1] The cross-section tables for erbium are not provided by MCNP-4B2.

Table 5.1.3-4. Chemical Composition of SS 304

| Element | Weight Percent Range[1] | Value Used |
|---|---|---|
| Carbon | 0.08 (max) | 0.08 |
| Chromium | 18 - 20 | 19.00 |
| Nickel | 8 - 10.5 | 9.25 |
| Manganese | 2.00 (max) | 2.00 |
| Phosphorus | 0.045 (max) | 0.045 |
| Sulfur | 0.03 (max) | 0.03 |
| Silicon | 1.00 (max) | 1.00 |
| Nitrogen | 0.10 (max) | 0.10 |
| Iron | Balance | 68.495 |
| Density[1] = 7.94 g/cm$^3$ | | |

[1] Ref. 7.5, p. 20.

Table 5.1.3-5. Gamma Source at 1-Year Decay Time for the TRIGA FLIP SNF Element [1]

| Upper Energy Boundary[2] (MeV) | Average Group Energy (MeV) | Gamma Intensity (photons/s) |
|---|---|---|
| 0.02 | 1.50E-02 | 3.850E+13 |
| 0.03 | 2.50E-02 | 8.610E+12 |
| 0.05 | 3.75E-02 | 9.140E+12 |
| 0.07 | 5.75E-02 | 7.920E+12 |
| 0.10 | 8.50E-02 | 5.520E+12 |
| 0.15 | 1.25E-01 | 6.770E+12 |
| 0.30 | 2.25E-01 | 4.570E+12 |
| 0.45 | 3.75E-01 | 2.360E+12 |
| 0.70 | 5.75E-01 | 1.840E+13 |
| 1.00 | 8.50E-01 | 1.150E+13 |
| 1.50 | 1.25E+00 | 1.150E+13 |
| 2.00 | 1.75E+00 | 5.820E+10 |
| 2.50 | 2.25E+00 | 2.280E+11 |
| 3.00 | 2.75E+00 | 1.050E+09 |
| 4.00 | 3.50E+00 | 1.120E+08 |
| 6.00 | 5.00E+00 | 2.610E+03 |
| 8.00 | 7.00E+00 | 3.000E+02 |
| 14.00 | 1.10E+01 | 3.450E+01 |
| Total | | 1.251E+14 |

[1] Ref. 7.11, p. B-7.
[2] Ref. 7.4, p. 15. Ref. 7.11, p. B-7 mistakenly lists the following two upper energy boundaries: 8 and 11 MeV for the average group energies 5 and 7 MeV, respectively.

Reference 7.11, p. B-8 provides only the neutron source at 20-year decay time for a TRIGA FLIP fuel rod. Using the activities of the radionuclides contributing to the neutron source at 1- and 20-year decay times, and the neuron intensity at 20-year decay time, the neutron intensity at 1-year decay time can be calculated. The neutron intensity from each contributing nuclide at 1-year decay time is the product of nuclide activity and corresponding neutron production cross section. Since the neutron production cross sections are the same at 1- and 20-year decay times for a given SNF, the neutron production cross section is determined from nuclide activity and neutron intensity at 20-year decay time. Table 5.1.3-6 provides nuclide activities and neutron intensities at 20-year decay time. Attachment III presents the results of neutron source intensity calculation for TRIGA SNF at 1-year decay time.

Table 5.1.3-6. Neutron Source at 20-Year Decay Time for the TRIGA FLIP SNF Element [1]

| Nuclide | 20-Year Decay Time | | |
|---|---|---|---|
| | Activity (Ci) | (α, n) Production (neutrons/s) | Spontaneous Fission (SF) Production (neutrons/s) |
| Bi-211 | 3.78E-08 | 5.51E-03 | |
| Po-212 | 1.72E-05 | 1.84E+01 | |
| Po-215 | 3.78E-08 | 2.57E-01 | |
| Rn-219 | 3.78E-08 | 9.24E-03 | |
| U-235 | 1.19E-04 | 9.71E-03 | 1.82E-02 |
| U-238 | 1.83E-05 | 9.49E-04 | 6.43E-01 |
| Pu-238 | 3.24E-00 | 1.31E+03 | |
| Pu-239 | 9.35E-02 | 4.66E+01 | 4.70E-00 |
| Pu-240 | 7.69E-02 | 2.70E+01 | |
| Am-241 | 7.80E-01 | 4.57E+02 | |
| Total | | 1.86E+03 | 5.36E-00 |

[1] Ref. 7.11, p. B-8.

Data from Reference 7.5 are accepted data, since this document references standard handbooks. Data from standard handbooks are established facts and are considered as accepted data. Data from References 7.3, 7.13, and 7.14 are also considered accepted data, as they are accepted within the scientific and engineering community and are technically defensible. References 7.4, 7.6, 7.7, 7.8, 7.9, 7.10, 7.11, and 7.12 provide existing data.

## 5.2 DESCRIPTION OF CALCULATIONS

The contents of the DOE SNF canister are homogenized inside the canister cavity, and the corresponding atomic densities (AD) are calculated according to the following formula:

$$AD = \frac{\text{mass}_{\text{element or isotope}} * N_A}{\text{volume}_{\text{region}} * \text{atomic mass}_{\text{element or isotope}}}$$

In the above equation, $N_A$ is the Avogadro's Number, which equals 6.0221367E+23 atoms/mol (Ref. 7.13, p. 59). The atomic masses are provided by Ref. 7.13, pp. 18, 20, 24, 28, 42, and 48. These calculations are shown in Attachment IV.

There are two different radiation sources in this calculation, uniformly originating from HLW glass and the homogenized contents of the DOE SNF canister. The total source intensity is used as tally multiplier, while the source intensity fractions of SRS HLW glass canisters and the TRIGA SNF are specified as cell-dependent distribution numbers. Attachment III contains source intensity fractions for the DOE SNF and SRS HLW glass canisters.

In radiation shielding analysis, particle population diminishes considerably due to attenuation in shield materials. Preliminary calculations of this case have shown that the gamma-ray intensity decreases by a factor of about 4 when passing through the inner barrier, and by a factor of about 100 when passing through the outer barrier. Therefore, the use of variance reduction techniques is required to increase sampling in regions of the tally. Geometric splitting/Russian roulette across cell boundaries is the variance reduction technique employed. 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 (Ref. 7.14, p. 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.

In this calculation, the photon flux averaged over a surface tally is used to calculate surface-averaged dose rates in rem/h on selected surfaces of the WP.

The geometric representation for the MCNP calculation is shown in Fig. 5.2-1. Figure 5.2-2 shows the surfaces and segments of WP used in dose rate calculations. The radial surfaces, cut by the bottom and top planes of HLW glass, are equally divided into five segments of 43.2253-cm height (Segments 2 through 6). Segment 1, of 87.8735-cm height, is the portion between the top of WP cavity and the top of HLW glass. The axial dose rates for the surfaces 1 m and 2 m from the WP outer surfaces are averaged on a segment delimited by the outer radius of the WP.

The contribution of the neutron dose to the total dose is negligible. Dose rate calculations for the codisposal WP of HLW glass and Fermi SNF (Ref. 7.4) have shown that the neutron dose contribution to the total dose is insignificant (the maximum dose rate on the outer radial surface of the WP was 10.48 ± 0.38 rem/h for the gamma radiation and 0.08 ± 0.0004 rem/h for the neutron radiation [Ref. 7.4, p. 22]). The neutron source intensities for the codisposal WP of the TRIGA SNF and for the codisposal WP of the Fermi SNF are 4.107E+8 neutrons/s (see Att. III) and 4.104E+8 neutrons/s (Ref. 7.4, p. III-2), respectively. Therefore, the dose rates due to neutron radiation are almost the same for the two WPs. On the other hand, the gamma source intensity for the codisposal WP of the TRIGA SNF is about twice as much as that for the codisposal WP of the Fermi SNF (3.58E+16 photons/s [see Att. III] versus 1.88E+16 photons/s [Ref. 7.4, p. III-2]). Therefore, the neutron dose contribution to the total dose will be even smaller for the codisposal WP of the TRIGA SNF.

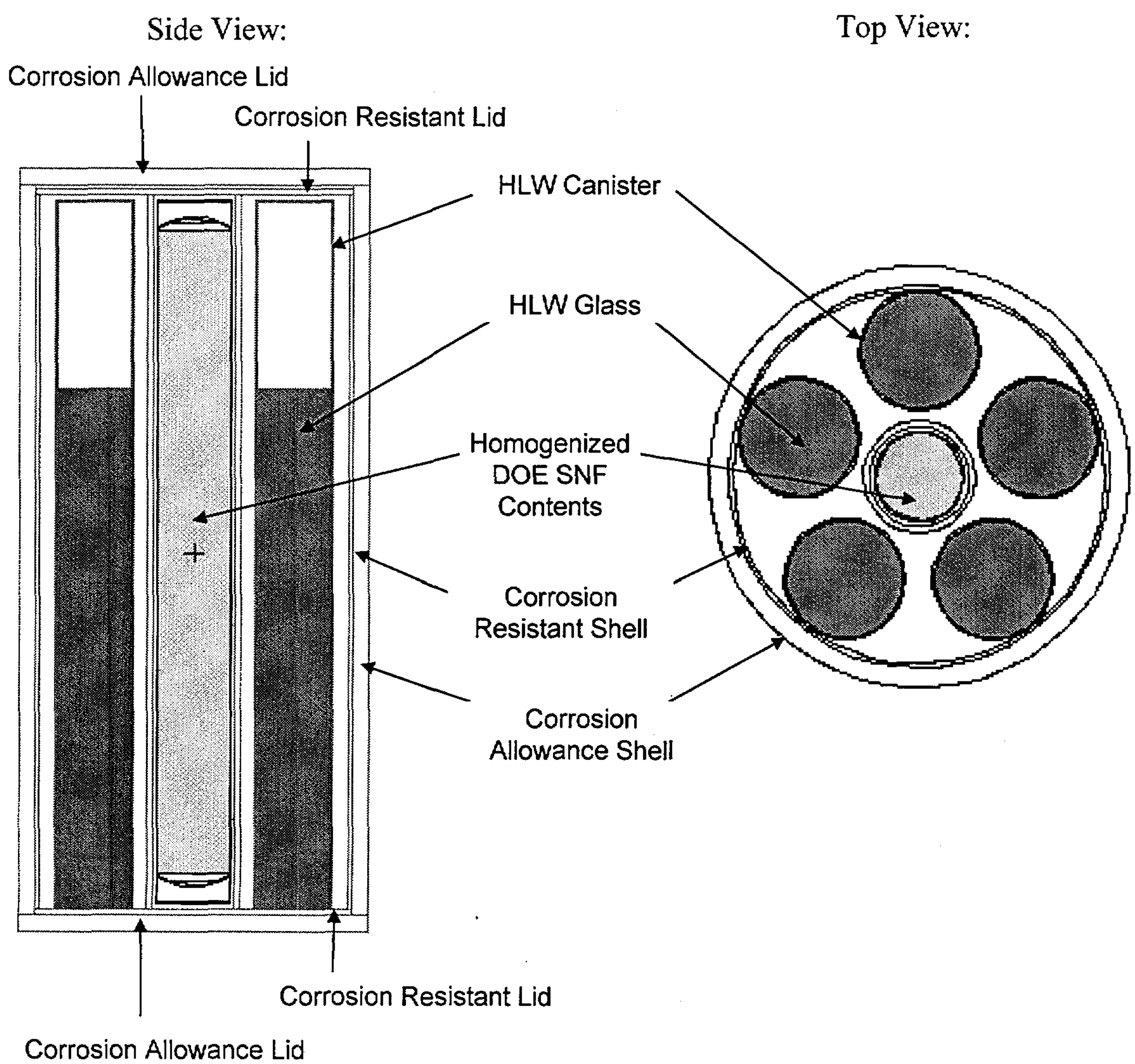


Figure 5.2-1. Cross Sections of the MCNP Geometric Representation

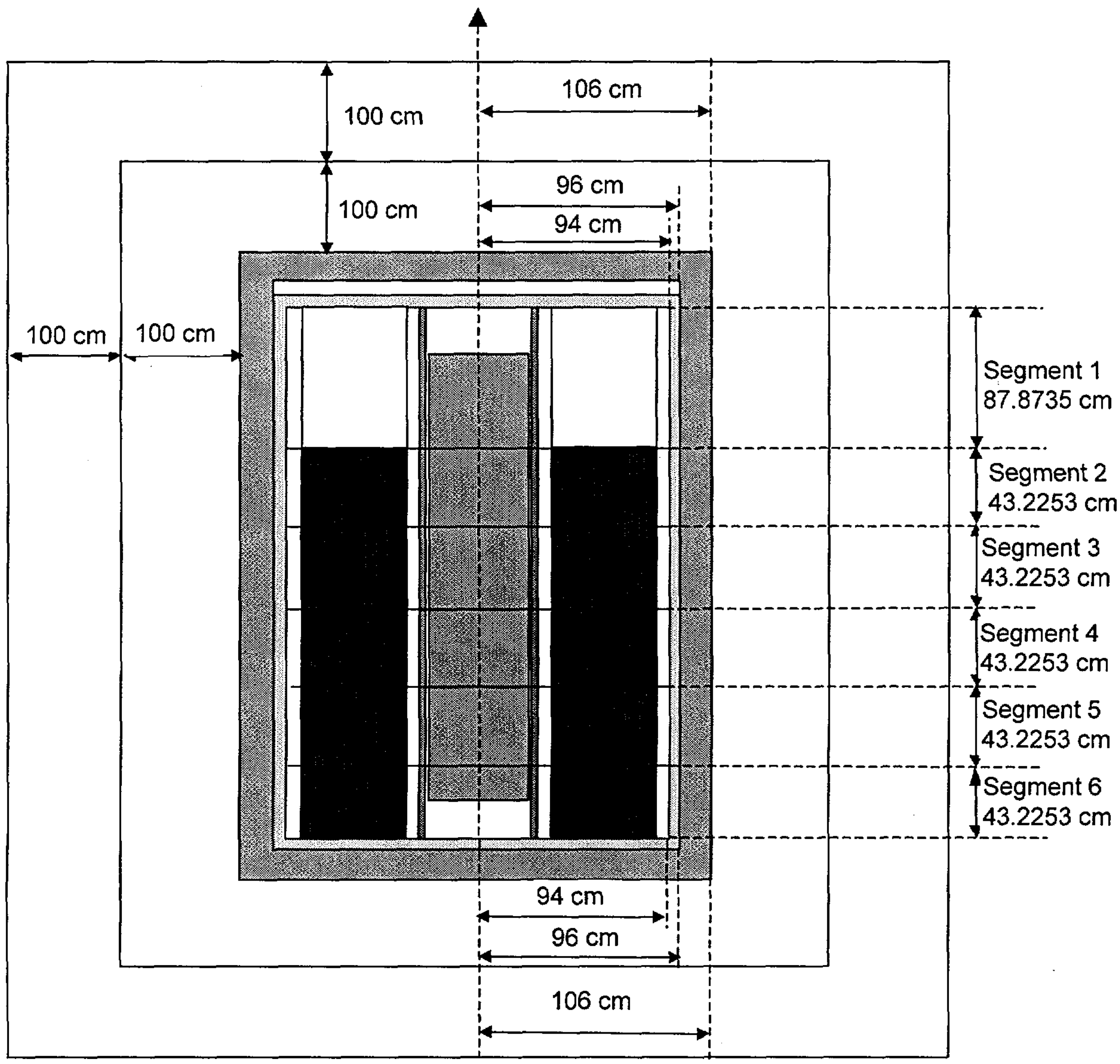


Figure 5.2-2. Surfaces and Segments of the WP Used in Dose Rate Calculations[1]

[1] Drawing not to scale

# 6. RESULTS

Existing data were used in the development of the results presented in this section. Therefore, the use of any result from this calculation for input into documents supporting procurement, fabrication, or construction is required to be identified and tracked as TBV in accordance with appropriate procedures.

Tables 6-1 through 6-6 show the radial and axial gamma dose rates on the WP and at the distances of 1 m and 2 m from the outer surface of the WP containing the SRS HLW glass and the TRIGA SNF.

Table 6-1. Gamma Dose Rates Averaged over the Inner Surface of Corrosion Resistant Shell

| Axial Location (see Fig. 5.2-2) | Dose Rate (rem/h) | Relative Error |
|---|---|---|
| Segment 1 | 2.7035E+03 | 0.0112 |
| Segment 2 | 7.7375E+03 | 0.0101 |
| Segment 3 | 8.8211E+03 | 0.0143 |
| Segment 4 | 8.6417E+03 | 0.0111 |
| Segment 5 | 8.6554E+03 | 0.0096 |
| Segment 6 | 7.3458E+03 | 0.0100 |

Table 6-2. Gamma Dose Rates Averaged over Inner Surface of Corrosion Allowance Shell

| Axial Location (see Fig. 5.2-2) | Dose Rate (rem/h) | Relative Error |
|---|---|---|
| Segment 1 | 7.1099E+02 | 0.0099 |
| Segment 2 | 1.9851E+03 | 0.0102 |
| Segment 3 | 2.2074E+03 | 0.0096 |
| Segment 4 | 2.1999E+03 | 0.0096 |
| Segment 5 | 2.1817E+03 | 0.0100 |
| Segment 6 | 1.8453E+03 | 0.0113 |

Table 6-3. Gamma Dose Rates Averaged over WP Outer Radial Surface

| Axial Location (see Fig. 5.2-2) | Dose Rate (rem/h) | Relative Error |
|---|---|---|
| Segment 1 | 7.3403E+00 | 0.0168 |
| Segment 2 | 1.2978E+01 | 0.0206 |
| Segment 3 | 1.4060E+01 | 0.0201 |
| Segment 4 | 1.3811E+01 | 0.0202 |
| Segment 5 | 1.3489E+01 | 0.0218 |
| Segment 6 | 1.0891E+01 | 0.0238 |

Table 6-4. Gamma Dose Rates Averaged over a Radial Surface 1 m from WP Outer Radial Surface

| Axial Location (see Fig. 5.2-2) | Dose Rate (rem/h) | Relative Error |
|---|---|---|
| Segment 1 | 3.1230E+00 | 0.0151 |
| Segment 2 | 4.6502E+00 | 0.0166 |
| Segment 3 | 5.3604E+00 | 0.0161 |
| Segment 4 | 5.3735E+00 | 0.0163 |
| Segment 5 | 4.8519E+00 | 0.0173 |
| Segment 6 | 3.4261E+00 | 0.0209 |

Table 6-5. Gamma Dose Rates Averaged over a Radial Surface 2 m from WP Outer Radial Surface

| Axial Location (see Fig. 5.2-2) | Dose Rate (rem/h) | Relative Error |
|---|---|---|
| Segment 1 | 1.8928E+00 | 0.0149 |
| Segment 2 | 2.5302E+00 | 0.0180 |
| Segment 3 | 2.6902E+00 | 0.0172 |
| Segment 4 | 2.8084E+00 | 0.0189 |
| Segment 5 | 2.5490E+00 | 0.0185 |
| Segment 6 | 2.0420E+00 | 0.0207 |

Table 6-6. Gamma Dose Rates Averaged over WP Axial Surfaces

| Axial Location (see Fig. 5.2-2) | Dose Rate (rem/h) | Relative Error |
|---|---|---|
| Upper surface of bottom corrosion resistant lid | 6.6258E+03 | 0.0109 |
| Upper surface of bottom corrosion allowance lid | 1.0829E+03 | 0.0128 |
| Bottom surface of WP | 3.5951E+00 | 0.0320 |
| Bottom surface 1 m from WP | 1.8957E+00 | 0.0385 |
| Bottom surface 2 m from WP | 1.0239E+00 | 0.0446 |
| Lower surface of top corrosion resistant lid | 1.3339E+03 | 0.0158 |
| Lower surface of top corrosion allowance lid | 3.1671E+02 | 0.0197 |
| Top surface of WP | 1.7518E+00 | 0.0411 |
| Top surface 1 m from WP | 1.0357E+00 | 0.0480 |
| Top surface 2 m from WP | 6.1704E-01 | 0.0594 |

## 7. REFERENCES

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7.11 Idaho National Engineering and Environmental Laboratory 1999. *TRIGA (UZrH) Fuel Characteristics for Disposal Criticality Analysis*. DOE/SNF/REP-048 Revision 0. Washington, D.C.: DOE. TIC: 244162.

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# 8. ATTACHMENTS

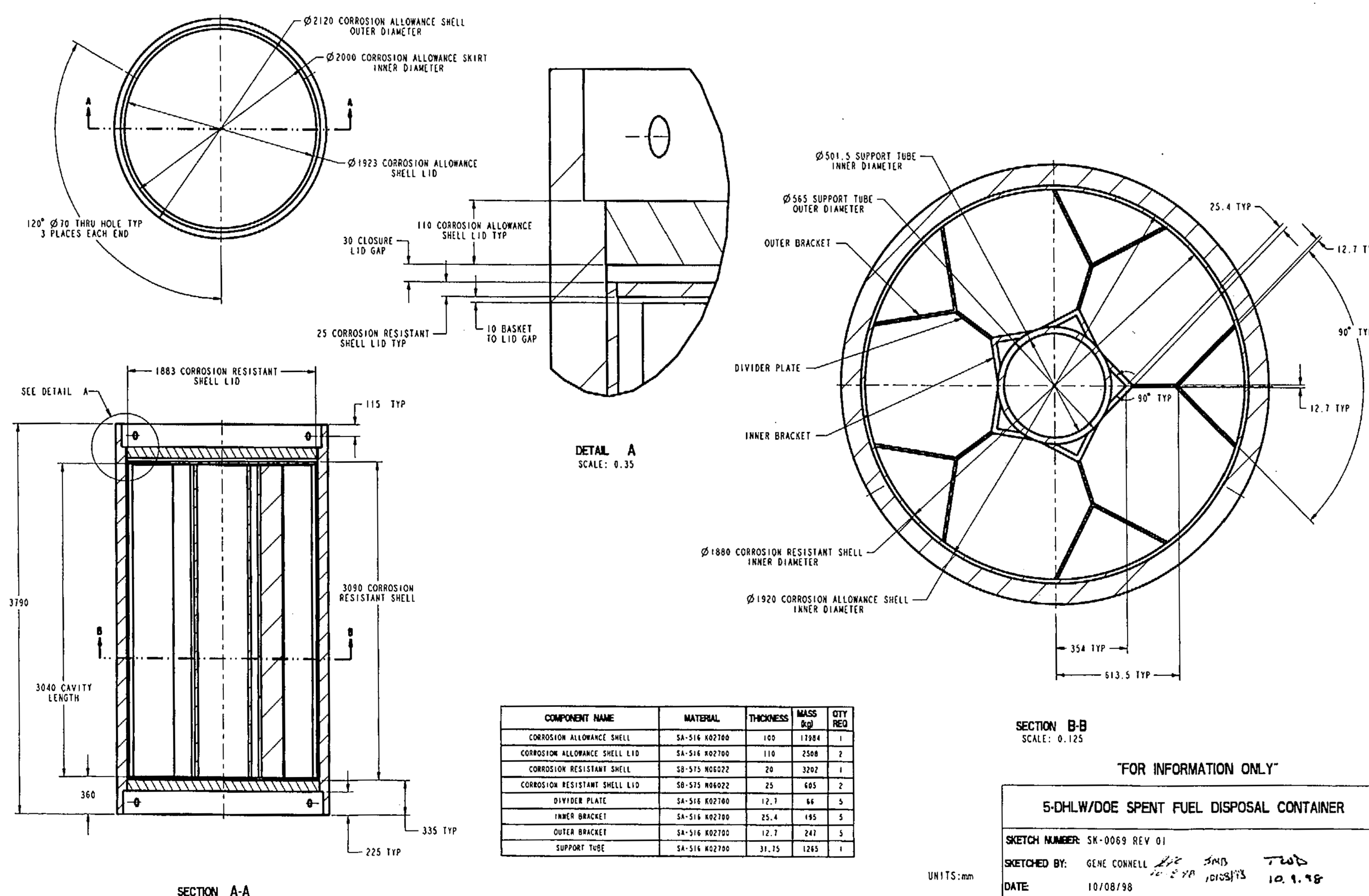
The hardcopy attachments are listed below in Table 8-1. Electronic output files are provided on Colorado Trakker® tapes (Ref. 7.15) and are listed in Table 8-2 below. Each output file is identified by its name, size (in bytes), and the date and time of last access. Note that for output files transferred from the HP workstation to the personal computer, the date and time will reflect the time of transfer. The actual date and time of run completion can be found in the output file.

Table 8-1. Attachments

| Description | Attachment Number | No. of Pages |
|---|---|---|
| 5-DHLW/DOE spent fuel disposal container SK-0069 REV 01 | I | 1 |
| TRIGA DOE SNF basket assembly SK-0124 REV 00 | II | 1 |
| Source intensity fractions of the TRIGA SNF and HLW glass | III | 1 |
| Atomic densities for the homogenized TRIGA DOE SNF canister contents | IV | 1 |

Table 8-2. MCNP Output Files

| Output File Name | Case | File Size (bytes) | File Date | File Time |
|---|---|---|---|---|
| TRIGAp.io | Gamma dose rate calculations | 181,982 | 07/19/99 | 1:40 p.m. |



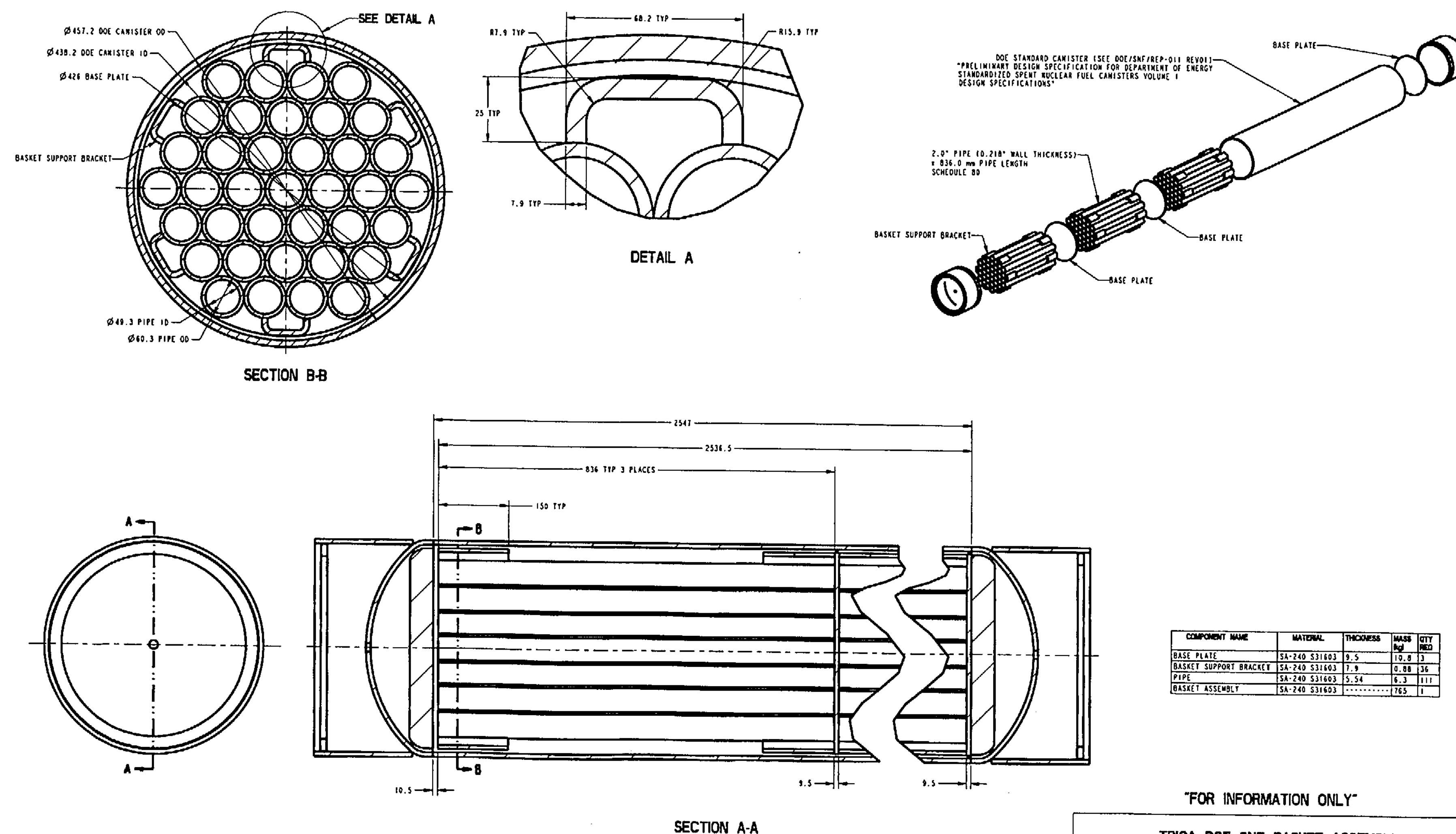

| COMPONENT NAME | MATERIAL | THICKNESS | MASS (kg) | QTY REQ |
|---|---|---|---|---|
| CORROSION ALLOWANCE SHELL | SA-516 K02700 | 100 | 17984 | 1 |
| CORROSION ALLOWANCE SHELL LID | SA-516 K02700 | 110 | 2508 | 2 |
| CORROSION RESISTANT SHELL | SB-575 N06022 | 20 | 3202 | 1 |
| CORROSION RESISTANT SHELL LID | SB-575 N06022 | 25 | 605 | 2 |
| DIVIDER PLATE | SA-516 K02700 | 12.7 | 66 | 5 |
| INNER BRACKET | SA-516 K02700 | 25.4 | 195 | 5 |
| OUTER BRACKET | SA-516 K02700 | 12.7 | 247 | 5 |
| SUPPORT TUBE | SA-516 K02700 | 31.75 | 1265 | 1 |

"FOR INFORMATION ONLY"

UNITS: mm

DO NOT SCALE

| 5-DHLW/DOE SPENT FUEL DISPOSAL CONTAINER | |
|---|---|
| SKETCH NUMBER: | SK-0069 REV 01 |
| SKETCHED BY: | GENE CONNELL |
| DATE: | 10/08/98 |
| FILE: | /home/pro_library/checkout/sketches/dhlw_5pk/SK-0069.dwg |




| COMPONENT NAME | MATERIAL | THICKNESS | MASS (kg) | QTY REQ |
|---|---|---|---|---|
| BASE PLATE | SA-240 S31603 | 9.5 | 10.8 | 3 |
| BASKET SUPPORT BRACKET | SA-240 S31603 | 7.9 | 0.88 | 36 |
| PIPE | SA-240 S31603 | 5.54 | 6.3 | 111 |
| BASKET ASSEMBLY | SA-240 S31603 | ---------- | 765 | 1 |

"FOR INFORMATION ONLY"

UNITS: mm
DO NOT SCALE FROM SKETCH

| TRIGA DOE SNF BASKET ASSEMBLY | |
|---|---|
| SKETCH NUMBER: | SK-0124 REV 00 |
| SKETCHED BY: | BRYAN HARKINS |
| DATE: | 03/19/99 |
| FILE: | /home/harkins/proe/fermi_37/SK-0124_triga.dwg |

BA 19Mar 99 SMB 03/19/99 TWD 3.19.99

Source Intensity Fractions of the TRIGA SNF and 3-m long SRS Canisters

Table III-1. TRIGA Neutron Source at 1-Year Decay Time

| Nuclide | 20-Year Decay Time[1] | | | 1-Year Decay Time | | |
|---|---|---|---|---|---|---|
| | Activity (Ci) | ($\alpha$, n) Production (neutrons/s) | SF Production (neutrons/s) | Activity[2] (Ci) | ($\alpha$, n) Production (neutrons/s) | SF Production (neutrons/s) |
| Bi-211 | 3.78E-08 | 5.51E-03 | | 3.94E-09 | 5.74E-04 | |
| Po-212 | 1.72E-05 | 1.84E+01 | | 5.90E-06 | 6.31E+00 | |
| Po-215 | 3.78E-08 | 2.57E-01 | | 3.94E-09 | 2.68E-02 | |
| Rn-219 | 3.78E-08 | 9.24E-03 | | 3.94E-09 | 9.63E-04 | |
| U-235 | 1.19E-04 | 9.71E-03 | 1.82E-02 | 1.19E-04 | 9.73E-03 | 1.82E-02 |
| U-238 | 1.83E-05 | 9.49E-04 | 6.43E-01 | 1.83E-05 | 9.50E-04 | 6.44E-01 |
| Pu-238 | 3.24E-00 | 1.31E+03 | | 3.75E+00 | 1.52E+03 | |
| Pu-239 | 9.35E-02 | 4.66E+01 | 4.70E-00 | 9.35E-02 | 4.66E+01 | 4.70E+00 |
| Pu-240 | 7.69E-02 | 2.70E+01 | | 7.66E-02 | 2.69E+01 | |
| Am-241 | 7.80E-01 | 4.57E+02 | | 1.03E-01 | 6.02E+01 | |
| Total | | 1.86E+03 | 5.36E-00 | | 1.66E+03 | 5.36E+00 |

[1] Ref. 7.11, p. B-8.
[2] Ref. 7.11, p. B-4.

1. Neutron source strength (peaking factor for TRIGA fuel: 1.25 [see Assumption 3.6])

Total neutron source strength (neutrons/s): TRIGA: 1.25*111*(1.66E+3 + 5.36E+0)=2.31E+5
SRS: 5*8.21E+7 =4.105E+8
Total: 4.105E+8 + 2.31E+5 = 4.107E+8

2. Gamma source strength fractions for source probability (SP) card: peaking factor for TRIGA fuel: 1.25 (see Assumption 3.6)

Total gamma source strength (photons/s): TRIGA: 1.25*111*1.251E+14=1.7358E+16
SRS: 5*3.68E+15 = 1.84E+16
Total: 1.84E+16 + 1.7358E+16 = 3.5758E+16

Fractions: SRS: 3.68E+15/3.5758E+16=0.1029
TRIGA: 1.7358+16/3.5758E+16=0.4855

Atomic Density Calculation

Total SS 316L mass (kg)=3*(235.6+10.8+10.6)=771
Total SS 304 mass (g) = 111*(270+530)=88800
Total Zr mass (g) = 111*(2060+63.7)=235730.7
Total graphite mass (g) =111*450=49950
Total U-235 mass (g) =111*137=15207
Total U-238 mass (g) =111*59=6549
DOE SNF canister cavity volume (cm$^3$)=$\pi$*21.9075^2*254.7=384029.3865

Table IV-1. Element or Isotope Mass Calculation for the Contents of DOE SNF Canister

| Element/ Isotope | SS 316L | | SS 304 | | Zr | Graphite | U-238 | U-235 | Total |
|---|---|---|---|---|---|---|---|---|---|
| | wt% | Mass (g) | wt% | Mass (g) | Mass (g) | Mass (g) | Mass (g) | Mass (g) | Mass (g) |
| C | 0.03 | 231.3 | 0.08 | 71.04 | | 49950 | | | 50,252.3 |
| Mn | 2 | 15,420 | 2 | 1,776 | | | | | 17,196 |
| P | 0.045 | 346.95 | 0.045 | 39.96 | | | | | 386.91 |
| S | 0.03 | 231.3 | 0.03 | 26.64 | | | | | 257.94 |
| Si | 1 | 7,710 | 1 | 888 | | | | | 8,598 |
| Cr | 17 | 131,070 | 19 | 16,872 | | | | | 147,942 |
| Ni | 12 | 92,520 | 9.25 | 8,214 | | | | | 100,734 |
| Mo | 2.5 | 19,275 | | | | | | | 19,275 |
| N | 0.1 | 771 | 0.1 | 88.8 | | | | | 859.8 |
| Fe | 65.295 | 503,424.5 | 68.495 | 60,823.56 | | | | | 564,248 |
| Zr | | | | | 235,730.7 | | | | 235,731 |
| U-235 | | | | | | | | 15,207 | 15,207 |
| U-238 | | | | | | | 6,549 | | 6,549 |

Table IV-2. Element or Isotope Atomic Densities for the Contents of the DOE SNF Canister

| Element/Isotope | Atomic Mass[1] (g) | Mass (g) | Atomic Density (atoms/b.cm) |
|---|---|---|---|
| C | 12.0107 | 50,252.34 | 6.5611E-03 |
| Mn | 54.938049 | 17,196 | 4.9084E-04 |
| P | 30.973761 | 386.91 | 1.9589E-05 |
| S | 32.066 | 257.94 | 1.2614E-05 |
| Si | 28.0855 | 8,598 | 4.8007E-04 |
| Cr | 51.9961 | 147,942 | 4.4618E-03 |
| Ni | 58.6934 | 100,734 | 2.6914E-03 |
| Mo | 95.94 | 19,275 | 3.1505E-04 |
| N | 14.00674 | 859.8 | 9.6260E-05 |
| Fe | 55.845 | 564,248 | 1.5844E-02 |
| Zr | 91.224 | 235,730.7 | 4.0522E-03 |
| U-235 | 235.043922 | 15,207 | 1.0146E-04 |
| U-238 | 238.050785 | 6,549 | 4.3141E-05 |
| Total | | | 3.5170E-02 |

[1] Ref. 7.13.