Enclosure 8 to E-56694

Proposed Amendment 17, Revision 0
Changes to the Standardized NUHOMS® System
Updated Final Safety Analysis Report
(Public Version)

1.13 U. S. Nuclear Regulatory Commission, Office of Nuclear Materials Safety and Safeguards, "Safety Evaluation Report for Nutech Horizontal Modular System for Irradiated Fuel Topical Report," March 28, 1986.

1.14 U. S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, "Safety Evaluation Report for a Design Change to the Transfer Cask for the Duke Power Company's Independent Spent Fuel Storage Installation," February 1990.

1.CoC-APPC CoC 1004 Appendix C, ASME Code Alternatives for the Standardized NUHOMS® Horizontal Modular Storage System, Amendment 17.

1.TS CoC 1004 Technical Specifications for the Standardized NUHOMS® Horizontal Modular Storage System, Amendment 17.

7.2.3.2 Determining Fuel Assembly Minimum Required Cooling Times Using Fitting Equations

Fuel qualification tables (FQTs) are developed to provide the minimum required cooling times needed for the authorized fuel assemblies for a given decay heat limit and/or radiological sources (combined total dose rates). As described in Section M.5.2.6, the FQTs are developed for the heavy metal loadings of 380, 475 and 492 kgU. (The 492 kgU FQTs do not apply to the 32PT DSC.) As described in Chapter T.5 and Y.5, the FQTs are developed for the 61BTH and 69BTH DSCs, respectively, for two heavy metal loadings – 170 KgU and 198 KgU. *When the 61BTH DSC is stored in an HSM-H, fitting equations are not needed.*

This section provides a method for calculating the minimum required cooling time for a given fuel assembly, with an intermediate heavy metal loading that is in between the two discrete heavy metal load values mentioned above. It is demonstrated that the minimum required cooling times can be calculated by using a simple fitting equation. Section 7.2.3.2.1 provides details on the fitting equation and how to determine the various terms of the equations. Section 7.2.3.2.2 provides several examples on how to use the fitting equation. Section 7.2.3.2.3 provides verification and validation information concerning the accuracy of the fitting equation results.

7.2.3.2.1 Fitting Equations for Determining Minimum Required FA Cooling Times

Determining the minimum required cooling time for a given FA using this method is a two-step process. Step one involves clearly identifying the variable value inputs needed for the fitting equation for the applicable system, some of which are user inputs and others are looked up in FQTs. Step two involves determining the minimum required cooling time for the FA in question in the applicable system by using the looked up variable value inputs with the applicable fitting equation.

First, it is necessary to clearly identifying the variable value inputs needed for the fitting equation. These values are listed in Table 7.2-12 below.

Equation 4 Example Solution

$$CT_{new} = 6.56 * \{[(\ln(180) - 5.14] * \mathit{10.0} - [\ln(180) - 5.29] * \mathit{8.0} = \mathit{8.57} \text{ years}\}$$

The minimum required cooling time determined for the 180 kgU FA is *8.6* years. The result of the fitting equation shall be rounded up to the next highest single decimal place.

7.2.3.2.3 Verification of Fitting Equation Methodology

7.2.3.2.3.1 PWR Fuel in 24PTH and 37PTH

Table M.5-45 provides examples of cooling time determinations for various kgU and burnup/enrichment using the fitting equation and the linear interpolation using cooling times determined in FQTs explicitly generated at 380 kgU and 492 kgU.

As shown in Table M.5-45, the predicted cooling times obtained using Fitting Equation 3 for 24PTH and 37PTH are in agreement with those obtained by the linear interpolation approach. For certain cases, the predicted cooling times by the fitting equations are slightly higher.

7.2.3.2.3.2 BWR Fuel in 61BTH (Standardized HSM) and 69BTH

Table M.5-47 provides examples of cooling time determinations for various kgU and burnup/enrichment using the fitting equations and the linear interpolation using cooling times determined in FQTs explicitly generated at 170 kgU and 198 kGU.

As shown in Table M.5-47, the predicted cooling times obtained using the fitting equations are in agreement with those obtained by the linear interpolation approach. For certain cases, the predicted cooling times by the fitting equations are slightly higher.

The FQTs included in the Technical Specifications are the most critical for controlling dose rates. The following methodology is used to select the FQTs included in the Technical Specifications:

1. *For each DSC, the hottest fuel assembly in the basket is determined. This decay heat results in the bounding source term. The FQT corresponding to this decay heat is included in the Technical Specifications. All fuel to be loaded in any basket location shall comply with this FQT.*
2. *For each DSC, the bounding HLZC is determined. Because the thermally hottest fuel on the periphery of the bounding HLZC dominates the dose rate, FQTs are also included in the Technical Specifications applicable to the peripheral region of the bounding HLZC. In many cases, the FQTs from (1) and (2) are the same.*

Prior to Amendment 17, a complete set of FQTs for all decay heats were included in the respective UFSAR section for each DSC. However, because most FQTs have little effect on dose rate, a site-specific evaluation may be performed to evaluate the dose rate from fuel that does not meet the minimum cooling time requirements of the UFSAR FQTs. For Amendment 17 and later amendments, FQTs for non-bounding decay heats are not added to the UFSAR. The 61BTH DSC FQTs in Chapter T.2 are legacy examples except where noted.

The FQTs cannot be used to determine decay heat. Therefore, providing only a limited number of FQTs in the Technical Specifications does not impact compliance with decay heat requirements. Decay heat for each fuel assembly to be loaded may be determined using ORIGEN (i.e., SAS2H, ORIGEN-ARP), NRC Regulatory Guide 3.54, or other acceptable methods.

For the following systems, a complete or partial set of FQTs are included in both the UFSAR and Technical Specifications: 24P, 52B, 61BT, 61BT/OS197L, and 32PT/OS197L.

For the following systems, a complete set of FQTs are included in the UFSAR, and a limited number of FQTs are provided in the Technical Specifications: 24PTH, 24PHB, 32PT, 32PTH1, 37PTH, 61BTH, and 69BTH DSCs. Note that the 24PTH, 32PT, 32PTH1, and 37PTH DSCs share a common set of FQTs determined in Appendix M.5 and provided in Appendix M.2. However, because the HLZCs are different for these DSCs, the FQTs provided in the Technical Specifications are uniquely determined for the 24PTH, 32PT, 32PTH1, and 37PTH DSCs using the methodology outlined above. Fuel Specifications tables in the Technical Specifications (e.g., Table 1-1e for the 32PT DSC) specify the fuel qualification table numbers applicable to their respective DSCs.

Technical Specification 1.1 defines INTACT FUEL ASSEMBLY as an assembly containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pin hole leaks. Non-cladding material damage is acceptable to the extent that the fuel assembly can be handled by normal means and the fuel assembly is retrievable after all normal and off-normal conditions. This is applicable to fuel assemblies to be loaded in the 24P, 24PHB, 52B, 61BT, 32PT, 24PTH, 61BTH, 32PTH1, 69BTH or 37PTH DSCs. The bases for this definition is that the criticality and confinement functions are maintained under normal, off-normal and accident conditions of storage. The criticality analyses documented for these DSCs considers that the fuel assembly geometry remains unchanged under accident conditions and sub-criticality is assured.

T.1 General Discussion

This Appendix to the NUHOMS® Updated Final Safety Analysis Report (UFSAR) addresses the Important to Safety aspects of adding the NUHOMS®-61BTH system to the Standardized NUHOMS® system described in the UFSAR.

The NUHOMS®-61BTH system is a modular canister based spent fuel storage and transfer system, similar to the Standardized NUHOMS®-61BT system described in the UFSAR. It is designed to accommodate up to 61 intact*,* or up to 16 damaged*, or* up to 4 failed fuel cans *(FFCs)* loaded with failed fuel with the remainder intact BWR fuel assemblies, with characteristics as described in Section T.2.1. *Additionally, 4 FFCs may be loaded with failed fuel with up to 12 additional damaged fuel assemblies and the balance intact.* Alternatively, 61 damaged fuels can also be stored with characteristics as described in Section T.2.1. *See Figure 1-25 of the Technical Specifications for details on damaged and failed fuel storage locations [1.TS].*

The NUHOMS®-61BTH Dry Shielded Canister (DSC) is a dual purpose (storage/transportation) DSC, with two alternate configurations, designated as NUHOMS®-61BTH Type 1 DSC or Type 2 DSC. The 61BTH DSC is shown in Figure T.1-1.

The geometry of the 61BTH Type 1 DSC is identical to the 61BT DSC described in Appendix K of the UFSAR. The maximum heat load of 22.0 kW is allowed in Type 1 DSC. An optional top grid assembly welded to the top of the fuel compartment assembly is provided (as shown in Figure T.1-2) in lieu of a hold down ring assembly. The 61BTH Type 2 DSC is provided with thicker cover plates to accommodate the higher internal pressures and the basket is provided with aluminum rails (as shown in Figure T.1-3) to accommodate the higher DSC heat loads of up to 31.2 kW. The Type 2 DSC incorporates the top grid assembly design in lieu of the top hold down ring.

The NUHOMS®-61BTH DSC basket is designed with three alternate neutron absorber plate materials: (a) borated aluminum alloy, (b) boron carbide/aluminum metal matrix composite (MMC) and (c) Boral®. For each neutron absorber material, the NUHOMS®-61BTH DSC basket is analyzed for six alternate basket configurations, depending on the boron loadings analyzed to accommodate the various fuel enrichment levels (designated as "A" for the lowest B-10 loading to "F" for the highest B-10 loading).

The 61BTH Type 1 DSC is stored in either the standardized horizontal storage module (HSM) (Model 80 or Model 102 or Model 152 or Model 202 as described in the UFSAR), or the HSM-H described in Appendix P.1, or the HSM-HS described in Appendix U.1. A loaded Type 1 61BTH DSC is transferred from a plant's fuel/reactor building either in the OS197/OS197H transfer cask (TC), described in the UFSAR or a modified version of the OS200 TC described in Appendix U.1, or a modified version of the OS197FC TC, designated as OS197FC-B. The 61BTH Type 2 DSC is to be stored in the HSM-H/HSM-HS and transferred in OS197FC-B or OS200FC TC only. The OS200 or OS200FC is fitted with an aluminum sleeve to accommodate the smaller diameter of the 61BTH DSC.

T.1.1 Introduction

The NUHOMS®-61BTH System is designed to store up to 61 intact (including reconstituted) or up to 16 damaged *assemblies (and remaining intact) or* up to 4 FFCs loaded with failed fuel with the remainder intact BWR fuel assemblies with or without fuel channels. *Additionally, 4 FFCs may be loaded with up to 12 additional damaged fuel assemblies with the balance intact.* Alternatively, 61 damaged fuels can be stored in the NUHOMS®-61BTH DSC. *See Figure 1-25 of the Technical Specifications [1.TS] for details on the allowable storage locations for damaged and failed fuel*. The fuel to be stored is limited to a maximum initial lattice average initial enrichment of 5.0 wt. %, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of 3.0 years *for the 61BTH Type 1 or 1.0 year for the 61BTH Type 2*. The design characteristics, including physical and radiological parameters of the payload, are described in Appendix T.2.

Reconstituted assemblies containing up to *40 irradiated* stainless steel rods per *DSC, or 10 per assembly,* or *an unlimited number of* lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods are acceptable for storage in 61BTH DSC as intact fuel assemblies.

Provisions have been made for storage of up to 61 damaged fuel assemblies in lieu of an equal number of intact assemblies in cells located at the outer edge of the 61BTH basket. Damaged BWR fuel assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means and the retrievability is ensured following the normal and off-normal conditions. The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is ensured following normal and off-normal conditions. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to ensure retrievability.

Provisions have also been made for storage of up to four failed fuel assemblies in the corner cells, along with up to 12 damaged fuel assemblies in the cells located at the outer periphery of the 61BTH basket and balance intact as described in Appendix T.2.

The NUHOMS®-61BTH System consists of the following new or modified components:

- A 61BTH DSC, with two alternate configurations, designated as Type 1 61BTH DSC or Type 2 61BTH, is described in detail in Section T.1.2. It provides confinement, an inert environment, structural support, heat rejection, and criticality control for the 61 BWR fuel assemblies,
- A modified HSM-H module, as described in Section T.1.2, or HSM Model 80/102/152/202, with no modifications to the configuration as described in UFSAR Chapter 1, is provided for environmental protection, shielding and heat rejection during storage,
- An OS197 or OS197H TC with no modifications to the configuration as described in UFSAR Chapter 1, or a modified version of the OS197FC TC, designated as OS197FC-B, described in Section T.1.2, is provided for onsite transfer of the 61BTH DSCs,
- An upgraded version of the HSM-H, designated as HSM-HS, is provided to allow storage of the NUHOMS®-61BTH DSC in locations where higher seismic levels exist. The HSM-HS design configuration, described in Appendix U.1, is modified to accommodate the smaller diameter of the NUHOMS®-61BTH DSC, and

T.2.1 Spent Fuel To Be Stored

As described in Appendix T.1, there are two alternate design configurations for the NUHOMS®-61BTH DSC; Type 1 and Type 2. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged BWR fuel assemblies as specified in Table T.2-1 and Table T.2-2. The fuel to be stored is limited to a maximum lattice average initial enrichment of 5.0 wt. % ^{235}U. The maximum allowable fuel assembly average burnup is limited to 62 GWd/MTU.

The NUHOMS®-61BTH DSC is also authorized to store fuel assemblies containing blended low enriched uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material *have an assumed* higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Reconstituted fuel assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or *an unlimited number of* lower enrichment UO_2 rods instead of zircaloy clad enriched UO_2 rods are acceptable for storage in 61BTH DSCs as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel *rods* per DSC is *40* with irradiated stainless steel rods. *The maximum number of rods is not limited* with UO_2 rods or Zr rods or Zr pellets or unirradiated stainless steel rods.

The NUHOMS®-61BTH DSCs can also accommodate up to a maximum of 61 damaged fuel assemblies placed in the fuel compartments located in accordance with Figure T.2-9. When loaded with failed fuel, the damaged fuel assemblies are to be loaded as shown in Figure T.2-9. Damaged BWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly, including non-cladding damage, is to be limited such that a fuel assembly is able to be handled by normal means. The extent of damage in the fuel rods is to be limited such that a fuel pellet is not able to pass through the damaged cladding during handling and retrievability is assured following normal and off-normal conditions. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

The NUHOMS®-61BTH Type 2 DSC, when used with the top grid assembly (Alternate 1) design, is also able to accommodate up to a maximum of four failed fuel assemblies encapsulated in individual failed fuel cans and placed in cells located at the outer edge of the DSC as shown in Figure T.2-9. Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Failed fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.

Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a rod storage basket are also considered as failed fuel. Loose fuel debris, not contained in a rod storage basket may also be placed in a failed fuel can for storage, provided the size of the debris is larger than the failed fuel can screen mesh opening and it is located at least 10 in. above the top of the bottom shield plug of the DSC.

Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its contents shall be less than 705 lbs.

A 61BTH DSC containing less than 61 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

The NUHOMS®-61BTH Type 1 DSC may store up to 61 BWR fuel assemblies arranged in any of the five alternate heat load zoning configurations shown in *Figure 1-17 through Figure 1-20 and Figure 1-25a of the Technical Specifications [1.TS]* with a maximum decay heat of 0.54 kW per assembly while restricting the maximum canister heat load to 22.0 kW. The NUHOMS®-61BTH Type 2 DSCs may store up to 61 BWR fuel assemblies arranged in any of the *13* alternate heat load zoning configurations shown in *Figure 1-17 through Figure 1-24 and Figure 1-25a and Figure 1-25b, and Figure 1-25d, Figure 1-25e, and Figure 1-25f [1.TS] of the CoC 1004 Technical Specifications* with a maximum decay heat of *1.7* kW per assembly and a maximum heat load of 31.2 kW per canister.

The NUHOMS®-61BTH DSC is designed with six alternate basket configurations based on the boron content in the poison plates as listed in Table T.2-3 or Table T.2-4 or Table T.2-4a (designated as "A" for the poison plates with the lowest B-10 loading to "F" for the highest B-10 loading). Three alternate poison materials are allowed: (a) borated aluminum alloy, (b) a boron carbide/aluminum metal matrix composite (MMC), or (c) Boral®. For criticality analysis, 90% of the B-10 content present in the borated aluminum alloy and MMC is credited, while only 75% of the B-10 content in Boral® is credited.

A summary of the minimum B-10 loadings required in the poison plates as a function of the maximum lattice average enrichment level of the fuel assembly to be stored in a given 61BTH basket type is presented in Table T.2-3 for intact fuel. Table T.2-4 for damaged fuel, and in Table T.2-4a for failed and damaged fuel.

The 61BTH Type 1 DSC has a minimum cooling time of 3 years when stored in the Standardized HSM, while the 61BTH Type 1 or 2 DSC has a minimum cooling time of 1 year when stored in an HSM-H. Because bounding transfer cask and HSM source terms occur for the fuel with the highest heat load, particularly when located on the periphery of the basket, only the FQT for the bounding fuel is treated as a limit and applied to all fuel in the basket. The fuel qualification methodology is discussed in more detail in Chapter 10.

The Standardized HSM is limited to HLZC 1, 2, 3, 4, or 9 using the 61BTH Type 1 DSC. For these HLZCs, the 0.54 kW fuel assembly results in the bounding source and this basket location occurs on the periphery. Therefore, the 0.54 kW FQT (TS Table 1-4e [1.TS]) is a limit for all fuel to be loaded in the 61BTH Type 1 DSC when stored in the Standardized HSM, and the remaining FQTs included in this chapter are provided as examples for other decay heats but are not limiting when the 61BTH Type 1 DSC is stored in a Standardized HSM. Interpolation of FQT cooling times is allowed for uranium loadings between 170 kgU and 198 kgU, as well as extrapolation into the unanalyzed region, as explained in the notes to TS Table 1-4e.

For the 61BTH Type 1 or 2 DSC stored in HSM-H (HLZC 1 through 13), the 1.7 kW fuel assembly results in the bounding source and occurs on the periphery. The FQT to be applied to the 61BTH Type 1 or 2 DSC when stored in an HSM-H is provided as TS Table 1-4f [1.TS]. The FQTs provided in this chapter are examples for other decay heats but are not limiting when the 61BTH DSC is stored in an HSM-H.

Fuel below the minimum enrichments defined by either TS Table 1-4e (for Standardized HSM) or TS Table 1-4f (for HSM-H) is classified as unanalyzed fuel (UF). Limitations on the number and location of UF are provided in TS Table 1-1t [1.TS].

The NUHOMS®-61BTH DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight allowed is 705 lbs for fuel assemblies with channels and 640 lbs for fuel assemblies without channels.

The maximum fuel cladding temperature limit of 400 ºC (752 ºF) is applicable to normal conditions of storage and all short term operations from the spent fuel pool to the ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-61BTH DSC per NUREG-1536 [2.1]. In addition, NUREG-1536 [2.1] does not permit repeated thermal cycling of the fuel cladding (limited to less than 10 cycles) with cladding temperature differences greater than 65°C (117°F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570°C (1058°F) is applicable to accidents or off-normal storage thermal transients [2.1].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses, including shielding, criticality, thermal and confinement. These evaluations are described in Appendices T.5, T.6, T.4 and T.7 respectively. The fuel assembly classes considered are listed in Table T.2-2. *GE 7x7 fuel is used as the design basis fuel assembly in the shielding analysis documented in Chapter T.5.* For criticality safety, the GE 10x10 fuel assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analyses for all other fuel assembly classes, except for GNF2 and ATRIUM 11 fuel assembly classes, are performed to determine the maximum enrichment allowed as a function of the fixed poison loading. The GNF2 and ATRIUM 11 fuel assembly classes are evaluated individually. For thermal analysis, the FANP 9x9-2 fuel assembly is limiting, since it has the lowest effective thermal conductivity. The confinement analysis is based on GE 7x7 fuel assembly, since it results in the least free volume inside the DSC cavity.

For calculating the maximum internal pressure in the NUHOMS®-61BTH DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged for off normal conditions, and 100% of the fuel rods will be damaged following a design basis accident event. A minimum of 100% of the fill gas and 30% of the fission gases within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [2.1].

The maximum internal pressures used in the structural analysis for the NUHOMS®-61BTH Type 1 DSC are 10, 20, and 65 psig for normal, off-normal and accident conditions, respectively, during storage and transfer operations. The maximum internal pressures for the 61BTH Type 2 DSC are 15, 20, and 120 psig for normal, off-normal and accident conditions, respectively during storage and transfer operations.

T.2.1.1 General Operating Functions

No change to Section 3.1.2.

T.2.5 Summary of NUHOMS®-61BTH System Design Criteria

T.2.5.1 61BTH DSC Design Criteria

The NUHOMS®-61BTH DSC is designed to store intact and/or damaged BWR fuel assemblies with assembly average burnup, lattice average initial enrichment and cooling time as described in Table T.2-1 and Table T.2-2. The maximum total heat generation rate of the stored fuel is limited to *1.7* kW per fuel assembly for the Type 2 DSC and 0.54 kW per fuel assembly for the Type 1 DSC. The maximum heat load per canister is limited to 31.2 kW for the Type 2 DSC and 22.0 kW for Type 1 DSC, in order to keep the maximum fuel cladding temperature below the limit [2.4] necessary to ensure cladding integrity. The fuel cladding integrity is assured by the NUHOMS®-61BTH DSC and basket design, which limits fuel cladding temperature and maintains a non-oxidizing environment in the DSC cavity as described in Chapters T.4 and T.7.

The NUHOMS®-61BTH DSC is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. A combination of fixed neutron absorbers and favorable geometry are employed to maintain the upper subcritical limit of 0.9415. The fixed neutron absorbers are in the form of plates made from either Borated Aluminum alloy or MMC or Boral®.

The NUHOMS®-61BTH DSC (shell and closure) is designed and fabricated as a Class 1 component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [2.2], and the alternative provisions to the ASME Code as described in Table T.3.1-2.

The basket is designed and fabricated in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG-3200 [2.2] and the alternative provisions to the ASME Code as described in Table T.3.1-2.

The principal design loadings for the NUHOMS®-61BTH DSC are provided in Table T.2-14. The applicable load combinations for the NUHOMS®-61BTH DSC are presented in Table T.2-11 and the corresponding stress criteria are presented in Table T.2-12 and Table T.2-13.

The NUHOMS®-61BTH system is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornadoes, lightning and floods. Chapter T.11 describes the NUHOMS®-61BTH DSC behavior under these accident conditions.

The NUHOMS®-61BTH DSC design, fabrication and testing are covered by Transnuclear's Quality Assurance Program, which conforms to the criteria in Subpart G of 10CFR72.

T.2.5.2 HSM-H Models 80, 102, 152, 202 and HSM-H Design Criteria

There is no change to the HSM Models 80, 102, 152, 202 design criteria as presented in Chapter 3 of the UFSAR for Models 80/102 and the appropriate appendix for Models 152/202. The maximum heat load allowed for storage of a 61BTH in these HSMs remains at 22 kW.

There is no change to the HSM-H design criteria presented in Appendix P.2 except for accommodating a payload (61BTH DSC) with a maximum decay heat load of 31.2 kW.

Table T.2-4b

The detailed information associated with this table can be found in CoC 1004 Technical Specifications Table 1-1x *[1.TS]*.

Table T.2-5

Example BWR Fuel Qualification Table for 0.22 kW per FA for the NUHOMS®-61BTH DSC

(Minimum years of cooling time after reactor core discharge for fuel with 198 kgU per FA)

BU	Assembly Average Initial Enrichment (wt. % U-235)																																	
GWd/MTU	0.9	1.2	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
20	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
23	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
25	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
28	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
30	10.5	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5
32				11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0
34				14.0	14.0	14.0	14.0	14.0	14.0	14.0	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5
36				16.5	16.0	16.0	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	14.5	14.5	14.5	14.5
38				19.5	19.0	19.0	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.5	18.0	18.0	18.0	18.0	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5
39				21.0	21.0	20.5	20.5	20.5	20.5	20.5	20.5	20.0	20.0	20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0
40						23.5	23.0	22.5	22.5	22.0	21.5	21.5	21.5	21.5	21.5	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5
41										23.5	23.5	23.0	23.0	23.0	23.0	23.0	22.5	22.5	22.5	22.5	22.5	22.5	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	21.5
42										24.5	24.5	24.5	24.5	24.5	24.5	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5
43										26.0	26.0	26.0	26.0	26.0	25.5	25.5	25.5	25.5	25.5	25.5	25.5	25.5	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.5
44										27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.0	27.0	27.0	27.0	27.0	27.0	27.0	27.0	26.5	26.5	26.5	26.5	26.5	26.0	26.5	26.0	26.0	26.0
45										29.0	29.0	29.0	29.0	29.0	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.5	28.0	28.0	28.0	28.0	28.0	27.5	27.5	27.5	27.5	27.5	27.5
46										30.5	30.5	30.5	30.5	30.0	30.0	30.0	30.0	30.0	30.0	30.0	30.0	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.0	29.0	29.0
47										31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.5	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	31.0	30.5	30.5	30.5	30.5	30.5	30.5
48										33.0	33.0	33.0	33.0	33.0	33.0	33.0	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.5	32.0	32.0	32.0	32.0	32.0	32.0	32.0
49										34.5	34.5	34.5	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	33.5	33.5	33.5	33.5	33.5	33.5	33.5	33.0	33.0	33.0	33.0
50										36.0	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.0	35.0	35.0	35.0	35.0	35.0	35.0	34.5	34.5	34.5	34.5	34.5	34.5	34.5	34.5
51										37.0	37.0	37.0	37.0	37.0	37.0	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0	36.0
52										38.5	38.0	38.0	38.0	38.0	38.0	38.0	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.5	37.0
53										39.5	39.5	39.5	39.5	39.5	39.5	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.0	39.5	39.0	38.5	38.5	38.5
54										41.0	41.0	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	39.5
55										41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.0	41.0	41.0	41.0	41.0
56										43.0	43.0	43.0	43.0	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5	42.5
57										44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	44.0	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5	43.5
58										45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	45.0	44.5	44.5	44.5	44.5
59										46.0	46.0	46.0	46.0	46.0	46.5	46.0	46.0	46.0	46.0	46.0	46.0	46.0	46.0	46.5	46.0	46.0	46.0	46.0	46.0	46.0	46.0	46.0	46.0	45.5
60										47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0	47.0
61										48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.0	48.5	48.0	48.0	48.0	48.0
62										49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5	49.5

(continued)

Note: The page that follows Table T.2-10 provides explanatory notes and limitations regarding the use of this table.

Table T.2-5
Example BWR Fuel Qualification Table for 0.22 kW per FA for the NUHOMS®-61BTH DSC

(Minimum years of cooling time after reactor core discharge for fuel with 170 kgU per FA)

BU GWd/MTU	Assembly Average Initial U-235 Enrichment, wt. %																
	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0
20	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0
23	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
25	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
28	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0
30	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0
32									8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0
34									10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0
36											11.5	11.5	11.0	11.0	11.0	11.0	10.5
38											13.5	13.5	13.5	13.0	13.0	13.0	13.0
39											15.0	14.5	14.5	14.5	14.0	14.0	14.0
40														16.0	15.5	15.5	15.5
41																	
42																	
43																	
44																	
45																	
46																	
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BU GWd/MTU	Assembly Average Initial U-235 Enrichment, wt. %																								
	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
23	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
25	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
28	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5
30	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
32	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0
34	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0
36	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	9.5	9.5	9.5	9.5
38	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	11.5	11.5	11.5	11.5	11.5	11.0	11.0
39	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0
40	15.0	15.0	14.5	14.5	14.5	14.5	14.5	14.5	14.0	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.5	13.5	13.5	13.5	13.5	13.5	13.5	13.5
41	16.5	16.0	16.0	16.0	16.0	16.0	15.5	15.5	15.5	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0
42	17.5	17.5	17.5	17.5	17.0	17.0	17.0	16.5	16.5	16.5	16.5	16.5	16.5	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.5	15.5	15.5	15.5
43	19.0	19.0	18.5	18.5	18.5	18.5	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0
44	20.5	20.0	20.0	20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5
45	21.5	21.5	21.5	21.5	21.0	21.0	21.0	21.0	21.0	20.5	20.5	20.0	20.0	20.0	20.0	20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5
46	23.0	23.0	22.5	22.5	22.5	22.5	22.5	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0
47	24.5	24.0	24.0	24.0	24.0	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5
48	25.5	25.5	25.5	25.5	25.0	25.0	25.0	25.0	25.0	24.5	24.5	24.0	24.0	24.0	24.0	24.0	24.0	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5
49	27.0	27.0	26.5	26.5	26.5	26.5	26.5	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0
50	28.5	28.0	28.0	28.0	28.0	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5
51	29.5	29.5	29.5	29.0	29.0	29.0	29.0	29.0	28.5	28.5	28.0	28.0	28.0	28.0	28.0	28.0	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5	27.5
52	31.0	31.0	30.5	30.5	30.0	30.0	30.0	30.0	30.0	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5	29.5
53	32.0	32.0	31.5	31.5	31.5	31.5	31.5	31.0	31.0	31.0	31.0	31.0	30.5	30.5	30.5	30.0	30.0	30.0	30.0	30.0	30.0	30.0	29.5	29.5	29.5
54	33.0	33.0	33.0	33.0	33.0	32.5	32.5	32.5	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0	32.0
55	34.5	34.5	34.0	34.0	34.0	33.5	33.5	33.5	33.5	33.5	33.5	33.0	33.0	33.0	33.0	33.0	33.0	32.5	32.5	32.5	32.0	32.0	32.0	32.0	32.0
56	35.5	35.5	35.0	35.0	35.0	35.0	35.0	35.0	34.5	34.5	34.5	34.5	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0	34.0
57	36.5	36.5	36.5	36.5	36.0	36.0	36.0	35.5	35.5	35.5	35.5	35.5	35.5	35.5	35.0	35.0	35.0	35.0	35.0	35.0	35.0	34.5	34.5	34.5	34.5
58	37.5	37.5	37.5	37.0	37.0	37.0	37.0	37.0	37.0	37.0	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5	36.5
59	38.5	38.5	38.5	38.5	38.5	38.0	38.0	38.0	38.0	38.0	38.0	38.0	38.0	37.5	37.5	37.5	37.0	37.0	37.0	37.0	37.0	37.0	37.0	37.0	36.5
60	39.5	39.5	39.5	39.5	39.5	39.5	39.5	39.0	39.0	39.0	39.0	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5	38.5
61	40.5	40.5	40.5	40.5	40.5	40.0	40.0	40.0	40.0	40.0	40.0	40.0	40.0	39.5	39.5	39.5	39.5	39.5	39.5	39.5	39.5	39.5	39.0	39.0	39.0
62	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.5	41.0	41.0	41.0	41.0	41.0	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5	40.5

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-6
Example BWR Fuel Qualification Table for 0.35 kW per FA for the NUHOMS®-61BTH DSC

(Minimum years of cooling time after reactor core discharge for fuel with 198 kgU per FA)

BU GWd/MTU	Assembly Average Initial Enrichment (wt. % U-235)																																
	0.9	1.2	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
25	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
28	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
30	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
32			6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
34				6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
36				6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
38				7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
39				7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
40						8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
41										8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
42										8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
43										9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0
44										9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5
45										10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0
46										11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5
47										12.0	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0
48										12.5	12.5	12.5	12.5	12.5	12.5	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.0	10.5	10.5	10.5	10.5
49										13.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5
50										14.5	14.5	14.5	14.5	14.5	14.5	14.5	14.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0
51										15.5	15.0	15.0	15.0	15.0	15.0	15.0	15.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0	13.0	13.0	13.0	13.0	13.0	13.0
52										16.5	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	15.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0
53										17.5	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.0	15.0	15.0	15.0	15.0	15.0
54										18.5	18.5	18.0	18.0	18.0	18.0	18.0	18.0	18.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	17.0	16.0	16.0	16.0	16.0	16.0	16.0
55										20.5	20.5	19.0	19.0	19.0	19.0	19.0	19.0	19.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	17.0	17.0	17.0	17.0	17.0	17.0
56										21.5	21.5	20.5	20.5	20.5	20.5	20.5	20.5	20.5	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	19.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0
57										22.5	22.5	22.5	21.5	21.5	21.5	21.5	21.5	21.5	21.5	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	20.0	19.0	19.0	19.0	19.0	19.0	19.0
58										22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	22.5	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	21.0	20.0	20.0	20.0	20.0
59										23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	23.5	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	22.0	21.0	21.0	21.0	21.0
60										24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	24.5	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	23.0	22.0	22.0	22.0	22.0
61										26.5	26.5	26.5	25.5	25.5	25.5	25.5	25.5	25.5	25.5	25.5	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	24.0	23.0	23.0	23.0	23.0
62										27.5	27.5	27.5	27.5	26.0	26.0	26.0	26.0	26.0	26.0	26.0	26.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	25.0	24.0	24.0

(continued)

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-6
Example BWR Fuel Qualification Table for 0.35 kW per FA for the NUHOMS®-61BTH DSC

(Minimum years of cooling time after reactor core discharge for fuel with 170 kgU per FA)

BU	Assembly Average Initial U-235 Enrichment, wt. %																																							
GWd/MTU	0.9	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
28	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5
30	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
32							5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
34							5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
36									5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
38									6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0
39									6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
40												7.0	7.0	7.0	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5
41																6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
42																7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
43																7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
44																7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5
45																8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
46																8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0
47																9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
48																9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
49																10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0
50																10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5
51																11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5
52																12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0
53																12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.0	10.0	10.0
54																13.5	13.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5
55																14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	13.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5
56																15.5	15.0	15.0	14.5	14.5	14.5	14.5	14.5	14.5	14.5	13.5	13.5	13.5	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0
57																16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.5	15.5	14.5	14.5	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.5	13.5	13.5	13.5	13.0	13.0
58																17.0	17.0	17.0	16.5	16.5	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.5	15.5	14.5	14.5	14.0	14.0	14.0	14.0	14.0	14.0	14.0	14.0
59																18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	16.5	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	14.5	14.0
60																19.0	19.0	18.5	18.5	18.5	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	16.5	16.0	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5
61																20.0	20.0	19.5	19.5	19.5	19.5	19.5	19.5	19.5	19.5	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	17.5	16.5	16.0	16.0
62																21.0	20.5	20.5	20.5	20.0	20.0	20.0	20.0	20.0	20.0	20.0	19.0	18.5	18.5	18.5	18.5	18.0	18.0	18.0	18.0	18.0	18.0	18.0	18.0	17.0

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-7

Example BWR Fuel Qualification Table for 0.393 kW per FA for the NUHOMS®-61BTH DSC

(Minimum years of cooling time after reactor core discharge for fuel with 198 kgU per FA)

BU	Assembly Average Initial Enrichment (wt. % U-235)																																
GWd/MTU	0.9	1.2	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
28	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
30	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
32				5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
34				5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5
36				6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
38				6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
39				6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
40						7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
41										7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0
42										7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
43										7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5
44										8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
45										8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0
46										8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5
47										9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
48										10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5
49										10.5	10.5	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0
50										11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0
51										11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5
52										12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5
53										13.5	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0
54										14.0	14.0	14.0	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5
55										15.0	15.0	14.5	14.5	14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0
56										16.0	16.0	15.5	15.5	15.5	15.0	15.0	15.0	15.0	15.0	14.5	14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0
57										17.0	16.5	16.5	16.5	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.0	15.0	14.5	14.5	14.5	14.5	14.5	14.5	14.0	14.0	14.0	14.0	14.0
58										18.0	17.5	17.5	17.5	17.5	17.0	17.0	16.5	16.5	16.5	16.5	16.5	16.0	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.0	15.0	14.5	14.5
59										19.5	18.5	18.5	18.0	18.0	18.0	17.5	17.5	17.5	17.5	17.0	17.0	17.0	17.0	16.5	16.5	16.5	16.0	16.0	16.0	16.0	16.0	16.0	15.5	15.5
60										20.0	19.5	19.5	19.5	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.5	18.0	17.5	17.5	17.5	17.0	17.0	17.0	17.0	17.0	16.5	16.5	16.5	16.5
61										20.5	20.5	20.5	20.5	20.5	20.0	19.5	19.5	19.5	19.0	19.0	19.0	18.5	18.5	18.5	18.5	18.5	18.0	18.0	18.0	18.0	17.5	17.5	17.5	17.5
62										21.5	21.5	21.0	21.0	21.0	21.0	20.5	20.5	20.5	20.0	20.0	20.0	19.5	19.5	19.5	19.0	19.0	19.0	19.0	19.0	19.0	18.5	18.5	18.5	18.0

(continued)

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-7
Example BWR Fuel Qualification Table for 0.393 kW per FA for the NUHOMS®-61BTH DSC

(Minimum years of cooling time after reactor core discharge for fuel with 170 kgU per FA)

BU	Assembly Average Initial U-235 Enrichment, wt. %																																							
GWd/MTU	0.9	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
30	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
32							4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
34							5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
36									5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
38									5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5
39									5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
40												6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
41																6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0
42																6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
43																6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
44																6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5
45																7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
46																7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0
47																7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
48																7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5
49																8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
50																8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0
51																9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
52																9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5
53																10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0
54																10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5
55																11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0
56																12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5
57																12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	9.5
58																13.5	13.0	13.0	13.0	13.0	13.0	13.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5
59																14.0	14.0	13.5	13.5	13.5	13.5	13.5	13.0	12.5	12.5	12.5	12.5	12.5	12.5	12.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.5
60																15.0	15.0	14.5	14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	12.0	12.0	12.0
61																16.0	15.5	15.5	15.0	15.0	15.0	15.0	15.0	15.0	14.0	14.0	14.0	13.5	13.5	13.5	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.5
62																16.5	16.5	16.0	16.0	16.0	16.0	16.0	16.0	16.0	15.0	15.0	14.5	14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.0	13.0	13.0	13.0	13.0

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-8

Example BWR Fuel Qualification Table for 0.48 kW per FA for the NUHOMS®-61BTH DSC

(Minimum years of cooling time after reactor core discharge for fuel with 198 kgU per FA)

BU	Assembly Average Initial Enrichment (wt. % U-235)																																
GWd/MTU	0.9	1.2	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
30	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
32				4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5
34				4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
36				5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
38				5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
39				5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
40						6.5	6.0	6.0	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
41										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
42										6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
43										6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
44										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
45										6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5
46										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
47										7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
48										7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0
49										7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
50										7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5
51										8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
52										8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0
53										9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
54										9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5
55										10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0
56										10.5	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5
57										11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	9.0	9.0
58										11.5	11.5	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.5	9.5	9.5	9.0	9.0
59										12.0	12.0	12.0	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5
60										13.0	12.5	12.5	12.5	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.0	10.0	10.0
61										13.5	13.5	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0	11.0	11.0	11.0	10.5
62										14.0	14.0	14.0	14.0	13.5	13.5	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.0	11.0

(continued)

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-8
Example BWR Fuel Qualification Table for 0.48 kW per FA for the NUHOMS®-61BTH DSC

(Minimum years of cooling time after reactor core discharge for fuel with 170 kgU per FA)

BU	Assembly Average Initial U-235 Enrichment, wt. %																																						
GWd/MTU	0.9	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
32							4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
34							4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5
36									4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
38									5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
39									5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
40												6.0	5.5	5.5	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0
41																5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
42																5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
43																5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
44																5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5
45																5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
46																5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
47																6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
48																6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0
49																6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
50																6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
51																6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5
52																7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
53																7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0
54																7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
55																8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5
56																8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
57																8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0
58																9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0
59																10.0	9.5	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5
60																11.0	10.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0
61																12.0	11.5	10.5	10.0	10.0	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0
62																13.0	12.5	11.5	11.0	10.5	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5	9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-9

The detailed information associated with this table can be found in CoC 1004 Technical Specifications Table 1-4e [1.TS].

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Table T.2-10

Example BWR Fuel Qualification Table for 0.7 kW per FA for the NUHOMS®-61BTH Type 2 DSC

(Minimum years of cooling time after reactor core discharge for fuel with 198 kgU per FA)

BU GWd/MTU	Assembly Average Initial Enrichment (wt. % U-235)																																	
	0.9	1.2	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	3.0	3.0	3.0	3.0	3.0	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32				3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34				3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36				3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0
38				3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
39				4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
40						5.0	5.0	4.5	4.5	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
41										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
42										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
43										5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
44										5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
45										4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
46										5.5	5.5	5.5	5.5	5.5	5.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0
47										6.0	6.0	6.0	6.0	6.0	5.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0
48										6.0	6.0	6.0	6.0	6.0	6.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
49										6.0	6.0	6.0	6.0	6.0	6.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
50										6.5	6.5	6.5	6.5	6.5	6.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
51										5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
52										5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5
53										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
54										5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
55										6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
56										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0
57										6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
58										6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5
59										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5
60										6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5
61										7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
62										7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0

(continued)

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-10
Example BWR Fuel Qualification Table for 0.7 kW per FA for the NUHOMS®-61BTH Type 2 DSC

(Minimum years of cooling time after reactor core discharge for fuel with 170 kgU per FA)

BU GWd/MTU	Assembly Average Initial U-235 Enrichment, wt. %																																						
	0.9	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32							3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34							3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36									3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
38									3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
39									3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
40												4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
41																3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
42																3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0
43																3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
44																4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
45																4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
46																4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
47																4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5
48																4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5
49																4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
50																4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
51																4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
52																4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
53																5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0
54																5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
55																5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
56																5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
57																5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5
58																5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5
59																5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
60																6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
61																6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0
62																6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Notes: Tables T.2-5 through T.2-10 and Tables T.2-17 through T.2-18

- *FQTs for 0.22 kW, 0.35 kW, 0.393 kW, 0.48 kW, 0.7 kW, 0.9 kW, and 1.2 kW are provided in this chapter as examples and are not limiting.*
- *When loaded in a Standardized HSM, the 61BTH Type 1 DSC (HLZC 1, 2, 3, 4, or 9) is limited by the FQT provided in TS Table 1-4e [1.TS]. TS Table 1-4e is based on 0.54 kW. A complete set of notes for application of TS Table 1-4e are provided in the TS.*
- *When loaded in an HSM-H, the 61BTH Type 1 or 2 DSC is limited by the FQT provided in TS Table 1-4f [1.TS]. A complete set of notes for application of TS Table 1-4f are provided in the TS.*

Table T.2-16
Deleted

Table T.2-17

Example BWR Fuel Qualification Table for 0.9 kW per FA for the NUHOMS®-61BTH Type 2 DSC

(Minimum years of cooling time after reactor core discharge for fuel with 198 kgU per FA)

BU GWd/MTU	Assembly Average Initial Enrichment (wt. % U-235)																																						
	0.9	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32							3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34							3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36									3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
38									3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
39									3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
40												3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
41																3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
42																3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
43							Not Analyzed									3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
44																3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
45																3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0
46																3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
47																4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
48																4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
49																4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
50																4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
51																4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5
52																4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
53																4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
54																4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
55																4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
56																4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0
57																5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0
58																5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
59																5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
60																5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5
61																5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5
62																5.5	5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	4.5	4.5

(continued)

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-17
Example BWR Fuel Qualification Table for 0.9 kW per FA for the NUHOMS®-61BTH Type 2 DSC

(Minimum years of cooling time after reactor core discharge for fuel with 170 kgU per FA)

BU	Assembly Average Initial Enrichment (wt. % U-235)														
GWd/MTU	0.9	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5
10	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32							3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34							3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36									3.0	3.0	3.0	3.0	3.0	3.0	3.0
38									3.0	3.0	3.0	3.0	3.0	3.0	3.0
39									3.0	3.0	3.0	3.0	3.0	3.0	3.0
40												3.0	3.0	3.0	3.0
41															
42															
43							Not Analyzed								
44															
45															
46															
47															
48															
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60															
61															
62															

BU GWd/MTU	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	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
15	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
20	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
23	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
25	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
28	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
30	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
32	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
34	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
36	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
38	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
39	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
40	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
41	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
42	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
43	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
44	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
45	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
46	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
47	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
48	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
49	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
50	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0
51	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
52	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
53	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
54	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
55	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
56	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5
57	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5
58	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
59	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
60	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
61	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
62	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-18

Example BWR Fuel Qualification Table for 1.2 kW per FA for the NUHOMS®-61BTH Type 2 DSC

(Minimum years of cooling time after reactor core discharge for fuel with 198 kgU per FA)

Burn-Up, GWD/MTU	Assembly Averaged Initial U-235 Enrichment, wt.%																																
	0.5	**0.8**	**0.9**	**1.6**	**1.7**	**1.8**	**1.9**	**2.0**	**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**
6	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
20	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
21			2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
31			2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
32					2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
34					2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
35					2.1	2.1	2.1	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
36							2.1	2.1	2.1	2.1	2.1	2.1	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
37							2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
38							2.3	2.2	2.2	2.2	2.2	2.2	2.2	2.1	2.1	2.1	2.1	2.1	2.1	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
39							2.3	2.3	2.3	2.3	2.3	2.3	2.3	2.2	2.2	2.2	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
40										2.4	2.4	2.4	2.4	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.0
41										2.5	2.5	2.4	2.4	2.3	2.3	2.3	2.3	2.3	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.1	2.1	2.1	2.1	2.1	2.1
42										2.5	2.5	2.5	2.5	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	2.2	2.2	2.2	2.2	2.2
43										2.6	2.6	2.6	2.6	2.4	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	2.3	2.2	2.2	2.2	2.2
44										2.7	2.7	2.6	2.6	2.5	2.5	2.5	2.5	2.4	2.4	2.4	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
45										2.7	2.7	2.7	2.7	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4	2.4	2.4	2.4	2.4	2.4	2.4	2.3	2.3
46										2.8	2.8	2.8	2.8	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4	2.4	2.4	2.4
47										2.9	2.9	2.8	2.8	2.7	2.7	2.7	2.7	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5
48										2.9	2.9	2.9	2.9	2.8	2.8	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5
49										3.0	3.0	3.0	3.0	2.8	2.8	2.8	2.8	2.8	2.8	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.6	2.6	2.6	2.6	2.6	2.6	2.6
50										3.1	3.1	3.0	3.0	2.9	2.9	2.9	2.9	2.8	2.8	2.8	2.8	2.8	2.8	2.8	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.6	2.6
51										3.2	3.2	3.1	3.1	3.0	3.0	2.9	2.9	2.9	2.9	2.9	2.9	2.9	2.8	2.8	2.8	2.8	2.8	2.8	2.8	2.7	2.7	2.7	2.7	2.7
52										3.2	3.2	3.2	3.2	3.1	3.0	3.0	3.0	3.0	3.0	3.0	2.9	2.9	2.9	2.9	2.9	2.9	2.8	2.8	2.8	2.8	2.8	2.8	2.8	2.8
53										3.3	3.3	3.3	3.2	3.1	3.1	3.1	3.1	3.1	3.0	3.0	3.0	3.0	3.0	3.0	2.9	2.9	2.9	2.9	2.9	2.9	2.9	2.8	2.8	2.8
54										3.4	3.4	3.3	3.3	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.1	3.0	3.0	3.0	3.0	3.0	3.0	3.0	2.9	2.9	2.9	2.9	2.9
55										3.5	3.5	3.4	3.4	3.3	3.3	3.2	3.2	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.0	3.0	3.0	3.0	3.0	3.0	3.0
56										3.5	3.5	3.5	3.5	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	3.0	3.0	3.0
57										3.6	3.6	3.6	3.5	3.4	3.4	3.4	3.4	3.3	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
58										3.7	3.7	3.6	3.6	3.5	3.5	3.5	3.4	3.4	3.4	3.4	3.4	3.3	3.3	3.3	3.3	3.3	3.3	3.2	3.2	3.2	3.2	3.2	3.2	3.2
59										3.8	3.8	3.7	3.7	3.6	3.6	3.5	3.5	3.5	3.5	3.5	3.4	3.4	3.4	3.4	3.4	3.3	3.3	3.3	3.3	3.3	3.3	3.3	3.2	3.2
60										3.9	3.9	3.8	3.8	3.7	3.6	3.6	3.6	3.6	3.6	3.5	3.5	3.5	3.5	3.5	3.4	3.4	3.4	3.4	3.4	3.4	3.3	3.3	3.3	3.3
61										3.9	3.9	3.9	3.9	3.7	3.7	3.7	3.7	3.7	3.6	3.6	3.6	3.6	3.6	3.5	3.5	3.5	3.5	3.5	3.4	3.4	3.4	3.4	3.4	3.4
62										4.0	4.0	4.0	4.0	3.8	3.8	3.8	3.8	3.7	3.7	3.7	3.7	3.7	3.6	3.6	3.6	3.6	3.6	3.5	3.5	3.5	3.5	3.5	3.4	3.4
Enr. Wt.%	**0.5**	**0.8**	**0.9**	**1.6**	**1.7**	**1.8**	**1.9**	**2.0**	**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**

(continued)

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

Table T.2-18
Example BWR Fuel Qualification Table for 1.2 kW per FA for the NUHOMS®-61BTH Type 2 DSC

(Minimum years of cooling time after reactor core discharge for fuel with 170 kgU per FA)

Burn-Up, GWD/MTU	Assembly Averaged Initial U-235 Enrichment, wt.%																																
	0.5	**0.8**	**0.9**	**1.6**	**1.7**	**1.8**	**1.9**	**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**
6	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
20	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
21			2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
31			2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
32					2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
35					2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
36							2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
39							2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
40									2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
41									2.2	2.2	2.2	2.1	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
42									2.2	2.2	2.2	2.2	2.1	2.1	2.1	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
43									2.3	2.3	2.3	2.3	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
44									2.4	2.4	2.3	2.3	2.2	2.2	2.2	2.2	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.0	2.0	2.0	2.0	2.0	2.0	2.0
45									2.4	2.4	2.4	2.4	2.3	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.0
46									2.5	2.5	2.4	2.4	2.3	2.3	2.3	2.3	2.3	2.3	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.2	2.1	2.1	2.1	2.1	2.1	2.1
47									2.5	2.5	2.5	2.5	2.4	2.4	2.4	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	2.2	2.2	2.2	2.2	2.2
48									2.6	2.6	2.6	2.6	2.4	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
49									2.7	2.7	2.6	2.6	2.5	2.5	2.5	2.5	2.4	2.4	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
50									2.7	2.7	2.7	2.7	2.6	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4	2.4	2.4	2.4	2.4	2.4	2.3	2.3	2.3	2.3
51									2.8	2.8	2.8	2.7	2.6	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4	2.4	2.4	2.4	2.4	2.4
52									2.9	2.9	2.8	2.8	2.7	2.7	2.7	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.4	2.4	2.4
53									2.9	2.9	2.9	2.9	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.5	2.5	2.5	2.5	2.5	2.5
54									3.0	3.0	2.9	2.9	2.8	2.8	2.8	2.8	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.6	2.5
55									3.1	3.1	3.0	3.0	2.9	2.9	2.8	2.8	2.8	2.8	2.8	2.8	2.8	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.6	2.6	2.6	2.6	2.6
56									3.1	3.1	3.1	3.1	2.9	2.9	2.9	2.9	2.9	2.9	2.8	2.8	2.8	2.8	2.8	2.8	2.8	2.7	2.7	2.7	2.7	2.7	2.7	2.7	2.7
57									3.2	3.2	3.1	3.1	3.0	3.0	3.0	3.0	2.9	2.9	2.9	2.9	2.9	2.9	2.8	2.8	2.8	2.8	2.8	2.8	2.8	2.7	2.7	2.7	2.7
58									3.2	3.2	3.2	3.2	3.1	3.1	3.0	3.0	3.0	3.0	3.0	3.0	2.9	2.9	2.9	2.9	2.9	2.9	2.8	2.8	2.8	2.8	2.8	2.8	2.8
59									3.3	3.3	3.3	3.3	3.1	3.1	3.1	3.1	3.1	3.1	3.0	3.0	3.0	3.0	3.0	3.0	2.9	2.9	2.9	2.9	2.9	2.9	2.9	2.8	2.8
60									3.4	3.4	3.3	3.3	3.2	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.0	3.0	3.0	3.0	3.0	3.0	3.0	2.9	2.9	2.9	2.9	2.9
61									3.5	3.5	3.4	3.4	3.3	3.3	3.2	3.2	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.1	3.0	3.0	3.0	3.0	3.0	3.0	3.0	2.9
62									3.5	3.5	3.5	3.5	3.3	3.3	3.3	3.3	3.3	3.3	3.2	3.2	3.2	3.2	3.2	3.1	3.1	3.1	3.1	3.1	3.1	3.1	3.0	3.0	3.0
Enr. Wt.%	**0.5**	**0.8**	**0.9**	**1.6**	**1.7**	**1.8**	**1.9**	**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**

Note: The page that follows Table T.2-10 provides the explanatory notes and limitations regarding the use of this table.

The detailed information associated with this figure can be found in CoC 1004 Technical Specifications Figure 1-25 *[1.TS]*.

Figure T.2-9

The detailed information associated with this figure can be found in CoC 1004 Technical Specifications Figure 1-25a *[1.TS]*.

Figure T.2-10

The detailed information associated with this figure can be found in CoC 1004 Technical Specifications Figure 1-25b *[1.TS]*.

Figure T.2-11

T.4 Thermal Evaluation

T.4.1 Discussion

This chapter presents the thermal evaluations which demonstrate that the NUHOMS®-61BTH System meets the thermal requirements of 10 CFR Part 72 for the dry storage of spent fuel. The NUHOMS®-61BTH System is designed to passively reject decay heat during storage and transfer for normal, off-normal and accident conditions while maintaining temperatures and pressures within specified regulatory limits.

The elevation of the ISFSI location up to 5000 feet with respect to sea level has negligible impact on the thermal performance of the NUHOMS® -61BTH System. Therefore, 61BTH System can be used for elevations up to 5000 feet with respect to sea level.

Several thermal design criteria are established for the thermal analysis of the 61BTH DSC basket as discussed below.

- Maximum temperatures of the confinement structural components must not adversely affect the confinement function,
- Maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short term fuel loading and transfer operations including vacuum drying and helium backfilling of the 61BTH DSC per Interim Staff Guidance (ISG) No. 11, Revision 3 [4.15]. In addition, ISG-11 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65 °C (117 °F) during drying and backfilling operations,
- Maximum fuel cladding temperature limit of 570 °C (1058 °F) is applicable to accidents or off-normal thermal transients [4.15],
- The maximum DSC cavity internal design pressures are as follows:
    - For 61BTH Type 1 DSC, the maximum design pressures for normal, off-normal and accident conditions are 10 psig, 20 psig, and 65 psig, respectively,
    - For 61BTH Type 2 DSC, the maximum design pressures for normal, off-normal and accident conditions are 15 psig, 20 psig, and 120 psig, respectively,
- A total of *thirteen (13)* heat load zoning configurations (HLZCs) are allowed for the 61BTH DSCs as shown in *Figures 1-17 through 1-24 and 1-25a, 1-25b, 1-25d, 1-25e, and 1-25f of the Technical Specifications [1.TS]*.

A summary of the two 61BTH DSC configurations analyzed in this chapter is shown below:

DSC Type	61BTH Type 1 DSC		61BTH Type 2 DSC	
Maximum Heat Load (kW)	19.4	22.0	27.4	31.2
Heat Load Zoning Configuration(1)	#3, #4	#1, #2, #9	#8	#5, #6, #7, #10, *#11, #12, #13*
Neutron Absorber Material	Borated Aluminum or BORAL® or MMC	Borated Aluminum or MMC	Borated Aluminum or BORAL® or MMC	Borated Aluminum or MMC
Top Grid/ Holddown Ring	Original 61BT or Alternate Designs		Alternate Designs	
R45 Rail Design	Steel Rail and Aluminum Shim		Steel Rail and 5/8″ Aluminum/Shims (on 3 sides)	
R90 Rail Design	Steel Rail and Aluminum Shim		Aluminum Rail(3) and 1/4″ Steel Plate	
Transfer Cask	OS197, OS197H or OS197FC-B or OS200		OS197FC-B or OS200FC	
Storage Module	HSM(2) or HSM-H or HSM-HS		HSM-H or HSM-HS	

Notes:

(1) Refer to Section T.4.6.1 for a discussion on the applicability of Heat Load Zoning Configuration (HLZC) for each DSC type.

(2) Models 80, 102, 152, and 202

(3) Aluminum Rail may consist of plate and shims.

For the 61BTH Type 1 and Type 2 DSCs, sensitivity thermal analyses for the cask in vertical orientation when inside the fuel building at 120 °F without insolation are performed to determine bounding configuration for thermal analyses.

The comparison of the sensitivity analysis results shows that for both the 61BTH Type 1 and Type 2 DSCs, the fuel cladding and component temperatures for the borated aluminum neutron absorber configurations with higher heat loads bound the corresponding temperatures for the BORAL® neutron absorber configurations with lower heat loads. Therefore, to bound the analysis results, a complete set of thermal analyses are required only for the borated aluminum neutron absorber option with higher heat loads for both 61BTH Type 1 and Type 2 DSCs.

The thermal evaluations presented herein include steady state and transient analyses of the thermal response of the NUHOMS®-61BTH System components to a defined set of thermal loading conditions. These loading conditions envelope the thermal conditions expected during all normal, off-normal, and postulated accident loading, transfer and dry storage operations for the design basis thermal conditions as defined in Section T.2. The applicable allowable temperatures are presented and comparisons are made with calculated temperatures as the basis for acceptance.

The temperature profiles of the TCs and the 61BTH DSC shell and top and bottom cover plates and shield plugs obtained from the results of the OS197FC-B TC analysis with 19.4 kW, 22.0 kW, 27.4 kW, and 31.2 kW heat loads are used in thermal stress calculations.

T.4.5.3 OS197FC-B TC Thermal Model Results

The maximum temperature results for the 61BTH DSC shell assemblies and TC components during transfer are presented in Table T.4-7 through Table T.4-9. These results are for 31.2 kW and 22.0 kW heat loads. The DSC shell temperatures are then used as boundary conditions in the 61BTH DSC basket analysis presented in Section T.4.6.

This section presents the thermal evaluation of transfer operations for 61BTH DSC with HLZCs 1 through 8. Thermal evaluation of 61BTH DSC for HLZCs 9 and 10 is presented in Section T.4.6.10. *Thermal evaluation of 61BTH Type 2 DSC for HLZCs 11, 12 and 13 is presented in Section T.4.6.13.*

T.4.5.3.1 Normal and Off-Normal Conditions Results

Table T.4-7 presents the maximum steady state component temperatures for the configuration of the TC with a 61BTH Type 1 DSC with 22.0 kW and 19.4 kW of decay heat. All component temperatures are well below their associated maximum allowable limits. Figure T.4-14 illustrates the temperature distribution within the TC at steady-state conditions during vertical transfer operations with no insolation and 120 °F ambient.

Transient analyses are performed to determine the time limit for DSC transfer operations for 61BTH Type 2 DSC with a decay heat load higher than 22.0 kW up to 31.2 kW. The analyses assume that the transient analysis begins with water in the TC/DSC annulus and that with the TC in a vertical orientation (i.e., no credit is taken for heat transferred through the canister rails). At time = 0, the annulus water is assumed to be drained and the bolting of the TC top cover is initiated. This causes the system to heat up. Figure T.4-17 illustrates the predicted thermal response of the DSC and TC for this transient, assuming a decay heat load of 31.2 kW in a 61BTH Type 2 DSC. Figure T.4-17 also shows the steady state results of the same case. Based on targeted DSC shell temperatures of approximately 405 °F (for HLZC 7) and 445 °F (for HLZCs 5, 6 and 8) to avoid excessive fuel cladding temperatures, the transient analysis indicates that approximately 15 and 28 hours, respectively, are available to transfer the DSC into the HSM-H or take some other corrective actions. The anticipated corrective actions are:

- Complete the transfer of the DSC from the TC to the HSM-H, or
- Unbolt the TC top cover plate and flood the TC/DSC annulus with water if the TC is vertical, or
- Use of an external fan to circulate the air in the TC/DSC annulus if the TC is horizontal, or
- Return the TC to the TC handling area, unbolt the TC top cover plate and reflood the TC/DSC annulus with clean water.

forging exceeding the temperatures for a short duration. That justification is also applicable for this case.

The results of the DSC shell temperatures are used in the DSC basket and fuel cladding temperature models and the results documented in Section T.4.6 show that all the basket and fuel cladding material temperature limits are satisfied. The results of the TC temperatures are used in Section T.3 to show that thermal stresses in the TC are also within these allowables. Tables T.4-7 through T.4-11 list the DSC shell maximum allowable limit of 800 °F for long term. Similarly, Table T.4-11 also lists 1000 °F as the maximum allowable limit for short term. Note that these limits are not the thermal analyses limits. They are limits for the structural analyses in Section T.3 to ensure structural integrity for stainless steel components.

The allowable duration for the transfer operations (defined as from the time when the water in the TC-DSC annulus is drained to when the DSC is loaded into the storage module) will vary depending only on the DSC type and the heat load configuration. For simplicity of operations, a single time limit is used for all ambient conditions and TC orientations (i.e., longer times are available for the non-controlling conditions). The following table summarizes the permissible operational conditions:

Fuel Basket Type	DSC Heat Load Zoning Configuration	Transfer Time Limit (1), (2) (4)
Type 1	All configurations ($\leq$ 22 kW)	No time limit
Type 2	HLZC #1, 2,3, 4 and 9(5) ($\leq$ 22 kW)	No time limit
	HLZC #5, 6 ($\leq$ 31.2 kW)	26.0 Hours(3)
	HLZC #7, 10(5), *11, 12, 13(6)* ($\leq$ 31.2 kW)	13.0 Hours(3)
	HLZC #8 ($\leq$ 27.4 kW)	26.0 Hours(3)

Notes:

(1) Transfer time is defined as from the time when the TC DSC annulus water is drained to when the DSC is loaded into the storage module.

(2) The listed allowable transfer times are valid for all ambient conditions and TC orientations.

(3) Initiate recovery operations such as air circulation if the operation time exceeds the limit. Two hours is considered sufficient time to initiate the air circulation option.

(4) The transfer operation time limit is reset only if the transfer cask annulus is refilled with water.

(5) Thermal evaluation of 61BTH DSC for HLZCs 9 and 10 is presented in Section T.4.6.10.

(6) Thermal evaluation of 61BTH Type 2 DSC for HLZCs 11, 12 and 13 is presented in Section T.4.6.13.

T.4.5.5 Evaluation of OS200/OS200FC TC with 61BTH DSCs

This section presents the thermal evaluations which demonstrate that the NUHOMS® OS200/OS200FC TC meets the thermal requirements of 10CFR72 for the onsite transfer of spent fuel when used to transfer the 61BTH DSCs between the fuel building and the horizontal storage module (HSM/HSM-H/HSM-HS) at the ISFSI site. The OS200 TC is designed to passively remove the decay heat loading from the DSCs under normal, off-normal, and accident conditions while maintaining fuel cladding temperatures and DSC internal pressures within specified regulatory limits [4.15]. The applicable design criteria are listed in Section T.4.1.

The table below presents the 61BTH DSC types along with the maximum decay heat loads, HLZCs and the applicable transfer time limits for the OS200/OS200FC TC system.

61BTH DSCs in OS200/OS200FC TC

DSC Type	Maximum Heat Load, (KW)	HLZC #	Transfer Cask	Transfer Time Limit[1][2][4]
61BTH Type 1 DSC	19.4	3, 4	OS200/OS200FC	No time limit
	22	1,2, 9[5]		
61BTH Type 2 DSC	≤ 22	1,2,3, 4, 9[5]	OS200/OS200FC	
	27.4	8	OS200FC	26.0 hours[3]
	31.2	5, 6		26.0 hours[3]
		7, 10[5], *11, 12, 13*[6]	OS200FC	13.0 hours[3]

Notes:

(1) Transfer time is defined as from the time when the TC/DSC annulus water is drained to when the DSC is loaded into the storage module.

(2) The listed allowable transfer times are valid for all ambient conditions and TC orientations.

(3) Initiate recovery operations such as air circulation if the operation time exceeds the limit. Two hours is considered sufficient time to initiate the air circulation option.

(4) The transfer operation time limit is reset only if the transfer cask annulus is refilled with water.

(5) Thermal evaluation of 61BTH DSC for HLZCs 9 and 10 is presented in Section T.4.6.10.

(6) *Thermal evaluation of 61BTH Type 2 DSC for HLZCs 11, 12 and 13 is presented in Section T.4.6.13.*

T.4.5.6 Benchmarking of the OS200FC TC with the 32PTH1 DSC ANSYS Model to SINDA/FLUINT Model

Thermal Desktop® and SINDA/FLUINT models were used in the Appendix P, Section P.4.5, Appendix T, Section T.4.5 and Appendix U, Section U.4.5 of the UFSAR to analyze the thermal performance of the OS197/OS200 TCs with and without air circulation. In order to use ANSYS computer code to simulate the thermal performance with and without air circulation for the OS200/OS200FC TCs, respectively with the 61BTH DSCs, the ANSYS model is validated in this section by benchmarking it against the Thermal Desktop® and SINDA/FLUINT models. This benchmarked model provides the basis for the thermal analysis of the 61BTH DSC in the OS200/OS200FC TC using ANSYS model.

The OS200FC TC loaded with 32PTH1 DSC and heat loads of 31.2 kW and 40.8 kW are analyzed in Appendix U, Section U.4.5 with Thermal Desktop® and SINDA/FLUINT models and are considered for the benchmarking to envelope the conditions expected for the 61BTH DSC with a maximum heat load of 31.2 kW. The following criteria are considered for the maximum differences between ANSYS and SINDA/FLUINT models for the benchmarking purposes.

(1) ±5°F for the maximum DSC shell temperature.

(2) ±5% for the heat removed by air circulation.

(3) ±10°F for the air exit temperature.

T.4.5.6.1 Methodology for Thermal Evaluation of the OS200FC TC with the 32PTH1 DSC using ANSYS Model

T.4.6 NUHOMS®-61BTH DSC Thermal Analysis

The thermal analysis of the NUHOMS® 61BTH DSC is based on finite element models developed using the ANSYS computer code [4.22]. The methodology used is identical to that used for 24PTH DSC modeling described in Appendix P, Section P.4.6. ANSYS is a comprehensive thermal, structural and fluid flow analysis package. It is a finite element analysis code capable of solving steady state and transient thermal analysis problems in one, two or three dimensions. Heat transfer via a combination of conduction, radiation and convection can be modeled by ANSYS.

T.4.6.1 Heat Load Zoning Configurations

A total of *thirteen (13)* HLZCs are allowed for the 61BTH DSCs as shown in *Figures 1-17 through 1-24 and 1-25a, 1-25b, 1-25d, 1-25e, and 1-25f of the Technical Specifications [1.TS]*. The following table summarizes the different types of DSCs, the maximum total heat loads, and the different HLZCs that can be used for each DSC type.

61BTH DSC Type	**Neutron Absorber Type**	**Max Heat Load (kW)**	**HLZC 3 (19.4 kW)**	**HLZC 4 (19.4 kW**	**HLZC 1, 2, 9 (22.0 kW)**	**HLZC 8 (27.4 kW)**	**HLZC 5, 6, 7, 10, *11, 12, 13* (31.2 kW)**
Type1	Borated Aluminum/BORAL®/ MMC	19.4	√	√			
	Borated Aluminum/MMC	22.0	√	√	√		
Type 2	Borated Aluminum/BORAL®/ MMC	27.4	√	√	√	√	
	Borated Aluminum/MMC	31.2	√	√	√	√	√

Section T.4.6.1 through Section T.4.6.9 present the thermal evaluation of 61BTH DSC with HLZCs 1 through 8. Thermal evaluation of 61BTH DSC for HLZC 9 and 10 is presented in Section T.4.6.10. *Thermal evaluation of 61BTH Type 2 DSC for HLZCs 11, 12 and 13 is presented in Section T.4.6.13.*

The above table shows that the maximum decay heat loads for the Type 1 and the Type 2 DSCs are 22.0 kW, and 31.2 kW, respectively. The checked (√) box indicates the HLZC that is allowed for use in each DSC configuration.

Consider the following example: In a 61BTH Type 2 DSC with BORAL® neutron absorber, the maximum decay heat load is 27.4 kW. The HLZCs that can be used with this DSC are 3, 4, 1, 2, and 8. Since HLZCs 3, 4, 1, and 2 are bounded by HLZC 8, only HLZC 8 needs to be considered for bounding fuel loading analysis.

$$V_{DSC\ cavity} = \frac{\pi \cdot ID^2_{DSC\ shell}}{4 \cdot L_{DSC\ cavity}}$$

where

$ID_{DSC\ shell}$ - DSC shell inside diameter,

$L_{DSC\ \mathrm{cavity}}$ - DSC cavity length.

The calculated 61BTH DSC cavity free volumes are shown in Table T.4-29.

Quantity of Helium Fill Gas in DSC

The DSC free volume is filled with a maximum 3.5 psig (2.5 psig +1) of helium after vacuum drying operation. The average steady state helium backfill temperature is used for the calculation of the helium fill gas quantity. Using the ideal gas law, the quantity of helium in each type of DSC is calculated and the results are presented in Table T.4-15.

Quantity of Helium Fill Gas in Fuel Rods per DSC

The volume of the helium fill gas in a GE12/GE14 10x10 fuel assembly, at cold unirradiated conditions is 2.066 in^3, and there are a maximum of 92 fuel pins in a fuel assembly. The GE12/GE14 10x10 fuel assembly has bounding (highest) number of fuel rods per assembly, which results in the highest quantity of helium fill gas in fuel rods per DSC. The maximum fill pressure is 140 psig (155 psia) and the fill temperature is assumed to be room temperature (70°F or 530°R). The mole quantity of fuel rod fill gas is given by:

$$n_{he} = \frac{(155\,psia)(6894.8\,Pa/psi)(61 \cdot 92 \cdot 2.066 in^3)(1.6387 \cdot 10^{-5}\,m^3/in^3)}{(8.314 J/(mol \cdot K))(530°R)(5/9\,K/°R)}$$

$$n_{he} = 83.0\,g - moles$$

Based on NUREG 1536 [4.5], the maximum fraction of the fuel pins that are assumed to rupture and release their fill and fission gas for normal, off-normal and accident events is 1, 10 and 100%, respectively. The amount of helium fill gas released per DSC for each of these conditions is summarized in Table T.4-30. For all of these events, 100% of the fill gas in each ruptured rod is assumed to be released.

Quantity of Fission Gas released as a Result of Irradiation in Fuel Rods per DSC

The GE12/GE14 10x10 fuel assembly used in the pressure calculations is assumed to have a maximum burnup of up to 62 GWd/MTU, which is the highest burnup proposed for the NUHOMS®-61BTH system configuration. The maximum burnup creates a bounding case for the amount of fission gas produced in a fuel rod during reactor operation.

The total amount of gases released per FA due to irradiation is 20.1 g-moles at reactor discharge (0 year cooling period), and increases to 20.2 g-moles after 5 years cooling period. The longer cooling period results in a slightly higher amount of gases, primarily from the increase in helium due to alpha decay of actinides. As mentioned in Section T.2.1, the minimum cooling time for the FAs to be loaded in the 61BTH Type 2 DSCs is 1 year. The amount of fission gases released per FA from the 5 year cooling period bounds that from the 1 year cooling period. Therefore it is conservatively considered that 20.2 g-moles is the total amount of fission gases released per FA due to irradiation.

T.4.6.13 Thermal Evaluation of NUHOMS® 61BTH Type 2 DSC with HLZCs 11, 12 and 13

This section presents the thermal evaluation of 61BTH Type 2 DSC with HLZCs 11, 12 and 13. The HLZCs 11 through 13 are shown in Figure 1-25d through Figure 1-25f of the Technical Specifications [1.TS]. The HLZCs 11 through 13 allow a maximum heat load of 31.2 kW per DSC.

As discussed in Section T.4.6.10, the thermal evaluations presented in Section T.4.4 for storage conditions and Section T.4.5 for transfer operations are performed by considering the maximum heat load either as a heat flux on the radial inner surface of the DSC or as a volumetric heat generation rate applied over a homogenized basket. Because of this approach, the thermal evaluations presented in Section T.4.4 and Section T.4.5 are not dependent on the HLZC, but are dependent only on the maximum heat load per DSC. Since the maximum heat load considered for HLZCs 11, 12 and 13 is 31.2 kW and remains bounded by those previously evaluated in Sections 4.4 and 4.5, the DSC shell temperature profiles for storage evaluation in Section T.4.4 and transfer evaluation in Section T.4.5 remain applicable for HLZCs 11, 12 and 13. For transfer operations, the thermal performance of 61BTH Type 2 DSC was evaluated at 15 hours for HZLC 7 and 28 hours for HLZCs 5, 6 and 8 in Section T.4.5.3.1. Similar to HLZC 7, this evaluation for HLZCs 11 through 13 considers the DSC shell temperature profiles at 15 hours.

Since no other changes are considered to the 61BTH Type 2 DSC except for the HLZC, the thermal evaluation of the 61BTH DSC Type 2 with HLZCs 11, 12 and 13 is based on a sensitivity study of the normal hot storage condition with 100°F ambient and the vertical transfer condition with 120°F ambient, using the 61BTH Type 2 DSC thermal model used in Section T.4.6.

For HLZCs 11, 12 and 13, since the total heat load is limited to 31.2 kW, all zones cannot be fully loaded with the maximum defined heat load per FA. A study of the loading patterns in [4.29] concludes that for a given decay heat load in a cask, loading assemblies with a higher decay heat load in the outermost compartments will result in lower peak fuel cladding temperature. Based on this study, the peak cladding temperature is maximized if the heat load is concentrated in the inner core compartments.

Section T.4.6.13.1 presents the thermal evaluation for HLZCs 11, 12 and 13 wherein the FAs in the inner zones are loaded with the highest allowed heat load and the heat load for FAs in the outer zones is adjusted to maintain the maximum heat load of 31.2 kW. To provide additional assurance that this results in the bounding configuration, an additional evaluation is presented in Section T.4.6.13.2 wherein the FAs in the outer zones are loaded with the highest allowed heat load and the heat load in the inner zones is adjusted to maintain the maximum heat load of 31.2 kW. HLZC 11 is considered for this study since it allows the maximum per FA heat load of 1.7 kW in the outer zones.

Section T.4.6.13.3 presents the thermal evaluation of HLZCs 11, 12 and 13 with damaged and failed FAs.

T.4.6.13.1 *Thermal Evaluation of HLZCs 11, 12 and 13 with Maximum Payloads in Inner Zones*

HLZCs 11, 12 and 13 can be loaded with a maximum heat load of 31.2 kW. As discussed above, the payload per FA in each zone is adjusted to ensure the total decay heat load per DSC equals 31.2 kW. In this section, the HLZC inner zones are loaded with maximum payloads while adjusting the payload of the outer zones as shown in the following table to obtain the bounding fuel cladding temperatures.

Bounding HLZCs with Highest Payloads in Inner Zones

Zone #	FA Decay Heat (kW)	No. of FA	Zone Decay Heat (kW)
HLZC 11 (Zone 1, 2, 4 and 5 are Inner Zones with maximum allowable heat load per FA, Zone 3 and 6 are Outer Zones with adjusted heat load per FA)			
Zone 1	0.393	9	3.54
Zone 2	0.393	20	7.86
Zone 3	0.63	6	3.78
Zone 4	0.7	12	8.4
Zone 5	0.48	8	3.84
Zone 6	0.63	6	3.78
Total Heat Load (kW)			31.2
HLZC 12 (Zone 1, 2, 4 and 5 are the Inner Zones with maximum allowable heat load per FA, Zone 3 is the Outer Zone with adjusted heat load per FA)			
Zone 1	0.3	9	2.7
Zone 2	0.393	20	7.86
Zone 3	0.5	12	6
Zone 4	0.9	12	10.8
Zone 5	0.48	8	3.84
Total Heat Load (kW)			31.2
HLZC 13 (Zone 1, 2 and 4 are the Inner Zones with maximum allowable heat load per FA, Zone 3 is the Outer Zone with adjusted heat load per FA)			
Zone 1	0.3	23	6.9
Zone 2	0.5	16	8
Zone 3	0.2875	8	2.3
Zone 4	1	14	14
Total Heat Load (kW)			31.2

The following table compares the maximum fuel cladding and DSC component temperatures for the 61BTH Type 2 DSC with HLZCs 11, 12 and 13 to the design basis values for HLZC 7 from Section T.4.6 during normal hot storage condition (DSC in HSM, 100 °F ambient) and during transfer condition (DSC in OS197FC-B TC, 120°F ambient, @ 15 Hrs). Typical temperature plots for HLZCs 11 through 13 during the normal storage condition and vertical storage condition are presented in Figure T.4-42 through Figure T.4-47.

Maximum Component Temperatures for 61BTH Type 2 DSC for Normal Storage in 100 °F Ambient

	DSC in HSM, 100 °F Ambient (°F)					
HLZC	**Fuel Cladding**	**Fuel Compartment**	**Al/Poison Plate**	**R45/R90 Rails**	**Top Grid**	**DSC Shell**
Design Basis [Tables T.4-12 and T.4-14]	719	690	689	514	496	434
HLZC 11	685	660	660	512	484	434
$\Delta T = (T_{HLZC\ 11} - T_{Design\ basis})$	-34	-30	-29	-2	-12	0
HLZC 12	665	645	645	512	480	434
$\Delta T=(T_{HLZC\ 12} - T_{Design\ basis})$	-54	-45	-44	-2	-16	0
HLZC 13	685	651	651	515	483	434
$\Delta T=(T_{HLZC\ 13} - T_{Design\ basis})$	-34	-39	-38	1	-13	0

Maximum Component Temperatures for 61BTH Type 2 DSC for Vertical Transfer, 120 °F Ambient

	DSC in TC, 120 °F Ambient, Vertical Transfer (°F)					
HLZC	**Fuel Cladding**	**Fuel Compartment**	**Al/Poison Plate**	**R45/R90 Rails**	**Top Grid**	**DSC Shell**
Design Basis [See Section T.4.6.10.2.1 for HLZC 7]	730	701	701	522	493	408
HLZC 11	696	672	671	520	481	408
$\Delta T = (T_{HLZC\ 11} - T_{Design\ basis})$	-34	-29	-30	-2	-12	0
HLZC 12	676	657	657	521	477	408
$\Delta T=(T_{HLZC\ 12} - T_{Design\ basis})$	-54	-44	-44	-1	-16	0
HLZC 13	693	663	663	524	480	408
$\Delta T=(T_{HLZC\ 13} - T_{Design\ basis})$	-37	-38	-38	2	-13	0

A comparison of the maximum fuel cladding temperature of 713℉ in Figure T.4-33 for Normal Storage, 100°F Insolation and 728℉ in Figure T.4-34 for Vertical Loading @ 28 Hr. to the design basis temperature of 719°F and 734°F, respectively in Table T.4-12 shows that the maximum reported fuel cladding temperatures are increased by 6°F. This increase is based on an evaluation to allow shims (a maximum of 6 plates/sheets) per Note 4 of Drawing NUH61BTH-2002-SAR. The same increase is also applied to the results presented in this evaluation of HLZCs 11 through 13 as seen from the difference in the maximum temperature reported within Figure T.4.-42 through Figure T.4-47 and the above table.

As shown in the above tables, maximum fuel cladding temperature of 696°F is reported for HLZC 11 during vertical transfer operations with sufficient margin to the temperature limit of 752°F and is 34°F lower compared to the design basis analysis. Maximum fuel cladding temperatures of 685°F is reported for HLZC 11 during storage conditions and it is also 34°F lower compared to the design basis analysis evaluated in Section T.4.6. The component temperatures for HLZCs 11, 12 and 13 always remain bounded by the design basis analysis evaluated in Section T.4.6, except for the R45/R90 rails temperature for HLZC 13. Minor temperature increases of 2°F and 1°F for HLZC 13 observed for the R45/R90 rails during transfer and storage conditions, respectively are insignificant and do not affect the thermal or structural performance of the 61BTH Type 2 DSC.

For normal/off-normal transfer operations, as discussed in Section T.4.6.13, HLZCs 11, 12 and 13 are evaluated at 15 hours similar to HLZC 7. Therefore, the maximum temperatures in the above table are compared to the maximum temperatures from HLZC 7 instead of the maximum temperatures listed for transfer operation in Tables T.4-12 and T.4-14 which are based on HLZC 5 at 28 hours. Similarly, comparing the maximum fuel cladding temperature of 696°F for HLZC 11 at 15 hours to the design basis maximum fuel cladding temperature of 734°F in Table T.4-12 shows a decrease of 38°F (versus 34°F when compared to HLZC 7 at 15 hours).

For both storage and transfer conditions, the maximum temperatures for the fuel cladding and fuel compartment decrease significantly when compared to the design basis temperatures due to the maximum allowable heat load of the inner zones. A review of Figure 1-23 [1.TS] for HLZC 7 shows that the inner zones are at a heat load of 0.48 kW compared to a maximum of 0.393 kW in HLZC 11 and 0.3 kW in HLZCs 12 and 13.

Based on this evaluation, the maximum fuel cladding temperatures for the 61BTH Type 2 DSC reported in Table T.4-12, Table T.4-17 and Table T.4-21 for normal, off-normal and accident conditions remain valid for the 61BTH Type 2 DSC with HLZCs 11, 12 and 13. Therefore, the time limits for transfer operations determined for a 61BTH Type 2 DSC with HLZC 7 in Section T.4.5.4 are also applicable to a 61BTH Type 2 DSC with HLZCs 11, 12 and 13.

In addition, since the maximum temperatures are lower, the average helium and fuel cladding temperatures for the design basis analysis from Section T.4.6 remain bounding for HLZC 11 through 13. Therefore, it is concluded that there is no adverse effect on the internal pressure evaluation in Section T.4.6.6.4 and the design basis pressures reported in Table T.4-16, Table T.4-20 and Table T.4-24 for normal, off-normal and accident conditions remain applicable.

T.4.6.13.2 Thermal Evaluation of HLZCs 11, 12 and 13 with Maximum Payloads in Outer Zones

As discussed in Section T.4.6.13 for HLZCs 11, 12 and 13, the payload per FA in each zone is adjusted to ensure that the total decay heat load remains below 31.2 kW. This section presents a sensitivity analysis with highest heat loads in the outer zones to evaluate the impact of basket component temperatures when the outer zones are loaded with the maximum heat load and the inner zones are adjusted to satisfy the total heat load limit of 31.2 kW. A review of the HLZCs 11 through 13 in Figure 1-25d through Figure 1-25f of the Technical Specifications [1.TS], HLZC 11 is identified to have the highest heat loads in the outer zones. Thus, an alternate configuration of HLZC 11 with maximum heat loads in the outer zones as shown in the following table is evaluated to find the bounding basket component temperatures.

Alternate HLZC 11 with Highest Payloads in Outer Zones

Zone #	***FA Decay Heat (kW)***	***No. of FA***	***Zone Decay Heat (kW)***
HLZC 11 (Zone 1, 2 and 5 are Inner Zones with adjusted heat load per FA, Zone 3, 4 and 6 are Outer Zones with maximum heat load per FA)			
Zone 1	*0*	*9*	*0*
Zone2	*0*	*20*	*0*
Zone 3	*1.6*	*6*	*9.6*
Zone 4	*0.7*	*12*	*8.4*
Zone 5	*0.375*	*8*	*3*
Zone 6	*1.7*	*6*	*10.2*
Total Heat load (kW)	*31.2*		

Because there is no change to the total heat load of the DSC, the same limiting cases for transfer operation and storage conditions identified in Section T.4.6.13 are re-evaluated with the alternate configuration of HLZC 11.

Maximum basket component temperatures for the bounding storage and transfer conditions in comparison to the design basis values are listed below.

Maximum Component Temperatures for 61BTH Type 2 DSC for Normal Storage in 100 °F Ambient

	DSC in HSM, 100 °F Ambient (°F)					
HLZC	***Fuel Cladding***	***Fuel Compartment***	***Al/Poison Plate***	***R45/R90 Rails***	***Top Grid***	***DSC Shell***
Design Basis [Tables T.4-12 and T.4-14]	*719*	*690*	*689*	*514*	*496*	*434*
Alternate HLZC 11	*646*	*545*	*544*	*505*	*434*	*434*
$\Delta T=(T_{Alternate,\ 11} - T_{Design\ basis})$	*-73*	*-145*	*-145*	*-9*	*-62*	*0*

Maximum Component Temperatures for 61BTH Type 2 DSC for Vertical Transfer, 120 °F Ambient

	DSC in TC, 120 °F Ambient, Vertical Transfer (°F)					
HLZC	***Fuel Cladding***	***Fuel Compartment***	***Al/Poison Plate***	***R45/R90 Rails***	***Top Grid***	***DSC Shell***
Design Basis [See Section T.4.6.10.2.1 for HLZC 7]	*730*	*701*	*701*	*522*	*493*	*408*
HLZC 11	*657*	*556*	*555*	*515*	*430*	*409*
$\Delta T=(T_{Alternate,\ 11} - T_{Design\ basis})$	-73	-145	-146	-7	-63	1

As shown in the above table, even with the highest payloads in the outer zones, the basket rail temperatures still remain bounded by the design basis temperatures. Thus, it is concluded that HLZCs 11 through 13 have no adverse impact on the design basis temperatures reported in Section T.4.6.

T.4.6.13.3 Thermal Analysis of HLZCs 11, 12 and 13 with Damaged and Failed Fuel Assemblies

As seen from Figure 1-25 [1.TS] up to 16 damaged (and remaining intact) or up to 4 failed (plus up to 12 additional damaged and balance intact) BWR FAs can be loaded per HLZCs 11, 12 and 13.

Intact and Damaged FA Only

HLZCs 11, 12 and 13 only allow up to a maximum of 16 damaged FAs with the remaining intact compared to a maximum of up to 61 damaged FAs in HLZCs 1 through 10. Section T.4.6.11 presents the thermal evaluation of up to 61 damaged FAs based on HLZC 7.

As discussed in Section T.4.6.11, the damaged FAs maintain their structural integrity during normal and off-normal conditions, and therefore there is no reconfiguration of the heat generating regions. Therefore, the analysis performed for HLZCs 11, 12 and 13 during normal and off-normal conditions with intact FAs will be bounding for damaged FAs during normal and off-normal conditions of storage and transfer. However, during postulated drop accident condition, the worst possible scenario is that the damaged FAs turning into rubble at the bottom of the DSC.

To account for any possible combination of intact/damaged fuel assemblies, Section T.4.6.11 evaluated various combinations of intact and damaged FAs including 45 intact/16 damaged FAs during accident conditions of transfer to evaluate the impact on the maximum fuel cladding temperature of the surrounding intact fuel assemblies based on the bounding HLZC 7 with maximum heat load of 0.54 kW per damaged FA. Based on the results presented in Section T.4.6.13.1, HLZC 7 remains bounding for HLZCs 11, 12 and 13. Therefore, no additional evaluation is required when loading up to 16 damaged FAs with the remaining intact FAs in HLZCs 11, 12 and 13 and the thermal evaluation in Section T.4.6.11 with 61 damaged FAs remains bounding.

Intact, Damaged and Failed FA

In addition to damaged FA, failed FA may also be stored per HLZCs 11, 12 and 13 with intact FAs. As seen from Figure 1-25 [1.TS], up to 4 failed (plus up to 12 additional damaged and balance intact) BWR FAs can be loaded. Section T.4.6.9 presents an evaluation for a similar configuration based on the bounding HLZC 7.

For normal and off-normal conditions, Section T.4.6.9 based on evaluation for the 61BT DSC in Appendix K, Section K.4.8 concludes that since the damaged and failed FAs are loaded in the outermost fuel compartment cells, they have negligible impact on intact fuel and basket components maximum temperatures for normal/off-normal conditions of storage and transfer.

For accident conditions, Section T.4.6.9 performed a thermal evaluation based on the bounding HLZC 7 and concludes in Section T.4.6.9.4 that there is no negative impact on the peak fuel cladding and DSC component temperatures when compared with the 61BTH DSC loaded with all intact FAs.

Since HLZC 7 remains bounding and because the same locations are considered in HLZCs 11, 12 and 13 for loading failed/damaged FAs along with intact FAs as evaluated in Section T.4.6.9, no additional analysis is required.

T.4.6.13.4 Impact of Alternate Material for Aluminum Base Plates and Increased Corner Gaps around R45 Transition Rails on HLZCs 11, 12 and 13

Section T.4.6.12 evaluates the impact of the following design changes:

1. *Increased corner gap between the R45 rail aluminum based plates from 0.10" and 0.20" (Figure T.4-27, Details C and D) to 0.35".*
2. *Allowing an option to fabricate Item 20/21 of Drawing NUH61BTH-2002-SAR from Aluminum 6061 compared to Aluminum 1100. See Note 29 of Drawing NUH61BTH-2002-SAR.*

Based on the evaluation in Section T.4.6.12 and the results presented in Section T.4.6.12.3, a maximum increase of 5°F is observed for the fuel cladding temperatures. Considering the same increase, the maximum fuel cladding temperature for HLZCs 11, 12 and 13 is 701°F (= 696°F + 5°F) and is significantly below the maximum temperature limit of 752°F.

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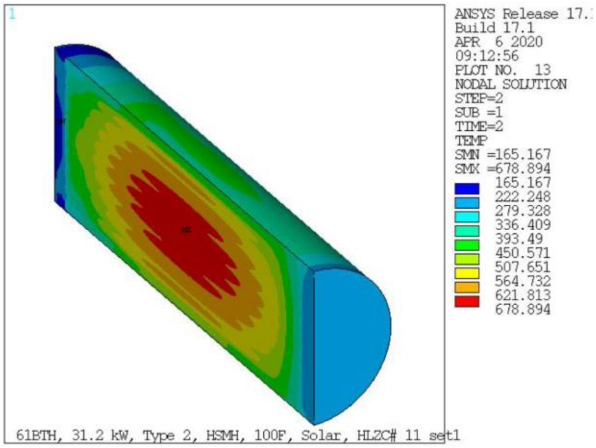
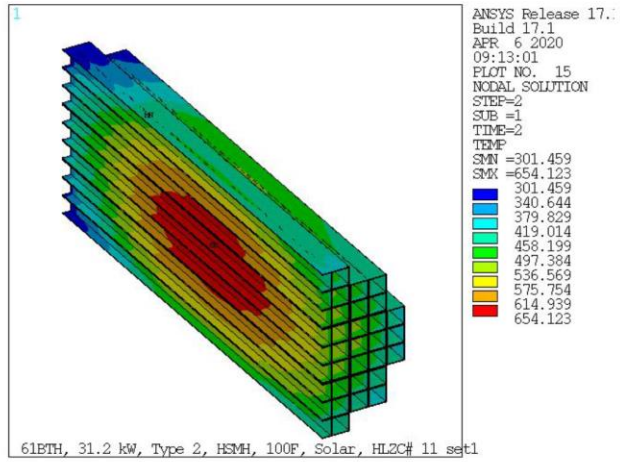
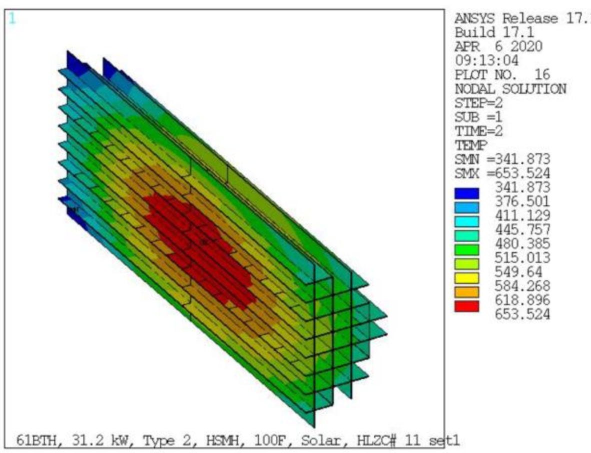


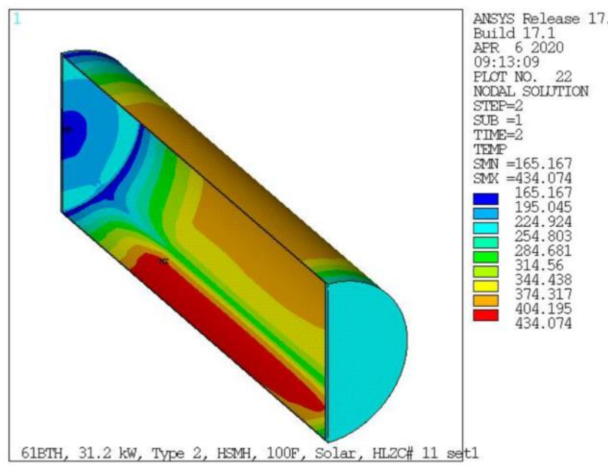
61BTH Type 2 DSC



Fuel Compartment

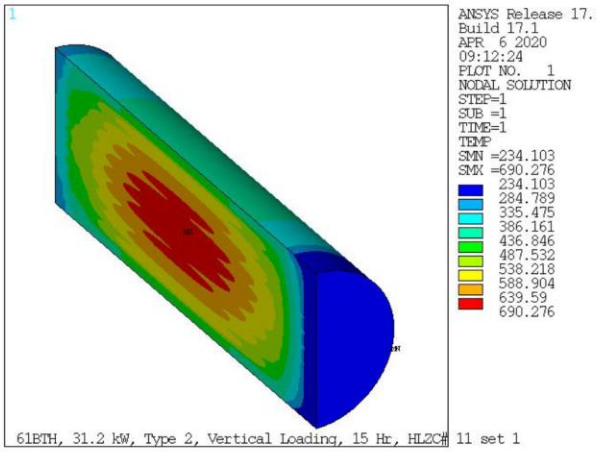


Neutron Absorber Plate

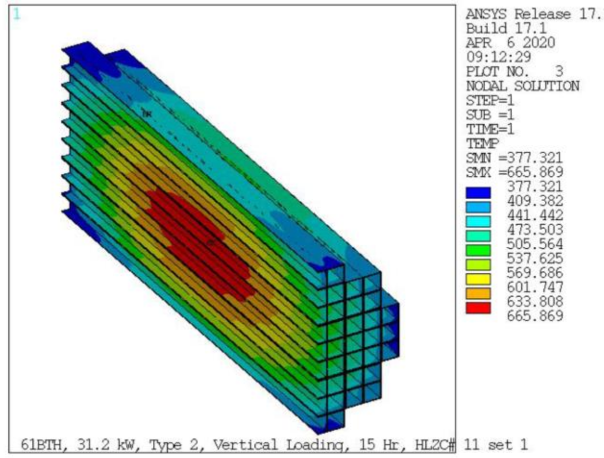


DSC Shell

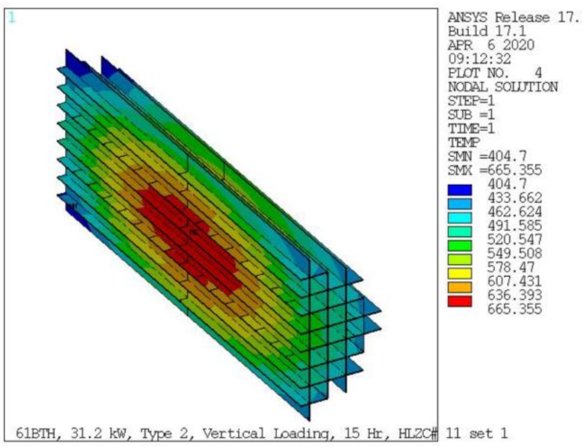
Figure T.4-42
Typical Temperature Plots for 61BTH DSC, Type 2 Basket
(Normal Storage @ 100°F, HLZC #11, 31.2 kW, Solar Insolation)



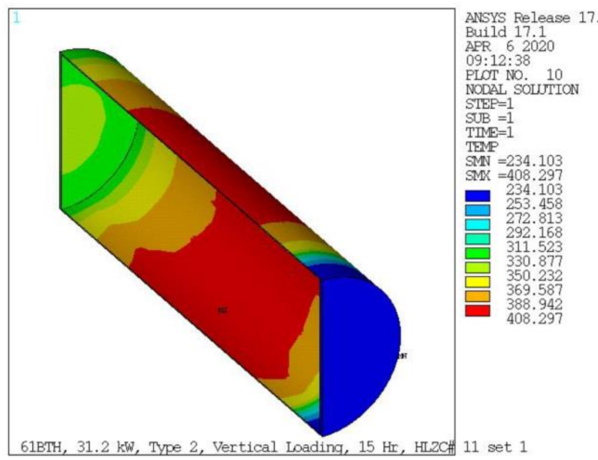
61BTH Type 2 DSC



Fuel Compartment

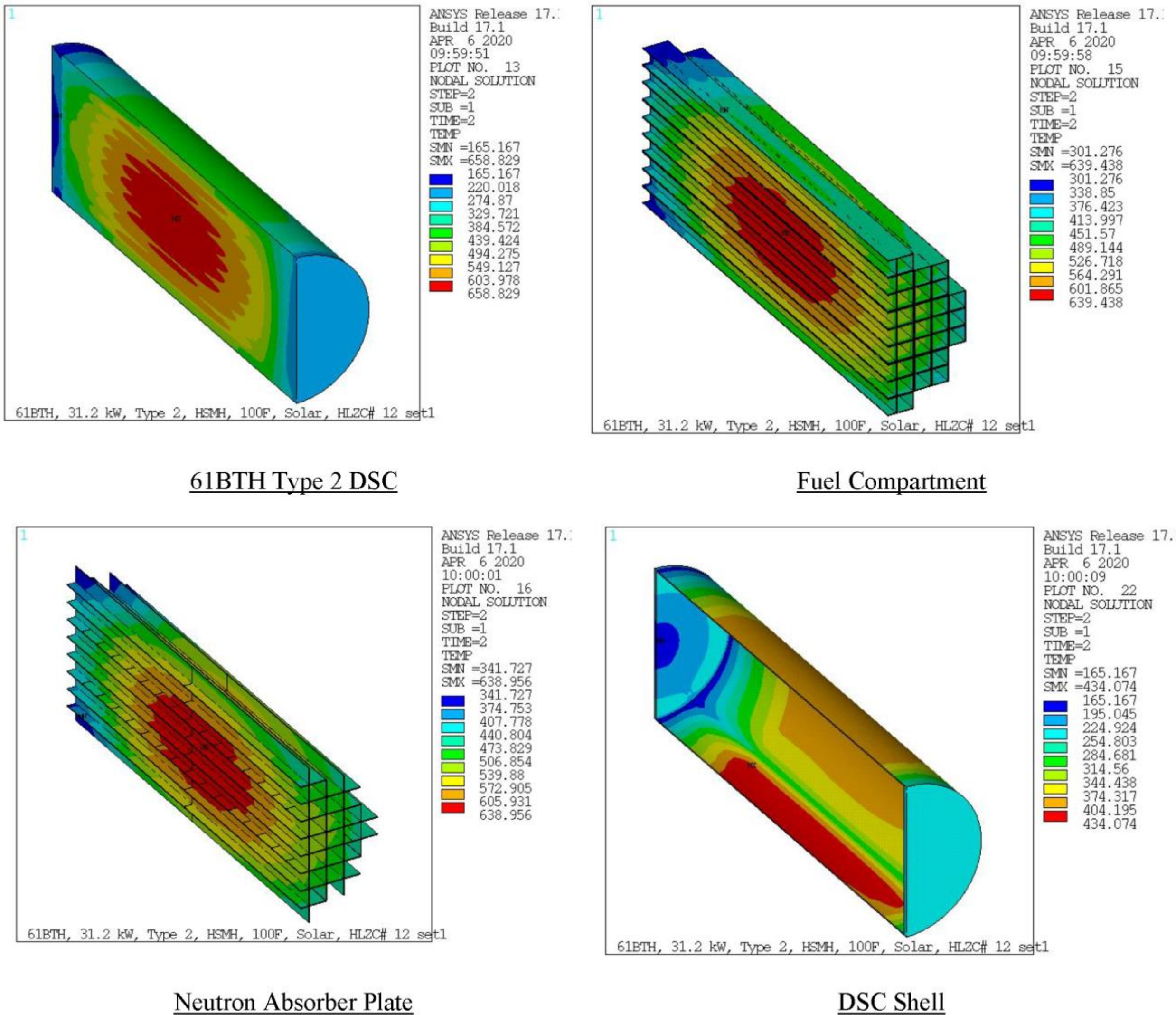


Neutron Absorber Plate



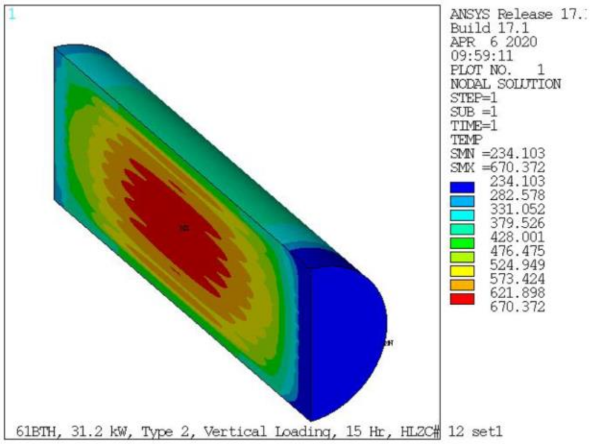
DSC Shell

Figure T.4-43
Typical Temperature Plots for 61BTH DSC, Type 2 Basket
(Vertical Transfer @ 120°F, HLZC #11, 31.2 kW, No Solar Insolation, 15 hr transient)

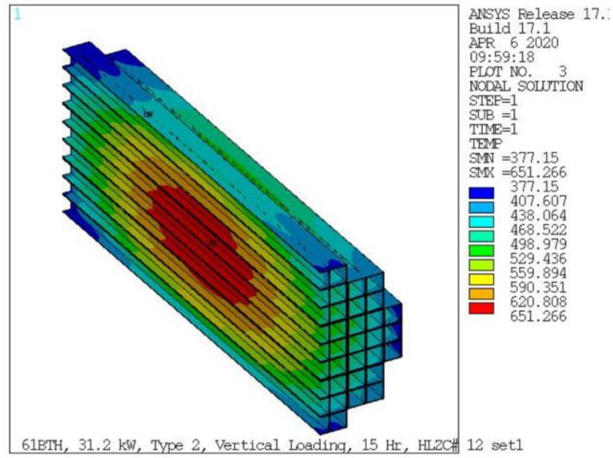


61BTH Type 2 DSC | Fuel Compartment | Neutron Absorber Plate | DSC Shell

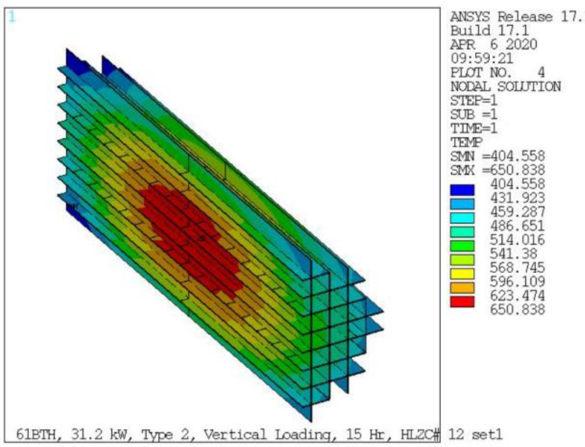
Figure T.4-44
Typical Temperature Plots for 61BTH DSC, Type 2 Basket
(Normal Storage @ 100°F, HLZC #12, 31.2 kW, Solar Insolation)



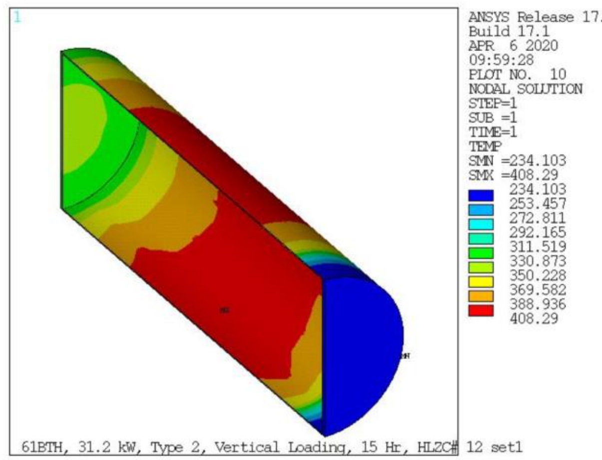
61BTH Type 2 DSC



Fuel Compartment

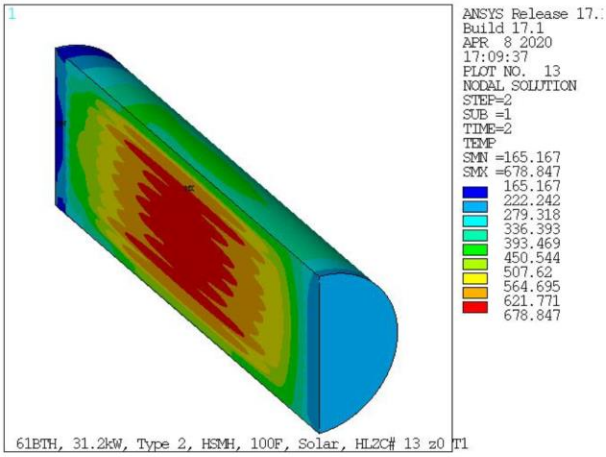


Neutron Absorber Plate

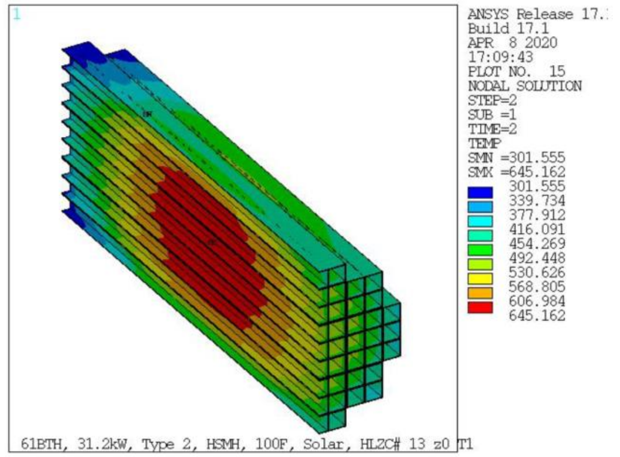


DSC Shell

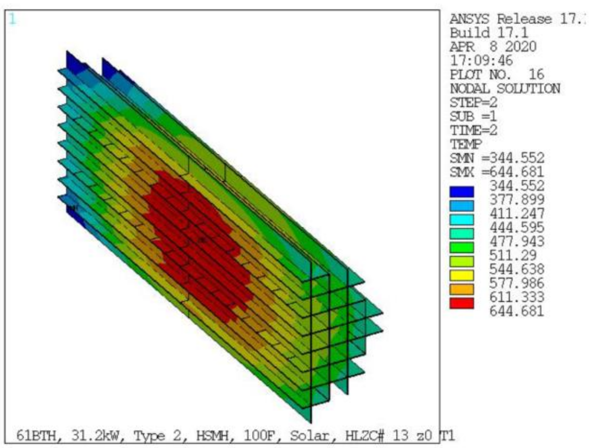
Figure T.4-45
Typical Temperature Plots for 61BTH DSC, Type 2 Basket
(Vertical Transfer @ 120°F, HLZC #12, 31.2 kW, No Solar Insolation, 15 hr transient)



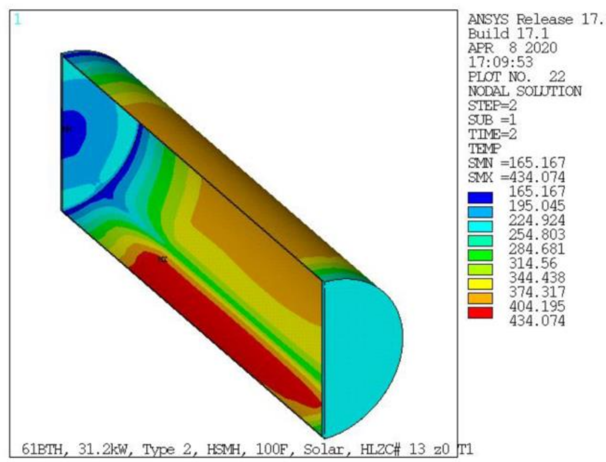
61BTH Type 2 DSC



Fuel Compartment

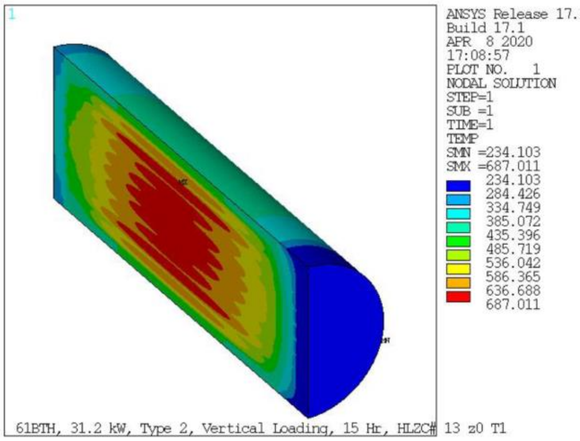


Neutron Absorber Plate

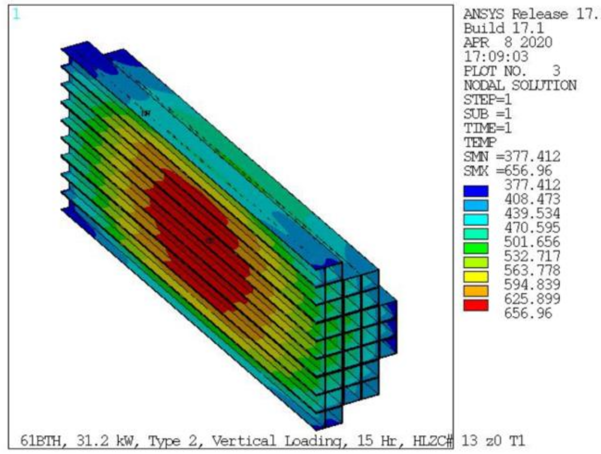


DSC Shell

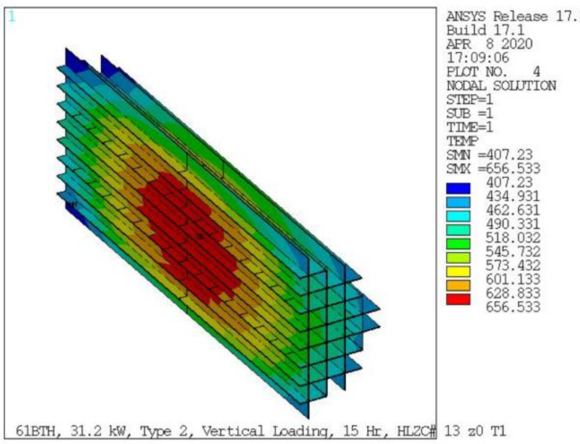
Figure T.4-46
Typical Temperature Plots for 61BTH DSC, Type 2 Basket
(Normal Storage @ 100°F, HLZC #13, 31.2 kW, Solar Insolation)



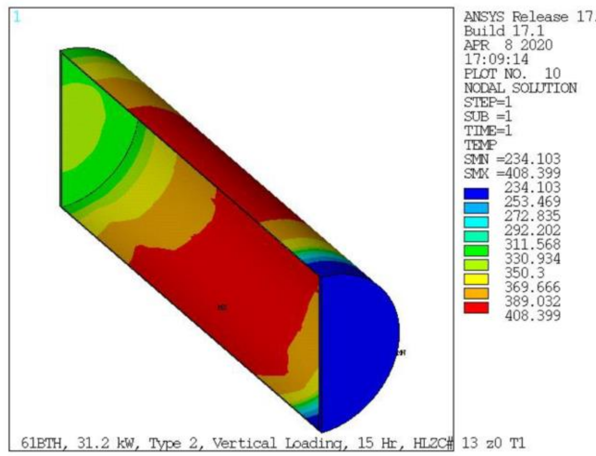
61BTH Type 2 DSC



Fuel Compartment



Neutron Absorber Plate



DSC Shell

Figure T.4-47
Typical Temperature Plots for 61BTH DSC, Type 2 Basket
(Vertical Transfer @ 120°F, HLZC #13, 31.2 kW, No Solar Insolation, 15 hr transient)

T.5 Shielding Evaluation

This chapter specifically addresses the shielding evaluation of the NUHOMS® 61BTH System with design basis BWR fuel loaded in a NUHOMS®-61BTH DSC. 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 UFSAR. The NUHOMS®-61BTH System consists of the NUHOMS® horizontal storage module (HSM) and HSM-H, the OS197 transfer cask (TC), and the 61BTH Type 1 and the Type 2 DSCs.

Site dose and occupation exposure analyses are described in Chapter T.10. Accident dose rate analyses are described in Chapter T.11.

The 61BTH DSC will be transferred using either the OS197/OS197H or a modified version of the OS197FC TC (OS197FC-B) if air circulation is required. The NUHOMS® 61BTH System will use either the *Standardized* HSM *(up to 22 kW/DSC)*, or the HSM-H (up to 31.2 kW/DSC) for storage.

The radiation shielding evaluation described below is for the NUHOMS® 61BTH Type 1 and Type 2 DSCs loaded in a NUHOMS® System TC. For heat levels below 22 kW *(i.e., HLZC 1, 2, 3, 4, and 9),* both DSC types can be transferred in any of the 3 transfer casks: 1) OS197, 2) OS197H, and 3) OS197FC-B. *The 61BTH Type 1 DSC may be* stored in any of these HSMs: 1) HSM Model 80, 2) HSM Model 102, 3) HSM Model 152, 4) HSM Model 202, and 5) HSM-H. *The HSM Model 80, HSM Model 102, HSM Model 152, and HSM Model 202 are collectively known as the Standardized HSM and are limited to 22 kW (i.e., HLZC 1, 2, 3, 4, and 9).* For heat loads exceeding 22 kW, the 61BTH Type 2 DSC must be used and can only be stored in HSM-H and transferred in the OS197FC-B TC. With respect to shielding performance, seven possible loading combinations are considered as listed below:

(1) 61BTH Type 1/2 DSC → OS197 (bounded by #3)
(2) 61BTH Type 1/2 DSC → OS197H (bounded by #3)
(3) 61BTH Type 1 DSC → OS197FC-B (bounds OS197/OS197H and Type 2 DSC)
(4) 61BTH Type 2 DSC → OS197FC-B (bounded by #3)
(5) 61BTH Type *1* DSC → HSM Model 80
(6) 61BTH Type *1* DSC → HSM Model 102*/152/202*
(7) 61BTH Type 1/2 DSC → HSM Model HSM-H

The design features of the HSM result in the occupational and site dose rates being as low as reasonably achievable (ALARA).

The NUHOMS® 61BTH DSC can also be stored within an upgraded HSM model, designated as HSM-HS as described in Appendix U of the UFSAR. From a shielding standpoint, the HSM-HS-module is identical to the HSM-H module. Therefore, all calculations performed with the HSM-H are applicable to the HSM-HS.

The NUHOMS® 61BTH DSC is also transferred in a modified version of the OS200 TC as described in Appendix U of the UFSAR. The OS200 TC is fitted with an aluminum sleeve to accommodate the smaller diameter 61BTH DSC.

The basket layout for Type 1 and Type 2 DSC configurations is identical except for the basket transition rails. Each DSC configuration is designed to store up to 61 intact BWR fuel assemblies or up to 61 damaged fuel assemblies in accordance with Figure T.2-9. For shielding purposes, radiological sources related to the 61BTH Type 2 bound the 61BTH Type 1 because assemblies with higher neutron and gamma sources are loaded in the outer zones. The presence of solid aluminum rails that fill the space between the peripheral fuel compartments and the DSC shell results in a more effectively shielded configuration for the Type 2 DSC. When such bounding neutron and gamma sources are placed in a Type 1 DSC, HSM and TC dose rates are bounding for all the shielding configurations. Therefore, the *HSM-H and OS197FC-B* shielding evaluation presented herein is performed for the hypothetical shielding configuration where radiological source terms bounding for *the* Type 2 DSC are analyzed with the Type 1 DSC *geometry*.

The NUHOMS® 61BTH Type 1 DSC is identical to the NUHOMS® 61BT DSC analyzed in UFSAR Appendix K, except for an optional redesigned basket hold-down ring. Relative to the existing 61BT DSC, the 61BTH Type 1 DSC allows for an increase in heat load from 18.3 kW to 22 kW, increase in maximum burnup from 40,000 MWd/MTU to 62,000 MWd/MTU, and an increase in maximum initial fuel enrichment from 4.4 wt. % U-235 to 5 wt. % U-235.

The 61BTH Type 2 DSC is also based on the basket design for the 61BT DSC with modifications to the shell assembly (cover plate thicknesses are increased to handle higher internal pressures) and to the basket transition rails to allow storage of fuel assemblies with a total heat load of up to 31.2 kW, with burnup of up to 62,000 MWd/MTU and with maximum initial fuel enrichments of up to 5 wt. % U-235. The Type 2 basket also incorporates the redesigned hold down ring.

The OS197FC-B is essentially the same as the OS197FC except that the lid and bottom have been modified to introduce air cooling design features to accommodate a higher decay heat load (>22 kW). 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 the shielding analysis, OS197FC-B TC is used to bound the OS197/OS197H 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.

There are a total of *13* possible heat load zoning configurations (HLZCs) for the 61BTH Type 1 and Type 2 DSCs. *Eight* out of *13* total DSC HLZCs are for Type 2 DSC only. The remaining five can be used with either DSC type; however, certain restrictions apply for Type 1 DSC in some cases. DSC HLZCs are depicted in *the Technical Specifications (TS) [1.TS], Figure 1-17 through Figure 1-24 (HLZC 1 through 8), Figure 1-25a and Figure 1-25b (HLZC 9 and 10), and Figure 1-25d through Figure 1-25f (HLZC 11 through 13).*

For loading of the 61BTH Type 1 DSC in a Standardized HSM, the fuel qualification tables *(FQTs)* shown in Table T.2-5 through Table T.2-10, Table T.2-*17 and* Table T.2-18 are developed for the design basis heavy metal loading of 0.198 MTU with SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] and for the heavy metal loading of 0.170 MTU with SAS2H/ORIGEN-S modules of SCALE 5.0 [5.20]. *Only the 0.54 kW FQT, which is the maximum fuel assembly heat load of the 61BTH Type 1 DSC and is located in a peripheral location, is a TS requirement, TS Table 1-4e. [1.TS]. The remaining FQTs in Chapter T.2 are provided "for information only."* Section 7.2.3.2 of Chapter 7 provides the methods for determining minimum required cooling times using fitting equations or linear interpolation for a given MTU between 0.170 MTU and 0.198 MTU. Also, Section T.5.2 provides the methods for determining the minimum required cooling times for combinations of burn-up and enrichments in "Not Analyzed" domain of the FQTs. Cells relevant to this domain are empty and shaded in the FQTs.

For the 61BTH DSC to be stored in an HSM-H, a single bounding FQT is provided in TS Table 1-4f [1.TS]. A simplified methodology is developed for the HSM-H FQT, and FQT interpolation/extrapolation is not needed.

The NUHOMS®-61BTH DSCs are designed to store BWR fuel assemblies with the characteristics described in Table T.2-1 and Table T.2-2, and associated tables and figures of Chapter T.2. The NUHOMS® 61BTH Type 2 DSC is also designed to store up to four failed fuel assemblies in the corner locations of the basket. Each failed fuel assembly is housed inside a failed fuel canister prior to loading in these designated positions within the basket.

The design basis BWR fuel source terms are derived from a bounding "generic" fuel assembly. The parameters of the bounding "generic" fuel assembly are selected in such a way that the resulting radiological and decay heat source terms bound those from all other fuel assembly types that are authorized for loading into the NUHOMS® 61BTH DSC. This "generic" fuel assembly shares many common features with the GE-2,3 7x7 Type G2A assembly *and is used to bound* the 8x8, 9x9, 10x10, and 11x11 fuel assemblies which are also authorized for loading into the NUHOMS®-61BT DSC. Its parameters are described in Section T.5.2. In addition, the maximum Co-59 content of each hardware region for the bounding fuel assembly type is used to determine the activation source for each fuel assembly region.

Maximum decay heat allowed for the 61BTH Type 1 DSC is 22 kW and a maximum heat load of 31.2 kW is allowed for the Type 2 DSC. The HLZCs to be used for the 61BTH Type 1 DSC are shown in *TS Figure 1-17 through Figure 1-20 (HLZC 1 through 4), and TS Figure 1-25a (HLZC 9) [1.TS]*. Fuel assemblies loaded in 61BTH Type 2 DSC can have a maximum decay heat of *1.7* kW per assembly. 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 that results in a maximum calculated dose rate on the surface of the HSM and/or TC side because the highest source fuel assemblies are on the outer periphery of the basket region where self-shielding due to adjacent assemblies is limited.

Source terms developed for the 61BTH Type 1 DSC are based on the bounding heat load configuration depicted in Figure T.5-1 and are used only in the Standardized HSM (HSM Model 80 and Model 102) analyses. These source terms are provided in Table T.5-10, Table T.5-16, and Table T.5-18. HSM Model 102 and Model 80 dose rates are provided in Table T.5-2 and Table T.5-3, respectively.

Source terms developed for the 61BTH Type 2 DSC are based on the bounding heat load configuration depicted in Figure T.5-1a and are used only in the HSM-H and OS197FC-B analyses. These source terms are provided in Table T.5-18a through Table T.5-18g. HSM-H dose rates are provided in Table T.5-1, and OS197FC-B dose rates are provided in Table T.5-4 and Table T.5-5.

Reconstituted and/or damaged fuel is also acceptable for the DSC payload. Reconstituted fuel may contain up to 10 *irradiated* solid stainless steel rods *per fuel assembly, or 40 rods per DSC*. 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 pronounced effect on the dose rates that are due to assemblies with cooling times less than 10 years. An additional 5 years of cooling time is required if reconstituted fuel with steel rods that are irradiated are present. Under normal conditions 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 damaged fuel.

The shielding evaluations for loading and transfer configurations documented herein are based on the OS197FC-B TC and are bounding for the OS200 TC. This is due to the fact that the neutron and gamma shielding material thicknesses are slightly higher for the OS200 TC.

Further, the aluminum sleeve employed to accommodate the smaller diameter 61BTH DSC within the OS200 TC also provides for slightly enhanced gamma shielding. Therefore, no additional shielding calculations are necessary for the OS200 TC.

The fuel qualification requirements for failed fuel assemblies limits the maximum heat load of failed fuel assemblies to *0.54 kW* per assembly *[1.TS]*. Therefore, the shielding evaluation for the basket containing failed fuel assemblies is bounded by that of the intact fuel assemblies,which *utilize* a maximum decay heat of *approximately 2.1* kW per assembly *in the failed fuel locations in the OS197FC-B and HSM-H analyses*. Further, the presence of the failed fuel canister also results in increased gamma shielding within the basket. The accident calculations performed for the damaged assemblies conservatively bound those for the failed fuel assemblies because the fuel assembly geometry under accident conditions is identical for both damaged and failed assemblies. Therefore, no additional shielding calculations are necessary for the failed fuel assemblies.

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

T.5.1 Discussion and Results

The shielding analyses for the HSM-H and OS197FC-B are performed using MCNP5 with source terms that conservatively bound HLZC 1 through 13. The shielding analysis for the Standardized HSM (i.e., Model 80 and Model 102) is performed using MCNP5 with source terms that conservatively bound 61BTH Type 1 DSC sources (HLZC 1, 2, 3, 4, and 9). All analysis is conservatively performed with 61BTH Type 1 DSC geometry, which has less shielding than the 61BTH Type 2 DSC.

Table T.5-1 summarizes the maximum and average dose rates for the NUHOMS®-61BTH design basis DSC loaded into the NUHOMS® HSM-H.

Table T.5-2 summarizes the maximum and average dose rates for the NUHOMS®-61BTH design basis DSC loaded into the NUHOMS® HSM Model 102.

Table T.5-3 provides a summary of the dose rates for the NUHOMS®-61BTH design basis DSC loaded into the NUHOMS® Model 80 HSM.

Table T.5-4 provides a summary of the dose rates on and around the OS197FC-B TC for transfer of the 61BTH design basis DSC under normal, off-normal and accident conditions.

Table T.5-5 provides a summary of the dose rates on and around the OS197FC-B TC for decontamination and welding operations for the 61BTH design basis DSC.

A discussion of the method used to determine the design basis fuel source terms is included in Section T.5.2. The shielding material densities are given in Section T.5.3. *Source terms for the HSM-H and OS197FC-B analyses are generated using the ORIGEN-ARP module of SCALE 6.0 [5.8]. Source terms for the Standardized HSM analyses are generated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1].* The shielding evaluation is performed with the MCNP Computer Code Version 5 [5.2, 5.3*, 5.4*] 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 T.5.5.

The NUHOMS®-61BTH DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. **[**

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T.5.2 Source Specification

The source term development described in this section was performed and updated in support of several amendments. While the methodologies are similar, there are distinct differences in methodology used to develop the 61BTH Type 1 and Type 2 DSC source terms. The original methodology used to develop 61BTH Type 1 DSC source terms is maintained in this chapter primarily to support the Standardized HSM dose rate analyses (both HSM and site dose). A simplified methodology is used to develop the 61BTH Type 2 DSC source terms used in the HSM-H and OS197FC-B analyses.

The 61BTH Type 1 DSC source terms were developed using the SAS2H module of SCALE 4.4 as part of the Amendment 10 development effort. At that time, FQTs performed the dual function of limiting both decay heat and dose rates. Because decay heat is directly proportional to the uranium loading of the fuel assembly, FQTs have traditionally been developed using the maximum uranium loading (0.198 MTU for BWR fuel) to ensure conservatively long FQT cooling times. These FQT cooling times were then treated as Technical Specification (TS) limits. Due to the decay heat controlling function, FQTs were developed for every heat load available in the HLZCs, even low heat loads that do not contribute appreciably to dose rate.

An FQT generated at the maximum uranium loading will penalize the minimum required cooling times for fuel assemblies with lower uranium loadings. To lower the cooling times for lower uranium loadings, 0.170 MTU FQTs were developed for Amendment 14. For uranium loadings in the range 0.170 MTU ≤ x ≤ 0.198 MTU, interpolation is allowed between cooling times. This methodology allowed more flexibility but doubled the number of FQTs and added complexity.

Source terms, particularly the neutron source, are sensitive to the burnup/enrichment combinations evaluated. The 61BTH Type 1 DSC FQTs were developed using many burnup/enrichment combinations that are practically impossible due to the physics of reactor systems. However, because unanalyzed burnup/enrichment combinations, while rare, originally could not be stored without an amendment, fuel qualification by extrapolation into the unanalyzed region was added in later amendments, which further increased the complexity for the end user.

As part of Amendment 16, which employed a graded approach, the decay heat limiting function of the FQTs was removed. The Licensee is responsible to certify that all fuel to be loaded meets the decay heat requirements, and using FQTs to limit decay heat is no longer necessary. FQTs currently provide only a dose rate limiting function. Due to that change in methodology, most FQTs were moved from the TS to Chapter T.2 as part of Amendment 16. FQTs are included in the TS only if they have a significant effect on dose rate and subsequent exposure. This approach is discussed in more detail in Chapter 10, Section B 10 TS 2.

To reduce complexity and maximize dose rates, 61BTH Type 2 DSC source terms are developed using highly conservative assumptions. A single FQT is developed only for the bounding source term. This FQT is developed to be simple for the Licensee to apply without the added complexity of extrapolation or interpolation.

To clarify the discussion, the 61BTH Type 1 and 2 DSC source term development sections are not integrated. The original source term methodology pertaining to the 61BTH Type 1 DSC is provided in Sections T.5.2.1.a through T.5.2.5. These subsections are used for:

- *Development of Standardized HSM source terms*
- *Development of 0.198 MTU and 0.170 MTU FQTs provided in Chapter T.2 for HLZC 1 through 10. The 0.54 kW FQT is provided in the TS as a limit when loading the 61BTH Type 1 DSC into a Standardized HSM [1.TS]*
- *Development of reconstituted fuel cooling time penalty.*

All source terms and FQT developed for the 61BTH Type 2 DSC are documented in Section T.5.2.6.

T.5.2.1.a 61BTH Type 1 DSC Source Terms

Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the design basis heavy metal weight of 0.198 MTU. The SAS2H/ORIGEN-S results are used to develop the fuel qualification tables listed in Table T.2-5 through Table T.2-10, Table T.2-*17 and* Table T.2-18 of Chapter T.2 and the design basis fuel source terms suitable for use in the *Standardized HSM* shielding calculations.

The GE-2,3 7x7 Type G2A assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading in the fuel and Co-59 content in the hardware regions as compared to the 8x8, other 9x9, 10x10, and 11x11 fuel assemblies which are also authorized contents of the NUHOMS®-61BTH DSC. The neutron flux during reactor operation is peaked in the active fuel region of the fuel assembly and drops off rapidly outside the active fuel region. Much of the fuel assembly hardware is outside of the active fuel or 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 active fuel region, the (gas) plenum region, and the top (nozzle) region. The GE 7x7 fuel assembly masses for each irradiation region are listed in Table T.5-6. The light elements that make up the various materials for the various fuel assembly materials are taken from Reference [5.6] and are listed in Table T.5-7. The design basis heavy metal loading is 0.198 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 active fuel regions, neutron flux (fluence) correction factors are applied to the light element composition for each region. The neutron flux correction factors which are from Reference [5.19] are given in Table T.5-8.

Evaluations of the existing light water reactor (LWR) fuel data with SAS2H and the 44-group ENDF/B-V library used in the calculation of the design basis source terms are documented in References [5.17] and [5.18]. 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 [5.14]. 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 [5.12], to obtain more data for burnups above 45 GWd/MTU.

There are cross-section data on about 1600 isotopes in the cross section libraries available for SAS2H. Only about 20 isotopes are primary concern when dealing with high burnup spent fuel [5.13, 5.15]. According to Reference [5.15] 95 % of the decay heat is dominated by fewer than 10 nuclides for LWR assemblies at five years of cooling. Eighty-five percent of the decay heat would be contributed by only four isotopes after 100 years.

Applicability of SAS2H for prediction of isotopic content in BWR assemblies was analyzed in [5.13]. A UO_2 sample was burned to 57 GWd/MTU in a BWR reactor. The sample U-235 enrichment was 4.97 wt. %. Also, the isotopic content of the discharged sample was measured experimentally. Measured content was reported for actinides and fission products. Among concentrations of 16 nuclides investigated, five agreed with the measured values to within ±5%.

The fuel qualification tables are generated based on the decay heat limits *for HLZC 1 through 10, as defined in the TS [1.TS]*. SAS2H is used to calculate the minimum required cooling time to the nearest 0.1 year (0.3 years at burnups greater than ~50 GWd/MTU when considering low, less than 0.35 kW/FA, decay heat powers) as a function of fuel 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.

Parameters that influence the source term calculations are fuel assembly power (expressed in MW/Fuel Assembly (MW/FA)) and the total time between cycles. Other depletion parameters like cycle length and number of cycles are derived from the target burnup, MTU loading and *fuel assembly* power. The most important parameter for the calculation of source terms is the *fuel assembly* power. *Fuel assembly* power for typical US-BWR fuel assemblies are ≤ 5 MW/FA. The source terms for this evaluation are calculated using a *fuel assembly* power that ranges from 7 MW/FA (at lower burnups) to 14 MW/FA (at higher burnups) and results in a conservative estimation of the source terms. The time between cycles utilized is 73 days and represents a typical downtime for US BWRs (60 to 90 days).

FQTs *for HLZC 1 through 10* are developed for two different uranium loadings: 0.170 MTU and 0.198 MTU. Because cooling times are selected to target specific decay heat values and decay heat is proportional to the uranium loading, the FQT cooling times decrease with decreasing uranium loading to maintain the same heat load. In most cases, the uranium loading of a fuel assembly will fall between the 0.170 MTU and 0.198 MTU values. In such cases, the cooling time interpolation methodology described in Section *7.2.3.2 of Chapter 7* may be employed.

Each FQT contains an unanalyzed zone marked in the FQTs as gray. Limited extrapolation of FQT cooling times into the unanalyzed regions is allowed. The extrapolation may be performed for a maximum difference of 4 GWd/MTU in burnup or 0.4 wt.% in enrichment. The extrapolation may be performed for either fixed enrichment (variable burnup, fixed FQT column) or fixed burnup (variable enrichment, fixed FQT row). The methodology is:

1. Perform a regression analysis on the FQT cooling times and associated variable (either burnup or enrichment). Note: All FQT cooling times in either the row or column of data being extrapolated shall be used, even if many of the cooling times are the same.

2. Develop a fitting equation for the data. A fourth-order polynomial with parameters having at least six significant digits to avoid rounding errors is recommended.

3. Use the fitting equation to compute the extrapolated cooling time at the desired enrichment or burnup.

4. Add 0.2 years as additional margin.

Because extrapolation may be performed on either an FQT row or column of data, in some cases extrapolation to the same FQT cell could be achieved using either data set. It is possible that the extrapolating equations with two alternative sets of parameters may result in slightly different predictions for the cooling times. However, either of the predicted values are legitimate to use because they are both conservative.

An example is provided for extrapolating the 170 kgU FQT for a 1.2 kW fuel assembly at a fixed burnup of 50 GWd/MTU. Note that only built-in capabilities of MS Excel® are utilized in this example. The cooling times are extracted from Table T.2-18 and summarized in the table below. The minimum enrichment in the analyzed region is 2.6 wt.%, and the cooling time for an enrichment of 2.2 wt.% is desired.

Enrichment (wt.%)	Cooling time (years)	Enrichment (wt.%)	Cooling time (years)
2.60	2.60	4.00	2.40
2.70	2.60	4.10	2.40
2.80	2.60	4.20	2.40
2.90	2.60	4.30	2.40
3.00	2.60	4.40	2.40
3.10	2.50	4.50	2.40
3.20	2.50	4.60	2.40
3.30	2.50	4.70	2.30
3.40	2.50	4.80	2.30
3.50	2.50	4.90	2.30
3.60	2.50	5.00	2.30
3.70	2.50		
3.80	2.50		
3.90	2.40		

The fitting equation Cooling Time as function of Enrichment using 4th order polynomial is:

CT = -0.016105* enr4 + 0.231001* enr3 - 1.219261* enr2 + 2.676368* enr + 0.572562

For enr = 2.2 wt.%, cooling time = 2.6 years. An additional 0.2 years is added to this value for a final extrapolated cooling time of 2.8 years.

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 *Standardized* HSM *(e.g., HSM Model 80 or 102)*. The 1-D discrete ordinates code ANISN [5.7] and the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [5.5] is used to determine the relative HSM 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 analysis purposes. For the HSM, the roof can be selected as the dose location. This approach, described in detail in Section T.5.2.4, is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® 24PTH described in Appendix P, Chapter P.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 MCNP models for the locations of interest. Note that ANISN code is not used to calculate any design basis dose rates. MCNP code models are used for calculating design basis dose rates.

A sample SAS2H/ORIGEN-S input file for the Active Fuel Region of 0.70 kW/FA assembly is listed in Section T.5.5.1. This case corresponds to 62 GWd/MTU, 2.6 wt. % U-235 and 7.1 -years cooling case.

Based on the ANISN response function dose rates, the bounding burnup, minimum initial enrichment, and cooling time combinations for the fuel assemblies used in the shielding analyses of the 61BTH Type 1 DSC in the HSM Model 80 or HSM Model 102 are as follows:

- *Central and inner intermediate zones (0.393 kW per assembly): 62 GWd/MTU, 2.6 wt.% U-235, 21.4-year cooled fuel*
- *Outer intermediate zone (0.480 kW per assembly): 25 GWd/MTU, 1.0 wt.% U-235, 3.2-year cooled fuel*
- *Peripheral zone (0.540 kW per assembly): 25 GWd/MTU, 0.9 wt.% U-235, 3.0-year cooled fuel*

The fuel assembly sources are modeled in the basket per Figure T.5-1.

T.5.2.1 Gamma Source Term for MCNP Models

T.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, active fuel, 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 active fuel region and only the light element source for the plenum, bottom, and top regions. Gamma source terms for the active fuel 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.

Design basis source terms used for the shielding analysis of the 61BTH DSC in HSM-102 are shown in Table T.5-10, Table T.5-16 and Table T.5-18.

T.5.2.1.2 Uncertainty in Gamma Source Terms

Almost 100% of the gamma spectrum from light elements is in the range of *1.0* to 1.33 MeV which corresponds exactly to the two 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 the gamma spectrum is dependent upon the energy. Photon rates computed for fission products tend to be more accurate than 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 greater than 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 ^{244}Cm, which is proportional to the quantity of ^{244}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].

T.5.2.2 Neutron Source Term for MCNP Models

One SAS2H/ORIGEN-S run is required for each burnup/initial enrichment/cooling time combination to determine the total neutron source term for the active fuel 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 90% 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 T.5-10, Table T.5-16, and* Table T.5-18. Neutron source terms for use in the MCNP shielding models are calculated by multiplying the fuel assembly source by the number of assemblies in the active fuel region and summing the terms from all the radial regions. The magnitude of the neutron source is also increased to account for the axial distribution in the fuel, as explained in *Section T.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 T.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 $(\mathrm{MeV})^{-1}$, as given in the MCNP manual [5.4].

T.5.2.3 Axial Peaking

The axial peaking factors for both neutron and gamma sources in BWR fuel are provided in Appendix K Section K.5.2.3. The same peaking factors are used in the MCNP analysis presented herein. The peaking factors for both neutron and gamma sources as a function of active fuel height are listed in Appendix K, Table *K.5-14*. These factors are directly applied to MCNP source input for the fuel region.

These factors in Appendix K, Table *K.5-14* are determined based on typical axial burnup distributions for BWR assemblies and based on typical axial water density distribution that occurs during core operation. Using the base SAS2H/ORIGEN-S input for the 7x7 BWR, selected as the design basis assembly for this application, neutron and gamma source terms are generated for axial zones as a function of burnup and moderator density. This estimates both the non-linear behavior of the neutron source with burnup and the core operating moderator density effects on the actinide isotopics (neutron source). This axial distribution is conservative at high burnup because the burnup distribution will flatten out with increased burnup resulting in a reduction in the overall peaking factor.

The average values of the axial peaking distributions are also provided in Appendix K, Table *K.5-14*. For the gamma distribution, the average value is 1.0. However, for the neutron distribution, the average value of the distribution is *1.326*. 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 fuel assembly as reported in *Table T.5-10, Table T.5-16, and* Table T.5-18 is multiplied by *1.326*.

T.5.2.4 ANISN Evaluation of Bounding Source Terms

As discussed above, the fuel qualification tables are generated based on the decay heat limits for *HLZC 1 through 10 [1.TS]*. SAS2H is used to calculate the minimum required cooling time as a function of fuel 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 *Standardized HSM*, the total source term, which includes the contribution from the fuel as well as the hardware in the entire fuel assembly (including end fittings) is used to calculate its total dose rate using the ANISN code. The methodology employed in the current analysis is similar to that used in Appendix P, Chapter P.5 for 24PTH DSCs. The notion behind using the response function for determination of the bounding combination is that if one burnup/enrichment/cooling time combination results in the highest dose rate at a selected location near or on the HSM surface it will result in the highest dose rate at any location near or on the HSM surface.

An ANISN model is developed only for the OS197 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. Because 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, water was used instead of NS-3 in ANISN models. Therefore, the resulting ANISN model is consistent with the OS197FC-B design at the side of the cask.

ANISN [5.7] determines the fluence of particles through 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 are 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 spacial 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 [5.5] and the ANSI/ANS-6.1.1-1977 flux-to-dose conversion factors [5.10] is chosen to generate the ANISN dose rates used to determine the relative strength of the various source terms from fuel assemblies and determine the design basis source terms for the HSM and TC. These design basis source terms are used with MCNP models of the 61BTH 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 generate the relative dose rates on the TC is similar to a cut through the center of the OS197FC-B TC side model used for the shielding evaluation. Figure T.5-2 provides a sketch for the ANISN model of the OS197FC TC centerline. A sample ANISN input file is included in Section T.5.5.5.

With the exception of the fuel region, the material densities used in the ANISN models are the same as those used in the MCNP models as provided in Table T.5-19. The ANISN and MCNP number densities in the fuel region differ because in the MCNP 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 T.5-20.

To reduce the number of ANISN calculations required, a "response function" is developed using ANISN. Separate response functions are developed for all the radial heat zones shown on Figure T.5-2b. It allows estimation of the relative contribution to the dose rate due to individual decay HLZC used.

To generate a neutron response function, the neutron radiation source for the ANISN model corresponds to a single particle emitted per second for each fuel assembly. The radius of the entire homogenized source region in the ANISN model is 71.58 cm. The axial extent of the gamma source region includes the bottom nozzle, active fuel, plenum and top nozzle zones. The total length of these zones is 447.55 cm. For the neutron ANISN models, the axial zone of the source corresponds to the active fuel region only, which has a length of 365.76 cm.

For the gamma response function, a separate ANISN model is executed with a single gamma emitted per second in each of the 18 CASK-81 gamma energy groups. The ANISN source volume and number of assemblies are used to calculate the ANISN source strength in units of particles per sec per unit volume. The neutron response function is generated in a similar fashion to the gamma response function, although only one ANISN input file is required because the neutron spectrum is adequately represented by the ^{244}Cm spectrum provided in Table T.5-21. 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 due to individual radial zones on the surface of the TC or HSM knowing only the magnitude of the neutron source.

Response functions are generated for each radial fuel zone shown on Figure T.5-2b. An effective compartment unit cell is derived by preserving the total fuel compartment area in the cask. This effective unit cell dimension is 6.26 inches. The one dimensional methodology employed in ANISN is not capable of accurately modeling the two dimensional nature of the radial zone distribution of the fuel compartments. To alleviate this issue, modified zone radii are employed so that the two dimensional shielding features of the radial zones are accounted for. Effective zone radii are assumed that represent the cylindrical regions such that the thickness of the shielding material the particle radiation traverses is preserved and hence would adequately simulate penetration of radiation to the surface of the HSM or transfer cask through each radial zone. The effective zone radii used in the ANISN models are shown on Figure T.5-2.

In order to preserve the volumetric source strength throughout the source regions, adjustment factors are applied to the calculated ANISN response functions. These adjustment factors are equal to the actual total compartment area divided by the ANISN zone area that represents such compartment in ANISN models.

As shown in Figure T.5-2b, radial zone 4 contains 12 assemblies that shield the inner zones only at the 0, 90, 180 and 270 degree corners of the cask. However at the 45, 135, 225 and 315 degree corners radial zone 4 does not shield the inner zones. Therefore, radial zone 4 is treated as void in the calculation of zones 1 through 3 response functions.

Response functions as well as MCNP calculations performed demonstrate that dose rates on or near HSM surfaces are dominated by gamma radiation. The gamma component is larger than the neutron component by a factor of 10 to 100. That implies the burnup, enrichment and cooling time combination resulting in the highest gamma radiation only dose rate when using HSM response functions will be bounding for HSM dose rates.

Therefore, it is appropriate to use transfer cask gamma response functions when determining burnup/enrichment/cooling time combinations related to HSM dose rates.

The response functions for the OS197FC TC are provided in Table T.5-22 through Table T.5-25. These response functions are used to compute the dose rate for each entry in the *0.393 kW, 0.48 kW, and 0.54 kW* fuel qualification tables*, consistent with Figure T.5-1.* For each qualification table, the burnup/enrichment/cooling time combination that results in the highest dose rate is selected as the design basis source.

T.5.2.5 Reconstituted Fuel

As explained in Section T.5.2, each fuel assembly may have up to 10 solid stainless steel rods that replace fuel rods. Reconstituted fuel assemblies typically generate lower decay heat than a standard assembly because fuel is replaced with steel. However, the reconstituted fuel produces higher dose rates due to the irradiated stainless steel that contains a strong ^{60}Co source. As the half-life of ^{60}Co is 5.27 years, after 10 years the ^{60}Co activity in the stainless steel rods is reduced to approximately a factor of four and the reconstituted assembly no longer generates higher dose rates than an equivalent standard fuel assembly. To bound this effect, the fuel qualification tables require an additional 5 years of cooling time for reconstituted fuel assemblies.

To validate this approach SAS2H runs are generated for reconstituted fuel assemblies. Dose rates are estimated using the response functions for radial zone number 4 (see Figure T.5-2b) shown in Table T.5-25. This zone is analyzed because it contributes the largest to the total dose rate at the side of the transfer cask *for* the 61BTH *Type 1* DSC.

The SAS2H model for reconstituted fuel is very similar to the model for standard fuel assemblies. The following changes are implemented to generate the SAS2H model for reconstituted fuel. First, the number of fuel rods is decreased from 49 to 39. Second, the power is adjusted to maintain the desired burnup corresponding to the initial heavy metal loading of 0.198 MTU*39/49, or 0.158 MTU. Lastly, using the material masses shown in Table T.5-6 the SAS2H light elements are modified to account for the 10 fuel rods that have been replaced with stainless steel rods.

It is assumed that reconstituted fuel is irradiated during the second and third cycles because the first cycle will always correspond to fresh fuel that cannot be loaded with reconstituted rods. To accurately model this behavior, two SAS2H models are generated for a subset of burnup and enrichment combinations used to generate the fuel qualification table for the 0.54 kW/assembly fuel assemblies (radial zone 4, see Figure T.5-2b). The first SAS2H model is for only one cycle of irradiation of 10 reconstituted rods, while the second 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) corresponding to three cycles, the effective source term of the reconstituted fuel assembly is generated (three irradiation cycles of fuel and two irradiation cycles of reconstituted fuel).

This source term is used with the response function shown in Table T.5-25 to calculate dose rates due to the reconstituted fuel assembly. If the dose rate of the reconstituted fuel assembly exceeds the dose rates due to the design basis source term of the standard fuel assembly, the cooling time of the reconstituted fuel assembly is increased until the design basis source term is bounding. The reconstituted fuel assembly was analyzed for all burnups at the lowest and highest enrichment to evaluate the cooling time increase as a function of enrichment. When the reconstituted fuel is examined in this fashion, no more than 5 additional years of cooling time is required for reconstituted fuel to be bounded by the design basis source. In most cases, the increase in cooling time is from 1 to 3 years.

Reconstituted fuel may contain up to 10 stainless steel rods that have been irradiated or an unlimited number of lower enriched UO_2 rods or Zircaloy rods or unirradiated stainless steel rods

that replace fuel rods. All replacement rods shall be of similar OD and length such that the equivalent amount of water is displaced as the original fuel rod in the fuel assembly matrix. The lower enriched UO_2 rods are of similar design and behavior as the standard 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.

Fuel assemblies reconstituted with Zircaloy replacement rods are bounded by the design basis source terms because Zircaloy has a low ^{59}Co content and therefore results in a much lower source term that the rod it replaces. Lower enriched UO_2 rods reduce the fuel assembly average initial enrichment. Using this reduced assembly average enrichment with the fuel qualification tables accurately addresses the actual source term for the reconstituted assembly. Finally, unirradiated stainless steel replacement rods contribute no source term and are therefore bounded by the intact fuel assembly source term on which the shielding analysis is based.

T.5.2.6 61BTH Type 2 DSC Source Terms

Source terms for the 61BTH Type 2 DSC to be used in the HSM-H and OS197FC-B dose rate analyses are developed in the following sections. Source terms are computed using the ORIGEN-ARP module of SCALE 6.0 [5.8] and the ge7x7-0 library. These source terms are developed to provide bounding dose rates while reducing the complexity used to develop 61BTH Type 1 DSC source terms documented in Section T.5.2.1a through Section T.5.2.2.

Shielding HLZC

Source terms are developed to bound HLZC 1 through 13, which are defined in the TS [1.TS]. A single HLZC for source term and shielding analysis purposes is provided in Figure T.5-1a. The various zones used in HLZC 1 through 13 are analyzed using four zones.

Inner zones 1 and 2 (37 fuel assemblies) are modeled at 0.5 kW/FA. Peripheral zones 3 and 4 each have 12 FAs. Zone 3 is modeled at 1.0 kW/FA, while zone 4 is modeled at 2.1 kW/FA for OS197FC-B sources and 2.2 kW/FA for HSM-H sources.

The shielding HLZC is developed to provide bounding dose rates at all locations:

- *The maximum FA heat load in any HLZC is 1.7 kW/FA for HLZC 11, which occurs on the periphery. This FA is modeled as 2.1 kW/FA (for OS197FC-B) and 2.2 kW/FA (for HSM-H) in the current analysis.*
- *In HLZC 11, peripheral fuel adjacent to the 1.7 kW/FA location is 0.7 kW/FA, which is modeled as 1.0 kW/FA in the current analysis.*
- *The total DSC is limited to 31.2 kW. The peripheral region alone is modeled as 37.2 kW for OS197FC-B analysis and 38.4 kW in HSM-H analysis. Total as-modeled heat is 55.7 kW/DSC for OS197FC-B analysis and 56.9 kW/DSC for HSM-H analysis.*
- *The maximum allowable heat load in the central zone is 0.48 kW/FA for HLZC 7, which is modeled at 0.5 kW/FA in the current analysis.*

Burnup/Enrichment Combinations

The source term, particularly the neutron component, is sensitive to the enrichment. Data is available from the GC-859 database [5.9] for approximately 139,000 BWR fuel assemblies discharged in the United States through June, 2013. These data are provided graphically on Table T.5-29. The heavy black line captures >99.5% of BWR fuel. The following empirical equations for minimum enrichment as a function of burnup correspond to the heavy black line on Table T.5-29:

- *E = 0.7 wt.% for BU ≤ 6 GWd/MTU*
- *E = 0.9 wt. % for 7 GWd/MTU ≤ BU ≤ 19 GWd/MTU*
- *E = BU/20 wt.% for 20 GWd/MTU ≤ BU ≤ 35 GWd/MTU*
- *E = BU/16 wt.% for 36 GWd/MTU ≤ BU ≤ 62 GWd/MTU*

When using these empirical equations, enrichment is rounded down to the nearest 0.1%. For example, for the maximum burnup of 62 GWd/MTU, E = 62/16 = 3.875%, rounded down to 3.8%. Fuel below the minimum enrichment for each burnup is classified as unanalyzed fuel (UF). Based on Table T.5-29, BWR UF is rare. Most DSCs will not contain UF, and if UF is present, only a small number of UF is likely.

UF may be stored but with additional restrictions. Because dose rates are dominated by fuel on the periphery, the peripheral region is limited to 4 UFs, and each UF must be circumferentially separated by 5 analyzed FAs. The peripheral region is illustrated on TS Figure 1-25c [1.TS]. No restrictions are placed on the number and location of UF in the inner basket locations. An additional 0.2 year cooling time penalty is imposed on all UF [1.TS]. Because the minimum cooling time is 1.0 year, a 0.2 year cooling time penalty is significant. Also, in the storage configuration, site dose rates are largely insensitive to the neutron source because the HSM-H is gamma-dominated. Alternately, the required cooling time for UF may be explicitly determined using the fuel qualification methodology described in this chapter.

Source Term Generation Methodology

Source terms are computed using the ORIGEN-ARP module of SCALE 6.0 [5.8] and the ge7x7-0 library for 1752 combinations of burnup and enrichment:

- *Burnup is varied from 6 GWd/MTU to 62 GWd/MTU in 1 GWd/MTU increments*
- *Enrichment is varied from the minimum enrichment for each burnup (as defined above) to the maximum of 5.0 wt.% in 0.1 wt.% increments.*
- *A constant power of 5.95 MW/FA (35 MW/MTU) is modeled. This power is conservatively high because fuel assembly power is typically reduced to significantly lower levels in the final burnup cycle. The target burnup is achieved by adjusting the cycle length.*
- *Light element masses are consistent with the light elements used to generate the 61BTH Type 1 DSC sources. The light element inputs conservatively feature high cobalt impurities consistent with older fuel assemblies, see Table T.5-7.*

- *All sources are generated using a uranium loading of 0.170 MTU rather than the maximum value of 0.198 MTU to add additional ~5% conservatism in dose rate, as explained in the following paragraphs.*

Studies performed for PWR fuel indicate that lowering the uranium loading from 0.492 MTU to 0.380 MTU for a fixed decay heat shortens the cooling times and reduces self-shielding for the same fuel assembly envelope, and the net effect is an increase in dose rates. See, for example, the footnotes to Table U.5-1 and Table U.5-2 for the 32PTH1 DSC. For BWR fuel, this dose rate effect is less pronounced for lower uranium loadings because the percentage difference in mass between 0.198 MTU and 0.170 MTU for BWR fuel is much smaller than the percentage difference in mass between 0.492 MTU and 0.380 MTU for PWR fuel.

For fixed decay heat, 0.170 MTU BWR sources increase the dose rates by approximately 5% for both the OS197FC-B and HSM-H compared to 0.198 MTU BWR sources. This 5% factor is determined by explicitly computing source terms for 1752 burnup/enrichment combinations for each zone for both 0.198 MTU and 0.170 MTU, and dose rates for each zone are computed and compared using response functions. Because the purpose of the FQT is to limit dose rates and dose rates are slightly larger for lower uranium loadings, all source terms and dose rates for the 61BTH Type 2 DSC analysis are performed for 0.170 MTU.

The influence of uranium loading on dose rate is a second-order effect. A dose rate increase of 5% is considered small and is generally within the uncertainty of a Monte Carlo dose rate analysis.

The source terms for 1752 burnup/enrichment/cooling time (BECT) combinations are determined for each zone using the following methodology:

1. *Source terms are determined for fixed decay heats of 0.5 kW/FA and 1.0 kW/FA by adjusting the cooling time for each burnup/enrichment combination. This results in candidate sources for zones 1, 2, and 3.*
2. *For zone 4, which is the bounding source term, candidate source terms are determined for the following fixed decay times:*
   a) *1.00 year for BU ≤ 47 GWd/MTU*
   b) *1.05 year for 48 GWd/MTU ≤ BU ≤ 49 GWd/MTU*
   c) *1.10 year for 50 GWd/MTU ≤ BU ≤ 51 GWd/MTU*
   d) *1.15 year for 52 GWd/MTU ≤ BU ≤ 53 GWd/MTU*
   e) *1.20 year for 54 GWd/MTU ≤ BU ≤ 55 GWd/MTU*
   f) *1.25 year for 56 GWd/MTU ≤ BU ≤ 58 GWd/MTU*
   g) *1.30 year for 59 GWd/MTU ≤ BU ≤ 62 GWd/MTU*
3. *Each zone is separately ranked by OS197FC-B (Table T.5-25a) and HSM-H (Table T.5-25b) response functions to determine the bounding burnup/enrichment combinations. Response function development is described in the following section.*
4. *For each BECT combination, the light-element only (i.e., Co-60) source is also determined. The end hardware (i.e., bottom end fitting, plenum, and top end fitting) sources are due to Co-60 and may peak at a different BECT than the active fuel sources. Because the OS197FC-B centerline dose rates are sensitive to Co-60 in the hardware regions, the hardware region sources are optimized separately than the active fuel sources.*

5. *For the bounding BECT combination in each zone, ORIGEN-ARP calculations are performed for the bottom end fitting, active fuel, plenum, and top end fitting using the light element masses appropriate for each fuel region.*

The zone 4 cooling times listed above are the basis for the FQT cooling times provided in TS Table 1-4f [1.TS]. Cooling times are fixed for all enrichments corresponding to a particular burnup value. Because a specific decay heat is not targeted in zone 4, each BECT combination may have a different decay heat. However, for cooling times exceeding 1.0 year, the decay heat is ≥ 1.8 kW/FA for all BECTs, which is conservative because the maximum allowed decay heat is 1.7 kW/FA. Because FQTs no longer serve a decay heat limiting function, it is conservative for dose rate calculations if the decay heat of the computed source term is larger than the allowed decay heat.

The total OS197FC-B response function dose rate is 2447 mrem/hr. The total response function dose rate is the sum of zones 1 through 4. The bounding BECT combination for each zone results in the largest dose rate for the BECT combinations considered. The response function dose rates near the bounding zone 4 burnup/enrichment combination are provided in Table T.5-30 as an example of the response function method. Several other BECT combinations result in similar dose rates. Zone 4 results in approximately 60% of the dose rate at the side of the cask, which exceeds the combined dose rate from zones 1, 2, and 3. OS197FC-B bounding BECT combinations in the active fuel region are:

1. *Zone 1: 62 GWd/MTU, 3.8%, 7.3 years (237 mrem/hr)*
2. *Zone 2: 62 GWd/MTU, 3.8%, 7.3 years (172 mrem/hr)*
3. *Zone 3: 62 GWd/MTU, 3.8%, 3.1 years (597 mrem/hr)*
4. *Zone 4: 62 GWd/MTU, 3.8%, 1.3 years (1441 mrem/hr)*

The OS197FC-B source is maximized for neutrons, as the maximum burnup of 62 GWd/MTU is modeled in each zone. In the accident analysis, it is customary to model the maximum neutron source because the neutron shield is lost. Therefore, the OS197FC-B sources are applicable to both normal conditions and accident conditions.

The total HSM-H response function dose rate is 22.5 mrem/hr. The total response function dose rate is the sum of zones 1 through 4. The bounding BECT combination for each zone results in the largest dose rate for the BECT combinations considered. The response function dose rates near the bounding zone 4 burnup/enrichment combination are provided in Table T.5-31 as an example of the response function method. Several other BECT combinations result in the same dose rate within round off and would also be acceptable for use as the design basis source. Zone 4 results in approximately 65% of the dose rate at the HSM-H roof, which exceeds the combined dose rate from zones 1, 2, and 3. HSM-H bounding BECT combinations in the active fuel region are:

1. *Zone 1: 62 GWd/MTU, 3.8%, 7.3 years (0.4 mrem/hr) (same BECT as OS197FC-B)*
2. *Zone 2: 9 GWd/MTU, 5.0%, 1.3 years (0.9 mrem/hr)*
3. *Zone 3: 18 GWd/MTU, 5.0%, 1.1 years (6.7 mrem/hr)*
4. *Zone 4: 47 GWd/MTU, 2.9%, 1.0 years (14.4 mrem/hr)*

The source term in the hardware (i.e., bottom end fitting, plenum, and top end fitting) is due primarily to Co-60, and the total light element source is directly proportional to the Co-60 activity. The bounding light element BECT combinations are:

1. *Zone 1: 35 GWd/MTU, 1.7%, 3.8 years*
2. *Zone 2: 35 GWd/MTU, 1.7%, 3.8 years*
3. *Zone 3: 35 GWd/MTU, 1.7%, 2.1 years*
4. *Zone 4: 62 GWd/MTU, 3.8%, 1.3 years*

OS197FC-B source terms for the 61BTH Type 2 DSC are provided in Table T.5-18a through Table T.5-18c. HSM-H source terms for the 61BTH Type 2 DSC are provided in Table T.5-18d through Table T.5-18g. A sample ORIGEN-ARP input file is provided in Section T.5.5.6.

The treatment of axial peaking is the same as described in Section T.5.2.3. The neutron source is scaled by 1.326 to account for the axial burnup profile. For clarity, the neutron source magnitude to be input to MCNP is provided in the source term tables.

Subcritical neutron multiplication is accounted for automatically by MCNP in a conservative manner because the fuel is modeled as 4.0% enriched. The only exception is for the wet decontamination models, where the neutron source is scaled by 1/(1-k), and k is assumed to be 0.94, which is conservative. The total neutron source to be used in wet and dry analysis is provided in the source term tables.

<u>MCNP Response Functions for the OS197FC-B and HSM-H</u>

Response functions used to "rank" BECT combinations are generated using MCNP. The MCNP response function methodology is analogous to the ANISN response function methodology described in Section T.5.2.4. MCNP is used to compute the dose rate for a unit source on the side of the OS197FC-B and roof of the HSM-H, see Figure T.5-2c and Figure T.5-2d. The HSM-H response function model is a simplification of the actual HSM-H geometry and represents the dose rate on the roof in the absence of vents. This simplification is acceptable, as the response functions are used only to rank the 1752 BECT combinations in each zone. The OS197FC-B and HSM-H response functions are provided in Table T.5-25a and Table T.5-25b.

To determine the "ranking" dose rate for each BECT, the total source is multiplied by the response function and the results are summed. The response function returns the dose rate from the entire zone (i.e., the candidate fuel assembly is placed in every zone location). The maximum response function dose rates are the basis for the design basis source selection.

<u>Reconstituted Fuel</u>

In Section T.5.2.5, a reconstituted fuel analysis is performed for the 61BTH Type 1 DSC with 0.54 kW/FA sources in the peripheral region. For 10 irradiated stainless steel reconstituted rods per fuel assembly (40 rods per DSC), it is concluded that 5 additional years of cooling are required for the dose rate to remain bounded by the design basis source. The conclusions from Section T.5.2.5 are conservatively applied to the 61BTH Type 2 DSC (HLZC 1 through 13).

Because the 61BTH Type 2 DSC has a minimum cooling time of 1 year, a 5 year cooling time penalty for reconstituted fuel will extend the minimum cooling time to 6 years. Therefore, this 5 year penalty has a larger effect on the 61BTH Type 2 DSC sources compared to the Type 1 DSC sources, as the 61BTH Type 1 DSC sources were derived for a minimum 3 year cooling time. Alternately, the Licensee can qualify fuel assemblies with fewer than 10 irradiated stainless steel rods and reduce the cooling time requirement.

T.5.4 Shielding Evaluation

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from 61 BWR fuel assemblies loaded into NUHOMS® -61BTH Type 1 DSC and Type 2 DSC are calculated with *MCNP 5 v 1.40* [5.4] and MCNP 5 v 1.20 [5.2,5.3] at various locations on and around the Transfer Casks (TC) and Horizontal Storage Modules (HSM), respectively. The following evaluation specifically addresses the NUHOMS®-61BTH Type 1 DSC in HSM Model 80, 102 or HSM-H or OS197/OS197H/OS197FC-B TC using the design basis source terms determined in Section T.5.2.

T.5.4.1 Computer Program

MCNP 5 [5.2, 5.3*, 5.4*] *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 *MCNP 5* 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.

T.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,

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

T.5.4.3 Cross Section Data

The cross-section data used is the continuous energy ENDF/B-VI provided with the MCNP code. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to account for the contributions from secondary gamma radiation (n,γ).

T.5.4.4 Flux-to-Dose-Rate Conversion

The flux distribution calculated by the MCNP code is converted to dose rates using flux-to-dose rate conversion factors from ANSI/ANS-6.1.1-1977 [5.10] given in Table T.5-26.

T.5.4.5 Methodology

The methodology used herein is summarized below.

1. Sources are developed for all fuel regions using the source term data developed in Section T.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).

2. Suitable shielding material densities are calculated for all regions modeled.

3. The 3-D Monte Carlo code MCNP is used to calculate dose rates on and around the HSM Model 102, *HSM-H,* and OS197/OS197H/OS197FC-B. The MCNP 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. HSM Model 80 results are determined by applying scaling factors to the HSM Model 102 results.

4. MCNP results are used to calculate offsite exposures (see Section T.10).

5. MCNP models are also generated to determine the effects of accident scenarios, such as loss of cask neutron shield, for the OS197FC-B TC model (Section T.11).

T.5.4.6 Assumptions

The following general assumptions are used in the analyses.

T.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 90% 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 T.5.2.2.
- The BWR heavy metal weight is *modeled as* 0.198 MTU per fuel assembly *in the HSM Model 80 and Model 102 analyses, and 0.170 MTU in the OS197FC-B and HSM-H analyses*.

T.5.4.6.2 HSM-H Dose Rate Analysis Assumptions

- The 61BTH 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. *Three configurations are evaluated.*
  (1) A rear configuration is used to compute rear dose rates and features reflective boundaries on the left and right sides of the model, see Figure T.5-3 through Figure T.5-5.
  (2) A side configuration is used to compute dose rates on the side (end) shield wall and features reflective boundaries on the left side and rear of the model, see Figure T.5-6 and Figure T.5-7.
  (3) A front/roof configuration is used to compute dose rates on the front/roof of the model and features reflective boundaries on the left, right, and rear surfaces, see Figure T.5-7a and Figure T.5-7b.
- Embedment and rebar in the HSM-H concrete are conservatively neglected.
- Penetrations on the exterior of the HSM-H modules for instrumentation and ease of installation are not modeled since they do not result in any significant change in the dose rate distribution and are covered by other modeling conservatisms.
- The borated neutron absorber sheets in the 61BTH DSC are modeled as aluminum.
- An axial source distribution is discussed in Section T.5.2.3 is utilized.
- Fuel is homogenized within the fuel compartment, although the 61BTH DSC basket is modeled explicitly.
- *1.5 inch wide fabrication gaps are modeled between all HSM-Hs, both side-to-side and front-to-front. The gap size is selected to bound gaps that could be encountered during ISFSI installation. At a reflective boundary, the gap is modeled as 1.5/2 = 0.75 inches.*

T.5.4.6.3 HSM Model 102 Dose Rate Analysis Assumptions

The dose rates for HSM Model 102 were also calculated using MCNP. Those dose rates are due to bounding 61BTH DSC Type 1 loading configuration sources. This configuration includes 0.393 kWt/FA assemblies in the central 25 fuel compartments. The next layer of 24 fuel compartments holds *0.48* kWt/FA assemblies, the outer compartments admit assemblies generating 0.54 kWt/FA. Note that this is also a fictitious loading configuration because more than 22.0 kWt/DSC heat load is not allowed for 61BTH DSC Type 1 and HSM Model 102 configuration. This "fictitious" configuration is depicted in Figure T.5-1. The same set of assumptions listed in Section T.5.4.6.2 applies to MCNP model of HSM Model 102*, although only a single plane of reflection is used on the left side of the model, and modeling additional gaps is not necessary because the HSM Model 102s are separated by 6 inch gaps*. 72.48

T.5.4.6.4 OS197FC-B TC Dose Rate Analysis Assumptions

- The 61BTH Type 1 DSC *geometry* is modeled within the OS197FC-B TC. It is assumed that fuel is placed inside of the Type 1 DSC in accordance with the Type 2 DSC *shielding HLZC depicted on Figure T.5-1a.* when calculating dose rates on and around the top and side of the transfer cask. Note that this is a hypothetical DSC loading combination. However, such an assumption results in the transfer cask dose rates that are bounding for all possible decay heat loading configurations. Only the OS197FC-B is modeled as it bounds the OS197 and OS197H TCs.
- Three inches of supplemental temporary neutron shielding and one inch of steel are assumed to be placed on top of the 61BTH Type 1 DSC cover plates during welding operations.
- During the accident case, the cask neutron shield (either water or NS-3) and the neutron shield jacket (outer steel skin) are assumed to be lost.
- The borated neutron absorber sheets in the 61BTH Type 1 and Type 2 DSC are modeled as aluminum.
- An axial source distribution and the *1.326* neutron scaling factor discussed in Section T.5.2.3 is utilized.
- Fuel is homogenized within the fuel assembly cross section, although the 61BTH Type 1 DSC basket is modeled explicitly.
- The OS197FC-B is equipped with channels to allow air flow through the bottom and the lid at the top. The air gaps formed by these channels are assumed to extend around the entire circumference of the cask.

T.5.4.7 Normal Condition Models

As stated above, only one MCNP shielding configuration is considered for the transfer cask and HSM shielding analyses: 61BTH Type 1 DSC. Such a shielding configuration is conservative and bounds other loading configurations. Unless otherwise indicated, the following discussion is related to the bounding 61BTH Type 1 DSC.

T.5.4.7.1 61BTH DSC in HSM-H

As described in Section T.5.4.6.2, separate models are developed for rear, side, and front/roof dose rates, with planes of reflection to maximize dose rates by including the dose rate contribution from adjacent HSM-Hs. Dose rates at hypothetical 1.5 inch gaps between HSM-Hs are explicitly evaluated. The concrete density is modeled at 140 pcf (2.243 g/cm^3). Source terms are consistent with Figure T.5-1a and Table T.5-18d through Table T.5-18g. This is a hypothetical combination, but it results in HSM dose rates that are bounding for all possible DSC/HSM shielding and source term combinations. These models are presented in Figure T.5-3 through Figure T.5-7*b*. The HSM-H length is designated as the z axis, the width as the x axis, and the height as the y axis. The HSM-H door is designated as the -z direction. The roof is the +y direction. *In the side model, the -x direction* is designated as a reflective boundary and an end shield wall (3 ft thick) is attached *in the +x direction*.

The bottom (bottom of bottom fitting) of the fuel assembly is assigned to a z plane at -224.06 cm. The center of the HSM-H is at (x,y,z)=(0,0,0). The 61BTH Type 1 DSC lid is *modeled approximately 8 inches* from the HSM-H rear wall, which places the bottom of the DSC at z=*-243.12* cm, about *24.33* cm from the door interior. The 61BTH DSC support rails are not included in the model. The heat shields are modeled as flat plates and horizontal vent "liner" plates (2 cm thick) are modeled in the top side vents.

The dose rates for the HSM-H are calculated based on a door design *consistent with the round-face design per the drawings in Chapter T.1, with 18.5 inches of concrete and 7-7/8 inches of steel at the door centerline*. 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. Bounding dose rates are provided in Table T.5-1. Gamma and neutron dose rates for the front, side shield wall and roof surface at the DSC centerline of the HSM-H are also plotted as a function of distance in Figure T.5-17 through Figure T.5-22, respectively.

Dose rates are elevated at the hypothetical 1.5 inch gaps between modules, as these gaps are radiation streaming paths. The front gap dose rates are bounded by the inlet vent dose rates, and the roof gap dose rates are bounded by the outlet vent dose rates. However, the maximum side dose rate occurs at the gap, as indicated in Table T.5-1, although the side gap dose rate (43.0 mrem/hr) is negligible compared to the maximum inlet vent dose rate (2081 mrem/hr). The rear gap dose rate is 180 mrem/hr, although the average rear dose rate is low (3.3 mrem/hr) because the gap is small in relation to the total rear surface area.

Average dose rates are also provided in Table T.5-1. Average dose rates (or fluxes) are used as input to the site dose evaluation in Chapter T.10. The average dose rates are computed on the surfaces of a box that encloses the HSM-H, including the vent covers. Gaps between HSM-Hs are included in the average dose rate results.

For the HSM-H, the system is gamma-dominated, as neutrons contribute little to the total dose rate. For example, the maximum inlet dose rate is 2074/2081 >99.5% due to gamma radiation. Fuel in the unanalyzed region of the FQT may be stored with additional restrictions per the TS [1.TS], and fuel in the unanalyzed region could have an elevated neutron source due to the low enrichment. However, HSM-H dose rates and consequently site dose rates are largely insensitive to the neutron dose rate.

An evaluation of *the optional* door configuration *described on the drawings in Chapter T.1* is performed on a door design with 25" thick concrete and 3" thick steel. This evaluation results in *dose rate perturbations* around the door region and has no impact on the average dose rates on the HSM-H front surface. Consequently, the effect on the ISFSI site dose rates, calculated in Chapter T.10, is negligible. Small changes to the steel (±1") or concrete (±6") thicknesses in the door region are not expected to have any significant effect on the front surface dose rates since the surface dose rates in the door region are lower than the HSM front average dose rates by more than a factor of 20.

A sample MCNP 5 model input file of HSM-H with 61BTH DSC is included in Section T.5.5.2.

T.5.4.7.2 61BTH DSC in HSM Model 102

Two three-dimensional MCNP models are developed for the 61BTH DSC Type 1 within a HSM-Model 102, one model is for neutrons and the other for gammas. Note that DSC Type 1 is loaded in accordance with the bounding for 61BTH DSC Type 1 heat loading configuration. This is a fictitious combination but it results in HSM dose rates that are bounding for all possible DSC/HSM shielding and source terms combinations pertinent to the Type 1 DSC. These models are presented in Figure T.5-8 through Figure T.5-10. The HSM length is designated as the z axis, the width as the x axis, and the height as the y axis. The HSM door is designated as the south side and the –z direction, with the west wall as the –x direction. The roof is the +y direction. The west wall is designated as a reflective boundary and an end shield wall (2 ft thick) is attached to the east wall.

The bottom (bottom of bottom fitting) of the fuel assembly is assigned to a z plane at -224.06 cm. The center of the HSM-H is at (x,y,z)=(0,0,0). The 61BTH DSC Type 1 lid is located 2.31" from the HSM Model 102 rear wall (z=252.30 cm) which places the bottom of the DSC at z=-242.92 cm, about 5.07" from the door interior. The 61BTH DSC support rails are not included in the model. Steel embodiments are modeled along vent perimeters as 1.5" thick plates.

Dose rates are calculated on thin cells surrounding the HSM Model 102 and are segmented into about 30 cm increments to capture the peak dose rates. Dose rates are also calculated at the inlet and outlet vents (in front and above frontal and top bird screens, respectively).

A sample MCNP 5 model input file of HSM Model 102 with 61BTH Type 1 DSC is included in Section T.5.5.3.

T.5.4.7.3 61BTH DSC in HSM Model 80

The dose rates for the 61BTH DSC in the HSM Model 80 were calculated utilizing ratios of gamma and neutron dose rates for the 61BT DSC in the Model 102 and Model 80 HSMs. The dose rates calculated for the 61BT DSC in the HSM Model 102 and Model 80 are taken from Appendix K, Chapter K.5. These dose rates are used to determine neutron and gamma dose rate ratios for the front, top, side and rear as shown in Table T.5-27 and Table T.5-28. (The surface averaged and maximum dose rates calculated for the 61BTH / Model 80 configuration are presented in Table T.5-3.) These ratios were then applied to the 61BTH DSC HSM Model 102 dose rates calculated in Section T.5.4.7.2. This method is reasonable, as the design basis source spectra for the 61BT canister is essentially equivalent to the 61BTH.

T.5.4.7.4 61BTH DSC in OS197FC-B TC

Two three-dimensional MCNP models are employed for *each* shielding *configuration* of the 61BTH DSC within an OS197FC-B TC, one model calculating for neutron dose rates and the other for gamma dose rates. These models are presented in Figure T.5-11 through Figure T.5-16. The z-axis in the MCNP models coincides with the axis of rotation of the cask and the 61BTH 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 61 neutron shield panel support angles, the 4 trunnions, relief valves, clevises, and eyebolts. With the exception of the 61 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 61 neutron shield support angles provide support for the neutron shield skin, which contains 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 *0.958* g/cm^3.

The trunnions penetrate the neutron shield, which locally changes the shielding configuration of the neutron shield. The trunnions are thick steel structures filled with an optional 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 *0.958* 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-B TC and 61BTH DSC are modeled in MCNP. The overall length of the OS197FC-B TC is 202.97". The outer diameter of the OS197FC-B TC is 85.50" (neutron shield included). The outer diameter excluding the neutron shield is 79.12". The bottom of the OS197FC-B TC is designed to mate with a 61BTH DSC. The overall length of the 61BTH DSC is 196.04" (excluding the grapple) and its outer diameter is 67.25". The bottom end of the 61BTH DSC is in contact with the structural shell assembly of the transfer cask. *The top shield plug is modeled with a radial clearance gap of 0.25" between the top shield plug and the internal DSC shell diameter. In the axial direction, the clearance gap sits above the support ring and below the inner top cover plate.* 72.48

The OS197FC-B TC has a ventilated top lid to facilitate air circulation. In MCNP, 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 the ram access cover.

Dose rates are provided in Table T.5-4. Dose rates at the sides, top, and bottom of this cask are presented graphically in Figure T.5-23 through Figure T.5-25.

A sample MCNP model input file for OS197FC-B TC with 61BTH DSC is included in Section 5.5.3.

T.5.4.8 Accident Condition Models

No accident condition models were developed for the HSM-H because no accident scenario in Chapter T.11 has been identified that would alter the dose rates provided in Table T.5-1. The HSM Model 102 in an array, in an accident condition, is assumed to slide next to an adjacent HSM and therefore double the gap on one side as described in Chapter T.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 T.11.

For the OS197FC-B TC, *a far-field* accident case *using the source terms provided in Table T.5-18a through Table T.5-18d* is performed which assumes that the neutron shield and steel neutron shield jacket (outer skin of each) has been torn off. A second case *using HLZC 6 source terms* is considered to analyze the effect of damaged fuel turning to rubble in the bottom of the cask following an accident. Figure T.5-15 and Figure T.5-16 show the MCNP fuel rubble accident model. The dose rates from fuel rubble exhibit local peaking however at far distances the accident dose rates without fuel rubble are conservative. Accident dose rates at 1m, 100m, and 500m from the side of the cask are presented in Table T.5-4.

T.5.4.9 OS197FC-B TC Models During Fuel Loading Operations

MCNP models are developed for the cask decontamination and welding operations during fuel loading using the 61BTH DSC. The dose rates calculated for these operations are utilized to estimate the occupational exposure which is documented in Chapter T.10.

Cask Decontamination. The 61BTH DSC and the OS197FC-B TC are assumed to be completely filled with water, including the region between 61BTH DSC and cask, which is referred to as the "TC/61BTH DSC annulus." The 61BTH DSC top shield plug and inner cover plate are assumed to be in place and the temporary shielding has not yet been installed (configuration prior to placement of the automated welding system (AWS)). *Because the cask is flooded with water, subcritical neutron multiplication is suppressed using the NONU card, and k = 0.94 is conservatively assumed. The neutron source is then scaled by 1/(1-k). The associated neutron source is indicated in Table T.5-18a through Table T.5-18d.* Results for this case are provided in Table T.5-5.

Welding and 61BTH DSC Draining. Before the start of welding operation, approximately 60% of the water in the DSC cavity is removed due to hydrogen generation considerations. A dry DSC cavity is assumed in all welding models to be conservative. Temporary shielding consisting of three inches of NS-3 and one inch of steel is assumed to cover the 61BTH DSC top shield plug. *This temporary shielding may be constructed of other materials that provide equivalent or better shielding.* In addition, the DSC outer top cover plate is not present. The cask/61BTH DSC annulus is assumed to remain completely filled with water. Results for this case are provided in Table T.5-5. 72.48

T.5.4.10 Impact on Dose Rates due to Reduced Density Concrete and Gaps between HSMs

1.5 inch fabrication gaps and 140 pcf concrete density are explicitly considered in the HSM-H analysis presented in this chapter, which is conservative. Dose reduction hardware may be installed to further reduce *vent* dose rates. *No dose reduction hardware is included in the models presented in this chapter.*

Proprietary Information on This Page
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Proprietary Information on Pages T.5-75 through T.5-91a
Withheld Pursuant to 10 CFR 2.390

Proprietary Information on Pages T.5-96a through T.5-96e
Withheld Pursuant to 10 CFR 2.390

T.5.6 References

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5.3 MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, Volume II: User's Guide, LA-CP-03-0245, 2003.

5.4 MCNP - A General Monte Carlo N-Particle Transport Code, Version 5, LA-UR-03-1987, April 2003.

5.5 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, August 1987.

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

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5.8 Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," ORNL/TM-2005/39, Version 6, January 2009.

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5.11 Not used.

5.12 U. S. Nuclear Regulatory Commission, "Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High Burnup LWR Fuel," NUREG/CR-6701, Published January 2001, ORNL/TM-2000/277.

5.13 B. D Murphy, "Prediction of Isotopic Composition of UO2 Fuel from a BWR: Analysis of the DU1 Sample from the Dodewaard Reactor," ORNL/TM-13687, October 1998.

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

Table T.5-1
Summary of NUHOMS®-61BTH DSC in HSM-H, Bounding Maximum and Average Dose Rates (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)	*121.5*	*0.2%*	*1.5*	*0.2%*	*123.0*	*0.2%*
HSM Roof Birdscreen	*1486*	*0.2%*	*15.7*	*0.2%*	*1502*	*0.2%*
HSM End (Side) Shield Wall Surface *(excluding gap)* (3)	*11.4*	*0.9%*	*0.1*	*1.5%*	*11.5*	*0.9%*
HSM Door Exterior Surface (centerline)	*0.6*	*3.2%*	*0.2*	*1.4%*	*0.8*	*2.5%*
HSM Front Birdscreen	*2074*	*5.9%*	*6.6*	*2.1%*	*2081*	*5.9%*

Dose Rate Location	Gamma 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 *(above vent covers, includes gaps)*	*122.0*	*0.1%*	*1.4*	*0.1%*	*123.4*	*0.1%*
HSM End (Side) Shield Wall Surface *(includes gaps)*	*3.0*	*0.2%*	*0.04*	*0.1%*	*3.1*	*0.2%*
HSM Front *(includes gaps)*	*52.6*	*0.7%*	*0.4*	*0.2%*	*53.0*	*0.7%*
HSM Back Shield Wall *(includes gaps)*	*3.2*	*3.3%*	*0.02*	*1.2%*	*3.3*	*3.2%*

Notes:

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

(2) Dose calculated using a 61BTH Type 2 DSC source loaded into a Type 1 DSC bounds dose rates from all specified DSC configurations. Dose rates can be higher by 6% to account for the use of grout during HSM fabrication and installation.

(3) *At a hypothetical 1.5 inch gap between back-to-back HSMs, the maximum dose rates are 42.4 mrem/hr gamma, 0.63 mrem/hr neutron, and 43.0 mrem/hr total.*

Table T.5-4
Summary of NUHOMS®-61BTH DSC, OS197FC-B TC Maximum Dose Rates During Transfer Operations

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)	*1.83E+03*	*0.3%*	*7.57E+02*	*0.2%*	*2.59E+03*	*0.2%*
Cask Top Axial Surface	*2.93E+02*	*4.1%*	*2.51E+01*	*1.2%*	*3.06E+02*	*4.0%*
Cask Bottom Axial Surface(2)	*5.10E+03*	*0.9%*	*1.05E+03*	*0.4%*	*6.15E+03*	*0.7%*
1 ft from Cask Side (Radial)	*1.24E+03*	*0.3%*	*4.74E+02*	*0.1%*	*1.72E+03*	*0.2%*
1 ft from Cask Top Axial Surface	*8.13E+01*	*5.0%*	*1.53E+01*	*0.9%*	*9.01E+01*	*4.5%*
1 ft from Cask Bottom Axial Surface	*2.67E+03*	*1.0%*	*3.77E+02*	*0.5%*	*3.05E+03*	*0.9%*
3 ft from Cask Side (Radial)	*7.64E+02*	*0.3%*	*2.67E+02*	*0.1%*	*1.03E+03*	*0.2%*
3 ft from Cask Top Axial Surface	*3.78E+01*	*6.3%*	*1.03E+01*	*0.5%*	*4.39E+01*	*5.4%*
3 ft from Cask Bottom Axial Surface	*9.63E+02*	*1.2%*	*1.06E+02*	*0.8%*	*1.07E+03*	*1.1%*
Cask 1 m (Radial) Accident Condition	*1.13E+03*	*1.5%*	*3.13E+03*	*0.5%*	*4.26E+03*	*0.5%*
Cask 100 m (Radial) Accident Condition	*5.32E-01*	*1.1%*	*1.21E+00*	*0.4%*	*1.75E+00*	*0.4%*
Cask 500 m (Radial) Accident Condition	*2.90E-03*	*1.5%*	*4.74E-03*	*1.0%*	*7.64E-03*	*0.8%*

Notes:

(1) Gamma and neutron dose rate peaks do not always occur at the 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 *743* mrem/hr gamma, *181* mrem/hr neutron for a total average dose rate of *924* mrem/hr.

Table T.5-5
Summary of NUHOMS®-61BTH DSC, OS197FC-B TC Maximum Dose Rates During Decontamination and Welding Operations

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)	*9.84E+02*	*0.4%*	*4.66E+02*	*0.5%*	*1.45E+03*	*0.3%*
Top Axial Surface	*2.00E+03*	*1.6%*	*9.58E+00*	*1.9%*	*2.00E+03*	*1.6%*
Cask Bottom Axial Surface(2)	*3.96E+03*	*1.0%*	*2.32E+01*	*1.4%*	*3.97E+03*	*1.0%*
1 ft from Cask Side (Radial)	*6.81E+02*	*0.4%*	*2.93E+02*	*0.4%*	*9.74E+02*	*0.3%*
1 ft from Top Axial Surface	*1.63E+03*	*1.5%*	*4.33E+00*	*7.1%*	*1.63E+03*	*1.5%*
1 ft from Cask Bottom Axial Surface	*2.06E+03*	*1.1%*	*1.15E+01*	*5.4%*	*2.06E+03*	*1.1%*
3 ft from Cask Side (Radial)	*4.23E+02*	*0.4%*	*1.63E+02*	*0.4%*	*5.87E+02*	*0.3%*
3 ft from Top Axial Surface	*9.72E+02*	*1.8%*	*3.92E+00*	*2.5%*	*9.72E+02*	*1.8%*
3 ft from Cask Bottom Axial Surface	*7.24E+02*	*1.5%*	*7.97E+00*	*1.8%*	*7.25E+02*	*1.5%*
Welding						
Cask Side Surface (Radial)	*1.67E+03*	*0.3%*	*5.43E+02*	*0.1%*	*2.21E+03*	*0.3%*
Top Axial Surface	*2.65E+03*	*4.3%*	*2.61E+01*	*3.4%*	*2.68E+03*	*4.2%*
Cask Bottom Axial Surface(3)	*5.11E+03*	*1.1%*	*9.36E+02*	*0.4%*	*6.04E+03*	*1.0%*
1 ft from Cask Side (Radial)	*1.14E+03*	*0.3%*	*3.40E+02*	*0.1%*	*1.48E+03*	*0.2%*
1 ft from Top Axial Surface	*1.12E+03*	*5.4%*	*1.12E+01*	*3.8%*	*1.13E+03*	*5.4%*
1 ft from Cask Bottom Axial Surface(3)	*2.66E+03*	*1.3%*	*3.36E+02*	*0.4%*	*3.00E+03*	*1.2%*
3 ft from Cask Side (Radial)	*7.08E+02*	*0.3%*	*1.92E+02*	*0.2%*	*8.99E+02*	*0.2%*
3 ft from Top Axial Surface	*5.99E+02*	*7.0%*	*6.46E+00*	*0.7%*	*6.05E+02*	*6.9%*
3 ft from Cask Bottom Axial Surface(3)	*9.46E+02*	*1.8%*	*9.22E+01*	*0.7%*	*1.04E+03*	*1.7%*

Notes:

(1) Gamma and neutron dose rate peaks do not always occur at the 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 *554* mrem/hr gamma, *1.5* mrem/hr neutron for a total average dose rate of *556* mrem/hr. *The peak neutron dose rate of 23.2 mrem/hr occurs at a radius of 250 cm and not at the centerline, the centerline neutron dose rate is < 10 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 *733* mrem/hr gamma, *155* mrem/hr neutron for a total average dose rate of *888* mrem/hr. Note that this bottom axial dose rate has no impact on the occupational exposure because no operations are performed near the bottom axial location.

Table T.5-6
BWR Fuel Assembly Material Mass

Hardware Item	Material	Average Mass, (kg/FA)	Comments
Active Fuel Zone, (**144.00** inch long, **4.73** g/FA total cobalt content)			
Cladding	Zircaloy-2	49.2	
Fuel Channel Sleeve	Zircaloy-4	37.1	
Grid Spacers	Zircaloy-4	1.95	7 spacers*~0.28 kg/spacer
Spacer Springs	Inconel X-750	0.36	7 springs*0.051 kg/spring
Channel Spring & Bolt	Inconel X-750	0.13	
Channel Fastener Guard	Stainless Steel	0.46	
Channel Spacer & Rivert	Stainless Steel	0.13	
Fuel	Uranium	198*(2)*	wt. of UO_2=224.643 kg.=0.198 mtu/0.8814
Gas Plenum Zone, (**12.93** inch long, **0.89** g/FA total cobalt content)			
Cladding	Zircaloy-2	4.89	
Fuel Channel	Zircaloy-4	0.00	
Plenum Springs	Stainless Steel	1.05	
Top End Fitting Zone, (**12.62** inch long, **4.51** g/FA total cobalt content)			
Upper Tie Plate	Stainless Steel	2.08	
Lock Tab Washers & Nuts	Stainless Steel	0.05	
Expansion Springs	Inconel X-750	0.43	
End Plugs	Zircaloy	1.26	
Bottom End Fitting Zone, (**6.65** inch long, **4.10** g/FA total cobalt content)			
Finger Springs	Inconel	0.05	
End Plugs	Zircaloy	1.26	
Lower Tie Plate	Stainless Steel	4.7	
Total, kgs.(1)		329.7	
Total, lbs.(1)		726.3	

Note 1: This mass is very conservative for the source term calculation because the maximum weight of fuel assembly with or without channel is limited to 705 lbs per Chapter T.2.

Note 2: In the OS197FC-B and HSM-H models, 170 kgU is conservatively modeled.

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

Element	Atomic Number	Material Composition, grams per kg of material				UO_2 Fuel, Grams/1.345 kgs
		Zircaloy-4	Inconel-718	Inconel X-750	Stainless Steel 304	
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*(1)*	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	-	-	-	1.00E-03

Note: (1) Modern fuel assemblies have significantly lower cobalt impurity than the values provided in this table.

Table T.5-9
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Table T.5-10
Gamma and Neutron Source Term for 0.48 kW Fuel, HSM Model 80 and 102 for the Modeled HLZC (Figure T.5-1)
(25 GWd/MTU, 1.0 wt. % U-235 and 3.2-Year Cooled Fuel)

E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0	to	0.05	1.5767e+11	1.0846e+15	1.9068e+11	1.2116e+11
0.05	to	0.1	1.6981e+10	2.4167e+14	5.6723e+09	1.2771e+10
0.1	to	0.2	5.1010e+09	2.1043e+14	9.0221e+09	4.0617e+09
0.2	to	0.3	2.6492e+08	5.9161e+13	5.3713e+08	2.1288e+08
0.3	to	0.4	5.0740e+08	4.5265e+13	1.9215e+09	4.3552e+08
0.4	to	0.6	5.3991e+09	4.1178e+14	4.0981e+10	5.2686e+09
0.6	to	0.8	2.8120e+09	8.1035e+14	2.1314e+10	2.7711e+09
0.8	to	1.0	1.0518e+11	1.4113e+14	4.5006e+10	4.6043e+10
1.0	to	1.33	4.9362e+12	7.2668e+13	1.4864e+12	3.7075e+12
1.33	to	1.66	1.3940e+12	2.3412e+13	4.1975e+11	1.0470e+12
1.66	to	2.0	1.0611e+05	1.3819e+12	4.3329e+04	1.0363e+05
2.0	to	2.5	3.3081e+07	3.3957e+12	9.9618e+06	2.4847e+07
2.5	to	3.0	5.1296e+04	1.1210e+11	1.5446e+04	3.8527e+04
3.0	to	4.0	1.4138e-12	1.3964e+10	4.5348e-16	1.0104e-11
4.0	to	5.0	0	3.7908e+06	0	0
5.0	to	6.5	0	1.5214e+06	0	0
6.5	to	8.0	0	2.9846e+05	0	0
8.0	to	10.0	0	6.3370e+04	0	0
Total Gamma			6.6241e+12	3.1054e+15	2.2213e+12	4.9472e+12
Total Neutron			1.10e+8 n/s/assembly			

Table T.5-11
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Table T.5-12
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Table T.5-13
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Table T.5-14
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Table T.5-15
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Table T.5-16
Gamma and Neutron Source Term for 0.54 kW Fuel, HSM Model 80 and 102 for the Modeled HLZC (Figure T.5-1)
(25 GWd/MTU, 0.9 wt. % U-235 and 3.0-Year Cooled Fuel)

E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0	to	0.05	1.6853e+11	1.2121e+15	2.1150e+11	1.2917e+11
0.05	to	0.1	1.7923e+10	2.7252e+14	5.9980e+09	1.3438e+10
0.1	to	0.2	5.3997e+09	2.3891e+14	9.6161e+09	4.2887e+09
0.2	to	0.3	2.8124e+08	6.7134e+13	5.7783e+08	2.2547e+08
0.3	to	0.4	5.3938e+08	5.1646e+13	2.0501e+09	4.6187e+08
0.4	to	0.6	5.7892e+09	4.5751e+14	4.3741e+10	5.6293e+09
0.6	to	0.8	3.0299e+09	8.4881e+14	2.2863e+10	2.9703e+09
0.8	to	1.0	1.2517e+11	1.5349e+14	5.3430e+10	5.4651e+10
1.0	to	1.33	5.2088e+12	7.8190e+13	1.5639e+12	3.9003e+12
1.33	to	1.66	1.4710e+12	2.5486e+13	4.4164e+11	1.1014e+12
1.66	to	2.0	2.1615e+05	1.6212e+12	8.9902e+04	2.1136e+05
2.0	to	2.5	3.49098+07	4.0106e+12	1.0482e+07	2.6139e+07
2.5	to	3.0	5.4129e+04	1.3173e+11	1.6252e+04	4.0531e+04
3.0	to	4.0	1.5221e-12	1.6406e+10	4.9681e-16	1.0817e-11
4.0	to	5.0	0	4.1984e+06	0	0
5.0	to	6.5	0	1.6850e+06	0	0
6.5	to	8.0	0	3.3055e+05	0	0
8.0	to	10.0	0	7.0184e+04	0	0
Total Gamma			7.0065e+12	3.4116e+15	2.3553e+12	5.2126e+12
Total Neutron			1.22e+8 n/s/assembly			

Table T.5-17
Deleted

Table T.5-18a
OS197FC-B Source Term, Zones 1 and 2

Burnup (GWd/MTU)			35	62	35	35
Enrichment (%)			1.7	3.8	1.7	1.7
Cooling Time (years)			3.8	7.3	3.8	3.8
E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0.01	to	0.05	1.012E+11	5.133E+14	1.270E+11	7.113E+10
0.05	to	0.1	1.734E+10	1.378E+14	5.043E+09	1.152E+10
0.1	to	0.2	5.286E+09	1.049E+14	8.711E+09	3.755E+09
0.2	to	0.3	2.832E+08	3.027E+13	5.968E+08	2.072E+08
0.3	to	0.4	5.308E+08	1.982E+13	1.922E+09	4.183E+08
0.4	to	0.6	5.662E+09	2.005E+14	4.025E+10	5.197E+09
0.6	to	0.8	2.959E+09	1.038E+15	2.101E+10	2.751E+09
0.8	to	1.0	8.226E+10	9.549E+13	3.403E+10	3.488E+10
1.0	to	1.33	5.014E+12	5.470E+13	1.321E+12	3.336E+12
1.33	to	1.6	1.416E+12	1.435E+13	3.730E+11	9.422E+11
1.66	to	2.0	7.219E+03	1.464E+11	3.340E+03	7.570E+03
2.0	to	2.5	3.388E+07	1.444E+11	8.924E+06	2.254E+07
2.5	to	3.0	2.895E+04	8.555E+09	7.625E+03	1.926E+04
3.0	to	4.0	2.137E-07	8.248E+08	4.307E-08	2.358E-06
4.0	to	5.0	2.670E-30	1.312E+07	7.595E-29	9.688E-30
5.0	to	6.5	7.693E-31	5.266E+06	2.188E-29	2.791E-30
6.5	to	8.0	9.785E-32	1.033E+06	2.783E-30	3.551E-31
8.0	to	10.0	1.306E-32	2.193E+05	3.715E-31	4.738E-32
Total Gamma			6.645E+12	2.209E+15	1.932E+12	4.408E+12
Total Neutron			3.777E+08 n/s/assembly (raw) 5.008E+08 n/s/assembly (dry, treated with 1.326 peaking factor) 8.347E+09 n/s/assembly (wet, treated with 1.326 peaking factor and k = 0.94)			

Table T.5-18b
OS197FC-B Source Term, Zone 3

Burnup (GWd/MTU)			35	62	35	35
Enrichment (%)			1.7	3.8	1.7	1.7
Cooling Time (years)			2.1	3.1	2.1	2.1
E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0.01	to	0.05	1.446E+11	1.294E+15	2.915E+11	1.061E+11
0.05	to	0.1	2.180E+10	3.925E+14	6.668E+09	1.448E+10
0.1	to	0.2	6.973E+09	3.449E+14	1.322E+10	5.011E+09
0.2	to	0.3	3.951E+08	9.612E+13	1.009E+09	2.950E+08
0.3	to	0.4	1.156E+09	7.146E+13	5.618E+09	9.387E+08
0.4	to	0.6	8.944E+09	9.188E+14	6.222E+10	8.238E+09
0.6	to	0.8	6.211E+09	1.804E+15	4.248E+10	5.518E+09
0.8	to	1.0	3.292E+11	3.675E+14	1.363E+11	1.397E+11
1.0	to	1.33	6.279E+12	1.184E+14	1.654E+12	4.178E+12
1.33	to	1.6	1.773E+12	4.283E+13	4.671E+11	1.180E+12
1.66	to	2.0	3.292E+06	2.027E+12	1.923E+06	3.448E+06
2.0	to	2.5	4.243E+07	4.499E+12	1.126E+07	2.824E+07
2.5	to	3.0	3.626E+04	1.525E+11	9.615E+03	2.413E+04
3.0	to	4.0	2.216E-07	1.409E+10	4.465E-08	2.445E-06
4.0	to	5.0	2.670E-30	1.548E+07	7.595E-29	9.688E-30
5.0	to	6.5	7.693E-31	6.213E+06	2.188E-29	2.791E-30
6.5	to	8.0	9.785E-32	1.219E+06	2.783E-30	3.551E-31
8.0	to	10.0	1.306E-32	2.588E+05	3.715E-31	4.738E-32
Total Gamma			8.571E+12	5.457E+15	2.680E+12	5.638E+12
Total Neutron			4.461E+08 n/s/assembly (raw) 5.915E+08 n/s/assembly (dry, treated with 1.326 peaking factor) 9.859E+09 n/s/assembly (wet, treated with 1.326 peaking factor and k = 0.94)			

Table T.5-18c
OS197FC-B Source Term, Zone 4

Burnup (GWd/MTU)			62	62	62	62
Enrichment (%)			3.8	3.8	3.8	3.8
Cooling Time (years)			1.3	1.3	1.3	1.3
E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0.01	to	0.05	2.038E+11	3.421E+15	5.417E+11	1.545E+11
0.05	to	0.1	2.716E+10	1.105E+15	9.644E+09	1.812E+10
0.1	to	0.2	9.343E+09	1.050E+15	2.064E+10	6.825E+09
0.2	to	0.3	6.479E+08	2.810E+14	2.110E+09	5.030E+08
0.3	to	0.4	3.397E+09	2.192E+14	1.895E+10	2.731E+09
0.4	to	0.6	1.725E+10	1.939E+15	9.793E+10	1.623E+10
0.6	to	0.8	4.105E+10	2.905E+15	2.545E+11	3.286E+10
0.8	to	1.0	6.918E+11	6.861E+14	2.839E+11	2.969E+11
1.0	to	1.33	7.720E+12	2.038E+14	2.036E+12	5.138E+12
1.33	to	1.6	2.180E+12	8.368E+13	5.749E+11	1.451E+12
1.66	to	2.0	5.836E+07	7.035E+12	4.801E+07	6.149E+07
2.0	to	2.5	5.258E+07	1.983E+13	1.699E+07	3.513E+07
2.5	to	3.0	4.491E+04	5.104E+11	1.440E+04	3.000E+04
3.0	to	4.0	3.748E-07	4.667E+10	7.700E-08	4.216E-06
4.0	to	5.0	3.125E-29	1.702E+07	1.049E-27	1.338E-28
5.0	to	6.5	9.006E-30	6.831E+06	3.023E-28	3.856E-29
6.5	to	8.0	1.145E-30	1.340E+06	3.845E-29	4.905E-30
8.0	to	10.0	1.529E-31	2.845E+05	5.132E-30	6.546E-31
Total Gamma			1.089E+13	1.192E+16	3.841E+12	7.117E+12
Total Neutron			4.919E+08 n/s/assembly (raw) 6.523E+08 n/s/assembly (dry, treated with 1.326 peaking factor) 1.087E+10 n/s/assembly (wet, treated with 1.326 peaking factor and k = 0.94)			

Table T.5-18d
HSM-H Source Term, Zone 1

Burnup (GWd/MTU)			35	62	35	35
Enrichment (%)			1.7	3.8	1.7	1.7
Cooling Time (years)			3.8	7.3	3.8	3.8
E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0.01	to	0.05	1.012E+11	5.133E+14	1.270E+11	7.113E+10
0.05	to	0.1	1.734E+10	1.378E+14	5.043E+09	1.152E+10
0.1	to	0.2	5.286E+09	1.049E+14	8.711E+09	3.755E+09
0.2	to	0.3	2.832E+08	3.027E+13	5.968E+08	2.072E+08
0.3	to	0.4	5.308E+08	1.982E+13	1.922E+09	4.183E+08
0.4	to	0.6	5.662E+09	2.005E+14	4.025E+10	5.197E+09
0.6	to	0.8	2.959E+09	1.038E+15	2.101E+10	2.751E+09
0.8	to	1.0	8.226E+10	9.549E+13	3.403E+10	3.488E+10
1.0	to	1.33	5.014E+12	5.470E+13	1.321E+12	3.336E+12
1.33	to	1.6	1.416E+12	1.435E+13	3.730E+11	9.422E+11
1.66	to	2.0	7.219E+03	1.464E+11	3.340E+03	7.570E+03
2.0	to	2.5	3.388E+07	1.444E+11	8.924E+06	2.254E+07
2.5	to	3.0	2.895E+04	8.555E+09	7.625E+03	1.926E+04
3.0	to	4.0	2.137E-07	8.248E+08	4.307E-08	2.358E-06
4.0	to	5.0	2.670E-30	1.312E+07	7.595E-29	9.688E-30
5.0	to	6.5	7.693E-31	5.266E+06	2.188E-29	2.791E-30
6.5	to	8.0	9.785E-32	1.033E+06	2.783E-30	3.551E-31
8.0	to	10.0	1.306E-32	2.193E+05	3.715E-31	4.738E-32
Total Gamma			6.645E+12	2.209E+15	1.932E+12	4.408E+12
Total Neutron			3.777E+08 n/s/assembly (raw) 5.008E+08 n/s/assembly (dry, treated with 1.326 peaking factor)			

Table T.5-18e
HSM-H Source Term, Zone 2

Burnup (GWd/MTU)			35	9	35	35
Enrichment (%)			1.7	5.0	1.7	1.7
Cooling Time (years)			3.8	1.3	3.8	3.8
E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0.01	to	0.05	1.012E+11	1.172E+15	1.270E+11	7.113E+10
0.05	to	0.1	1.734E+10	3.733E+14	5.043E+09	1.152E+10
0.1	to	0.2	5.286E+09	4.079E+14	8.711E+09	3.755E+09
0.2	to	0.3	2.832E+08	9.208E+13	5.968E+08	2.072E+08
0.3	to	0.4	5.308E+08	7.265E+13	1.922E+09	4.183E+08
0.4	to	0.6	5.662E+09	1.526E+14	4.025E+10	5.197E+09
0.6	to	0.8	2.959E+09	5.006E+14	2.101E+10	2.751E+09
0.8	to	1.0	8.226E+10	3.357E+13	3.403E+10	3.488E+10
1.0	to	1.33	5.014E+12	2.348E+13	1.321E+12	3.336E+12
1.33	to	1.6	1.416E+12	1.057E+13	3.730E+11	9.422E+11
1.66	to	2.0	7.219E+03	1.469E+12	3.340E+03	7.570E+03
2.0	to	2.5	3.388E+07	1.057E+13	8.924E+06	2.254E+07
2.5	to	3.0	2.895E+04	6.567E+10	7.625E+03	1.926E+04
3.0	to	4.0	2.137E-07	5.275E+09	4.307E-08	2.358E-06
4.0	to	5.0	2.670E-30	4.740E+03	7.595E-29	9.688E-30
5.0	to	6.5	7.693E-31	1.890E+03	2.188E-29	2.791E-30
6.5	to	8.0	9.785E-32	3.686E+02	2.783E-30	3.551E-31
8.0	to	10.0	1.306E-32	7.795E+01	3.715E-31	4.738E-32
Total Gamma			6.645E+12	2.851E+15	1.932E+12	4.408E+12
Total Neutron			1.603E+05 n/s/assembly (raw) 2.126E+05 n/s/assembly (dry, treated with 1.326 peaking factor)			

Table T.5-18f
HSM-H Source Term, Zone 3

Burnup (GWd/MTU)			35	18	35	35
Enrichment (%)			1.7	5.0	1.7	1.7
Cooling Time (years)			2.1	1.1	2.1	2.1
E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0.01	to	0.05	1.446E+11	2.240E+15	2.915E+11	1.061E+11
0.05	to	0.1	2.180E+10	7.168E+14	6.668E+09	1.448E+10
0.1	to	0.2	6.973E+09	7.694E+14	1.322E+10	5.011E+09
0.2	to	0.3	3.951E+08	1.780E+14	1.009E+09	2.950E+08
0.3	to	0.4	1.156E+09	1.402E+14	5.618E+09	9.387E+08
0.4	to	0.6	8.944E+09	4.105E+14	6.222E+10	8.238E+09
0.6	to	0.8	6.211E+09	1.178E+15	4.248E+10	5.518E+09
0.8	to	1.0	3.292E+11	1.053E+14	1.363E+11	1.397E+11
1.0	to	1.33	6.279E+12	5.387E+13	1.654E+12	4.178E+12
1.33	to	1.6	1.773E+12	2.333E+13	4.671E+11	1.180E+12
1.66	to	2.0	3.292E+06	3.102E+12	1.923E+06	3.448E+06
2.0	to	2.5	4.243E+07	1.926E+13	1.126E+07	2.824E+07
2.5	to	3.0	3.626E+04	1.580E+11	9.615E+03	2.413E+04
3.0	to	4.0	2.216E-07	1.325E+10	4.465E-08	2.445E-06
4.0	to	5.0	2.670E-30	6.005E+04	7.595E-29	9.688E-30
5.0	to	6.5	7.693E-31	2.405E+04	2.188E-29	2.791E-30
6.5	to	8.0	9.785E-32	4.710E+03	2.783E-30	3.551E-31
8.0	to	10.0	1.306E-32	9.988E+02	3.715E-31	4.738E-32
Total Gamma			8.571E+12	5.838E+15	2.680E+12	5.638E+12
Total Neutron			1.840E+06 n/s/assembly (raw) 2.440E+06 n/s/assembly (dry, treated with 1.326 peaking factor)			

Table T.5-18g
HSM-H Source Term, Zone 4

Burnup (GWd/MTU)			62	47	62	62
Enrichment (%)			3.8	2.9	3.8	3.8
Cooling Time (years)			1.3	1.0	1.3	1.3
E_{lower} (MeV)		E_{upper} (MeV)	Bottom Region (γ/s/assembly)	Fuel Region (γ/s/assembly)	Plenum Region (γ/s/assembly)	Top Region (γ/s/assembly)
0.01	to	0.05	2.038E+11	3.938E+15	5.417E+11	1.545E+11
0.05	to	0.1	2.716E+10	1.285E+15	9.644E+09	1.812E+10
0.1	to	0.2	9.343E+09	1.252E+15	2.064E+10	6.825E+09
0.2	to	0.3	6.479E+08	3.287E+14	2.110E+09	5.030E+08
0.3	to	0.4	3.397E+09	2.590E+14	1.895E+10	2.731E+09
0.4	to	0.6	1.725E+10	1.799E+15	9.793E+10	1.623E+10
0.6	to	0.8	4.105E+10	2.923E+15	2.545E+11	3.286E+10
0.8	to	1.0	6.918E+11	5.754E+14	2.839E+11	2.969E+11
1.0	to	1.33	7.720E+12	1.968E+14	2.036E+12	5.138E+12
1.33	to	1.6	2.180E+12	7.935E+13	5.749E+11	1.451E+12
1.66	to	2.0	5.836E+07	8.244E+12	4.801E+07	6.149E+07
2.0	to	2.5	5.258E+07	2.553E+13	1.699E+07	3.513E+07
2.5	to	3.0	4.491E+04	5.821E+11	1.440E+04	3.000E+04
3.0	to	4.0	3.748E-07	5.294E+10	7.700E-08	4.216E-06
4.0	to	5.0	3.125E-29	9.542E+06	1.049E-27	1.338E-28
5.0	to	6.5	9.006E-30	3.829E+06	3.023E-28	3.856E-29
6.5	to	8.0	1.145E-30	7.512E+05	3.845E-29	4.905E-30
8.0	to	10.0	1.529E-31	1.595E+05	5.132E-30	6.546E-31
Total Gamma			1.089E+13	1.267E+16	3.841E+12	7.117E+12
Total Neutron			2.750E+08 n/s/assembly (raw) 3.647E+08 n/s/assembly (dry, treated with 1.326 peaking factor)			

Table T.5-19
Shielding Material Densities

Assembly Region Material Densities

Element/ Isotope	Atomic Number	Number Density (atom/b-cm)				
		Bottom Fitting	Fuel (198 kgU)	*Fuel (170 kgU)*	Plenum	Top Fitting
C	6	5.846E-5	3.389E-7	*3.389E-7*	6.717E-6	1.396E-5
O	8	-	1.433E-2	*1.232E-2*	-	-
Si	14	3.208E-4	5.573E-6	*5.573E-6*	3.591E-5	1.123E-4
P	15	1.275E-5	7.392E-8	*7.392E-8*	1.465E-6	3.045E-6
Ti	22	4.876E-6	2.207E-6	*2.207E-6*	-	2.210E-5
Cr	24	3.239E-3	3.927E-5	*3.927E-5*	3.775E-4	8.903E-4
Mn	25	3.195E-4	1.852E-6	*1.852E-6*	3.671E-5	7.630E-5
Fe	26	1.077E-2	8.418E-5	*8.418E-5*	1.252E-3	2.624E-3
Ni	28	1.537E-3	6.079E-5	*6.079E-5*	1.632E-4	8.655E-4
Zr	40	2.534E-3	4.750E-3	*4.750E-3*	5.057E-3	1.335E-3
Sn	50	2.874E-5	5.388E-5	*5.388E-5*	5.736E-5	1.514E-5
Hf	72	1.318E-7	2.471E-7	*2.471E-7*	2.631E-7	6.946E-8
U-234	92	-	2.593E-6	*2.230E-6*	-	-
U-235	92	-	2.901E-4	*2.495E-4*	-	-
U-236	92	-	1.329E-6	*1.143E-6*	-	-
U-238	92	-	6.872E-3	*5.909E-3*	-	-
Total		1.882E-2	2.649E-2	*2.348E-2*	6.989E-3	5.958E-3

Table T.5-19
Shielding Material Densities

(Continued)

HSM Shielding Materials

Element/ Isotope	Atomic Number	Number Density (atom/b-cm)				
		Concrete[(1)]	Air	Carbon Steel	Stainless Steel	Aluminum/ BORAL
H	1	7.758E-3				
C	6		8.423E-9	3.907E-3		
N	7		3.897E-5			
O	8	4.312E-2	1.047E-5			
Na	11	1.021E-3				
Al	13	2.340E-3				6.031E-2
Si	14	1.557E-2				
Ar	18		2.330E-7			
K	19	6.768E-4				
Ca	20	2.852E-3				
Cr	24				1.743E-2	
Mn	25				1.736E-3	
Fe	26	3.015E-4		8.348E-2	5.935E-2	
Ni	28				7.720E-3	
Total		7.364E-2	4.969E-5	8.739E-2	8.624E-2	6.031E-2

Note (1): The concrete number densities reflect a density of 2.29 g/cm^3 , which is used in the HSM Model 102 analysis. In the HSM-H models, concrete is conservatively modeled at 2.243 g/cm^3 (140 pcf).

Table T.5-19
Shielding Material Densities
(Concluded)
TC Shielding Materials

Element/ Isotope	Atomic Number	Number Density (atom/b-cm)						
		NS-3	Water	Air	Lead	Carbon Steel	Stainless Steel	Aluminum/ BORAL
H	1	4.498E-2	*6.406E-2*					
B-10	5	6.077E-5						
B-11	5	2.446E-4						
C	6	9.595E-3		8.642E-9		3.939E-3	3.188E-4	
N	7			3.995E-5				
O	8	3.704E-2	*3.203E-2*	1.073E-5				
Al	13	6.887E-3						6.031E-2
Si	14	1.243E-3					1.702E-3	
P	15						6.947E-5	
Ar	18			2.388E-7				
Ca	20	1.454E-3						
Cr	24						1.747E-2	
Mn	25						1.741E-3	
Fe	26	1.042E-4				8.380E-2	5.854E-2	
Ni	28						7.739E-3	
Pb	82				3.296E-2			
Total		1.016E-1	*9.609E-2*	5.093E-5	3.296E-2	8.774E-2	8.758E-2	6.031E-2

Table T.5-25a
OS197FC-B MCNP Response Functions, Middle of Side Surface

E_{max} (MeV)	Zone 1 (mrem/hr)	Zone 2 (mrem/hr)	Zone 3 (mrem/hr)	Zone 4 (mrem/hr)
Primary Gamma				
0.05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.1	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.2	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.3	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.4	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.6	0.00E+00	2.14E-18	1.28E-17	9.73E-18
0.8	1.98E-17	3.09E-16	2.42E-15	3.01E-15
1	7.62E-16	8.03E-15	5.18E-14	6.39E-14
1.33	2.04E-14	1.24E-13	6.50E-13	7.99E-13
1.66	1.69E-13	7.61E-13	3.39E-12	4.18E-12
2	6.09E-13	2.30E-12	9.09E-12	1.12E-11
2.5	1.64E-12	5.39E-12	1.93E-11	2.38E-11
3	3.32E-12	9.87E-12	3.31E-11	4.06E-11
4	6.00E-12	1.64E-11	5.11E-11	6.25E-11
5	8.71E-12	2.23E-11	6.76E-11	8.22E-11
6.5	9.93E-12	2.59E-11	7.66E-11	9.31E-11
8	1.11E-11	2.76E-11	8.22E-11	9.94E-11
10	1.13E-11	2.91E-11	8.59E-11	1.04E-10
Neutron	3.03E-07	2.08E-07	2.77E-07	3.17E-07
Secondary Gamma	1.62E-07	9.06E-08	9.94E-08	1.04E-07

Note 1: To use the response functions, multiply the source for a single fuel assembly by the response function. The product is the dose rate from the entire zone (i.e., the input fuel assembly in all zone locations).

Note 2: Subcritical neutron multiplication included in the neutron and secondary gamma response functions.

Table T.5-25b
HSM-H MCNP Response Functions, Roof Surface

E_{max} (MeV)	Zone 1 (mrem/hr)	Zone 2 (mrem/hr)	Zone 3 (mrem/hr)	Zone 4 (mrem/hr)
Primary Gamma				
0.05	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.1	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.2	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.3	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.4	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.6	0.00E+00	0.00E+00	8.93E-19	7.41E-19
0.8	0.00E+00	3.97E-18	4.49E-17	4.39E-17
1	2.66E-18	4.94E-17	2.94E-16	3.88E-16
1.33	1.05E-16	5.93E-16	3.11E-15	3.91E-15
1.66	1.05E-15	4.46E-15	1.97E-14	2.53E-14
2	4.97E-15	1.94E-14	7.86E-14	9.95E-14
2.5	2.28E-14	7.77E-14	2.89E-13	3.64E-13
3	7.76E-14	2.43E-13	8.50E-13	1.08E-12
4	2.77E-13	8.09E-13	2.72E-12	3.46E-12
5	7.44E-13	2.16E-12	7.11E-12	9.16E-12
6.5	1.62E-12	4.67E-12	1.53E-11	2.00E-11
8	2.60E-12	8.02E-12	2.69E-11	3.58E-11
10	3.75E-12	1.19E-11	4.07E-11	5.45E-11
Neutron	2.25E-10	1.57E-10	2.16E-10	2.46E-10
Secondary Gamma	5.52E-10	3.22E-10	3.72E-10	3.98E-10

Note 1: To use the response functions, multiply the source for a single fuel assembly by the response function. The product is the dose rate from the entire zone (i.e., the input fuel assembly in all zone locations).

Note 2: Subcritical neutron multiplication included in the neutron and secondary gamma response functions.

Table T.5-29

Distribution of BWR Assemblies from 2013 EIA GC-859 Database

Burn-Up, GWd/ MTU	*Assembly Averaged Initial ^{235}U Enrichment, wt.%*																																							
	0.6	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	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
1																			1	2																				
2		50																	1	5											1									
3		365				2										2			7									1												
4		579														3	11		6	1									2											
5		298				1									1	2	19		11	38									8	1	2	6					1			
6		44		32		9									1	22	39		17	2	1	1	1				3				2	10								
7				40		135									2	57	3	2	19	20		1				1	3	1	4			4								
8				132	6	99									1	28	9	1	7	27			1				5		8					2						
9				156	17	103						2			21	50	27		31	29	9		2				7	1			2									
10				4	23	272				116		6	1		3	74	32		33	96	30		6				8			4	11					1				
11				8	70	305			8	144	8	8			2	95	54	1	29	84	7		14			2	32	2	2	3	4		1							
12				7	13	76			12	77		304			2	227	50	9	4	7	4		36		1	12	71		3		10		2	2						
13				52						168	4	171			12	441	105	2	4	14	26		110	2	2	11	114		6		15		7	5						
14				58						114	20	508			27	140	115	1	4	30	23		37	2	1	2	54	3	7	1	34	1	6	9	1		1			
15				15						24	128	197		2	137	141	108	3	9	25	8	2	14		2		57	1	4	8	30		6	17		1				
16				14							161	197	3		51	263	116	13	13	22	2	2	26	1		1	4		2	4	36		29	17						
17				18						32	50	289	12	6	53	662	95	35	4	35	5	17	12		1	2				6	62		46	7	1	1				
18				3						28	61	151	49		140	808	144	46	1	27	32	7	20			3			2	12	57		64	6						
19				5					1	8	80	249	57	6	275	726	154	92	4	121	52	19	4	1		2	1		5	9	24		3	6						
20									3		60	59	30	18	99	1047	229	111	9	417	96	20	7			1			5	13	11		3	10	1			1		
21									7		21	27	7	57	137	774	309	135	20	261	131	54	38	1		6	1	5	6	15	6		7	4	1					
22									1		35	20	3	4	85	943	209	143	44	264	164	112	82	7	1	3	4	6	7	6	1		21	7	4					
23									1		22	16	1		90	746	187	49	202	541	246	94	111	15	3	17	5	1	32	1		3	29	5	2	2				
24									8		26	24	4		80	574	302	75	224	609	434	223	203	12	5	32	3	1	48			5	34		1	3	1			
25									21	3	20	9	4	6	76	522	335	51	210	559	400	395	317	21	5	80	3	7	35			5	30	1	1					
26									7	5	16	8	1	17	103	460	152	45	300	649	439	442	474	83	30	52	11	6	34			3	92							
27											2	16	1	11	92	402	150	40	257	551	536	435	550	200	58	67	37	6	22			3	78	3	2	2				
28											6	6	1	27	3	274	181	31	265	492	752	457	1006	185	149	86	16	26	4	1		10	35	1		4				
29												7	1	15	8	169	139	35	273	348	988	289	1110	271	229	110	43	40	15			1	28		3					
30										1				24	12	126	60	7	135	183	925	276	1184	484	377	112	44	124	53	2			13		1	2				
31														24	4	38	41		144	60	797	229	1084	487	334	156	150	147	119	29	2	50	8	1	10	8				
32														3		24	17	2	36	76	686	206	1002	711	381	213	270	399	229	50	22	38	18	2	36	42		1		
33											4					9	5	1	12	76	507	177	707	879	497	391	207	484	297	54	14	74	53	3	15	1	4			
34																7			6	20	197	114	631	865	671	406	418	525	251	94	9	161	78	9	7	10	8			
35															24				1	8	105	89	307	612	451	495	776	431	310	152	18	63	125	22	11	6		1		
36																			24	4	3	81	127	404	470	731	1094	495	427	179	21	29	130	29	52	43	11	1		
37														4			1		14	4	2	34	79	269	373	892	1120	613	512	193	65	95	65	46	70	47	12	14		
38														14					11	6	1	4	36	286	468	565	909	594	484	196	72	184	160	211	220	90	68	13		
39														3						16	7	16	33	225	421	436	824	359	455	234	104	309	232	369	193	158	128	19	1	
40																					2		18	54	149	305	736	342	393	286	180	429	263	359	241	115	164	27		
41										1										4	2		34	25	77	132	508	454	381	284	381	500	356	405	405	147	165	91	1	
42																							7	47	41	111	346	263	287	506	385	532	365	709	632	368	196	39	2	
43																					1		7	51	19	72	288	151	245	430	266	407	507	761	1032	568	228	64	15	
44																					4			16	4	20	172	70	234	240	174	453	474	867	1537	1043	361	49	47	
45																									8	31	162	20	121	171	293	521	358	790	1192	1244	294	54	57	
46																										14	87	27	57	99	301	504	379	760	1009	913	242	37	8	
47																										3	62	27	20	94	226	460	364	704	1053	1180	158	43	8	
48																										1	24	18	5	81	134	266	219	674	1001	934	93	21	13	
49																											3	18	17	41	113	309	230	597	923	910	79	27	22	
50																											2	17	18	19	39	187	234	424	721	824	72	1	1	
51																											2	39	4	20	26	150	95	219	519	670	29	1		
52																											2	22		6	18	47	41	101	288	382	4			
53																											1	24		8	24	9	38	26	101	147				
54																												4		32	8	2		9	22	18				
55																														8					2	1				
56																																								
Enr.-->	0.6	0.7	0.8	0.9	1.0	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	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

Note: The heavy line represents the analyzed lower enrichment boundary.

Table T.5-30
OS197FC-B Response Function Results, Zone 4

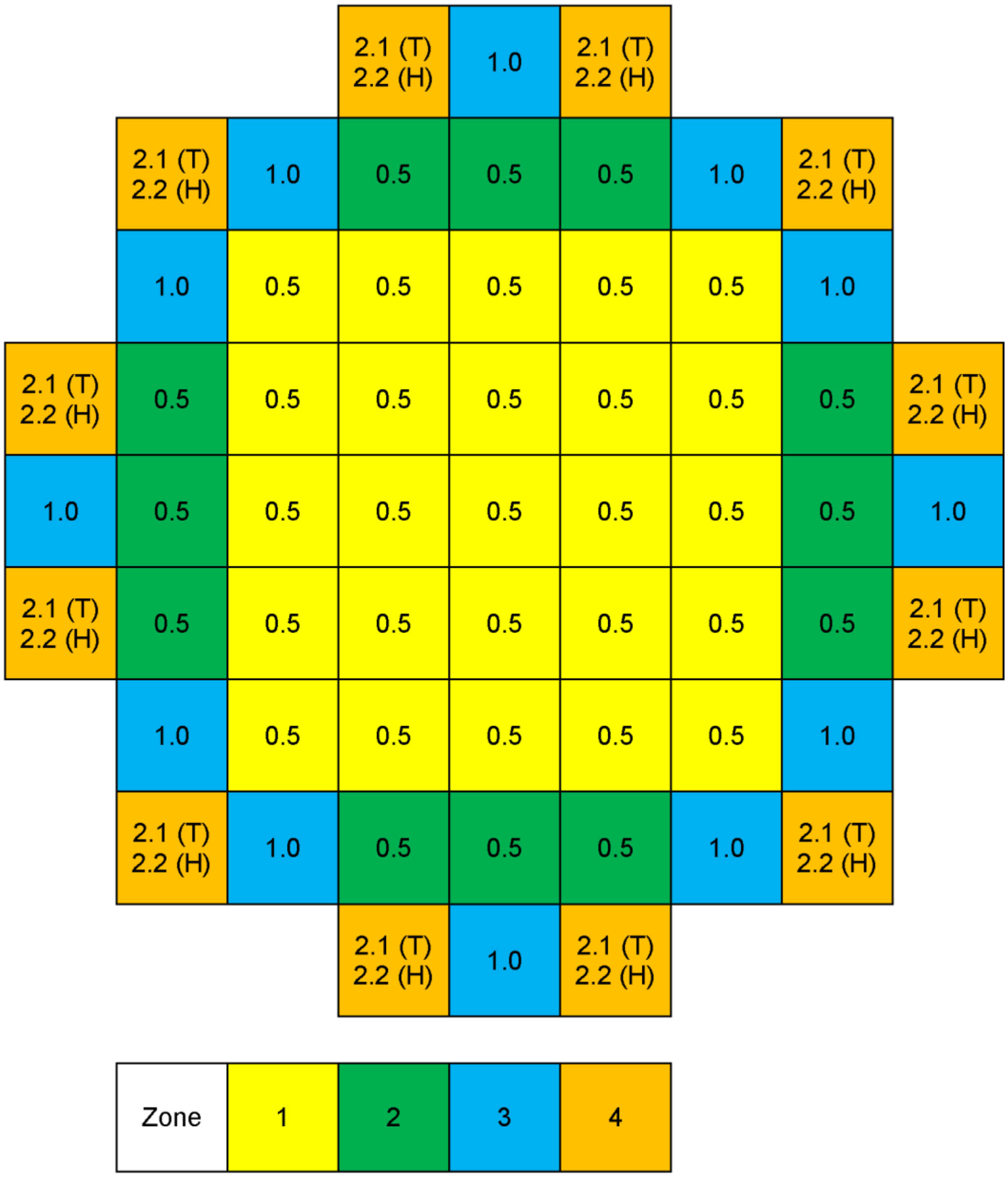
(Response function dose rate, side of OS197FC-B, mrem/hr)

BU (GWd/MTU)	Assembly Average Initial ^{235}U Enrichment, wt.%																					
	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
36	1165	1155	1145	1136	1127	1119	1111	1103	1096	1089	1082	1076	1070	1065	1059	1054	1049	1045	1040	1036	1032	1028
37	1191	1181	1171	1161	1151	1143	1134	1126	1119	1111	1104	1098	1092	1086	1080	1074	1069	1064	1059	1055	1051	1046
38	1218	1207	1196	1186	1176	1167	1158	1150	1142	1134	1127	1120	1113	1107	1100	1095	1089	1084	1079	1074	1069	1065
39	1245	1233	1222	1211	1201	1191	1182	1173	1165	1156	1149	1141	1134	1128	1121	1115	1109	1104	1098	1093	1088	1084
40	1272	1260	1248	1237	1226	1216	1206	1197	1188	1179	1171	1163	1156	1149	1142	1135	1129	1123	1118	1112	1107	1102
41	1300	1287	1274	1263	1251	1241	1230	1220	1211	1202	1194	1185	1177	1170	1163	1156	1149	1143	1137	1131	1126	1120
42	1327	1314	1301	1289	1277	1266	1255	1245	1235	1225	1216	1208	1199	1191	1184	1177	1170	1163	1157	1151	1145	1139
43	1356	1342	1329	1316	1303	1292	1280	1270	1259	1249	1240	1230	1222	1214	1206	1198	1191	1184	1177	1170	1164	1159
44	1384	1370	1356	1343	1330	1317	1306	1294	1283	1273	1263	1254	1244	1236	1227	1219	1211	1204	1197	1190	1184	1177
45		1398	1384	1370	1356	1344	1331	1319	1308	1297	1287	1277	1267	1258	1249	1240	1232	1225	1217	1210	1203	1197
46		1427	1412	1397	1383	1370	1357	1345	1333	1322	1311	1300	1290	1280	1271	1262	1254	1245	1238	1230	1223	1216
47			1440	1425	1411	1397	1383	1370	1358	1346	1335	1324	1313	1303	1293	1284	1275	1266	1258	1251	1243	1236
48				1410	1395	1381	1367	1354	1341	1329	1317	1306	1295	1284	1274	1264	1255	1246	1238	1230	1222	1214
49				1438	1423	1408	1394	1380	1367	1354	1342	1330	1318	1307	1297	1287	1277	1268	1259	1250	1242	1234
50					1408	1393	1378	1364	1351	1337	1325	1312	1301	1290	1279	1268	1258	1248	1239	1230	1222	1213
51					1436	1420	1405	1390	1376	1363	1349	1336	1324	1313	1301	1290	1280	1270	1260	1251	1242	1233
52						1407	1391	1376	1362	1348	1334	1321	1308	1296	1285	1273	1262	1252	1242	1232	1223	1214
53							1419	1403	1388	1373	1359	1346	1333	1320	1308	1296	1285	1274	1263	1253	1243	1234
54							1407	1391	1375	1360	1346	1332	1318	1305	1293	1280	1269	1258	1247	1236	1226	1216
55								1418	1402	1387	1371	1357	1343	1329	1316	1304	1292	1280	1268	1258	1247	1237
56									1351	1335	1320	1305	1290	1276	1263	1250	1238	1226	1214	1203	1192	1182
57									1380	1363	1347	1332	1317	1302	1289	1275	1262	1250	1238	1226	1215	1204
58										1394	1378	1362	1346	1331	1317	1303	1289	1276	1264	1252	1240	1229
59										1387	1370	1353	1338	1322	1307	1293	1279	1266	1253	1240	1228	1217
60											1400	1383	1366	1350	1335	1320	1305	1292	1278	1265	1253	1241
61												1411	1394	1378	1362	1346	1331	1317	1303	1290	1277	1264
62												1441	1423	1406	1390	1374	1359	1344	1329	1315	1302	1289

Table T.5-31
HSM-H Response Function Results, Zone 4

(Response function dose rate, roof of HSM-H, mrem/hr)

BU (GWd/MTU)	Assembly Average Initial ^{235}U Enrichment, wt.%																						
	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
36	13.0	13.0	12.9	12.9	12.8	12.8	12.7	12.7	12.7	12.6	12.6	12.5	12.5	12.5	12.5	12.4	12.4	12.4	12.3	12.3	12.3	12.3	12.2
37		13.2	13.1	13.1	13.0	13.0	12.9	12.9	12.8	12.8	12.8	12.7	12.7	12.6	12.6	12.6	12.6	12.5	12.5	12.5	12.4	12.4	12.4
38		13.3	13.3	13.2	13.2	13.1	13.1	13.0	13.0	13.0	12.9	12.9	12.8	12.8	12.8	12.7	12.7	12.7	12.7	12.6	12.6	12.6	12.6
39			13.5	13.4	13.4	13.3	13.3	13.2	13.2	13.1	13.1	13.0	13.0	13.0	12.9	12.9	12.9	12.8	12.8	12.8	12.8	12.7	12.7
40				13.6	13.5	13.5	13.4	13.4	13.3	13.3	13.2	13.2	13.2	13.1	13.1	13.1	13.0	13.0	13.0	12.9	12.9	12.9	12.8
41				13.8	13.7	13.6	13.6	13.5	13.5	13.4	13.4	13.4	13.3	13.3	13.2	13.2	13.2	13.1	13.1	13.1	13.0	13.0	13.0
42					13.9	13.8	13.7	13.7	13.6	13.6	13.6	13.5	13.5	13.4	13.4	13.3	13.3	13.3	13.2	13.2	13.2	13.1	13.1
43					14.0	14.0	13.9	13.9	13.8	13.8	13.7	13.7	13.6	13.6	13.5	13.5	13.5	13.4	13.4	13.3	13.3	13.3	13.3
44						14.1	14.1	14.0	14.0	13.9	13.9	13.8	13.8	13.7	13.7	13.6	13.6	13.6	13.5	13.5	13.5	13.4	13.4
45							14.2	14.2	14.1	14.0	14.0	14.0	13.9	13.9	13.8	13.8	13.7	13.7	13.7	13.6	13.6	13.5	13.5
46							14.4	14.3	14.3	14.2	14.1	14.1	14.0	14.0	14.0	13.9	13.9	13.8	13.8	13.7	13.7	13.7	13.6
47								14.4	14.4	14.3	14.3	14.2	14.2	14.1	14.1	14.0	14.0	14.0	13.9	13.9	13.8	13.8	13.8
48									14.0	14.0	13.9	13.8	13.8	13.7	13.7	13.7	13.6	13.6	13.5	13.5	13.4	13.4	13.4
49									14.1	14.1	14.0	14.0	13.9	13.9	13.8	13.8	13.7	13.7	13.6	13.6	13.6	13.5	13.5
50										13.7	13.7	13.6	13.5	13.5	13.4	13.4	13.4	13.3	13.3	13.2	13.2	13.1	13.1
51										13.8	13.8	13.7	13.7	13.6	13.6	13.5	13.5	13.4	13.4	13.3	13.3	13.2	13.2
52											13.4	13.4	13.3	13.2	13.2	13.1	13.1	13.0	13.0	13.0	12.9	12.9	12.8
53												13.5	13.4	13.4	13.3	13.3	13.2	13.2	13.1	13.1	13.0	13.0	12.9
54												13.1	13.1	13.0	13.0	12.9	12.9	12.8	12.8	12.7	12.7	12.6	12.6
55													13.2	13.1	13.1	13.0	13.0	12.9	12.9	12.8	12.8	12.7	12.7
56														12.3	12.3	12.2	12.2	12.1	12.1	12.0	12.0	11.9	11.9
57														12.5	12.4	12.3	12.3	12.2	12.2	12.1	12.1	12.1	12.0
58															12.6	12.5	12.5	12.4	12.3	12.3	12.2	12.2	12.1
59															12.3	12.2	12.1	12.1	12.0	12.0	11.9	11.9	11.8
60																12.3	12.3	12.2	12.2	12.1	12.1	12.0	12.0
61																	12.4	12.3	12.3	12.2	12.2	12.1	12.1
62																	12.5	12.5	12.4	12.4	12.3	12.3	12.2



Note: Heat displayed in kW. This configuration bounds HLZC 1 through 13 for shielding performance. The decay heat in zone 4 is approximately 2.1 kW in the OS197FC-B TC models (T) and 2.2 kW in the HSM-H models (H).

Figure T.5-1a
Heat Load Zoning Configuration Utilized for HSM-H and OS197FC-B Evaluation

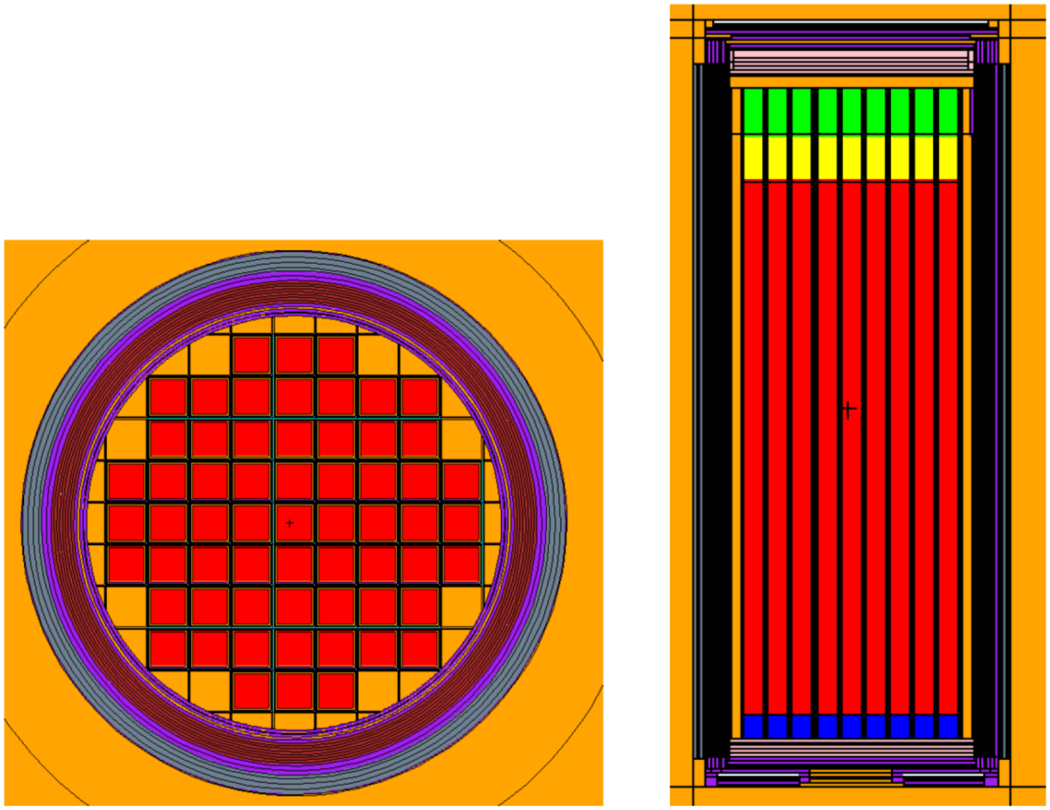


Figure T.5-2c
MCNP OS197FC-B Response Function Model

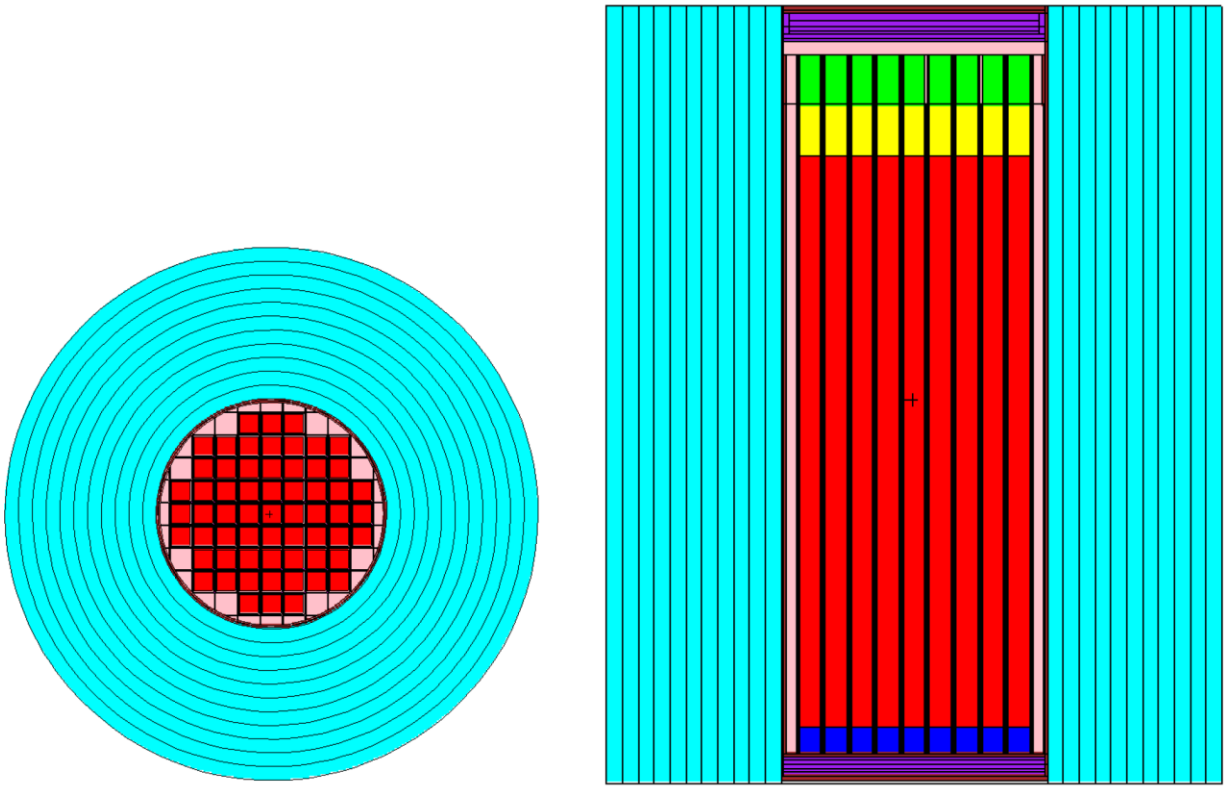
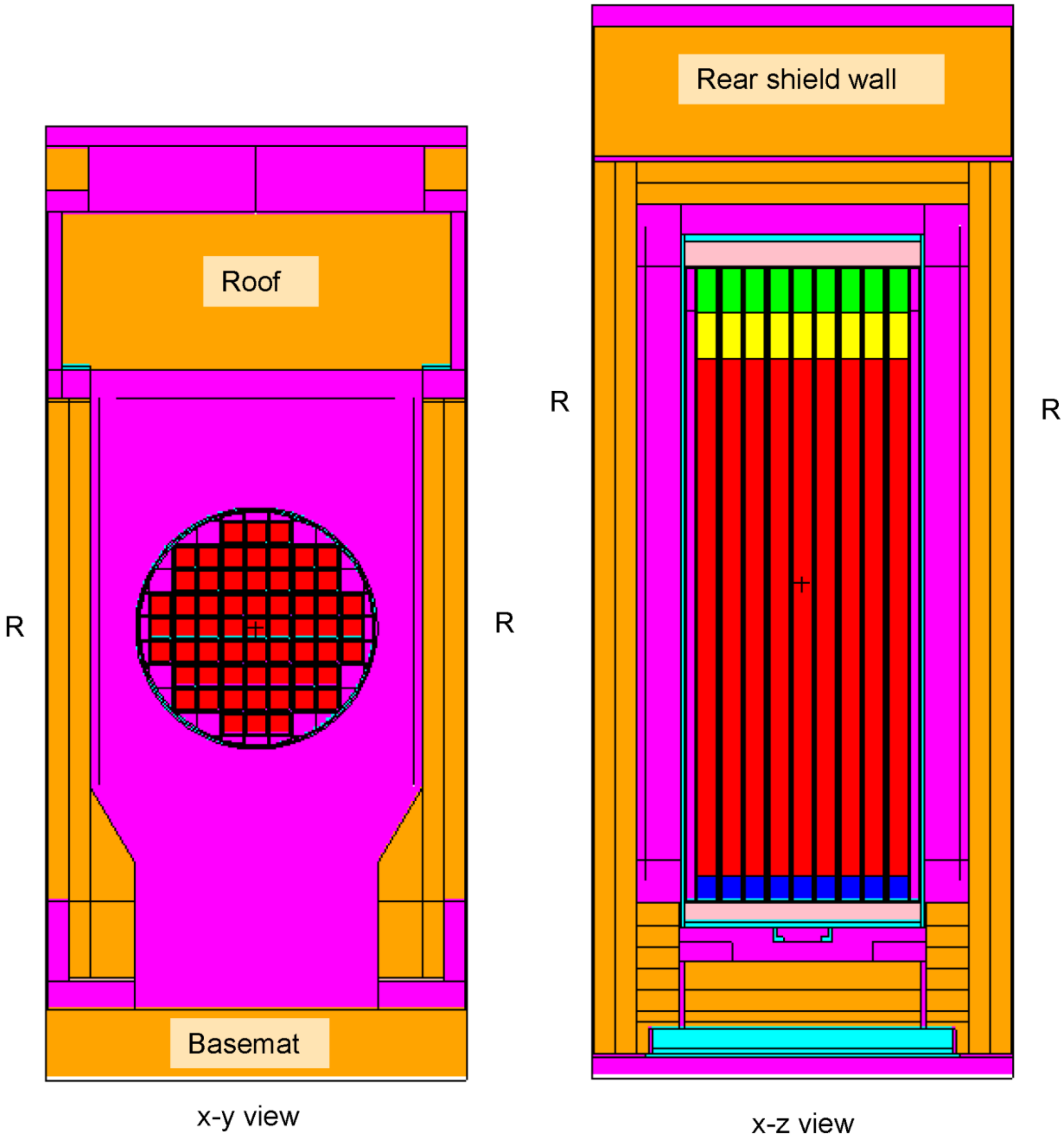


Figure T.5-2d
MCNP HSM-H Response Function Model



Reflective boundaries are marked "R."

Figure T.5-3
MCNP HSM-H Rear Model

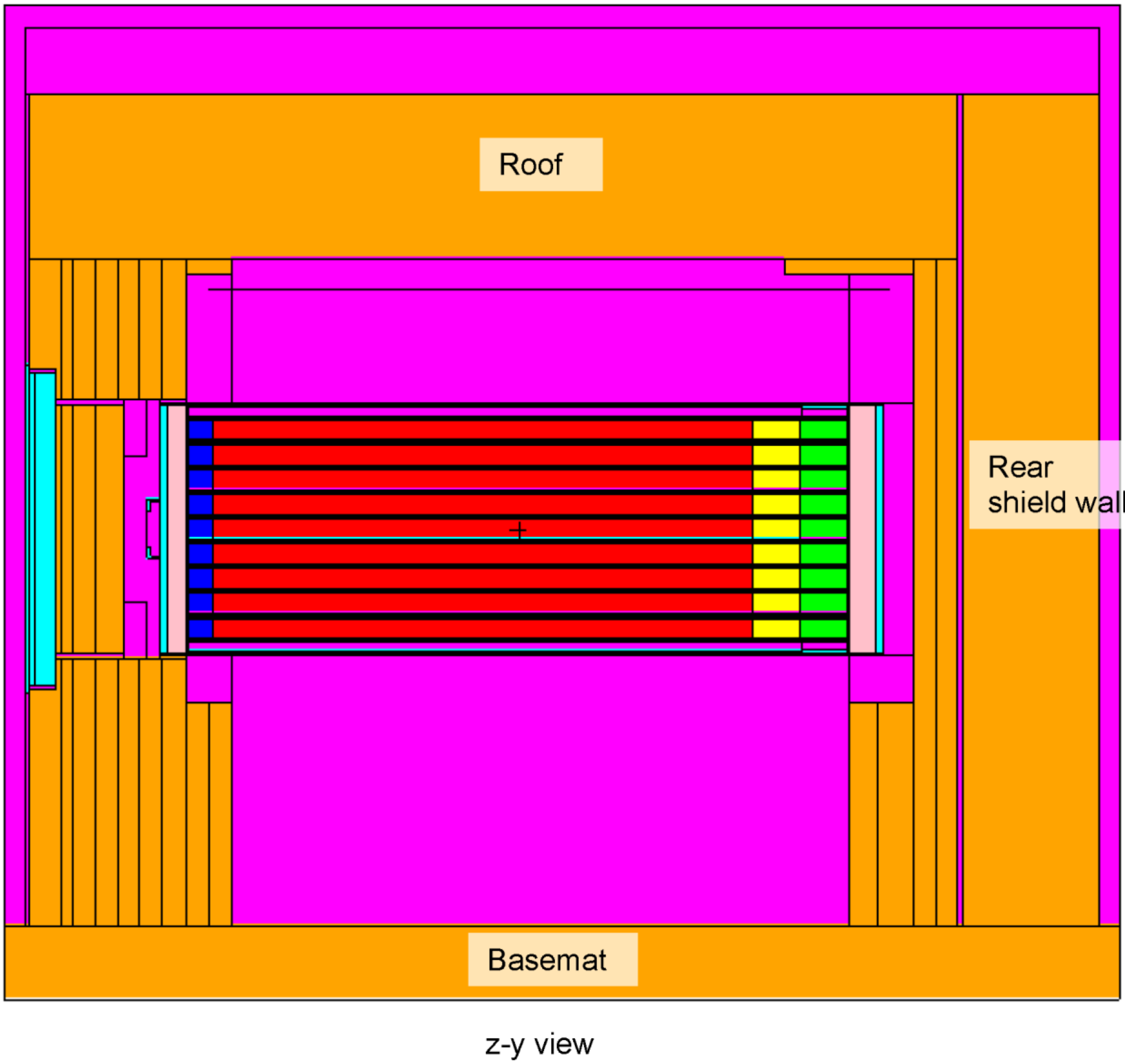
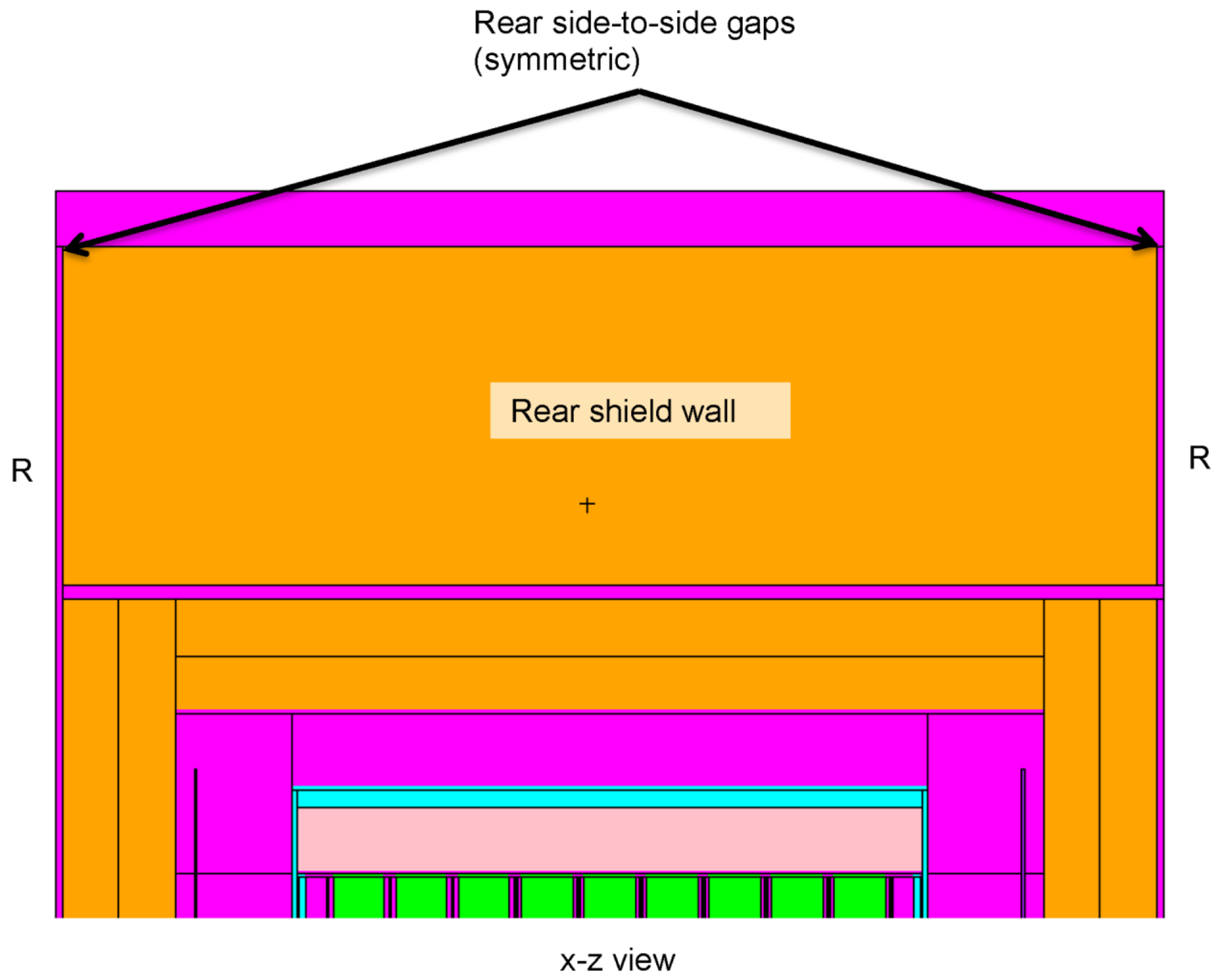


Figure T.5-4
MCNP HSM-H Rear Model (z-y View)



Reflective boundaries are marked “R.”

Figure T.5-5
MCNP HSM-H Rear Model, Close-Up Showing Gaps

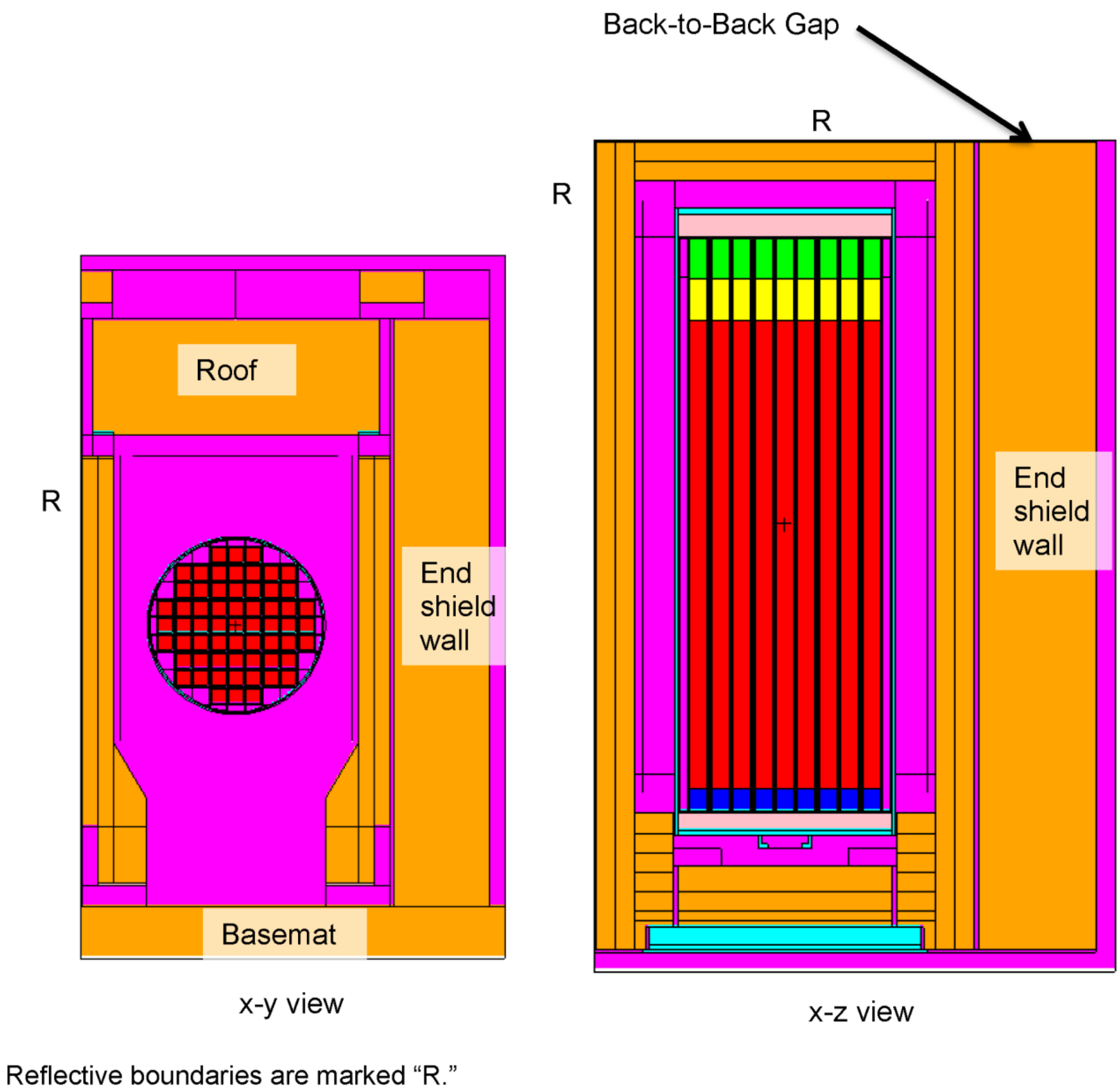
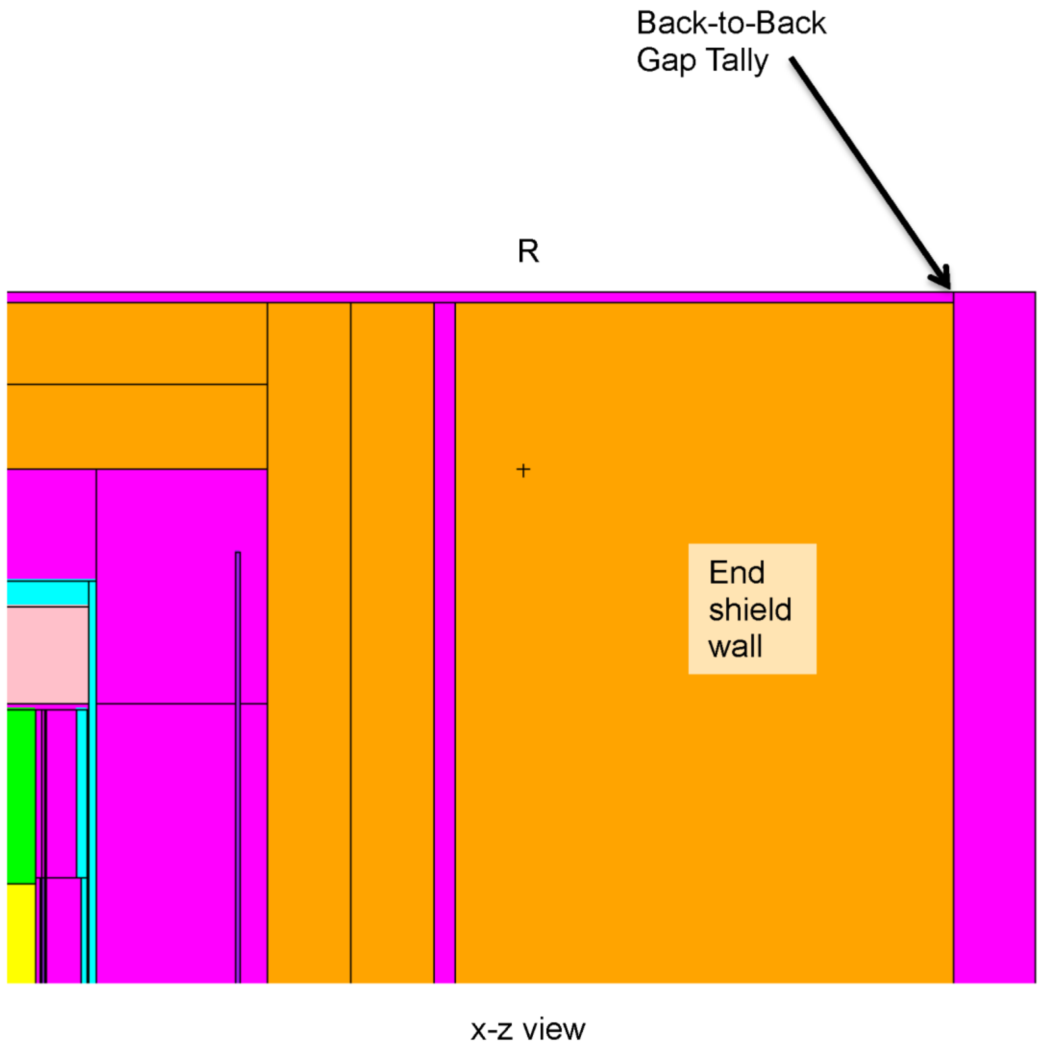


Figure T.5-6
MCNP HSM-H Side Model



Reflective boundaries are marked “R.”

Figure T.5-7
MCNP HSM-H Side Model, Close-Up Showing Gap

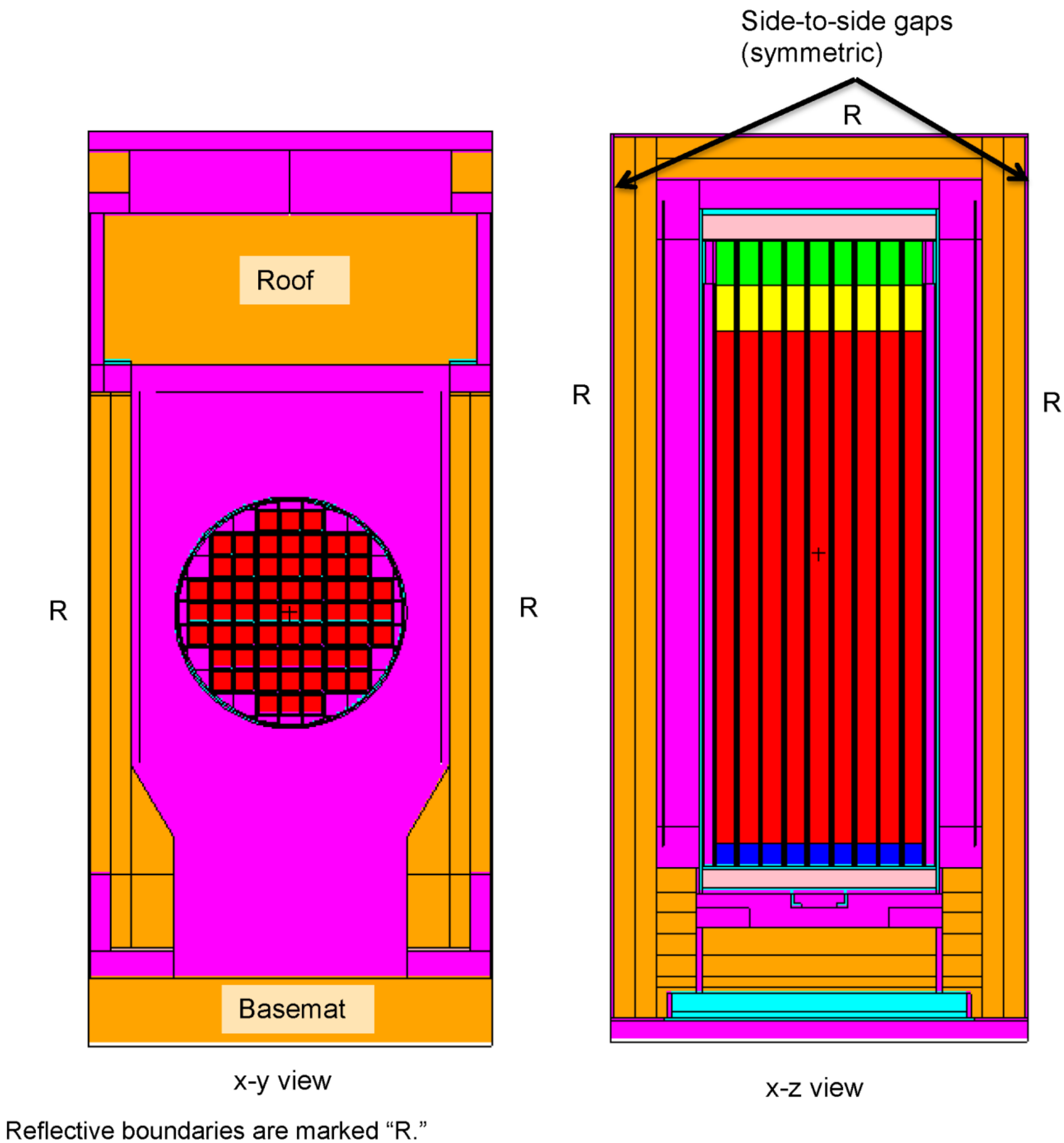
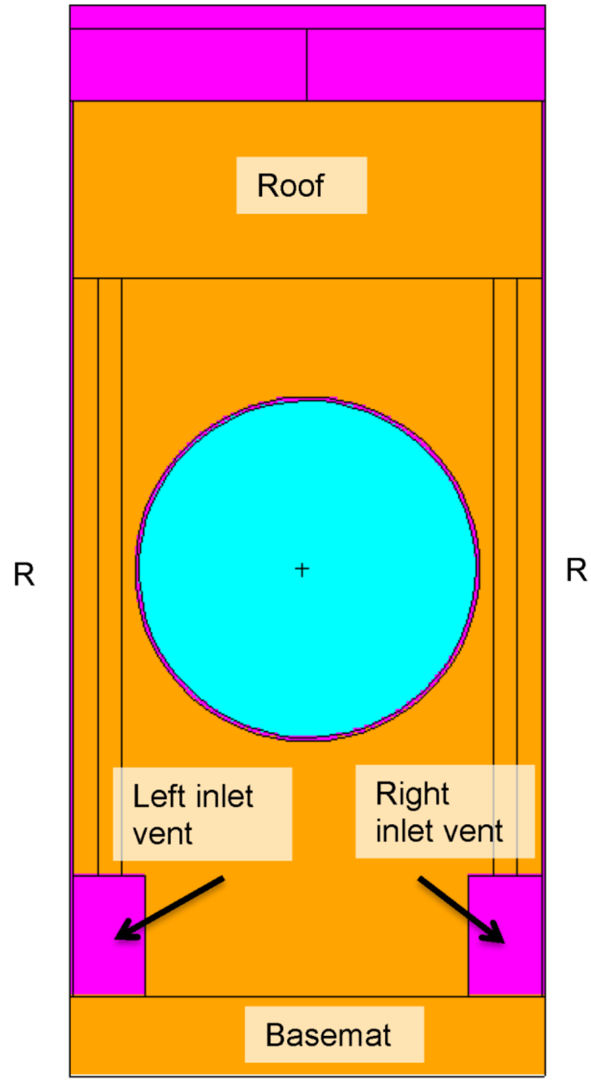


Figure T.5-7a
MCNP HSM-H Front/Roof Model



x-y view

Reflective boundaries are marked "R."

Figure T.5-7b
MCNP HSM-H Front/Roof Model Showing Inlet Vents

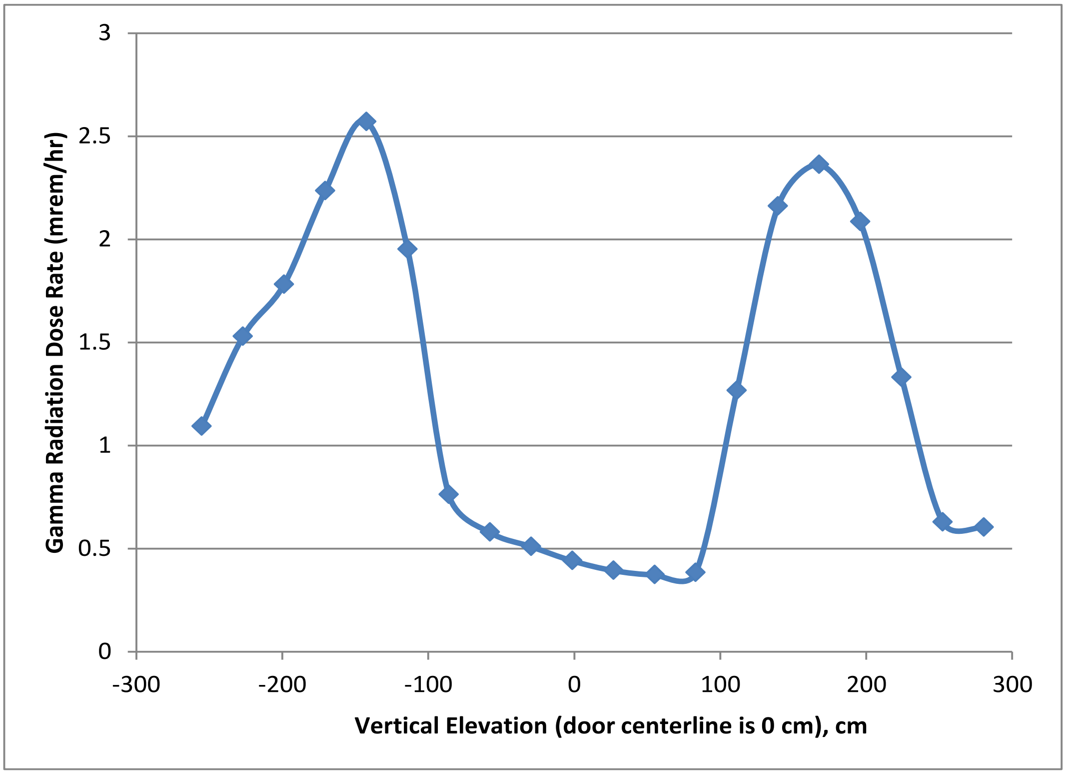


Figure T.5-17
HSM-H with 61BTH DSC, Gamma Radiation Dose Rate along HSM-H Front Centerline in Vertical Elevation

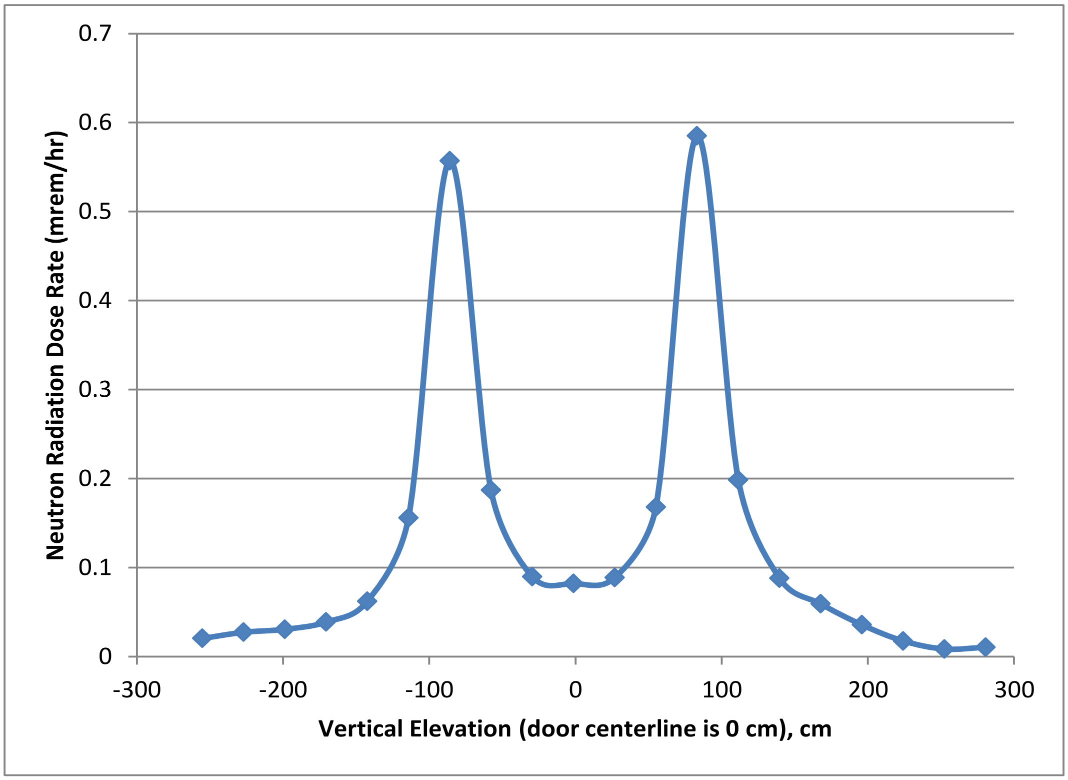
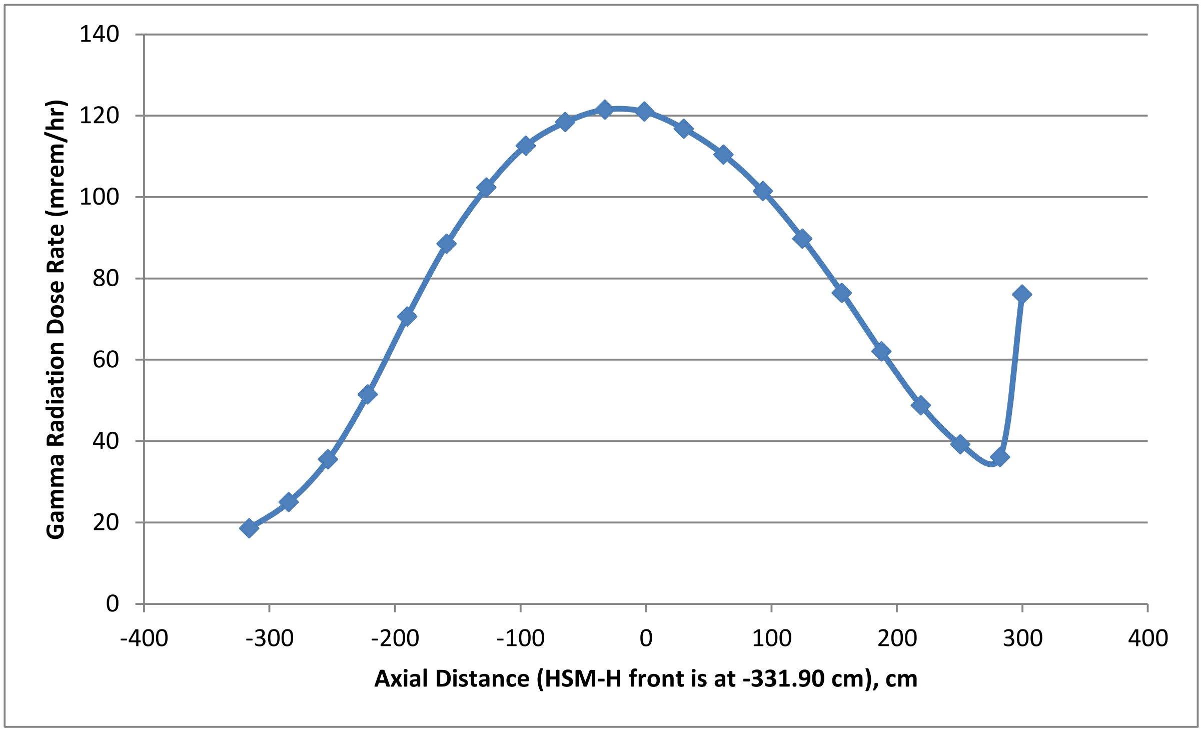
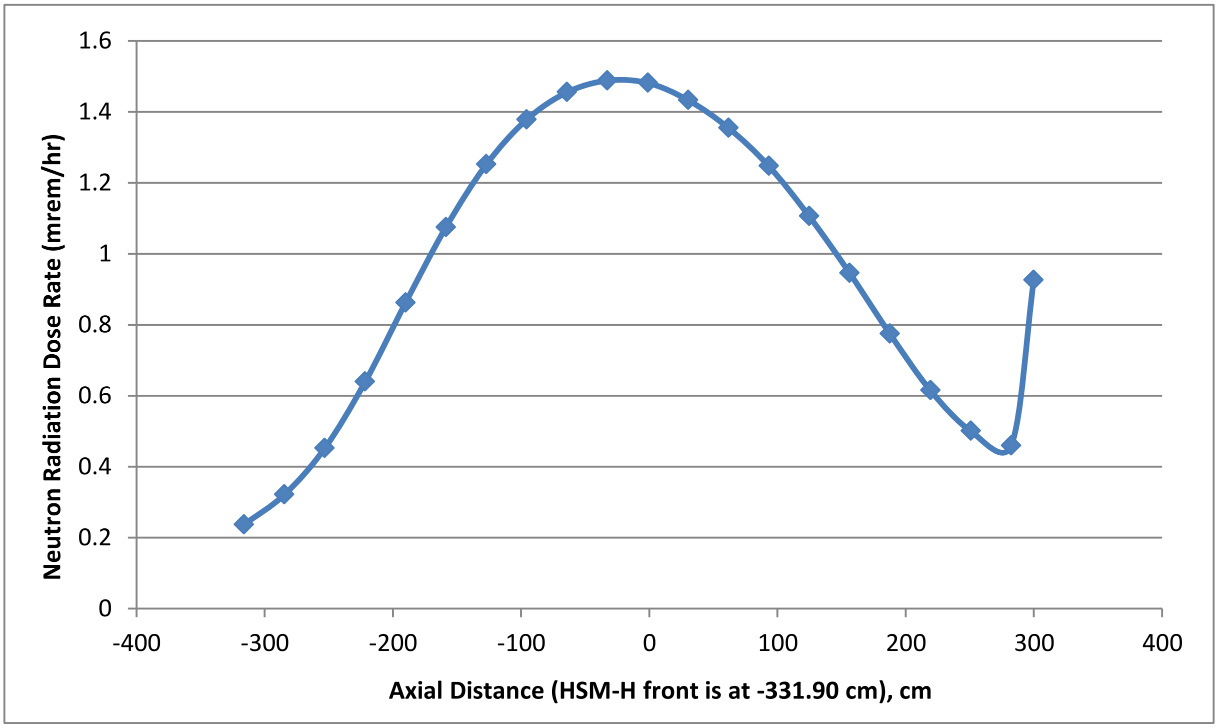


Figure T.5-18
HSM-H with 61BTH DSC, Neutron Radiation Dose Rate along HSM-H Front Centerline in Vertical Elevation



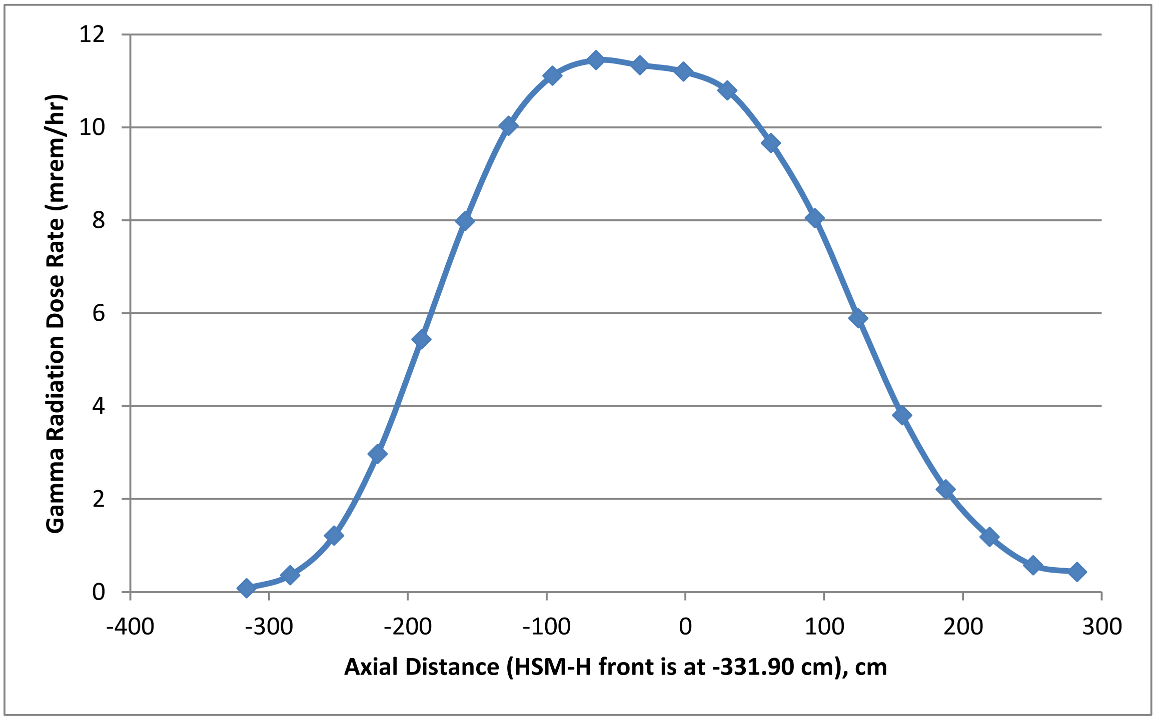
The dose rate increase at z = 300 cm is due to the assumed gap.

Figure T.5-19
HSM-H with 61BTH DSC, Gamma Radiation Dose Rate at Roof Centerline



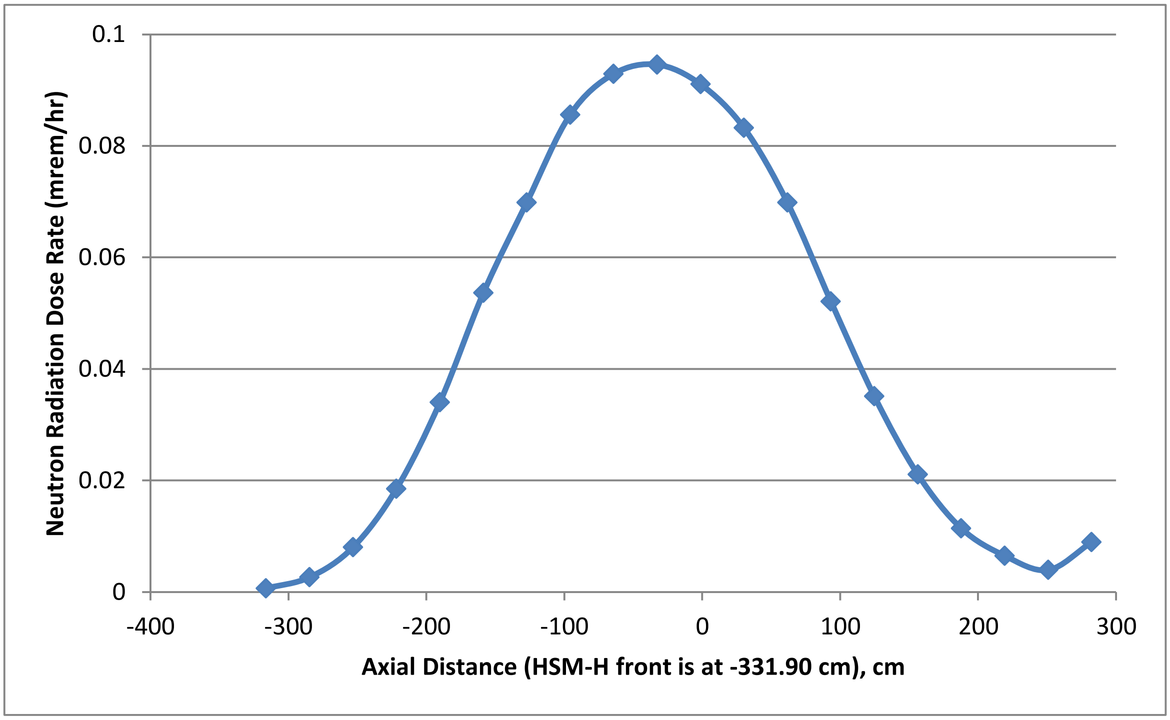
The dose rate increase at z = 300 cm is due to the assumed gap.

Figure T.5-20
HSM-H with 61BTH DSC, Neutron Dose Rate at Roof Centerline



The peak gamma dose rate at the assumed gap (42.4 mrem/hr at z = 300 cm) is not shown on the figure.

Figure T.5-21
HSM-H with 61BTH DSC, Side Shield Wall Surface at DSC Centerline Gamma Radiation Dose Rate



The peak neutron dose rate at the assumed gap (0.635 mrem/hr at z = 300 cm) is not shown on the figure.

Figure T.5-22
HSM-H with 61BTH DSC, Side Shield Wall Surface at DSC Centerline Neutron Radiation Dose Rate

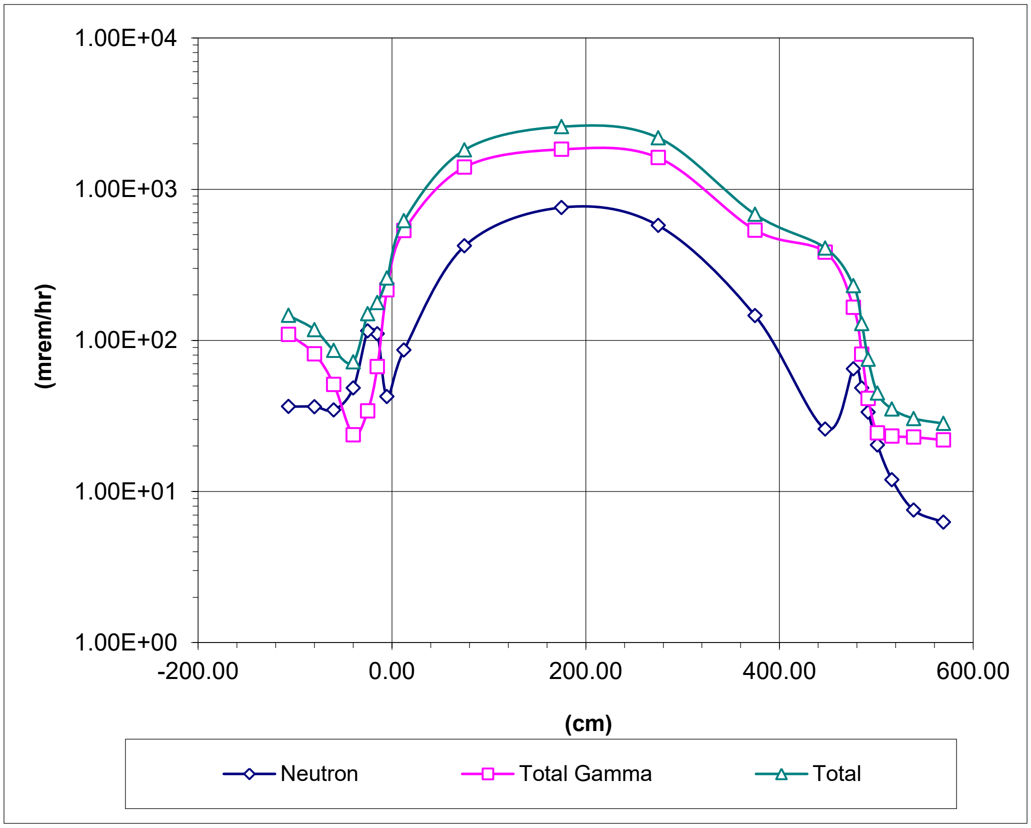


Figure T.5-23
OS197FC-B TC with 61BTH DSC, Side Surface Dose Rate, Normal Conditions

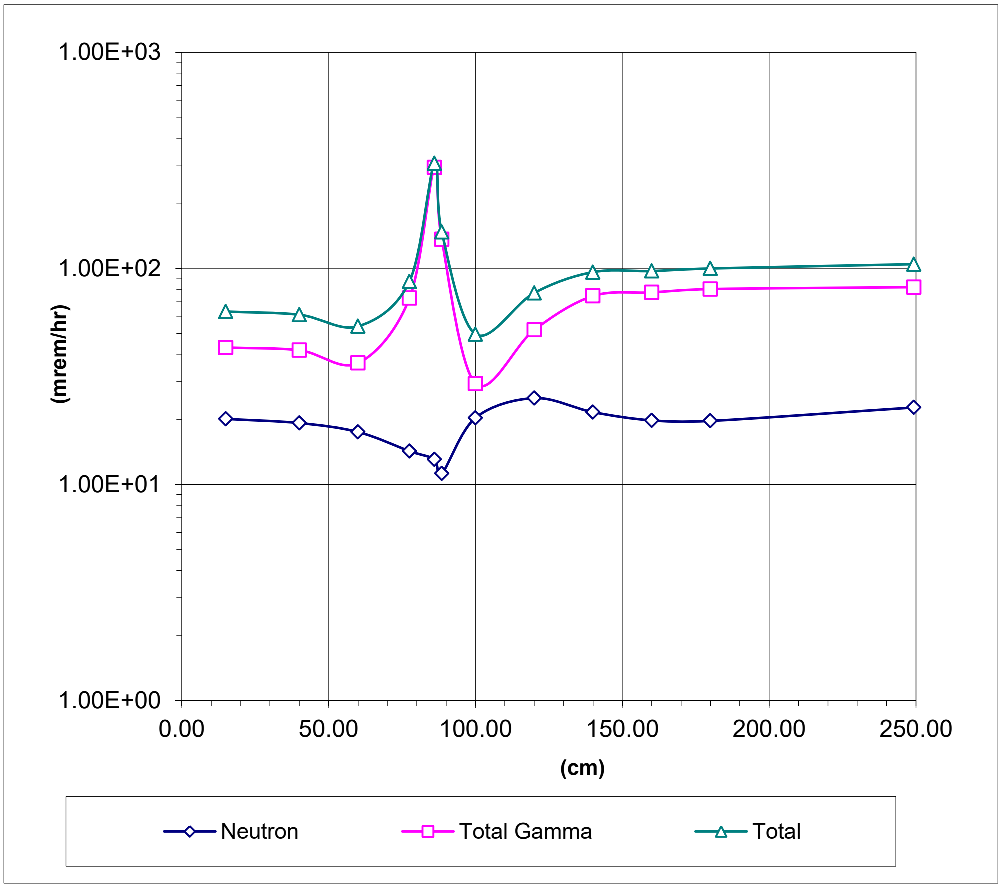


Figure T.5-24
OS197FC-B TC with 61BTH DSC, Top Surface Dose Rate, Normal Conditions

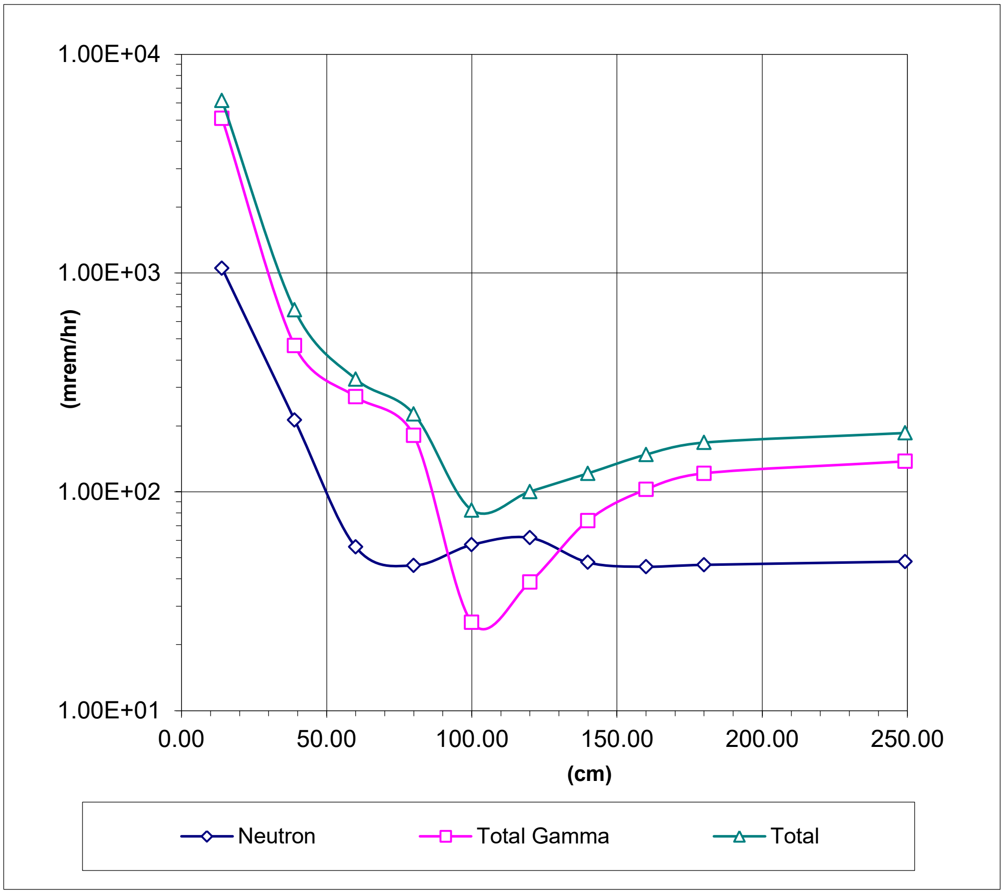


Figure T.5-25
OS197FC-B TC with 61BTH DSC, Bottom Surface Dose Rate, Normal Conditions

T.8.1.2 DSC Fuel Loading

1. Lift the TC/DSC and position it over the cask loading area of the spent fuel pool in accordance with the plant's 10CFR50 cask handling procedures.

2. Lower the cask into the fuel pool until the bottom of the cask is at the height of the fuel pool surface. As the cask is lowered into the pool, spray the exterior surface of the cask with demineralized water.

3. Place the cask in the designated location of the fuel pool.

4. Disengage the lifting yoke from the cask lifting trunnions and move the yoke. Spray the lifting yoke with clean water if it is raised out of the fuel pool.

5. The potential for fuel misloading is essentially eliminated through the implementation of procedural and administrative controls. The controls instituted to ensure that damaged and/or intact fuel assemblies are placed into a known cell location within a DSC, will typically consist of the following:

   - A TC/DSC loading plan is developed to verify that the failed, damaged, and/or intact fuel assemblies meet the burnup, enrichment and cooling time parameters of Technical Specification 2.1.
   - The loading plan is independently verified and approved before the fuel load.
   - A fuel movement schedule is then written, verified and approved based upon the loading plan. All fuel movements from any rack location are performed under strict compliance with the fuel movement schedule.

     CAUTION: HLZCs may have asymmetric decay heat loading zones. Verify TC/DSC orientation prior to initiating fuel movement.
   - If loading damaged fuel assemblies, verify that the required number of bottom end caps are installed in appropriate fuel compartment tube locations before fuel load.
   - If failed fuel is to be loaded in the DSC, place the empty failed fuel cans (refer to drawing NUH61BTH-72-1105) in the appropriate locations in the 61BTH DSC. (Note: If the failed fuel is to be loaded into the failed fuel can prior to loading into the DSC, skip this step.)

6. Prior to insertion of a spent fuel assembly into the DSC, the identity of the assembly is to be verified by two individuals using an underwater video camera or other means. Read and record the fuel assembly identification number from the fuel assembly and check this identification number against the DSC loading plan which indicates which fuel assemblies are acceptable for dry storage.

7. Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Steps 6 and 7 for each SFA loaded into the DSC. A maximum of 61 damaged fuel or 4 failed fuel assemblies may be loaded into the appropriate 2x2 compartments of the 61BTH DSC basket per Technical Specification 2.1. If loading failed fuel, ensure that the failed fuel can lids are installed. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly in the DSC. If

T.10 Radiation Protection

Section 7.4.1 discusses the anticipated cumulative dose exposure to site personnel during the fuel handling and transfer activities associated with utilizing one NUHOMS® HSM for storage of one DSC. Chapter 5 describes in detail the NUHOMS® operational procedures, several of which involve potential exposure to personnel. This section of the Appendix provides occupational exposure and off-site dose rates from a NUHOMS®-61BTH Type 1 or 2 DSC stored in each of the following HSM types: NUHOMS® HSM-H, Model 80, Model 102, Model 152, or Model 202.

T.10.1 Occupational Exposure

The expected occupational dose for placing a canister of spent fuel into dry storage is based on the operational steps outlined in Table 7.4-1 of the UFSAR. The total exposure for the occupational dose due to placing a single NUHOMS®-61BTH DSC into storage is conservatively estimated to be *3.33* person-rem as described in Table T.10-1. This value bounds the exposure for loading either a 61BTH-Type 1 or 61BTH-Type 2 DSC into storage. This is a very conservative estimate because the dose rates on and around the 61BTH DSCs used in these calculations are based on very conservative assumptions for the design-basis source terms and analyses models. The calculated exposures are due mainly to the expected gamma dose rate during preparation for welding.

The NUHOMS®-61BTH System loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table T.10-1. This information is used as the basis for estimating the total occupational exposure associated with one fuel load. This evaluation is performed for the storage of either the design-basis NUHOMS®-61BTH Type 1 or 2 DSC in an HSM. The loading operations are identical for the 61BTH-Type 1 and 61BTH-Type 2 DSC. The dose rates applicable for each operation are based on the results presented in Section T.5.4 of the UFSAR for loading operations. Engineering judgment and operational experience are used to estimate dose rates that were not explicitly evaluated. This evaluation assumes that a transfer trailer/skid with an integral ram is used for the DSC transfer operations. Licensees may elect to use different equipment and/or different procedures. Each licensee must evaluate any such changes in accordance with its ALARA program.

Unique steps are sometimes necessary at the individual site to load the canister, complete closure operations and place the canister in the HSM. Specifically, the licensee may choose to modify the sequence of operations in order to achieve reduced dose rates for a larger number of steps, with the end result of reduced total exposure. The only requirement is that the licensee practice ALARA with respect to the total exposure received for a loading campaign. These estimated durations, manloading and dose rates are not limits.

The amount of time required to complete some operations as identified in Table T.10-1 may be greater than the actual amount of time spent in a radiation field. The process of vacuum drying the DSC includes setting up the vacuum drying system (VDS), verifying that the VDS is operating correctly, evacuating the DSC cavity, monitoring the DSC pressure, and disconnecting the VDS from the DSC. Of these tasks, only setup and removal of the VDS require a worker to spend time near the DSC. The most time consuming task, evacuating the DSC, does not require anyone to be present near the DSC at all. The total exposure calculated for each task is therefore not necessarily equal to the number of workers multiplied by the total time required, multiplied by a dose rate. The exposure estimation for each task correctly accounts for cases such as vacuum drying and assumes that good ALARA practices are followed.

The results of the evaluations of the 61BTH are presented in Table T.10-1.

T.10.2 Off-Site Dose Calculations

Calculated dose rates in the immediate vicinity of the NUHOMS®-61BTH System are presented in Chapter T.5, which provides a detailed description of source term configuration, analysis models and bounding dose rates. The bounding dose rates are based upon contributions from the design basis fuel. Off-site dose rates and annual doses are presented in this section. This evaluation determines the neutron and gamma-ray off-site dose rates (including skyshine) in the vicinity of two generic ISFSI layouts containing design-basis fuel in the NUHOMS®-61BTH DSCs.

The first generic ISFSI evaluated is a 2x10 back-to-back array of HSM-Hs loaded with design-basis fuel in NUHOMS®-61BTH DSCs. The second generic layout evaluated is two 1x10 front-to-front arrays of HSM-Hs loaded with design-basis fuel in NUHOMS®-61BTH DSCs. This evaluation provides results for distances ranging from 6.1 to 600 meters from each face of the two arrays of HSMs. Similar calculations are performed for the NUHOMS®-61BTH DSCs within HSM Model 102s, although the source term used is different because the HSM Model 102 is limited to 24.0 kW decay heat (see Figure T.5-1 of Chapter T.5). Dose rates for the NUHOMS®-61BTH Type 1 DSCs within HSM Model 80s filled with design basis fuel assemblies are generated by scaling the HSM Model 102 results. The dose rates that are obtained with the HSM Model 80 are conservatively applicable to HSM Model 152 and 202.

The total annual exposure for each ISFSI layout as a function of distance from each face is given in Table T.10-2, Table T.10-3, and Table T.10-4 for the HSM-H, HSM Model 102, and the HSM Model 80, respectively. These data are also plotted in Figure T.10-1, Figure T.10-2, and Figure T.10-3 for the HSM-H, HSM Model 102, and HSM Model 80, respectively. The total annual exposure estimates assume 100% occupancy for 365 days.

The Monte Carlo computer code MCNP [10.1] calculates the dose rates at the specified locations around the arrays of HSM-Hs and HSM Model 102s. The results of these calculations provide an example of how to demonstrate compliance with the relevant radiological requirements of 10CFR20 [10.2], 10CFR72 [10.3], and 40CFR190 [10.4] for a specific site. Each site must perform site specific calculations to account for the actual layout of the HSMs and fuel source.

The assumptions used to generate the geometry of the ISFSIs for the MCNP analyses are summarized below. The following discussion applies to both the HSM-H and HSM Model 102 analyses.

- The 20 HSMs in the 2x10 back-to-back array are modeled as a box enveloping the 2x10 array of HSMs including the shield walls on the two sides of the array. MCNP starts the source particles on the surfaces of the box. The interiors of the HSMs and shield walls are modeled as air. Most particles that enter the interiors of the HSMs and shield walls will therefore pass through unhindered.

- The 20 HSMs in the two 1x10 front-to-front arrays are modeled as two boxes which envelop each 1x10 array of HSMs including the shield walls on the two sides and back of each array. The interiors of one array of HSMs and shield walls are modeled as air (the “source” array). Most particles that enter the interiors of these HSMs and shield walls will therefore pass

through unhindered. The other 1x10 array (the "shield" array) is modeled as concrete to simulate the shielding provided by the second array of HSMs for the direct radiation from the front of the opposing 1x10 array. The dose rates around the ISFSI are then generated using superposition.

- The ISFSI approach slab is modeled as concrete. Because the ground composition has, at best, only a secondary impact on the dose rates at the detectors, any differences between this assumed layout and the actual layout would not have a significant effect on the site dose rates.

- The "universe" is a sphere surrounding the ISFSI. To account for skyshine, the radius of this sphere (r=500,000 cm) is more than 10 mean free paths for neutrons and 50 mean free paths for gammas greater than that of the outermost surface, thus ensuring that the model is of a sufficient size to include all interactions, including skyshine, affecting the dose rate at the detectors.

The assumptions used for the MCNP analyses are summarized below.

- The HSM-H surface sources are bootstrapped (input to provide an equivalent boundary condition) using *the surface fluxes that correspond to the average dose rates provided in Table T.5-1. These surface fluxes are computed on the surface of a box that encloses the HSM-H, including the vent covers. These surface fluxes account for adjacent HSMs using reflective boundary conditions, as described in Section T.5.4.*

- The HSM Model 102 front and roof average dose rates are increased prior to input in the site dose calculation. *In the dose rate analysis documented in Section T.5.4, the HSM Model 102 is analyzed with an adjacent HSM on only one side. In the ISFSI configuration, there is typically an adjacent HSM on both sides. Therefore, the average front and roof dose rates as reported in Section T.5.4 are increased prior to input into the site dose analysis to estimate the effects of having an adjacent HSM on both sides.* The front and roof HSM Model 102 gamma dose rates used in the site dose analysis are 17.34 and 27.14 mrem/hr, respectively. The front and roof HSM Model 102 neutron dose rates used in the site dose analysis are 0.72 and 0.86 mrem/hr, respectively.

- MCNP starts the source particles on the ISFSI array surface with initial directions following a cosine distribution. Radiation fluxes outside thick shields such as the HSM walls and roof tend to have forward peaked angular distributions; therefore, a cosine function is a reasonable approximation for the starting direction distribution. Vents through shielding regions such as the HSM vents tend to collimate particles such that a semi-isotropic assumption would not be appropriate.

- Point detectors determine the dose rates on the four sides of the ISFSI as a function of distance from the ISFSI. All detectors represent the dose rate at three feet above ground level.

- Source information required by MCNP includes gamma-ray and neutron spectra for the HSM array surfaces, total gamma-ray and neutron activities for each HSM array face and total gamma-ray and neutron activities for the entire ISFSI. *The approach used for the HSM-H and HSM Model 102 is slightly different. For the HSM-H analysis, the neutron and gamma-ray spectra are explicitly computed on each face as part of the dose rate analysis documented in Chapter T.5 and applied on each surface. Gamma-ray flux calculations for the HSM-H are shown in Table T.10-5. Similar calculations for neutrons are shown in Table T.10-6.*

- *For the HSM Model 102 analysis, the* neutron and gamma-ray spectra are determined using MCNP tallies averaged over the HSM roof (including vents) using the design-basis active fuel neutron and gamma fuel sources. Use of the roof gives an average spectrum for the vented surfaces at which the dose rates are the highest. For gammas that penetrate the roof block, the thicker shield increases the dose rate importance of the higher energy gamma-rays from the fuel because the thicker shield filters out the lower energy particles. This roof spectrum is also used for neutrons for convenience, as gammas dominate neutrons in the site dose analysis. The HSM spectra are normalized to a one mrem/hour source using the flux-to-dose-factors from Reference [10.5]. These normalized spectra are then input into the MCNP ERG source variable.

- The probability of a particle being born on a given surface is proportional to the total activity of that surface. *For the HSM-H, the activity (or outward current) of each surface is determined by multiplying the total flux (particles/s-cm^2) by the area of the face (cm^2) and dividing by two. The factor of two is a consequence of the cosine directional distribution, as current = flux/2 for a cosine distribution.* This calculation is performed for the roof, sides, back and front of the HSM. The sum of the surface activities is then input as the tally multiplier.

- Gamma-ray spectrum calculations for the HSM Model 102 are shown in Table T.10-7. Both the group-wise dose rates and fluxes are taken directly from the MCNP runs. The "Input Current" column in Table T.10-7 is simply half the roof flux in each group, divided by the total dose rate and represents the roof current normalized to one mrem per hour. Similar calculations for neutrons are shown in Table T.10-8. *The surface activities are then computed as the average dose rate multiplied by the normalization constant from Table T.10-7 (702.4 γ/cm^2-s per mrem/hr gamma) and Table T.10-8 (69.7 n/cm^2-s per mrem/hr for neutron).*

The assumptions used to generate the HSM Model 80 dose rates are summarized below.

- For the HSM Model 80 analysis, the dose rates calculated for the HSM Model 102 ISFSI are simply scaled. The HSM Model 80 and Model 102 are geometrically identical, the only differences being the steel vent liners and thicker door for the HSM Model 102. The MCNP site dose calculation output provides the contribution of each surface source to the total dose rate. Therefore, the HSM Model 80 dose rates may be computed by multiplying each dose rate component by the ratio of the HSM Model 80 to HSM Model 102 surface source and summing the results. These ratios are provided in Section T.5.4.7.3.

T.10.2.1 Activity Calculations

The methodologies used to develop the surface activities are described in the previous section. Activity calculations are performed for the HSM-H and HSM Model 102, as the activity is required in the MCNP input. Similar activity calculations are not performed for the Model 80 because the site dose rates for the Model 80 are generated by scaling rather than by MCNP.

2x10 Back-to-Back Array

A box that envelops the HSM array and shield walls, as modeled in MCNP, approximates the 2x10 back-to-back array of HSMs. The dimensions of the box also include the width of the HSM end shield walls.

Two 1x10 Front-to-Front Arrays

A box that envelops the HSM array and shield walls, as modeled in MCNP, approximates the two 1x10 arrays of HSMs. The dimensions of the box also include the width of the HSM end and back shield walls.

The surface activities are summarized in Table T.10-9 and Table T.10-10 for the HSM-H and HSM Model 102, respectively.

T.10.2.2 Dose Rates

Dose rates are calculated for distances of 6.1 meters (20 feet) to 600 meters from the edges of the two ISFSI designs. The HSM is modeled in MCNP as a box representing the HSM arrays.

Neutron and gamma-ray sources are placed on each HSM surface (including shield walls) using the spectra and activities determined above. The angular distribution of source particles is modeled as a cosine distribution. *For the HSM-H, capture gamma-rays are computed for completeness, although their contribution to the total dose rate is negligible (< 1%). For the HSM Model 102, the* contribution of capture gamma-rays has been neglected, as has the contribution of bremsstrahlung electrons. The inclusion of coherent scattering greatly increases the variance in a problem with point detector tallies without improving the accuracy of the calculation. Thus, coherent scattering of photons is ignored.

The MCNP models of the ISFSI layouts are described herein. For the 2x10 back-to-back array of HSM-Hs with end shield walls, the "box" dimensions are as follows. The total width is 1260 cm. The length of the "box" is 3129 cm and the height of the "box" is *610* cm. For the HSM Model 102, these dimensions are 1209 cm, 3221 cm, and 457 cm.

For the two 1x10 front-to-front arrays of HSM-Hs with end and back shield walls, the "box" dimensions for each array are as follows. The total width is 721 cm. The length of the "box" is 3129 cm and the height of the "box" is *610* cm. The two 1x10 arrays are 1067 cm (35 feet) apart. For the HSM Model 102, these dimensions are 665 cm, 3221 cm, and 457 cm.

Point detectors are placed at the following locations as measured from each face of the "box": 6.095 m (20 feet), 10 m, 20 m, 30 m, 40 m, 50 m, 60 m, 70 m, 80 m, 90 m, 100 m, 200 m, 300 m, 400 m, 500 m, and 600 m. Each point detector is placed 91.4 cm (3 feet) above the ground.

Table T.10-1
Occupational Exposure Summary, 61BTH System

Location	Task Description	# of workers	Duration (hr)	Area Dose Rate (mrem/hr)	Total Exposure (person-mrem)
Auxiliary Building and Fuel Pool	Place the DSC into the Transfer Cask	3	1	*0*	*0*
	Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal	2	2	*0*	*0*
	Fill the DSC Cavity with Water	1	6	*0*	*0*
	Place the Cask Containing the DSC in the Fuel Pool	5	0.5	*0*	*0*
	Verify and Load the Candidate Fuel Assemblies into the DSC	3	5	*0*	*0*
	Place the Top Shield Plug on the DSC	2	1	*0*	*0*
	Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area	5	0.5	*0*	*0*
		1	0.033	*415*	*14*
		1	0.667	*268*	*179*
Cask Decon-tamination Area	Decontaminate the Outer Surface of the Cask	1	1.75	*268*	*469*
		1	1	*0*	*0*
	Decontaminate the Top Region of the Cask and DSC	1	0.5	*418*	*209*
		1	0.5	*60*	*30*
	Drain Water from the DSC	1	0.083	*415*	*35*
	Remove Cask/DSC Annulus Seal and Set-Up Welding Machine	1	0.167	*834*	*139*
		1	0.75	*219*	*164*
		1	0.5	*157*	*78*
	Weld the Inner Top Cover to the DSC Shell and Perform NDE (PT)	2	6	*0*	*0*
		1	0.33	*420*	*140*
	Drain the DSC Cavity	1	0.25	*219*	*55*
		1	0.017	*420*	*7*
		1	0.5	*0*	*0*
	Vacuum Dry and Backfill the DSC with Helium	1	0.5	*157*	*78*
		2	30	*0*	*0*
	Helium Leak Test the Shield Plug Weld	2	1	*0*	*0*
	Seal Weld the Prefabricated Plugs to the Vent and Siphon Ports and Perform NDE (PT)	1	0.5	*219*	*110*
	Fit-Up the DSC Outer Top Cover Plate	1	0.25	*420*	*105*
		1	0.5	*219*	*110*
	Weld the Outer Top Cover Plate to DSC Shell and Perform NDE (PT)	1	1	*157*	*157*
		1	0.167	*420*	*70*
		2	14	*0*	*0*
		1	0.333	*420*	*140*
	Install the Cask Lid	2	0.667	*183*	*244*
Reactor/ Fuel Building Bay	Ready the Cask Support Skid and Transfer Trailer for Service	2	2	*0*	*0*
	Place the Cask onto the Skid and Trailer	2	0.25	*320*	*160*
	Secure the Cask to the Skid	1	0.25	*320*	*80*
ISFSI Site	Ready The Cask Support Skid and Transfer Trailer for Service	2	2	*negligible*	*0*
	Transfer the Cask to ISFSI	6	1	*negligible*	*0*
	Position the Cask in Close Proximity with the HSM	3	1	*negligible*	*0*
	Remove the Cask Lid	2	0.67	*90*	*121*
	Align and Dock the Cask with the HSM	2	0.25	*238*	*119*
	Position and Align Ram with Cask	2	0.5	*180*	*180*
	Remove Ram Access Cover Plate	1	0.083	*806*	*67*
	Transfer the DSC from the Cask to the HSM	3	0.5	*negligible*	*0*
	Lift the Ram Back onto the Trailer and Un-Dock the Cask from the	2	0.083	*117*	*19*
	Install HSM Access Door	2	0.5	*55*	*55*
	Totals	**N/A**	***87***	***N/A***	***3331***

Total estimated dose is *3.33* person-rem per 61BTH canister load.
This dose bounds the expected dose for the 61BTH Type 1 DSC and Type 2 DSC canister loads in the HSM-H, HSM Model 102 and HSM Model 80.

Table T.10-2
Total Annual Exposure, 61BTH within HSM-H

2x10 Back to Back Array

Distance (meters)	Front Total Dose (mrem)	1σ Uncertainty (mrem)	1σ Relative Uncertainty
6.1	*156,477*	*31*	*0.02%*
10	*98,575*	*29*	*0.03%*
20	*39,967*	*28*	*0.1%*
30	*21,036*	*14*	*0.1%*
40	*12,822*	*16*	*0.1%*
50	*8,532*	*7*	*0.1%*
60	*6,042*	*6*	*0.1%*
70	*4,454*	*5*	*0.1%*
80	*3,383*	*5*	*0.1%*
90	*2,623*	*4*	*0.2%*
100	*2,063*	*3*	*0.2%*
200	*310*	*1*	*0.4%*
300	*67*	*0.4*	*0.6%*
400	*17*	*0.1*	*0.8%*
500	*5*	*0.1*	*1.0%*
600	*2*	*0.03*	*1.5%*

Distance (meters)	Side Total Dose (mrem)	1σ Uncertainty (mrem)	1σ Relative Uncertainty
6.1	*23,515*	*14*	*0.1%*
10	*17,336*	*12*	*0.1%*
20	*10,120*	*8*	*0.1%*
30	*6,867*	*8*	*0.1%*
40	*4,961*	*6*	*0.1%*
50	*3,755*	*15*	*0.4%*
60	*2,888*	*5*	*0.2%*
70	*2,281*	*4*	*0.2%*
80	*1,826*	*5*	*0.3%*
90	*1,471*	*3*	*0.2%*
100	*1,198*	*3*	*0.2%*
200	*212*	*1*	*0.6%*
300	*48*	*0.4*	*0.8%*
400	*13*	*0.1*	*0.9%*
500	*4*	*0.1*	*2.3%*
600	*1*	*0.02*	*1.8%*

Two 1x10 Front to Front Arrays

Distance (meters)	Back Total Dose (mrem)	1σ Uncertainty (mrem)	1σ Relative Uncertainty
6.1	*25,007*	*13*	*0.1%*
10	*19,692*	*19*	*0.1%*
20	*12,011*	*10*	*0.1%*
30	*8,165*	*7*	*0.1%*
40	*5,905*	*6*	*0.1%*
50	*4,439*	*6*	*0.1%*
60	*3,420*	*5*	*0.1%*
70	*2,695*	*4*	*0.2%*
80	*2,152*	*4*	*0.2%*
90	*1,742*	*4*	*0.2%*
100	*1,420*	*3*	*0.2%*
200	*248*	*1*	*0.4%*
300	*57*	*0.3*	*0.6%*
400	*16*	*0.9*	*5.8%*
500	*5*	*0.1*	*1.1%*
600	*1*	*0.02*	*1.3%*

Distance (meters)	Side Total Dose (mrem)	1σ Uncertainty (mrem)	1σ Relative Uncertainty
6.1	*62,710*	*31*	*0.05%*
10	*36,284*	*21*	*0.1%*
20	*15,258*	*13*	*0.1%*
30	*9,026*	*11*	*0.1%*
40	*6,128*	*11*	*0.2%*
50	*4,470*	*11*	*0.3%*
60	*3,360*	*6*	*0.2%*
70	*2,606*	*5*	*0.2%*
80	*2,081*	*16*	*0.8%*
90	*1,666*	*4*	*0.3%*
100	*1,355*	*7*	*0.5%*
200	*231*	*1*	*0.4%*
300	*54*	*1*	*1.0%*
400	*14*	*0.2*	*1.2%*
500	*4*	*0.1*	*2.5%*
600	*1*	*0.03*	*2.0%*

Table T.10-5
HSM-H Gamma-Ray *Flux* Calculation Results

E (MeV)	***Front Flux (g/cm^2-s)***	***Roof Flux (g/cm^2-s)***	***Rear Flux (g/cm^2-s)***	***Side Flux (g/cm^2-s)***
1.00E-02	*0.00E+00*	*0.00E+00*	*0.00E+00*	*0.00E+00*
3.00E-02	*2.35E+00*	*9.72E+00*	*8.24E-02*	*1.50E-01*
5.00E-02	*1.81E+03*	*4.41E+03*	*9.53E+01*	*6.49E+01*
7.00E-02	*1.60E+04*	*3.30E+04*	*9.18E+02*	*4.61E+02*
1.00E-01	*3.40E+04*	*6.41E+04*	*1.80E+03*	*9.05E+02*
1.50E-01	*3.85E+04*	*5.97E+04*	*1.79E+03*	*8.59E+02*
2.00E-01	*1.95E+04*	*2.70E+04*	*1.11E+03*	*4.08E+02*
2.50E-01	*1.01E+04*	*1.63E+04*	*4.95E+02*	*2.47E+02*
3.00E-01	*5.17E+03*	*1.09E+04*	*4.34E+02*	*1.60E+02*
3.50E-01	*2.98E+03*	*7.51E+03*	*2.68E+02*	*1.10E+02*
4.00E-01	*1.65E+03*	*5.20E+03*	*1.44E+02*	*8.41E+01*
4.50E-01	*8.95E+02*	*3.63E+03*	*1.07E+02*	*6.98E+01*
5.00E-01	*4.87E+02*	*2.62E+03*	*7.53E+01*	*6.01E+01*
5.50E-01	*3.49E+02*	*2.00E+03*	*4.51E+01*	*5.46E+01*
6.00E-01	*2.31E+02*	*1.52E+03*	*2.77E+01*	*4.69E+01*
6.50E-01	*1.59E+02*	*1.21E+03*	*3.01E+01*	*4.29E+01*
7.00E-01	*1.40E+02*	*9.86E+02*	*2.48E+01*	*3.86E+01*
8.00E-01	*1.58E+02*	*1.53E+03*	*1.36E+01*	*6.85E+01*
1.00E+00	*2.01E+02*	*2.08E+03*	*1.69E+01*	*1.09E+02*
1.40E+00	*1.86E+02*	*2.44E+03*	*1.17E+01*	*1.54E+02*
1.80E+00	*9.05E+01*	*1.40E+03*	*3.31E+00*	*1.02E+02*
2.20E+00	*5.53E+01*	*9.63E+02*	*2.05E+00*	*7.31E+01*
2.60E+00	*2.96E+01*	*5.81E+02*	*1.06E+00*	*4.17E+01*
2.80E+00	*1.33E+00*	*1.88E+01*	*5.36E-02*	*1.75E+00*
3.25E+00	*1.47E+00*	*1.98E+01*	*5.81E-02*	*1.93E+00*
3.75E+00	*3.26E-01*	*6.92E+00*	*2.70E-02*	*7.40E-01*
4.25E+00	*1.31E-01*	*3.22E+00*	*9.01E-03*	*2.54E-01*
4.75E+00	*0.00E+00*	*0.00E+00*	*0.00E+00*	*7.04E-04*
5.00E+00	*0.00E+00*	*0.00E+00*	*0.00E+00*	*1.00E-03*
Total	*1.33E+05*	*2.49E+05*	*7.42E+03*	*4.16E+03*

Table T.10-6
HSM-H Neutron *Flux* Calculation Results

E (MeV)	***Front Flux (n/cm²-s)***	***Roof Flux (n/cm²-s)***	***Rear Flux (n/cm²-s)***	***Side Flux (n/cm²-s)***
2.50E-08	*8.21E+00*	*3.63E+01*	*5.65E-01*	*1.05E+00*
1.00E-07	*2.20E+01*	*1.02E+02*	*1.58E+00*	*3.01E+00*
1.00E-06	*6.42E+00*	*3.24E+01*	*4.30E-01*	*7.24E-01*
1.00E-05	*2.83E+00*	*1.41E+01*	*1.67E-01*	*2.07E-01*
1.00E-04	*2.63E+00*	*1.22E+01*	*1.52E-01*	*1.79E-01*
1.00E-03	*2.39E+00*	*1.00E+01*	*1.34E-01*	*1.49E-01*
1.00E-02	*2.11E+00*	*7.74E+00*	*1.07E-01*	*1.18E-01*
1.00E-01	*2.47E+00*	*6.31E+00*	*8.91E-02*	*1.02E-01*
5.00E-01	*1.91E+00*	*3.73E+00*	*6.09E-02*	*6.94E-02*
1.00E+00	*5.54E-01*	*1.21E+00*	*2.05E-02*	*3.31E-02*
2.50E+00	*3.14E-01*	*1.19E+00*	*2.07E-02*	*6.48E-02*
5.00E+00	*4.49E-02*	*2.30E-01*	*3.15E-03*	*1.07E-02*
7.00E+00	*5.66E-03*	*3.36E-02*	*3.23E-04*	*2.79E-03*
1.00E+01	*5.82E-04*	*4.28E-03*	*4.46E-05*	*2.80E-04*
1.40E+01	*6.11E-05*	*4.04E-04*	*2.52E-06*	*2.03E-05*
2.00E+01	*3.43E-06*	*1.68E-05*	*1.23E-07*	*1.35E-06*
Total	*5.18E+01*	*2.27E+02*	*3.33E+00*	*5.73E+00*

Table T.10-9
Summary of ISFSI Surface Activities, 61BTH DSC within HSM-H

2x10 Back-to-Back Array

Source	Area (cm^2)	Neutron Activity (neutrons/sec)	Gamma-Ray Activity (γ/sec)
Roof	*3,942,392.1*	*4.477E+08*	*4.914E+11*
Front 1	*1,907,609.1*	*4.945E+07*	*1.266E+11*
Front 2	*1,907,609.1*	*4.945E+07*	*1.266E+11*
Side 1	*767,998.5*	*2.199E+06*	*1.610E+09*
Side 2	*767,998.5*	*2.199E+06*	*1.610E+09*
Total	***9,293,607.2***	***5.510E+08***	***7.478E+11***

Two 1x10 Front-to-Front Arrays

Source	Area (cm^2)	Neutron Activity (neutrons/sec)	Gamma-Ray Activity (γ/sec)
Roof	*2,257,337.4*	*2.563E+08*	*2.814E+11*
Front	*1,907,609.1*	*4.945E+07*	*1.266E+11*
Back	*1,907,609.1*	*3.175E+06*	*7.083E+09*
Side 1	*439,741.1*	*1.259E+06*	*9.219E+08*
Side 2	*439,741.1*	*1.259E+06*	*9.219E+08*
Total	***6,952,037.7***	***3.115E+08***	***4.169E+11***

Table T.10-11
MCNP Front Detector Dose Rates for 2x10 Array, 61BTH DSC within HSM-H

Distance (meters)	Gamma Dose Rate (mrem/hr)	Gamma MCNP 1σ Uncertainty	Neutron Dose Rate (mrem/hr)	Neutron MCNP 1σ Uncertainty	Total Dose Rate (mrem/hr)	Combined MCNP 1σ Uncertainty
6.1	*1.76E+01*	*0.02%*	*2.36E-01*	*0.1%*	*1.79E+01*	*0.02%*
10	*1.11E+01*	*0.03%*	*1.57E-01*	*0.1%*	*1.13E+01*	*0.03%*
20	*4.49E+00*	*0.1%*	*7.20E-02*	*0.2%*	*4.56E+00*	*0.1%*
30	*2.36E+00*	*0.1%*	*4.05E-02*	*0.2%*	*2.40E+00*	*0.1%*
40	*1.44E+00*	*0.1%*	*2.54E-02*	*0.2%*	*1.46E+00*	*0.1%*
50	*9.57E-01*	*0.1%*	*1.70E-02*	*0.3%*	*9.74E-01*	*0.1%*
60	*6.78E-01*	*0.1%*	*1.18E-02*	*0.3%*	*6.90E-01*	*0.1%*
70	*5.00E-01*	*0.1%*	*8.56E-03*	*0.4%*	*5.08E-01*	*0.1%*
80	*3.80E-01*	*0.1%*	*6.38E-03*	*0.7%*	*3.86E-01*	*0.1%*
90	*2.95E-01*	*0.2%*	*4.78E-03*	*0.5%*	*2.99E-01*	*0.2%*
100	*2.32E-01*	*0.2%*	*3.66E-03*	*0.5%*	*2.35E-01*	*0.2%*
200	*3.49E-02*	*0.4%*	*5.02E-04*	*1.4%*	*3.54E-02*	*0.4%*
300	*7.54E-03*	*0.6%*	*1.25E-04*	*3.6%*	*7.67E-03*	*0.6%*
400	*1.94E-03*	*0.8%*	*3.94E-05*	*3.6%*	*1.98E-03*	*0.8%*
500	*5.59E-04*	*1.0%*	*1.50E-05*	*4.3%*	*5.74E-04*	*1.0%*
600	*1.87E-04*	*1.5%*	*6.48E-06*	*4.9%*	*1.93E-04*	*1.5%*

Table T.10-12
MCNP Back Detector Dose Rates for the Two 1x10 Arrays, 61BTH DSC within HSM-H

Distance (meters)	Gamma Dose Rate (mrem/hr)	Gamma MCNP 1σ Uncertainty	Neutron Dose Rate (mrem/hr)	Neutron MCNP 1σ Uncertainty	Total Dose Rate (mrem/hr)	Combined MCNP 1σ Uncertainty
6.1	*2.77E+00*	*0.1%*	*8.36E-02*	*0.1%*	*2.85E+00*	*0.1%*
10	*2.18E+00*	*0.1%*	*6.79E-02*	*0.1%*	*2.25E+00*	*0.1%*
20	*1.33E+00*	*0.1%*	*4.14E-02*	*0.2%*	*1.37E+00*	*0.1%*
30	*9.05E-01*	*0.1%*	*2.68E-02*	*0.2%*	*9.32E-01*	*0.1%*
40	*6.56E-01*	*0.1%*	*1.83E-02*	*0.2%*	*6.74E-01*	*0.1%*
50	*4.94E-01*	*0.1%*	*1.28E-02*	*0.3%*	*5.07E-01*	*0.1%*
60	*3.81E-01*	*0.1%*	*9.32E-03*	*0.4%*	*3.90E-01*	*0.1%*
70	*3.01E-01*	*0.2%*	*6.86E-03*	*0.3%*	*3.08E-01*	*0.2%*
80	*2.40E-01*	*0.2%*	*5.22E-03*	*0.6%*	*2.46E-01*	*0.2%*
90	*1.95E-01*	*0.2%*	*4.01E-03*	*0.5%*	*1.99E-01*	*0.2%*
100	*1.59E-01*	*0.2%*	*3.17E-03*	*0.8%*	*1.62E-01*	*0.2%*
200	*2.79E-02*	*0.4%*	*4.54E-04*	*1.3%*	*2.83E-02*	*0.4%*
300	*6.33E-03*	*0.6%*	*1.22E-04*	*2.8%*	*6.45E-03*	*0.6%*
400	*1.80E-03*	*6.0%*	*3.98E-05*	*3.4%*	*1.84E-03*	*5.8%*
500	*5.03E-04*	*1.1%*	*1.42E-05*	*3.8%*	*5.17E-04*	*1.1%*
600	*1.62E-04*	*1.3%*	*7.07E-06*	*10.4%*	*1.70E-04*	*1.3%*

Table T.10-13
MCNP Side Detector Dose Rates, 61BTH DSC within HSM-H

2x10 Back-to-Back Array

Distance (meters)	Gamma Dose Rate (mrem/hr)	Gamma MCNP 1σ Uncertainty	Neutron Dose Rate (mrem/hr)	Neutron MCNP 1σ Uncertainty	Total Dose Rate (mrem/hr)	Combined MCNP 1σ Uncertainty
6.1	*2.60E+00*	*0.1%*	*8.36E-02*	*0.1%*	*2.68E+00*	*0.1%*
10	*1.92E+00*	*0.1%*	*6.29E-02*	*0.1%*	*1.98E+00*	*0.1%*
20	*1.12E+00*	*0.1%*	*3.56E-02*	*0.2%*	*1.16E+00*	*0.1%*
30	*7.61E-01*	*0.1%*	*2.27E-02*	*0.3%*	*7.84E-01*	*0.1%*
40	*5.51E-01*	*0.1%*	*1.53E-02*	*0.3%*	*5.66E-01*	*0.1%*
50	*4.18E-01*	*0.4%*	*1.07E-02*	*0.3%*	*4.29E-01*	*0.4%*
60	*3.22E-01*	*0.2%*	*7.80E-03*	*0.4%*	*3.30E-01*	*0.2%*
70	*2.55E-01*	*0.2%*	*5.79E-03*	*0.4%*	*2.60E-01*	*0.2%*
80	*2.04E-01*	*0.3%*	*4.35E-03*	*0.5%*	*2.08E-01*	*0.3%*
90	*1.65E-01*	*0.2%*	*3.44E-03*	*0.7%*	*1.68E-01*	*0.2%*
100	*1.34E-01*	*0.2%*	*2.67E-03*	*0.8%*	*1.37E-01*	*0.2%*
200	*2.38E-02*	*0.6%*	*3.85E-04*	*1.4%*	*2.42E-02*	*0.6%*
300	*5.42E-03*	*0.8%*	*1.00E-04*	*2.5%*	*5.52E-03*	*0.8%*
400	*1.43E-03*	*0.9%*	*3.50E-05*	*3.9%*	*1.47E-03*	*0.9%*
500	*4.35E-04*	*2.4%*	*1.27E-05*	*4.8%*	*4.48E-04*	*2.3%*
600	*1.39E-04*	*1.8%*	*4.95E-06*	*4.9%*	*1.44E-04*	*1.8%*

Two 1x10 Front-to-Front Arrays

Distance (meters)	Gamma Dose Rate (mrem/hr)	Gamma MCNP 1σ Uncertainty	Neutron Dose Rate (mrem/hr)	Neutron MCNP 1σ Uncertainty	Total Dose Rate (mrem/hr)	Combined MCNP 1σ Uncertainty
6.1	*7.04E+00*	*0.05%*	*1.23E-01*	*0.2%*	*7.16E+00*	*0.05%*
10	*4.06E+00*	*0.1%*	*8.31E-02*	*0.2%*	*4.14E+00*	*0.1%*
20	*1.70E+00*	*0.1%*	*4.24E-02*	*0.2%*	*1.74E+00*	*0.1%*
30	*1.00E+00*	*0.1%*	*2.62E-02*	*0.3%*	*1.03E+00*	*0.1%*
40	*6.82E-01*	*0.2%*	*1.74E-02*	*0.4%*	*7.00E-01*	*0.2%*
50	*4.98E-01*	*0.3%*	*1.22E-02*	*0.4%*	*5.10E-01*	*0.3%*
60	*3.75E-01*	*0.2%*	*8.78E-03*	*0.5%*	*3.84E-01*	*0.2%*
70	*2.91E-01*	*0.2%*	*6.48E-03*	*0.7%*	*2.97E-01*	*0.2%*
80	*2.33E-01*	*0.8%*	*4.87E-03*	*0.6%*	*2.38E-01*	*0.8%*
90	*1.86E-01*	*0.3%*	*3.78E-03*	*0.7%*	*1.90E-01*	*0.3%*
100	*1.52E-01*	*0.5%*	*2.98E-03*	*0.9%*	*1.55E-01*	*0.5%*
200	*2.59E-02*	*0.4%*	*4.29E-04*	*1.6%*	*2.64E-02*	*0.4%*
300	*6.06E-03*	*1.0%*	*1.08E-04*	*2.5%*	*6.17E-03*	*1.0%*
400	*1.58E-03*	*1.2%*	*4.30E-05*	*9.9%*	*1.63E-03*	*1.2%*
500	*4.92E-04*	*2.6%*	*1.38E-05*	*4.9%*	*5.06E-04*	*2.5%*
600	*1.55E-04*	*2.1%*	*5.91E-06*	*7.7%*	*1.61E-04*	*2.0%*

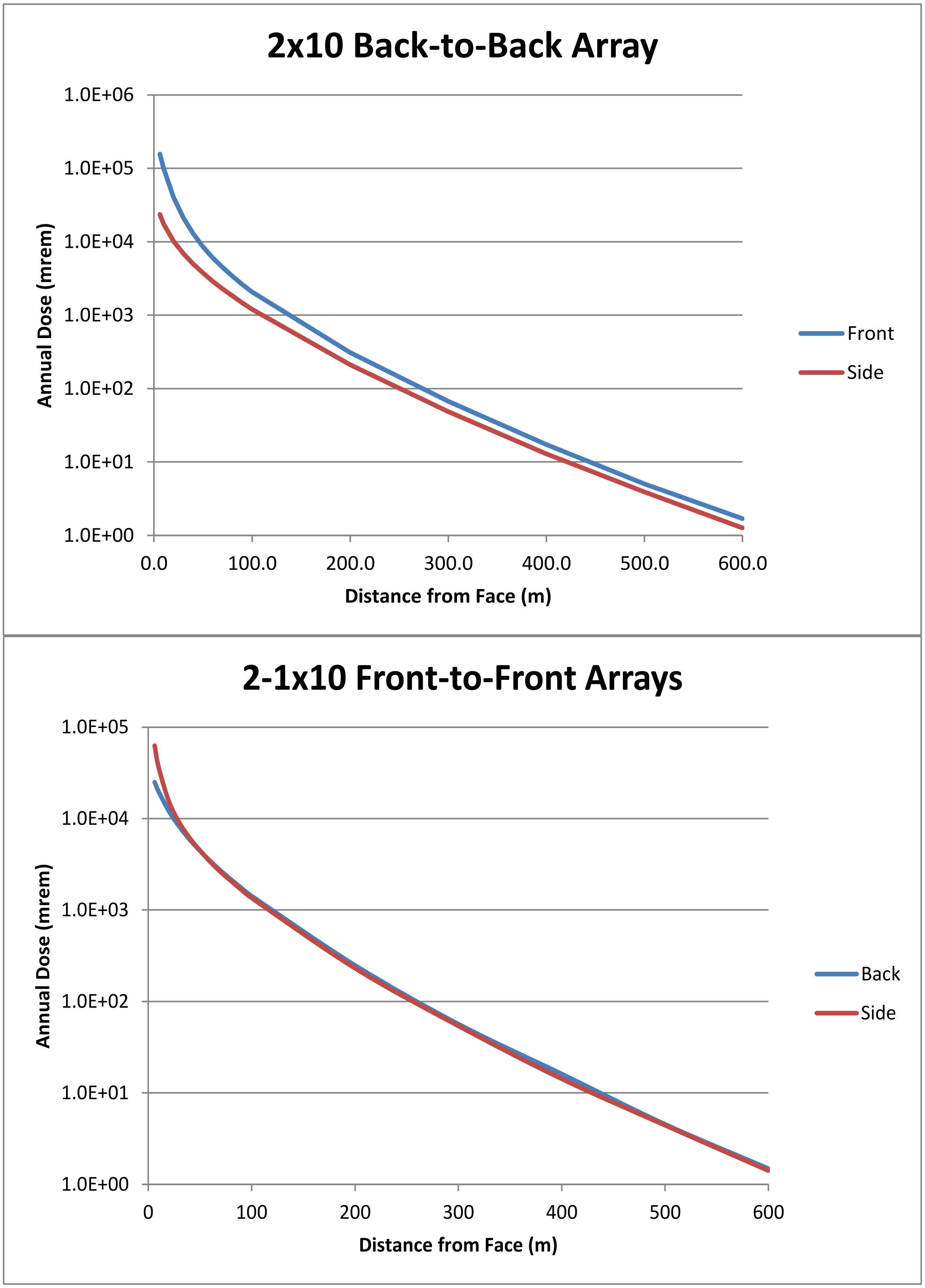


Figure T.10-1
Annual Exposure from the ISFSI as a Function of Distance, 61BTH DSC within HSM-H

T.11.2 Postulated Accidents

T.11.2.1 Reduced HSM Air Inlet and Outlet Shielding

This event is described in UFSAR Section 8.2 for Models 80/102 or in their respective appendices for Models 152/202.

T.11.2.1.1 Cause of Accident

No change to the cause of the accident for Standardized HSM as described in UFSAR Section 8.2.1.1 for Models 80/102 or in their respective appendices for Models 152/202.

For the HSM-H, HSM-HS and HSM Models 152 and 202, this accident is not credible since the an array composed of these HSM types is designed with the elimination of 6-inch gaps between the adjacent HSMs. These HSM types are placed next to each other and even in the unlikely event of large settlement of the ISFSI foundation, shifting of adjacent HSMs occurring and causing these HSMs to separate is not credible.

T.11.2.1.2 Accident Analysis

There are no structural consequences that affect the safe operation of the NUHOMS®-61BTH system resulting from the separation of the Standardized HSM. The thermal effects of this accident result from the blockage of Standardized HSM air inlet and outlet openings. However, the effect on the NUHOMS®-61BTH Type 1 DSC, Standardized HSM and fuel temperatures is bounded by the complete blockage of air inlet and outlet openings described in Section T.11.2.7. The radiological consequences of this accident are described in the paragraph below.

T.11.2.1.3 Accident Dose Calculations

The off-site radiological effects that result from a partial loss of adjacent Standardized HSM shielding is an increase in the air scattered (skyshine) and direct doses from the 12 inch gap between the separated HSMs. The air scattered (skyshine) and direct doses are reduced from the gap between the HSMs that are in contact with each other. On-site radiological effects result from an increase in the direct radiation during recovery operations and increased skyshine radiation. Table 8.2-2 shows the comparisons of the increased dose rate as a function of distance due to the reduced shielding effects of the adjacent HSM for the 24P DSC with 5-year cooled design basis fuel. Table T.11–1 provides a similar table for the NUHOMS®-61BTH *Type 1 DSC in the Standardized HSM*. For the NUHOMS®-61BTH system the dose increase to a person located 100 meters away from the NUHOMS® installation for eight hours a day for five days (estimated recovery time) would be 8.8 mrem. The increased dose to an off-site person for 24 hours a day for five days located 600 m away would be about 0.05 mrem. Thus, the 10CFR72.106 requirements for this postulated event are met.

T.11.2.1.4 Corrective Actions

No change. See Section UFSAR 8.2.1.4 for Models 80/102 or in their respective appendices for Models 152/202.

There is no change to the determination of the tornado wind and tornado missile loads acting on the Standardized HSM or HSM-H as detailed in Section T.2.2.1.

There is no change to the determination of the tornado wind and tornado missile loads acting on the HSM-H/HSM-HS as detailed in Appendix P, Section P.2.2.1.

T.11.2.3.2 Accident Analysis

An evaluation of the HSM and transfer cask with respect to tornado winds and tornado missiles is presented in Section 8.2.2. Changes to this analysis, as a result of the addition of the NUHOMS®-61BTH DSC, are presented in Section T.3.7.1. Evaluation of the Standardized HSM and TC with respect to tornado missile is also presented in Section 8.2.2. The tornado missile impact evaluation of the HSM-H is presented in the following sections.

The evaluation of the HSM-H for the effect of DBT wind pressure loads is addressed in Section P.3.7.1.1.

The missile impact analysis presented in Section P.11.2.3.2.1 is applicable here. Therefore, a loaded HSM-H rotates a maximum of 0.60° from vertical. The loaded HSM-H is stable against overturning as tip-over does not occur until the CG rotates past the edge point (point B, Figure T.11-1) to an angle of more than 24.65° [= $\tan^{-1}(52.0/118.77)$].

The tornado missile impact evaluation of the HSM-H/HSM-HS presented in Section U.11.2.3.2.1 is not changed.

T.11.2.3.3 Accident Dose Calculations

The increase in the dose rates at the localized impact location following the missile impact accident is expected to be bounded by the dose rates at the HSM-H vents, calculated to be *2081* mrem/hour in Table T.5-1, since the structural analysis results demonstrate that there is no full penetration. This represents an increase in the roof centerline dose rate by a factor *of 2081/123 ≈ 17* and is conservative.

For the purpose of this calculation, it is conservatively assumed that the affected area is twice the area of impact ~ 1.6 ft^2. The approximate surface areas at the HSM-H front is 140 ft^2, at the HSM-H roof is 200 ft^2 and that at the HSM-H side is 280 ft^2. The impact area, therefore, represents approximately 0.6% to 1.2% of the surface area of the HSM-H, and the average dose rate on the surface of the impacted HSM will not increase appreciably. This increase does not significantly affect the ISFSI site dose rates and the results from Section T.10.2 for a 2x10 array of undamaged HSMs (specifically Table T.10-*11*) can be utilized to determine the exposure from a damaged HSM. This method is conservative because the missile impact will affect at most a single HSM, while a 2x10 array has approximately 20 front and 20 roof vents. The total dose rate is then the dose rate of the damaged HSM summed with the dose rate of the undamaged HSMs in the array, or twice the dose rate of the undamaged array using the conservative assumptions outlined above.

The dose received by a person located 100 meters away from the ISFSI for the assumed 8-hour duration would be less than 5 mrem (2*8 hours**HSM-H* dose rate at 100m, *0.235* mrem/hour) with a 2x10 array of HSMs. As an additional conservatism, *bounding HSM-H* dose rates are used. The dose to an offsite person located 500 meters away for the assumed 8-hour duration would be less than 0.01 mrem (2*8 hours**HSM-H* dose rate at 500m, *5.74*E-04 mrem/hour) with a 2x10 array of HSMs.

T.11.2.5.2 Accident Analysis

The evaluation of the NUHOMS®-61BTH DSC shell and basket assemblies due to an accidental drop is presented in Section T.3.7.4. As documented in Chapter T.3.7, the TCs have been evaluated for a payload that bounds the 24PTH DSC payload, and therefore is not affected by the 61BTH DSC. As shown in Section T.3.7.4, the DSC shell and basket stress intensities are within the appropriate ASME Code Service Level D allowable limits and maintains their structural integrity.

For the case of an OS197/OS197H transfer cask liquid neutron shield, a complete loss of neutron shield was evaluated at the 100ºF ambient condition with full solar load in Chapter T.4. It is conservatively assumed that the neutron shield jacket is still present but all the liquid is lost. The maximum DSC shell temperature is 544ºF. The maximum OS197/OS197H cask inner liner, OS197/OS197H cask outer shell, and OS197/OS197H cask neutron shield jacket temperatures are 428ºF, 392ºF and 267ºF, respectively, for 61BTH Type 2 DSC with 31.2 kW decay heat load as shown in Table T.4-10. The fuel cladding temperatures are below their limit as shown in Table T.4-25. Accident thermal conditions, such as loss of the liquid neutron shield, need not be considered in the load combination evaluation. Rather the peak stresses resulting from the accident thermal conditions must be less than the allowable fatigue stress limit for 10 cycles from the appropriate fatigue design curves in Appendix I of the ASME Code. Similar analyses of other NUHOMS® TCs have shown that fatigue is not a concern. Therefore, these thermal stresses in a TC with a liquid neutron shield need not be evaluated for the accident condition.

As documented in Section U.3.3.7.4, the OS200/OS200FC transfer cask has been evaluated for the 32PTH1 DSC payload, which bounds that for the 61BTH DSC.

For the OS200/OS200FC transfer cask, the assessment of a complete loss of neutron shield, presented in Section U.11.2.5.2, is not changed.

T.11.2.5.3 Accident Dose Calculations for Loss of Neutron Shield

The postulated accident condition for the onsite OS197 TC assumes that after a drop event, the water in the neutron shield is lost. The loss of neutron shield is modeled using the normal operation models described in Section T.5.4 by replacing the neutron shield with air. As discussed in Chapter T.5, the evaluation with the OS197 TC is bounding for the OS200 TC. *Potentially damaged fuel is modeled as intact in the far-field accident models because it is determined in Chapter T.5 that far-field dose rates are larger when fuel is modeled as intact.* The accident condition dose rates from Chapter T.5, are summarized in Table *T.5-4* for the bounding 61BTH DSC Type 1 loaded with design basis fuel.

Unanalyzed fuel (UF) has enrichments below what is typically observed for BWR fuel, see the discussion in Section T.5.2.6. Because UF predominantly affects the neutron source, the computed exposures are conservatively doubled to account for the potential presence of UF. The dose received by a person located 100 meters away from the NUHOMS® 61BTH system installation for an assumed 8 hour duration would be less than *30 mrem (1.75 mrem/hr * 8 hours * 2)* mrem with the OS197FC-B. The increased dose to an offsite person located 500 meters away for the assumed 8 hour duration would be less than *0.2 mrem (7.64E-3 mrem/hr * 8 hours * 2)* with both the OS197FC-B TC with NUHOMS® 61BTH DSC. These exposures are well within the limits of 10CFR72.106 for an accident condition.

T.11.2.5.4 Corrective Action

No change to Section 8.2.5.4.

T.11.2.6 Lightning

No change. The evaluation presented in Section 8.2.6 is not affected by the addition of the NUHOMS®-61BTH DSC to the NUHOMS® system.

T.11.2.7 Blockage of Air Inlet and Outlet Openings

This accident conservatively postulates the complete blockage of the ventilation air inlet and outlet openings of the Standardized HSM, HSM-H, or HSM-HS.

T.11.2.7.1 Cause of Accident

No change to Section 8.2.7.1.

T.11.2.7.2 Accident Analysis

This event is evaluated in Section 8.2.7.2 for Standardized HSM with 24 kW heat load and is addressed in Section U.11.2.7.2 for the HSM-HS. The maximum heat load (22 kW) in the Type 1 61BTH DSC within a Standardized HSM is bounded by 24 kW. Therefore, the evaluation presented in Section 8.2.7.2 is also applicable to the Standardized HSM with the 61BTH DSC.

The thermal evaluation of this event is presented in Chapter T.4 for HSM-H and a 61BTH DSC. As discussed in Appendix T, Section T.4.4.2, the thermal evaluation for the HSM-H is also applicable to the HSM-HS. The temperatures determined in Chapter T.4 are used in the structural evaluation of this event, which is presented in Sections T.3.7.7 and T.3.4.4.3 for HSM-H and 61BTH DSC.

The section below describes the additional analyses performed to demonstrate the acceptability of the system with the NUHOMS®-61BTH DSC.

T.11.2.7.3 Accident Dose Calculations

There are no off-site dose consequences as a result of this accident. The only significant dose increase is that related to the recovery operation. Based on the results presented in Chapter T.5,

Table T.5-1 and Table T.5-3, the bounding average dose on HSM front or roof is *123.4* mrem/hr and 58.4 mrem/hr for the HSM-H and Standardized HSM, respectively.

It is conservatively estimated that the on-site workers will receive an additional dose of no more than *987 mrem (123.4 mrem/hr * 8 hours)* during an estimated eight hour period that may be required for removal of debris from the inlet and outlet vent openings. These exposures are well within the limits of 10CFR72.106 for an accident condition.

T.11.2.7.4 Corrective Action

No change to Section 8.2.7.4.

T.11.2.8 DSC Leakage

The NUHOMS®-61BTH DSC is designed as a pressure retaining containment boundary to prevent leakage of contaminated materials. The analyses of normal, off-normal, and accident conditions have shown that no credible conditions can breach the DSC shell or fail the double seal welds at each end of the DSC. The NUHOMS®-61BTH DSC is designed and tested to be leak tight [11.2]. Therefore DSC leakage is not considered a credible accident scenario. See Chapter T.7 for additional details on the confinement evaluation.

T.11.2.9 Accident Pressurization of DSC

T.11.2.9.1 Cause of Accident

The bounding internal pressurization of the NUHOMS®-61BTH DSC is postulated to result from cladding failure of the spent fuel in combination with the transfer accident case with the loss of sunshield and liquid neutron shield in the transfer cask under extreme ambient temperature conditions of 117ºF and maximum insolation and the consequent release of spent fuel rod fill gas and free fission gas. The evaluation conservatively assumes that 100% of the fuel rods have failed.

Table T.11-1
Comparison of Total Dose Rates for *Standardized* HSM with and without Adjacent HSM Shielding Effects

Distance from Nearest HSM Wall, 2x10 Array (meters)	Normal Case Dose Rate(1) (mrem/hr)	Accident Case Dose Rate(1) (mrem/hr)
10	5.1	10.2
100	0.11	0.22
500	5.3×10^{-4}	1.1×10^{-3}
600	1.9×10^{-4}	3.8×10^{-4}

(1) Air scattered plus direct radiation

Table T.11-2
Deleted