**Enclosure 8 to E-50037**

**CoC 1004 Amendment 15 Application, Revision 2**
**UFSAR Changed Pages**
**(Public Version)**

For the TC/DSC Lifting Heights as a Function of Low Temperature and Location, the basis for the low temperature and height limits is ANSI N14.6-1986 and -1993 paragraph 4.2.6, which requires at least 40 °F higher service temperature than nil ductility transition (NDT) temperature for the TC. In the case of the standardized TC, the test temperature is -40 °F; therefore, although the NDT temperature is not determined, the material will have the required 40 °F margin if the ambient temperature is 0 °F or higher. This assumes the material service temperature is equal to the ambient temperature.

The basis for the low temperature limit for the DSC is NUREG/CR-1815. The basis for the handling height limits is the NRC evaluation of the structural integrity of the DSC to drop heights of 80 inches and less.

For the NUHOMS®-24P, 52B and 61BT *Systems*, the basis for the high temperature limit is PNL-6189 for the fuel clad limit, the manufacturer's specification for neutron shield, and the design basis pressure of the TC internal cavity pressure. For the NUHOMS®-32PT, 24PHB and 24PTH *Systems*, the fuel cladding limits are based on ISG-11, Revision 2. For the NUHOMS®-61BTH *System* and the NUHOMS®-61BT *System* with FANP 9x9-2 fuel assemblies, the fuel cladding limits are based on ISG-11 Revision 3.

For the NUHOMS®-69BTH and 37PTH *Systems*, the fuel cladding limits are based on NUREG-1536, Revision 1.

For the TC/DSC Transfer at High Ambient Temperatures (32PTH1 DSC Only), the fuel cladding limits are based on ISG-11 Revision 3 (Reference 4).

*The basis for using a solar shield during transfer operations when ambient temperatures exceed 100 °F (106 °F for 32PTH1) is to prevent direct solar radiation on the exterior surface of the cask and to ensure that component temperatures remain below the allowable limits described in the UFSAR. With the solar shield in place, the temperatures on the exterior surface of the cask remain below those evaluated in the UFSAR.*

*The solar shield while not required for transfer operations with ambient temperatures below 100 °F (106 °F for 32PTH1) may also be used to protect the cask from rain or snow or other ambient conditions.*

B 10.5.3.2 Cask Drop

BASES

The basis for this specification is Section 8.2.5, "Accidental Cask Drop."

## M.1.5 Supplemental Data

The following Transnuclear drawings are enclosed:

1. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Main Assembly, Drawing NUH-32PT-1001-SAR.

2. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Shell Assembly, Drawing NUH-32PT-1002-SAR.

3. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Basket Assembly Plate Options 1, Drawing NUH-32PT-1003-SAR.

4. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Basket Assembly Tube Options 2, Drawing NUH-32PT-1004-SAR.

5. NUHOMS®-32PT Transportable Storage Canister for PWR Fuel, Aluminum Transition Rails, Drawing NUH-32PT-1006-SAR.

6. *NUHOMS®-32PT Transportable Storage Canister for PWR Fuel Failed Fuel Can Assembly, Drawing NUH32PT-1007-SAR*

7. *NUHOMS®-32PT Transportable Canister for PWR Fuel Damaged Fuel End Caps, Drawing NUH32PT-1008-SAR.*

8 7 6 5 4 3 2 1

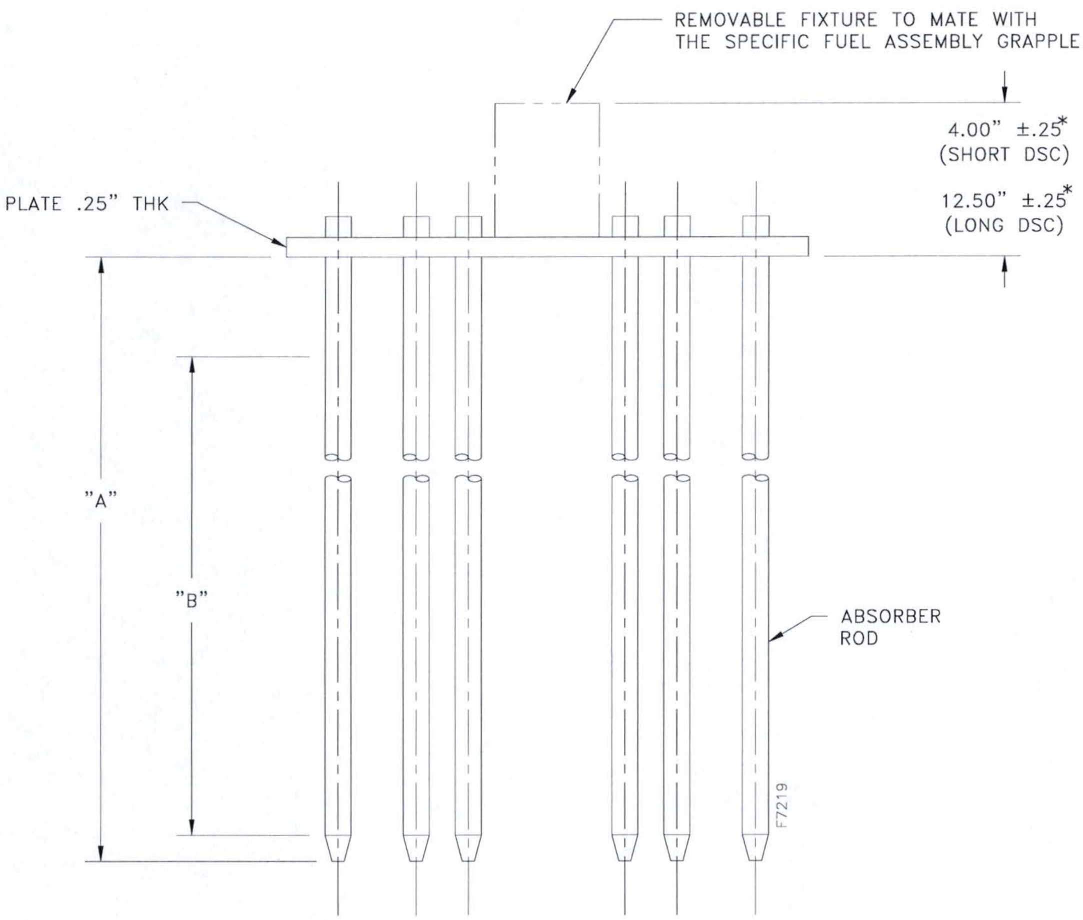
H G F E D C B A

**Proprietary and Security Related Information for Drawing NUH32PT-1008-SAR, Rev. 0A Withheld Pursuant to 10 CFR 2.390**

H G F E D C B A

8 7 6 5 4 3 2 1

All indicated changes are in response to RAI 7-1




*THESE DIMENSIONS ARE FOR USE WITH WESTINGHOUSE 17x17 FUEL ASSEMBLIES. DIMENSIONS WILL VARY AS REQUIRED BY FUEL ASSEMBLY TYPE.

| DIMENSION | FUEL ASSEMBLY TYPE | | | | |
|---|---|---|---|---|---|
| | WE 17x17 | B&W 15x15 | WE 15x15 | CE 14x14 | WE 14x14 |
| ABSORBER ROD OD NOMINAL (IN) | .362 | .438 | .450 | .975 | .432 |
| MINIMUM ABSORBER ROD DIMENSION "A"(IN) | 156 | 160 | 156 | 143 | 156 |
| ABSORBER STACK HEIGHT, "B" (IN) | 151 | 151 | 150 | 129 | 150 |
| CLAD THICKNESS NOMINAL (IN) | .018 | .022 | .023 | .049 | .022 |
| No. OF RODS | 24 | 16 | 20 | 5 | 16 |
| MATERIAL | 304 SST | 304 SST | 304 SST | 304 SST | 304 SST |

**Figure M.1-2**
**Poison Rod Assemblies (*$B_4C$-PRAs or AIC-PRAs*)**

*The NUHOMS®-32PT DSCs can also accommodate up to a maximum of 28 damaged fuel assemblies placed in cells located in the DSC as shown in Figures M.2-1 through M.2-3a. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks. 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 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.*

*The structural analysis for damaged fuel cladding described in Appendix M.3 demonstrates that the cladding does not undergo additional degradation under normal and off-normal conditions of storage. The criticality analysis described in Appendix M.6 limits the allowable contents for damaged fuel assemblies based on worst case geometry and material reconfigurations. The shielding analysis described in Appendix M.5 demonstrates that the worst configuration for damaged fuel assemblies is bounded by that of the intact fuel assemblies.*

*The NUHOMS®-32PT DSC is designed to accommodate up to a maximum of 8 failed fuel assemblies encapsulated in individual failed fuel cans (FFCs) and placed in cells located at the outer edge of the DSC as shown in Figure M.2-2. 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 an FFC 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 inches above the top of the bottom shield plug of the DSC.*

*Several different configurations with varying number of damaged fuel assemblies or failed fuel assemblies (balanced by intact fuel assemblies) are allowed for loading in the NUHOMS®-32PT DSC as shown in Figure M.2-7 through Figure M.2-9.*

The NUHOMS®-32PT DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than $UO_2$ fuel pellets except for the presence of a 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.

Throughout this Appendix BPRAs are considered as being representative of all CCs, unless specifically excluded.

For calculating the maximum internal pressure in the NUHOMS®-32PT DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged

## M.3.5 Fuel Rods

*All the fuel assembles that are designed to be loaded into the NUHOMS® 32PT DSC are similar to those evaluated for the NUHOMS® 37PTH DSC. The geometry, model data, loads, and boundary conditions used for the 37PTH DSC are all identical and applicable for the NUHOMS® 32PT DSC. Therefore, bounding fuel side and corner drop accident conditions for the 37PTH DSC performed in Sections Z.3.5.2 and Z.3.5.3, respectively, are also applicable to the 32PT DSC. Similarly, the normal and off-normal condition damaged fuel evaluation performed in Section Z.3.6.3 for the 37PTH DSC are considered bounding for the 32PT DSC.*

*The 37PTH DSC uses material allowable evaluated at 713 °F for side drop evaluations. The 32PT fuel cladding is made of a zirconium based alloy material, which includes "ZIRLO" and "Optimized ZIRLO." The maximum fuel cladding temperature for the 32PT DSC is 720 °F as provided in Section M.4.9.1.2. Therefore, the side drop results from Table Z.3.5-4 (at 713 °F) are compared against the reduced allowable yield strength of 93,950 psi for the zirconium based alloy at 725 °F. Results for the 32PT DSC are shown in Table M.3.5-1.*

*Corner drop and damaged fuel are evaluated at 750 °F for the 37PTH DSC. This condition bounds the maximum fuel cladding temperature of 720 °F for the 32PT DSC. Therefore, evaluations performed in Sections Z.3.5.3 and Z.3.6.3 for the 37PTH DSC for corner drop are applicable to the 32PT DSC. Similarly the structural evaluations performed for damaged fuel cladding in section Z.3.6.3 for the 37PTH DSC are applicable to the 32PT DSC. Therefore, the retrievability of the damaged fuel assemblies is assured under normal and off-normal loads.*

*The structural analysis documented in Appendix Z.3.6.3 conservatively evaluates a limiting configuration with a single rod and the spacer grids in designated locations without any support from fuel compartment to provide assurance of limiting additional cladding damage. The changes to fuel assembly configuration do not have impact on retrievability due to damage to spacer grids, as long as the assembly is able to be handled by normal means and the retrievability is ensured following the 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. The criticality analysis documented in Section M.6.4 also considers the impact of damage to the fuel assembly that includes missing and damaged grid spacers, which results in limiting the enrichment of these fuel assemblies. Therefore, additional configurations are not evaluated herein. Licensees can perform specific evaluations to demonstrate retrievability using actual configurations.*

**Table M.3.5-1**
**Summary of Stress Results for Side Drop**

| 75g Accident Load | WE 14x14 Std/ZCA | WE 15x15 Std/ZC | CE 16x16 SCE | WE 17x17 OFA/LOPAR/ RFA |
|---|---|---|---|---|
| **Max Bending Stress, $S_b$ (psi)** | 73,255 | 70,237 | 32,308 | 70,640 |
| **Internal Pressure (psi)** | 2,235 | 2,235 | 2,235 | 2,235 |
| **$S_{press}$ (psi)** | 11,200 | 10,271 | 8,729 | 8,381 |
| **Combined Stress (psi)(1)** | 84,455 | 80,508 | 41,037 | 79,021 |
| **Yield Stress at 725 °F (psi)(2)** | 93,950 | 93,950 | 93,950 | 93,950 |

Notes:

(1) Maximum combined stress results are taken from Table Z.3.5-4.

(2) The yield strength of 93,950 psi for zircaloy cladding tube at 725 °F.

*M.4.12.2 Thermal Evaluations for Failed Fuel Assemblies*

*The NUHOMS®-32PT DSC can accommodate up to 8 failed fuel cans (FFCs) in the corner locations of heat load zoning configuration (HLZC) #2 as shown in Figure M.4-2. Failed FAs are encapsulated in individual FFCs that are designed to fit into the existing 32PT basket fuel compartments at the corner locations. For this configuration, the maximum heat load per FFC is limited to 0.8 kW.*

*This section evaluates the thermal performance of the NUHOMS® -32PT DSC loaded with up to 8 FFCs during storage and transfer conditions.*

*M.4.12.2.1 Thermal Evaluation for Normal and Off-Normal Conditions*

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All indicated changes are in response to RAI 4-1

*Boundary conditions for the limiting case (normal transfer, 100 °F ambient with insolation) with HLZC #2 are taken from Section M.4.4.1.8. The DSC shell temperature profile based on the total DSC heat load of 24 kW from Section M.4.4.1.8 is conservatively applied for the limiting case with failed FAs.*

*Table M.4.12.2-1 lists the maximum temperatures of fuel cladding for intact FAs and the 32PT DSC components for the limiting case (normal transfer, 100 °F ambient with insolation) with 8 FFCs in HLZC#2 compared with the design basis values (normal transfer, 100 °F ambient with insolation) with all intact FAs in HLZC # 2 reported in Section M.4.4.2.*

***Table M.4.12.2-1***
***Maximum Temperature of Fuel Cladding and 32PT DSC Components without and with Failed FAs in FFCs for Limiting Case HLZC # 2***

| | ***32 intact FAs*** *(Tables M.4-2, M.4-4)* | ***24 intact FAs and 8 FFCs*** | ***Maximum ΔT*** |
|---|---|---|---|
| *Intact FA, °F* | *705* | [ | ] |
| *Grid, °F* | *691* | | |
| *Rail, °F* | *473* | | |
| *Poison, °F* | *691* | | |
| *Shell, °F* | *445* | | |
| *FFC wall, °F* | *n/a* | [ | ] |

[

]

*For the limiting case (normal transfer, 100 °F ambient with insolation) with 8 FFCs in HLZC#2, the average temperature of helium in the DSC cavity is* [ ]. *This temperature is below the design basis value of 550°F used in the maximum normal operating pressure evaluation from Section M.4.4.4.6 and maximum off-normal operating pressure evaluation from Section M.4.5.4.*

*Therefore, the design basis internal pressures in the 32PT DSC cavity for normal and off-normal operating conditions reported in Table M.4-7 and Table M.4-12 are bounding for the 32PT DSC with any combination of intact FAs and up to 8 FFCs in HLZC#2.*

*It is concluded that the NUHOMS®-32PT DSC for storage of up to 8 FFCs with HLZC # 2 meet all applicable normal and off-normal thermal requirements.*

For the limiting accident condition with 8 FFCs in HLZC#2, the average temperature of helium in the DSC cavity of [ ] is below the design basis value of 705 °F (see Section M.4.6.5). Therefore, the design basis values of the internal pressures in the 32PT DSC cavity for accident conditions as reported in Table M.4-15 are bounding for the 32PT DSC with HLZC #2 with 8 FFCs.

It is concluded that the NUHOMS®-32PT DSC for the storage of up to 8 failed FAs in FFCs of HLZC # 2 maintains confinement during the postulated accident condition.

M.4.12.3 Thermal Evaluations for Heat Load Zoning Configuration (HLZC) # 4

This section evaluates the thermal performance of the NUHOMS® -32PT DSC loaded in HLZC # 4 during storage and transfer conditions. The NUHOMS®-32PT DSC can be loaded with the maximum heat load per FA up to 2.2 kW in HLZC #4 as shown in Figure M.4-3a. For this configuration, the maximum heat load per DSC is limited to 24 kW.

The maximum heat load of 24 kW for HLZC #4 is equal to the maximum allowable heat load specified for HLZCs #1 and #2 as shown in Figures M.4-1 and M.4-2. Since no other changes are considered to the 32PT DSC except for the HLZC, the thermal evaluation of the 32PT DSC with HLZC #4 is based on a sensitivity analysis. A review of the maximum fuel cladding and basket component temperatures listed in Tables M.4-2 through M.4-5 and Tables M.4-8 through M.4-11 indicates that, among all the normal and off-normal conditions of storage and transfer operations, the normal transfer operation with HLZC #1 (DSC horizontal in cask, 100 °F ambient) represents the bounding case. Therefore, a sensitivity analysis based on the normal transfer operation (DSC horizontal in cask, 100 °F ambient) is selected to evaluate the thermal performance of the 32PT DSC with HLZC #4.

All thermal properties for the materials used in this sensitivity analysis are the same as those described in Section M.4.2.

Boundary conditions for the limiting case (DSC horizontal in cask, 100 °F ambient) with HLZC #4 are taken from Section M.4.4.1.8.

A review of the HLZC #4 in Figure M.4-3a indicates that if each fuel assembly is loaded at the maximum allowed heat load, the total heat load of the DSC would be *30.8* kW and will exceed the maximum allowable heat load of 24 kW for the 32PT DSC. However, to bound the various options that could be used to adjust the maximum heat load to 24 kW in HLZC # 4, the maximum heat load per fuel assembly listed in Figure M.4-3a is used in this evaluation. This is conservative since the total heat load of the DSC considered is *30.8* kW compared to the maximum allowable heat load of 24 kW.

Heat generation rates are calculated using the same methodology as described in Section M.4.4.1.4. Peaking factors and the minimum active fuel length are also the same as those described in Section M.4.4.1.4 with a correction factor of 1.013575 applied in the full-length model.

Table M.4.12.3-1 presents the maximum temperatures of fuel cladding and DSC components for the 32PT DSC with HLZC #4 from the sensitivity analysis compared to the design basis values for the normal transfer operation (DSC horizontal in cask, 100 °F ambient) with HLZC #1 as discussed in Section M.4.4.2 and listed in Tables M.4-2 and M.4-3.

As seen from Table M.4.12.3-1, the maximum fuel cladding temperature for the 32PT DSC with HLZC #4 is *705* °F with significant margin to the temperature limit of 752 °F. Further, the maximum fuel cladding temperature is also *15* °F lower than that determined previously for the design basis evaluation with HLZC #1. Therefore, the maximum fuel cladding temperatures of the 32PT DSC with HLZC #1 listed in Table M.4-2 remain bounding for the 32PT DSC with HLZC #4 for normal conditions.

**Table M.4.12.3-1**
**Maximum Component Temperatures for 32PT DSC for Normal Transfer, 100°F Ambient**

| HLZC | $T_{fuel,max}$ (°F) | $T_{grid,max}$ (°F) | $T_{rail,max}$ (°F) | $T_{Al,max}$ (1) (°F) |
|---|---|---|---|---|
| HLZC #1, Design Basis (See Tables M.4-2 and M.4-3) | 720 | 705 | 471 | 705 |
| HLZC #4 | *705* | *694* | *491* | *694* |
| ΔT, °F ($T_{HLZC\ \#4} - T_{Design\ Basis}$) | *-15* | *-11* | *20* | *-11* |

Similarly, for the basket components, the maximum temperatures determined for the 32PT DSC with HLZC #4 remain bounded by the maximum temperatures determined for HLZC #1 except for the rail wherein a temperature increase of *20* °F is observed. However, there is no impact of this increase on the performance of this component as discussed in Section M.3.6.1.3.2.

Table M.4.12.3-1 shows that normal and off-normal transfer operations of the 32PT DSC with HLZC #4 are bounded by design basis load cases for normal and off-normal conditions. Similarly for accident conditions, the maximum temperatures of fuel cladding and DSC components listed in Table M.4-13 and M.4-14 for accident conditions remain bounding for the 32PT DSC with HLZC #4.

The average helium temperature in DSC cavity for the 32PT DSC with HLZC # 4 is *559* °F (*1019* °R). This is a *9* °F increase compared to the design basis average helium temperature of 550°F (1010 °R) as noted in Section M.4.4.4.6. This small change has a negligible effect for the design basis maximum internal pressures listed in Table M.4-7, Table M.4-12 and Table M.4-15 for normal, off-normal and accident conditions.

*Thermal evaluation of the 32PT DSC with damaged fuel assemblies in Appendix M.4.12.1 shows that up to 28 damaged fuel assemblies can be stored in the 32PT DSC with HLZC #1 through HLZC # 3.*

*Based on the discussion presented in Appendix M.4.12.1.1, storage of damaged FAs within the 32PT DSC does not impact either the normal or the off-normal conditions. Similarly, there is no impact of storing damaged FAs per HLZC # 4 for normal/ off-normal conditions.*

*However as discussed in Appendix M.4.12.1.2, there is a potential for damaged FAs to further degrade during accident conditions. While storing damaged FAs, HZLC # 4 differs from HLZC #1 evaluated in Appendix M.4.12.1 as follows:*

1. *HLZC # 4 only allows up to 20 damaged FAs compared to 28 for HLZC # 1 through 3. A review of the results presented in Table M.4.12.1-1 shows that the number of damaged FAs has a significant impact on the maximum fuel cladding temperature of the intact FAs. Since the number of damaged FAs is significantly lower for HLZC # 4, this change is conservative.*

2. *Within HLZC # 4, the maximum decay heat load per FA in Zone 1 is 0.4 kW compared to 0.63 kW per FA in HLZC # 1 analyzed in Appendix M.4.12.1.*

   *A reduction in the maximum decay heat load per FA in Zone 1 is conservative and will help further reduce the maximum fuel cladding temperature during accident conditions.*

3. *Within HLZC # 4, the maximum decay heat load per FA in locations identified as Zone 3 and 4 is higher compared to that of HLZC # 1.*

   *For HLZC # 4, the maximum decay heat load per FA is 2.2 kW in Zone 3 and 1.7 kW in Zone 4 whereas the maximum decay heat load per FA in Zone 2 for HLZC # 1 is 0.87 kW. However, as seen from Table M.4.12.3-1, the maximum fuel cladding temperature for HLZC # 1 is higher. This is due to the reduction in the decay heat load per FA of the inner compartments while increasing the decay heat load per FA of the outer compartments. Since the outer compartments with higher decay heat load per FA in HLZC # 4 are limited to storage of intact fuel assemblies, a similar behavior will be observed.*

*Therefore, the thermal evaluation of the 32PT DSC with damaged fuel assemblies in Appendix M.4.12.1 remains bounding for HLZC # 4.*

*Based on the above discussion, no further evaluations are required for the 32PT DSC with HLZC #4 and all design criteria described are satisfied.*

*M.4.12.4 Thermal Evaluation for 32 Single-Poison- Plate Configuration*

*As discussed in Section M.4.4.1.1, the thermal evaluation of the 32PT DSC for the second alternate poison plate configuration consisting of either 24 poison plates paired with aluminum chevrons or 24 single poison plates is bounded by the thermal evaluation of the basket with the original poison plate configuration in Section M.4.4 through Section M.4.7. This section evaluates the thermal performance of the third alternate poison plate configuration consisting of 32 single-poison-plates.*

*All thermal properties used for the evaluation are the same as those listed in Section M.4.2. The material properties for the 32 single poison plates are the same as those specified in Section M.4.3.*

*Similar to the previously evaluated 24 single-poison-plate configuration shown in Figure M.4-22, the design change for the 32 single-poison-plate configuration is limited to the replacement of eight aluminum chevrons with eight single poison plates. Since no other changes are considered to the 32PT DSC, the thermal evaluation of the 32PT DSC with this design change is based on a sensitivity study. A review of HLZC #1 through HLZC #4 shown in Figure M.4-1 through Figure M.4-3a indicates that HLZC #3 is the bounding HLZC since it has the highest heat load per fuel assembly stored in the inner compartments among all HLZCs.*

*A review of the maximum fuel cladding temperatures for HLZC # 3 listed in Tables M.4-1, M.4-2, M.4-8 and M.4-13, the normal transfer operation (DSC horizontal transfer in cask, 100 °F, HLZC # 3) listed in Table M.4-2 represents the limiting case due to its smallest margin to the fuel cladding temperature limit among all storage and transfer conditions. Therefore, a sensitivity analysis based on the limiting case is selected to re-evaluate the thermal performance of the 32PT DSC with 32 single-poison-plate configuration.*

*The 32PT DSC thermal model described in Section M.4.4.1.1 is modified by*

- *selecting the elements representing the single aluminum plates or the poison plate paired with aluminum chevrons and the contact gaps between them , and*
- *modifying the material property to that of poison plates listed in Section M.4.3.*

*The boundary condition and heat generation for the limiting case (DSC horizontal transfer in cask, 100 °F, HLZC # 3) described in Section M.4.4.1 are used in the sensitivity evaluation.*

### *M.5.4.15 Impact on Dose Rates due to Reduced Density Concrete and Gaps between HSMs*

A bounding analysis is performed by employing a minimum concrete density of 140 pounds per cubic foot (pcf) in the HSM-H MCNP model combined with a maximum gap of 1.5 inches between adjacent HSM-H modules and shield walls to determine the effect on maximum and average dose rates due to a fully loaded 32PTH1 DSC. These calculations are documented in Appendix U.5, Section U.5.4.10. The ratios shown in Appendix U.5, Table U.5-18 and Table U.5-19 can be used as scaling factors to increase the maximum and surface-average dose rates of the *32PT DSC* in the HSM to account for low density concrete and 1.5-inch gaps during HSM fabrication and installation. Note that the HSM concrete contains high density rebar, which is not credited in the MCNP models. Further, the modules are installed adjacent to each other such that there will not be a "uniform" gap of 1.5 inches. Ignoring the effect due to increased vent dose rates, the increase in the average dose rates caused by both the maximum postulated uniform gaps and the minimum postulated concrete density is expected to be less than 20% at the front and roof surfaces of the HSM module. Dose reduction hardware may be installed to further reduce these dose rates.

### *M.5.4.16 Shielding Analysis with a Loading of 0.380 MTU per Fuel Assembly, Updated Bounding Source Terms, and New HLZC 4*

*The 32PT/OS197TC and 32PT/HSM dose rates computed by DORT and reported in Tables M.5-3, M.5-4, M.5-5, and M.5-23 must be scaled up to account for the following effects: (1) HLZC 4, which bounds HLZC 2 used in the DORT runs, (2) the source terms developed in Section M.5.2.7, which are more penalizing than the design basis source terms used in the DORT analysis, and (3) 0.380 MTU fuel assemblies, which results in a self-shielding penalty compared to the 0.475 MTU fuel assemblies used in the DORT analysis.*

*Dose rate calculations are performed using 3-D MCNP5 models. The approach is performed as follows: First, MCNP models are created to supplement the DORT analysis contained in this appendix. The maximum gamma, neutron, and total side on contact dose rates for the 32PT-L125 DSC in the OS197 TC using MCNP5 and 0.475 MTU fuel with HLZC 4 are calculated for the normal transfer configuration.*

*The MCNP5 model geometry is similar to Figure M.5-1, although the transition rails are modeled as solid aluminum, consistent with the licensing drawings. (The model depicted in Figure M.5-1 is used only to justify the DORT basket model homogenization assumptions, as detailed in Section M.5.4.7.) The HLZC 4 loading configuration is asymmetric, resulting in the heat load being higher in the upper hemisphere compared to the lower hemisphere. Only the upper right quadrant is modeled, with reflective boundaries modeled on the west and south surfaces. Modeling quarter symmetry is conservative because it results in a total basket heat load of (0.4+0.6*5+2.2*2)*4 = 31.2 kW rather than the HLZC 4 total of 24 kW.*

*The models are then benchmarked by comparing the MCNP results to those found using DORT with 0.475 MTU fuel for the normal transfer configuration. This comparison verifies that the MCNP models produce results which are conservatively representative for the 32PT-L125 DSC in the OS197 TC.*

All indicated changes are in response to RAI 6-5

- *The occupational exposure for the TC loading and storage operations is to be scaled by* $\boxed{1.25.}$

*These scaling factors are included as footnotes in the dose rate results summarized in Table M.5-3, M.5-4, M.5-5, and M.5-23.*

*These scaling factors are also employed to scale the occupational exposure and generic site dose (2X10 back-to-back and front-to-front arrays) results calculated for the 32PT System in Appendix M.10, and to scale the dose rate consequences of accidents for the 32PT System in Appendix M.11.*

All indicated changes are in response to RAI 7-4

*The AIC PRA absorber material is also allowed for the intact fuels, which is a very high thermal neutron absorber containing silver-indium-cadmium (AIC) alloy. No credit is taken for residual indium or cadmium isotopes in AIC PRAs. Table M.2-4a lists the minimum silver content employed for AIC PRAs for the WE 17x17 fuel class.*

*The CE 14x14, CE 15x15, WE 14x14 and WE 17x17 damaged fuel classes are qualified for transferring and storage in the NUHOMS®-32PT DSC. Several different configurations with varying number of damaged fuel assemblies (balanced by intact fuel assemblies) are allowed for loading in the NUHOMS®-32PT DSC. The CE 14x14, CE 15x15, WE 14x14 and WE 17x17 failed fuels are qualified for transferring and storage in the NUHOMS®-32PT DSC. Failed fuels are placed in failed fuel cans (FFCs) closed at top and bottom ends with an end cap. A maximum of eight failed fuel assemblies (balanced by intact fuel assemblies) are allowed for loading in the NUHOMS®-32PT DSC. The CC and PRA are not allowed for the fuel loading configurations containing either damaged fuels or the failed fuels. The results from the WE 17x17 fuel class for damaged and failed fuel configurations are applicable to the CE 15x15 fuel class.*

*For WE17, WE14 and CE14, above five damaged fuel configurations are analyzed to find the most reactive one which is used for the maximum initial enrichment determination. The methodology used in analysis of damaged fuel is similar to that employed for the NUHOMS®-32PTH1 DSC. The details of damaged fuel analysis methodology are given in Table M.6-72.*

*Determination of the Most Reactive Failed Configuration*

*The NUHOMS®-32PT DSC is designed to accommodate up to a maximum of eight failed fuel assemblies encapsulated in FFCs. Failed fuel rods are fuel rods that have been removed from another damaged or failed fuel assembly and placed in a secondary container. These include breached rods, grossly breached rods, other defective rods and fuel debris.*

*The FFCs are modeled as stainless steel containers containing failed fuel array. For conservatively simulating breached rods, the fuel rods are modeled without cladding, but the layer of full density fresh water in the fuel-cladding gap is retained. Rod pitch studies are performed on these failed fuel rods similar to that performed for damaged fuel to determine optimum fuel rod pitch.*

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*Additional sensitivity analyses performed for 32 failed WE 17x17 FAs with full rod lattice (17x17 failed rods) and reduced lattice (16x16 failed rods) show the maximum reactivity occurs for the reduced lattice (16x16) similar to the nominal lattice (17x17) when empty rod locations are replaced with fuel rods, as shown in Table M.6-74.*

*For WE17, WE14, and CE14, the maximum allowable enrichment is determined for each fuel type as a function of soluble boron concentration and basket type.*

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M.6.4.3 Studies with Radial Variation of Enrichment

The next set of analyses is for the Exxon/ANF 15x15 CE fuel assembly. These calculations justify the adequacy of the uniform maximum lattice average fuel enrichment assumed in the analysis. These calculations were carried out using a soluble boron concentration of 2500 ppm and a poison loading of 6.30 mg B-10/cm$^2$ with moderator densities (MD) varying from 40% to 100% of full density. The 16 poison plate configuration was utilized for these analyses and, the results are applicable to the 20 and 24 poison plate configuration and all basket types.

The different variable enrichments used in these calculations are shown in Table M.6-25. A plot of the fuel assembly with the radial variation in enrichment is shown in Figure M.6-17 and Figure M.6-18. The results for this evaluation are listed in Table M.6-26. Due to the higher initial enrichment values utilized to model the loading patterns 2A, 2B and 2C, the $k_{eff}$ values shown in Table M.6-26 are higher than USL. The purpose of these calculations is to determine the sensitivity of $k_{eff}$ due to radial variation of enrichment only and not to USL; hence, the results shown in Table M.6-26 are acceptable for these sensitivities studies.

The results demonstrate that the assumption of the uniform maximum lattice average enrichment is both appropriate and conservative. The $k_{eff}$ of the system with the uniform maximum lattice average enrichment assumption is greater than that of the non-uniform case at all MD.

M.6.4.4 Determination of the Maximum Initial Enrichment for each Fuel Class

*Determination of the Maximum Initial Enrichment for Intact Fuels*

The most reactive configuration as determined in Section M.6.3.1 above, is with solid aluminum in the transition rail region, nominal poison/aluminum plate thickness, minimum basket structure

All indicated changes are in response to RAI 7-3

***WE 17x17 Class Assemblies***

The most reactive WE 17x17 class assembly is the WE 17x17 LOPAR/standard assembly as demonstrated in Table M.6-6. The results for the WE 17x17 class assembly calculations for the 20 poison plate configuration are listed in Table M.6-13 and Table M.6-14 for cases without and with BPRAs, respectively. The results for the WE 17x17 class assembly calculations for the 16 poison plate configuration are listed in Table M.6-27 and Table M.6-28 for cases without and with BPRAs, respectively. The results for the WE 17x17 class assembly calculations for the 24 poison plate Type A/B/C/D basket configuration are listed in Table M.6-29 and Table M.6-30 for cases without and with BPRAs, respectively. The 24 poison plate Type A1 and A2 basket configuration results are listed in Tables M.6-51 and M.6-57, respectively.

*Additional configurations with the 24 poison plate design include $B_4C$ PRAs (04 and 08 in number) and AIC PRAs (04 and 08 in number). Criticality analysis for the 32 poison plate configuration is performed with and without BPRAs. These configurations are evaluated at 2500 ppm and 2800 ppm soluble boron concentrations, and the maximum allowable enrichments are determined. The results for these configurations are given in Table M.6-63.*

*The poison rod radius is modeled as 0.902 cm in the KENO model. Considering the initial AIC composition is Ag 80 wt. %, In 15 wt. % and Cd 5 wt. % with a density of 10.17 g/cm$^3$, the initial linear mass of Ag is 6.14 g/cm per rod of AIC. 40% of the initial linear mass of Ag, or 2.46 g/cm, is required, see Table M.2-4a and a fraction of the required amount of Ag, 75% for this poison rod radius, is credited in the KENO model.*

*A sensitivity analysis with a smaller poison pellet radius is performed for the Type B1-r case with 4-RCCA presented in Table M.6-1 (Part 3 of 3): WE17 fuel assembly without CC 4.60 wt% in presence of 2800 ppm boron. The maximum $k_{eff}$ is 0.9370 (80% water density – 2800 ppm boron), Table M.6-63 (Part 4 of 5). The sensitivity is performed for a smaller poison pellet radius of 0.43307 cm (or 0.86614 cm diameter), see AIC diameter Control Rod specification for WE 17x17 fuel assembly in Table 4 of NUREG/CR-6759, with 90% of the 2.46 g/cm required linear mass of Ag in the poison rod. The maximum $k_{eff}$ is 0.9372 (75% water density – 2800 ppm boron). The $k_{eff}$ is statistically equivalent to that of the base case. The sensitivity analysis shows that the required 2.46 g/cm initial linear mass of Ag in AIC is appropriate for the criticality safety of the 32PT DSC when AIC RCCA is credited for the configurations shown in Table M.6-1 (Part 3 of 3).*

***B&W 15x15 Class Assemblies***

The most reactive B&W 15x15 class assembly is the B&W 15x15 Mark B assembly as demonstrated in Table M.6-6. The results for the B&W 15x15 class assembly calculations for the 20 poison plate configuration are listed in Table M.6-15 and Table M.6-16 for cases without and with BPRAs, respectively. The results for the B&W 15x15 class assembly calculations for the 16 poison plate configuration are listed in Table M.6-31 and Table M.6-32 for cases without and with BPRAs, respectively. The results for the B&W 15x15 class assembly calculations for the 24 poison plate Type A/B/C/D basket configuration are listed in Table M.6-33 and Table M.6-34 for cases without and with BPRAs, respectively. The 24 poison plate Type A1 and A2 basket configuration results are listed in Tables M.6-49 and M.6-55, respectively.

### *CE 15x15 Class Assemblies*

The most reactive CE 15x15 class assembly is the CE 15x15 Palisades assembly as demonstrated in Table M.6-6. The results for the CE 15x15 class assembly calculations for the 20 poison plate configuration are listed in Table M.6-17 for cases without BPRAs. BPRAs are not authorized to be stored with CE 15x15 class assemblies.

The addition of plugging cluster assemblies, i.e., steel rods, into each of the eight guide tubes of a CE 15x15 class assembly reduces the maximum reactivity of the payload. The introduction of the steel rods displaces both moderator and soluble boron within the assemblies. The plugging clusters are assumed to extend approximately 1 inch into the top of the assembly's active fuel region, and the resulting change in the maximum reactivity is less than the statistical uncertainties of the calculations. To demonstrate the *effect* of displacing the borated water on system reactivity, CE 15x15 Palisades cases with the highest fuel enrichments and highest soluble boron loadings (2300, 2400, and 2500 ppm boron) were reevaluated with steel in the guide tubes. Two scenarios were evaluated: full length steel rods and 1 inch long steel rods. The

calculated reactivity of the models are shown in Table M.6-44 and Table M.6-45 (the $k_{KENO}$ and $1\sigma$ values in columns 2 and 3 are from Table M.6-17). As shown therein, the addition of plugging clusters reduces reactivity of the system regardless of the length of the plugging cluster. These results also apply to the 16 and 24 poison plate configurations.

The results for the CE 15x15 class assembly calculations for the 16 and 24 poison plate configuration as well as the 24 position plate and Type A basket configuration as a function of boron loading are listed in Table M.6-41. The results for the 24 poison plate and Type A1 and A2 basket configuration are listed in Tables M.6-48 and M.6-54, respectively.

*Additional configurations with the 24 poison plate design (at 2800 ppm) and the 32 poison plate design are analyzed. The maximum allowable enrichments are determined for a soluble boron concentration at 2500 ppm and 2800 ppm. The result for criticality analysis of these configurations is given in Table M.6-64.*

***WE 15x15 Class Assemblies***

The most reactive WE 15x15 class assembly is the WE 15x15 Standard assembly as demonstrated in Table M.6-6. The results for the WE 15x15 class assembly calculations for the 20 poison plate configuration are listed in Table M.6-18 for cases without BPRAs. The results for the WE 15x15 class assembly calculations for the 16 poison plate configuration are listed in Table M.6-36 for cases without BPRAs. The results for the WE 15x15 class assembly calculations for the 24 poison plate Type A/B/C/D basket configuration are listed in Table M.6-37 for cases without BPRAs. The 24 poison plate Type A1 and A2 basket configuration results, with and without BPRAs, are listed in Tables M.6-50 and M.6-56, respectively.

***CE 14x14 Class Assemblies***

The most reactive CE 14x14 class assembly is the CE 14x14 Fort Calhoun assembly as demonstrated in Table M.6-6. The results for the CE 14x14 class assembly calculations for the 20 poison plate configuration as a function of boron loading are listed in Table M.6-19 for cases without BPRAs.

The results for the CE 14x14 class assembly calculations for the 16 poison plate configuration as well as the 24 position plate and Type A/B/C/D basket configuration as a function of boron loading are listed in Table M.6-42. The results for the 24 poison plate and Type A1 and A2 basket configuration results, with and without BPRAs, are listed in Tables M.6-47 and M.6-53, respectively.

***WE 14x14 Class Assemblies***

The most reactive WE 14x14 class assembly is the WE 14x14 ZCA/ZCB assembly as demonstrated in Table M.6-6. The results for the WE 14x14 class assembly calculations for the 20 poison plate configuration as a function of boron loading are listed in Table M.6-20 for cases without BPRAs.

The results for the WE 14x14 class assembly calculations for the 16 poison plate configuration as well as the 24 position plate and Type A/B/C/D basket configuration as a function of boron

loading are listed in Table M.6-43. The results for the 24 poison plate and Type A1 and A2 basket configuration results, with and without BPRAs, are listed in Tables M.6-46 and M.6-52, respectively.

<u>Determination of the Maximum Initial Enrichment for Damaged fuels</u>

***WE 17x17 Damaged Fuel Assemblies***

*WE 17x17 damaged fuel assemblies are allowed for loading in the NUHOMS®-32PT DSC with the 24 (A1 and A2 basket type) poison plates design, as well as the 32 poison plates design (A1-32 and A2-32 basket types). The most reactive configuration (MRC) is ascertained by means of the criticality analysis of all damaged fuel scenarios previously discussed, and it was shown that rod pitch expansion yields the MRC. The MRC is then used to determine the maximum allowable enrichments for various basket types at different values of soluble boron concentration. The results for damaged fuel configurations analyzed with WE 17x17 fuel are given in Table M.6-65. The maximum allowable enrichments for the A1, A2, A1-32, and A2-32 basket types are shown in Table M.6-68.*

*The 28 damaged, 4 intact WE 17x17 FA loading configuration is examined in the 24-poison plate Type A1 DSC with 2500 ppm of soluble boron in the water for the sensitivity analysis with the CE 15x15 fuel. The maximum $k_{eff}$ of 0.9386 occurs at a maximum enrichment of 4.05 wt. % with 70% water density in the model, as shown in Table M.6-68. The sensitivity analysis for the CE 15x15 fuel class is performed in the most reactive configuration, which is a maximum pitch within the 32PT compartment opening. The maximum $k_{eff}$ of 0.9371 occurs at a maximum enrichment of 4.05 wt. % with 75% water density in the model.*

*The maximum planar average enrichments determined for the WE 17x17 fuel class using the 28 damaged, 4 intact loading configuration reported in Table M.6-61 are applicable to the CE 15x15 fuel class.*

***WE 14x14 Damaged Fuel Assemblies***

*For WE 14x14 assemblies, a similar procedure is followed to establish that the rod pitch study for a configuration with 32 damaged WE 14x14 assemblies is the most reactive out of all the damaged fuel scenarios discussed previously. The MRC is used to establish the maximum allowable enrichments for basket type A1, as well as A2, at different values of soluble boron concentration. The results for damaged fuel configurations analyzed with WE 14x14 fuel are given in Table M.6-66. The maximum allowable enrichments for the A1 and A2 basket types are given in Table M.6-69.*

***CE 14x14 Damaged Fuel Assemblies***

*For CE14 damaged fuel assemblies, the double shearing is the most reactive configuration, which is followed by the rod pitch expansion with a $\Delta k_{eff}$ of 0.0010. Considering the small difference in $k_{eff}$, the rod pitch expansion is selected as the MRC for CE 14x14, same as for WE17x17 and WE14x14. The established MRC is used to determine the maximum allowable enrichment for basket type A1 and A2 with 16, 28 and 32 damaged fuels assemblies. The loading configurations for 16 and 28 damaged fuels are shown in Figure M.2-7 and M.2-8. The results for the damaged fuel configurations analyzed with CE 14x14 fuel are given in Table M.6-67. The results for loading curves are given in Table M.6-70.*

*Determination of the Maximum Initial Enrichment for Failed fuels*

***WE 17x17 Failed Fuel***

*As is previously described in Section M.6.4.2 for CE 14x14 and WE 14x14 classes, the most reactive configuration is determined using a 17x17 failed fuel pin array with the optimum pin pitch. Once the MRC is determined, the maximum allowable enrichment has been deduced for*

*A1, A2, A1-32 and A2-32 basket types with a 14x14, 15x15, 16x16 and a 17x17 failed fuel array size at 2500 and 2800 ppm. The results for the criticality analysis of WE 17x17 failed fuel are given in Table M.6-71.*

*The 8 failed, 24 intact WE 17x17 fuel (15x15 failed fuels array) loading configuration is examined in the 24-poison plate Type A1 DSC with 2500 ppm of soluble boron in the water for the sensitivity analysis with the CE 15x15 fuel. The maximum $k_{eff}$ of 0.9345 occurs at a maximum enrichment of 4.10 wt. % as shown in Table M.6-71. The sensitivity analysis for the CE 15x15 fuel class is performed in the most reactive configuration, which is a maximum pitch within the 32PT compartment opening. The maximum $k_{eff}$ of 0.9349 occurs at a maximum enrichment of 4.10 wt. %.*

*The maximum planar average enrichments determined for the WE 17x17 fuel class using the 8 failed, 24 intact loading configuration reported in Table M.6-62 are applicable to the CE 15x15 fuel class. Note that the 16x16 and 17x17 lattice results are not applicable to the CE 15x15 fuel class.*

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***WE 14x14 Failed Fuel***

*The most reactive configuration is determined using a 14x14 failed fuel pin array with the optimum pin pitch as described in Section M.6.4.2. The MRC is subsequently used to determine the maximum allowable enrichment values for basket types A1 and A2 at 1800 ppm, 2100 ppm, 2300 ppm and 2500 ppm for WE 14x14 failed fuel loading curve. The results for the criticality analysis of WE 14x14 failed fuel in A1 and A2 baskets are given in Table M.6-72.*

***CE 14x14 Failed Fuel***

*The most reactive configuration is determined using the 14x14 failed fuel pin array with the optimum pin pitch as described in Section M.6.4.2. The MRC is subsequently used to determine the maximum allowable initial enrichment for the A1 and A2 basket types at 2600 ppm soluble boron concentration. The results for the criticality analysis of CE 14x14 failed fuel are given in Table M.6-73.*

*When intact, damaged, or failed fuel are loaded per Technical Specification Figure 1-4b, the maximum enrichments for the three fuel types are restricted with the lowest enrichment shown across Table M.6-1, Table M.6-61 and Table M.6-62 considering the same poison plate configuration and soluble boron loading.*

M.6.4.5 Criticality Results

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Intact Fuel by SCALE 4.4

Table M.6-21 lists the bounding results for all conditions of storage. The highest calculated $k_{eff}$, including $2\sigma$ uncertainty, is for the WE 14x14 assembly in the 20 poison plate configuration with an initial U-235 enrichment of 3.85 wt. %, no PRAs, 2000 ppm boron, and 60% moderator density. The maximum allowed initial enrichment for each assembly type/PRA configuration is listed in Table M.6-1.

All indicated changes are in response to RAI 7-4

### *Table M.6-61*
### *Maximum Initial Enrichment for Each Configuration, wt. % U-235 – Damaged Fuel*
*(Part 1 of 2)*

| Assembly Class and Type | Soluble Boron Loading (ppm) | 28 Damaged Fuels | |
|---|---|---|---|
| | | 24 Poison Plate Configuration | |
| | | Type A1 | Type A2 |
| WE 17x17 fuel assembly (without CC) | 2500 | 4.05 | 4.20 |
| CE 15x15 fuel assembly (without CC) | 2800 | 4.30 | 4.45 |
| **Assembly Class and Type** | **Soluble Boron Loading (ppm)** | **32 Damaged Fuels** | |
| | | **24 Poison Plate Configuration** | **24 Poison Plate Configuration** |
| | | **Type A1** | **Type A1** |
| WE 14x14 fuel assembly (without CC) | 1800 | 3.80 | 3.95 |
| | 2100 | 4.10 | 4.25 |
| | 2300 | 4.30 | 4.45 |
| | 2500 | 4.50 | 4.65 |
| CE 14x14 fuel assembly (without CC) | 1800 | 3.60 | 3.75 |
| | 2100 | 3.90 | 4.05 |
| | 2300 | 4.10 | 4.25 |
| | 2500 | 4.30 | 4.45 |
| | 2600 | 4.35 | 4.55 |
| **Assembly Class and Type** | **Soluble Boron Loading (ppm)** | **28 Damaged Fuels** | |
| | | **32 Poison Plate Configuration** | |
| | | **Type A1-32** | **Type A2-32** |
| WE 17x17 fuel assembly (without CC) | 2500 | 4.45 | 4.65 |
| CE 15x15 fuel assembly (without CC) | 2800 | 4.70 | 4.95 |

### *Table M.6-61*
### *Maximum Initial Enrichment for Each Configuration, wt. % U-235 – Damaged Fuel*
*(Part 2 of 2)*

| Assembly Class and Type | Soluble Boron Loading (ppm) | 28 Damaged Fuels | |
|---|---|---|---|
| | | 24 Poison Plate Configuration | |
| | | Type A1 | Type A2 |
| CE 14x14 fuel assembly (without CC) | 1800 | 3.70 | 3.85 |
| | 2100 | 4.00 | 4.15 |
| | 2300 | 4.20 | 4.35 |
| | 2500 | 4.40 | 4.50 |
| | 2600 | 4.45 | 4.65 |
| **Assembly Class and Type** | **Soluble Boron Loading (ppm)** | **16 Damaged Fuels** | |
| | | **24 Poison Plate Configuration** | |
| | | **Type A1** | **Type A2** |
| CE 14x14 fuel assembly (without CC) | 1800 | 3.80 | 3.95 |
| | 2100 | 4.10 | 4.25 |
| | 2300 | 4.30 | 4.45 |
| | 2500 | 4.50 | 4.70 |
| | 2600 | 4.60 | 4.80 |

All indicated changes are in response to RAI 7-5

**Table M.6-62**
***Maximum Initial Enrichment for Each Configuration, wt. % U-235 – Failed Fuel***

| *Assembly Class and Type* | *FFC Array Size* | *Soluble Boron Loading (ppm)* | *8 Failed Fuels* | |
|---|---|---|---|---|
| | | | *24 Poison Plate Configuration* | |
| | | | *Type A1* | *Type A2* |
| *WE 17x17 fuel assembly (without CC)*<br>*CE 15x15 fuel assembly (without CC)* | *14x14* | *2500* | *4.15* | *4.30* |
| | | *2800* | *4.40* | *4.60* |
| | *15x15* | *2500* | *4.10* | *4.25* |
| | | *2800* | *4.40* | *4.55* |
| | *16x16(*)* | *2500* | *4.05* | *4.20* |
| | | *2800* | *4.35* | *4.50* |
| | *17x17(*)* | *2500* | *4.00* | *4.15* |
| | | *2800* | *4.30* | *4.45* |
| *WE 14x14 fuel assembly (without CC)* | *14x14* | *1800* | *4.15* | *4.30* |
| | | *2100* | *4.50* | *4.70* |
| | | *2300* | *4.75* | *4.95* |
| | | *2500* | *4.95* | *5.00* |
| *CE 14x14 fuel assembly (without CC)* | *14x14* | *2600* | *4.70* | *4.90* |
| *Assembly Class and Type* | *FFC Array Size* | *Soluble Boron Loading (ppm)* | *8 Failed Fuels* | |
| | | | *32 Poison Plate Configuration* | |
| | | | *Type A1-32* | *Type A2-32* |
| *WE 17x17 fuel assembly (without CC)*<br>*CE 15x15 fuel assembly (without CC)* | *14x14* | *2500* | *4.55* | *4.75* |
| | | *2800* | *4.85* | *5.00* |
| | *15x15* | *2500* | *4.50* | *4.70* |
| | | *2800* | *4.80* | *5.00* |
| | *16x16(*)* | *2500* | *4.45* | *4.65* |
| | | *2800* | *4.75* | *4.95* |
| | *17x17(*)* | *2500* | *4.40* | *4.60* |
| | | *2800* | *4.65* | *4.90* |

*(*) Not applicable to CE 15x15 fuel assembly*

All indicated changes are in response to RAI 7-4

*Table M.6-68*
*Damaged Fuel Criticality Evaluation - WE17*
*(Part 1 of 2)*

| Basket Type A1: 28-Damaged Fuel Assemblies | | | |
|---|---|---|---|
| **2500 ppm** | **Optimum Rod Pitch** | **4.05 wt. % U-235** | |
| **IMD (%)** | $k_{KENO}$ | $\sigma_{KENO}$ | $k_{EFF}$ |
| 60 | 0.9289 | 0.0007 | 0.9302 |
| 65 | 0.9339 | 0.0007 | 0.9353 |
| 70 | 0.9373 | 0.0007 | 0.9387 |
| 75 | 0.9344 | 0.0007 | 0.9358 |
| 100 | 0.9197 | 0.0006 | 0.9209 |
| **2800 ppm** | **Optimum Rod Pitch** | **4.30 wt. % U-235** | |
| **IMD (%)** | $k_{KENO}$ | $\sigma_{KENO}$ | $k_{EFF}$ |
| 60 | 0.9320 | 0.0008 | 0.9336 |
| 65 | 0.9351 | 0.0007 | 0.9365 |
| 70 | 0.9358 | 0.0006 | 0.9370 |
| 75 | 0.9350 | 0.0006 | 0.9362 |
| 100 | 0.9149 | 0.0007 | 0.9163 |
| **Basket Type A2: 28-Damaged Fuel Assemblies** | | | |
| **2500 PPM** | **Optimum Rod Pitch** | **4.20 wt. % U-235** | |
| **IMD (%)** | $k_{KENO}$ | $\sigma_{KENO}$ | $k_{EFF}$ |
| 70 | 0.9341 | 0.0006 | 0.9354 |
| 75 | 0.9361 | 0.0006 | 0.9373 |
| 80 | 0.9349 | 0.0007 | 0.9362 |
| 90 | 0.9309 | 0.0006 | 0.9322 |
| 100 | 0.9227 | 0.0007 | 0.9240 |
| **2800 PPM** | **Optimum Rod Pitch – 28-Damaged model** | **28- Damaged fuels: 4.45 wt. % U-235** | |
| **IMD (%)** | $k_{KENO}$ | $\sigma_{KENO}$ | $k_{EFF}$ |
| 70 | 0.9332 | 0.0007 | 0.9347 |
| 75 | 0.9349 | 0.0006 | 0.9361 |
| 80 | 0.9331 | 0.0007 | 0.9345 |
| 90 | 0.9277 | 0.0006 | 0.9289 |
| 100 | 0.9183 | 0.0006 | 0.9195 |

All indicated changes are in response to RAI 7-4

***Table M.6-68***
***Damaged Fuel Criticality Evaluation - WE17***
*(Part 2 of 2)*

| Basket Type A1-32: 32 Damaged Fuel Assemblies | | | |
|---|---|---|---|
| **2500 ppm** | **Optimum Rod Pitch** | **4.45 wt. % U-235** | |
| **IMD (%)** | $k_{KENO}$ | $\sigma_{KENO}$ | $k_{EFF}$ |
| 70 | 0.9345 | 0.0006 | 0.9358 |
| 75 | 0.9350 | 0.0007 | 0.9364 |
| 80 | 0.9361 | 0.0006 | 0.9373 |
| 90 | 0.9341 | 0.0006 | 0.9354 |
| 100 | 0.9258 | 0.0006 | 0.9271 |
| **2800 ppm** | **Optimum Rod Pitch** | **4.70 wt. % U-235** | |
| **IMD (%)** | $k_{KENO}$ | $\sigma_{KENO}$ | $k_{EFF}$ |
| 70 | 0.9331 | 0.0007 | 0.9345 |
| 75 | 0.9344 | 0.0007 | 0.9357 |
| 80 | 0.9342 | 0.0006 | 0.9354 |
| 90 | 0.9283 | 0.0007 | 0.9297 |
| 100 | 0.9196 | 0.0008 | 0.9211 |
| **Basket Type A2-32: 32 Damaged Fuel Assemblies** | | | |
| **2500 PPM** | **Optimum Rod Pitch** | **4.65 wt. % U-235** | |
| **IMD (%)** | $k_{KENO}$ | $\sigma_{KENO}$ | $k_{EFF}$ |
| 70 | 0.9329 | 0.0008 | 0.9344 |
| 75 | 0.9342 | 0.0007 | 0.9356 |
| 80 | 0.9348 | 0.0007 | 0.9362 |
| 90 | 0.9334 | 0.0007 | 0.9349 |
| 100 | 0.9280 | 0.0007 | 0.9293 |
| **2800 PPM** | **Optimum Rod Pitch** | **4.95 wt. % U-235** | |
| **IMD (%)** | $k_{KENO}$ | $\sigma_{KENO}$ | $k_{EFF}$ |
| 70 | 0.9343 | 0.0007 | 0.9357 |
| 75 | 0.9360 | 0.0007 | 0.9373 |
| 80 | 0.9356 | 0.0007 | 0.9370 |
| 90 | 0.9316 | 0.0007 | 0.9329 |
| 100 | 0.9253 | 0.0007 | 0.9267 |

All indicated changes are in response to RAI 7-5

*Table M.6-74*
***Sensitivity with WE 17x17 fuels – Full Lattice and Reduced Lattice with Deletion of Rods***

| *Model Description* | $k_{KENO}$ | $1\sigma$ | $k_{eff} = k_{KENO} + 2\sigma$ |
|---|---|---|---|
| ***WE 17x17 Fuel Assembly, 4.40 wt. %, 32 poison plates, Type A1 (15 mg B-10/cm$^2$), 2500 ppm*** | | | |
| *17x17 failed rod lattice* | | | |
| *17x17, Pitch = 0.496", IMD=60%* | *0.9568* | *0.0007* | *0.9582* |
| *17x17, Pitch = 0.496", IMD=60%* | *0.9616* | *0.0006* | *0.9629* |
| *17x17, Pitch = 0.496", IMD=70%* | *0.9663* | *0.0006* | *0.9675* |
| *17x17, Pitch = 0.496", IMD=75%* | *0.9705* | *0.0007* | *0.9719* |
| ***17x17, Pitch = 0.496", IMD=80%*** | ***0.9740*** | ***0.0007*** | ***0.9753*** |
| *17x17, Pitch = 0.496", IMD=90%* | *0.9725* | *0.0007* | *0.9739* |
| *17x17, Pitch = 0.496", IMD=100%* | *0.9682* | *0.0007* | *0.9696* |
| *17x17, 0.496", Remove 02 Rods, IMD=80%* | *0.9700* | *0.0007* | *0.9714* |
| *17x17, 0.496", Remove 04 Rods, IMD=80%* | *0.9689* | *0.0008* | *0.9705* |
| *17x17, 0.496", Remove 06 Rods, IMD=80%* | *0.9663* | *0.0006* | *0.9676* |
| *17x17, 0.496", Remove 08 Rods, IMD=80%* | *0.9637* | *0.0006* | *0.9649* |
| *17x17, 0.496", Remove 10 Rods, IMD=80%* | *0.9625* | *0.0007* | *0.9639* |
| *16x16 failed rod lattice* | | | |
| *16x16, Pitch = 0.527", IMD=60%* | *0.9384* | *0.0009* | *0.9402* |
| *16x16, Pitch = 0.527", IMD=60%* | *0.9435* | *0.0007* | *0.9448* |
| *16x16, Pitch = 0.527", IMD=70%* | *0.9459* | *0.0006* | *0.9471* |
| *16x16, Pitch = 0.527", IMD=75%* | *0.9441* | *0.0006* | *0.9453* |
| ***16x16, Pitch = 0.527", IMD=80%*** | ***0.9444*** | ***0.0007*** | ***0.9457*** |
| *16x16, Pitch = 0.527", IMD=90%* | *0.9383* | *0.0007* | *0.9397* |
| *16x16, Pitch = 0.527", IMD=100%* | *0.9301* | *0.0006* | *0.9313* |
| *16x16, 0.527", Remove 02 Rods, IMD=80%* | *0.9425* | *0.0006* | *0.9437* |
| *16x16, 0.527", Remove 04 Rods, IMD=80%* | *0.9389* | *0.0007* | *0.9404* |
| *16x16, 0.527", Remove 06 Rods, IMD=80%* | *0.9367* | *0.0007* | *0.9380* |
| *16x16, 0.527", Remove 08 Rods, IMD=80%* | *0.9326* | *0.0007* | *0.9340* |
| *16x16, 0.527", Remove 10 Rods, IMD=80%* | *0.9281* | *0.0006* | *0.9294* |

All indicated changes are in response to RAI 8-2

## M.8.1 Procedures for Loading the Cask

Process flow diagrams for the NUHOMS® System operation are presented in Figure M.8-1 and Figure M.8-2. The location of the various operations may vary with individual plant requirements. The following steps describe the recommended generic operating procedures for the standardized NUHOMS® System.

### M.8.1.1 Preparation of the TC and DSC

**NOTE**: If using the OS200 TC for transfer of the NUHOMS®-32PT DSC, verify that it has been fitted with an internal aluminum sleeve (refer to Drawing NUH-08-8004-SAR provided in Appendix U, Section U.1.5). This step, if required, can be performed at any time prior to placing the DSC in the TC.

1. Prior to placement in dry storage, the candidate intact *and damaged and failed* fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification 2.1. Depending on the length of the fuel assemblies to be loaded, fuel spacers may be placed within the DSC to reduce the fuel assembly/DSC cavity gap in consideration of Part 71requirements. There are no requirements for fuel spacers under Part 72. Fuel spacers, if used, may be placed below the assembly, above the assembly, or both, and shall be evaluated for any adverse impact.

2. Prior to being placed in service, the TC is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification 5.4.2.d. Prior to being placed in service, the DSC should have the top shield plug, inner top cover and outer top cover test fitted and removed.

3. Place the TC in the vertical position in the cask decon area using the cask handling crane and the TC lifting yoke.

4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.

5. Remove the TC top cover plate and examine the cask cavity for any physical damage and ready the cask for service.

5a. If using the OS200 TC to load, verify that a cask spacer of appropriate height (refer to Drawing NUH-08-8005-SAR provided in Appendix U, Section U.1.5) is placed at the bottom of the TC.

6. Examine the DSC for any physical damage which might have occurred since the receipt inspection was performed. The DSC is to be cleaned and any loose debris removed.

7. Using a crane, lower the DSC into the cask cavity by the internal lifting lugs and rotate the DSC to match the cask and DSC alignment marks.

8. Fill the cask-DSC annulus with clean, demineralized water. Place the inflatable seal into the upper cask liner recess and seal the cask-DSC annulus by pressurizing the seal with compressed air.

8a. Place and verify that the bottom fuel assembly spacers, if required, are present in the fuel cells. Optionally, this step may be performed at any prior time.

9. *If damaged fuel assemblies are included in a specific campaign, place the required number of bottom end caps into the cell locations per Technical Specifications. Place and verify that the bottom fuel assembly spacers, if required, are present in the fuel cells. Optionally, this step may be performed at any prior time.*

10. *If failed fuel is to be loaded in the DSC, place the empty failed fuel can in the appropriate locations in the DSC in the appropriate locations in the DSC per Technical Specifications.*

11. Fill the DSC cavity with water from the fuel pool or an equivalent source which meets the requirements of Technical Specification 3.2.1. If loading 32PT-S100 or 32PT-L100 DSC (qualified for 100-ton crane capacity), drain neutron shield water from the TC.

    **NOTE:** A TC/DSC annulus pressurization tank filled with demineralized water as described above is connected to the top vent port of the TC via a hose to provide a positive head above the level of water in the TC/DSC annulus. This is an optional arrangement, which provides additional assurance that contaminated water from the fuel pool will not enter the TC/DSC annulus, provided a positive head is maintained at all times.

12. Position the cask lifting yoke and engage the cask lifting trunnions.

13. Visually inspect the yoke lifting hooks to insure that they are properly positioned and engaged on the cask lifting trunnions.

14. Connect the vacuum drying system (VDS) or optional liquid pump to the siphon port of the DSC and position the connecting hose such that the hose will not interfere with loading (yoke, fuel, shield plug, rigging, etc.). A flowmeter must be installed at a suitable location as part of this connection.

15. Move the scaffolding away from the cask as necessary.

16. Lift the cask just far enough to allow the weight of the cask to be distributed onto the yoke lifting hooks. Reinspect the lifting hooks to insure that they are properly positioned on the cask trunnions.

17. Optionally, secure a sheet of suitable material to the bottom of the TC to minimize the potential for ground-in contamination. This may also be done prior to initial placement of the cask in the decon area.

18. Prior to the cask being lifted into the fuel pool, the water level in the pool should be adjusted as necessary to accommodate the cask/DSC volume. If the water placed in the DSC cavity was obtained from the fuel pool, a level adjustment may not be necessary.

M.8.1.2 DSC Fuel Loading

1. Lift the cask/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 location of the fuel pool designated as the cask loading area.

4. Disengage the lifting yoke from the cask lifting trunnions and move the yoke clear of the cask. Spray the lifting yoke with clean demineralized water as 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 *failed/damaged and/ or intact* fuel assemblies, Control Components (CCs) and Poison Rod Assemblies (PRAs), if required, are placed into a known cell location within a DSC, will typically consist of the following:

   - A cask/DSC loading plan is developed to verify that the *failed/damaged and/ or intact* fuel assemblies, and CCs, if applicable, meet the burnup, enrichment and cooling time parameters of Technical Specification 2.1. If PRAs are determined to be needed by Technical Specification 2.1, record the number required and the DSC cell location for each of the PRAs on the loading plan
   - The loading plan is independently verified and approved before the fuel load.

All indicated changes are in response to RAI 8-2

- 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 of the fuel movement schedule.
- *If loading damaged fuel assemblies, verify that the required number of bottom end caps are installed in appropriate fuel compartment tube locations.*
- *If loading failed fuel, verify that the required number of failed fuel cans are installed in the appropriate locations, or, once loaded with fuel, are installed in the appropriate locations in the basket.*

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

7. Position the fuel assembly for insertion into the selected DSC storage cell and load the fuel assembly. Repeat Step 6 for each SFA loaded into the DSC. *Damaged fuel assemblies or failed fuel assemblies may be loaded into the basket per Technical Specifications.* If applicable, insert the required number of PRAs at specific locations called out in the loading plan. After the DSC has been fully loaded, check and record the identity and location of each fuel assembly and BPRAs, if applicable, in the DSC. Also record the location of each PRA inserted in the DSC (if applicable). *If loading damaged fuel assemblies, place top end caps over each damaged fuel assembly placed into the basket. If loading failed fuel, ensure that the failed fuel can lids are installed.*

8. After all the SFAs, BPRAs, and PRAs, if applicable, have been placed into the DSC and their identities verified, position the lifting yoke with the rigging cables connected to the top shield plug, adjust the rigging cables as needed to level the top shield plug and lower the shield plug onto the DSC.

   **CAUTION:** Verify that all the lifting height restrictions as a function of temperature specified in Technical Specification 5.3.1.A can be met in the following steps which involve lifting of the TC.

9. Visually verify that the top shield plug is properly seated onto the DSC.

10. Position the lifting yoke with the TC trunnions and verify that it is properly engaged.

11. Raise the TC to the pool surface. Prior to raising the top of the cask above the water surface, stop vertical movement.

12. Inspect the top shield plug to verify that it is properly seated onto the DSC. If not, lower the cask and reposition the top shield plug. Repeat Steps 11 and 12 as necessary.

13. Continue to raise the TC from the pool and spray the exposed portion of the cask with demineralized water until the top region of the cask is accessible.

14. Drain any excess water from the top of the DSC shield plug back to the fuel pool.

15. Check the radiation levels at the center of the top shield plug and around the perimeter of the cask.

16. If loading 32PT-S100 or 32PT-L100 DSC (qualified for 100-ton crane capacity), drain approximately 750 gallons of water (as indicated on the flowmeter) from the DSC back into the fuel pool or other suitable location using the VDS or optional liquid pump. Consistent with ISG-22 [8.7] guidance and Technical Specification 3.1.1, helium at 1-3 psig is used to backfill the DSC with an inert gas as water is being removed from the DSC.

All indicated changes are in response to RAI 8-2

38. Disengage the lifting yoke from the cask and lift the top shield plug from the DSC.

39. *If the DSC contains damaged or failed fuel, remove the top end cap or failed fuel can lid. Then remove the fuel or rod storage basket from the DSC and place the fuel or rod storage basket into the racks.* Remove the fuel from the DSC and place the fuel into the spent fuel racks.

40. Lower the top shield plug onto the DSC.

41. Visually verify that the top shield plug is properly positioned onto the DSC.

42. Engage the lifting yoke onto the cask trunnions.

43. Visually verify that the yoke lifting hooks are properly engaged with the cask trunnions.

44. Lift the cask by a small amount and verify that the lifting hooks are properly engaged with the trunnions.

45. Lift the cask to the pool surface. Prior to raising the top of the cask to the water surface, stop vertical movement and inspect the top shield plug to ensure that it is properly positioned.

46. Spray the exposed portion of the cask with demineralized water.

47. Visually inspect the top shield plug of the DSC to insure that it is properly seated onto the cask. If the top shield plug is not properly seated, lower the cask back to the fuel pool and reposition the plug.

48. Drain any excess water from the top of the top shield plug into the fuel pool.

49. Lift the cask from the pool. As the cask is rising out of the pool, spray the cask with demineralized water.

50. Move the cask to the cask decon area.

51. Check radiation levels around the perimeter of the cask. The cask exterior surface should be decontaminated if necessary.

52. Place scaffolding around the cask so that any point along the surface of the cask is easily accessible to personnel.

53. Ready the DSC vacuum drying system (VDS).

54. Connect the VDS to the vent port with the system open to atmosphere. Also connect the VDS to the siphon port and connect the other end of the system to the liquid pump. The pump discharge should be routed to the plant radwaste system or the spent fuel pool.

55. Open the valves on the vent port and siphon port of the VDS.

56. Activate the liquid pump.

57. Once the water stops flowing out of the DSC, deactivate the pump.

All indicated changes are in response to RAI 6-5

# M.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.

## M.10.1 Occupational Exposure

*The occupational exposure results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section M.5.4.16. Loading 0.380 MTU fuel results in an increase in occupational exposure of 25%.*

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. The total exposure for the occupational dose due to placing a single NUHOMS®-32PT DSC into storage is conservatively estimated to be 1.8 person-rem (32PT-S125/32PT-L125 DSC configuration) and 3.8 person-rem (32PT-S100/32PT-L100 DSC configuration). This is a very conservative estimate because the dose rates on and around the 32PT DSC used in these calculations are based on very conservative assumptions for the design-basis source terms (32PT-S100/32PT-L100 DSC with Heat Loading Zoning Configuration 2 from Chapter M.2). As in Section M.5, no credit is taken for the thicker door described in Section 8.1.1.6 or for any steel liners around the vent openings for the occupational exposure analysis. The calculated exposures for both configurations are due mainly to the expected gamma dose rate during preparation for welding. The increased calculated exposure for the 32PT-S100/32PT- L100 configuration is due to the thinner shield plug and due to draining the NUHOMS®-32PT DSC to meet a 32PT-S100/32PT-L100 DSC weight limit as described in Section M.8.

The NUHOMS®-32PT System loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table M.10-1 and Table M.10-2 for the 32PT-S125/32PT-L125 DSC and 32PT-S100/32PT-L100 DSC configurations respectively. 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 one design-basis NUHOMS®-32PT DSC in an HSM. The dose rates applicable for each operation are based on the results presented in Section M.5.4 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. 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 is sometimes far 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,

All indicated changes are in response to RAI 6-5

# Table M.10-1
## Occupational Exposure Summary
## (32PT-S125/32PT-L125 DSC configuration)

| Task | Number of Workers | Completion Time (hours) | Dose Rate (mrem/hr) | Exposure (mrem) |
|---|---|---|---|---|
| **Location: Auxiliary Building and Fuel Pool** | | | | |
| Ready the DSC and Transfer Cask for Service(1) | 2 | 4 | 0 | - |
| Place the DSC into the Transfer Cask(1) | 3 | 1 | 2 | 6 |
| Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal | 2 | 2 | 2 | 8 |
| Fill the DSC Cavity with Water(2) | 1 | 6 | 2 | 1 |
| Install Lifting Yoke and Connect VDS | 2 | 0.5 | 2 | 2 |
| Place the Cask Containing the DSC in the Fuel Pool | 5 | 0.5 | 2 | 5 |
| Verify and Load the Candidate Fuel Assemblies into the DSC | 3 | 8 | 2 | 48 |
| Place the Top Shield Plug on the DSC | 2 | 1 | 2 | 4 |
| Raise the Cask/DSC to the Fuel Pool Surface | 5 | 0.5 | 2 | 5 |
| Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area | 2 | 0.5 | 20 | 20 |
| **Location: Cask Decon Area** | | | | |
| Decontaminate the Outer Surface of the Cask (on the hook)(3) | 3 | 1 | varies | 65 |
| Cask Decontamination (in the decon area)(3) | 3 | 1 | varies | 110 |
| Remove the Cask/DSC Annulus Seal and Set-up Welder(3) | 2 | 1.5 | varies | 88 |
| Drain the DSC Cavity(3) | 2 | 0.5 | varies | 33 |
| Weld the Inner Top Cover to the DSC Shell and Perform NDE(3) | 2 | 6 | varies | 71 |
| Vacuum Dry and Backfill the DSC with Helium(3) | 2 | 16 | varies | 33 |
| Helium Leak Test the Shield Plug Weld | 2 | 1 | 2 | 4 |
| Seal Weld the Prefabricated Plugs to the Vent and Siphon Port and Perform NDE | 1 | 1 | 241 | 241 |
| Drain Cask/DSC Annulus(3) | 1 | 0.25 | varies | 23 |
| Install DSC Outer Top Cover Plate(3) | 2 | 1 | varies | 209 |
| Weld the Outer Top Cover Plate to DSC Shell and Perform NDE(3) | 2 | 16 | varies | 112 |
| Install the Cask Lid | 2 | 1 | 15 | 30 |
| **Location: Reactor /Fuel Building Bay** | | | | |
| Ready the Cask Support Skid and Transfer Trailer for Service(1) | 2 | 2 | 0 | - |
| Place the Cask Onto the Skid and Secure(2) | 3 | 0.5 | 225 | 225 |
| **Location: ISFSI Site** | | | | |
| Ready the HSM and Hydraulic Ram System for Service(1) | 2 | 2 | 0 | - |
| Transfer the Cask to the ISFSI(1) | 6 | 1 | 0 | - |
| Position the Cask in Close Proximity with the HSM(1) | 3 | 1 | 0 | - |
| Remove the Cask Lid | 3 | 1 | 25 | 75 |
| Align and Dock the Cask with the HSM | 2 | 0.25 | 225 | 113 |
| Position and Align Ram with Cask(3) | 2 | 0.5 | varies | 28 |
| Remove the RAM Access Cover Plate | 2 | 0.25 | 121 | 61 |
| Transfer the DSC from the Cask to the HSM(1) | 3 | 0.5 | 0 | - |
| Un-Dock the Cask from the HSM | 2 | 0.083 | 752 | 125 |
| Install the HSM Access Door | 2 | 0.5 | 63 | 63 |
| **Total** | | **80** | | **1,808** |

**Total estimated dose is 1.8 person-rem per canister load**

**Total estimated completion time is 80 hrs**

(1) Performed away from any significant radiation sources.
(2) Personnel are not present throughout this activity.
~~(3) Dose rates and locations vary during this task.~~
*Total estimated dose increases by 25% when loading 0.380 MTU/FA fuel.*

All indicated changes are in response to RAI 6-5

## Table M.10-2
## Occupational Exposure Summary
## (32PT-S100/32PT-L100 DSC configuration)

| Task | Number of Workers | Completion Time (hours) | Dose Rate (mrem/hr) | Exposure (mrem) |
|---|---|---|---|---|
| **Location: Auxiliary Building and Fuel Pool** | | | | |
| Ready the DSC and Transfer Cask for Service(1) | 2 | 4 | 0 | - |
| Place the DSC into the Transfer Cask(1) | 3 | 1 | 2 | 6 |
| Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal | 2 | 2 | 2 | 8 |
| Fill the DSC Cavity with Water(2) | 1 | 6 | 2 | 1 |
| Install Shield Plug and Connect VDS | 2 | 0.5 | 2 | 2 |
| Place the Cask Containing the DSC in the Fuel Pool | 5 | 0.5 | 2 | 5 |
| Verify and Load the Candidate Fuel Assemblies into the DSC | 3 | 8 | 2 | 48 |
| Place the Top Shield Plug on the DSC | 2 | 1 | 2 | 4 |
| Raise the Cask/DSC to the Fuel Pool Surface | 5 | 0.5 | 2 | 5 |
| Drain Water from DSC Cavity | 1 | 1 | 100 | 100 |
| Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area | 2 | 0.5 | 20 | 20 |
| **Location: Cask Decon Area** | | | | |
| Decontaminate the Outer Surface of the Cask (on the hook)(3) | 3 | 1 | varies | 680 |
| Fill Cask Neutron Shield and DSC Cavity | 1 | 0.1 | 57 | 6 |
| Cask Decontamination (in the decon area)(3) | 3 | 1 | varies | 112 |
| Remove the Cask/DSC Annulus Seal and Set-up Welder(3) | 2 | 1.5 | varies | 191 |
| Drain the DSC Cavity(3) | 2 | 0.5 | varies | 84 |
| Weld the Inner Top Cover to the DSC Shell and Perform NDE(3) | 2 | 6 | varies | 139 |
| Vacuum Dry and Backfill the DSC with Helium(3) | 2 | 16 | varies | 84 |
| Helium Leak Test the Shield Plug Weld | 2 | 1 | 2 | 4 |
| Seal Weld the Prefabricated Plugs to the Vent and Siphon Port and Perform NDE | 1 | 1 | 650 | 650 |
| Drain Cask/DSC Annulus(3) | 1 | 0.25 | varies | 58 |
| Install DSC Outer Top Cover Plate(3) | 2 | 1 | varies | 527 |
| Weld the Outer Top Cover Plate to DSC Shell and Perform NDE(3) | 2 | 16 | varies | 214 |
| Install the Cask Lid | 2 | 1 | 25 | 50 |
| **Location: Reactor /Fuel Building Bay** | | | | |
| Ready the Cask Support Skid and Transfer Trailer for Service(1) | 2 | 2 | 0 | - |
| Place the Cask Onto the Skid and Secure(2) | 3 | 0.5 | 226 | 226 |
| **Location: ISFSI Site** | | | | |
| Ready the HSM and Hydraulic Ram System for Service(1) | 2 | 2 | 0 | - |
| Transfer the Cask to the ISFSI(1) | 6 | 1 | 0 | - |
| Position the Cask in Close Proximity with the HSM(1) | 3 | 1 | 0 | - |
| Remove the Cask Lid | 3 | 1 | 27 | 81 |
| Align and Dock the Cask with the HSM | 2 | 0.25 | 226 | 113 |
| Position and Align Ram with Cask(3) | 2 | 0.5 | varies | 50 |
| Remove the RAM Access Cover Plate | 2 | 0.25 | 274 | 137 |
| Transfer the DSC from the Cask to the HSM(1) | 3 | 0.5 | 0 | - |
| Un-Dock the Cask from the HSM | 2 | 0.083 | 788 | 131 |
| Install the HSM Access Door | 2 | 0.5 | 96 | 96 |
| **Total** | | **81** | | **3,831** |

**Total estimated dose is 3.8 person-rem per canister load**

**Total estimated completion time is 80 hrs**

(1) Performed away from any significant radiation sources.
(2) Personnel are not present throughout this activity.
(3) Dose rates and locations vary during this task.
*Total estimated dose increases by 25% when loading 0.380 MTU/FA fuel.*

All indicated changes are in response to RAI 6-5

## N.2.1 Spent Fuel to be Stored

There are two design configurations for the NUHOMS®-24PHB DSC: the 24PHBS and 24PHBL, which are nearly identical to the standard and long cavity 24P DSCs, respectively. Each of the DSC configurations is designed to store 24 intact PWR fuel assemblies, including reconstituted assemblies or up to 4 damaged and balance intact PWR fuel assemblies (including reconstituted) with characteristics described in Table N.2-1. The 24PHB DSC is designed to store intact or damaged B&W 15x15, intact WE 17x17, intact WE 15x15, intact CE 14x14, and intact WE 14x14 Class PWR fuel assemblies as specified in Table N.2.1. Control Components (CCs) and damaged fuel assemblies are allowed only in the B&W 15x15 class fuel assembly. Replacement assemblies by other manufacturers are also allowed provided they meet limiting features listed in Table N.2-1.

*Damaged PWR 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 retrievability is ensured following 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.*

The NUHOMS®-24PHB DSC may store PWR fuel assemblies arranged in one of two alternate Heat Load Zoning Configurations with a maximum decay heat of 1.3 kW per assembly and a maximum heat load of 24 kW per DSC. The Heat Load Zoning Configurations are shown in Figure N.2-1 and Figure N.2-2. The NUHOMS®-24PHB DSC is vacuum dried and backfilled with helium at the time of loading. The maximum (bounding) fuel assembly weight of 1682 lbs with a CC is identical to the NUHOMS®-24P DSC design.

The maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the 24PHB DSC per the guidance provided in NUREG-1536 [2.1]. In addition, NUREG-1536 does not permit thermal cycling of the fuel cladding with 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 thermal transients [2.1].

The information provided in Table N.2-1 is based on the design basis B&W 15x15 fuel which is the bounding fuel assembly. The types of spent fuel considered in Appendix N include the following:

- B&W 15x15 Mark B2, B3, B4, B4Z, BZ, B5, B5Z, B6, B7, B8, B9, B10, B10D, B10E, B10F, B10G, B10L, B11, and B11A fuel assemblies, with or without CCs.
- B&W 15x15 reconstituted fuel assemblies with a maximum of 10 stainless steel rods per assembly or unlimited number of lower enrichment $UO_2$ rods instead of zirconium-alloy clad enriched $UO_2$ rods. The stainless steel rods are assumed to have two thirds the irradiation time as the zirconium-alloy 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

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

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

*The above discussion on the applicability of SCALE 4.4/SAS2H to compute gamma and neutron high-burnup source terms is also largely applicable to SCALE 6.0/ORIGEN-ARP because both code systems utilize ORIGEN-S for the depletion calculation. For PWR fuel assemblies, SAS2H and ORIGEN-ARP generate comparable source terms for equivalent program inputs. The TRITON T-DEPL module of SCALE 6.0 is used to generate ORIGEN-ARP libraries applicable to B&W 15x15 fuel assemblies. These libraries are then used by ORIGEN-ARP to compute gamma and neutron source terms.*

*Oak Ridge National Laboratory has benchmarked TRITON based on measured data from six different PWRs. This benchmarking is documented in NUREG/CR-6968 [5.16], NUREG/CR-7012 [5.17], and NUREG/CR-7013 [5.18] and includes measurement samples up to a burnup of 78.3 GWd/MTU. A summary of experimental samples including the number of samples for each isotope utilized in the benchmark analysis is provided in Table P.5-27. The benchmark references show that TRITON computed results agree well with experiments, thus verifying the use of SCALE 6.0/ORIGEN-ARP to compute gamma and neutron source terms for high-burnup fuel (burnup ≤ 62 GWd/MTU).*

*Further, the significant documentation on SCALE TRITON/ORIGEN-ARP benchmark, for example NUREG/CR-6969, NUREG/CR-7162 and ORNL paper ("Analysis of isotopic assay data from the MALIBU program," ORNL paper for the International Conference on Reactor Physics, Switzerland, 2009) indicates that TRITON/ORIGEN-ARP and associated cross-section library are appropriate for decay heat and source term calculations.*

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

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

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

A bounding analysis is performed by employing a minimum concrete density of 140 pounds per cubic foot (pcf) in the HSM-H MCNP model combined with a maximum gap of 1.5 inches between adjacent HSM-H modules and shield walls to determine the effect on maximum and average dose rates due to a fully loaded 32PTH1 DSC. These calculations are documented in Appendix U.5, Section U.5.4.10. The ratios shown in Appendix U.5, Table U.5-18 and Table U.5-19 can be used as scaling factors to increase the maximum and surface-average dose rates of the 24PTH in the HSM-H to account for low density concrete and 1.5-inch gaps during HSM fabrication and installation. Note that the HSM-H concrete contains high density rebar which is not credited in the MCNP models. Further, the modules are installed adjacent to each other such that there will not be a "uniform" gap of 1.5 inches. Ignoring the effect due to increased vent dose rates, the increase in the average dose rates caused by both the maximum postulated uniform gaps and the minimum postulated concrete density is expected to be less than 20% at the front and roof surfaces of the HSM-H module. Dose reduction hardware may be installed to further reduce these dose rates.

P.5.4.11 *Shielding Analysis with a Loading of 0.380 MTU per Fuel Assembly*

*As discussed in Section P.5.4, additional shielding analysis is performed with a reduced Uranium loading of 0.380 MTU per fuel assembly. The objective of this analysis is to determine the impact that reduced uranium loading has on system dose rates. The results of this analysis are employed to scale the dose rate results for the 24PTH System (all DSCs). For this purpose, the MCNP4C2 models used for the 0.490 MTU analyses are rerun using MCNP5 with updated source terms as described in P.5.2.6, and with updated material specifications to reflect the reduction in MTU. MCNP5 calculations are performed for the 24PTH-L DSC inside the HSM-H in the normal storage configuration, and dose rate scaling factors are derived using the same methodology as that described in Appendix U, Section U.5.4.12. MCNP5 calculations are also performed for the 24PTH-L DSC inside the OS197 TC in the decontamination and welding configurations, and in the normal and accident transfer configurations, and dose rate and occupational exposure scaling factors are derived using the same methodology as that described in Appendix U, Section U.5.4.12. These results are also applicable to the 24PTH-S and 24PTH-S-LC DSCs. Based on the updated results, six scaling factors are determined and are summarized as follows:*

All indicated changes are in response to RAI 6-4

**Table P.5-27**
***Summary of Experimental Samples as a Function of Burnup Range***

| | | Low Burnup (B < 45 GWd/MTU) | | High Burnup (B > 45 GWd/MTU) | |
|---|---|---|---|---|---|
| **Power Plant** | **Reference** | **No of Samples** | **Range (GWd/MTU)** | **No of Samples** | **Range (GWd/MTU)** |
| Takahama-3 | NUREG/CR-6968 Reference [5.16] | 14 | 8.55 – 42.16 | 2 | 47.03 – 47.25 |
| Three Mile Island - 1 | | 10 | 22.80 – 44.8 | 9 | 50.10 – 55.70 |
| Calvert Cliffs | | 3 | 27.35 – 44.34 | - | - |
| Vandellós II | NUREG/CR-7013 Reference [5.18] | 1 | 42.50 | 5 | 54.85 – 78.30 |
| Gösgen | NUREG/CR-7012 Reference [5.17] | 1 | 31.10 | 5 | 46.00 –70.30 |
| GKN II | | - | - | 1 | 54.10 |
| Total | | 29 | - | 22 | - |

***Number of Samples Used for Each Isotope***

| | Cm-242 | Cm-244 | Cs-134 | Cs-137 | Ce-144 | Eu-154 | Ru-106 | Sm-147 | Sr-90 |
|---|---|---|---|---|---|---|---|---|---|
| **Gosgen** | 4 | 6 | 6 | 6 | 6 | 6 | 6 | 6 | 6 |
| **GKN II** | 1 | 1 | - | - | 1 | 1 | - | 1 | - |
| **Takahama 3** | 16 | 16 | 16 | 16 | 16 | 16 | 16 | 6 | - |
| **Vandellos II** | - | 6 | 6 | 6 | 5 | 5 | 5 | 5 | - |
| **TMI** | 8 | 8 | 8 | 19 | - | - | - | 19 | - |
| **Calvert Cliffs** | - | - | 3 | 3 | - | 3 | - | 3 | 3 |

$\cos(24.65+\alpha- 90) = 0.00433 +0.907411 = 0.911741$

$90-\alpha = 24.85-24.25 = 0.60$

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 P.11-1) to an angle of more than 24.65° [$= \tan^{-1}(52.0/118.77)$].

P.11.2.3.3 Accident Dose Calculations

The NUHOMS®-24PTH DSC is designed and tested as a leak-tight containment boundary according to the criteria of ANSI N14.5. As shown in Section P.11.1.1.1, the tornado wind and tornado missiles do not breach the containment boundary. Therefore, there is no increase in site boundary dose *as a result of leakage from the DSC* due to this accident event. *However, a small increase in the site dose rate may occur due to damage to the HSM-H.*

*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 1400 mrem/hour in Table P.5-1 (1237 mrem/hr*1.13 ~ 1400 mrem/hr), 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 greater than 20 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 P.10.2 for a 2x10 array of undamaged HSMs (specifically Table P.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*Model 102 dose rate at 100m, 1.09E-01 mrem/hour*1.13 scaling factor) with a 2x10 array of HSMs. As an additional conservatism, Model 102 dose rates are used because these dose rates bound HSM-H dose rates. The dose to an offsite person located 500 meters away for the assumed 8-hour duration would be less than 0.02 mrem (2*8 hours*Model 102 dose rate at 500m, 7.43E-04 mrem/hour*1.13 scaling factor) with a 2x10 array of HSMs.*

P.11.2.4 Flood

P.11.2.4.1 Cause of Accident

No change to Section 8.2.4.1.

P.11.2.4.2 Accident Analysis

No change to the HSM Model 102 analysis presented in Section 8.2.4.2.

The HSM-H and DSCs are evaluated for flooding in Section P.3.7.3. The DSC is designed and tested to be leak tight to the criteria of ANSI N14.5. The stresses in the DSC due to the design basis flood are well below the allowable stresses for Service Level C of the ASME Code Subsection NB [11.5]. Therefore, the NUHOMS®-24PTH DSC will withstand the design basis flood without breach of the confinement boundary.

The evaluation of the HSM-H/HSM-HS for flooding, presented in Section U.3.7.3, is not changed.

P.11.2.4.3 Accident Dose Calculations

The radiation dose due to flooding of the HSMs (HSM-H or HSM Model 102) is negligible. The NUHOMS®-24PTH DSC is designed and tested as a leak-tight containment boundary. Flooding does not breach the containment boundary. Therefore radioactive material inside the DSC will

All indicated changes are in response to RAI 8-3

## T.1.1 Introduction

The NUHOMS®-61BTH *System* is designed to store up to 61 intact (including reconstituted) or up to 16 damaged with up to 4 *failed fuel cans* (FFCs) loaded with failed fuel with the remainder intact BWR fuel assemblies with or without fuel channels. Alternatively, 61 damaged fuels can be stored in the NUHOMS®-61BTH DSC as shown in Figure T.2-9. 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. The design characteristics, including physical and radiological parameters of the payload, are described in Appendix T.2.

Reconstituted assemblies containing up to 10 replacement stainless steel rods per assembly or up to 61 lower enrichment $UO_2$ rods instead of Zircaloy clad enriched $UO_2$ rods are acceptable for storage in 61BTH DSC as intact fuel assemblies with a slightly longer cooling time than that required for a standard assembly. The maximum number of reconstituted fuel assemblies per DSC is four with stainless steel rods or 61 with $UO_2$ rods.

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 Transfer Cask (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

by changing the Units 15, 17, 18, 19, 21, 22, 24, 25, 27, 29, 31, 50, 53, 55 and 57 such that the 0.300″ poison material is modeled as 0.125″ borated aluminum poison and the remaining as aluminum. The KENO Unit 25 is split into Units 125 and 225 so that the poison and aluminum are modeled separately. Similar treatment is accorded to Units 27, 29 and 31. This parametric evaluation scoping is also extended to include Boral® material as the poison. The Boral® material is modeled based on the specification shown in Table T.6-1 for a B-10 loading of 36 $mg/cm^2$ and a thickness of 0.064 inch. The results of this evaluation demonstrate that the effect of modeling the paired plates is statistically insignificant. The results also show that the effect of the various poison plate material specifications is also statistically insignificant as long as the amount of absorber material present in the model does not change. The statistically insignificant results arising due to the variation in the thickness of the poison plates (0.300-inch, 0.125-inch, and 0.064-inch) and hence, the thickness of the aluminum plates indicates that there is no reactivity effect due to the modeling of cladding materials for the poison (like for Boral®). These results are shown in Table T.6-7.

The KENO model implementation of the paired plates has been effected in three ways in the same input model. For Units 15, 17, 18, 19, 21, 22 and 24 the poison in the paired plates is surrounded by aluminum. For Units 25, 27, 29 and 31 the poison and the aluminum are modeled as two distinct plates. For Units 50, 53, 55 and 57 the aluminum in the paired plates is surrounded by poison. Though the geometry in the actual basket for the paired plates is expected to be similar to what is modeled in Unit 25, this representation provides further insight to the results shown in Table T.6-7 and the conclusions herein with regard to variation in the poison plate and aluminum plate thicknesses. Therefore, treatment of paired plates does not result in any significant change in the system reactivity.

The ninth set of calculations determines the effect of basket rail modeling on the system reactivity. In order to obtain an acceptable, yet conservative model for both Type 1 and Type 2 DSC basket designs, the peripheral rails were modeled with solid aluminum and 7.5 cm (approximately 5.9 inches diameter, about 15% less than the actual water volume fraction at those locations) water holes in the eight corner positions. The configuration is similar to that shown in Appendix K, Table K.6-4. The water holes (circular cross section) were also modeled using water squares to determine the effect due to the "hole" geometry. The results of this evaluation indicate that the assumption of solid aluminum rails with water holes conservatively bounds the internal moderator rails for the Type 1 DSC basket. It is also clear from the results that the use of solid aluminum as a rail material alone results in an overly conservative model. These results are shown in Table T.6-7. The water area calculations are shown in Section T.6.6.2.

Finally, minimum boron loading in the poison plate as a function of lattice average initial enrichment is evaluated. These models represent the most reactive intact fuel assembly (GE12, 10x10) with a minimum assembly-to-assembly pitch, nominal shell thickness, minimum paired plate thickness, minimum fuel clad OD, minimum fuel cell width with full internal and optimum external moderator density. Moreover, the calculational criticality analysis KENO model is also based on internal moderator gaps, paired plates with minimum poison thickness, solid aluminum rails with water holes that bounds both the Type 1 and Type 2 DSC basket designs.

*Note that the criticality results for GNF2 and ATRIUM 11 fuel assemblies are obtained using the subject fuel assemblies instead of a representative model. The criticality analysis for the GNF2*

All indicated changes are in response to RAI 7-7

*fuel assembly is performed as a sensitivity to the GE12 10x10 fuel assembly on the basis of three DSC basket types – Types 1 and 2 with A, C, and F poison loadings. The results are provided in Table T.6-26 and Table T.6-27.*

*The analyses performed for the three DSC basket type poison loadings show that the maximum enrichments determined for the GE12 10x10 intact fuels are applicable to the GNF2 fuel assembly. These analyses demonstrate that the GNF2 fuel is similar in reactivity to the GE12 10x10 fuel; hence, the maximum enrichments determined for the GE12 10x10 fuel for the DSC basket type poison loadings B, D, and E are also applicable to the GNF2 fuel.*

*For the ATRIUM 11 fuel assembly, criticality evaluations are performed using the Type 2F basket, and the results indicate that the maximum allowable enrichment in is reduced by 0.55 wt. % U-235 from that allowed for the GE12 fuel assembly. The results are provided in Table T.6-28.*

The *B-10* areal density in the poison plate (and hence thickness of the poison plate) is varied to determine the maximum lattice average fuel assembly enrichment. Thus, these cases can be

One important difference between the design basis intact assembly models and the damaged assembly models is in the treatment of axial boundary conditions. In the intact assembly models, the axial boundary conditions are reflective and therefore, the fuel assembly length was essentially infinite axially (conservative modeling). In the case of the damaged assembly models, a fuel assembly active length of 144″ was utilized with an additional 12.5″ of shifted damaged row of rods and water boundary conditions axially. The active fuel length of the GE 10x10 assembly is 150″ and even though it is expected to contain at least 6″ length of blankets, it is necessary to evaluate the effect of an increase in the active fuel length, if any, on the system reactivity.

Case ID 25 was modified to create the Case ID 26 where the active fuel length was increased to 150″. These results are also shown in Table T.6-11 and indicate that there is no significant change in the system reactivity due to the modeling of shorter active fuel length.

Case ID 23 was modified to create Case ID 27 where the outermost damaged rows of rods was modeled without cladding (clad replaced with internal moderator). This configuration is expected to result in an increase in $k_{eff}$ because of the fact that fuel assemblies, in general, are undermoderated. Therefore, increasing the moderation by replacing the clad with moderator will result in an increase in $k_{eff}$. Variations to the above case included modeling the two outermost rows of rods with bare fuel, Case ID 28. The results of these cases are also shown in Table T.6-11. The most reactive configuration for the double shear cases is based on the Case ID 28, as expected, since it contains more bare rods.

The optimum rod pitch case, Case ID 29, was based on a modification to Case ID 23 except that there was no "UP" modeling included in the model. The "Dancoff" factor to be used to describe the damaged lattice is obtained from the rod array cases documented in Table T.6-10. A comparison of the $k_{eff}$ obtained from this case, also shown in Table T.6-11, with the design basis double shear case from the previous evaluation (Case ID 28) clearly shows that the worst case damaged assembly configuration is based on the optimum rod pitch model.

Finally, minimum boron loading in the poison plate as a function of lattice average initial enrichment is evaluated. These models represent the DSC with the most reactive damaged fuel assemblies (GE12, 10x10, optimum pitch, 95 rods) for both the 4 and 16 assembly loading configurations with the most reactive configuration determined in the previous analyses. The remaining locations are loaded with the most reactive intact fuel assembly (GE12, 10x10) with the most reactive configuration determined in Section T.6.4.2B. The calculational criticality analysis KENO model is also based on internal moderator gaps, paired plates with minimum poison thickness, solid aluminum rails with water holes that bounds both the Type 1 and Type 2 DSC basket designs. Above all, all the damaged assembly criticality models for the paired plates are based on the "correct" arrangement (as described at the end of Section T.6.4.2B) of these plates. All the damaged assembly calculations are carried out with the borated aluminum poison.

*Note that the criticality results for GNF2 and ATRIUM 11 fuel assemblies are obtained using the subject fuel assemblies instead of a representative model. The criticality analysis for the GNF2 fuel assembly is performed as a sensitivity to the GE12 10x10 fuel assembly on the basis of three DSC basket types – Types 1 and 2 with A, C, and F poison loadings. The results for four*

*damaged fuel assemblies are provided in Table T.6-29 and Table T.6-30, while for those with 16 damaged fuel assemblies are provided in Table T.6-32 and Table T.6-33.*

*The analyses performed for the three DSC basket type poison loadings show that the maximum enrichments determined for the GE12 10x10 damaged fuel are applicable to the GNF2 fuel assembly. These analyses demonstrate that the GNF2 fuel is similar to the GE12 10x10 fuel in reactivity; hence, the maximum enrichments determined for the GE12 10x10 fuel for the DSC basket type poison loadings B, D, and E are also applicable to the GNF2 fuel.*

*For the ATRIUM 11 fuel assembly, only up to four damaged fuel assemblies are authorized for storage with the remaining intact. The criticality evaluations are performed using the Type 2F basket and the results indicate that the maximum allowable enrichment is reduced by 0.55 wt. % U-235 from that allowed for the GE12 fuel assembly. The results are provided in Table T.6-31.*

A sensitivity calculation was performed to determine the effect of the specification of the second lattice or the "Dancoff" factor in the criticality analyses. This is due to the fact that both the intact and damaged fuel assemblies can be specified as the second lattice. The "Dancoff" factor for the intact fuel assemblies is 2.6461172E-01 while that for the damaged fuel assemblies is 1.4377643E-01.

may slide into and increase the reactivity of the system as this region does not have basket poison plates. The length of 16 inches is conservatively chosen to bound the length of top grid axial length, which for current designs is maximum 14.5 inches (Appendix T.1). The most reactive configuration search includes the variation of assembly position inside the compartment and of the material filling around the failed fuel compartments in the 16 inches axial region outside the basket. This 16-inch axial region variation is evaluated using water, aluminum, or an approximate model of the top grid is used in order to cover a wide range of design variations that might affect the top grid region. The input models for configurations with 4 failed fuel assemblies in Table T.6-14 are constructed by changing the model of the design basis input for loading configuration with 4 damaged assemblies to represent the failed fuel loaded in the designated peripheral compartments in basket. Similarly, the input models for configurations with 4 failed fuel and 12 damaged fuel assemblies in Table T.6-19 are constructed by changing the model of the design basis input for loading configuration with 16 damaged assemblies to represent the failed fuel loaded in the designated peripheral compartments in basket. The KENO model plot of the 4 failed/57 intact assembly configuration is shown in Figure T.6-7. The input file listing for this case is provided in Section T.6.6.6. The KENO model plot of the 4 failed/12 damaged/45 intact assembly configuration is shown in Figure T.6-8. The input file listing for this case is also provided in Section T.6.6.6.

Finally, minimum boron loading in the poison plate as a function of lattice average initial enrichment is evaluated. These models represent the DSC with the most reactive failed fuel compartment configuration (GE, 10x10, optimum pitch, de-cladded rods, no rods removed, as shown in Table T.6-18), most reactive damaged fuel assemblies (GE, 10x10, optimum pitch, 95 rods, as shown in, Table T.6-10) in both loading configurations investigated, with 4 failed fuel compartments, and with 4 failed fuel compartments and 12 damaged assemblies. The remaining locations are loaded with the most reactive intact fuel assembly (GE12, 10x10) with a minimum assembly-to-assembly pitch, nominal shell thickness, minimum fuel clad OD, minimum poison thickness, minimum fuel cell width with full internal and optimum external moderator density. The calculational criticality analysis KENO model is also based on internal moderator gaps and solid aluminum rails with water holes that bounds Type 2 basket design. All the failed fuel calculations are carried out with the borated aluminum poison.

*Note that the criticality results for GNF2 fuel assemblies are obtained using the subject fuel assemblies instead of a representative model. The criticality analysis for the GNF2 fuel assembly is performed as a sensitivity to the GE12 10x10 fuel assembly on the basis of three DSC basket types – Types 1 and 2 with A, C, and F poison loadings. The results are provided in Tables T.6-34 and T.6-35.*

*The analyses performed for the three DSC basket type poison loadings show that the maximum enrichments determined for the GE12 10x10 failed fuels are applicable to the GNF2 fuel assembly. These analyses demonstrate that the GNF2 fuel is similar to the GE12 10x10 fuel in reactivity; hence, the maximum enrichments determined for the GE12 10x10 fuel for the DSC basket type poison loadings B, D, and E are also applicable to the GNF2 fuel.*

The *B*-10 areal density in the poison plate is varied to determine the maximum lattice average fuel assembly enrichment. Thus, these cases can be used to specify the minimum *B*-10 poison plate loading (and the appropriate thickness) as a function of maximum lattice average assembly enrichment. The results are reported in Tables T.6-20 and T.6-21.

The dry case calculations are performed for the most reactive initial enrichment/poison plate loading combination. The case selected for performing the dry calculations is based on the most reactive fully flooded case (100% internal and external moderator density). For the dry cases,

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 600 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 greater than 20 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.10-17) 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.*

All indicated changes are in response to RAI 7-6

*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*Model 80 dose rate at 100m, 1.06E-01 mrem/hour) with a 2x10 array of HSMs. As an additional conservatism, Model 80 dose rates are used because these dose rates bound HSM-H dose rates. 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*Model 80 dose rate at 500m, 5.25E-04 mrem/hour) with a 2x10 array of HSMs.*

All indicated changes are in response to RAI 6-5

### U.1.1 Introduction

The NUHOMS®-32PTH1 System is designed to store up to 32 (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, and WE 14x14 class PWR fuel assemblies. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235, a maximum assembly average burnup of 62 GWd/MTU, and a minimum cooling time of *2.0* years. Each of the 32PTH1 DSC types is designed to store up to 32 Control Components (CCs) which include burnable poison rod assemblies (BPRAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), and neutron source assemblies (NSAs). The design characteristics, including physical and radiological parameters of the payload, are described in Chapter U.2.

Reconstituted 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 or Zr rods or Zr pellets or unirradiated stainless steel rods are acceptable for storage in 32PTH1 DSC as intact fuel assemblies. The maximum number of reconstituted fuel assemblies with irradiated stainless steel rods per DSC is four.

Provisions have been made for storage of up to 16 damaged fuel assemblies in lieu of an equal number of intact assemblies in the cells located at the center of the 32PTH1 basket. Damaged PWR 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 assure retrievability.

Provisions have also been made for storage of up to *four failed fuel cans (*FFCs) in cells located at the corners of the interior 4x4 compartment cells of the 32PTH1 basket *or up to 16 FFCs* in a checkerboard pattern, as described in Chapter U.2.

The NUHOMS®-32PTH1 System consists of the following new or modified components:

- A 32PTH1 DSC, with three alternate configurations, described in detail in Section U.1.2, provides confinement, an inert environment, structural support, and criticality control for the 32 PWR fuel assemblies,
- A modified HSM-H module, described in Section U.1.2, is provided for environmental protection, shielding and heat rejection during storage, and
- OS200 or OS200FC TC for onsite transfer of the 32PTH1 DSCs.

The NUHOMS®-32PTH1 System requires the use of non-safety related auxiliary transfer equipment similar to those described in Section 1.3.2.2 (for OS200 TC) and Appendix P (for OS200FC TC) of the UFSAR. There is no change to any of the design features of the auxiliary

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Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding, and criticality evaluations for these rods are bounded by those of the regular fuel rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Reconstituted 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, or Zr rods or Zr pellets, or unirradiated stainless steel rods are acceptable for storage in 32PTH1 DSC 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 assemblies per DSC is four with irradiated stainless steel replacement rods or 32 with $UO_2$ replacement rods.

The NUHOMS®-32PTH1 DSCs can also accommodate up to a maximum of 16 damaged fuel assemblies placed in the center cells of the DSC as shown in Figure U.2-1 through Figure U.2-3. Damaged PWR 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, *including non-cladding damage*. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means and retrievability is assured ~~following 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 which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

The NUHOMS®-32PTH1F DSC, an alternate version of the NUHOMS®-32PTH1 DSC, is designed to accommodate failed fuel in up to a maximum of four failed fuel cans (*FFC*s) placed in the corner cells of the interior 4x4 compartment cells of the basket, as shown in Figure U.2-5 or *up to 16 FFCs* in a checkerboard loading as shown in Figure U.2-3. Failed fuel is defined as fuel rods that have been removed from a fuel assembly and placed in a secondary container, such as a rod storage basket. Failed fuel 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. Individual fuel rods that are not failed can be stored directly in the FFC without a secondary container such as an RSB. The maximum number of fuel rods that may be stored in a failed fuel can is 100, with a total uranium loading limited to 250 kg initial uranium. The total weight of the failed fuel can plus all its contents shall be less than 1715 lbs, 1625 lbs, and 1665 lbs, for the 32PTH1-L, 32PTH1-M, and 32PTH1-S DSCs, respectively.

*transfer among the DSC shell, heat shields, and HSM-H; and heat dissipation from the HSM-H and the vent outlet via convection and radiation to the ambient.*

*Section U.4.11.1.4 presents the computer-aided design (CAD) model and the meshes for the HSM-H with the 32PTH1 Type 1 DSC basket within an external air domain in ANSYS ICEM CFD [4.46]. The meshes are imported into ANSYS FLUENT [4.45] to develop a CFD model for thermal evaluation.*

*Section U.4.11.1.5 describes the detailed setup of the CFD model in ANSYS FLUENT [4.45].*

*U.4.11.1.4* *Computer-Aided Design and Meshing*

Proprietary Information on Pages U.4-53t1 and U.4-53t2
Withheld Pursuant to 10 CFR 2.390

All indicated changes are in response to RAI 4-3

*To ensure the high quality of the mesh, the guidelines on grids and grid design from NUREG-2152 [4.49] are considered in choosing the mesh parameters and techniques.*

- *Maintain the expansion ratio between two consecutive cells below 1.3 in the regions where high gradients of temperature and velocity are expected or material changes. In solid regions, the expansion ratio is allowed to be 2 or higher.*
- *Avoid highly skewed elements with angles less than 45 degrees or larger than 135 degrees, especially in critical regions. In this mesh, around 99% of the elements have an angle between 45 and 135 degrees, and around 70% of the total elements have an angle between 81 and 99 degrees.*
- *Keep the aspect ratio of most elements less than 20 except for those in the near wall regions. In this mesh, 95% elements maintain the aspect ratio below 20, and 56% elements have the aspect ratio smaller than 5.*
- *Use finer and high-quality mesh in critical regions with high temperature and velocity gradients or with significant changes in geometry, such as regions near the DSC outer surfaces.*
- *Ensure sufficient resolution in the near-wall regions adjacent to the wall to capture the large variations in the flow. The low-Reynolds k-ε turbulence model in ANSYS FLUENT [4.45] enables the viscosity-affected region to be resolved with a mesh all the way to the wall. In the mesh model, a minimum of 15 elements is applied in the radial direction across the narrow region with 1.0" thickness around the outer surface of the DSC shell, where the first layer element is specified as 0.05" and grows with stretching factor smaller than 1.1. Furthermore, CFD results show that the average $y^+$ at the outer surface of 32PTH1 Type 1 DSC shell is 1.23. The dimensionless wall distance $y^+$ is defined as:*

$$y^+ = \frac{\rho y U_\tau}{\mu},$$

- *where $\rho$ is the fluid density, $\mu$ is the fluid viscosity, y is the element size, $U_\tau = \sqrt{\tau_w / \rho}$ is the friction velocity, and $\tau_w$ is the wall shear stress.*

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Proprietary Information on Pages U.4-53w through U.4-53x6
Withheld Pursuant to 10 CFR 2.390

All indicated changes are in response to RAI 4-3


4.48. *ANSYS Design Modeler, Version 14.0, ANSYS, Inc.*

4.49. *U.S. NRC, Office of Nuclear Material Safety and Safeguards, "Computational Fluid Dynamics Best Practice Guidelines for Dry Cask Applications-Final Report," NUREG-2152, Rev. 0, March 2013.*

4.50. *I. E. Idelchik, Handbook of Hydraulic Resistance, 3rd Edition, Begell House, Inc., 1996.*

4.51. *S. Suffield, J. Cuta, J. Fort, B. Collins, H. Adkins and E. Siciliano, "Thermal Modeling of NUHOMS HSM-15 and HSM-1 Storage Modules at Calvert Cliffs Nuclear Power Station ISFSI," PNNL-21788, 2012.*

4.52. *American Society of Mechanical Engineers, "Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer," ASME V&V 20-2009, November 30th, 2009.*

### *U.5.4.12 Shielding Analysis with a Loading of 0.380 MTU per Fuel Assembly*

*As discussed in Section U.5.4, additional shielding analysis is performed with reduced FA Uranium loading of 0.380 MTU/FA. The objective of this analysis is to determine the impact that reducing the FA Uranium loading has on system dose rates, site dose, and occupational exposure. The results of this analysis are employed to scale the dose rate, site dose, and occupational exposure results for the 32PTH1 System. For this purpose, the MCNP5 models employed for the 0.490 MTU/FA analyses are rerun with updated source terms as described in Section U.5.2.6, and with updated material specifications to reflect the reduction in MTU/FA. The material specification update for the 0.380 MTU/FA analysis uses the same active fuel stack height as that of the 0.490 MTU/FA models, and the material density is reduced accordingly to achieve the 0.380 MTU/FA total mass. This approach is very conservative in that actual 0.380 MTU/FA assemblies use full pellet density UO2 pellets which are stacked to a lower active fuel stack height. By reducing the density instead of reducing the stack height, self-shielding is significantly reduced below that of actual 0.380 MTU/FA assemblies, resulting in conservatively high dose rate estimates.*

*MCNP5 calculations are performed for the 32PTH1 DSC inside the HSM-H in the normal storage configuration. MCNP5 calculations are also performed for the 32PTH1 DSC inside the OS200 TC in the decontamination and welding configurations, and in the normal and accident transfer configurations. The resulting dose rates, site dose, and occupational exposure are compared to the 0.490 MTU/FA dose rates, site dose, and occupational exposure to determine scaling factors for each of these configurations.*

*Storage System Dose Rates*

*For the 32PTH1/HSM-H storage configuration, a comparison is shown in Table U.5-28 of the average 0.490 MTU/FA dose rates reported in Table U.5-1 and the average 0.380 MTU/FA dose rates. These results are grouped into two categories:*

1. *HSM roof and front dose rates, which are labeled as "R+F,"*

2. *HSM back and side shield wall dose rates, which are labeled as "SSW."*

*A scaling factor is calculated for each of the 4 average dose rates shown in the table labeled as "Derived Scaling Factors" and the maximum value for each category (R+F and SSW) is chosen and labeled as "Maximum Derived Scaling Factors". The maximum derived R+F scaling factor is 1.18, or an 18% increase, and the maximum derived SSW scaling factor is 1.36, or a 36% increase.*

*A comparison is shown in Table U.5-29 of the maximum 0.490 MTU/FA dose rates reported in Table U.5-1 and the maximum 0.380 MTU/FA dose rates. The maximum dose rates are categorized identically as the average dose rates, with the Roof Centerline, the Roof Bird Screen, the Door Exterior Surface, and the Front Bird Screen in the R+F category, and the End (Side) Shield Wall Surface in the SSW category. The Maximum Derived Scaling Factors from Table U.5-28 are applied to the 0.490 MTU/FA maximum dose rate values of each category in Table U.5-29 to verify the validity of using the scaling factor derived from average dose rates to*

*predict maximum dose rates. In Table U.5-29, the "Scaled 0.380 MTU/FA Results" column is the maximum 0.380 MTU/FA dose rate data produced in this manner. Comparing the scaled 0.380 MTU/FA results with the calculated 0.380 MTU/FA results demonstrates that the R+F scaling factor always generates a bounding dose rate, whereas the SSW scaling factor under predicts the dose rate by ~6%. The SSW scaling factor result is judged to be sufficiently close to the actual value so as to be acceptable for use.*

*Storage System Site Dose*

*As stated above, the 32PTH1/HSM front average dose rate increases by ≈18% when loading 0.380 MTU/FA FA's (the R+F scaling factor is 1.18). Since the site dose value is dominated by the HSM front average dose rate, the site dose value will also increase by ≈18% when loading 0.380 MTU/FA FA's.*

*Storage System Dose Rates and Site Dose Scaling Factors*

*Based on these results, three scaling factors are reported below. They are implemented to scale the existing 0.490 MTU/FA storage dose rate and site dose values to determine bounding values for FA loadings containing from 0.380 MTU/FA up to but not including 0.490 MTU/FA:*

1. *The dose rates for the HSM front and roof are to be scaled by 1.18.*

2. *The dose rates for the HSM side and rear are to be scaled by 1.36.*

3. *The site dose is to be scaled by 1.18.*

*These scaling factors are included as footnotes in the dose rate results summarized in Table U.5-1. These scaling factors are also employed to scale the generic site dose (2X10 back-to-back and front-to-front arrays) results calculated for the 32PTH1 system in Appendix U.10, and to scale the dose rate consequences of accidents for the 32PTH1 system in Appendix U.11.*

*Transfer System Dose Rates*[1]

*For the 32PTH1/OS200 TC transfer system, maximum dose rates for the three transfer configurations at both 0.490 MTU/FA and 0.380 MTU/FA are compared:*

- *Table U.5-30 shows a comparison of the maximum 0.490 MTU/FA dose rates originally reported in Table U.5-2 and the maximum 0.380 MTU/FA dose rates for the normal transfer configuration.*
- *Table U.5-31 shows a comparison of the maximum 0.490 MTU/FA dose rates originally reported in Table U.5-3 and the maximum 0.380 MTU/FA dose rates for the decontamination configuration.*
- *Table U.5-32 shows a comparison of the maximum 0.490 MTU/FA dose rates originally reported in Table U.5-3 and the maximum 0.380 MTU/FA dose rates for the welding configuration.*

[1] *For the discussion in this section, the fully loaded 32PTH1 DSC placed into the OS200 TC is referred to as the "Transfer System."*

*The results in each of these tables are grouped into three categories:*

1. *OS200 TC side dose rates, which are labeled as "Side,"*

2. *OS200 TC top dose rates, which are labeled as "Top,"*

3. *OS200 TC bottom dose rates, which are labeled as "Bottom."*

*A scaling factor is calculated for each dose rate group (top, side, and bottom) for each configuration (normal transfer, decontamination, and welding). These scaling factors are labeled as "Derived Scaling Factors." The maximum value for each category in each configuration is chosen and labeled as "Maximum Derived Scaling Factors." The maximum derived scaling factors are as follows:*

- *For the normal transfer configuration,*
  - *The side scaling factor is 1.14, or a 14% increase,*
  - *The top scaling factor is 1.60, or a 60% increase,*
  - *The bottom scaling factor is 1.51, or a 51% increase.*
- *For the decontamination configuration,*
  - *The side scaling factor is 1.10, or a 10% increase,*
  - *The top scaling factor is 1.59, or a 59% increase,*
  - *The bottom scaling factor is 1.91, or a 91% increase.*
- *For the welding configuration,*
  - *The side scaling factor is 1.24, or a 24% increase,*
  - *The top scaling factor is 1.63, or a 63% increase,*
  - *The bottom scaling factor is 1.54, or a 54% increase.*

*The highest bounding scaling factor for each cask position from the three configurations listed above is conservatively chosen to scale all dose rates from that cask position in all configurations. The bounding TC side scaling factor is the welding configuration factor of 1.24, the bounding TC top scaling factor is the welding configuration factor of 1.63, and the bounding TC bottom scaling factor is the decontamination configuration factor of 1.91.*

*<u>Transfer System Dose Rate Scaling Factors</u>*

*Based on these results, three scaling factors are reported below. They are implemented to scale the existing 0.490 MTU/FA normal transfer, welding and decontamination dose rates to determine bounding values for FA loadings containing from 0.380 MTU/FA up to but not including 0.490 MTU/FA:*

- *The dose rates for the TC side are to be scaled by 1.24,*
- *The dose rates for the TC top are to be scaled by 1.63,*

- *The dose rates for the TC bottom are to be scaled by 1.91.*

*These scaling factors are included as footnotes in the dose rate results summarized in Tables U.5-2, Table U.5-3, Table U.5-21, and Table U.5-22.*

*The 0.380 MTU/FA OS200 TC accident transfer configuration dose rates are bound by the 0.490 MTU/FA analysis results, and do not require updating.*

*Transfer System Occupational Exposure*

*The 0.380 MTU/FA normal transfer, decontamination, and welding dose rates from Tables U.5-30, U.5-31, and U.5-32 respectively are substituted into the area dose rate fields of the chapter U.10 occupational exposure summary table, Table U.10-1, and the total occupational exposure from loading 0.380 MTU/FA fuel is calculated. A comparison is shown in Table U.5-33 of the 0.490 MTU/FA area dose rates, total exposure by operational step, and cumulative operational exposure originally reported in Table U.10-1 and the 0.380 MTU/FA area dose rates, total exposure by operational step, and cumulative operational exposure. At the bottom of Table U.5-33, the 0.490 MTU/FA cumulative operational exposure is 1934 person-millirem and the 0.380 MTU/FA cumulative operational exposure is 2425 person-millirem. Loading 0.380 MTU/FA fuel results in an increase in occupational exposure of 25% (which is a scaling factor of 1.25) when compared to loading 0.490 MTU/FA fuel.*

*Transfer System Occupational Exposure Scaling Factors*

*Based on these results, the scaling factor of 1.25 is implemented to scale the existing 0.490 MTU/FA transfer system pool to pad loading operation occupational exposure to determine the bounding value for FA loadings containing from 0.380 MTU/FA up to but not including 0.490 MTU/FA. This scaling factor is employed to scale the occupational exposure results calculated for the 32PTH1 system in Appendix U.10.*

All indicated changes are in response to RAI 6-5

*Table U.5-28*
*Comparison of Bounding Average Dose Rates with NUHOMS®-32PTH1 Bounding DSC in HSM-H with 0.490 MTU/FA and 0.380 MTU/FA to Derive Dose Rate Scaling Factors for Storage*

| Dose Rate Location | Average Gamma (mrem/hour) | | Average Neutron (mrem/hour) | | Average Total (mrem/hour) | | | | - |
|---|---|---|---|---|---|---|---|---|---|
| | 0.490 MTU | 0.380 MTU | 0.490 MTU | 0.380 MTU | 0.490 MTU | 0.380 MTU | DR Change | Derived Scaling Factors | Maximum Derived Scaling Factors |
| HSM Roof | 11.27 | 13.47 | 0.62 | 0.51 | 11.89 | 13.98 | +2.09 | 1.18 | Max R+F = 1.18 |
| HSM Front | 14.87 | 12.60 | 0.31 | 0.23 | 15.19 | 12.83 | -2.36 | 0.84 | |
| HSM End (Side) Shield Wall Surface | 0.36 | 0.50 | 0.02 | 0.02 | 0.38 | 0.52 | +0.14 | 1.36 | Max SSW= 1.36 |
| HSM Back Shield Wall | 0.07 | 0.07 | 0.00 | 0.00 | 0.07 | 0.07 | 0.00 | 1.06 | |

*Table U.5-29*
*Verification of Derived Dose Rate Scaling Factors for Predicting Maximum Dose Rates for NUHOMS®-32PTH1 Bounding DSC in HSM-H with 0.380 MTU/FA for Storage*

| Dose Rate Location | Maximum Gamma (mrem/hour) | | Maximum Neutron (mrem/hour) | | Maximum Total (mrem/hour) | | | | Scaling Factor Applied to 0.490 MTU/FA Max Total |
|---|---|---|---|---|---|---|---|---|---|
| | 0.490 MTU | 0.380 MTU | 0.490 MTU | 0.380 MTU | 0.490 MTU | 0.380 MTU | DR Change | Scaled 0.380 MTU/FA Results | |
| HSM Roof (centerline) | 14.13 | 16.49 | 0.74 | 0.59 | 14.86 | 17.08 | +2.22 | 17.47 | R+F 1.18 scaling factor is applied |
| HSM Roof Birdscreen | 114.71 | 123.93 | 6.47 | 0.04 | 121.17 | 123.97 | +2.8 | 142.45 | |
| HSM Door Exterior Surface (centerline) | 0.61 | 0.18 | 0.08 | 0.20 | 0.70 | 0.38 | -0.32 | 0.82 | |
| HSM Front Birdscreen | 471.28 | 476.86 | 5.89 | 4.78 | 477.17 | 481.64 | +4.47 | 560.96 | |
| HSM End (Side) Shield Wall Surface | 1.49 | 2.15 | 0.06 | 0.04 | 1.54 | 2.19 | +0.65 | 2.09 | SSW 1.36 scaling factor is applied |

*Table U.5-30*

***Comparison of Bounding Maximum Dose Rates with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Dose Rate Scaling Factors for the Transfer Configuration***

| *Transfer Configuration* | *Maximum Gamma (mrem/hour)* | | *Maximum Neutron (mrem/hour)* | | *Maximum Total (mrem/hour)* | | | *Maximum Derived Scaling Factors* |
|---|---|---|---|---|---|---|---|---|
| *Dose Rate Location* | *0.490 MTU/FA* | *0.380 MTU/FA* | *0.490 MTU/FA* | *0.380 MTU/FA* | *0.490 MTU/FA* | *0.380 MTU/FA* | *Derived Scaling Factor* | |
| *Cask Side Surface (Radial)* | *407.0* | *501.2* | *202.0* | *136.3* | *609.0* | *637.6* | *1.05* | *Max Side = 1.14* |
| *1.5 ft from Cask Side (Radial)* | *239.0* | *314.1* | *123.0* | *82.9* | *362.0* | *397.0* | *1.10* | |
| *3 ft from Cask Side (Radial)* | *161.0* | *222.1* | *83.9* | *57.4* | *245.0* | *279.5* | *1.14* | |
| *Cask Top Axial Surface* | *232.0* | *363.8* | *38.6* | *29.9* | *251.0* | *380.7* | *1.52* | *Max Top = 1.60* |
| *1.5 ft from Cask Top Axial Surface* | *47.1* | *85.0* | *20.7* | *16.2* | *59.5* | *95.2* | *1.60* | |
| *3 ft from Cask Top Axial Surface* | *29.5* | *45.3* | *14.4* | *10.3* | *37.9* | *52.6* | *1.39* | |
| *Cask Bottom Axial Surface* | *2150.0* | *4129.6* | *1400.0* | *806.3* | *3550.0* | *4935.9* | *1.39* | *Max Bottom = 1.51* |
| *1.5 ft from Cask Bottom Axial Surface* | *925.0* | *1732.0* | *343.0* | *184.2* | *1270.0* | *1916.2* | *1.51* | |
| *3 ft from Cask Bottom Axial Surface* | *465.0* | *661.2* | *140.0* | *125.9* | *605.0* | *710.9* | *1.18* | |

*Table U.5-31*
*Comparison of Bounding Maximum Dose Rates with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Dose Rate Scaling Factors for the Decontamination Configuration*

| Decontamination Configuration | Maximum Gamma (mrem/hour) | | Maximum Neutron (mrem/hour) | | Maximum Total (mrem/hour) | | | Maximum Derived Scaling Factors |
|---|---|---|---|---|---|---|---|---|
| Dose Rate Location | 0.490 MTU/FA | 0.380 MTU/FA | 0.490 MTU/FA | 0.380 MTU/FA | 0.490 MTU/FA | 0.380 MTU/FA | Derived Scaling Factor | |
| Cask Side Surface (Radial) | 346.0 | 435.1 | 373.0 | 314.6 | 719.0 | 749.7 | 1.04 | Max Side = 1.10 |
| 1.5 ft from Cask Side (Radial) | 202.0 | 267.3 | 229.0 | 194.7 | 431.0 | 462.0 | 1.07 | |
| 3 ft from Cask Side (Radial) | 135.0 | 185.6 | 158.0 | 135.4 | 292.0 | 321.0 | 1.10 | |
| Top Axial Surface | 832.0 | 1320.7 | 9.3 | 9.9 | 833.0 | 1321.5 | 1.59 | Max Top = 1.59 |
| 1.5 ft from Top Axial Surface | 615.0 | 978.1 | 5.6 | 5.6 | 616.0 | 978.7 | 1.59 | |
| 3 ft from Top Axial Surface | 427.0 | 675.5 | 3.6 | 3.6 | 428.0 | 676.5 | 1.58 | |
| Cask Bottom Axial Surface | 1700.0 | 3254.8 | 68.3 | 133.6 | 1770.0 | 3388.5 | 1.91 | Max Bottom = 1.91 |
| 1.5 ft from Cask Bottom Axial Surface | 731.0 | 1371.3 | 18.2 | 42.4 | 749.0 | 1413.7 | 1.89 | |
| 3 ft from Cask Bottom Axial Surface | 368.0 | 662.1 | 8.1 | 11.9 | 376.0 | 662.1 | 1.76 | |

### *Table U.5-32*
### *Comparison of Bounding Maximum Dose Rates with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Dose Rate Scaling Factors for the Welding Configuration*

| Welding Configuration | Maximum Gamma (mrem/hour) | | Maximum Neutron (mrem/hour) | | Maximum Total (mrem/hour) | | | Maximum Derived Scaling Factors |
|---|---|---|---|---|---|---|---|---|
| Dose Rate Location | 0.490 MTU/FA | 0.380 MTU/FA | 0.490 MTU/FA | 0.380 MTU/FA | 0.490 MTU/FA | 0.380 MTU/FA | Derived Scaling Factor | |
| Cask Side Surface (Radial) | 309.0 | 424.0 | 147.0 | 97.2 | 456.0 | 521.2 | 1.14 | Max Side = 1.24 |
| 1.5 ft from Cask Side (Radial) | 185.0 | 270.3 | 89.9 | 59.4 | 275.0 | 329.7 | 1.20 | |
| 3 ft from Cask Side (Radial) | 127.0 | 193.2 | 62.3 | 41.3 | 189.0 | 234.6 | 1.24 | |
| Top Axial Surface | 720.0 | 1202.1 | 42.1 | 36.8 | 762.0 | 1238.9 | 1.63 | Max Top = 1.63 |
| 1.5 ft from Top Axial Surface | 425.0 | 653.3 | 17.4 | 17.4 | 442.0 | 670.7 | 1.52 | |
| 3 ft from Top Axial Surface | 294.0 | 459.9 | 11.7 | 11.8 | 306.0 | 470.6 | 1.54 | |
| Cask Bottom Axial Surface | 2170.0 | 4133.4 | 1220.0 | 695.1 | 3390.0 | 4828.5 | 1.42 | Max Bottom = 1.54 |
| 1.5 ft from Cask Bottom Axial Surface | 927.0 | 1731.4 | 300.0 | 167.8 | 1230.0 | 1899.2 | 1.54 | |
| 3 ft from Cask Bottom Axial Surface | 466.0 | 668.9 | 121.0 | 93.1 | 587.0 | 706.1 | 1.20 | |

All indicated changes are in response to RAI 6-5

## *Table U.5-33*
## *Comparison of Bounding Maximum Dose Rates and Cumulative Doses with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Occupational Exposure Scaling Factor for Pool to Pad Loading Operations*

| Location | Task Description | Area Dose Rate w/0.490 MTU/FA (mrem/hr) | Total Exposure w/0.490 MTU/FA (person-mrem) | Area Dose Rate w/0.380 MTU/FA (mrem/hr) | Total Exposure w/0.380 MTU/FA (person-mrem) | Area Dose Rate and Total Exposure Change |
|---|---|---|---|---|---|---|
| Auxiliary Building and Fuel Pool | Place the DSC into the Transfer Cask | 2 | 8 | 2 | 8 | 1.00 |
| | Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal | 2 | 6 | 2 | 6 | 1.00 |
| | Fill the DSC Cavity with Water (borated for PWRs) | 2 | 12 | 2 | 12 | 1.00 |
| | Place the Cask Containing the DSC in the Fuel Pool | 2 | 5 | 2 | 5 | 1.00 |
| | Verify and Load the Candidate Fuel Assemblies into the DSC | 2 | 30 | 2 | 30 | 1.00 |
| | Place the Top Shield Plug on the DSC | 2 | 4 | 2 | 4 | 1.00 |
| | Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area | 2 | 5 | 2 | 5 | 1.00 |
| | | 199 | 7 | 229 | 8 | 1.15 |
| | | 146 | 98 | 169 | 112 | 1.15 |
| Cask Decontamination Area | Decontaminate the Outer Surface of the Cask | 146 | 256 | 169 | 295 | 1.15 |
| | | 2 | 2 | 2 | 2 | 1.00 |
| | Decontaminate the Top Region of the Cask and DSC | 195 | 97 | 293 | 147 | 1.51 |
| | | 64 | 32 | 90 | 45 | 1.41 |
| | Drain Water from the DSC | 199 | 17 | 229 | 19 | 1.15 |
| | Remove Cask/DSC Annulus Seal and Set-Up Welding Machine | 337 | 56 | 515 | 86 | 1.53 |
| | | 87 | 65 | 123 | 92 | 1.41 |
| | | 72 | 36 | 97 | 49 | 1.35 |
| | Weld the Inner Top Cover to the DSC Shell and Perform NDE (PT) | 2 | 24 | 2 | 24 | 1.00 |
| | | 169 | 56 | 251 | 84 | 1.49 |
| | Drain the Cask/DSC Annulus and the DSC Cavity | 87 | 22 | 123 | 31 | 1.41 |
| | | 169 | 3 | 251 | 4 | 1.49 |
| | | 2 | 1 | 2 | 1 | 1.00 |
| | Vacuum Dry and Backfill the DSC with Helium | 72 | 36 | 97 | 49 | 1.35 |
| | | 2 | 120 | 2 | 120 | 1.00 |
| | Helium Leak Test the Shield Plug Weld | 2 | 4 | 2 | 4 | 1.00 |
| | Seal Weld the Prefabricated Plugs to the Vent and Siphon Port and Perform NDE (PT) | 87 | 44 | 123 | 61 | 1.41 |
| | Fit-Up the DSC Top Cover Plate | 169 | 42 | 251 | 63 | 1.49 |
| | | 87 | 44 | 123 | 61 | 1.41 |
| | Weld the Outer Top Cover Plate to DSC Shell and Perform NDE (PT) | 72 | 72 | 97 | 97 | 1.35 |
| | | 169 | 28 | 251 | 42 | 1.49 |
| | | 2 | 56 | 2 | 56 | 1.00 |
| | | 169 | 56 | 251 | 84 | 1.49 |
| | Install The Cask Lid | 93 | 124 | 115 | 153 | 1.23 |

All indicated changes are in response to RAI 6-5

*Table U.5-33*

***Comparison of Bounding Maximum Dose Rates and Cumulative Doses with NUHOMS®-32PTH1 Bounding DSC in OS200 TC with 0.490 MTU/FA and 0.380 MTU/FA to Derive Occupational Exposure Scaling Factor for Pool to Pad Loading Operations***

*(Continued)*

| Location | Task Description | Area Dose Rate w/0.490 MTU/FA (mrem/hr) | Total Exposure w/0.490 MTU/FA (person-mrem) | Area Dose Rate w/0.380 MTU/FA (mrem/hr) | Total Exposure w/0.380 MTU/FA (person-mrem) | Area Dose Rate and Total Exposure Change |
|---|---|---|---|---|---|---|
| Reactor/ Fuel Building Bay | Ready the Cask Support Skid and Transport Trailer for the Service | 2 | 8 | 2 | 8 | 1.00 |
| | Place the Cask onto the Skid and Trailer | 136 | 68 | 158 | 79 | 1.16 |
| | Secure the Cask to the Skid | 136 | 34 | 158 | 40 | 1.16 |
| ISFSI Site | Ready The Cask Support Skid and Transport Trailer for the Service | negligible | 0 | negligible | 0 | - |
| | Transport the Cask to ISFSI | negligible | 0 | negligible | 0 | - |
| | Position the Cask in Close Proximity with the HSM | negligible | 0 | negligible | 0 | - |
| | Remove the Cask Lid | 42 | 56 | 47 | 63 | 1.12 |
| | Align and Dock the Cask with the HSM | 108 | 54 | 130 | 65 | 1.20 |
| | Position and Align Ram with Cask | 174 | 174 | 222 | 222 | 1.28 |
| | Remove Ram Access Cover Plate | 562 | 47 | 757 | 63 | 1.35 |
| | Transfer the DSC from the Cask to the HSM | negligible | 0 | negligible | 0 | - |
| | Lift the Ram Back onto the Trailer and Un-Dock the Cask from the | 56 | 9 | 65 | 11 | 1.17 |
| | Install HSM Access Door | 15 | 15 | 15 | 15 | 1.00 |
| | Totals | - | 1934 | - | 2425 | 1.25 |

All indicated changes are in response to RAI 6-5

## U.10.1 Occupational Exposure

*The occupational exposure results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section U.5.4.12. Loading 0.380 MTU fuel results in an increase in occupational exposure of 25%.*

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® 32PTH1 DSC loaded with design basis fuel assemblies into storage is conservatively estimated to be 2 person-rem as summarized in Table U.10-1. This is a very conservative estimate because the dose rates on and around 32PTH1 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® 32PTH1 system loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table U.10-1. This information is used as the basis for estimating the total occupational exposure associated with one fuel load. The dose rates applicable for each operation are based on the results presented in Section U.5.4 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-H. 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 U.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 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.

Localized regions of elevated dose rates should be anticipated and minimized with good ALARA practices. Such regions exist due primarily to radiation streaming, including for example, streaming through the cask/DSC annulus, the ventilation paths in OS200 lid and the DSC vent/siphon ports.

All indicated changes are in response to RAI 6-5

## Table U.10-1
## Occupational Exposure Summary, 32PTH1 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 | 2 | 2 | 2 | 8 |
| | Fill the Cask/DSC Annulus with Clean Water and Install the Inflatable Seal | 3 | 1 | 2 | 6 |
| | Fill the DSC Cavity with Water | 1 | 6 | 2 | 12 |
| | Place the Cask Containing the DSC in the Fuel Pool | 5 | 0.5 | 2 | 5 |
| | Verify and Load the Candidate Fuel Assemblies into the DSC | 3 | 5 | 2 | 30 |
| | Place the Top Shield Plug on the DSC | 2 | 1 | 2 | 4 |
| | Remove the Cask/DSC from the Fuel Pool and Place them in the Decon Area | 5 | 0.5 | 2 | 5 |
| | | 1 | 0.033 | 199 | 7 |
| | | 1 | 0.667 | 146 | 98 |
| Cask Decontamination Area | Decontaminate the Outer Surface of the Cask | 1 | 1.75 | 146 | 256 |
| | | 1 | 1 | 2 | 2 |
| | Decontaminate the Top Region of the Cask and DSC | 1 | 0.5 | 195 | 97 |
| | | 1 | 0.5 | 64 | 32 |
| | Drain Water from the DSC | 1 | 0.083 | 199 | 17 |
| | Remove Cask/DSC Annulus Seal and Set-Up Welding Machine | 1 | 0.167 | 337 | 56 |
| | | 1 | 0.75 | 87 | 65 |
| | | 1 | 0.5 | 72 | 36 |
| | Weld the Inner Top Cover to the DSC Shell and Perform NDE (PT) | 2 | 6 | 2 | 24 |
| | | 1 | 0.33 | 169 | 56 |
| | Drain the Cask/DSC Annulus and the DSC Cavity | 1 | 0.25 | 87 | 22 |
| | | 1 | 0.017 | 169 | 3 |
| | | 1 | 0.5 | 2 | 1 |
| | Vacuum Dry and Backfill the DSC with Helium | 1 | 0.5 | 72 | 36 |
| | | 2 | 30 | 2 | 120 |
| | Helium Leak Test the Shield Plug Weld | 2 | 1 | 2 | 4 |
| | Seal Weld the Prefabricated Plugs to the Vent and Siphon Ports and Perform NDE (PT) | 1 | 0.5 | 87 | 44 |
| | Fit-Up the DSC Outer Top Cover Plate | 1 | 0.25 | 169 | 42 |
| | | 1 | 0.5 | 87 | 44 |
| | Weld the Outer Top Cover Plate to DSC Shell and Perform NDE (PT) | 1 | 1 | 72 | 72 |
| | | 1 | 0.167 | 169 | 28 |
| | | 2 | 14 | 2 | 56 |
| | | 1 | 0.333 | 169 | 56 |
| | Install the Cask Lid | 2 | 0.667 | 93 | 124 |
| Reactor/ Fuel Building Bay | Ready the Cask Support Skid and Transfer Trailer for Service | 2 | 2 | 2 | 8 |
| | Place the Cask onto the Skid and Trailer | 2 | 0.25 | 136 | 68 |
| | Secure the Cask to the Skid | 1 | 0.25 | 136 | 34 |
| 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 | 42 | 56 |
| | Align and Dock the Cask with the HSM | 2 | 0.25 | 108 | 54 |
| | Position and Align Ram with Cask | 2 | 0.5 | 174 | 174 |
| | Remove Ram Access Cover Plate | 1 | 0.083 | 562 | 47 |
| | 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 HSM | 2 | 0.083 | 56 | 9 |
| | Install HSM Access Door | 2 | 0.5 | 15 | 15 |
| **Totals** | | **N/A** | **87** | **N/A** | **1934** |

Total estimated dose is 2 person-rem per 32PTH1 canister load.

*Total estimated dose increases by 25% when loading 0.380 MTU/FA*

U.11.2.3.1 Cause of Accident

No change to the description presented in UFSAR Section 8.2.2.1. 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.

U.11.2.3.2 Accident Analysis

An evaluation that investigates the effect of the addition of the NUHOMS® 32PTH1 DSC, is presented in Chapter U.3, Section U.3.7.1. The evaluation of the HSM-H for the effect of DBT wind pressure loads is addressed in Section U.3.7.1.1. The tornado missile impact evaluation of the HSM-H / HSM-HS is presented in the following sections.

U.11.2.3.2.1 HSM-H/HSM-HS Missile Impact Analysis

No change to the missile impact evaluation presented in Appendix P, Section P.11.2.3.2.1.

To accommodate the longest 32PTH1 DSC inside the HSM-H / HSM-HS cavity, the concrete thickness of the shield door is reduced from 22.5 inches to 18.5 inches. The missile evaluations of the shielded composite door, described in Section P.11.2.3.2.1, do not take credit for the 22.5 inch concrete thickness that is structurally composite with the 7.875 inch total steel plate thickness. Therefore, the shielded door evaluations for missile loadings as presented in Section P.11.2.3.2.1 remain applicable for the shielded door with reduced concrete thickness.

In addition, as shown in the drawings for the HSM-H / HSM-HS in Section U.1.5, an optional door has been added to the HSM-H / HSM-HS design. The optional door has an additional 6.875 inches concrete thickness but the steel thickness is reduced to 3 inches.

As noted above, the evaluation of the shielded door includes an additional missile (8" diameter armor piercing artillery shell with a mass of 280 lbs and impact velocity of 508 fps). The controlling missile evaluations require a minimum steel thickness of 2.5 inches. Therefore, a door with 3 inch steel thickness is qualified for missile loads.

U.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 *600* mrem/hour in Table U.5-1 *(477.17 mrem/hr*1.18 ~ 600 mrem/hr)*, 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 greater than 20 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 surface area 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 U.10.2 for a 2x10 array of undamaged HSMs (specifically Table U.10-7) 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.*

All indicated changes are in response to RAI 7-6

*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*dose rate at 100m, 8.75E-02 mrem/hour*1.18 scaling factor) with a 2x10 array of HSMs. 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*dose rate at 500m, 1.83E-04 mrem/hour*1.18 scaling factor) with a 2x10 array of HSMs.*

U.11.2.3.4 Corrective Actions

After excessive high winds or a tornado, the HSM-H/HSM-HS and OS200 TC would be inspected for damage. Any debris would be removed. Any damage resulting from impact with a missile would be evaluated to determine if the system was still within the licensed design basis.

U.11.2.4 Flood

This event is described in UFSAR Section 8.2.4.

U.11.2.4.1 Cause of Accident

No change. See UFSAR Section 8.2.4.1.

U.11.2.4.2 Accident Analysis

The HSM-H / HSM-HS and DSCs are evaluated for flooding in Section U.3.7.3. The DSC is designed and tested to be leak tight to the criteria of ANSI N14.5 [11.2]. The stresses in the DSC due to the design basis flood are well below the allowable stresses for Service Level C of the ASME Code Subsection NB [11.5]. Therefore, the NUHOMS® 32PTH1 DSC will withstand the design basis flood without breach of the confinement boundary.

U.11.2.4.3 Accident Dose Calculations

The radiation dose due to flooding of the HSM-H is negligible. The NUHOMS® 32PTH1 DSC is designed and tested as a leak-tight containment boundary. Flooding does not breach the containment boundary. Therefore radioactive material inside the DSC will remain sealed in the DSC and, therefore, will not contaminate the encroaching flood water.

U.11.2.4.4 Corrective Actions

No change. See UFSAR Section 8.2.4.4.

U.11.2.5 Accidental TC Drop

This event is described in UFSAR Section 8.2.5.

U.11.2.5.1 Cause of Accident

See Section U.3.7.4.

### Y.2.1 Spent Fuel to be Stored

#### Y.2.1.1 Intact or Damaged Fuel

As described in Appendix Y.1, the NUHOMS®-69BTH DSC is designed to store intact (including reconstituted) and/or damaged (boiling water reactor) BWR fuel assemblies as specified in Table Y.2-1 and Table Y.2-2. The fuel to be stored is limited to a maximum lattice average initial enrichment of 5.0 wt. % U-235. The maximum allowable fuel assembly average burnup is limited to 62 GWd/MTU. The minimum required cooling time for fuel to be stored with 170 kgU/FA and 198 kgU/FA is explicitly specified as a function of burnup and enrichment in Tables Y.2-5 through Y.2-17b. For fuel with a kgU/FA loading between these two values, the minimum required cooling time for fuel to be stored as a function of burnup and enrichment is determined by using the instructions provided in the the notes and examples following Table Y.2-16.

The NUHOMS®-69BTH DSC is also authorized to store fuel assemblies containing blended low enriched uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity affects only 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 69 lower enrichment $UO_2$ rods instead of zircaloy clad enriched $UO_2$ rods are acceptable for storage in 69BTH 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 assemblies per DSC is four with irradiated stainless steel rods or 69 with $UO_2$ rods or Zr rods or Zr pellets or unirradiated stainless steel rods.

The NUHOMS®-69BTH DSCs can also accommodate up to a maximum of 24 damaged fuel assemblies placed in the four outer "six compartment" arrays located at the outer edge of the DSC as shown in Figure Y.2-7. 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 is to be limited such that the fuel assembly, *including non-cladding damage,* is able to be handled by normal means *and the retrievability is assured following the normal and off-normal conditions*. Missing fuel rods are allowed. *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 which store damaged fuel assemblies are provided with top and bottom end caps.

A 69BTH DSC containing less than 69 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, aluminum blocks that approximate the weight and center of gravity of a fuel assembly.

All indicated changes are in response to RAI 6-5

**Welding and 32PTH1-DSC Draining:** Before the start of welding operation, approximately 60% of the water in the DSC cavity is removed due to hydrogen generation. A dry DSC cavity is assumed in all welding models to be conservative. Temporary shielding consisting of three inches of NS-3 and one inch of steel is assumed to cover the 32PTH1 DSC inner top cover plate. In addition, the DSC outer top cover plate is not present. The cask/32PTH1 DSC annulus is assumed to remain completely filled with water. Results for this case are provided in Table Z.5-3.

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

A bounding analysis is performed by employing a minimum concrete density of 140 pounds per cubic foot (pcf) in the HSM-H MCNP model combined with a maximum gap of 1.5 inches between adjacent HSM-H modules and shield walls to determine the effect on maximum and average dose rates due to a fully loaded 32PTH1 DSC. These calculations are documented in Appendix U.5, Section U.5.4.10. The ratios shown in Appendix U.5, Table U.5-18 and Table U.5-19 can be used as scaling factors to increase the maximum and surface-average dose rates of the 37PTH in the HSM-H to account for low density concrete and 1.5" gaps. Note that the HSM-H concrete contains high density rebar which is not credited in the MCNP models. Further, the modules are installed adjacent to each other such that there will not be a "uniform" gap of 1.5 inches. Ignoring the effect due to increased vent dose rates, the increase in the average dose rates caused by both the maximum postulated uniform gaps and the minimum postulated concrete density is expected to be less than 20% at the front and roof surfaces of the HSM-H module. Dose reduction hardware may be installed to further reduce these dose rates.

*Z.5.4.11 Shielding Analysis with a Loading of 0.380 MTU per Fuel Assembly*

*As discussed in Section Z.5.4, additional shielding analysis is performed with a reduced Uranium loading of 0.380 MTU per fuel assembly. The objective of this analysis is to determine the impact that reduced Uranium loading has on system dose rates. The results of this analysis are employed to scale the dose rate results for the 37PTH System. For this purpose, the MCNP5 models employed for the 0.490 MTU analyses are rerun with updated source terms as described in Section Z.5.2.5, and with updated material specifications to reflect the reduction in MTU. MCNP5 calculations are performed for the 37PTH DSC inside the HSM-H, and dose rate scaling factors are derived using the same methodology as that described in Appendix U, Section U.5.4.12. As described in Section Z.5.4.5, the TC dose rates obtained from Section U.5 for the 32PTH1 DSC are applicable to the 37PTH DSC. The TC analysis for the 0.380 MTU loading is documented in Appendix U, Section U.5.4.12 and includes decontamination, welding, normal and accident conditions of transfer dose rates, and occupational exposure. The resulting dose rates, and occupational exposure are compared to the 0.490 MTU dose rates, and occupational exposure to determine scaling factors for these configurations.*

All indicated changes are in response to RAI 6-5

*Based on the updated results, six scaling factors are determined and are summarized as follows:*

- *The dose rates for the HSM-H front and roof are to be scaled by 1.25.*
- *The dose rates for the HSM-H side and rear are to be scaled by 1.35.*
- *The site dose for the HSM is to be scaled by 1.25.*
- *The dose rates for the TC for normal, welding and decontamination are to be scaled as follows:*
    - o *by 1.24 for the side,*
    - o *by 1.63 for the top,*
    - o *by 1.91 for the bottom.*
- *The dose rates for the TC for accidents are bound by the 0.490 MTU analysis results, and do not require updating.*
- *The occupational exposure for TC loading and storage operations is to be scaled by 1.25.*

*These scaling factors are included as footnotes in the dose rate results summarized in Table Z.5-1, Table Z.5-2, Table Z.5-3, and Table Z.5-19.*

*These scaling factors are also employed to scale the occupational exposure and generic site dose (2X10 back-to-back and front-to-front arrays) results calculated for the 37PTH system in Section Z.10, and to scale the dose rate consequences of accidents for the 37PTH system in Section Z.11*

All indicated changes are in response to RAI 6-5

Z.10.1 Occupational Exposure

*The occupational exposure results shown herein do not account for loading of 0.380 MTU fuel, which is described in Section Z.5.4.11. Loading 0.380 MTU fuel results in an increase in occupational exposure of 25%.*

The expected occupational dose for placing a canister of spent fuel into dry storage is based on the operational steps outlined in Chapter 7, Table 7.4-1 of the UFSAR. The total exposure for the occupational dose due to placing a single NUHOMS® 37PTH DSC loaded with design basis fuel assemblies into storage is estimated to be 2 person-rem as summarized in Table Z.10-1. The calculated exposures are due mainly to the expected gamma dose rate during preparation for welding.

The NUHOMS® 37PTH system loading operations, the number of workers required for each operation, and the amount of time required for each operation are presented in Table Z.10-1. This information is used as the basis for estimating the total occupational exposure associated with one fuel load. The dose rates applicable for each operation are based on the results presented in Appendix U.5, Section U.5.4 for loading operations. Based on the NUHOMS®-37PTH system HSM-H dose rates in Appendix Z.5, Section Z.5.4, it is conservative to bound the NUHOMS®-37PTH system by the NUHOMS®-32PTH1 system. 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-H. 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 Z.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 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.

Localized regions of elevated dose rates should be anticipated and minimized with good ALARA practices. Such regions exist due primarily to radiation streaming, including for example, streaming through the cask/DSC annulus, the ventilation paths in OS200 TC lid, and the DSC vent/siphon ports.

The results of the evaluations of the NUHOMS® 37PTH are presented in Table Z.10-1.

All indicated changes are in response to RAI 6-5

## Table Z.10-1
## Occupational Exposure Summary, 37PTH 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. | 2 | 2 | 2 | 8 |
| | Fill the cask/DSC annulus with clean water and install the inflatable seal. | 3 | 1 | 2 | 6 |
| | Fill the DSC cavity with water. | 1 | 6 | 2 | 12 |
| | Place the cask containing the DSC in the fuel pool. | 5 | 0.5 | 2 | 5 |
| | Verify and load the candidate fuel assemblies into the DSC. | 3 | 5 | 2 | 30 |
| | Place the top shield plug on the DSC. | 2 | 1 | 2 | 4 |
| | Remove the cask/DSC from the fuel pool and place them in the decon area. | 5 | 0.5 | 2 | 5 |
| | | 1 | 0.033 | 199 | 7 |
| | | 1 | 0.667 | 146 | 98 |
| **Cask Decontamination Area** | Decontaminate the outer surface of the cask. | 1 | 1.75 | 146 | 256 |
| | | 1 | 1 | 2 | 2 |
| | Decontaminate the top region of the cask and DSC. | 1 | 0.5 | 195 | 97 |
| | | 1 | 0.5 | 64 | 32 |
| | Drain water from the DSC. | 1 | 0.083 | 199 | 17 |
| | Remove cask/DSC annulus seal and set-up welding machine. | 1 | 0.167 | 337 | 56 |
| | | 1 | 0.75 | 87 | 65 |
| | | 1 | 0.5 | 72 | 36 |
| | Weld the inner top cover to the DSC shell and perform NDE (PT). | 2 | 6 | 2 | 24 |
| | | 1 | 0.33 | 169 | 56 |
| | Drain the cask/DSC annulus and the DSC cavity. | 1 | 0.25 | 87 | 22 |
| | | 1 | 0.017 | 169 | 3 |
| | | 1 | 0.5 | 2 | 1 |
| | Vacuum dry and backfill the DSC with helium. | 1 | 0.5 | 72 | 36 |
| | | 2 | 30 | 2 | 120 |
| | Helium leak test the shield plug weld. | 2 | 1 | 2 | 4 |
| | Seal weld the prefabricated plugs to the vent and siphon ports and perform NDE (PT). | 1 | 0.5 | 87 | 44 |
| | Fit-up the DSC outer top cover plate. | 1 | 0.25 | 169 | 42 |
| | | 1 | 0.5 | 87 | 44 |
| | Weld the outer top cover plate to DSC shell and perform NDE (PT). | 1 | 1 | 72 | 72 |
| | | 1 | 0.167 | 169 | 28 |
| | | 2 | 14 | 2 | 56 |
| | | 1 | 0.333 | 169 | 56 |
| | Install the cask lid. | 2 | 0.667 | 93 | 124 |
| **Reactor/Fuel Building Bay** | Ready the cask support skid and transport trailer for service. | 2 | 2 | 2 | 8 |
| | Place the cask onto the skid and trailer. | 2 | 0.25 | 136 | 68 |
| | Secure the cask to the skid. | 1 | 0.25 | 136 | 34 |
| **ISFSI Site** | Ready the cask support skid and transport trailer for service. | 2 | 2 | negligible | 0 |
| | Transport 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 | 42 | 56 |
| | Align and dock the cask with the HSM. | 2 | 0.25 | 108 | 54 |
| | Position and align ram with cask. | 2 | 0.5 | 174 | 174 |
| | Remove ram access cover plate. | 1 | 0.083 | 562 | 47 |
| | 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 HSM. | 2 | 0.083 | 56 | 9 |
| | Install HSM access door. | 2 | 0.5 | 15 | 15 |
| | **Totals** | **N/A** | **87** | **N/A** | **1934** |

Total estimated dose is 2 person-rem per 37PTH canister load.

*Total estimated dose increases by 25% when loading 0.380 MTU/FA fuel.*

All indicated changes are in response to RAI 7-6

Z.11.2.3.2 Accident Analysis

There is no change to the missile impact evaluation presented in Appendix U, Section U.11.2.3.2.

Z.11.2.3.2.1 HSM-H/HSM-HS Missile Impact Analysis

There is no change to the missile impact evaluations presented in Appendix U, Section U.11.2.3.2.1.

Z.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 600 mrem/hour in Appendix Z.5, Table Z.5-1 *(477.17 mrem/hr*1.25 ~ 600 mrem/hr),* 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 greater than 20 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 surface area 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 Z.10.2 for a 2x10 array of undamaged HSMs (specifically Table Z.10-7) 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*dose rate at 100m, 8.75E-02 mrem/hour*1.25 scaling factor) with a 2x10 array of HSMs. 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*dose rate at 500m, 1.83E-04 mrem/hour*1.25 scaling factor) with a 2x10 array of HSMs.*

Z.11.2.3.4 Corrective Actions

There is no change to the corrective actions presented in Appendix U.11, Section U.11.2.3.4.

Z.11.2.4 Flood

This event is described in Chapter 8, Section 8.2.4.

Z.11.2.4.1 Cause of Accident

No change. See Chapter 8, Section 8.2.4.1.

Z.11.2.4.2 Accident Analysis

There is no change to the accident analysis presented for HSM-H and HSM-HS in Appendix U.11, Section U.11.2.4.2. The DSC is designed and tested to be leak tight to the criteria of ANSI N14.5 [11.2]. The stresses in the DSC due to the design basis flood are well below the allowable

Listing of Computer Files Contained in Enclosure 10

| Disk ID No. (size) | Discipline | System/Component | File Series (topics) | Number of Files |
|---|---|---|---|---|
| Enclosure 10<br>One Computer Hard Drive<br>(9.46 GB) | Thermal | 32PTH1 Type 1 DSC in HSM-H | **NUH32PTH1-0433-000/1-LC1a-CoarseMesh Directory:**<br>Input and output files for the steady-state run of the HSM-H loaded with the 32PTH1 Type 1 DSC basket assembly under the bounding storage condition. | 16 |
| | Criticality | 32PT | **RAI7-4 Directory**<br>Input and output files for sensitivity runs mentioned in the RAI response<br>• Damaged fuel | |
| | | | 28-damaged\CE15 folder | 14 |
| | | | 28-damaged\WE17 folder | 14 |

# Enclosure 11 to E-50037

# Responses to
# Request for Additional Information
# (Public Version)

## 4 – Thermal Evaluation

**1. Clarify the following statement in Section M.4.12.2.1, pp. M4-51f:**

For **off-normal condition**, with 8 FFCs in HLZC#2, the average temperature of helium is bounded by the limiting normal condition with ambient temperature of 100°F due to using sunshade for ambient temperature of 117°F.

It is not evident from the Final Safety Analysis Report (FSAR) text why this statement is accurate given than text in a previous part of this section states:

..., the normal transfer condition (100°F ambient) represents the bounding normal and **off-normal condition** for storage and transfer...

AND

Boundary conditions for the limiting case (100°F ambient with insolation) with HLZC #2 are taken from Section M.4.4.1.8.

The staff cannot discern which of these statement is complete and correct for the purposes of calculating the average temperature of the helium in the Dry Shielded Canister (DSC) cavity used for pressure calculations.

This information is needed to determine compliance with 10 CFR 72.236(l).

**RESPONSE TO RAI 4-1**

The limiting case for the normal and off-normal conditions of storage and transfer operations with intact fuel assemblies (FAs) in HLZC #2 is the normal transfer condition (100 °F ambient with insolation) based on the thermal results listed in Tables M.4-2 and M.4-8.

This limiting case for thermal evaluation with intact FAs in HLZC #2 is also the limiting case (normal transfer, 100 °F ambient with insolation) for thermal evaluation with failed FAs for both normal and off-normal conditions. Therefore, the average helium temperature calculated for the limiting case with 8 FFCs in HLZC #2 provides the bounding average helium temperature for both normal and off-normal conditions of storage and transfer with 8 FFCs in HLZC #2.

FSAR Section M.4.12.2.1 has been revised to provide the requested clarification discussed above. In addition, a typographical error has been corrected in the second paragraph of Section M.4.12.2.1 to change "32PTH" to "32PT."

**Application Impact:**

FSAR Section M.4.12.2.1 has been revised as described in the response.

**2. Provide computer files for the ANSYS FLUENT sensitivity study of the 32PTH1 Type 1 DSC HLZC#6.**

The ANSYS FLUENT computer files for the Standardized NUHOMS system were presented for the thermal analysis of the 32PTH1 Type 1 DSC with HLZC#5 and HLZC#6. Upon review of those files, the NRC staff was unable to review the results of 2 the files contained in the compressed directories as they appeared to be corrupted. Specifically, the NRC Staff was unable to execute file 'HSMH-32PTH1-TWOHALFFULL-NORMAL-8000.DAT properly.

This information is needed to determine compliance with 10 CFR 72.236(l).

**RESPONSE TO RAI 4-2**

The requested ANSYS FLUENT computer files for the 32PTH1 Type 1 Dry Shielded Canister (DSC) with the bounding storage condition have been re-submitted in Enclosure 10 (as listed in Enclosure 9) in response to this RAI.

**Application Impact:**

No changes as a result of this question.

**3. Provide additional FSAR text regarding the ANSYS FLUENT methodology used in performing the sensitivity analysis for HLZC#6.**

The text of the Standardized NUHOMS FSAR repeatedly references sections of the NUHOMS EOS System when presenting the CFD modeling methodology used. In particular, Sections U.4.11.1.3, Methodology and U4.11.1.5, CFD Modeling do not have enough information, along with missing ANSYS FLUENT computer files identified above, to make a safety finding.

When making an incorporation by reference, the information being incorporated should be clear and specific information with a narrative describing how that specific information applies to the current amendment. The applicant should provide this narrative which supports the conclusion that the method used in the NUHOMS EOS application is adequate to be used with the Standardized NUHOMS design to demonstrate that the peak cladding temperature and canister pressure remains within the design limits.

This information is needed to determine compliance with 10 CFR 72.236(l).

**RESPONSE TO RAI 4-3**

The computational fluid dynamics (CFD) methodology used in this application follows the same methodology as that used in Certificate of Compliance (CoC) No. 1042 for the NUHOMS® EOS System, which was recently approved by NRC. In addition to referencing the corresponding sections in NUHOMS® EOS application, more details of this CFD methodology have been added to each section. Specifically, FSAR Section U.4.11.1.3 for the overview of the methodology, Section U.4.11.1.4 for the computer-aided design and meshing, and Section U.4.11.1.5 for the CFD modeling have been expanded.

**Application Impact:**

FSAR Sections U.4.11.1.3, U.4.11.1.4, and U.4.11.1.5 have been revised as described in the response.

## 6 – Shielding Evaluation

**1. Clarify how the solar shield is used for As Low As Reasonably Achievable (ALARA) purposes.**

The applicant states on page 10-32 of the Safety Analysis Report (SAR):

"The solar shield while not required for transfer operations with ambient temperatures below 100 °F (106 °F for 32PTHJ) may also be used for ALARA purposes to protect the cask from rain or snow or other ambient conditions."

The ALARA principle is designed to minimize the dose to radiation workers and the general public. It is not clear to the staff how protecting the cask from rain or snow is related to ALARA.

**RESPONSE TO RAI 6-1**

The phrase "for ALARA purposes" will be removed from the cited statement in the Technical Specification bases in Section B 10.5.3.1 of the FSAR.

**Application Impact:**

FSAR Section B 10.5.3.1 has been revised as described in the response.

**2. Justify the proposed storage of up to 28 damaged fuel assemblies or 8 failed fuel cans is bounded by the 32PT Dry Shielded Canister (DSC) containing all intact fuel assemblies.**

On the following pages of the SAR, the applicant states:

Page M.1-1:

"The NUHOMS-32PT System is designed to accommodate up to 32 intact, up to 28 damaged, or up to 8 failed fuel cans, with characteristics as described in Chapter M2."

Page M.1-2:

"The NUHOMS®-32PT DSC system is designed to store intact and/or damaged and/or failed standard Pressurized Water Reactor (PWR) fuel assemblies with or without Control Components (CCs). The NUHOMS®-32PT DSC system is designed for a maximum heat load of 24 kW/canister and a maximum of 2.2 kW/assembly when heat load zoning is considered. The fuel which may be stored in the NUHOMS®-32PT DSC is presented in Section M.2. Provisions have been made for storage of up to 28 damaged fuel assemblies in lieu of an equal number of intact assemblies in cells, other than the four cells located at the center of the 32PT basket as described in Section M2. The DSC basket cells that store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability. Provisions have also been made for storage of up to 8 [failed fuel cans] FFCs in cells located at the outside corners compartment cells of the 32PT basket as described in Chapter M2."

Page M2.3a:

"The NUHOM System-32PT DSC is designed to accommodate up to a maximum of 8 failed fuel assemblies encapsulated in individual failed fuel cans (FFCs) and placed in cells located at the outer edge of the DSC as shown in Figure M2-2."

In the previous amendments 13 and 14 the maximum fuel assembly heat load was 1.2 KM/assembly and Maximum 24KW/canister without any damage fuel or failed fuel. In amendment 15, the 28 damaged fuels can be stored in 32 PT with 4 intact fuels in the center of the cask. Provide a detailed source term and dose rate calculation for the storage of up to 28 damaged fuels assemblies or up to 8 failed fuel cans stored in a 32PT cask.

The shielding analysis in the SAR does not include an analysis considering the potential for source redistribution for failed fuel, or any reference to the calculation package for failed fuel. The applicant needs to revise the SAR to include a shielding analysis for the 32PT cask with failed fuel configuration.

The staff needs this information to determine if the TN 32PT cask with the amended design meets the regulatory requirements of 10 CFR 72.236(a).

**RESPONSE TO RAI 6-2**

The NUHOMS®-32PT System is designed to store up to 32 intact, up to 28 damaged, or up to 8 failed fuel cans as described in FSAR Appendix M.2. The four heat load zone configurations (HLZCs) are shown in Figure 1-2 through Figure 1-4 of the Technical Specifications (TS), and the maximum decay heat per DSC is 24 kW.

The shielding analysis in FSAR Appendix M.5 is performed using design basis source terms derived from the intact bounding fuel B&W 15x15. The shielding analysis for HLZC #2 is performed assuming, conservatively, 16 fuel assemblies in the outer ring at 1.2 kW for a total heat load of 28.8 kW, as opposed to the 24 kW heat load limit. Furthermore, the shielding analysis for HLZC #4 assumes, conservatively, that the upper dry shielded canister (DSC) hemisphere heat load zones represent the entire DSC loading, for a total heat load of 31.2 kW, as opposed to the 24 kW heat load limit.

Damaged fuel is defined in Table 1-1e of the TS. Damaged fuels are fuel 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 in such a way that a fuel assembly is able to be handled by normal means. As such, since the source term is not impacted and axial redistribution is unlikely, damaged fuel has essentially no impact on the dose rate. Damaged fuels are allowed in the four HLZCs of the NUHOMS®-32PT.

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 is housed inside a FFC placed in designated positions. Failed fuel is authorized for loading into HLZC #2 only. The maximum heat load for failed fuel is 0.8 kW per storage location, as opposed to the 1.2 kW per storage location allowed for Zone 2 of HLZC #2. Therefore, the shielding evaluation for HLZC #2 containing failed fuel is bounded by modeling intact fuel in the failed fuel locations, assuming sixteen intact fuel assemblies at 1.2 kW in the peripheral positions, for a total heat load of 28.8 kW as mentioned above.

The shielding analysis performed in FSAR Appendix M.5 is appropriate for all the requested configurations.

**Application Impact:**

No changes as a result of this RAI.

3. **Provide justification for the various increases in the occupational exposure from the loading of 0.380 MTU fuel.**

In the SAR statements or Table's footnote the dose rates at different locations of the cask are increased by different scaling factors. There is no detail of how these factors are obtained. Provide the justification for all factors that are used in the SAR and effect of dose rate increase on site boundary dose rate.

The staff needs this information to determine the TN 32PT cask with the amended design meets the regulatory requirements of 10 CFR 72.236(a).

**RESPONSE TO RAI 6-3**

The response to this RAI is included in the response to RAI 6-5, since the content of both RAIs are related to the same underlying concept. Both RAIs address applying dose rate, site dose, and occupational exposure scaling factors to examine the impact on these reported values when loading 0.380 MTU FAs into the 24PTH, 32PTH1, 32PTH, and 37PTH PWR storage and transfer systems. Please refer to the response for RAI 6-5 to address the content of this RAI.

**Application Impact:**

No changes as a result of this RAI.

**4. Justify the applicability of the isotopes used in code benchmarking for shielding analysis.**

It is stated on Page P.5-7 in the SAR:

"Oak Ridge National Laboratory has benchmarked TRITON based on measured data from six different PWRs. This benchmarking is documented in NUREGICR-6968 [5.16}, NUREGICR-7012 [5.17], and NUREGICR-7013 [5.18} and includes measurement samples up to a burnup of 78.3 GWd/MTU. A summary of experimental samples utilized in the benchmark analysis is provided in Table P.5-27. The benchmark references show that TRITON computed results agree well with experiments, thus verifying the use of SCALE 6. O/ORIGEN-ARP to compute gamma and neutron source terms for high-burnup fuel (burnup ≤ 62 GW d/MTU). Because SAS2H and ORIG EN-ARP compute similar source terms for PWR fuel, the overall uncertainty in the gamma and neutron source terms developed above for SAS2H (±5% for gammas and ±11 % for neutrons) is also applicable to the SCALE 6.0/0RIGEN-ARP generated source terms."

The SAS2H is a diffusion theory code and TRITON is a transport theory code; therefore, benchmarking for TRITON code uncertainty is not applicable to SAS2H code uncertainty. The code benchmarking provided in NUREG/CR-7108 and NURG/CR-7109 are for isotopes considered for burnup credit. The group of isotopes for shielding source term is different than those considered for burnup credit. NUREG/CR-6700 provides ranking of isotopes that are important to the shielding and burnup credit. Provide a justification for applying the uncertainty from SAS2H code in the development of the source term.

The staff needs this information to determine the TN 32PT cask with the amended design meets the regulatory requirements of 10 CFR 72.236(a).

**RESPONSE TO RAI 6-4**

A review of NUREG/CR-6700 suggests that the use of an experimental database encompassing the following radionuclides would be appropriate for validating a computer code used for decay heat and radiation source calculations: Cm-244, Cm-246, Cs-134, Co-60, Eu-154, Pr-144/Ce-144, Rh-106/Ru-106, Ba-137m/Cs-137, Y-90/Sr-90, Pu-238, Pu-240, and Am-241.

FSAR Table P.5-27 provides the summary of the experimental samples as function of the burnup range for TRITON/ORIGEN-ARP. The samples are from six different pressurized water reactors (PWRs), and are documented in NUREG/CR-6968, NUREG/CR-7013, and NUREG/CR-7012. Table 6-4-1 provides the number of samples from each reactor, for each isotope important for shielding.

**Table 6-4-1**
**Number of Samples Used for Each Isotope**

| | Cm-242 | Cm-244 | Cs-134 | Cs-137 | Ce-144 | Eu-154 | Ru-106 | Sm-147 | Sr-90 |
|---|---|---|---|---|---|---|---|---|---|
| **Gosgen** | 4 | 6 | 6 | 6 | 6 | 6 | 6 | 6 | 6 |
| **GKN II** | 1 | 1 | - | - | 1 | 1 | - | 1 | - |
| **Takahama 3** | 16 | 16 | 16 | 16 | 16 | 16 | 16 | 6 | - |
| **Vandellos II** | - | 6 | 6 | 6 | 5 | 5 | 5 | 5 | - |
| **TMI** | 8 | 8 | 8 | 19 | - | - | - | 19 | - |
| **Calvert Cliffs** | - | - | 3 | 3 | - | 3 | - | 3 | 3 |

As shown in Table 6-4-1, the benchmark includes numerous isotopes important for shielding.

In addition, NUREG/CR-6969, documenting the SCALE/TRITON (TRITON with ENDF/B-V cross-section library and ORIGEN-S) benchmark based on quality radiochemical assay data from two international programs, ARIANE and REBUS, for PWR fuels encompassing high burnup fuel up to 60 GWd/MTU, shows that the important contributors to decay heat and radiation source are adequately predicted: Am-241, Cm-244, Cs-137 within a 5 to 6% range (Am-241 is about 29% range with REBUS), Eu-154 (over-predicted) and Cs-134 (under-predicted) within 8 to 9% range, while plutonium isotopes such as Pu-238, Pu-240 and Pu-242 are within 5% range compared to measurements.

NUREG/CR-7162 documents the analysis of experimental radiochemical assay data from modern boiling water reactor (BWR) assemblies. The computational benchmark is performed with TRITON and ENDF/B-VII cross-section library. Overall, the results are similar to those presented in NUREG/CR-6969.

"Analysis of isotopic assay data from the MALIBU program," ORNL paper for the International Conference on Reactor Physics, Switzerland, 2009, presents the isotopic analysis of two fuel segments at 47 and 68 GWd/MTU. The computational analysis of the measurements is performed with TRITON and ENDF/B-V cross-section library. Overall, the results are consistent with results obtained from other evaluations, with Am-241, Cm-244, Eu-154 within 10% range, Cs-134 slightly above 10%; Cs-137 and Sr-90 are well-predicted, as well as plutonium isotopes relevant to decay heat and radiation, except for Pu-238, which is under predicted by 10%.

Based on the information provided, there is sufficient documentation on SCALE TRITON/ORIGEN-ARP benchmarks and associated cross-section libraries to establish that TRITON/ORIGEN-ARP is appropriate for decay heat and source term calculations. SAR Section P.5.2 has been modified to incorporate the benchmarking elements discussed in this response, and to remove any reference made to SAS2H.

### Application Impact:

FSAR Section P.5.2 has been revised as described in the response.

**5. Provide justification for scaling factors used in the tables for loading 0.380 MTU FAs.**

There is no justification for the scaling factors. Provide justification for how you arrive at these scaling factors.

The staff needs this information to evaluate the dose rates of the TN amendment 15 request with respect to demonstration of meeting the regulatory requirements of 10 CFR 72.236(a).

**RESPONSE TO RAI 6-5**

Because the content of RAI 6-3 and RAI 6-5 is related to the same underlying concept, this response is written to address both RAIs. The concept involves applying dose rate, site dose, and occupational exposure scaling factors to examine the impact on these reported values when loading 0.380 metric ton uranium (MTU) fuel assemblies (FAs) into the 24PTH, 32PTH1, 32PT, and 37PTH pressurized water reactor (PWR) storage and transfer systems.

One of the objectives of Amendment 15 is to authorize the loading of 0.380 MTU FAs into the 24PTH, 32PTH1, 32PT, and 37PTH PWR storage and transfer systems. The shielding analyses performed for these systems verify the impact on the storage and transfer dose rates, occupational exposures, and site doses when compared to the currently authorized upper-bound MTU FAs.

It was concluded that, for loading 0.380 MTU FAs, reporting these new higher values as scaling factors in footnotes of the existing FSAR tables would ease readability, rather than duplicating tables for a second set of storage and transfer dose rates, occupational exposures, and site doses for 0.380 MTU/FA loading and for the currently authorized upper-bound MTU FAs.

The shielding analyses for 0.380 MTU/FA are documented in FSAR Section U.5.4.12 for the 32PTH1 System, Section P.5.4.11 for the 24PTH System, Section M.5.4.16 for the 32PT System, and Section Z.5.4.11 for the 37PTH System. These sections provide the details on the scaling factors, and the justification for their use when reporting the impact on storage and transfer dose rates, occupational exposures, and site doses when loading 0.380 MTU FAs into these PWR systems.

The content of FSAR Section U.5.4.12 for the 32PTH1 System have been modified to provide detail about the way in which these scaling factors are derived and applied to the various tables throughout the FSAR. Also, FSAR Section P.5.4.11 for the 24PTH System, Section M.5.4.16 for the 32PT System, and Section Z.5.4.11 for the 37PTH System have been modified to add that the scaling factors for these systems are derived and applied in the same manner as is described in Section U.5.4.12.

**Application Impact:**

FSAR Sections M.5.4.16, M.10.1, P.5.11, U.5.4.12, U.10.1, Z.5.4.11 and Z.10.1 have been revised as described in the response.

FSAR Tables M.10-1, M.10-2, U.5-27, U.10-1, and Z.10-1 have been revised as described in the response.

**6. Provide justification for using the 32PTH1 DSC response function for the 24 PTH DSC HLZC#6.**

Page P.5-15 of the SAR stated:

"Because the original FQTs have been replaced with unified FQTs (Technical Specifications Tables I-3a through I-3p), the design basis source terms developed in Section P.5.2 are obsolete because they are based upon burnup, enrichment, and cooling time combinations that are no longer applicable. Therefore, SCALE6. O/ORJGEN-ARP design basis source terms are developed based on the unified FQTs. The FQTs are documented in Section M5.2.6. The methodology used to develop the SCALE6. O/ORIGEN-ARP design basis source terms for HLZC#2 is the same as described in Section P.5.2. The ANISN transfer cask and HSM response functions developed in Section P.5.2.4 are used to evaluate the source terms for each FQT burnup, enrichment, and cooling time (BECT) combination. The BECT combination that results in the maximum dose rate is selected as the design basis source. For HLZC#6, because the heat load zone configuration is not uniform, the three-zone 32PTH1 DSC response functions from Chapter U5, Tables U5-15 and U5-16, are used. Because the response functions are only used to rank the BECT combinations, using the 32PTH1 DSC response functions for the 24PTH DSC HLZC#6 is acceptable."

However, the applicant provides no justification for the applicability of the response function to the 24 PTH DSC HLZC#6. Provide the justification for using the three zone response functions for 32 PTH1 in the Tables U5-15 and 16 for 24PTH DSC HLZC#6 or develop a new response function for this specific cask loading configuration.

The staff needs this information to determine the TN 32PT cask with the amended design meets the regulatory requirements of 10 CFR 72.236(a).

**RESPONSE TO RAI 6-6**

Response Functions are essentially matrix operators that convert source "vectors" to dose rate "vectors." Components of the source "vectors" are intensities of the source terms in designated discrete (for gamma radiation source) energy groups. Components of the dose rate "vectors" are dose rate values around the shielded package side. Therefore, items of the response function matrix represent "source to dose rate" conversion factors for a given package shielding and source configuration/geometry. Response Functions can be employed for various applications, in particular for source term ranking, i.e., for determination of bounding source for given range of BECT combinations.

The three-zone response functions for the 32PTH1 Dry Shielded Canister (DSC) in FSAR Table U.5-15 and Table U.5-16 are developed for the three zones shown in FSAR Figure U.5-3 and reproduced below as Figure 6-6-1.

Heat load zone configuration (HLZC) #6 for the 24PTH DSC provided in Technical Specifications (TS) Figure 1-15a and reproduced below as Figure 6-6-2, includes three zones: Zone 4 (peripheral zone), Zone 5 (corner and central zones) and Zone 6 (inner zone).

Note that both the 24PTH and 32PTH1 DSCs are transferred and stored in systems that possess essentially the same or very similar shielding properties.

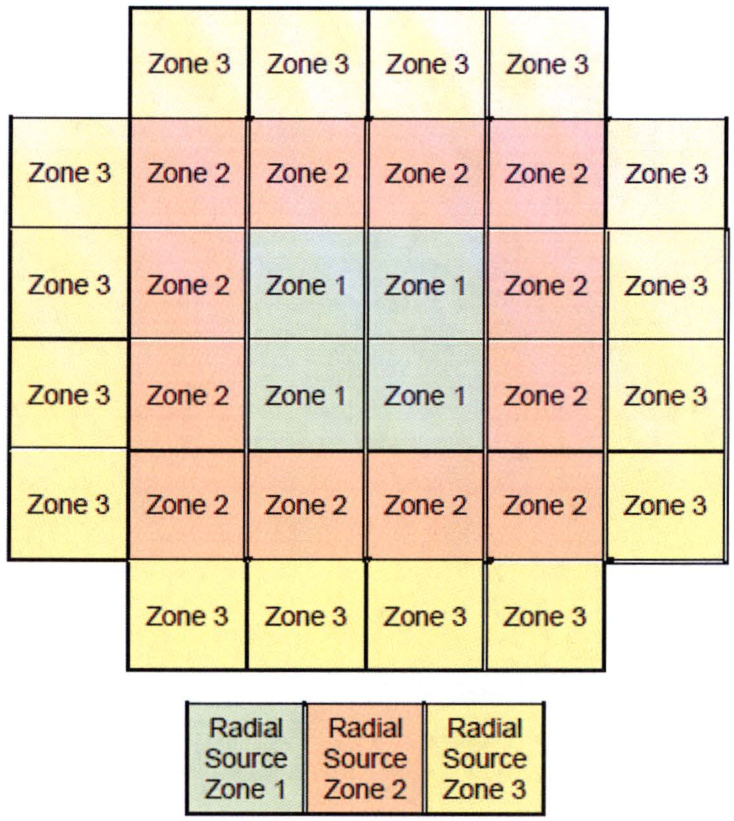
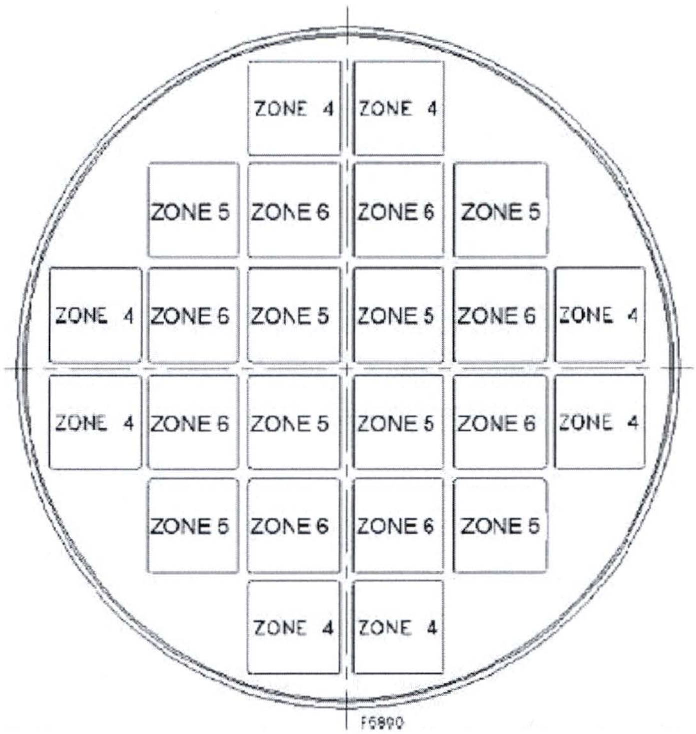



**Figure 6-6-1**
**Response Function Zoning**




| | Zone 4 | Zone 5 | Zone 6 |
|---|---|---|---|
| Maximum Decay Heat (kW/FA) | 1.3 | 0.6 | 2.5 |
| Maximum Decay Heat per Zone (kW) | 10.4 | 4.8 | 20.0 |

**Figure 6-6-2**
**HLZC#6 for 24PTH-S and 24PTH-L DSCs with Type 1 Basket**

**Application Impact:**

No changes as a result of this RAI.

**7. Provide justification for an increase in the peak roof dose rates.**

In the Page P.11-13 of the SAR stated that:

"P.11.2.3.4 Corrective Actions

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 1400 mrem/hour in Table P.5-1, since the structural analysis results demonstrate that there is no full penetration. This represents an increase in the peak roof dose rates by a factor greater than 20 and is conservative."

The applicant, however, provides no justification that the calculated 1400mrem/hour is bounding. The applicant needs to justify that using Horizontal Storage Module (HSM-H) vent dose rate bounds the roof top dose rate following a missile impact accident.

The staff needs this information to proceed with its review of the TN amendment 15 request with respect to the demonstration of meeting the regulatory requirements of 10 CFR 72.236(d).

**RESPONSE TO RAI 6-7**

The methodology used to assess the dose rate due to the missile impact accident was originally developed to address RAI 11-4 from Amendment 10 of CoC 1004 for the 61BTH DSC [1]. This methodology has since been used for other dry shielded canisters (DSCs) used with the horizontal storage module (HSM)-H, including 24PTH, 32PTH1, 37PTH, and 69BTH.

No explicit dose rate calculation due to the missile impact is performed because the walls of the HSM are not perforated and hence any dose rate at the area of impact should be bounded by dose rates from a vent, as the vent fully penetrates an HSM wall. The primary concern in the accident is not the HSM dose rate, which is not limited, but the exposure to an individual at the site boundary. When assessing the exposure to an individual at the site boundary, the dose rate of the entire 2x10 array of undamaged HSMs is used to represent the dose rate of a single damaged HSM. This approach is highly conservative because a missile impact that damages more than a single HSM is not credible, while a 2x10 array features approximately 20 front vents and 20 roof vents. As indicated in FSAR Section P.11.2.3.4, the exposure 100 m from the damaged HSM is conservatively estimated to be less than 5 mrem over the 8-hour recovery period. Because the exposure is small despite the large conservatisms employed, further refinement of the method is not warranted.

FSAR Sections P.11.2.3.3 and P.11.2.3.4 (24PTH), T.11.2.3.3 (61BTH), U.11.2.3.3 (32PTH1), and Z.11.2.3.3 (37PTH), have been modified to provide clarification and to more clearly highlight the conservatism of the method.

**Reference:**

[1] E-25506, Letter from Robert Grubb to the U.S. Nuclear Regulatory Commission, "Revision 1 to Transnuclear, Inc. (TN) Application for Amendment 10 to the Standardized NUHOMS® System (Docket No. 72-1004; TAC NO. L24052)," November 7, 2007 (ML073180235).

**Application Impact:**

FSAR Sections P.11.2.3.3 and P.11.2.3.4, T.11.2.3.3, U.11.2.3.3, and Z.11.2.3.3 have been modified as described in the response.

**8. Provide the source terms and shielding calculation for the two new fuels, GNF-2 and ATRIM 11 and provide justification for Table T.2-3 and T.2-4 footnote.**

In Table T.2-3:

"For ATRIUM 11 fuel assemblies, the U-235 wt. % enrichment is reduced by 0.55%."

And Table T.2-4:

"ATRIUM 11 fuel assemblies are authorized for storage only in the Type 2F basket with a maximum of 4 damaged intact fuel assemblies.

For ATRIUM 11 fuel assemblies, the U-235 wt. % enrichment is reduced by 0.55%."

The two new fuels GNF-2 and ATRIM-11 are added as authorizes content. There is no justification that if these fuels are bounded with any previous fuels authorized as content of 61BTH cask system. Provide the source term and shielding calculations for these new fuels, or if they bounded with any previous authorized contents. , Justify the factors in the footnotes of the Tables T.2.3 and 4 that used for ATRIUM-11 fuels.

The staff needs this information to determine the TN 32PT cask with the amended design meets the regulatory requirements of 10 CFR 72.236(a).

**RESPONSE TO RAI 6-8**

Radiological sources for the NUHOMS® 61BTH System are calculated using a geometry and material model described in the UFSAR Section T.5-2, as a "generic" design of a boiling water reactor (BWR) assembly. This "generic" fuel assembly shares many common features with the GE-2,3 7x7 Type G2A assembly. 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. The shielding analysis results show that the dose rates near the horizontal storage module (HSM) are almost entirely due to gamma radiation. Further, per Table T.5-4 of the UFSAR, more than 60% of the dose rates on the exterior surfaces of the transfer cask at normal conditions are also due to gamma radiation. Therefore, the cobalt content in the hardware materials of the BWR fuel assembly is a critical parameter for calculating bounding dose rates. Based on the information provided in UFSAR Table T.5-6 and Table T.5-7, the cobalt content for the "generic" BWR fuel assembly is calculated to be 16 grams per assembly.

The cobalt content in hardware materials presented in Table T.5-7 of the UFSAR is based on information from [1 and 2]. These references provide data for assemblies that were manufactured prior to the cobalt control (cobalt content reduction) program in materials of light water reactor (LWR) fuel assemblies. Fuel assemblies manufactured after the 1990s have lower cobalt content in their materials that what is employed herein as described in [3]. GNF-2 and ATRIUM 11 are newer assembly designs, and have lower net cobalt content in comparison to the "generic" BWR assembly. Using a 500 ppm cobalt content in steel and 1000 ppm cobalt content in Inconel, the total cobalt content for these fuel assemblies is calculated to be 8 grams per assembly. Therefore, the source terms and consequently, the dose rates calculated using the "generic" BWR fuel design remain bounding for GNF-2 and ATRIUM 11 assembly designs.

[1] U.S. Department of Energy [(*)], "Characteristics of Potential Repository Wastes," Office of Civilian Radioactive Waste Management, DOE/RW-0184-R1, Volume 1, ORNL, July 1992.

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

[3] Karl-Heinz Neeb[(‡)], "The radiochemistry of nuclear power plants with light water reactors," Berlin, New York: de Grueter, 1997. ISBN 3-11-013242-7.

Footnotes to FSAR Table T.2.3 and Table T.2.4 for the ATRIUM 11 fuel are based on the criticality requirement, see FSAR Appendix T.6.

**Application Impact:**

No changes as a result of this RAI.

[*] Cobalt content of hardware materials and elemental composition of various materials can be found in Section 2.7.2 on page 118 and Table 2.7.3 on page 123, respectively.

[†] This reference provides a composition of various materials in LWR reactors.

[‡] This reference provides data on Co-60 content in materials of fuel assemblies and nuclear reactors. In particular, it provides noticeably lower cobalt content in the fuel assemblies hardware than Reference [2]. Look for example at page 364, Section 4.4.4.5 and page 317-318.

**7 - Criticality Evaluation**

1. **Revise the Safety Analysis Report (SAR) to clarify if Figure M.1-2 is applicable to both $B_4C$ Poison Rod Assemblies (PRAs) and silver-indium-cadmium (AIC) PRAs, or $B_4C$ PRAs only. If Figure M.1-2 is applicable only to $B_4C$ PRAs, provide a separate figure showing the arrangement of AIC PRAs. In either case, provide a specification for the minimum AIC absorber material rod diameter.**

Figure M.1-2 shows the layout and dimensions for PRAs to be used in the 32PT Dry Shielded Canister (DSC). However, it is not clear if this figure applies to $B_4C$ PRAs only, or to both $B_4C$ and AIC PRAs. The Figure should be revised to indicate which PRA types it is applicable to. If it is not intended to be applicable to the AIC PRA, a separate figure should be provided. In either case, the minimum AIC absorber material rod diameter should be specified. For $B_4C$ PRAs, the minimum absorber material rod diameter is defined by the rod outer diameter and clad thickness dimensions, as the absorber material is $B_4C$ powder which fills the entire clad interior volume. The AIC absorber alloy is a solid material, however, which will necessarily have a smaller diameter than that defined by the clad dimensions. The minimum diameter of AIC absorber material should be limited to the dimension shown to maintain the cask subcritical in the criticality analysis.

This information is needed to ensure that the Standardized NUHOMS® 32PT DSC will continue to meet the criticality safety requirements of 10 CFR 72.124.RESPONSE TO RAI 8-1

As noted by the staff, EPRI-1000975, Boric Acid Corrosion Guidebook, Revision 1 observes that the test results obtained from the Moscow Power Institute (1970) vary as a function of test duration in such a way that the average corrosion rate is higher for short test durations.

The TC and DSC are only kept in the spent fuel pool for a short period of time, typically less than 24 hours (see Safety Analysis Report (SAR) Section 8.1.2). Therefore, selecting an average corrosion rate of 0.008 mm/y resulting from a 1000-hour test may not have been conservative in the previous evaluation presented in SAR Section 8.2.5.3.

Safety Analysis Report Section 8.2.5.3 has been revised to provide an updated assessment of the corrosion based on the relevant conditions for use of the system and the data provided in EPRI -1000975, Boric Acid Corrosion Guidebook, Revision 1. Table 8-36 has been added to provide the relevant corrosion rates taken from EPRI-1000975. Safety Analysis Report Section 8.7 has also been revised to replace NUREG/CR-6875 with EPRI-1000975 for reference 8-42. The updated corrosion assessment concludes again that the thickness reduction due to corrosion does not impact the structural performance.

**RESPONSE TO RAI 7-1**

The purpose of FSAR Figure M.1-2 is to display the poison rods design that could be used for additional criticality control, along with geometry, required soluble boron in spent fuel pool, and fixed neutron poison material in the fuel basket. Figure M.1-2 illustrates both $B_4C$ poison rods and silver-indium-cadmium (AIC) poison rods. While the dimensions shown in Figure M.1-2 illustrate the poison rods design, the critical requirements for the two poison rods designs are the minimum $B_4C$ content per rod for the $B_4C$ poison rod assembly (PRA) (in g/cm) and the minimum silver content per rod for the AIC PRA (in g/cm) as shown, respectively, in FSAR Table M.2-4 and Table M.2-4a. The title of FSAR Figure M.1-2 has been revised for clarity in response to this RAI.

**Application Impact:**

FSAR Figure M.1-2 has been revised as described in the response.

**2. Clarify the enrichment limits for damaged and failed fuel in Tables 1-1g2 and 1-1g3 of the Technical Specifications.**

Tables 1-1g2 and 1-1g3 of the Technical Specifications for the 32PT DSC provide enrichment limits for damaged and failed fuel, but do not specify if these limits are only for damaged or failed fuel, or for all fuel, including intact fuel, loaded in the same DSC with damaged or failed fuel. These tables should be modified to clarify which assemblies these enrichment limits apply to. If the limit is different for the damaged or intact fuel loaded in the same DSC as failed fuel, or intact fuel loaded in the same DSC as damaged fuel, modify the Technical Specification tables to specify the enrichment limits appropriately.

This information is needed to ensure that the Standardized NUHOMS® 32PT DSC will continue to meet the criticality safety requirements of 10 CFR 72.124.

**RESPONSE TO RAI 7-2**

Technical Specifications Table 1-1g2 and Table 1-1g3 for the 32PT dry shielded canister (DSC) have been updated to clarify that they provide the enrichment limits for all the fuel assemblies: intact or damaged or failed.

**Application Impact:**

TS Tables 1-1g2 and 1-1g3 have been revised as described in the response.

3. **Clarify the amount of credit taken in the 32PT DSC criticality analysis for the silver content in the AIC PRAs, in relation to the amount of silver required per Note 3 of Table 1-1h of the Technical Specifications.**

Section M.6 of the SAR states that 40% of the actual silver content is credited as being present, with 75% of that credited in the criticality analysis. However, Note 3 of Table 1-1h of the Technical Specifications specifies 2.46 grams of silver per centimeter of absorber rod in the AIC PRA. The footnote to Table M.2-4a of the SAR states that 75% of that value is credited in the criticality analysis. It is not clear how the minimum 2.46 grams per centimeter of absorber rod is determined, since it appears that, based on the dimensions in Figure M.1-2, this value represents roughly 44% of the density of elemental silver (at 10.49 g/cm3), or roughly 56% of the density of silver in typical 80% silver, 15% indium, 5% cadmium AIC alloys. Revise the SAR to state the expected density of absorber material in the AIC PRA, as well as the minimum diameter of absorber material in the rods of the AIC PRA. Note that conservatism in the modeling of AIC material is used to account for the lack of representative benchmark experiments with AIC material in the computer code validation analysis.

This information is needed to ensure that the Standardized NUHOMS® 32PT DSC will continue to meet the criticality safety requirements of 10 CFR 72.124.

**RESPONSE TO RAI 7-3**

The critical requirement for the silver-indium-cadmium (AIC) poison rod assembly (PRA) is the minimum linear mass of the silver content, Ag, per rod, 2.46 g/cm, as shown in Table 1-1h of the Technical Specifications (TS) or FSAR Table M.2-4a.

The initial AIC composition is Ag 80 wt.%, In 15 wt.%, and Cd 5 wt.% and its density is 10.17 g/cm$^3$. Given that the poison rod radius modeled as 0.4902 cm in the KENO model, the initial linear mass of Ag is 6.14 g/cm per rod of AIC. 40% of the initial linear mass of Ag, or 2.46 g/cm, is required (see Table 1-1h of the TS or Table M.2-4a of FSAR Appendix M.2. 75% of the required amount of Ag is credited in the KENO model for criticality analysis, i.e., 1.85 g/cm, as indicated in Note (1) of Table M.2-4a.

A sensitivity analysis performed with a smaller poison pellet dimension using the AIC Diameter Control Rod specification for WE 17x17 fuel assembly is shown in Table 4 of NUREG/CR-6759.

The sensitivity analysis shows that the required 2.46 g/cm initial linear mass of Ag in AIC is appropriate for the criticality safety of the 32PT DSC when an AIC rod cluster control assembly is credited for the configurations shown in Table M.6-1 (Part 3 of 3) of FSAR Appendix M.6.

FSAR Section M.6.4.4 has been revised to incorporate the sensitivity analysis information described in this response.

**Application Impact:**

FSAR Section M.6.4.4 has been revised as described in the response.

**4. Revise the application to demonstrate that the CE 15x15 fuel class is bounded by the WE 17x17 fuel class for damaged and failed fuel assemblies in the 32PT DSC.**

Section M.6.4.4 of the SAR states that no explicit calculations are performed for the CE 15x15 fuel class for damaged or failed fuel configurations. The results from the WE 17x17 fuel class are conservatively applied for this purpose. However, the SAR does not demonstrate that the WE 17x17 fuel class bounds the CE 15x15 fuel class for these configurations. The application should be revised to supply this demonstration.

This information is needed to ensure that the Standardized NUHOMS® 32PT DSC will continue to meet the criticality safety requirements of 10 CFR 72.124.

**RESPONSE TO RAI 7-4**

Additional analyses have been performed to demonstrate that the CE 15x15 fuel class is bounded by the WE 17x17 fuel class for damaged and failed fuel assemblies, i.e., the maximum planar average enrichments determined for WE 17X17 in damaged fuel and failed fuel configurations are applicable to the CE 15X15 fuel class. Final Safety Analysis Report (FSAR) Sections M.6.4.2 and M.6.4.4 have been revised to delete the statements that indicate that no explicit calculations are performed for the CE 15x15 fuel class for damaged or failed fuel configurations. Instead, results have been provided as indicated in the description below.

For damaged fuel assemblies (FAs), the criticality analysis result shows that the WE 17X17 damaged fuel is bounding, because the maximum $k_{eff}$ for WE 17X17 is 0.9386, while the maximum $k_{eff}$ for CE 15X15 is 0.9371.

For failed FAs, results of the criticality sensitivity analysis show that the enrichments determined for WE17 failed fuel are applicable to CE15 failed fuel since the maximum $k_{eff}$ for the two fuel types are statistically equivalent, 0.9345 for WE17 and 0.9349 for CE15.

These analyses demonstrate that the maximum planar average enrichments determined for the WE 17X17 fuel class for damaged and failed fuel loading configurations are applicable to the CE 15X15 fuel class.

**Application Impact:**

FSAR Sections M.6.4.2 and M.6.4.4 have been revised as described in the response.

FSAR Tables M.6-61 and M.6-72 have been revised as described in the response.

5. **Revise the failed fuel evaluation for the 32PT DSC in the SAR to consider smaller array sizes of fuel, representative of less than a full fuel assembly. Also, revise Technical Specifications Table 1-1g3 to specify enrichment limits in terms of the most limiting array size.**

The failed fuel evaluation considers several array sizes of unclad fuel rods at optimum pitch. However, the failed fuel canister may contain any amount of fuel material up to the mass of a full fuel assembly, meaning that there may be much less material than evaluated. The system may be more reactive with a smaller array of less fissile material, at optimum pitch. Also, Technical Specifications Table 1-1g3 contains enrichment limits for fuel types based on the particular array sizes evaluated in the application. Since failed fuel has no requirements to be in or maintain a particular fuel geometry, it is unclear how a cask user would determine the array size for failed fuel. The enrichment limits for damaged fuel should be revised to be expressed in terms of a single, most reactive array size, which would not need to be verified prior to loading.

This information is needed to ensure that the Standardized NUHOMS® 32PT DSC will continue to meet the criticality safety requirements of 10 CFR 72.124.

**RESPONSE TO RAI 7-5**

A comprehensive evaluation of PWR most reactive failed fuel assembly (FA) configuration was performed and documented in FSAR Section P.6.4.2(G) in support of failed fuel criticality analysis for the 24PTH DSC in Amendment 13. The determination of the most reactive failed FA configurations included the most reactive rod pitch study, and the most reactive lattice configurations, including reduced lattices for BW 15x15, CE14x14, CE 15x15x15, CE16x16, WE14x14, WE15x15 and WE 17x17 fuel classes, and rod pitch with deletion of rods. The study concluded that, due to presence of soluble boron in the moderator, the maximum reactivity occurred for reduced lattices similarly to nominal lattices when empty rod locations are replaced with fuel rods (see FSAR Table P.6-30). Additional sensitivity analyses have been performed to demonstrate that the above conclusion remains applicable for the 32PT DSC.

An additional analysis has been performed for 32 failed WE 17x17 FAs at 4.40 wt. % enrichment in a 32-poison plate Type A1 DSC and with 2500 ppm of soluble boron in the moderator. The analysis includes a case with a full lattice of 17x17 rods and a case with a reduced lattice of 16x16 rods; both cases are performed with and without removal of rods. The results for the two cases are presented in new FSAR Table M.6-74 and described in revised FSAR Section M.6.4.2. The results show the maximum reactivity occurs for the reduced lattice (16x16) similar to the nominal lattice (17x17) when empty rod locations are replaced with fuel rods.

The most reactive failed fuel configuration used in FSAR Section M.6.4.4 for determining the maximum planar average enrichments for Type A1 and Type A2 DSCs with 8 failed FAs is appropriate.

In addition, it is clarified that the failed fuel configurations authorized for the 32PT DSC are not based on generic rod arrays but applicable to specific fuel assembly classes that are specified to intact fuel assemblies.

FSAR Table M.6-62 has been revised to indicate that the 16x16 and 17x17 lattice results are not applicable to the CE 15x15 fuel class. Table 1-1g3 of the TS has been revised to provide the maximum planar average enrichments for authorized fuel class loadings with eight failed fuel assemblies.

In addition, TS Tables 1-1g, 1-1g1, 1-1g2, and 1-1g3 have been revised to clarify the control component (CC) configurations consistent with the analysis.

**Application Impact:**

FSAR Section M.6.4.2 and Table M.6-62 have been revised as described in the response.

FSAR Table M.6-74 has been added as described in the response.

TS Tables 1-1g, 1-1g1, 1-1g2, and 1-1g3 have been revised as described in the response.

**6. Revise the SAR and Technical Specifications, as necessary, to clarify if failed fuel canisters may be stored in a 32PT DSC with both intact and damaged fuel. If failed fuel may be stored with both intact and damaged fuel, provide an analysis with a 32PT DSC containing both failed and damaged fuel to demonstrate that this configuration remains subcritical, and clarify which enrichment limits apply to this configuration.**

Table 1-1e of the Technical Specifications states that up to eight (8) failed fuel assemblies may be stored, with the remainder intact and/or damaged fuel assemblies, empty slots, or dummy assemblies. However, the criticality analysis in Section M.6 of the SAR only evaluates damaged fuel or failed fuel, without evaluating a mixture of both. Additionally, it is not clear which enrichment limit applies for the situation of mixed intact, damaged, and failed fuel. The SAR and Technical Specifications should be revised to clarify these enrichment limits.

This information is needed to ensure that the Standardized NUHOMS® 32PT DSC will continue to meet the criticality safety requirements of 10 CFR 72.124.

**RESPONSE TO RAI 7-6**

A maximum of eight failed fuel assemblies may be stored in a 32PT dry shielded canister (DSC), for which the balance may be intact fuels and/or damaged fuels, empty slots or dummy assemblies, as shown in Figure 1-4b of the Technical Specifications (TS).

Table 1-1g through Table 1-1g3 of the TS show the maximum planar average initial enrichments for various combinations of intact, or intact and damaged, or intact and failed fuels loadings.

When intact and damaged and failed fuels are loaded per TS Figure 1-4b, the maximum enrichments for all the fuel types are restricted with the lowest enrichment shown in Table 1-1g1 through Table 1-1g3, considering the same poison plate configuration and soluble boron loading.

For example, for WE 17x17 fuel, 24 poison plates, Type A2 basket, up to eight failed fuel assemblies with 17x17 rods, and 2500 ppm soluble boron, the maximum enrichment of 4.15% is applied to all the fuels if intact or damaged or failed fuels are loaded (4.15% being the lowest enrichment across Table 1-1g1, Table 1-1g2 and Table 1-1g3 for the WE 17x17 fuel, 24 poison plates, Type A2 for the loading configurations stated above).

FSAR Section M.6.4.4 and TS Table 1-1g3 have been revised to clarify information about enrichment limits.

**Application Impact:**

FSAR Section M.6.4.4. has been revised as described in the response.

TS Table 1-1g3 has been revised as described in the response.

7. **Revise the Technical Specifications for the 61BTH DSC to limit GNF2 fuel to the Type 1 and 2 DSC with A, C, and F basket types evaluated in the criticality analysis, or revise the criticality analysis to demonstrate that the Type 1 and 2 DSCs with B, D, and E basket types loaded with GNF2 fuel will be subcritical.**

Section T.6 of the SAR states that GNF2 fuel criticality evaluations are performed for six configurations - Type 1 and 2 DSCs with A, C, and F basket types - to demonstrate that GE12 fuel assembly enrichment limits are applicable to GNF2 fuel. However, the Technical Specifications appear to show that the GE12 enrichment limits also apply to GNF2 fuel in Type 1 and 2 DSCs with B, D, and E basket types. The SAR should be revised to provide an analysis demonstrating that the GE12 enrichment limits are applicable to GNF2 fuel in Type 1 and 2 DSCs with B, D, and E basket types, or the Technical Specifications should be revised to limit GNF2 fuel to Type 1 and 2 DSCs with A, C, and F basket types.

This information is needed to ensure that the Standardized NUHOMS® 32PT DSC will continue to meet the criticality safety requirements of 10 CFR 72.124.

**RESPONSE TO RAI 7-7**

The maximum lattice average enrichments for the NUHOMS® 61BTH dry shielded canister (DSC) as a function of the minimum B-10 requirements (Table 1-1v, Table 1-1w, Table 1-1w1 and Table 1-1x of the Technical Specifications), are determined using GE12 10x10 as the most reactive fuel lattice (see Section T.6.4.2(A) of the FSAR). The criticality analysis for the GNF2 fuel assembly is performed in Appendix T.6 as a sensitivity to the GE12 10x10 fuel type using three DSC basket types - Types 1 and 2 with A, C, and F poison loadings.

Note that the Type F DSC is selected because it allows the maximum enrichment loading, 5 wt%. The Type A DSC is selected because it is the lowest poison loading, and the Type C DSC is analyzed as an additional data point between the Type A and Type F DSCs.

The analyses performed for the three DSC basket-type poison loadings show that the maximum enrichments determined for the GE12 10x10 fuel (intact, damaged, and failed loadings) are applicable to the GNF2 fuel assembly. These analyses demonstrate that the GNF2 fuel is similar in reactivity to the GE12 10x10 fuel; hence, the maximum enrichments determined for the GE12 10x10 fuel for the DSC basket-type poison loadings B, D and E are also applicable to the GNF2 fuel.

Clarification has been added to the description for the most reactive configuration for intact, damaged and failed fuel in FSAR Appendix T, Sections T.6.4.2(B), T.6.4.2(C) and T.6.4.2(D).

**Application Impact:**

FSAR Section T.6.4.2 has been revised as described in the response.

## 8 - Materials Evaluation

1. **Provide the drawings for the damaged fuel top and bottom end caps. Change #4 indicates that damaged fuel assemblies in the 32PT Dry Shielded Canister (DSC) are confined within top and bottom end caps and failed fuel assemblies loaded within individual failed fuel canisters. The amendment request included drawings for the transportable canister for pressurized water reactor (PWR) Fuel Failed Fuel can assembly (NUH32PT-1007-SAR). The drawings for the end caps were not provided.**

   This information is necessary to assure compliance with 10 CFR 72.236(b) and 10 CFR 72.236(g).

   **RESPONSE TO RAI 8-1**

   Final Safety Analysis Report (FSAR) Drawing NUH32PT-1008 SAR for the NUHOMS® - 32PT dry shielded canister (DSC) damaged fuel top and bottom end caps was added to the drawings in FSAR Section M.1.5, as requested.

   **Application Impact:**

   FSAR Drawing NUH32PT-1008 SAR has been added as described in the response.

**2. Provide updated Final Safety Analysis Report (USFAR) change pages for Appendix M.8 Operating Systems that include a description of the loading and unloading operations for the 32PT DSC loaded with damaged fuel.**

This information is necessary to assure compliance with 10 CFR 72.236(g).

**RESPONSE TO RAI 8-2**

Final Safety Analysis Report (FSAR) Appendix M, Chapter M.8 - Operating Systems Procedures (Sections M.8.1.1, M.8.1.2, and M.8.2.2) have been revised to provide a description of the loading and unloading operating procedures for damaged and failed fuel assemblies for the 32PT DSC, as applicable.

**Application Impact:**

FSAR Sections M.8.1.1, M.8.1.2, and M.8.2.2 have been revised as described in the response.

**3. Update the Technical Specifications Table 1-1e to include the following information in the definition of Fuel Damage: 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 ensured following normal and off-normal conditions.**

The applicant provided this description of damaged fuel for the 32PT DSC in the UFSAR change pages in section M2.1 Spent fuel to be stored.

This information is necessary to assure compliance with 10 CFR 72.236(g).

**RESPONSE TO RAI 8-3**

As requested, Technical Specification Table 1-1e has been revised to clarify the definition of Fuel Damage by adding the following information on fuel rod damage:

"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."

Similar changes have been made to the definition of fuel damage in TS Tables 1-1i, 1-1j, 1-1l, 1-1t, 1-1aa, 1-1gg and 1-1ll.

In addition, the definition of fuel damage has been clarified for other DSCs in the FSAR by adding the sentence above to FSAR Sections N.2.1, U.1.1, U.2.1, T.1.1, Y.2.1 and Z.2.1.

**Application Impact:**

FSAR Sections N.2.1, U.1.1, U.2.1, T.1.1, Y.2.1 and Z.2.1 have been revised as described in the response.

TS Tables 1-1e, 1-1i, 1-1j, 1-1l, 1-1t, 1-1aa, 1-1gg and 1-1ll have been revised as described in the response.

**4. Provide additional information to clarify the handling of the damaged fuel using normal means. Specifically address the potential for fuel fragments smaller than the size of a pellet to be released from damaged fuel rods, during handling and retrieval operations. compliance with the criticality, shielding, thermal, and structural requirements, and normal handling and retrieval from the cask.**

The applicant stated:

> *Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks. 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 ensured following normal and off-normal conditions.*

The staff note that the applicants approach to storing damaged fuel is not consistent with the guidance in NUREG-1536 Revision 1 Section 8.6 C. Canning of Damaged Fuel which states that the purpose of a can designed for damaged fuel is to (1) confine gross fuel particles, debris, or damaged assemblies to a known volume within the cask; (2) to demonstrate that compliance with the criticality, shielding, thermal, and structural requirements are met; and (3) permit normal handling and retrieval from the cask.

This information is necessary to assure compliance with 10 CFR 72.236(g).

**RESPONSE TO RAI 8-4**

As limited by their definition, damaged fuel assemblies are required to be handled by normal means and are confined to their respective compartments by means of top and bottom end caps. Damaged fuel assemblies do not contain sheared fuel rods or contain missing major sub-components like top and bottom nozzles that impact their ability to be handled by normal means during loading. Normal handling refers to the use of the crane and grapple to handle and load damaged fuel assemblies.

From the standpoint of NUREG 1536 Revision 1, the damaged fuel assemblies for the NUHOMS® -32PT are more similar to the "undamaged" fuel assemblies where their geometry is still in the form of intact bundles. For completeness, the "failed" fuel assemblies for the NUHOMS® -32PT are more similar to the "damaged" fuel assemblies per NUREG 1536 Revision 1.

The fuel compartment and the top and bottom end caps together form the "acceptable alternative, per NUREG-1536 Revision 1" for confinement of damaged fuel assemblies. If fuel particles smaller than a pellet are released from the damaged assembly, the top and bottom end caps provide for the confinement of gross fuel particles to a known volume. The bottom end cap is designed to be removable and contains socket for the use of a handling tool. The gross fuel particles can be retrieved from the bottom end cap.

The structural analysis for damaged fuel cladding described in Appendix M.3 demonstrates that the cladding does not undergo additional degradation under normal and off-normal conditions of storage. The criticality analysis described in Appendix M.6 limits the allowable contents for damaged fuel assemblies based on worst case geometry and material reconfigurations. The shielding analysis described in Appendix M.5 demonstrates that the worst configuration for damaged fuel assemblies is bounded by that of the intact fuel assemblies.

Updated Final Safety Analysis Report (UFSAR) Appendix M, Section M.2.1 has been revised to provide this clarification on damaged fuel safety analyses.

In addition, UFSAR Appendix M, Section M.3.5 has been revised to clarify that the retrievability of damaged fuel is ensured under both normal and off-normal loads.

**Application Impact:**

UFSAR Section M.2.1 and Section M.3.5 have been revised as described in the response.