TRANSNUCLEAR

AMENDMENT NO. 8

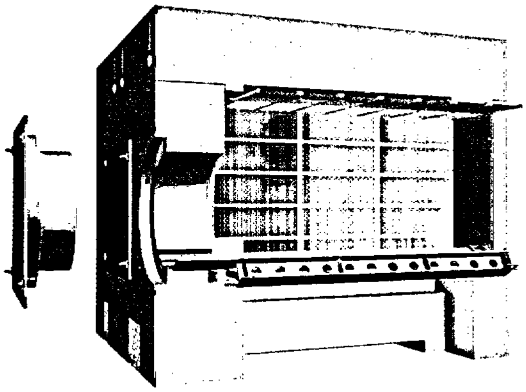
ADDITION OF 24PTH DSC

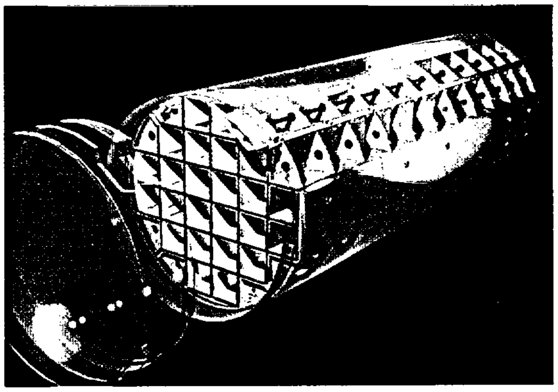
TO

NUHOMS® COC 1004

DSC TO STANDARDIZED NUHOMS® SYSTEM

SYSTEM





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NON-PROPRIETARY

P.5 Shielding Evaluation

The radiation shielding evaluation for the Standardized NUHOMS® System (during loading, transfer and storage) for the other NUHOMS® canisters is discussed in other sections and appendices of the FSAR. The following radiation shielding evaluation specifically addresses the shielding evaluation of the NUHOMS® 24PTH system with design-basis PWR fuel and control components (CCs) loaded in a NUHOMS®-24PTH DSC.

The shielding analysis is carried out for the three DSC configurations (24PTH-L, 24PTH-S, and 24PTH-S-LC) of the NUHOMS®-24PTH system described in Section P.1. The 24PTH-L and 24PTH-S DSCs are transferred either in the OS197/OS197H Transfer Cask (TC) or the OS197FC TC depending upon the heat load and stored in the HSM-H. The 24PTH-S-LC DSC is transferred in the Standardized TC and stored in either the HSM-H or HSM-Model 102. The seven possible loading combinations are listed below:

(1) 24PTH-L DSC → OS197FC TC (bounds OS197/OS197H TCs)
(2) 24PTH-L DSC → HSM-H
(3) 24PTH-S DSC → OS197FC TC (bounded by #1)
(4) 24PTH-S DSC → HSM-H (bounded by #2)
(5) 24PTH-S-LC DSC → Standardized TC
(6) 24PTH-S-LC DSC → HSM-H (bounded by #7)
(7) 24PTH-S-LC DSC → HSM-Model 102

The design of HSM-H is similar to HSM Model 102 except the HSM-H has improved shielding performance due to the following design features:

- Elimination of 6" uniform gap between adjacent modules,
- Innovative shielded inlet and outlet ventilation openings,
- Increased concrete thickness in roof, front and backwalls and shield walls, and
- Increased shielding in the HSM door.

These design features results in the occupational and site dose rates ALARA.

The basket layout for the three DSC configurations is identical except for the length of the DSC components and the shield plug design. The 24PTH-S DSC and 24PTH-L DSC differ in DSC and cavity length, while the 24PTH-S-LC DSC and 24PTH-S DSC differ in cavity length due to a different shield plug design. The 24PTH-L/S has carbon steel shield plugs, while the 24PTH-S-LC has thinner lead shield plugs to increase cavity length to allow for greater fuel lengths in a shorter canister.

Each DSC configuration is designed to store up to 24 intact (and up to 12 damaged, with remaining intact) PWR fuel assemblies. The 24PTH-L and 24PTH-S-LC DSCs are also designed to store up to 24 intact standard PWR fuel assemblies with or without CC; such as burnable poison rod assemblies (BPRAs), Control Rod Assemblies (CRAs), Thimble Plug

Assemblies (TPAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), and Neutron Source Assemblies (NSAs); the 24PTH-S DSC will not store CC. For shielding purposes, the 24PTH-L bounds the 24PTH-S DSC because of the additional gamma source due to the CC. Therefore, the shielding evaluation presented herein is not performed for the 24PTH-S DSC. Based on the results of Fuel Qualification Tables described in Section P.5.2, fuel with CC requires one more year of cooling time. To assure that this evaluation is conservative, the fuel source terms are not adjusted to account for the additional decay required to accommodate the CC.

Dose rates are calculated for the 24PTH-L DSC within HSM-H. Dose rates are also estimated for the 24PTH-S-LC within a HSM-Model 102. As the HSM-Model 102 provides less shielding than the HSM-H, shielding estimates are not made for the 24PTH-S-LC within HSM-H as the dose rates provided bound this scenario.

The design of the OS197FC TC is identical to the design of OS197/OS197H TC except that the OS197FC TC has a modified top lid. For shielding analysis of 24PTH-S and –L DSCs, OS197FC TC is used to bound the OS197/OS197FC TC also because the design features in the TC radial direction are identical for all three TCs; and OS197FC top axial geometry bounds other TCs.

The design-basis PWR fuel source terms are derived from the bounding fuel, B&W 15x15 Mark B assembly design as described in Section P.5.2.

The NUHOMS®-24PTH DSCs is designed to store PWR fuel assemblies and CC with the characteristics described in Table P.2-1. The 24PTH-S/L DSCs have a maximum decay heat of 2.0 kW per assembly and a maximum heat load of 40.8 kW per canister. Fuel in the 24PTH-S/L DSCs may be stored in four alternate heat zoning configurations as shown in Figure P.2-1 through Figure P.2-4. The 24PTH-S-LC DSC has a maximum decay heat of 1.5 kW per assembly and a maximum heat load of 24 kW per canister. The heat zoning configuration to be used for the 24PTH-S-LC DSC is shown in Figure P.2-5. Note that while the B&W, CE, and Westinghouse fuel designs are specifically listed, storing reload fuel designed by other manufacturers is also allowed provided an analysis is performed to demonstrate that the limiting features listed in Table P.2-1 and Table P.2-3 bound the specific manufacturer's replacement fuel. The limiting features are burnup, initial enrichment, cooling time, number of fuel rods, cobalt impurities in the hardware and initial heavy metal weight.

The design-basis fuel source terms for this evaluation are defined as the source terms from fuel with the burnup/initial enrichment/cooling time combination given in Table P.2-6 through Table P.2-9 (without CC) and located in the basket as shown in Figure P.2-1 through Figure P.2-5 that give the maximum dose rate on the surface of the HSM and/or TC. This approach is consistent with the method used to generate the fuel qualification tables for the Standardized NUHOMS®-24P and -52B DSC designs as described in Section 7.2.3, or 32PT DSC design as described in Appendix M. The design basis fuel source term is then added to the design basis CC source term (Table P.5-12) to create the total fuel assemblies plus CC source term used in the calculations.

For the 24PTH-L DSC, Heat Load Zoning Configuration 2 (Figure P.2-2) is the configuration that produces the highest dose rates on the surfaces of the HSM-H and OS197FC TC as compared to configurations 1, 2 and 3 because the highest source fuel assemblies are on the outer periphery of the basket region where self-shielding due to adjacent assemblies is limited. This configuration 2 consists of 20 2.0 kW fuel assemblies located in the outer regions of the DSC. For the 24PTH-S-LC, which has only one heat load zoning configuration (Configuration 5, Figure P.2-5). To bound the shielding analysis for heat load zoning configuration 5, fuel assemblies with a decay heat of 1.5 kW at all 24 location is used. This results in a shielding analysis corresponding to a total of 36 kW decay heat per DSC which is very conservative because the total decay heat in 24PTH-S-LC DSC is limited to 24kW. These bounding gamma and neutron source terms are then used in the radiation shielding models to conservatively calculate dose rates on and around the NUHOMS®-24PTH system.

The bounding burnup, minimum initial enrichment and cooling time combinations for the fuel assemblies used in the shielding analyses of the 24PTH-L DSC in the HSM-H and the OS197FC TC are as follows:

- Dose rates with 24PTH-L DSC in HSM-H: 41 GWd/MTU, 3.3 wt. % U-235, 3.0-year cooled fuel
- Dose rates with 24PTH-L DSC in OS197FC TC: 62 GWd/MTU, 3.4 wt. % U-235, 5.6-year cooled fuel

The bounding burnup, minimum initial enrichment and cooling time combinations for the fuel assemblies used in the shielding analysis of the 24PTH-S-LC DSC are as follows:

- Dose rates with 24PTH-S-LC DSC in Standardized TC: 32 GWd/MTU, 2.6 wt. % U-235, 3.0-year cooled fuel
- Dose rates with 24PTH-S-LC DSC in HSM-Model 102: 32 GWd/MTU, 2.6 wt. % U-235, 3.0-year cooled fuel (same as for Standardized TC)

Note that for the 24PTH-L DSC, the source terms are different for calculating dose rate when in HSM-H and OS197FC TC. However, for the 24PTH-S-LC DSC, the source terms are the same for calculating the dose rates when in HSM-Model 102 and Standardized TC. The method of selecting the bounding source terms is explained in detail in Section P.5.2.

The design basis CC source term that envelopes all CCs allowed in the 24PTH DSCs is taken from Appendix J for BPRAs with burnups up to 36 GWd/MTU. While Appendix J was developed to specifically address the additional source from a BPRA, this source term is selected as the bounding source term for all CCs. The TPAs and ORAs do not extend into the active fuel region of a fuel assembly. Therefore, they are limited to the source term equivalent to the top plus plenum region source term of a BPRA. However, tube conservative, the full total source term of BPRA is used in the shielding analysis to bound all CCs. The total per canister source term allowed for these CCs is shown in Table P.2-2. The source term energy distribution is shown in Table P.5-12. Any CC to be stored in a 24PTH DSC must be bounded by this source term.

Reconstituted and/or damaged fuel is also acceptable for the DSC payload. Reconstituted fuel may contain up to 10 solid stainless steel rods or unlimited number of lower enriched UO2 rods that replace damaged fuel rods. Note that lower enriched UO2 rods are of similar design and behavior as the standard fuel rods aside from the uranium enrichment. The reconstituted rods can be at any location in the fuel assemblies and the reconstituted assemblies can be placed anywhere in the basket. Reconstituted fuel has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added if reconstituted rods are present. Damaged fuel has essentially no impact on the dose rate as the source term would not be impacted and gross axial source redistribution is not likely. Therefore, shielding analysis results with intact fuel are also applicable to the damaged fuel.

The methodology, assumptions, and criteria used in this evaluation are summarized in the following subsections.

P.5.1 Discussion and Results

All 24PTH-L DSC MCNP calculations are performed for heat-load zoning configuration 2 which includes 20 design-basis PWR fuel assemblies (with CC) using 2.0 kW fuel. All 24PTH-S-LC DSC MCNP calculations are performed for 24 design-basis PWR fuel assemblies (with CC) using 1.5 kW fuel.

Table P.5-1 summarizes the maximum and average dose rates for the NUHOMS®-24PTH-L DSC loaded into the NUHOMS® HSM-H.

Table P.5-2 summarizes the maximum and average dose rates for the NUHOMS®-24PTH-S-LC DSC loaded into the NUHOMS® HSM-Model 102. Note that the HSM-H is more heavily shielded than the HSM-Model 102(thicker roof, shield walls, front and back wall including HSM door); therefore, HSM-Model 102 is conservatively modeled to bound HSM-H.

Table P.5-3 provides a summary of the dose rates on and around the OS197FC TC for transfer of the 24PTH-L DSC under normal, off-normal and accident conditions.

Table P.5-4 provides a summary of the dose rates on and around the OS197FC TC for decontamination and welding operations for the 24PTH-L DSC.

Table P.5-5 provides a summary of the dose rates on and around the Standardized TC for transfer of the 24PTH-S-LC DSC under normal, off-normal and accident conditions.

A discussion of the method used to determine the design-basis fuel source terms is included in Section P.5.2. The design basis CC source term which is from Appendix J is shown in Table P.5-12. The shielding material densities are given in Section P.5.3. The method used to determine the dose rates due to design-basis fuel assemblies with CC in the various NUHOMS®-24PT DSC design configurations is provided in Section P.5.4. Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The shielding evaluation is performed with the MCNP4C2 [5.2] code with the ENDF/B-VI cross section library. Sample input files used for calculating neutron and gamma source terms and dose rates are included in Section P.5.5.

P.5.2 Source Specification

Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The SAS2H/ORIGEN-S results are used to develop the fuel qualification tables listed in Table P.2-6 through Table P.2-13 and the design-basis fuel source terms suitable for use in the shielding calculations. The thermal and radiological source terms for the CCs which are taken from Appendix J, are shown in Table P.5-12.

The B&W 15x15 assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading and CO-59 content of the hardware regions as compared to the 14x14, other 15x15, and 17x17 fuel assemblies which are also authorized contents of the NUHOMS®-24PTH DSC. The neutron flux during reactor operation is peaked in the in-core region of the fuel assembly and drops off rapidly outside the in-core region. Much of the fuel assembly hardware is outside of the in-core region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the in-core region, the (gas) plenum region, and the top (nozzle) region. The B&W 15x15 fuel assembly masses for each irradiation region are listed in Table P.5-6. The light elements that make up the various materials for the various fuel assembly materials are taken from reference [5.4] and are listed in Table P.5-7. The design-basis heavy metal weight is 0.490 MTU. These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the In-Core regions neutron flux (fluence) correction factors are applied to light element composition for each region. The neutron flux correction factors which are from Reference [5.15] are given in Table P.5-8.

Evaluations of the existing data with SAS2H and the 44-group ENDF/B-V library used in the analysis are documented in References [5.11] and [5.12]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [5.13]. In fact, for the case with 46,460 MWd/MTU burnup, the isotopic predictions are all within 2% of those measured. There are ongoing efforts, some of which are documented in Reference [5.10], to obtain more data for burnups above 45 GWd/MTU. There is no reason to expect that the ongoing evaluations of the higher burnup fuel will result in less favorable comparisons.

As noted in References [5.14] and [5.10], there is no public data for the neutron component currently available that bounds a fuel burnup of up to 62 GWd/MTU. However, as documented in Reference [5.14] and confirmed in the SAS2H analysis, the total neutron source with increasing burnup is more and more dominated by spontaneous fission neutrons. Reviewing the output from the SAS2H runs, the neutron source term is due almost entirely to the spontaneous fission of Cm-244 (~98% of all neutrons both spontaneous fission and (α,n)). After reviewing the measured Cm-244 content compared to the Cm-244 content predicted by SAS2H and the 44-group ENDF/B-V library documented in References [5.11] and [5.12] for burnups up to 46,460 MWd/MTU, it is readily apparent that the calculated values are within ±11 % of the measured values, with most of the predicted values within ±5% of the measured. Finally, there is no observed trend as a function of burnup in the data that would indicate that the predicted Cm-244 content is significantly different at higher burnups.

As documented in Reference [5.14] and as observed in preparing the fuel qualification tables, the gamma dose rate increases nearly linearly with burnup relative to the direct gamma component and the neutron dose rate increases with burnup to the fourth power. Therefore, as burnups go beyond 45 GWd/MTU, the contribution from neutron (and associated n,γ) components to the total dose rates measured on the surfaces of the DSC, TC and HSM (HSM-H and HSM Model 102) increase in relative importance to that of the gamma component. However, this increase in the importance of the neutron source term has a relatively minor effect on the area dose rates on and around the HSM as these are dominated by the gamma component as shown in Table P.5-1 and Table P.5-2. The surface dose rates on the HSM are dominated by the gamma component because the HSM is constructed of thick reinforced concrete, which is an excellent neutron shield. Therefore, even a postulated substantial increase in the neutron source term would have a relatively minor effect on the site dose rate evaluation presented in Section P.10 of the amendment application.

For the TC, the neutron source term has a relatively minor effect on the area dose rates during most of the cask handling operations, since the DSC cavity and the annulus between the TC and DSC is filled with water and most of the work is done around the top of the cask. The neutron component is of more importance on and around the TC during transfer operations but, in general, only represents a small portion of the total dose rate on the sides and top of the TC. While the neutron dose rate on the bottom of the TC is slightly higher than gammas, relatively little occupational dose is received from this area. The dose rates for the design basis fuel on the surfaces of HSM and TC are shown in Table P.5-1 through Table P.5-5. These tables show that gamma dose rates are substantially higher than neutron dose rates. Therefore, the neutron component of the dose is a relatively minor fraction of the total occupational and site boundary dose.

The occupational exposure calculations demonstrate that most of the dose received by workers during cask loading and transfer operations is due to the gammas on and around the cask. The only surface of the TC that is dominated by neutrons is at the bottom of the cask. A small fraction of the total occupational exposure is due to the doses around the bottom of the cask because very little work is performed on or around the bottom of the cask with fuel in the TC.

As discussed above, any impact of uncertainties in source terms is expected to be negligible for the 24PTH system. Therefore, isotopic depletion calculations with SAS2H for fuel burned above 45 GWd/MTU are appropriate.

The fuel qualification tables are generated based on the decay heat limits for the various heat load zoning configurations shown in Figure P.2-1 through Figure P.2-5. SAS2H is used to calculate the minimum required cooling time to the nearest 0.1 year as a function of assembly initial enrichment and burnup for each decay heat limit. These cooling times are rounded up to the nearest 0.5 year increment in the final fuel qualification tables. Because the decay heat generally increases slightly with decreasing enrichment for a given burnup, it is conservative to assume that the required cooling time for a higher enrichment assembly is the same as that for a lower enrichment assembly with the same burnup. The required cooling time for initial enrichments that fall between any two SAS2H runs are assumed to be that of the lower enrichment case results.

Fuel qualification tables for fuel without CC are listed in Table P.2-6 through Table P.2-9. However, some assemblies will contain a CC, which adds up to 8 watts of decay heat per assembly. Therefore, an additional set of fuel qualification tables are developed, as shown in Table P.2-10 through Table P.2-13, for fuel that contains CC. The fuel qualification tables for fuel with CC have slightly longer cooling times when compared to fuel without CC.

Reconstituted and/or damaged fuel is also acceptable for the DSC payload. Reconstituted fuel may contain up to 10 solid stainless steel rods that replace fuel rods. Reconstituted fuel has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added if reconstituted stainless steel rods are present. If the cooling time is greater than 10 years, no additional cooling time is needed. Additional discussion on the method used to analyze reconstituted fuel is provided in Section P.5.2.5. Damaged fuel has essentially no impact on the dose rate as the source term would not be impacted and gross axial source redistribution is not likely.

The design-basis source terms are defined as the burnup/initial enrichment/cooling time combination given in the fuel qualification tables that result in the maximum dose rate on the surface of the HSM (either type) or TC (all types). Note that for a given DSC design, the design basis HSM source will not necessarily be the same as the corresponding design basis TC source. The 1-D discrete ordinates code ANISN [5.5] and the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3] is used to determine the HSM and TC dose rate for each entry in the fuel qualification tables and thereby determine the design basis source. As ANISN is a 1-D code, a single dose location must be selected for both the HSM and TC for analysis purposes. For the HSM, the roof is selected as the dose location, and for the TC the cask side is selected as the dose location. This approach, described in detail in Section P.5.2.4, is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® canister designs described in Section 7.2.3 and Appendix M.5. The radiological source terms generated in the SAS2H/ORIGEN-S runs are used in the ANISN evaluations to calculate the surface dose rates. The ANISN models are similar to the appropriate MCNP4C2 models for the locations of interest.

Heat load zoning configuration 2 (Figure P.2-2) produced the bounding total surface dose rate for both the HSM-H and OS197FC TC using the 24PTH-L DSC. The 24PTH-L DSC, HSM-H design-basis source terms are from fuel with 41 GWd/MTU burnup, an initial enrichment of 3.3 wt. % U-235 and 3-years cooling. The 24PTH-L DSC, OS197FC design-basis source terms are from fuel with 62 GWd/MTU burnup, an initial enrichment of 3.4 wt. % U-235 and 5.6-years cooling.

The heat load zoning configuration selected for the shielding analysis of the 24PTH-S-LC DSC bounds the actual heat load configuration shown in Figure P.2-5 because 1.5 kW fuel is placed in all 24 locations. The 24PTH-S-LC DSC, Standardized TC design-basis source terms are from fuel with 32 GWd/MTU burnup, an initial enrichment of 2.6 wt. % U-235 and 3.0-years cooling.

A sample SAS2H/ORIGEN-S input file for the In-Core Region for the 41 GWd/MTU, 3.3 wt. % U-235 and 3-years cooling case is listed in Section P.5.5.1. Input for reconstituted fuel would be similar, except for a reduced number of fuel pins from 208 to 198, light element masses that reflect reconstituted rods, and slightly different power input to maintain the same burnup for a reduced fuel mass.

P.5.2.1 Gamma Source Term for MCNP

P.5.2.1.1 Design Basis Gamma Fuel Assembly Source Terms

Once the design basis burnup/enrichment/cooling time combinations have been determined for each shielding configuration of interest, four SAS2H/ORIGEN-S runs are required for each combination to determine gamma source terms for the four fuel assembly regions (i.e., bottom, in-core, plenum and top). The only difference between the runs is in Block #10 "Light Elements" of the SAS2H input and the 82$$ card in the ORIGEN-S input. Each run includes the appropriate Light Elements for the region being evaluated and the 82$$ card is adjusted to have ORIGEN-S output the total gamma source for the in-core region and only the light element source for the plenum, bottom, and top regions. Gamma source terms for the in-core region include contributions from actinides, fission products, and activation products. The bottom, plenum and top nozzle regions include the contribution from the activation products in the specified region only. The SAS2H/ORIGEN-S gamma ray source is output in the CASK-81 energy group structure.

A design basis source is developed for each decay heat (1.5 and 2.0 kW) and shielding structure combination used in the shielding analysis. Note that for a given decay heat, the design basis TC and HSM source may or may not be the same. For the heat load configurations analyzed, four design basis sources are required:

(1) 2.0 kW fuel in 24PTH-L DSC/OS197FC TC,
(2) 2.0 kW fuel in 24PTH-L DSC/HSM-H,
(3) 1.5 kW fuel in 24PTH-S-LC DSC/Standardized TC, and
(4) 1.5 kW fuel in 24PTH-S-LC DSC/HSM-Model 102.

The results for 2.0 kW fuel in a 24PTH-L DSC/OS197FC TC (62 GWd/MTU, 3.4 wt. % U-235 and 5.6-years cooling) are shown in Table P.5-9. The results for 2.0 kW fuel in a 24PTH-L DSC/HSM-H (41 GWd/MTU, 3.3 wt. % U-235 and 3-years cooling) are shown in Table P.5-10. Finally, the results for 1.5 kW fuel are the same for either a 24PTH-S-LC DSC/Standardized TC or HSM-Model 102 (32 GWd/MTU, 2.6 wt. % U-235 and 3.0-years cooling) and are shown in Table P.5-11.

P.5.2.1.2 Design Basis CC Source Terms

The design basis CC source terms are taken from Appendix J of the FSAR and are listed in Table P.5-12. All CC to be stored in the DSC must be bounded by this source. Gamma source terms for use in the MCNP shielding models are calculated by adding the fuel assembly sources to the CC source and multiplying the sum by the number of assemblies in the model.

The design basis CC source terms are developed based upon an examination of the following three BPRA types (1) B&W 15 X 15 (2 cycles, 5 year cooled), (2) WE 17 X 17 Pyrex Burnable Absorber (2 cycles, 10 year cooled), and (3) WE 17 X 17 WABA Burnable Absorber (2 cycles, 10 year cooled). All BPRA types are irradiated to a burnup of 36 GWd/MTU using ORIGEN2. The final design basis CC source term is a hybrid of the worst-case results for the top, plenum, and core regions for these three BPRA types. The core region is taken from the WE 17 X 17 Pyrex BPRA, while the top and plenum regions are taken from the B&W 15 X 15 BPRA.

High burnup BPRAs may have a burnup up to 45 GWd/MTU. As the design basis CC source term assumes only 36 GWd/MTU, additional cooling time is required for high-burnup CCs so that the CC source remains bounded by the design basis. Calculations show that high-burnup B&W 15 X 15 BPRAs are acceptable for storage after 8 years of decay time. Both WE 17 X 17 Pyrex Burnable Absorber and WE 17 X 17 WABA Burnable Absorber high-burnup BPRAs are acceptable for storage after 13 years of decay time.

All other CCs, including BPRA types other than the three analyzed above, must be examined on a case by case basis to demonstrate that they are bounded by the design basis CC source. Specifically, the maximum allowed CC gamma source is 9.3E+14 γ/s/canister for BPRAs, NSAs, CRAs, and IFBAs, and 9.8E+13 γ/s/canister for TPAs.

P.5.2.1.3 Uncertainty in Gamma Source Terms

Almost 100% of the gamma spectrum from light elements is in the range of 0.70 to 1.33 MeV which corresponds exactly to two the most prominent lines of ^{60}Co. As for fission products, the main contributors after six years with a fraction greater then 5% in the range of 0.01 to 0.90 MeV are: ^{90}Sr, ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, ^{154}Eu, and ^{155}Eu. Contributions from ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, and ^{154}Eu are dominant in the range of 0.90 to 1.50 MeV. ^{106}Rh, ^{147}Sm, and ^{142}Ce are the strongest emitters at energies greater then 2.0 MeV. The accuracy of gamma spectrum is dependent upon the energy. Photon rates computed for fission products tend to be more accurate then those for actinides because the calculation of their inventory has less uncertainty [5.1].

Shortly after discharge the emission at higher energies is dominated by actinides. This is true for energies >4 MeV at all cooling times and energy above 3.5 MeV for cooling times after 10 years [5.1]. The major part of this emission comes from ^{244}Cm. Thus the uncertainty for energy groups of order 3.0 MeV and greater is bounded with the precision with which the inventory of ^{244}Cm is calculated. Per SCALE 4.4 [5.1], reported experimental ^{244}Cm densities are accurate within ± 20%. The gamma emission intensity from Cm, which is proportional to the quantity of Cm in the actinide inventory, is bounded by this value. Uncertainty in the source strength in the gamma energy range 0.5 to 2.5 MeV is in the vicinity of 10 to 15 % [5.1].

P.5.2.2 Neutron Source Term for MCNP

One SAS2H/ORIGEN-S run is required for each burnup/initial enrichment/cooling time combination to determine the total neutron source term for the in-core regions. At discharge the neutron source is almost equally produced from ^{242}Cm and ^{244}Cm. The other strong contributor is ^{252}Cf, which is approximately 1/10 of the Cm intensity, but its share vanishes after 6 years of cooling time because the half-life of ^{252}Cf is 2.65 years. The half-lives of ^{242}Cm and ^{244}Cm are

163 days and 18 years, respectively. Contributions from the next strongest emitters, ^{238}Pu and ^{240}Pu, are lower by a factor of 1000 and 100, respectively, relative to ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein.

The magnitude of the neutron source is provided as the final row in the gamma source term tables, see Table P.5-9, Table P.5-10, and Table P.5-11. Neutron source terms for use in the MCNP shielding models are calculated by multiplying the assembly source by the number of assemblies in the in-core region of interest (20 or 24). The magnitude of the neutron source is also increased to account for the axial distribution in the fuel, as explained in Section P.5.2.3.

The fixed source spectrum in MCNP is assumed to follow a ^{244}Cm spontaneous fission spectrum for all of the calculations in Appendix P.5. It is based on the following relationship:

$$p(E) = C \exp(-E/a)\sinh(bE)^{1/2}$$

with input parameters a=0.906 MeV and b=3.848 MeV^{-1}, as given in the MCNP manual [5.2].

P.5.2.3 Axial Peaking

Axial burnup peaking factors for PWR fuel are taken from Reference [5.6]. These peaking factors are assumed to match the gamma axial source distribution because the gamma source is proportional to burnup. The neutron source is approximately proportional to the fourth power of the burnup. Therefore, the axial neutron source distribution may be determined as the fourth power of the axial burnup profile.

Axial peaking changes with increasing burnup. As the design basis source occurs at different burnups for the various decay heat and shielding configurations, different axial peaking factors are selected for the various TC and HSM calculations. The axial peaking factors used are provided in Table P.5-13. The OS197FC TC calculations use peaking factors for a burnup >46 GWd/MTU because the design basis source for 2.0 kW fuel in a TC occurs at a burnup of 62 GWd/MTU. To simplify input preparation, the Standardized TC calculations, which use the 1.5 kW source term at a burnup of 32 GWd/MTU, use the same axial peaking factors as the OS197FC TC calculations. The uncertainty of this simplification is within the uncertainty of the calculational methods employed.

The HSM-H calculation uses peaking factors for a burnup in the range 38<BU<42 GWd/MTU because the design basis source for 2.0 kW fuel in an HSM-H occurs at a burnup of 41 GWd/MTU. The HSM-Model 102 calculation does not utilize MCNP, rather, result are obtained by scaling the HSM-Model 102 results from Appendix N. As such, the peaking factors listed in Table N.5-14 of Appendix N are used by default in the HSM-Model 102 analysis.

The neutron and gamma peaking factors are shown as a function of the core height in Table P.5-13. These factors are directly applied to each MCNP interval in the fuel region.

The average values of the axial peaking distributions are also provided in Table P.5-13. For the gamma distribution, the average value is 1.0. However, for the neutron distribution, the average value of the distribution is greater than 1.0. The average value of the axial neutron distribution may be interpreted as the ratio of the true total neutron source in an assembly to the neutron source calculated by SAS2H/ORIGEN-S for an average assembly burnup. Therefore, to properly correct the magnitude of the neutron source, the neutron source per assembly as reported in Table P.5-9, Table P.5-10, and Table P.5-11 is multiplied by the average value of the neutron source distribution as reported in Table P.5-13.

P.5.2.4 ANISN Evaluation for Bounding Source Terms

As discussed above, the fuel qualification tables are generated based on the decay heat limits for the various heat load zoning configurations shown in Figure P.2-1 through Figure P.2-5. SAS2H is used to calculate the minimum required cooling time as a function of assembly initial enrichment and burnup for each decay heat limit. To determine which combination of burnup, wt. % initial enrichment and cooling time result in the bounding dose rates on the surface of the HSM-H and OS197FC TC, the total source term, which includes the contribution from the fuel as well as the hardware in the entire assembly (including end fittings) is used to calculate its total ANISN dose rate on the HSM-H roof and OS197FC TC radial model using the ANISN code.

An ANISN TC model is developed only for the OS197FC TC. The side shielding through the OS197FC TC is identical to the side shielding through the Standardized TC except that the Standardized TC has NS-3 rather than water as the neutron shield. The thickness of NS-3 and water are identical between the two casks, and the shielding properties of NS-3 and water are also similar. Therefore, it is assumed that the OS197FC model is sufficient to determine the design basis source term for the Standardized TC.

Also, only one ANISN HSM model is developed, for the HSM-H. It is assumed that this model is sufficient to determine the source term for the HSM-Model 102, which has a roof thickness of 3' compared to 3'8" for the HSM-H. This difference of concrete thickness will have minimal impact on the results of the ANISN evaluation because ANISN is used only to compare the relative strength of the source terms for each entry in the fuel qualification table.

The CC contribution is fixed and is included in the design basis shielding evaluation as such and therefore is not included in this ANISN evaluation.

ANISN [5.5] determines the fluence of particles throughout one-dimensional geometric systems by solving the Boltzmann transport equation using the method of discrete ordinates. Particles can be generated by either particle interaction with the transport medium or extraneous sources incident upon the system. Anisotropic cross-sections can be expressed in a Legendre expansion of arbitrary order.

The ANISN code implements the discrete ordinates method as its primary mode of operation. Balance equations are solved for the flow of particles moving in a set of discrete directions in each cell of a space mesh and in each group of a multigroup energy structure. Iterations are performed until all implicitness in the coupling of cells, directions, groups, and source regeneration is resolved.

ANISN coupled with the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.3] and the ANSI/ANS-6.1.1-1977 flux-to-dose conversion factors [5.8] is chosen to generate the ANISN dose rates used to determine the relative strength of the various source terms from fuel assemblies to determine the design basis source terms for the HSM (both types) and TC (all types). These design basis source terms are used with MCNP4C2 models of the 24PTH system to calculate the bounding system dose rates. ANISN provides an efficient method to select the design basis source terms.

The surface dose rates are calculated using ANISN models to perform the evaluation for the fuel assembly parameters in the fuel qualification table. The ANISN model used to calculate the relative dose rates on the HSM-H surface is similar to a cut through the center of the MCNP4C2 HSM-H roof model used for the shielding evaluation. The ANISN model used to generate the relative dose rates on the TC is similar to a cut through the center of the MCNP4C2 OS197FC TC side model used for the shielding evaluation. Figure P.5-1 and Figure P.5-2 provide sketches for the ANISN models of the HSM-H roof and OS197FC TC centerline, respectively. An example ANISN input file is included in Section P.5.5.4.

With the exception of the fuel region, the material densities used in the ANISN models are the same as those used in the MCNP4C2 models as provided in Table P.5-14. The ANISN and MCNP4C2 number densities in the fuel region differ because in the MCNP4C2 models, the basket is modeled explicitly, while in the ANISN models the basket is homogenized with the fuel. The ANISN number densities for the fuel/basket region are provided in Table P.5-15.

To simplify the number of ANISN calculations required, a "response function" is developed using ANISN. A separate response function is developed for both the OS197FC TC and HSM-H. To generate a gamma response function, a separate ANISN model is executed with a single gamma per assembly in each of the 18 CASK-81 gamma energy groups. As ANISN requires the source in particles per second per unit volume, the volume of the homogenized source region is 6.87E+06 cm^3 (r=72.1 cm and h=420.7 cm, including the top and bottom nozzle regions), resulting in a gamma source of 24/6.87E+06=3.493E-06 γ/s-cm^3. Once the dose rate resulting from a single gamma per assembly is known for each energy group, the dose rate for an arbitrary gamma source can be determined simply by multiplying the source strength in each group by the dose rate contribution for that group and summing the results.

The neutron response function is generated in a similar fashion to the gamma response function, although only one ANISN neutron file is required because the neutron spectrum is adequately represented by the Cm-244 spectrum provided in Table P.5-16. Therefore, the ANISN model is executed with one neutron per assembly. As ANISN requires the source in particles per second per unit volume, the volume of the homogenized source region is 5.90E+06 cm^3 (r=72.1 cm and h=361.42 cm, height for the fuel region only). The resulting neutron source for ANISN is provided in Table P.5-16. The dose rate from secondary capture gammas is calculated in addition to the neutron dose rate. This method allows for the calculation of the neutron and capture gamma dose rate on the surface of the OS197FC TC or HSM-H knowing only the magnitude of the neutron source.

The response functions for the OS917FC TC and HSM-H are provided in Table P.5-17 and Table P.5-18, respectively. These response functions are used to compute the dose rate for each entry in the fuel qualification tables. For each qualification table, the burnup/enrichment/cooling time combination that results in the highest dose rate is selected as the design basis source.

The results of the ANISN response function evaluation are given in Table P.5-23 and Table P.5-24 for the 2.0 kW OS197FC TC and HSM-H cases, respectively. The results for the 1.5 kW OS197FC TC and HSM-H cases are given in Table P.5-25 and Table P.5-26, respectively. Note that the 1.5 kW results are assumed to be applicable to the Standardized TC and HSM-Model 102. The maximum dose rate for each table corresponds to the design basis source for that decay heat and shielding configuration.

Note also that the values presented in Table P.5-23 though Table P.5-26 are based upon decay heats rounded to the nearest 0.1 year and not the final decay heats as presented in the fuel qualification tables, which have been conservatively rounded up to the nearest 0.5 year, as the design basis sources were selected prior to the rounding process.

P.5.2.5 Reconstituted Fuel

As explained in Section P.5.2, reconstituted fuel assemblies may contain up to 10 stainless steel rods that replace damaged fuel rods. Because steel rods replace fuel rods, the decay heat of a reconstituted assembly is typically less than the decay heat of an equivalent standard assembly. Conversely, because steel contains Co-59 which activates to form Co-60, for low cooling times a reconstituted assembly typically generates higher dose rates than an equivalent standard assembly. As the half-life of Co-60 is 5.27 years, after 10 years the Co-60 activity has reduced by almost a factor of four and a reconstituted assembly no longer generates higher dose rates than an equivalent standard assembly. To bound this effect, the fuel qualification tables require that for reconstitute rods with cooling times less than 10 years, additional one year of cooling time is required. For cooling times of 10 years or greater, no additional cooling time is required to bound the reconstituted fuel with steel rods.

To quantify this statement, additional SAS2H runs are generated for reconstituted assemblies. For each burnup and enrichment corresponding to a transition point in a fuel qualification table (i.e., the point where the cooling time experiences a change of 0.5 years), reconstituted assembly SAS2H models are developed.

The SAS2H input files for a reconstituted assembly are very similar to the input files for a standard assembly except for the following changes: (1) The number of fuel rods is reduced from 208 to 198, (2) the POWER input variable is adjusted to maintain the correct burnup for the reduced fuel loading, and (3) the light elements change to reflect that 10 fuel rods have been replaced with steel rods. The constituent masses of the reconstituted fuel assembly required for the SAS2H input is provided in Table P.5-6.

Note that a reconstituted rod cannot be irradiated for more than two cycles because the first cycle will always contain fresh, undamaged fuel. To accurately model this behavior, two SAS2H models are generated for each transition point. The first SAS2H model is for only one cycle of irradiation of 10 reconstituted rods, while the second SAS2H model is for three cycles of irradiation of 10 reconstituted rods. By subtracting the single cycle source term of the reconstituted rods from the total source term (fuel and reconstituted rods) for three cycles, the source term for three cycle irradiation of fuel and two cycle irradiation of reconstituted rods is generated.

This source term is inserted into the HSM-H and OS197FC TC response functions to determine the dose rates for comparison to the design basis source dose rates. If the reconstituted fuel dose rate for either the HSM-H or OS197FC TC exceeded the dose rate with design basis fuel, an additional 0.5 year of cooling time is added to the reconstituted fuel source term. When the reconstituted fuel is examined in this fashion, no more than one additional year of cooling time is required for reconstituted fuel to be bounded by the design basis source if the decay time listed in the fuel qualification table is less than 10 years. After a cooling time greater than 10 years the effects of reconstituted fuel become insignificant.

P.5.3 Material Densities

The material weights given in Table P.5-6 for the fuel are used to calculate material densities for in-core, plenum, top and bottom regions of the fuel assembly. In addition, while the design basis fuel source terms account for the source from 24 CC, the material densities used in MCNP models of 24PTH system conservatively ignore the presence of CC, thereby any self-shielding provided by the CC materials is conservatively ignored.

In order to account for subcritical multiplication, an initial enrichment of 5.0 wt. % U-235 is used to calculate the amount of U-235 in the shielding models.

Material densities used in the various MCNP models are summarized in Table P.5-14. Material densities for the homogenized fuel/basket region used only in the ANISN models are summarized in Table P.5-15.

P.5.4 Shielding Evaluation

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from 20 or 24 fuel assemblies with CCs are calculated with the MCNP4C2 Code [5.2] at various location on and around the NUHOMS® -24PTH DSCs, HSM, and TC.

The following shielding evaluation discussion specifically addresses the NUHOMS®-24PTH-L in an HSM-H or OS197FC TC, and the 24PTH-S-LC DSC in an HSM-Model 102 or Standardized TC using the design-basis source terms determined in Section P.5.2.

P.5.4.1 Computer Program

MCNP4C2 [5.2] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP4C2 very versatile and easy to use include a powerful general source; an extensive collection of cross-section data; and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems.

P.5.4.2 Spatial Source Distribution

The source components are:

- The neutron sources due to the active fuel region,
- The gamma source due to the active fuel region,
- The gamma source due to the plenum,
- The gamma source due to the top region,
- The gamma source due to the bottom region,
- The gamma source due to the CC in the active fuel region,
- The gamma source due to the CC in the plenum region, and
- The gamma source due to the CC in the top region.

Axial peaking is accounted for in the active fuel region by inputting an axial shape, as discussed in Section P.5.2.3.

P.5.4.3 Cross Section Data

The cross-section data used is the continuous energy ENDF/B-VI provided with the MCNP4C2 code. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to be made to account for secondary gamma radiation (n,γ), if desired. All of the transfer cask dose rate calculations account for the dose rate due to secondary gamma radiation. For the HSM-H dose rate calculation, the dose rate contribution from the secondary gamma radiation is ignored because it is insignificant.

P.5.4.4 Flux-to-Dose-Rate Conversion

The flux distribution calculated by the MCNP4C2 code is converted to dose rates using flux-to-dose rate conversion factors from ANSI/ANS-6.1.1-1977 [5.8] given in Table P.5-19.

P.5.4.5 Methodology

The methodology used in the shielding analysis of the 24PTH system has been updated compared to the DORT methodology previously used to support NUHOMS® storage and transportation applications. In the current analysis, the 2-D DORT code has been replaced with the 3-D MCNP4C2 code. MCNP4C2 allows for explicit 3-D modeling of any shielding configuration and reduces the number of approximations needed in comparison to the 2-D DORT methodology. The methodology used herein is summarized below.

1. Sources are developed for all fuel regions using the source term data developed in Section P.5.2. Source regions include the active fuel region, bottom end fitting (including all materials below the active fuel region), plenum, and top end fitting (including all materials above the active fuel region). Sources for CC are added group-by-group to the fuel sources.
2. Suitable shielding material densities are calculated for all regions modeled.
3. The 3-D discrete ordinates code MCNP4C2 [5.2] is used to calculate dose rates on and around the HSM-H, OS197FC TC, and Standardized TC. The MCNP4C2 code is selected because of its ability to handle thick, multi-layered shields and account for streaming through both the HSM-H air vents and cask/DSC annulus using 3-D geometry. Note: MCNP4C2 is not used to calculate dose rates on the HSM-Model 102 surfaces. Rather, HSM-Model 102 dose rates are determined by simply scaling HSM-Model 102 results from Appendix N.5, as explain in Section P.5.4.7.2.
4. MCNP4C2 results are used to calculate offsite exposures (see Section P.10).
5. MCNP4C2 models are also generated to determine the effects of accident scenarios, such as loss of cask neutron shield, for both the OS197FC TC and Standardized TC models (Section P.11).

P.5.4.6 Assumptions

The following general assumptions are used in the analyses.

P.5.4.6.1 Source Term Assumptions

- The primary neutron source in LWR spent fuel is the spontaneous fission of ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein and is assumed to follow the ^{244}Cm fission spectrum provided in Section P.5.2.2.

- Surface gamma dose rates are calculated for the HSM and cask surfaces using the actual photon spectrum applicable for each case.

- The PWR heavy metal weight is assumed to be 0.490 MTU per assembly to bound existing PWR fuel designs.

P.5.4.6.2 HSM-H Dose Rate Analysis Assumptions

- The 24PTH-L DSC and fuel assemblies are positioned as close to the HSM-H front door as possible to maximize the HSM-H front wall dose rates.

- Planes of reflection are used to simulate adjacent HSM-Hs.

- Embedments and rebar in the HSM-H concrete are conservatively neglected.

- The borated neutron absorber sheets in the 24PTH-L DSC are modeled as aluminum.

- Axial source distribution assumed as shown in Table P.5-13.

- Fuel is homogenized within the fuel compartment, although the 24PTH-L DSC basket is modeled explicitly.

P.5.4.6.3 HSM-Model 102 Dose Rate Analysis Assumptions

- The HSM-Model 102 is not modeled in MCNP. Dose rates from the HSM-Model 102 analysis from Appendix N.5 are scaled appropriately to account for both the increase in source and decrease in DSC shield plug thicknesses as described in Section P.5.4.7.2.

- The relative change in the dose rate due to the decreased shield plug thicknesses can be estimated by treating the shield plugs as infinite planes and taking the ratio of attenuation.

- As it is estimated that the side and roof dose rates presented in Appendix N.5 are highly conservative because the fuel and basket are homogenized, additional scaling factors of 0.67 and 0.8 are introduced for gamma and neutron dose rates, respectively, at the side and roof of the HSM-Model 102 as described in Section P.5.4.7.2.

P.5.4.6.4 OS197FC TC and Standardized TC Dose Rate Analysis Assumptions

- The 24PTH-L DSC is modeled within the OS197FC TC. The 24PTH-S DSC is not modeled because it is bounded by the 24PTH-L DSC. The 24PTH-S-LC DSC is modeled within the Standardized TC.
- Only the OS197FC is modeled for the welding operation. Three inches of supplemental neutron shielding and one inch of steel are assumed to be placed on top of the 24PTH-L DSC cover plates during welding.
- During the accident case, the cask neutron shield (either water or NS-3) and the neutron shield jacket (outer steel skin) is assumed to be lost.
- The borated neutron absorber sheets in the 24PTH-L DSC and 24PTH-S-LC DSC are modeled as aluminum.
- Axial source distribution assumed as shown in Table P.5-13.
- Fuel is homogenized within the fuel compartments, although the 24PTH-L DSC and 24PTH-S-LC DSC baskets are modeled explicitly.
- In the OS197FC TC model, the gap in the cask lid is assumed to extend around the entire circumference of the lid.

P.5.4.7 Normal Condition Models

Three classes of MCNP4C2 models are developed: (1) 24PTH-L DSC in HSM-H, (2) 24PTH-L DSC in OS197FC TC, and (3) 24PTH-S-LC DSC in Standardized TC. A fourth scenario, 24PTH-S-LC DSC in HSM-Model 102, is analyzed by scaling the results from a similar analysis in Appendix N.5. These models are described in subsequent sections.

P.5.4.7.1 24PTH-L DSC in HSM-H

Two three-dimensional MCNP4C2 models are developed for the 24PTH-L DSC within a HSM-H, one model for neutrons and the other for gammas. These models are presented in Figure P.5-3 through Figure P.5-7. The HSM-H length is designated as the x axis, the width as the y axis, and the height as the z axis. The HSM-H door is designated as the south side and the –x direction, with the east wall as the –y direction. The roof is the +z direction. The east wall is designated as a reflective boundary and an end shield wall (3 ft thick) is attached to the west wall.

The bottom (bottom of bottom fitting) of the fuel assembly is assigned to an x plane at -213.84 cm. The center of the HSM-H is at y=0 and z=0. The 24PTH-L DSC lid is located 5" from the HSM-H rear wall (x=254.84 cm) which places the bottom of the DSC at x=-232.69 cm, about 9.5 in from the door interior. The 24PTH-L DSC support rails are not included in the model. The heat shields are modeled as flat plates without fins or louvers, and horizontal vent "liner" plates (2 cm thick) are modeled in the top side vents.

Dose rates are calculated on thin cells surrounding the HSM-H and are segmented into 30 cm increments to capture the peak dose rates. Dose rates are also calculated at the inlet and outlet vents. Dose rates for this scenario are provided in Table P.5-1. Dose rates for the front, roof, and side shield wall surface at DSC centerline of the HSM-H are also plotted as a function of distance in Figure P.5-16 through Figure P.5-18 respectively.

A sample MCNP4C2 model input file of HSM-H with 24PTH-L DSC is included in Section P.5.5.2.

P.5.4.7.2 24PTH-S-LC DSC in HSM-Model 102

MCNP4C2 was not used for the shielding analysis of the HSM-Model 102. Rather, dose rates on the surface of the HSM-Model 102 were estimated by appropriately scaling the HSM-Model 102 results from Appendix N.5 to properly account for the differences in source and DSC designs.

First, the scaling factors due to the higher source are determined. These scaling factors are generated by comparing the sources from Appendix N.5 to the design basis sources for the HSM-Model 102 from Table P.5-11. The design basis gamma source from Appendix N.5 is for a decay heat of 1.3 kW/assembly, 46.1 GWd/MTU burnup, 3.2 wt.% U-235, and 5.5 years cooled, as shown in Appendix N.5, Table N.5-10. The design basis neutron source is 9.65E+08 n/s, from Table N.5-13 of Appendix N.5.

The neutron source scaling factor would simply be the ratio of the total source strengths, or 0.22, because both neutron sources follow the Cm-244 spectrum. As can be seen, the neutron source from Appendix N.5 is stronger than the 24PTH neutron source by this factor. For the gammas, it is necessary to compare the sources on a group by group basis because the spectra are different.

Using the HSM-H response function from Table P.5-18, it may be demonstrated that the gamma dose rate on the HSM surface is dominated by gammas in groups 29, 31, and 32 (i.e, 1-1.33 MeV, 1.33-1.66 MeV, and 2-2.5 MeV), so further analysis is limited to these three groups. The gamma scaling factor is then generated by weighting the dose rate fraction of each of these three energy groups for the Appendix N.5 gamma source by the ratio of gammas in each energy group and summing the results. Using this methodology, the gamma source scaling factor is approximately 1.6.

Note that the gamma source scaling factor of 1.6 is valid only when the source is dominated by the fuel, such as the HSM-Model 102 roof, side, front vent, and roof vent. At the front and back of the HSM-Model 102, where dose rates are assumed to have roughly equal contributions from the nozzles and fuel, the scaling factor must be reexamined. The dose rate from the nozzles is primarily from Co-60, which is in energy groups 31 and 32. The gamma ratio for these energy groups is approximately 1.04, so that the gamma energy scaling factors at the front and back are approximately 0.5*1.04+0.5*1.6*24/20~1.6. The factor 24/20 is introduced at the ends because the 24PTH-S-LC DSC contains 24 assemblies, while Appendix N.5 utilized only 20 design basis assemblies. The scaling factor at the HSM-Model 102 side do not need this 24/20 factor because the center four assemblies are self-shielded.

In summary, the neutron source scaling factor is 0.22 (everywhere), while the gamma source scaling factor is 1.6 (everywhere). Note, however, that these scaling factors do not account for the differences between the 24PTH-S-LC DSC and the 24PHB DSC modeled in Appendix N.5. The 24PTH-S-LC DSC has 5.5 cm of steel and 7.0 cm of lead on the bottom end, while the 24PHB has 6.3 cm of steel and 8.9 cm of lead on the bottom end. Conversely, the 24PTH-S-LC DSC has 9.0 cm of steel and 10.9 cm of lead on the top end, while the 24PHB DSC has 7.6 cm of steel and 10.1 cm of lead on the top end. Note that the 24PTH-S-LC DSC and the 24PHB DSC have the same side dimensions so that the combined source and geometrical scaling factor at the side is 1.6.

To estimate the geometrical scaling factors at the ends, it is assumed that attenuation of gamma radiation in the shield plugs is similar to that of gamma radiation in an infinite slab. Macroscopic cross-sections in stainless steel and lead for gamma and neutron radiation in different energy groups are given in Table P.5-20. Similar to the source ratio of gamma and neutrons per energy group as explained in the preceeding paragraphs, the geometric scaling factors can be estimated by using an attenuation ratio between the 24PHB and 24PTH-S-LC DSCs. The attenuation is calculated using a simple exponential relation:

$\exp(-\sum_{i=1}^{n} t_i \Sigma_j^i)$, here ***i=1,..,n*** is number of different shielding media, t_i - is thickness of media ***i***, Σ_j^i -is a macroscopic cross section for CASK 81 energy group ***j*** in media ***i***. By using this equation, the attenuation ratio is determined for each energy group at both the front and back of the HSM-Model 102. Using this approach and focusing on the energy groups of interest as noted above, the total source and geometric scaling factors are determined to be 6.8 for the front gamma and 0.6 for the rear gamma. The total source and geometric scaling factors for neutrons are estimated as be 0.4 for the front and 0.1 for the rear.

The gamma and neutron scaling factors at the side (or roof), 1.6 and 0.22, respectively, may be further reduced to account for the basket homogenization in the model presented in Appendix N.5. It has been demonstrated [5.9] that basket homogenization results in conservatism of 67% and 80% for gamma and neutron dose rates, respectively, when explicit and homogenized MCNP basket models are compared. If these homogenization corrections are applied to the side (or roof) scaling factors, the final scaling factors become 1.1 and 0.2 for gammas and neutrons, respectively.

Therefore, the final scaling factors that consider both changes in source, DSC, and basket homogenization effects are as follows:

- HSM-Model 102 side, roof, roof birdscreen, and front birdscreen: 1.1 for gammas and 0.2 for neutrons

- HSM-Model 102 front: 6.8 for gammas and 0.4 for neutrons

- HSM-Model 102 back: 0.6 for gammas and 0.1 for neutrons

Using these scaling factors, the average and maximum dose rates on the surface of the HSM-Model 102 are provided in Table P.5-21 and Table P.5-22.

P.5.4.7.3 24PTH-L DSC in OS197FC TC

Two three-dimensional MCNP4C2 models are employed for shielding analyses of the 24PTH-L DSC within an OS197FC TC, one model for neutrons and the other for gammas. These models are presented in Figure P.5-8 through Figure P.5-11. The z-axis in the MCNP models coincides with the axis of rotation of the cask and the 24PTH-L DSC. Select features within the cask and on its surface are neglected because they produce only localized effects and have minimal impact on operational dose rates. Examples of neglected features include the 24 neutron shield panel support angles, the 4 trunnions, relief valves, clevises, and eyebolts. With the exception of the 24 neutron shield support angles and the trunnions, the balance of these items are local features that increase the shielding in a small area without replacing any of the shielding material which is included in the model. The additional shielding material that these features provide is not smeared into the bulk shielding, nor is any credit taken for it in the occupational exposure calculation. The 24 neutron shield support angles provide support for the neutron shield skin, which contains the water for the neutron shield. The steel that forms these angles is not smeared with the water in the neutron shield; rather it is modeled as water. This is conservative for gamma radiation because water is less than one seventh the density of steel. The density of the neutron shield water used in the cask MCNP models is 0.96 g/cm^3. The resultant reduction in the hydrogen density as compared to full density water results in the water attenuating the neutron dose rate at about the same rate as that for full density steel. Therefore, replacing the steel with the lower density water results in little to no effect on the neutron dose rate outside the cask.

The trunnions penetrate the neutron shield, which locally changes the shielding configuration of the neutron shield. The trunnions are thick steel structures filled with NS-3 neutron shielding material. These structures protrude well past the neutron shield and are made of materials which provide more gamma shielding and comparable neutron shielding as compared to the 0.96 g/cm^3 water that these replace. In addition, with the exception of the neutron shield support angles, none of these features is located near the axial center of the cask where the surface dose rate is the largest due to the axial peaking of the fuel.

Design features relevant to the shielding analysis of the OS197FC TC and 24PTH-L DSC are modeled in MCNP4C2. The overall length of the OS197FC TC is 202.97". The outer diameter of the OS197FC TC is 85.50" (neutron shield included). The outer diameter excluding the neutron shield is 79.12". The bottom of the OS197FC TC is designed to mate with a 24PTH-L DSC. The overall length of the 24PTH-L DSC is 192.55" (excluding the grapple) and its outer diameter is 67.19". The bottom end of the 24PTH-L DSC is in contact with the structural shell assembly of the transfer cask.

The OS197FC TC has a ventilated top lid to facilitate air circulation using fan as described in Section P.4.7. In MCNP4C2, the ventilation cutouts in the top cover assembly are modeled as complete annular gaps. The supporting steel around the bolts is not included for modeling convenience and conservatism in the results. Likewise, the neutron shielding in the top lid is also reduced to the inner radial dimension to conservatively account for the bolt cutouts. Use of cone adapters and cask spacers during air circulation will offset shielding lost by the removal of ram access cover.

Dose rates for this scenario are provided in Table P.5-3. Dose rates at the sides, top, and bottom of this cask are presented graphically in Figure P.5-19 through Figure P.5-21.

A sample MCNP4C2 model input file for OS197FC TC with 24PTH-L DSC is included in Section P.5.5.3.

P.5.4.7.4 24PTH-S-LC in Standardized TC

Two three-dimensional MCNP4C2 models are employed for shielding analyses of the 24PTH-S-LC DSC within a Standardized TC, one model for neutrons and the other for gammas. These models are presented in Figure P.5-12 through Figure P.5-15. The z-axis in the MCNP models coincides with the axis of rotation of the cask and the 24PTH-DSC. Select features within the cask and on its surface are neglected because they produce only localized effects and have minimal impact on operational dose rates. Examples of neglected features include the 24 neutron shield panel support angles, the 4 trunnions, relief valves, clevises, and eyebolts as justified in Section P.5.4.7.3. Design features relevant to the shielding analysis of the cask and DSC are modeled in MCNP.

The overall length of the standardized transfer cask is 192.97". The outer diameter of the cask is 85.50" (neutron shield included). The outer diameter excluding the neutron shield is 79.12". The bottom of the transfer cask is designed to mate with a DSC. The overall length of the 24PTH-S-LC is 186.55", and its outer diameter is 67.19". The bottom end of the 24PTH-DSC is in contact with the structural shell assembly of the transfer cask.

Dose rates for this scenario are provided in Table P.5-5. Dose rates at the sides, top, and bottom of this cask are presented graphically in Figure P.5-22 through Figure P.5-24.

P.5.4.8 Accident Models

No accident models were developed for the HSM-H because no accident scenario in Chapter P.11 has been identified that would alter the dose rates provided in Table P.5-1. For the HSM-Model 102 in an array, in an accident condition HSM-Model 102 is assumed to slide next to an adjacent HSM and therefore double the gap on one side as described in Chapter P.11. It is further conservatively assumed the dose rates from the array double as a result of this accident. The HSM-Model 102 accident analysis and results are provided in Chapter P.11.

For both the OS197FC TC and Standardized TC, accident cases are performed assuming the neutron shield and steel neutron shield jacket (outer skin) of each have been torn off. Accident dose rates at 1m, 100m, and 500m from the side of the cask are presented in Table P.5-3 and Table P.5-5 for the OS197FC TC and Standardized TC, respectively.

P.5.4.9 OS197FC TC Models During Fuel Loading Operations

MCNP4C2 models are developed for the cask decontamination and welding operations during fuel loading using the 24PTH-L DSC. As the dose rates from this cask with 24PTH-L DSC bounds the 24PTH-S-LC DSC due to the higher source term used in 24PTH-L DSC, calculations are not performed for the loading operations with Standardized TC with 24PTH-S-LC DSC.

Cask Decontamination. The 24PTH-L DSC and the OS197FC TC are assumed to be completely filled with water, including the region between 24PTH-DSC and cask, which is referred to as the "cask/24PTH-DSC annulus." The 24PTH-DSC inner cover plate is assumed to be in place and the temporary shielding has not yet been installed. Results for this case are provided in Table P.5-4.

Welding and 24PTH-L DSC Draining. Before the start of welding operation, approximately 60% of the water in the DSC cavity is removed due to hydrogen generation. A dry DSC cavity is assumed in all welding models to be conservative. Temporary shielding consisting of three inches of NS3 and one inch of steel is assumed to cover the 24PTH-L DSC top shield plug. In addition, the DSC outer top cover plate is not present. The cask/24PTH-DSC annulus is assumed to remain completely filled with water. Results for this case are provided in Table P.5-4.

P.5.5 Appendix

Section P.5.5.1 includes a sample SAS2H/ORIGEN-S code input file used for the NUHOMS®-24PTH system. The MCNP4C2 code models are described in Section P.5.4. Section P.5.5.2 includes a sample MCNP4C2 code input file used for the HSM-H analysis. Section P.5.5.3 includes a sample MCNP4C2 code input file used for the OS197FC TC analysis. Section P.5.5.4 includes a sample ANISN code input file used for the HSM-H analysis.

INPUT Files are Proprietary, and intentionally removed.

P.5.5.1 Sample SAS2H/ORIGEN-S Input File

P.5.5.2 Sample HSM-H MCNP4C2 Model

P.5.5.3 Sample OS197FC TC MCNP4C2 Model

P.5.5.4 Sample ANISN Model (TC –Group 23)

P.5.6 References

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5.2 "Monte Carlo N-Particle Transport Code System," CCC-701, Oak Ridge National Laboratory, RSICC Computer Code Collection, June 2001.

5.3 CASK-81 - 22 Neutron, 18 Gamma-Ray Group, P3, Cross Sections for Shipping Cask Analysis," DLC-23, Oak Ridge National Laboratory, RSIC Data Library Collection, June 1987.

5.4 Ludwig, S.B., and J.P. Renier, "Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018 Oak Ridge National Laboratory, December 1989.

5.5 "ANISN-ORNL - One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering", CCC-254, Oak Ridge National Laboratory, RSIC Computer Code Collection, April 1991.

5.6 "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," NUREG/CR-6801, Oak Ridge National Laboratory.

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5.8 "American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society, LaGrange Park, Illinois, March 1977.

5.9 K.Ueki, N. Nariyama, A. Ohashi, A. Yamaji. "Measurement of Dose-Equivalent Rates around a Cask and Monte Carlo Analysis with Actual Configuration of Fuel Basket". Journal of Nuclear Science and Technology, (Supplement 1, p.324-328), Atomic Energy Society of Japan, March 2000, ISSN 0022-3131.

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5.11 MD DeHart and OW Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, September 1996.

5.12 OW Hermann, SM Bowman, MC Brady, CV Parks, "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, March 1995.

5.13 Japan Atomic Energy Research Institute, "Technical Development on Burn-up Credit for Spent LWR Fuels," JAERI-Tech 2000-071, September 21, 2000.

5.14 U.S. Nuclear Regulatory Commission, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High Burnup LWR Fuel," NUREG/CR-6700, Published January 2001, ORNL/TM-2000/284.

5.15 "Characteristics of Potential Repository Waste," DOE/RW-0184-R21, Volume 1, Oak Ridge National Laboratory, Tennessee, July 1992.

Table P.5-1
Summary of NUHOMS®-24PTH-L DSC in HSM-H, Maximum and Average Dose Rates, Configuration 2 (2)

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
HSM Roof (centerline)	2.01	0.038	0.5	0.018	20.6	0.037
HSM Roof Birdscreen	205.8	0.019	4.1	0.012	209.9	0.018
HSM End (Side) Shield Wall Surface	3.4	0.081	0.1	0.016	3.5	0.079
HSM Door Exterior Surface (centerline)	1.3	0.143	0.1	0.524	1.3	0.139
HSM Front Birdscreen	1232.0	0.068	5.5	0.076	1237.0	0.068

Dose Rate Location	Average (mrem/hr)	Gamma MCNP 1σ Error	Average Neutron (mrem/hr)	Neutron MCNP 1σ Error	Average Total (mrem/hr)	Total MCNP 1σ Error
HSM Roof	19.9	0.010	0.5	0.006	20.3	0.010
HSM End (Side) Shield Wall Surface	1.0	0.016	0.1	0.033	1.1	0.015
HSM Front	10.3	0.090	0.1	0.066	10.4	0.089
HSM Back Shield Wall	0.6	0.074	0.1	0.025	0.6	0.074

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) Dose calculated using Configuration 2 in 24PTH-L DSC bounds configurations 1, 3 and 4.

Table P.5-2
Summary of NUHOMS®-24PTH-S-LC DSC in HSM-Model 102, Maximum and Average Dose Rates, Configuration 5

Dose Rate Location	Maximum Gamma (mrem/hr)	Maximum Neutron (mrem/hr)	Maxium Total (mrem/hr)
HSM Roof (centerline)	59.3	0.2	59.5
HSM Roof Birdscreen	976.5	3.2	979.7
HSM End (Side) Shield Wall Surface	266.9	0.3	267.2
HSM Door Exterior Surface (centerline)	60.6	1.6	62.2
HSM Front Birdscreen	489.6	2.5	492.1
HSM Back Shield Wall	2.5	0.02	2.5

Dose Rate Location	Average Gamma (mrem/hr)	Average Neutron (mrem/hr)	Average Total (mrem/hr)
HSM Roof	47.3	0.2	47.5
HSM End (Side) Shield Wall Surface	31.7	0.1	31.8
HSM Front	45.6	0.9	46.5
HSM Back Shield Wall	0.8	0.01	0.8

Table P.5-3
Summary of NUHOMS®-24PTH-L DSC, OS197FC TC Maximum Dose Rates During Transfer Operations, Configuration 2

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total[(1)] (mrem/hr)	Total MCNP 1σ Error
Cask Side Surface (Radial)	7.45E+02	0.0180	7.56E+02	0.0120	1.50E+03	0.0108
Cask Top Axial Surface	2.37E+02	0.0566	4.48E+01	0.0499	2.61E+02	0.0523
Cask Bottom Axial Surface	1.66E+03(2)	0.0353	2.57E+03(2)	0.0246	4.23E+03(2)	0.0204
1 ft from Cask Side (Radial)	4.76E+02	0.0179	4.82E+02	0.0107	9.58E+02	0.0104
1 ft from Cask Top Axial Surface	7.86E+01	0.0741	3.11E+01	0.0455	9.58E+01	0.0636
1 ft from Cask Bottom Axial Surface	9.80E+02	0.0355	9.44E+02	0.0340	1.92E+03	0.0246
3 ft from Cask Side (Radial)	2.85E+02	0.0163	2.78E+02	0.0096	5.63E+02	0.0095
3 ft from Cask Top Axial Surface	3.95E+01	0.1189	1.47E+01	0.0550	5.05E+01	0.0972
3 ft from Cask Bottom Axial Surface	3.44E+02	0.0341	2.79E+02	0.0600	6.23E+02	0.0328
Cask 1 m (Radial) Accident Condition	3.10E+02	0.0011	3.19E+03	0.0124	3.51E+03	0.0128
Cask 100 m (Radial) Accident Condition	1.50E-01	0.0366	5.10E-01	0.0134	6.61E-01	0.0124
Cask 500 m (Radial) Accident Condition	4.95E-04	0.8299	4.10E-04	0.0305	9.05E-04	0.0194

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 340 mrem/hr gamma, 419 mrem/hr neutron for a total average dose rate of 758 mrem/hr.

Table P.5-4
Summary of NUHOMS®-24PTH-L DSC, OS197FC TC Maximum Dose Rates During Decontamination and Welding Operations, Configuration 2

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
Decontamination						
Cask Side Surface (Radial)	4.34E+02	0.0210	8.23E+02	0.0069	1.26E+03	0.0085
Top Axial Surface	7.83E+02	0.0272	3.14E-01	0.2858	7.83E+02	0.0272
Cask Bottom Axial Surface	1.15E+03(2)	0.0478	5.83E+01(2)	0.0126	1.21E+03(2)	0.0455
1 ft from Cask Side (Radial)	2.82E+02	0.0206	5.32E+02	0.0060	8.14E+02	0.0082
1 ft from Top Axial Surface	5.93E+02	0.0262	1.05E+01	0.0304	5.93E+02	0.0262
1 ft from Cask Bottom Axial Surface	7.07E+02	0.0486	2.73E+01	0.0215	7.29E+02	0.0471
3 ft from Cask Side (Radial)	1.68E+02	0.0187	3.14E+02	0.0054	4.83E+02	0.0074
3 ft from Top Axial Surface	4.02E+02	0.0305	9.40E+00	0.0099	4.03E+02	0.0305
3 ft from Cask Bottom Axial Surface	2.54E+02	0.0494	1.86E+01	0.0085	2.62E+02	0.0480
Welding						
Cask Side Surface (Radial)	6.22E+02	0.0224	5.46E+02	0.0123	1.17E+03	0.0132
Top Axial Surface	8.56E+02	0.0264	3.37E+01	0.0658	8.84E+02	0.0256
Cask Bottom Axial Surface	1.64E+03(3)	0.0397	2.69E+03(3)	0.0297	4.34E+03(3)	0.0238
1 ft from Cask Side (Radial)	4.06E+02	0.0217	3.51E+02	0.0108	7.58E+02	0.0127
1 ft from Top Axial Surface	6.48E+02	0.0371	2.42E+01	0.0814	6.69E+02	0.0360
1 ft from Cask Bottom Axial Surface	9.78E+02	0.0401	9.23E+02	0.0395	1.90E+03	0.0282
3 ft from Cask Side (Radial)	2.47E+02	0.0191	2.05E+02	0.0097	4.52E+02	0.0113
3 ft from Top Axial Surface	4.44E+02	0.3175	1.33E+01	0.0815	4.51E+02	0.3124
3 ft from Cask Bottom Axial Surface	3.41E+02	0.0386	2.51E+02	0.0663	5.92E+02	0.0358

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 238 mrem/hr gamma, 13 mrem/hr neutron for a total average dose rate of 251 mrem/hr.

(3) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 331 mrem/hr gamma, 417 mrem/hr neutron for a total average dose rate of 748 mrem/hr. Note that this bottom axial dose rate has no impact on the occupational exposure because no operations are performed near bottom axial location.

Table P.5-5
Summary of NUHOMS®-24PTH-S-LC DSC, Standardized TC Maximum Dose Rates During Transfer Operations, Configuration 5

Dose Rate Location	Maximum Gamma (mrem/hr)	Gamma MCNP 1σ Error	Maximum Neutron (mrem/hr)	Neutron MCNP 1σ Error	Maximum Total(1) (mrem/hr)	Total MCNP 1σ Error
Cask Side Surface (Radial)	4.19E+02	0.0600	1.81E+02	0.0178	5.77E+02	0.0273
Cask Top Axial Surface	3.01E+01	0.0894	8.03E+00	0.0778	3.25E+01	0.0829
Cask Bottom Axial Surface	4.42E+03(2)	0.1154	3.30E+02(2)	0.0276	4.75E+03(2)	0.1074
1 ft from Cask Side (Radial)	2.95E+02	0.0536	1.14E+02	0.0176	3.78E+02	0.0242
1 ft from Cask Top Axial Surface	2.06E+01	0.0251	5.44E+00	0.0576	2.43E+01	0.0224
1 ft from Cask Bottom Axial Surface	2.31E+03	0.1261	1.17E+02	0.0308	2.43E+03	0.1200
3 ft from Cask Side (Radial)	1.89E+02	0.0449	9.20E+01	0.0158	2.31E+02	0.0368
3 ft from Cask Top Axial Surface	1.06E+01	0.0201	2.73E+00	0.0567	1.31E+01	0.0165
3 ft from Cask Bottom Axial Surface	8.82E+02	0.1398	3.33E+01	0.0403	9.15E+02	0.1347
Cask 1 m (Radial) Accident Condition	3.44E+02	0.0018	4.18E+02	0.0219	7.62E+02	0.0258
Cask 100 m (Radial) Accident Condition	1.67E-01	0.0872	6.76E-02	0.0232	2.35E-01	0.0308
Cask 500 m (Radial) Accident Condition	6.20E-04	1.1638	5.34E-05	0.0341	6.74E-04	0.0279

Notes:

(1) Gamma and Neutron dose rate peaks do not always occur at same location therefore the total dose rate is not always the sum of the gamma plus neutron dose rate.

(2) The peak bottom surface dose rate is directly below the grapple ring cut out in the bottom of the cask. The bottom average dose rates, including the grapple area, are 730 mrem/hr gamma, 61 mrem/hr neutron for a total average dose rate of 791 mrem/hr.

Table P.5-6
PWR Fuel Assembly Material Mass

Fuel Assembly Region, length	Fuel Assembly Part	Material	Standard Mass (kg)	Reconstituted Mass (kg)
Top Nozzle, 6.23 in.	Top Nozzle/Misc. Steel	SS-304	9.2	9.2
	Hold Down Spring	Inconel-718	1.8	1.8
Plenum, 8.73 in.	Upper Spring	Inconel-718	4.3	4.3
	Upper End Cap	Zircaloy-4	1.0	1.0
	Encompassing Clad.	Zircaloy-4	5.8	5.5
	Upper End Grid	Inconel-718	1.1	1.1
	Stainless Steel Rods	SS304	na	1.7
In-core Region, 142.29 in.	Fuel Stack	UO_2	490	466
	Encompassing Clad.	Zircaloy-4	101.1	96.2
	Encompassing Guide Tube	Zircaloy-4	6.3	6.3
	Spacer Grids	Inconel-718	5.0	5.0
	Grid Supports	Zircaloy-4	0.64	0.64
	Stainless Steel Rods	SS304	na	27.2
Bottom Nozzle, 8.38 in.	Lower End Plug	Zircaloy-4	8.9	8.5
	Encompassing Guide Tube	Zircaloy-4	0.1	0.1
	Lower Guide Tube Plugs	Zircaloy-4	1.4	1.4
	Lower End Fitting	SS 304	8.2	8.2
	Lower End Grid	Inconel-718	1.1	1.1
	Stainless Steel Rods	SS304	na	0.5

Table P.5-7
Elemental Composition of LWR Fuel-Assembly Structural Materials

Element Atomic Number		Material Composition, grams per kg of material				
		Zircaloy-4	Inconel-718	Inconel X-750	Stainless Steel 304	U0$_2$ Fuel
H	1	1.30E-02	-	-	-	-
Li	3	-	-	-	-	1.00E-03
B	5	3.30E-04	-	-	-	1.00E-03
C	6	1.20E-01	4.00E-01	3.99E-01	8.00E-01	8.94E-02
N	7	8.00E-02	1.30E+00	1.30E+00	1.30E+00	2.50E-02
O	8	9.50E-01	-	-	-	1.34E+02
F	9	-	-	-	-	1.07E-02
Na	11	-	-	-	-	1.50E-02
Mg	12	-	-	-	-	2.00E-03
Al	13	2.40E-02	5.99E+00	7.98E+00	-	1.67E-02
Si	14	-	2.00E+00	2.99E+00	1.00E+01	1.21E-02
P	15	-	-	-	4.50E-01	3.50E-02
S	16	3.50E-02	7.00E-02	7.00E-02	3.00E-01	-
Cl	17	-	-	-	-	5.30E-03
Ca	20	-	-	-	-	2.00E-03
Ti	22	2.00E-02	7.99E+00	2.49E+01	-	1.00E-03
V	23	2.00E-02	-	-	-	3.00E-03
Cr	24	1.25E+00	1.90E+02	1.50E+02	1.90E+02	4.00E-03
Mn	25	2.00E-02	2.00E+00	6.98E+00	2.00E+01	1.70E-03
Fe	26	2.25E+00	1.80E+02	6.78E+01	6.88E+02	1.80E-02
Co	27	1.00E-02	4.69E+00	6.49E+00	8.00E-01	1.00E-03
Ni	28	2.00E-02	5.20E+02	7.22E+02	8.92E+01	2.40E-02
Cu	29	2.00E-02	9.99E-01	4.99E-01	-	1.00E-03
Zn	30	-	-	-	-	4.03E-02
Zr	40	9.79E+02	-	-	-	-
Nb	41	-	5.55E+01	8.98E+00	-	-
Mo	42	-	3.00E+01	-	-	1.00E-02
Ag	47	-	-	-	-	1.00E-04
Cd	48	2.50E-04	-	-	-	2.50E-02
In	49	-	-	-	-	2.00E-03
Sn	50	1.60E+01	-	-	-	4.00E-03
Gd	64	-	-	-	-	2.50E-03
Hf	72	7.80E-02	-	-	-	-
W	74	2.00E-02	-	-	-	2.00E-03
Pb	82	-	-	-	-	1.00E-03
U	92	2.00E-04	-	-	-	8.81E+02

Table P.5-8
Flux Scaling Factors By Fuel Assembly Region

Fuel Assembly Region	Flux Factor
Bottom	0.20
In-Core	1.00
Plenum	0.20
Top	0.10

Table P.5-9
Gamma and Neutron Source Term for 2.0 kW Fuel in TC (62 GWd/MTU, 3.4 wt. % U-235 and 5.6-Year Cooled Fuel)

CASK-81 Energy Group	E_{upper} (MeV)	E_{mean} (MeV)	Top Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Bottom Region (γ/s/assembly)
23	10	9	0.000E+00	0.000E+00	9.552E+05	0.000E+00
24	8	7.25	0.000E+00	0.000E+00	4.499E+06	0.000E+00
25	6.5	5.75	0.000E+00	0.000E+00	2.293E+07	0.000E+00
26	5	4.5	0.000E+00	0.000E+00	5.713E+07	0.000E+00
27	4	3.5	7.138E-09	4.281E-08	1.340E+10	8.458E-09
28	3	2.75	1.202E+05	3.815E+05	1.060E+11	1.731E+05
29	2.5	2.25	7.749E+07	2.460E+08	2.381E+12	1.116E+08
30	2	1.83	8.772E+01	3.566E+02	1.362E+12	2.444E+02
31	1.66	1.495	3.265E+12	1.037E+13	9.292E+13	4.703E+12
32	1.33	1.165	1.156E+13	3.671E+13	3.191E+14	1.666E+13
33	1	0.9	5.101E+10	2.421E+10	5.713E+14	8.878E+10
34	0.8	0.7	1.392E+09	3.349E+10	3.869E+15	4.036E+10
35	0.6	0.5	3.940E+07	4.876E+10	1.269E+15	7.483E+10
36	0.4	0.35	6.225E+08	4.161E+09	8.345E+13	4.255E+09
37	0.3	0.25	4.751E+08	2.063E+09	1.234E+14	1.533E+09
38	0.2	0.15	9.560E+09	3.960E+10	4.333E+14	2.797E+10
39	0.1	0.075	3.962E+10	1.264E+11	5.390E+14	5.798E+10
40	0.05	0.025	3.172E+11	1.128E+12	2.724E+15	6.408E+11
Total Gamma			1.525E+13	4.848E+13	1.003E+16	2.230E+13
Total Neutron			1.67E+09 n/s/assembly			

Table P.5-10
Gamma and Neutron Source Term for 2.0 kW Fuel in HSM-H (41 GWd/MTU, 3.3 wt. % U-235 and 3.0-Year Cooled Fuel)

CASK-81 Energy Group	E_{upper} (MeV)	E_{mean} (MeV)	Top Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Bottom Region (γ/s/assembly)
23	10	9	0.000E+00	0.000E+00	2.227E+05	0.000E+00
24	8	7.25	0.000E+00	0.000E+00	1.049E+06	0.000E+00
25	6.5	5.75	0.000E+00	0.000E+00	5.346E+06	0.000E+00
26	5	4.5	0.000E+00	0.000E+00	1.332E+07	0.000E+00
27	4	3.5	2.202E-09	1.321E-08	4.290E+10	2.610E-09
28	3	2.75	1.070E+05	3.402E+05	3.473E+11	1.540E+05
29	2.5	2.25	6.902E+07	2.194E+08	1.371E+13	9.935E+07
30	2	1.83	7.993E+05	2.573E+06	4.527E+12	1.188E+06
31	1.66	1.495	2.909E+12	9.245E+12	1.033E+14	4.186E+12
32	1.33	1.165	1.030E+13	3.274E+13	3.130E+14	1.482E+13
33	1	0.9	2.486E+11	8.047E+10	6.585E+14	4.365E+11
34	0.8	0.7	9.034E+08	3.577E+10	3.430E+15	4.780E+10
35	0.6	0.5	8.728E+07	5.848E+10	1.722E+15	8.964E+10
36	0.4	0.35	5.548E+08	4.380E+09	1.713E+14	4.821E+09
37	0.3	0.25	4.235E+08	2.027E+09	2.281E+14	1.657E+09
38	0.2	0.15	8.515E+09	3.815E+10	8.340E+14	2.929E+10
39	0.1	0.075	3.529E+10	1.130E+11	9.470E+14	5.208E+10
40	0.05	0.025	2.933E+11	1.115E+12	4.275E+15	7.613E+11
Total Gamma			1.380E+13	4.343E+13	1.270E+16	2.043E+13
Total Neutron			3.87E+08 n/s/assembly			

Table P.5-11

Gamma and Neutron Source Term for 1.5 kW Fuel in TC or HSM (32 GWd/MTU, 2.6 wt. % U-235 and 3.0-Year Cooled Fuel)

CASK-81 Energy Group	E_{upper} (MeV)	E_{mean} (MeV)	Top Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Bottom Region (γ/s/assembly)
23	10	9	0.000E+00	0.000E+00	1.206E+05	0.000E+00
24	8	7.25	0.000E+00	0.000E+00	5.681E+05	0.000E+00
25	6.5	5.75	0.000E+00	0.000E+00	2.900E+06	0.000E+00
26	5	4.5	0.000E+00	0.000E+00	7.220E+06	0.000E+00
27	4	3.5	6.500E-10	3.900E-09	3.370E+10	7.700E-10
28	3	2.75	9.209E+04	2.928E+05	2.730E+11	1.325E+05
29	2.5	2.25	5.940E+07	1.890E+08	1.070E+13	8.550E+07
30	2	1.83	7.007E+05	2.250E+06	3.550E+12	1.040E+06
31	1.66	1.495	2.500E+12	7.960E+12	7.970E+13	3.600E+12
32	1.33	1.165	8.860E+12	2.820E+13	2.480E+14	1.280E+13
33	1	0.9	1.970E+11	6.410E+10	4.490E+14	3.460E+11
34	0.8	0.7	7.560E+08	2.900E+10	2.530E+15	3.860E+10
35	0.6	0.5	7.590E+07	4.730E+10	1.220E+15	7.240E+10
36	0.4	0.35	4.770E+08	3.630E+09	1.340E+14	3.940E+09
37	0.3	0.25	3.640E+08	1.710E+09	1.770E+14	1.370E+09
38	0.2	0.15	7.330E+09	3.220E+10	6.460E+14	2.430E+10
39	0.1	0.075	3.040E+10	9.720E+10	7.360E+14	4.470E+10
40	0.05	0.025	2.510E+11	9.460E+11	3.320E+15	6.330E+11
Total Gamma			1.185E+13	3.735E+13	9.550E+15	1.752E+13
Total Neutron			2.10E+08 n/s/assembly			

Table P.5-12
Design-Basis CC Source Terms

CASK-81 Energy Group	E_{upper} (MeV)	E_{mean} (MeV)	Top Region γ/s/CC	Plenum Region γ/s/CC	Fuel Region γ/s/CC
23	10	9	0.000E+00	0.000E+00	0.000E+00
24	8	7.25	0.000E+00	0.000E+00	0.000E+00
25	6.5	5.75	0.000E+00	0.000E+00	0.000E+00
26	5	4.5	0.000E+00	0.000E+00	0.000E+00
27	4	3.5	3.947E-15	6.520E-14	7.266E-18
28	3	2.75	3.942E+04	2.179E+04	5.495E+05
29	2.5	2.25	1.274E+07	7.040E+06	1.775E+08
30	2	1.83	9.577E+01	8.812E+01	9.153E-05
31	1.66	1.495	7.138E+11	3.946E+11	9.953E+12
32	1.33	1.165	1.690E+12	9.340E+11	2.356E+13
33	1	0.9	6.951E+09	4.180E+09	4.699E+09
34	0.8	0.7	4.155E+09	1.783E+10	2.835E+09
35	0.6	0.5	3.235E+07	3.854E+10	4.508E+08
36	0.4	0.35	7.014E+07	2.431E+10	9.776E+08
37	0.3	0.25	2.060E+08	3.482E+09	2.872E+09
38	0.2	0.15	1.217E+09	3.534E+09	1.697E+10
39	0.1	0.075	8.191E+09	5.176E+09	1.142E+11
40	0.05	0.025	8.484E+10	1.382E+11	1.170E+12
Total Gamma/sec/CC			2.509E+12	1.564E+12	3.483E+13

Table P.5-13
Source Term Peaking Factor Summary

		OS197FC TC or Standardized TC		HSM-H	
Active Fuel Zone Number	**Active Fuel Zone Center (% of height)**	**Gamma Profile**	**Neutron Profile**	**Gamma Profile**	**Neutron Profile**
1	2.78	0.573	0.108	0.660	0.190
2	8.33	0.917	0.707	0.936	0.768
3	13.89	1.066	1.291	1.045	1.193
4	19.44	1.106	1.496	1.080	1.360
5	25	1.114	1.540	1.091	1.417
6	30.56	1.111	1.524	1.093	1.427
7	36.11	1.106	1.496	1.092	1.422
8	41.69	1.101	1.469	1.090	1.412
9	47.22	1.097	1.448	1.089	1.406
10	52.78	1.093	1.427	1.088	1.401
11	58.33	1.089	1.406	1.088	1.401
12	63.89	1.086	1.391	1.086	1.391
13	69.44	1.081	1.366	1.084	1.381
14	75	1.073	1.326	1.077	1.345
15	80.56	1.051	1.220	1.057	1.248
16	86.11	0.993	0.972	0.996	0.984
17	91.67	0.832	0.479	0.823	0.459
18	97.22	0.512	0.069	0.525	0.076
Average		1.000	1.152	1.000	1.127

Table P.5-14
Shielding Material Densities

Assembly Region Material Densities

Element	Atomic Number	Number Density (atom/b-cm)			
		Bottom End Fitting	Fuel	Plenum	Top End Fitting
O	8	-	1.35E-02	-	-
Al	13	1.31E-05	3.61E-06	6.39E-05	2.98E-05
Ti	22	9.88E-06	2.72E-06	4.80E-05	2.24E-05
Cr	24	1.88E-03	6.62E-05	1.06E-03	2.99E-03
Mn	25	1.65E-04	-	-	2.49E-04
Fe	26	5.96E-03	8.45E-05	1.29E-03	9.17E-03
Ni	28	1.21E-03	1.44E-04	2.54E-03	2.22E-03
Zr	40	6.23E-03	3.79E-03	3.89E-03	-
Mo	42	1.85E-05	5.08E-06	8.99E-05	4.19E-05
Sn	50	7.81E-05	4.75E-05	4.88E-05	-
U-235	92	-	3.39E-04	-	-
U-238	92	-	6.37E-03	-	-
Total		1.56E-02	2.43E-02	9.03E-03	1.47E-02

Other Shielding Materials

Element	Atomic Number	Number Density (atom/b-cm)							
		NS-3	Concrete	Water	Air	Lead	Carbon Steel	Stainless Steel	Aluminum/ BORAL
H	1	4.498E-02	7.767E-03	6.393E-02					
B-10	5	3.054E-04							
C	6	9.595E-03							
N	7				3.587E-05				
O	8	3.704E-02	4.317E-02	3.203E-02	9.534E-06				
Na	11		1.022E-03						
Al	13	6.887E-03	2.343E-03						6.071E-02
Si	14	1.243E-03	1.559E-02						
K	19		6.776E-04						
Ca	20	1.454E-03	2.855E-03						
Cr	24							1.743E-02	
Fe	26	1.042E-04	3.019E-04				8.465E-02	6.128E-02	
Ni	28							7.511E-03	
Pb	82					3.296E-02			
Total		1.016E-01	7.373E-02	9.596E-02	4.540E-05	3.296E-02	8.465E-02	8.622E-02	6.071E-02

Table P.5-15
Material Densities for Fuel/Basket Region Used in ANISN Models

Element	Atomic Number	Number Density (atom/b-cm)
O	8	1.012E-02
Al	13	1.017E-02
Ti	22	2.039E-06
Cr	24	1.779E-03
Mn	25	1.723E-04
Fe	26	6.004E-03
Ni	28	8.256E-04
Zr	40	2.846E-03
Mo	42	3.817E-06
Sn	50	3.567E-05
U-235	92	2.548E-04
U-238	92	4.786E-03
Total		3.700E-02

Table P.5-16
Neutron Source for ANISN Calculation

CASK-81 Energy Group	E_{upper} (MeV)	E_{mean} (MeV)	Normalized Cm-244 Fission Source	ANISN Neutron Source (n/s-cm^3)
1	1.49E+01	1.36E+01	2.018E-04	5.103E-10
2	1.22E+01	1.11E+01	1.146E-03	4.339E-09
3	1.00E+01	9.09E+00	4.471E-03	1.193E-08
4	8.18E+00	7.27E+00	1.768E-02	5.949E-08
5	6.36E+00	5.66E+00	4.167E-02	1.507E-07
6	4.96E+00	4.51E+00	5.641E-02	1.992E-07
7	4.06E+00	3.54E+00	1.197E-01	5.001E-07
8	3.01E+00	2.74E+00	9.616E-02	4.095E-07
9	2.46E+00	2.41E+00	2.256E-02	1.001E-07
10	2.35E+00	2.09E+00	1.227E-01	5.168E-07
11	1.83E+00	1.47E+00	2.110E-01	9.210E-07
12	1.11E+00	8.30E-01	1.794E-01	8.165E-07
13	5.50E-01	3.31E-01	1.138E-01	3.762E-07
14	1.11E-01	5.72E-02	1.301E-02	1.621E-11
15	3.35E-03	1.97E-03	6.555E-05	0.000E+00
16	5.83E-04	3.42E-04	4.765E-06	0.000E+00
17	1.01E-04	6.50E-05	3.134E-07	0.000E+00
18	2.90E-05	1.96E-05	4.527E-08	0.000E+00
19	1.01E-05	6.58E-06	9.759E-09	0.000E+00
20	3.06E-06	2.09E-06	1.521E-09	0.000E+00
21	1.12E-06	7.67E-07	3.353E-10	0.000E+00
22	4.14E-07	2.12E-07	9.683E-11	0.000E+00

Table P.5-17
ANISN Response Function for the OS197FC TC

Response Function Parameter for CASK81 Energy Groups Neutrons/ Gammas	Lower Boundary of Energy Group, (MeV)	Upper Boundary of Energy Group, (MeV)	Middle of OS197FC TC Side Surface		
			Neutron (mrem/hr)	Gamma (mrem/hr)	Total (mrem/hr)
Neutrons	0.00E+00	2.0E+01	2.37E-07	6.51E-08	3.02E-07
Gamma Group 40	0.00E+00	5.00E-02	0.00E+00	0.00E+00	0.00E+00
Gamma Group 39	5.00E-02	1.00E-01	0.00E+00	8.27E-41	8.27E-41
Gamma Group 38	1.00E-01	2.00E-01	0.00E+00	8.85E-30	8.85E-30
Gamma Group 37	2.00E-01	3.00E-01	0.00E+00	9.11E-22	9.11E-22
Gamma Group 36	3.00E-01	4.00E-01	0.00E+00	3.14E-18	3.14E-18
Gamma Group 35	4.00E-01	6.00E-01	0.00E+00	1.39E-17	1.39E-17
Gamma Group 34	6.00E-01	8.00E-01	0.00E+00	1.19E-15	1.19E-15
Gamma Group 33	8.00E-01	1.00E+00	0.00E+00	4.54E-14	4.54E-14
Gamma Group 32	1.00E+00	1.33E+00	0.00E+00	6.78E-13	6.78E-13
Gamma Group 31	1.33E+00	1.66E+00	0.00E+00	3.69E-12	3.69E-12
Gamma Group 30	1.66E+00	2.00E+00	0.00E+00	9.89E-12	9.89E-12
Gamma Group 29	2.00E+00	2.50E+00	0.00E+00	2.12E-11	2.12E-11
Gamma Group 28	2.50E+00	3.00E+00	0.00E+00	3.50E-11	3.50E-11
Gamma Group 27	3.00E+00	4.00E+00	0.00E+00	5.05E-11	5.05E-11
Gamma Group 26	4.00E+00	5.00E+00	0.00E+00	6.07E-11	6.07E-11
Gamma Group 25	5.00E+00	6.50E+00	0.00E+00	6.43E-11	6.43E-11
Gamma Group 24	6.50E+00	8.00E+00	0.00E+00	6.06E-11	6.06E-11
Gamma Group 23	8.00E+00	1.00E+01	0.00E+00	4.91E-11	4.91E-11

Table P.5-18
ANISN Response Function for the HSM-H

Response Function Parameter for CASK81 Energy Groups Neutrons/ Gammas	Lower Boundary of Energy Group, (MeV)	Upper Boundary of Energy Group, (MeV)	Middle of HSM-H Roof Centerline		
			Neutron (mrem/hr)	Gamma (mrem/hr)	Total (mrem/hr)
Neutrons	0.00E+00	2.0E+01	5.38E-11	1.14E-10	1.68E-10
Gamma Group 40	0.00E+00	5.00E-02	0.00E+00	0.00E+00	0.00E+00
Gamma Group 39	5.00E-02	1.00E-01	0.00E+00	1.32E-34	1.32E-34
Gamma Group 38	1.00E-01	2.00E-01	0.00E+00	2.67E-23	2.67E-23
Gamma Group 37	2.00E-01	3.00E-01	0.00E+00	3.16E-21	3.16E-21
Gamma Group 36	3.00E-01	4.00E-01	0.00E+00	6.69E-20	6.69E-20
Gamma Group 35	4.00E-01	6.00E-01	0.00E+00	3.69E-18	3.69E-18
Gamma Group 34	6.00E-01	8.00E-01	0.00E+00	5.15E-17	5.15E-17
Gamma Group 33	8.00E-01	1.00E+00	0.00E+00	3.67E-16	3.67E-16
Gamma Group 32	1.00E+00	1.33E+00	0.00E+00	3.24E-15	3.24E-15
Gamma Group 31	1.33E+00	1.66E+00	0.00E+00	1.85E-14	1.85E-14
Gamma Group 30	1.66E+00	2.00E+00	0.00E+00	6.86E-14	6.86E-14
Gamma Group 29	2.00E+00	2.50E+00	0.00E+00	2.44E-13	2.44E-13
Gamma Group 28	2.50E+00	3.00E+00	0.00E+00	6.95E-13	6.95E-13
Gamma Group 27	3.00E+00	4.00E+00	0.00E+00	2.18E-12	2.18E-12
Gamma Group 26	4.00E+00	5.00E+00	0.00E+00	5.49E-12	5.49E-12
Gamma Group 25	5.00E+00	6.50E+00	0.00E+00	1.18E-11	1.18E-11
Gamma Group 24	6.50E+00	8.00E+00	0.00E+00	2.10E-11	2.10E-11
Gamma Group 23	8.00E+00	1.00E+01	0.00E+00	3.06E-11	3.06E-11

Table P.5-19
Flux to Dose Rate Conversion Factors

Neutron		Gamma	
E (MeV)	(mrem/hr)/(n/cm^2/s)	E (MeV)	(mrem/hr)/(γ/cm^2/s)
2.50E-08	3.67E-03	0.01	3.96E-03
1.00E-07	3.67E-03	0.03	5.82E-04
1.00E-06	4.46E-03	0.05	2.90E-04
1.00E-05	4.54E-03	0.07	2.58E-04
1.00E-04	4.18E-03	0.1	2.83E-04
0.001	3.76E-03	0.15	3.79E-04
0.01	3.56E-03	0.2	5.01E-04
0.1	2.17E-02	0.25	6.31E-04
0.5	9.26E-02	0.3	7.59E-04
1	1.32E-01	0.35	8.78E-04
2.5	1.25E-01	0.4	9.85E-04
5	1.56E-01	0.45	1.08E-03
7	1.47E-01	0.5	1.17E-03
10	1.47E-01	0.55	1.27E-03
14	2.08E-01	0.6	1.36E-03
20	2.27E-01	0.65	1.44E-03
		0.7	1.52E-03
		0.8	1.68E-03
		1	1.98E-03
		1.4	2.51E-03
		1.8	2.99E-03
		2.2	3.42E-03
		2.6	3.82E-03
		2.8	4.01E-03
		3.25	4.41E-03
		3.75	4.83E-03
		4.25	5.23E-03
		4.75	5.60E-03
		5	5.80E-03
		5.25	6.01E-03
		5.75	6.37E-03
		6.25	6.74E-03
		6.75	7.11E-03
		7.5	7.66E-03
		9	8.77E-03
		11	1.03E-02
		13	1.18E-02
		15	1.33E-02

Table P.5-20
Gamma and Neutron Macroscopic Cross Sections

Group	SS304 (cm^{-1})	Lead (cm^{-1})
40	178.309	545.092
39	5.902	46.968
38	1.566	25.729
37	0.949	6.716
36	0.782	3.211
35	0.660	1.762
34	0.562	1.111
33	0.498	0.851
32	0.438	0.682
31	0.386	0.574
30	0.350	0.522
29	0.319	0.488
28	0.294	0.470
27	0.271	0.466
26	0.253	0.472
25	0.241	0.483
24	0.233	0.501
23	0.234	0.532
22	1.117	0.368
21	0.955	0.368
20	0.937	0.372
19	0.924	0.369
18	0.917	0.369
17	0.913	0.369
16	0.888	0.369
15	0.770	0.366
14	0.791	0.354
13	0.318	0.246
12	0.240	0.190
11	0.248	0.176
10	0.278	0.193
9	0.291	0.215
8	0.298	0.233
7	0.303	0.250
6	0.319	0.248
5	0.316	0.231
4	0.304	0.192
3	0.279	0.164
2	0.252	0.162
1	0.225	0.171

Table P.5-21
Surface Average Dose Rates on HSM-Model 102 with 24PTH-S-LC DSC

Surfaces	Dose Components	Dose Rate from Table N.5-4 of Appendix N (mrem/hr)	Scaling factors	Dose Rate (mrem/hr)
Back	Gamma	1.3	0.6	0.8
	Neutron	0.1	0.1	0.01
Front (excluding bird screen) (1)	Gamma	4.9	6.8	33.5
	Neutron	2.0	0.4	0.9
Roof (excluding bird screen) (1)	Gamma	25.4	1.1	27.1
	Neutron	0.5	0.2	0.1
Side	Gamma	29.6	1.1	31.7
	Neutron	0.5	0.2	0.1
Front Bird Screen	Gamma	261.3	1.1	279.6
	Neutron	6.2	0.2	1.1
Roof Bird Screen	Gamma	408.9	1.1	437.5
	Neutron	9.0	0.2	1.5

Note (1): If the front average dose rate includes the contribution from the front birdscreen, the dose rates are 45.6 mrem/hr for gammas and 0.9 mrem/hr for neutron radiation. Likewise, if the roof average dose rate includes the contribution from the roof birdscreen, the dose rate is 47.3 mrem/hr for gammas and 0.2 mrem/hr for neutron radiation.

Table P.5-22
Maximum Dose Rates on HSM-Model 102 with 24PTH-S-LC DSC

Surfaces	Dose Components	Dose Rate from Table N.5-4 of Appendix N (mrem/hr)	Scaling factors	Dose Rate (mrem/hr)
Back	Gamma	4	0.6	2.5
	Neutron	0.1	0.1	0.02
Front	Gamma	9	6.8	60.6
	Neutron	4	0.4	1.6
Roof	Gamma	55	1.1	59.3
	Neutron	1.0	0.2	0.2
Side	Gamma	250	1.1	266.9
	Neutron	2.0	0.2	0.3
Front Bird Screen	Gamma	458	1.1	489.6
	Neutron	14[1]	0.2	2.5
Roof Bird Screen	Gamma	913	1.1	976.5
	Neutron	18	0.2	3.2

Note (1): Not calculated in appendix N.5. Estimated here as approximately twice the average dose rate.

Table P.5-23 OS917FC TC Total Dose Rates (mrem/hr) at Cask Centerline for 2.0 kW Case(1)

Burn-Up, GWD/MTU	Maximum Assembly Average Initial U-235 Enrichment, wt %																										
	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	258.4	235.0	231.2	227.6	224.3	221.1	218.2	215.4	212.8	210.3	208.0	205.7	203.6	201.6	199.7	197.9	196.2	194.6	193.0	191.5	190.1	188.7	187.3	186.1	184.9	183.7	182.6	181.5	180.4	179.4	178.4	177.5
15	409.4	373.3	367.4	361.7	356.4	351.4	346.7	342.2	338.0	333.9	330.1	326.5	323.1	319.8	316.7	313.7	310.9	308.2	305.6	303.1	300.7	298.5	296.3	294.2	292.2	290.2	288.3	286.5	284.8	283.1	281.5	279.9
20	580.7	529.6	521.0	512.8	505.1	497.8	490.9	484.4	478.2	472.2	466.6	461.3	456.3	451.4	446.8	442.4	438.2	434.2	430.4	426.7	423.2	419.8	416.5	413.4	410.4	407.5	404.8	402.1	399.5	397.0	394.6	392.2
25			695.5	684.4	673.8	663.6	654.0	644.9	636.2	627.9	620.1	612.6	605.5	598.7	592.2	586.0	580.1	574.4	569.1	563.9	558.9	554.2	549.6	545.2	540.9	536.9	533.0	529.2	525.6	522.1	518.7	515.4
28				798.7	786.1	773.8	762.5	751.4	741.0	730.9	721.6	712.5	703.9	695.7	687.9	680.4	673.2	666.4	659.9	653.6	647.6	641.8	636.3	631.0	625.9	621.0	616.3	611.7	607.3	603.1	599.0	595.1
30							839.5	827.3	815.4	804.1	793.4	783.3	773.8	764.5	755.6	747.1	739.0	731.3	723.9	716.8	710.0	703.5	697.3	691.3	685.5	680.0	674.7	669.5	664.6	659.8	655.2	650.8
32								906.8	893.8	881.3	869.2	858.0	847.0	836.6	826.5	817.0	807.9	799.2	790.9	782.9	775.4	768.0	761.0	754.3	747.8	741.5	735.5	729.8	724.2	718.9	713.7	708.7
34									976.4	962.3	949.0	936.1	924.0	912.2	901.2	890.6	880.5	870.6	861.3	852.2	843.6	835.5	827.7	820.0	812.8	805.8	799.0	792.6	786.3	780.3	774.5	769.0
36										1047.6	1032.7	1018.3	1004.9	992.0	979.6	967.6	956.4	945.4	935.0	924.8	915.2	906.1	897.2	888.8	880.5	872.8	865.3	857.9	851.0	844.3	837.9	831.6
38												1105.1	1089.9	1075.7	1061.7	1048.5	1035.9	1023.6	1012.1	1000.9	990.1	980.0	970.0	960.6	951.6	942.9	934.3	926.3	918.7	911.2	904.0	896.9
39												1149.8	1134.1	1119.0	1104.4	1090.4	1077.0	1064.4	1052.1	1040.4	1029.1	1018.1	1008.0	998.0	988.2	979.0	970.2	961.6	953.6	945.6	937.9	930.6
40												1195.9	1179.1	1163.2	1148.1	1133.3	1119.3	1105.9	1093.1	1080.5	1068.7	1057.2	1046.5	1036.0	1025.7	1016.0	1006.7	997.6	989.0	980.6	972.7	964.9
41												1195.3	1178.3	1161.7	1192.6	1177.3	1162.5	1148.6	1134.9	1121.8	1109.5	1097.4	1085.9	1074.6	1064.1	1053.9	1044.1	1034.5	1025.4	1016.8	1008.1	1000.1
42												1196.2	1178.6	1161.4	1190.4	1174.4	1159.1	1144.6	1130.6	1117.1	1150.9	1138.3	1126.3	1114.5	1103.4	1092.6	1082.1	1072.3	1062.7	1053.3	1044.3	1035.9
43												1198.2	1180.0	1206.4	1189.4	1172.9	1157.2	1142.1	1127.5	1158.8	1145.4	1132.2	1119.8	1107.7	1096.1	1085.0	1121.2	1110.6	1100.5	1090.9	1081.5	1072.2
44												1202.0	1182.9	1207.2	1189.6	1172.8	1156.6	1140.9	1126.0	1155.0	1141.0	1127.5	1114.6	1102.1	1135.8	1123.9	1112.8	1101.8	1091.4	1081.5	1071.7	1062.4
45														1168.8	1191.4	1174.1	1157.3	1141.3	1167.8	1152.9	1138.3	1124.5	1110.9	1098.2	1085.6	1117.3	1105.8	1094.6	1083.9	1073.3	1063.5	1053.6
46														1213.3	1194.6	1176.4	1159.4	1142.6	1167.5	1151.8	1136.9	1122.5	1108.7	1137.7	1124.6	1112.1	1100.0	1088.5	1077.4	1066.8	1056.4	1090.1
47														1180.4	1199.1	1180.6	1162.7	1145.6	1168.2	1152.3	1136.8	1122.1	1107.7	1094.0	1121.5	1108.3	1096.0	1084.1	1072.4	1061.5	1092.8	1082.4
48														1188.0	1204.9	1185.9	1167.5	1149.8	1132.7	1154.3	1138.2	1123.0	1108.2	1133.2	1119.3	1105.9	1093.2	1080.8	1068.8	1057.6	1086.9	1076.2
49														1197.0	1212.1	1192.3	1173.7	1155.1	1174.3	1157.2	1140.8	1125.0	1109.7	1095.2	1118.8	1104.8	1091.6	1078.8	1066.5	1093.9	1082.5	1071.1
50																1200.4	1180.9	1162.2	1143.9	1161.8	1144.7	1128.6	1112.7	1134.1	1119.3	1105.3	1091.4	1078.4	1065.5	1091.2	1079.1	1067.4
51																1209.5	1189.5	1169.9	1151.4	1167.6	1149.9	1133.3	1116.8	1101.1	1121.3	1106.7	1092.6	1078.9	1065.8	1089.5	1077.0	1065.4
52																1187.6	1199.1	1179.3	1160.0	1174.5	1156.5	1139.0	1122.3	1106.1	1124.7	1109.6	1094.9	1081.0	1067.3	1089.3	1076.6	1064.4
53																1200.3	1210.3	1189.7	1169.8	1150.7	1164.2	1146.2	1129.0	1112.4	1129.1	1113.7	1098.4	1083.9	1069.9	1056.6	1077.2	1064.7
54																1213.6	1192.2	1201.3	1180.9	1161.0	1142.2	1154.7	1136.9	1119.6	1103.0	1118.9	1103.3	1088.3	1074.0	1059.8	1078.9	1065.9
55																1228.0	1205.0	1185.1	1193.0	1172.9	1153.2	1134.2	1145.9	1128.0	1111.1	1094.5	1109.1	1093.8	1078.9	1064.5	1050.8	1037.3
56																1192.1	1169.4	1146.9	1124.5	1105.6	1086.1	1067.5	1049.2	1031.7	1014.9	998.9	983.3	968.2	1085.1	1070.2	1055.9	1072.3
57																1235.5	1212.8	1190.2	1167.7	1145.4	1126.2	1106.2	1087.3	1069.1	1051.6	1034.6	1018.3	1002.7	987.6	972.9	959.0	945.5
58																1254.4	1228.7	1206.3	1208.0	1188.6	1166.3	1147.1	1125.1	1107.4	1089.0	1071.5	1054.5	1038.0	1022.2	1007.1	992.2	978.2
59																1225.9	1225.3	1225.5	1200.2	1204.3	1182.1	1160.0	1165.1	1146.1	1127.6	1109.2	1091.5	1074.2	1057.7	1042.0	1026.6	1011.7
60																1182.2	1178.9	1154.2	1198.0	1173.1	1198.2	1176.2	1157.3	1135.5	1143.0	1121.3	1104.3	1111.5	1094.3	1077.8	1061.7	1046.2
61																1269.6	1244.6	1219.7	1194.9	1173.2	1148.6	1127.1	1105.7	1087.5	1068.3	1048.2	1030.3	1013.1	996.2	980.2	964.5	949.2
62																1270.8	1268.2	1240.4	1238.5	1213.7	1192.0	1167.4	1146.0	1124.7	1106.5	1085.3	1067.3	1049.4	1031.6	1014.6	998.2	982.3

Note(1): Maximum value for burnup of 62 GWd/MTU, 3.4 wt.% U-235, used only to determine the design basis source term.

Table P.5-24 HSM-H Total Dose Rates (mrem/hr) at Roof for 2.0 kW Case(1)

Burn-Up, GWD/MT	Maximum Assembly Average Initial U-235 Enrichment, wt %																														
	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	1.8	1.7	1.7	1.7	1.7	1.7	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.4	1.4	1.4	1.4	1.4
15	2.9	2.7	2.7	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4	2.4	2.4	2.4	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.2	2.2	2.2	2.2	2.2
20	4.0	3.7	3.7	3.7	3.6	3.6	3.6	3.5	3.5	3.5	3.4	3.4	3.4	3.4	3.3	3.3	3.3	3.3	3.3	3.2	3.2	3.2	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.1	3.1
25			4.8	4.7	4.7	4.7	4.6	4.6	4.5	4.5	4.5	4.4	4.4	4.4	4.3	4.3	4.3	4.2	4.2	4.2	4.2	4.1	4.1	4.1	4.1	4.1	4.0	4.0	4.0	4.0	4.0	4.0
28				5.4	5.4	5.3	5.3	5.2	5.2	5.1	5.1	5.1	5.0	5.0	5.0	4.9	4.9	4.9	4.8	4.8	4.8	4.7	4.7	4.7	4.7	4.6	4.6	4.6	4.6	4.6	4.5	4.5
30							5.7	5.7	5.6	5.6	5.5	5.5	5.5	5.4	5.4	5.3	5.3	5.3	5.2	5.2	5.2	5.1	5.1	5.1	5.1	5.0	5.0	5.0	5.0	4.9	4.9	4.9
32								6.1	6.1	6.0	6.0	5.9	5.9	5.9	5.8	5.8	5.7	5.7	5.7	5.6	5.6	5.6	5.5	5.5	5.5	5.4	5.4	5.4	5.4	5.3	5.3	5.3
34									6.6	6.5	6.5	6.4	6.4	6.3	6.3	6.2	6.2	6.1	6.1	6.1	6.0	6.0	6.0	5.9	5.9	5.9	5.8	5.8	5.8	5.7	5.7	5.7
36										7.0	6.9	6.9	6.8	6.8	6.7	6.7	6.6	6.6	6.5	6.5	6.5	6.4	6.4	6.3	6.3	6.3	6.2	6.2	6.2	6.2	6.1	6.1
38												7.3	7.3	7.2	7.2	7.1	7.1	7.0	7.0	6.9	6.9	6.9	6.8	6.8	6.7	6.7	6.7	6.6	6.6	6.6	6.5	6.5
39												7.6	7.5	7.5	7.4	7.4	7.3	7.3	7.2	7.2	7.1	7.1	7.0	7.0	7.0	6.9	6.9	6.9	6.8	6.8	6.8	6.7
40												7.8	7.8	7.7	7.7	7.6	7.5	7.5	7.4	7.4	7.4	7.3	7.3	7.2	7.2	7.1	7.1	7.1	7.0	7.0	7.0	6.9
41												7.6	7.6	7.5	7.9	7.8	7.8	7.7	7.7	7.6	7.6	7.5	7.5	7.5	7.4	7.4	7.3	7.3	7.3	7.2	7.2	7.2
42												7.5	7.4	7.4	7.7	7.7	7.6	7.5	7.5	7.4	7.8	7.8	7.7	7.7	7.6	7.6	7.6	7.5	7.5	7.4	7.4	7.4
43												7.3	7.3	7.6	7.5	7.5	7.4	7.4	7.3	7.7	7.6	7.6	7.5	7.5	7.4	7.4	7.8	7.7	7.7	7.7	7.6	7.6
44												7.2	7.1	7.4	7.4	7.3	7.3	7.2	7.2	7.5	7.4	7.4	7.3	7.3	7.7	7.6	7.6	7.5	7.5	7.5	7.4	7.4
45														6.9	7.2	7.2	7.1	7.0	7.4	7.3	7.3	7.2	7.2	7.1	7.1	7.4	7.4	7.3	7.3	7.3	7.2	7.2
46														7.1	7.1	7.0	7.0	6.9	7.2	7.1	7.1	7.0	7.0	7.3	7.3	7.2	7.2	7.2	7.1	7.1	7.0	7.4
47														6.7	6.9	6.9	6.8	6.8	7.0	7.0	6.9	6.9	6.8	6.8	7.1	7.1	7.0	7.0	6.9	6.9	7.2	7.2
48														6.5	6.8	6.7	6.7	6.6	6.6	6.8	6.8	6.7	6.7	7.0	6.9	6.9	6.9	6.8	6.8	6.7	7.0	7.0
49														6.4	6.7	6.6	6.6	6.5	6.8	6.7	6.7	6.6	6.6	6.5	6.8	6.7	6.7	6.7	6.6	6.9	6.9	6.8
50																6.5	6.4	6.4	6.3	6.6	6.5	6.5	6.4	6.7	6.6	6.6	6.6	6.5	6.5	6.8	6.7	6.7
51																6.4	6.3	6.3	6.2	6.5	6.4	6.4	6.3	6.3	6.5	6.5	6.4	6.4	6.3	6.6	6.6	6.5
52																6.0	6.2	6.2	6.1	6.3	6.3	6.2	6.2	6.1	6.4	6.3	6.3	6.3	6.2	6.5	6.4	6.4
53																5.9	6.1	6.1	6.0	6.0	6.2	6.1	6.1	6.0	6.3	6.2	6.2	6.1	6.1	6.0	6.3	6.2
54																5.9	5.8	6.0	5.9	5.9	5.8	6.0	6.0	5.9	5.9	6.1	6.1	6.0	6.0	5.9	6.2	6.1
55																5.8	5.7	5.7	5.9	5.8	5.8	5.7	5.9	5.8	5.8	5.8	6.0	5.9	5.9	5.8	5.8	5.8
56																5.3	5.3	5.2	5.2	5.1	5.1	5.0	5.0	5.0	4.9	4.9	4.8	4.8	5.8	5.7	5.7	5.9
57																5.4	5.4	5.4	5.3	5.3	5.2	5.2	5.1	5.1	5.0	5.0	5.0	4.9	4.9	4.8	4.8	4.8
58																5.4	5.3	5.3	5.4	5.4	5.3	5.3	5.2	5.2	5.2	5.1	5.1	5.0	5.0	5.0	4.9	4.9
59																5.0	5.1	5.2	5.2	5.3	5.3	5.2	5.4	5.3	5.3	5.2	5.2	5.2	5.1	5.1	5.0	5.0
60																4.6	4.6	4.6	5.0	4.9	5.2	5.2	5.1	5.1	5.2	5.2	5.1	5.3	5.2	5.2	5.2	5.1
61																4.9	4.9	4.8	4.8	4.8	4.7	4.7	4.6	4.6	4.5	4.5	4.5	4.4	4.4	4.4	4.3	4.3
62																4.8	4.9	4.8	4.9	4.9	4.8	4.8	4.7	4.7	4.7	4.6	4.6	4.5	4.5	4.5	4.4	4.4

Note(1): Maximum value for burnup of 41 GWd/MTU, 3.3 wt.% U-235, used only to determine the design basis source term.

Table P.5-25 OS197FC TC Total Dose Rates (mrem/hr) at Cask Centerline for 1.5 kW Case(1)

Burn-Up, GWD/MTU	Maximum Assembly Average Initial U-235 Enrichment, wt %																														
	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	258.4	235.0	231.2	227.6	224.3	221.1	218.2	215.4	212.8	210.3	208.0	205.7	203.6	201.6	199.7	197.9	196.2	194.6	193.0	191.5	190.1	188.7	187.3	186.1	184.9	183.7	182.6	181.5	180.4	179.4	178.4	177.5
15	409.4	373.3	367.4	361.7	356.4	351.4	346.7	342.2	338.0	333.9	330.1	326.5	323.1	319.8	316.7	313.7	310.9	308.2	305.6	303.1	300.7	298.5	296.3	294.2	292.2	290.2	288.3	286.5	284.8	283.1	281.5	279.9
20	580.7	529.6	521.0	512.8	505.1	497.8	490.9	484.4	478.2	472.2	466.6	461.3	456.3	451.4	446.8	442.4	438.2	434.2	430.4	426.7	423.2	419.8	416.5	413.4	410.4	407.5	404.8	402.1	399.5	397.0	394.6	392.2
25			695.5	684.4	673.8	663.6	654.0	644.9	636.2	627.9	620.1	612.6	605.5	598.7	592.2	586.0	580.1	574.4	569.1	563.9	558.9	554.2	549.6	545.2	540.9	536.9	533.0	529.2	525.6	522.1	518.7	515.4
28				798.7	786.1	773.8	762.5	751.4	741.0	730.9	721.6	712.5	703.9	695.7	687.9	680.4	673.2	666.4	659.9	653.6	647.6	641.8	636.3	631.0	625.9	621.0	616.3	611.7	607.3	603.1	599.0	595.1
30							839.5	827.3	815.4	804.1	793.4	783.3	773.8	764.5	755.6	747.1	739.0	731.3	723.9	716.8	710.0	703.5	697.3	691.3	685.5	680.0	674.7	669.5	664.6	659.8	655.2	650.8
32								906.8	893.8	881.3	869.2	858.0	847.0	836.6	826.5	817.0	807.9	799.2	790.9	782.9	775.4	768.0	761.0	754.3	747.8	741.5	735.5	729.8	724.2	718.9	713.7	708.7
34									901.6	888.2	875.4	862.8	886.3	875.0	864.0	853.5	843.5	833.8	824.7	816.0	843.6	835.5	827.7	820.0	812.8	805.8	799.0	792.6	786.3	780.3	774.5	769.0
36										898.3	884.5	871.3	858.6	846.5	867.3	856.1	845.4	835.0	825.1	815.7	840.4	831.6	823.0	814.9	806.9	799.5	792.2	785.1	813.6	807.0	800.5	794.4
38												882.9	869.3	856.2	843.7	831.6	850.4	839.3	828.7	818.3	808.8	799.4	821.6	812.7	804.4	796.2	788.5	781.0	773.6	799.3	792.4	785.7
39												861.6	875.8	862.4	849.2	836.9	825.1	842.7	831.7	821.0	811.1	801.0	822.1	813.0	804.3	795.7	787.7	779.8	772.4	796.6	789.3	782.5
40												870.2	855.9	869.4	855.9	842.9	830.5	818.9	835.6	824.5	814.0	803.9	794.3	784.9	804.8	796.2	787.8	779.6	771.8	764.5	757.4	780.4
41												879.9	865.0	850.6	863.3	849.8	837.0	824.8	813.0	828.9	818.0	807.3	797.4	787.7	778.5	797.4	788.6	780.4	772.3	764.6	757.1	778.8
42												865.7	874.9	860.2	845.7	832.3	844.3	831.8	819.4	807.9	822.6	811.9	801.4	791.3	781.8	799.4	790.4	781.8	773.3	765.3	757.5	750.1
43												877.8	861.9	870.4	855.7	841.4	827.9	839.2	826.6	814.4	803.0	817.1	806.2	795.8	785.6	776.2	766.9	758.1	775.3	767.0	758.8	751.1
44												890.5	874.1	858.4	843.2	851.5	837.4	824.1	811.0	822.1	810.1	798.6	787.2	800.9	790.7	780.6	771.0	761.9	777.9	769.3	761.0	753.0
45														871.0	855.2	840.2	825.7	834.1	820.6	807.7	817.9	806.1	794.4	783.6	772.9	785.8	775.9	766.5	757.2	748.3	739.9	755.5
46														863.2	847.3	852.6	837.5	823.2	830.7	817.5	804.5	792.2	802.4	790.9	779.9	769.3	781.7	771.6	762.3	753.1	744.1	758.9
47														877.8	861.4	845.4	829.9	835.4	820.9	807.3	814.7	802.1	789.7	799.0	787.7	776.6	766.2	755.9	746.1	758.4	749.1	740.5
48														874.0	857.2	859.6	843.8	828.5	833.2	819.3	805.5	792.5	799.7	787.5	775.7	784.8	773.8	763.2	753.1	743.4	754.9	745.7
49														872.5	855.2	856.2	840.0	842.7	827.6	812.9	817.6	804.0	791.1	778.4	785.5	773.9	762.7	751.9	760.9	750.9	741.1	731.6
50																854.8	838.2	839.5	823.9	826.6	812.1	798.3	803.0	790.0	777.4	765.6	753.9	761.0	750.4	739.9	748.7	739.0
51																855.7	854.9	838.5	822.4	823.8	808.9	811.5	797.9	784.5	789.1	776.7	764.7	753.1	742.3	749.0	738.9	729.0
52																858.3	856.2	839.3	838.6	822.8	807.7	809.1	794.7	797.7	784.4	771.5	759.4	764.3	752.9	741.9	731.0	720.9
53																862.6	859.2	841.9	840.2	823.9	808.3	793.1	793.8	779.7	781.8	768.8	756.1	759.5	747.9	736.4	741.7	730.9
54																868.4	850.0	846.1	829.0	826.6	810.6	809.6	794.6	780.1	780.8	767.3	754.4	756.7	744.6	748.3	737.0	726.0
55																874.1	856.8	852.0	834.2	817.3	814.5	798.7	796.8	796.0	781.5	767.8	768.7	755.7	743.4	745.7	734.2	723.1
56																870.1	863.9	846.5	841.0	823.4	819.4	803.5	800.8	785.5	784.1	769.9	756.1	756.4	743.6	744.9	732.9	721.5
57																871.1	861.2	854.7	837.6	831.0	813.7	809.3	793.3	790.1	775.1	773.2	759.2	758.2	745.3	732.7	733.4	721.6
58																880.4	860.4	843.5	833.3	826.5	822.9	804.9	800.1	796.2	780.7	765.9	763.6	749.5	748.4	735.3	734.9	722.8
59																871.7	864.0	853.4	846.1	826.3	819.2	802.6	786.1	791.8	776.2	772.2	768.9	754.6	741.0	739.2	738.0	714.2
60																877.0	865.9	854.9	847.1	827.6	820.0	812.6	796.2	789.1	773.7	768.8	753.8	749.9	746.8	733.1	730.9	718.2
61																880.3	868.8	849.5	841.2	830.1	822.1	802.9	795.1	790.5	783.1	767.3	752.2	737.4	743.2	739.5	725.9	713.1
62																877.2	865.3	853.6	845.1	826.0	814.5	806.3	798.1	790.1	774.2	766.4	761.5	746.2	741.3	727.3	723.1	710.1

Note(1): Maximum value for burnup of 32 GWd/MTU, 2.6 wt.% U-235, used only to determine the design basis source term.

Table P.5-26 HSM-H Total Dose Rates (mrem/hr) at Roof for 1.5 kW Case(1)

Burn-Up, GWD/MT	Maximum Assembly Average Initial U-235 Enrichment, wt %																														
	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	1.8	1.7	1.7	1.7	1.7	1.7	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.4	1.4	1.4	1.4	1.4
15	2.9	2.7	2.7	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4	2.4	2.4	2.4	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.2	2.2	2.2	2.2	2.2
20	4.0	3.7	3.7	3.7	3.6	3.6	3.6	3.5	3.5	3.5	3.4	3.4	3.4	3.4	3.3	3.3	3.3	3.3	3.3	3.2	3.2	3.2	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.1	3.1
25			4.8	4.7	4.7	4.7	4.6	4.6	4.5	4.5	4.5	4.4	4.4	4.4	4.3	4.3	4.3	4.2	4.2	4.2	4.2	4.1	4.1	4.1	4.1	4.1	4.0	4.0	4.0	4.0	4.0	4.0
28				5.4	5.4	5.3	5.3	5.2	5.2	5.1	5.1	5.1	5.0	5.0	5.0	4.9	4.9	4.9	4.8	4.8	4.8	4.7	4.7	4.7	4.7	4.6	4.6	4.6	4.6	4.6	4.5	4.5
30							5.7	5.7	5.6	5.6	5.5	5.5	5.5	5.4	5.4	5.3	5.3	5.3	5.2	5.2	5.2	5.1	5.1	5.1	5.1	5.0	5.0	5.0	5.0	4.9	4.9	4.9
32								6.1	6.1	6.0	6.0	5.9	5.9	5.9	5.8	5.8	5.7	5.7	5.7	5.6	5.6	5.6	5.5	5.5	5.5	5.4	5.4	5.4	5.4	5.3	5.3	5.3
34									5.9	5.8	5.8	5.8	6.0	6.0	5.9	5.9	5.8	5.8	5.8	5.7	6.0	6.0	6.0	5.9	5.9	5.9	5.8	5.8	5.8	5.7	5.7	5.7
36										5.7	5.6	5.6	5.5	5.5	5.7	5.7	5.6	5.6	5.6	5.5	5.8	5.7	5.7	5.7	5.6	5.6	5.6	5.5	5.8	5.8	5.8	5.8
38												5.4	5.4	5.3	5.3	5.2	5.4	5.4	5.4	5.3	5.3	5.2	5.5	5.5	5.4	5.4	5.4	5.3	5.3	5.6	5.5	5.5
39												5.1	5.3	5.2	5.2	5.1	5.1	5.3	5.3	5.2	5.2	5.2	5.4	5.4	5.3	5.3	5.3	5.2	5.2	5.4	5.4	5.4
40												5.0	5.0	5.2	5.1	5.1	5.0	5.0	5.2	5.2	5.1	5.1	5.0	5.0	5.2	5.2	5.2	5.1	5.1	5.1	5.0	5.3
41												5.0	4.9	4.9	5.1	5.0	5.0	4.9	4.9	5.1	5.0	5.0	5.0	4.9	4.9	5.1	5.1	5.0	5.0	5.0	4.9	5.2
42												4.7	4.9	4.8	4.8	4.7	4.9	4.9	4.8	4.8	5.0	4.9	4.9	4.8	4.8	5.0	5.0	4.9	4.9	4.9	4.9	4.8
43												4.7	4.6	4.8	4.7	4.7	4.6	4.8	4.8	4.7	4.7	4.9	4.8	4.8	4.7	4.7	4.7	4.6	4.8	4.8	4.8	4.7
44												4.6	4.6	4.5	4.5	4.6	4.6	4.6	4.5	4.7	4.6	4.6	4.5	4.7	4.7	4.6	4.6	4.6	4.8	4.7	4.7	4.7
45														4.5	4.5	4.4	4.4	4.5	4.5	4.4	4.6	4.5	4.5	4.5	4.4	4.6	4.5	4.5	4.5	4.5	4.4	4.6
46														4.3	4.3	4.4	4.3	4.3	4.4	4.4	4.3	4.3	4.4	4.4	4.4	4.3	4.5	4.5	4.4	4.4	4.4	4.5
47														4.3	4.2	4.2	4.2	4.3	4.2	4.2	4.3	4.3	4.2	4.4	4.3	4.3	4.3	4.2	4.2	4.3	4.3	4.3
48														4.1	4.1	4.2	4.1	4.1	4.2	4.2	4.1	4.1	4.2	4.2	4.1	4.2	4.2	4.2	4.2	4.1	4.3	4.2
49														4.0	3.9	4.0	4.0	4.1	4.0	4.0	4.1	4.0	4.0	4.0	4.1	4.1	4.0	4.0	4.1	4.1	4.0	4.0
50																3.9	3.8	3.9	3.9	4.0	3.9	3.9	4.0	4.0	3.9	3.9	3.9	4.0	3.9	3.9	4.0	4.0
51																3.7	3.8	3.8	3.7	3.8	3.8	3.9	3.8	3.8	3.9	3.9	3.8	3.8	3.8	3.9	3.8	3.8
52																3.6	3.7	3.7	3.7	3.7	3.7	3.7	3.7	3.8	3.7	3.7	3.7	3.8	3.7	3.7	3.7	3.7
53																3.5	3.6	3.6	3.6	3.6	3.5	3.5	3.6	3.5	3.6	3.6	3.6	3.6	3.6	3.6	3.7	3.6
54																3.4	3.4	3.5	3.4	3.5	3.4	3.5	3.5	3.4	3.5	3.5	3.4	3.5	3.5	3.6	3.5	3.5
55																3.4	3.3	3.4	3.3	3.3	3.4	3.3	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4	3.4
56																3.2	3.3	3.2	3.3	3.2	3.3	3.2	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.3
57																3.1	3.1	3.2	3.1	3.2	3.1	3.2	3.1	3.2	3.2	3.2	3.2	3.2	3.2	3.2	3.2	3.2
58																3.0	3.0	3.0	3.0	3.0	3.1	3.0	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.1
59																2.9	2.9	2.9	3.0	2.9	2.9	2.9	2.9	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.1	3.0
60																2.8	2.8	2.8	2.9	2.8	2.8	2.9	2.8	2.9	2.8	2.9	2.8	2.9	2.9	2.9	2.9	2.9
61																2.7	2.7	2.7	2.7	2.7	2.8	2.7	2.7	2.8	2.8	2.8	2.7	2.7	2.8	2.8	2.8	2.8
62																2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7

Note(1): Maximum value for burnup of 32 GWd/MTU, 2.6 wt.% U-235, used only to determine the design basis source term.

Figure Withheld Under 10 CFR 2.390

Figure P.5-1
ANISN HSM-H Model

Figure Withheld Under 10 CFR 2.390

Figure P.5-2
ANISN OS197FC TC Model

Figure Withheld Under 10 CFR 2.390

Figure P.5-3
24PTH-L DSC Within HSM-H, Side View at Centerline of DSC

[xxx] = surface numbers, all dimensions without units are in cm

Figure Withheld Under 10 CFR 2.390

Figure P.5-4
24PTH-L DSC Within HSM-H, Head-on View at X=0

[xxx] = surface numbers, all dimensions without units are in cm

Figure Withheld Under 10 CFR 2.390

Figure P.5-5
24PTH-L DSC Within HSM-H, Head-on View Showing Top Vents

Figure Withheld Under 10 CFR 2.390

Figure P.5-6
24PTH-L DSC Within HSM-H, Head-on View at Lid End of DSC (X=225 cm)

Figure Withheld Under 10 CFR 2.390

Figure P.5-7
24PTH-L DSC Within HSM-H, Head-on View at Bottom End of DSC (X=-225 cm)

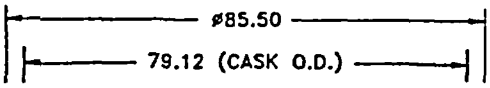


Figure Withheld Under 10 CFR 2.390

dimensions are in inches

Figure P.5-8
24PTH-L DSC Within OS197FC TC, Axial View of Transfer Model

Figure Withheld Under 10 CFR 2.390

Note: All dimensions are in inches.

Figure P.5-9
24PTH-L DSC Within OS197FC TC, Top View of Transfer Model Showing Cask Lid with Gap, Top Nozzle, and Plenum

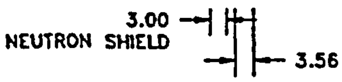
Figure Withheld Under 10 CFR 2.390

Figure P.5-10
24PTH-L DSC Within OS197FC TC, Bottom View of Transfer Model Showing Cask Bottom and Bottom Nozzle

Figure Withheld Under 10 CFR 2.390

Figure P.5-11
24PTH-L DSC Within OS197FC TC, Radial Cut View of Transfer Models Showing Fuel Locations

Figure Withheld Under 10 CFR 2.390



Note: All dimensions are in inches.

Figure P.5-12
24PTH-S-LC DSC Within Standardized TC, Axial View of Transfer Model

Figure Withheld Under 10 CFR 2.390

Figure P.5-13
24PTH-S-LC DSC Within Standardized TC, Top View of Transfer Model Showing Cask Lid with Gap, Top Nozzle, and Plenum

Figure Withheld Under 10 CFR 2.390

Note: All dimensions are in inches.

Figure P.5-14
24PTH-S-LC DSC Within Standardized TC, Bottom View of Transfer Model Showing Cask Bottom and Bottom Nozzle

Figure Withheld Under 10 CFR 2.390

Figure P.5-15
24PTH-S-LC DSC Within Standardized TC, Radial Cut Views of Transfer Model Showing Fuel Locations

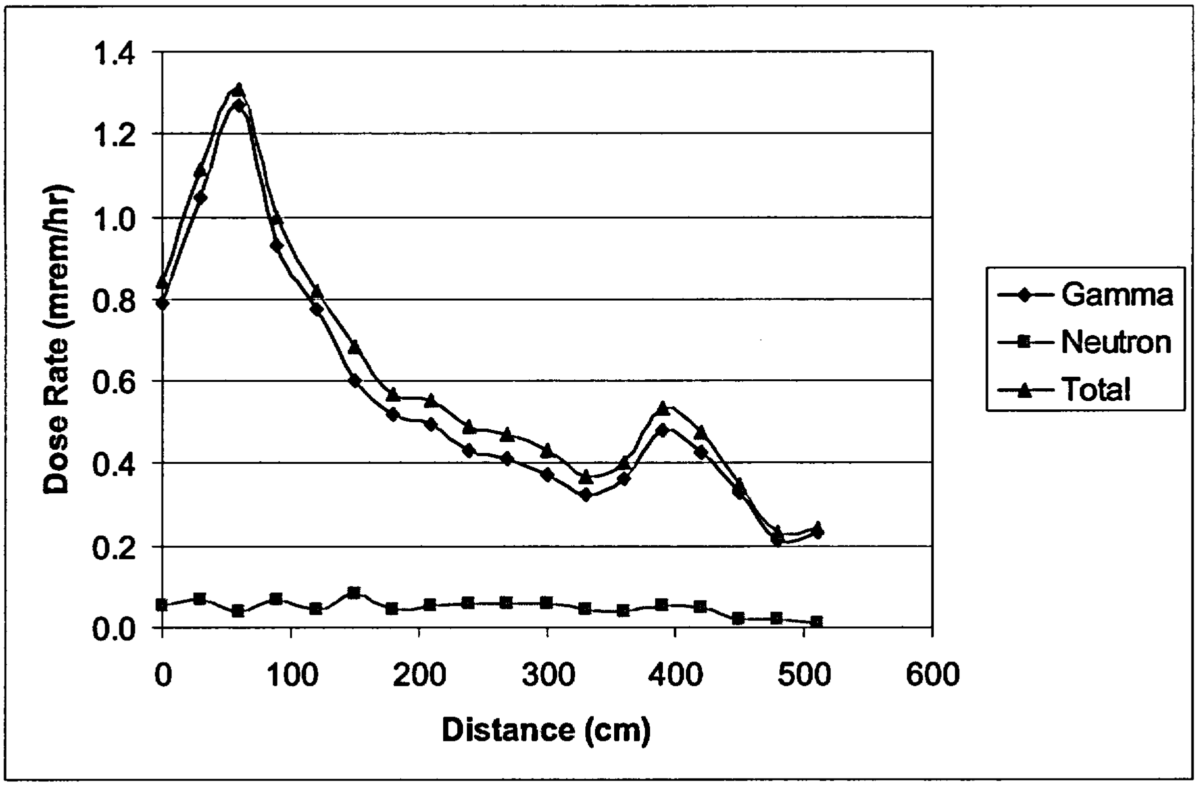


Figure P.5-16
HSM-H with 24PTH-L DSC, Front Door Centerline Dose Rate

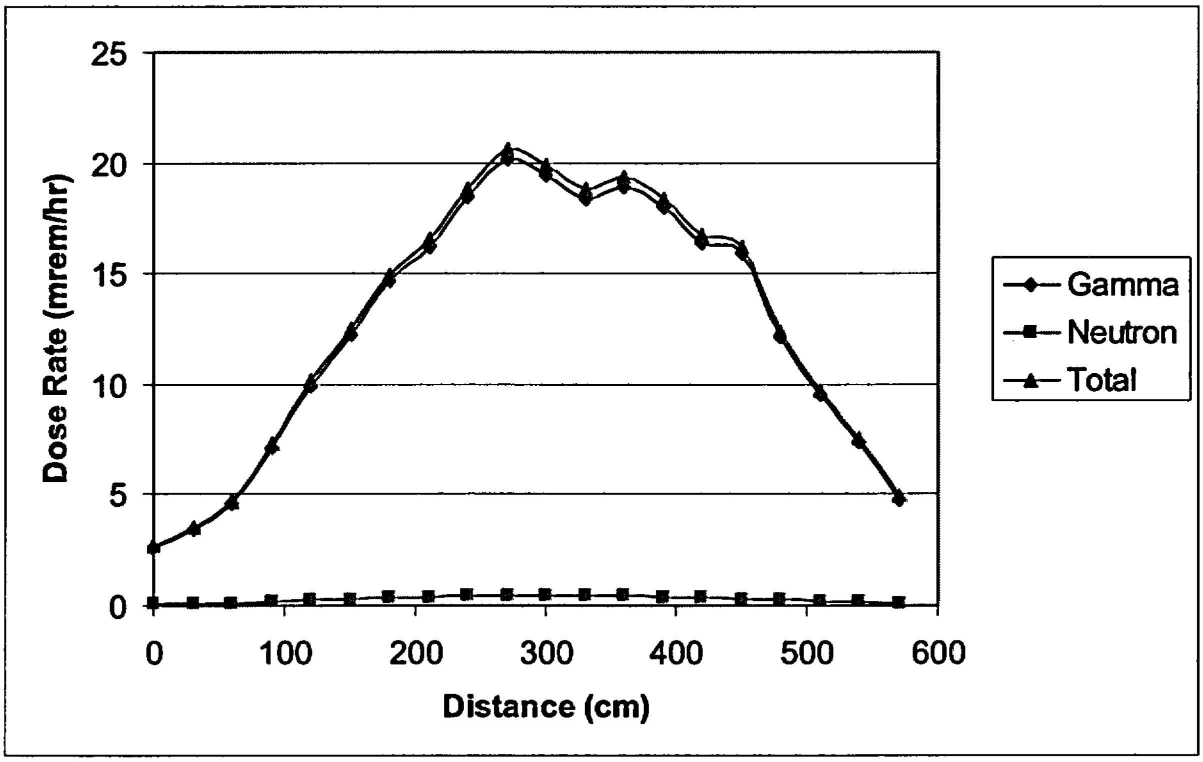


Figure P.5-17
HSM-H with 24PTH-L DSC, Roof Centerline Dose Rate

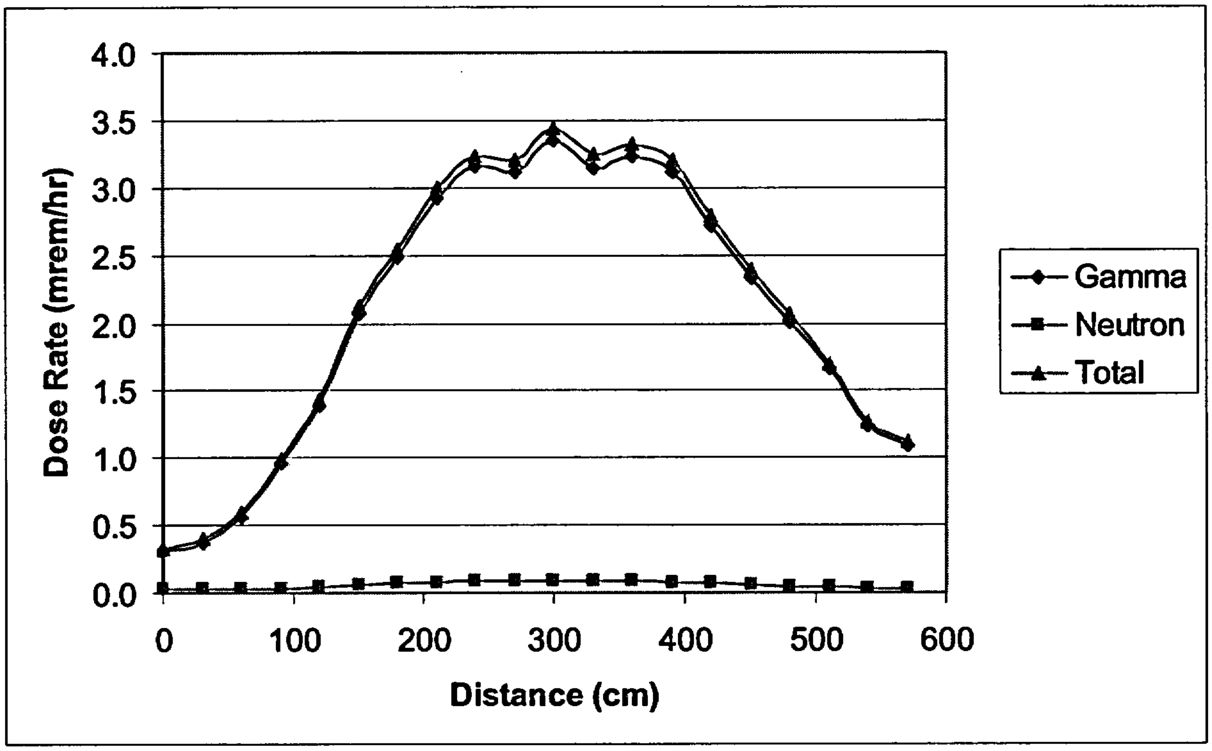


Figure P.5-18
HSM-H with 24PTH-L DSC, Side Shield Wall Surface at DSC Centerline Dose Rate

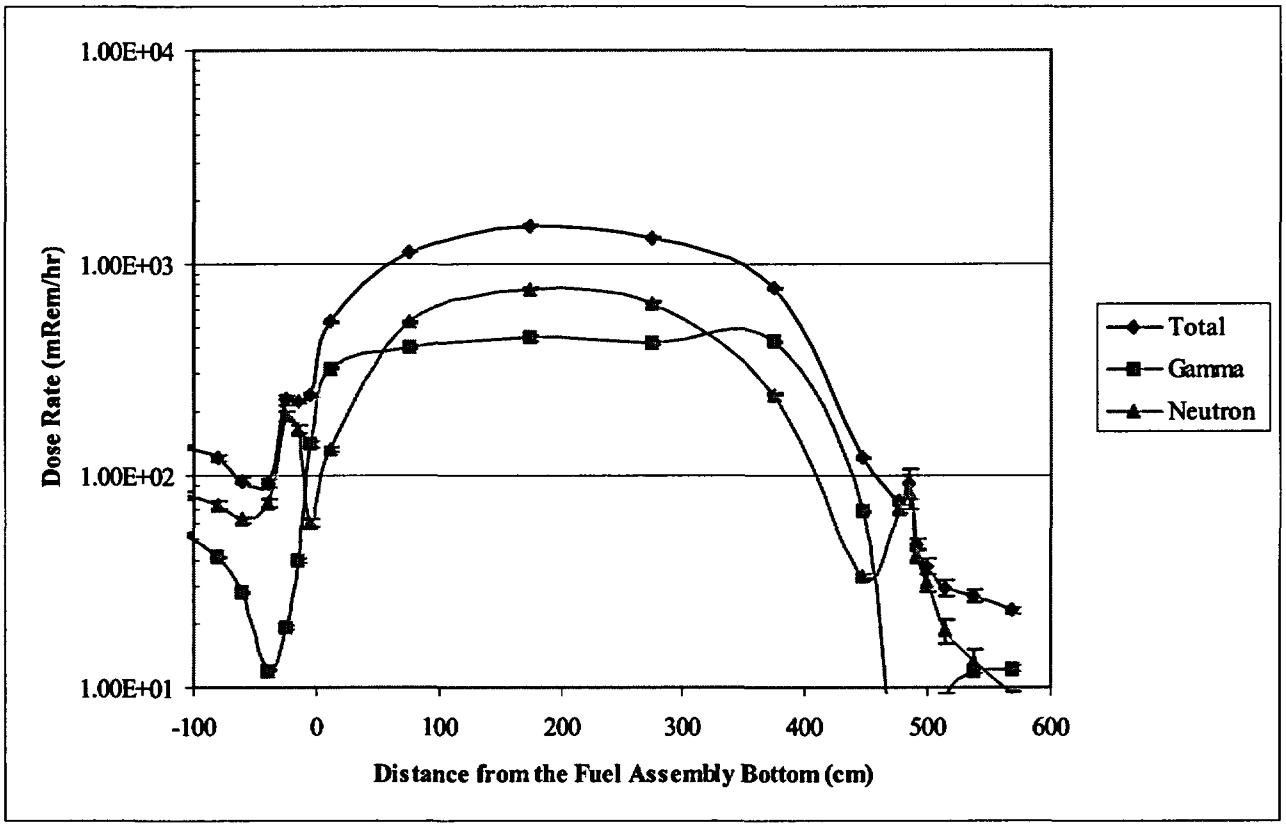


Figure P.5-19
OS197FC TC with 24PTH-L DSC, Side Surface Dose Rate

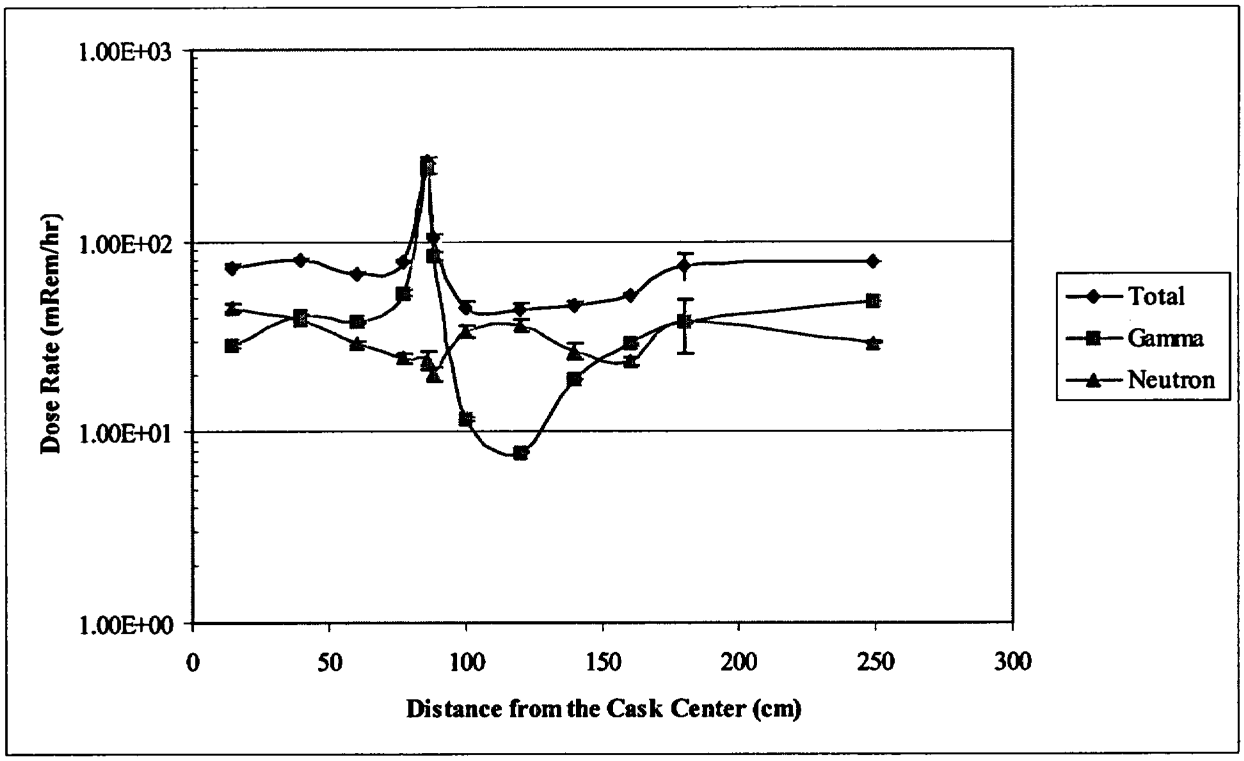


Figure P.5-20
OS197FC TC with 24PTH-L DSC, Top Surface Dose Rate

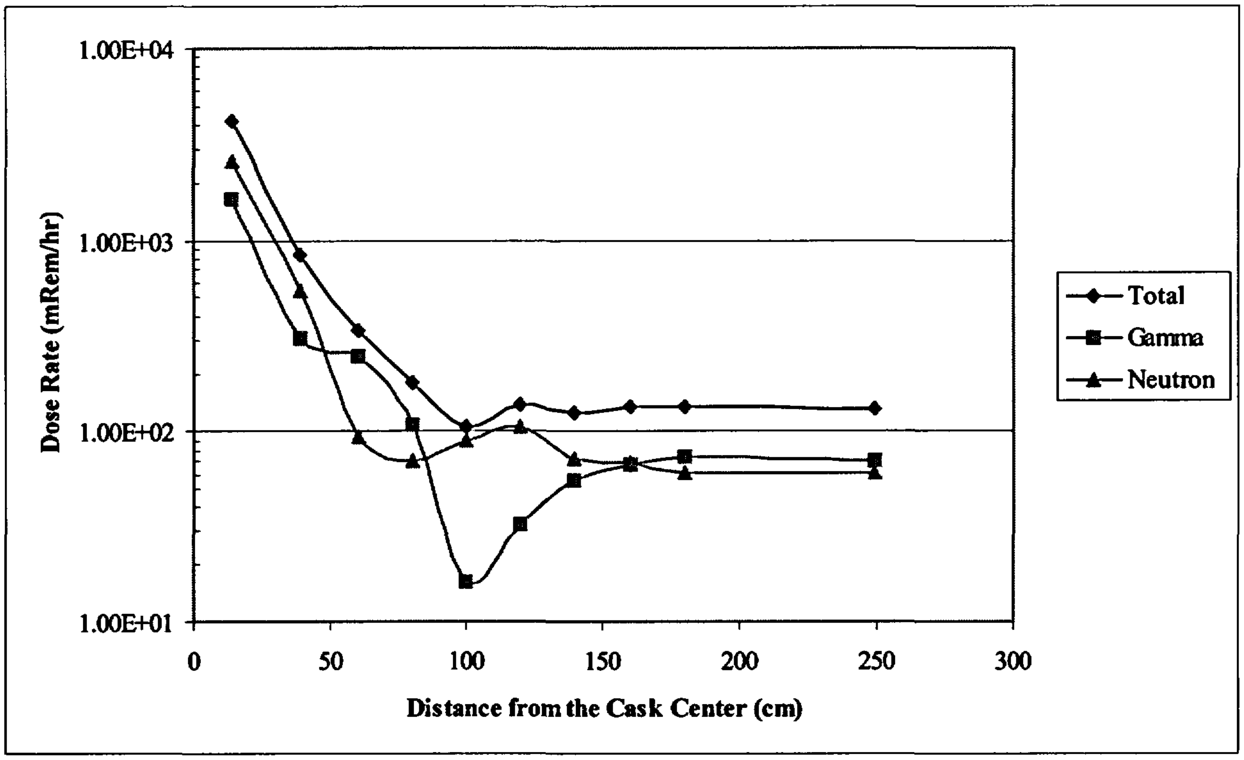


Figure P.5-21
OS197FC TC with 24PTH-L DSC, Bottom Surface Dose Rate

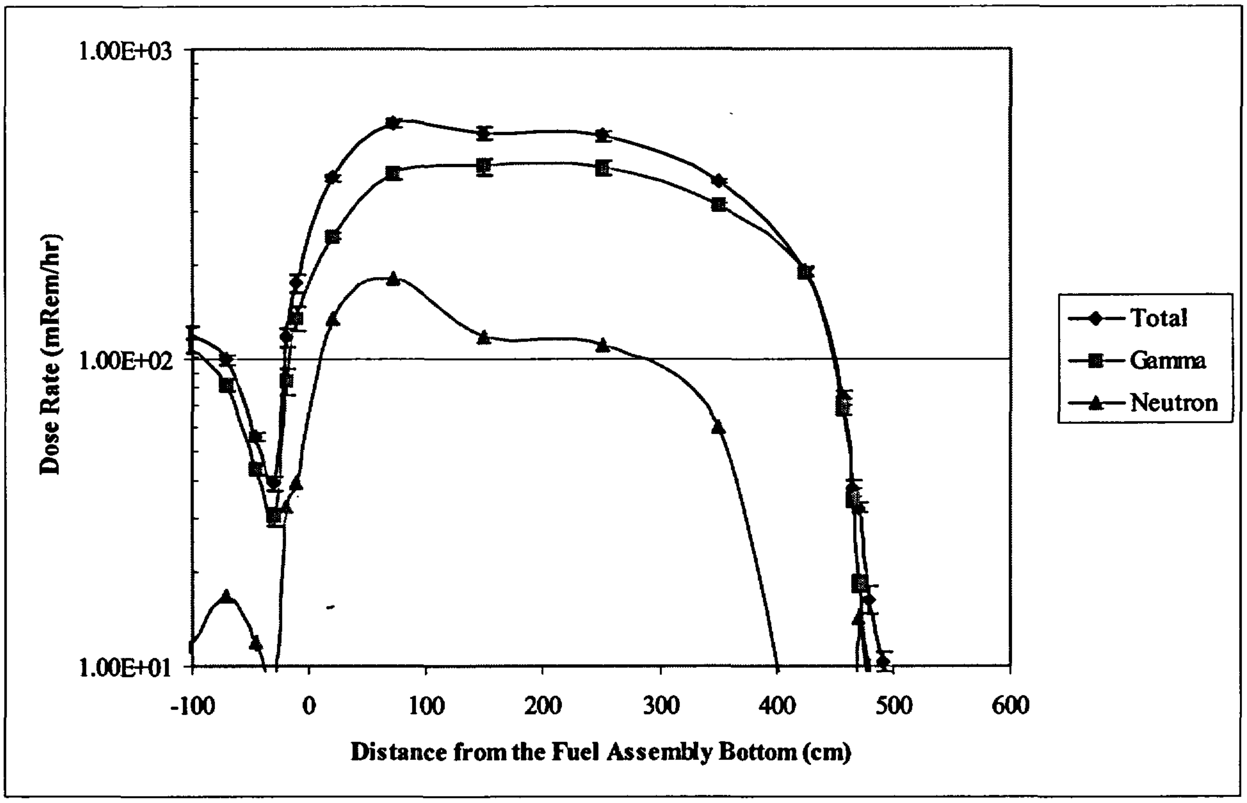


Figure P.5-22
Standardized Transfer Cask with 24PTH-S-LC DSC, Side Surface Dose Rate

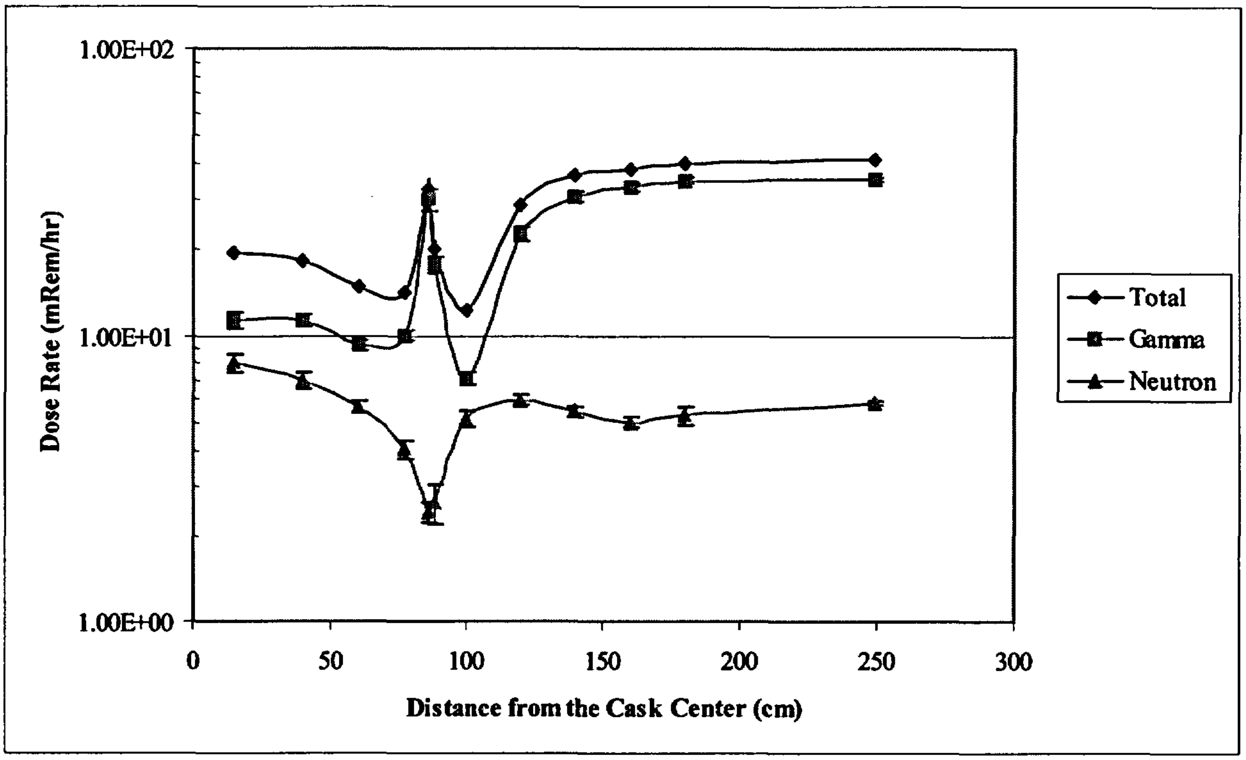


Figure P.5-23
Standardized Cask with 24PTH-S-LC DSC, Top Surface Dose Rate

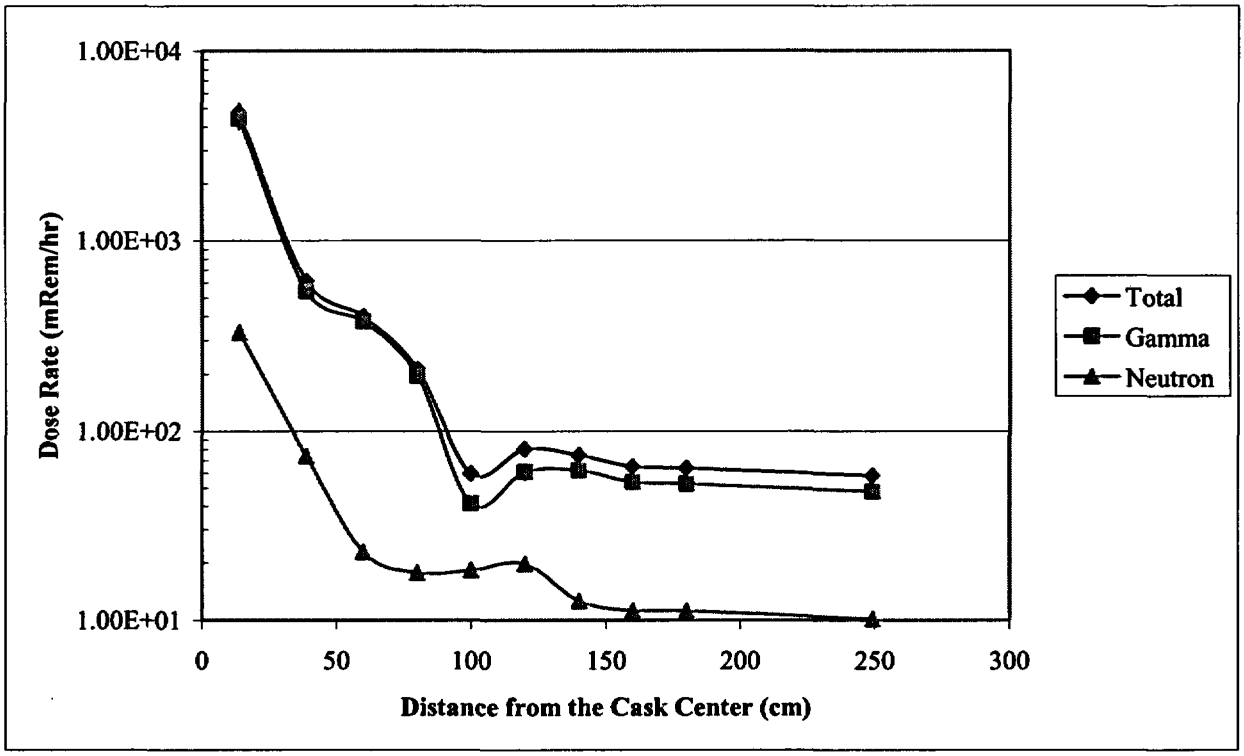


Figure P.5-24
Standardized Cask with 24PTH-S-LC DSC, Bottom Surface Dose Rate