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Changed Pages for NUHOMS®-MP197 Safety Analysis Report, Revision 17A (Non-Proprietary)

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NUHOMS®-MP197 TRANSPORTATION PACKAGE SAFETY ANALYSIS REPORT

Revision *17A*

August 2016

AREVA TN

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Revision Log

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0	4/2001	Original Issue for CoC Revision 0
1	1/2002	Various Changes CoC Revision 0
2	2/2002	Various Changes CoC Revision 0
3	4/2002	Various Changes CoC Revision 0
4	5/2002	Various Changes CoC Revision 0
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6	6/2009	Various Changes for CoC Revision 3
7	4/2010	Various Changes for CoC Revision 3
8	7/2010	Various Changes for CoC Revision 3
9	3/2011	Various Changes for CoC Revision 3
10	8/2011	Consolidated SAR Submittal for CoC Revision 3
11	9/2011	Various Changes for CoC Revision 4
12	2/2012	Application for CoC Revision 7
13	8/2012	Various Changes for CoC Revision 7
14	8/2013	Various Changes for CoC Revision 7
15	1/2014	Various Changes for CoC Revision 7
16	3/2014	Various Changes for CoC Revision 7
17	4/2014	Various Changes for CoC Revision 7
17A	8/2016	Application for CoC Revision 8 Revised pages as follows: SAR pages A.1.4.8-3a, A.1.4.8.13a, A.1.4.8-14, A.1.4.9-i, A.1.4.9-2, A.1.4.9-2a, A.1.4.9-3, A.1.4.9-9, A.1.4.9-9a, A.1.4.9-10, A.1.4.10-1, A.1.4.10-6 SAR pages A.2.13.11-ii, A.2.13.11-1 through A.2.13.11-6, A.2.13.11-12, A.2.13.11-13, A.2.13.11-18, A.2.13.11-22 through A.2.13.11-24, A.2.13.11-31, A.2.13.11-41, A.2.13.11-43 SAR pages A.3-i through A.3-v, A.3-1, A.3-3, A.3-4, A.3-6, A.3-7, A.3-28, A.3-55, A.3-62, A.3-77, A.3-78, A.3-80, A.3-81, A.3-97, A.3-98, A.3-99, A.3-102, A.3-175, A.3-176 SAR pages A.5-i through A.5-v, A.5-1, A.5-2, A.5-3, A.5-5, A.5-7, A.5-8, A.5-11, A.5-12d, A.5-20b, A.5-29a, A.5.-29c, A.5-29d through A.5-29h, A.5-32b, A.5-34, A.5-37, A.5-38, A.5-40, A.5-41, A.5-41a, A.5-75, A.5-80l, A.5-80m, A.5-80p, A.5-80q, A.5-80t, A.5-105

Revision Log

Rev. No.	*Date*	*Description*
		Revised drawings as follows: *Drawing MP197HB-71-1005* *Drawing NUH69BTH-71-1004* *New pages as follows:* *Proprietary Information Notice* *Revision Log* *SAR pages A.1.4.9-9b through A.1.4.9-9g, A.1.4.9-15a* *SAR pages A.3-60a, A.3-102a, A.3-102b* *SAR pages A.5-2a, A.5-38a, A.5-41b, A.5-41c, A.5-42a, A.5-79b, A.5-102a*

Where,

Δt = cooling time to be added to the appropriate value in Table A.1.4.8-6 or Table A.1.4.8-7 (years)
m = the uranium loading of the fuel assembly to be shipped (kg), and
B = additional cooling time (years), obtain from Table A.1.4.8-7a

This equation applies only if the "B" value is defined in Table A.1.4.8-7a. Additional cooling is not required for burnup and enrichment combinations that fall in the zone labeled "no additional cooling time required." Therefore, no additional cooling is required for burnups < *49* GWd/MTU.

This equation can return a negative value for Δt, which indicates that no additional cooling is required. In such a scenario, $\Delta t \equiv 0$ years. Application of the method is illustrated in the following examples.

Example #1: Fuel with a burnup of 62 GWd/MTU, enrichment of 2.6 wt. % U-235, and uranium loading of 198 kgU is to be loaded in the peripheral zone of the Type 1 DSC. From Table A.1.4.8-6, the minimum cooling time is 29.5 years. From the equation above, $\Delta t = 26.13*\ln(1) + 10.5 = 10.5$ years. Therefore, the minimum required cooling time for this fuel assembly in the peripheral zone is $29.5+10.5 = 40.0$ years.

Example #2: Fuel with a burnup of *58* GWd/MTU, enrichment of 4.0 wt. % U-235, and uranium loading of 185 kgU is to be loaded in the peripheral zone of the Type 1 DSC. From Table A.1.4.8-6, the minimum cooling time is *11.5* years. From the equation above, $\Delta t = 26.13*\ln(185/198)+1.0$ years = -0.8 years $\equiv$ 0 years. Therefore, the minimum required cooling time for this fuel assembly in the peripheral zone is *11.5* years. No additional cooling time is required because the reduction in uranium loading is sufficient to reduce the dose rate to acceptable levels.

The cooling times determined by Table A.1.4.8-6 or Table A.1.4.8-7 (as supplemented by equation A.1.4.8-1) are developed to meet dose rate limits only. The minimum cooling time for an assembly is the longer of that given by these supplemented FQT tables and the cooling time required to meet the decay heat restrictions provided in Figure A.1.4.8-1 through Figure A.1.4.8-8 for the heat load zoning configuration selected.

A.1.4.8.4 References

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Subsections NB, NG and NF, 1998 edition including 2000 Addenda.

Table A.1.4.8-7a
"B" Parameters to Determine Additional Cooling Time for Fuel in Peripheral Compartments (years)

BU	***Assembly Average Initial Enrichment (wt.% U-235)***																								
GWd/MTU	***2.6***	***2.7***	***2.8***	***2.9***	***3.0***	***3.1***	***3.2***	***3.3***	***3.4***	***3.5***	***3.6***	***3.7***	***3.8***	***3.9***	***4.0***	***4.1***	***4.2***	***4.3***	***4.4***	***4.5***	***4.6***	***4.7***	***4.8***	***4.9***	***5.0***
49	*1.5*																								
50	*2.5*	*1.5*	*0.5*																						
51	*4.0*	*3.0*	*2.0*	*1.0*																					
52	*5.0*	*4.0*	*3.5*	*2.5*	*1.5*	*0.5*																			
53	*6.0*	*5.5*	*4.5*	*4.0*	*3.0*	*2.0*	*1.0*	*0.5*										*No Additional Cooling Required*							
54	*7.0*	*6.0*	*5.5*	*5.0*	*4.0*	*3.5*	*2.5*	*1.5*	*1.0*																
55	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*3.5*	*3.0*	*2.0*	*1.0*	*0.5*														
56	*8.5*	*8.0*	*7.0*	*6.5*	*6.0*	*5.5*	*5.0*	*4.0*	*3.5*	*2.5*	*2.0*	*1.0*													
57	*9.0*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*3.5*	*3.0*	*2.0*	*1.0*	*0.5*											
58	*9.5*	*9.0*	*8.5*	*8.0*	*7.5*	*7.5*	*6.5*	*6.0*	*5.5*	*5.0*	*4.0*	*3.5*	*2.5*	*2.0*	*1.0*										
59	*9.5*	*9.5*	*9.0*	*8.5*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*4.0*	*3.0*	*2.5*	*1.5*	*0.5*								
60	*10.0*	*10.0*	*9.5*	*9.0*	*9.0*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.5*	*5.0*	*4.0*	*3.5*	*3.0*	*2.0*	*1.0*	*0.5*						
61	*10.5*	*10.0*	*10.0*	*9.5*	*9.5*	*9.0*	*8.5*	*8.5*	*7.5*	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*4.0*	*3.0*	*2.5*	*2.0*	*1.0*					
62	*10.5*	*10.5*	*10.5*	*10.0*	*10.0*	*9.5*	*9.0*	*9.0*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.5*	*5.0*	*4.5*	*3.5*	*3.0*	*2.0*	*1.5*	*0.5*			
Enr. wt.%	***2.6***	***2.7***	***2.8***	***2.9***	***3.0***	***3.1***	***3.2***	***3.3***	***3.4***	***3.5***	***3.6***	***3.7***	***3.8***	***3.9***	***4.0***	***4.1***	***4.2***	***4.3***	***4.4***	***4.5***	***4.6***	***4.7***	***4.8***	***4.9***	***5.0***

Table A.1.4.8-8
BWR Assembly Decay Heat for Heat Load Configurations

The decay heat (DH) in watts is expressed as:

$$F1 = -59.1 + 23.4*X1 - 21.1*X2 + 0.280*X1^2 - 3.52*X1*X2 + 12.4*X2^2$$
$$DH = F1*Exp(\{[1-(1.2/X3)]* -0.720\}*[(X3-4.5)^{0.157}]*[(X2/X1)^{-0.132}]) + 10$$

where,

F1 Intermediate function
X1 Assembly burnup in GWD/MTU
X2 Initial enrichment in wt. % U-235
X3 Cooling time in years (minimum *5* years)

A uranium loading of 198 kg is employed in the calculation of the decay heat equation. Alternatively, the decay heat can be calculated without employing the decay heat equation, using an approved methodology with actual spent fuel parameters instead of bounding spent fuel parameters.

Appendix A.1.4.9
NUHOMS®-69BTH DSC

TABLE OF CONTENTS

A.1.4.9.1 NUHOMS®-69BTH DSC Description A.1.4.9-1
A.1.4.9.2 NUHOMS®-69BTH Fuel Basket A.1.4.9-1
A.1.4.9.3 NUHOMS®-69BTH DSC Contents A.1.4.9-2
A.1.4.9.4 References A.1.4.9-2a

LIST OF TABLES

Table A.1.4.9-1 BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-69BTH DSC A.1.4.9-3
Table A.1.4.9-2 BWR Fuel Assembly Design Characteristics for the NUHOMS®-69BTH DSC A.1.4.9-4
Table A.1.4.9-3 BWR Fuel Assembly Lattice Average Initial Enrichment v/s Minimum B10 Requirements for the NUHOMS®-69BTH DSC Poison Plates A.1.4.9-6
Table A.1.4.9-4 BWR Fuel Qualification Table for the NUHOMS®-69BTH DSC (Uranium Loading ≤ 198 kg) A.1.4.9-7
Table A.1.4.9-5 BWR Fuel Qualification Table for the NUHOMS®-69BTH DSC (Uranium Loading ≤ 182 kg) A.1.4.9-8
Table A.1.4.9-5a "B" Parameters to Determine Additional Cooling Time for Fuel in Peripheral Compartments (years) A.1.4.9-9a
Table A.1.4.9-5b BWR Fuel Qualification Table for NUHOMS®-69BTH DSC using Type "F" Poison in HLZC 8: A.1.4.9-9b
Table A.1.4.9-6 BWR Assembly Decay Heat for Heat Load Configurations A.1.4.9-10

LIST OF FIGURES

Figure A.1.4.9-1 Location of Damaged Fuel Assemblies Inside 69BTH DSC A.1.4.9-11
Figure A.1.4.9-2 Heat Load Zoning Configuration No. 1 for 69BTH Basket A.1.4.9-12
Figure A.1.4.9-3 Heat Load Zoning Configuration No. 2 for 69BTH Basket A.1.4.9-13
Figure A.1.4.9-4 Heat Load Zoning Configuration No. 3 for 69BTH Basket A.1.4.9-14
Figure A.1.4.9-5 Heat Load Zoning Configuration No. 4 for 69BTH Basket A.1.4.9-15
Figure A.1.4.9-5a Heat Load Zoning Configuration No. 8 for 69BTH DSC Basket A.1.4.9-15a
Figure A.1.4.9-6 Peripheral and Inner Fuel Locations for the 69BTH DSC A.1.4.9-16

Shear keys, welded to the inner wall of the DSC, mate with notches in the basket support rails to prevent the basket from rotating during normal operations. Also a hold down ring is installed above the basket to prevent the basket from moving axially during transport.

The NUHOMS®-69BTH DSC is designed with six alternate basket configurations based on the boron content in the poison plates as listed in Table A.1.4.9-3 (designated as "A" for the poison plates with the lowest B10 loading to "F" for the highest B10 loading). Three alternate poison materials are allowed: (a) Borated Aluminum alloy, (b) Boron Carbide/Aluminum Metal Matrix Composite (MMC), or (c) Boral®. The poison plates provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

A.1.4.9.3 NUHOMS®-69BTH DSC Contents

The NUHOMS®-69BTH DSC is designed to transport 69 intact, or up to 24 damaged and the remainder intact, standard BWR fuel assemblies with or without fuel channels. The NUHOMS®-69BTH DSC can transport intact or damaged BWR fuel assemblies with the characteristics described in Table A.1.4.9-1. *The fuel to be transported 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 70 GWd/MTU.* Damaged BWR fuel assemblies are fuel assemblies containing 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 will still be able to be handled by normal means. Missing fuel rods are allowed.

The fuel assemblies considered are listed in Table A.1.4.9-2.

A.1.4.9.3.1 Heat Load Zone Configurations 1, 2, 3, and 4

The basket is divided into inner and peripheral zones, as shown in Figure A.1.4.9-6. Fuel in the inner zone is governed by the cooling times presented in Table A.1.4.9-4 and Table A.1.4.9-5 for fuel loadings ≤198 kgU and ≤182 kgU, respectively. These cooling times are determined to meet dose rate limits. For fuel in the peripheral zone, certain burnup and enrichment combinations require additional cooling time governed by the following equation:

Peripheral Compartment Cooling = Fuel Qualification Table Cooling + Δt

$$\Delta t = 26.13 \cdot \ln\left(\frac{m}{198}\right) + B \qquad \text{(A.1.4.9-1)}$$

Where,

Δt = cooling time to be added to the appropriate value in Table A.1.4.9-4 (years). (Do not add Δt to the values from Table A.1.4.9-5)
m = the uranium loading of the fuel assembly to be shipped (kg),
B = additional cooling time (years), obtain*ed* from Table A.1.4.9-5a

This equation applies only if the "B" value is defined in Table A.1.4.9-5a. Additional cooling is not required for burnup and enrichment combinations that fall in the zone labeled "no additional cooling time required." *No* additional cooling *time* is required for burnups < *49* GWd/MTU.

This equation can return a negative value for Δt, which indicates that no additional cooling is required. In such a scenario, $\Delta t \equiv 0$ years. Application of the method is illustrated in the following examples.

Example #1: Fuel with a burnup of 62 GWd/MTU, enrichment of 2.6 wt. % U-235, and uranium loading of 198 kgU is to be loaded in the peripheral zone. From Table A.1.4.9-4, the minimum cooling time is 18.5 years. From the equation above, $\Delta t = 26.13*\ln(1) + 10.5 = 10.5$ years. Therefore, the minimum required cooling time for this fuel assembly in the peripheral zone is $18.5+10.5 = 29.0$ years.

Example #2: Fuel with a burnup of *58* GWd/MTU, enrichment of 4.0 wt. % U-235, and uranium loading of 185 kgU is to be loaded in the peripheral zone. From Table A.1.4.9-4, the minimum cooling time is *7.5* years. From the equation above, $\Delta t = 26.13*\ln(185/198)+1.0$ years = -0.8 years $\equiv$ 0 years. Therefore, the minimum required cooling time for this fuel assembly in the peripheral zone is *7.5* years. No additional cooling time is required because the reduction in uranium loading is sufficient to reduce the dose rate to acceptable levels.

The cooling times determined by Table A.1.4.9-4 (as supplemented by equation A.1.4.9-1) or Table A.1.4.9-5 *(as supplemented by equation A.1.4.9-1)* are developed to meet dose rate limits only. The minimum cooling time for an assembly is the longer of that given by Table A.1.4.9-4 or Table A.1.4.9-5 *(as supplemented by equation A.1.4.9-1)* and the cooling time required to meet the decay heat restrictions provided in Figure A.1.4.9-2 through Figure A.1.4.9-5 for the heat load zoning configuration selected.

*Example #3: Fuel with a burnup of 70 GWd/MTU, enrichment of 3.7 wt. % U-235, and uranium loading of 180 kgU is acceptable for loading after 17.5-year cooling time, from Table A.1.4.9-5, when placed in inner positions. When placed in peripheral positions, the required cooling time is 27.5-year cooling time calculated as 19.5 (from Table A.1.4.9-4) + 26.13 * ln (180/198) (from Equation A.1.4.9-1) + 10.5 (from Table A.1.4.9-5a).*

A.1.4.9.3.2 Heat Load Zone Configuration 8

The Heat Load Zone Configuration 8 applies only to the 69 BTH DSC Type F, Table A.1.4.9-3. The 69BTH DSC basket using Heat Load Zone Configuration 8 is divided into five zones, as shown in Figure A.1.4.9-5a for fuel loadings ≤ 188 kgU. Fuel in zone 1 is governed by the cooling times presented in Table A.1.4.9-5b, Part 1. Fuel in zone 2 is governed by the cooling times presented in Table A.1.4.9-5b, Part 2. Fuel in zone 3 is governed by the cooling times presented in Table A.1.4.9-5b, Part 3. Fuel in zone 4 is governed by the cooling times presented in Table A.1.4.9-5b, Part 4. Fuel in zone 5 is governed by the cooling times presented in Table A.1.4.9-5b, Part 5. These cooling times are determined to meet dose rate limits in NCT and HAC. There are no supplemental cooling time tables generated for the 69BTH DSC basket using Heat Load Zone Configuration 8.

A.1.4.9.4 References

1. American Society of Mechanical Engineers, ASME Boiler And Pressure Vessel Code, Section III, Division 1 - Subsections NB, NG and NF, 2004 edition including 2006 Addenda.

Table A.1.4.9-1
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-69BTH DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.9-2. Damaged fuel assemblies beyond the definition contained below are not authorized for transport.
Damaged Fuel	Damaged BWR fuel assemblies are assemblies containing 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 will still be able to be handled by normal means. Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
RECONSTITUTED FUEL ASSEMBLIES[(1)]**:** • Maximum Number of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum Number of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum Number of Reconstituted Assemblies per DSC with unlimited number of low enriched UO_2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	 4 4 69
Number of Intact Assemblies	≤69
Number and Location of Damaged Assemblies	Up to 24 damaged fuel assemblies, with balance intact or dummy assemblies, are authorized for transport in 69BTH DSC. Damaged fuel assemblies may only be transported in the four outer "6-compartment" arrays as shown in Figure A.1.4.9-1. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Channels	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Maximum Assembly Weight with Channels	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS: Maximum Lattice Average Initial Enrichment (wt. % U-235)	Per Table A.1.4.9-3.
Allowable Heat Load Zoning Configurations for each 69BTH DSC	Per Figure A.1.4.9-2 or Figure A.1.4.9-3 or Figure A.1.4.9-4 or Figure A.1.4.9-5 *or Figure A.1.4.9-5a.*
Fuel Assembly Average Burnup and minimum Cooling Time	Per Table A.1.4.9-4 (as supplemented by equation A.1.4.9-1 for peripheral compartments) or Table A.1.4.9-5 *or Table A.1.4.9-5b.*
Decay Heat per DSC	Per Figure A.1.4.9-2 or Figure A.1.4.9-3 or Figure A.1.4.9-4 or Figure A.1.4.9-5 *or Figure A.1.4.9-5a.*
Minimum B10 Content in Poison Plates	Per Table A.1.4.9-3.

(1) Reconstituted rods shall displace an amount of water equal to or greater than that displaced by the original fuel rods in the active fuel region of the fuel assembly.

Notes, Table A.1.4.9-4 and Table A.1.4.9-5:

- BU = Assembly Average burnup.
- For fuel assemblies with natural uranium blankets greater than 8 inches at the top and/or bottom end, BU=Maximum Planar Average Burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with a lattice average initial enrichment less than 0.7 (or less than the minimum provided above for each burnup) or greater than 5.0 wt. % U-235 is unacceptable for transportation.
- Fuel with an assembly average burnup greater than *70* GWd/MTU is unacceptable for transportation.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transportation after 6-years cooling.
- For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 for fuel assemblies in the 24 peripheral locations of the canister with cooling times less than 10 years. No adjustment of cooling time is required for fuel assemblies in other locations or for those that have cooled for more than 10 years.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235, a Uranium loading of 190 kg, and a burnup of 41.5 GWd/MTU is acceptable for transport after a 6-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- The minimum cooling times from Table A.1.4.9-4 shall be utilized for fuel assemblies with a Uranium loading greater than 182 kg.
- Fuel in the peripheral compartments may require additional cooling time per equation A.1.4.9-1*; the additional cooling is to be added to the minimum cooling time in Table A.1.4.9-4.*

Table A.1.4.9-5a
"B" Parameters to Determine Additional Cooling Time for Fuel in Peripheral Compartments (years)

BU	***Assembly Average Initial Enrichment (wt.% U-235)***																								
GWd/MTU	**2.6**	**2.7**	**2.8**	**2.9**	**3.0**	**3.1**	**3.2**	**3.3**	**3.4**	**3.5**	**3.6**	**3.7**	**3.8**	**3.9**	**4.0**	**4.1**	**4.2**	**4.3**	**4.4**	**4.5**	**4.6**	**4.7**	**4.8**	**4.9**	**5.0**
49	*1.5*																								
50	*2.5*	*1.5*	*0.5*																						
51	*4.0*	*3.0*	*2.0*	*1.0*																					
52	*5.0*	*4.0*	*3.5*	*2.5*	*1.5*	*0.5*																			
53	*6.0*	*5.5*	*4.5*	*4.0*	*3.0*	*2.0*	*1.0*	*0.5*										*No Additional Cooling Required*							
54	*7.0*	*6.0*	*5.5*	*5.0*	*4.0*	*3.5*	*2.5*	*1.5*	*1.0*																
55	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*3.5*	*3.0*	*2.0*	*1.0*	*0.5*														
56	*8.5*	*8.0*	*7.0*	*6.5*	*6.0*	*5.5*	*5.0*	*4.0*	*3.5*	*2.5*	*2.0*	*1.0*													
57	*9.0*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*3.5*	*3.0*	*2.0*	*1.0*	*0.5*											
58	*9.5*	*9.0*	*8.5*	*8.0*	*7.5*	*7.5*	*6.5*	*6.0*	*5.5*	*5.0*	*4.0*	*3.5*	*2.5*	*2.0*	*1.0*										
59	*9.5*	*9.5*	*9.0*	*8.5*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*4.0*	*3.0*	*2.5*	*1.5*	*0.5*								
60	*10.0*	*10.0*	*9.5*	*9.0*	*9.0*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.5*	*5.0*	*4.0*	*3.5*	*3.0*	*2.0*	*1.0*	*0.5*						
61	*10.5*	*10.0*	*10.0*	*9.5*	*9.5*	*9.0*	*8.5*	*8.5*	*7.5*	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*4.0*	*3.0*	*2.5*	*2.0*	*1.0*					
62	*10.5*	*10.5*	*10.5*	*10.0*	*10.0*	*9.5*	*9.0*	*9.0*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.5*	*5.0*	*4.5*	*3.5*	*3.0*	*2.0*	*1.5*	*0.5*			
63	*Not Analyzed*											*7.5*	*7.5*	*7.0*	*6.5*	*6.0*	*5.0*	*4.5*	*4.0*	*3.5*	*3.0*	*2.0*	*1.5*	*0.5*	
64												*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.5*	*5.5*	*5.0*	*4.5*	*4.0*	*3.5*	*2.5*	*2.0*	*1.0*
65												*9.0*	*8.5*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.5*	*5.0*	*4.5*	*3.5*	*3.0*	*2.5*
66												*9.5*	*9.0*	*9.0*	*8.5*	*8.0*	*8.0*	*7.0*	*6.5*	*6.5*	*6.0*	*5.5*	*4.5*	*4.0*	*3.5*
67												*10.0*	*9.5*	*9.0*	*9.0*	*8.5*	*8.5*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.5*	*5.0*	*4.5*
68												*10.0*	*10.0*	*9.5*	*9.5*	*9.0*	*9.0*	*8.5*	*8.0*	*8.0*	*7.5*	*7.0*	*6.5*	*6.0*	*5.5*
69												*10.5*	*10.0*	*10.0*	*10.0*	*9.5*	*9.0*	*9.0*	*8.5*	*8.5*	*8.0*	*7.5*	*7.0*	*7.0*	*6.5*
70												*10.5*	*10.5*	*10.5*	*10.0*	*10.0*	*10.0*	*9.5*	*9.0*	*9.0*	*8.5*	*8.0*	*8.0*	*7.5*	*7.0*
Enr. wt.%	**2.6**	**2.7**	**2.8**	**2.9**	**3.0**	**3.1**	**3.2**	**3.3**	**3.4**	**3.5**	**3.6**	**3.7**	**3.8**	**3.9**	**4.0**	**4.1**	**4.2**	**4.3**	**4.4**	**4.5**	**4.6**	**4.7**	**4.8**	**4.9**	**5.0**

Table A.1.4.9-5b
BWR Fuel Qualification Table for NUHOMS®-69BTH DSC using Type "F" Poison in HLZC 8: Zone 1 with 188 kgU/FA
(Minimum required years of cooling time after reactor core discharge for fuel with 188 kgU per FA)
(Part 1 of 5)

BU,	Assembly Average Initial Enrichment (wt.% U-235)																																		
GWd/MTU	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
10	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
15	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
16	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
17	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
18	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
19	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
20	2.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
21	5.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
22	9.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
23	13.5	2.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
24	17.0	4.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
25	20.5	8.0	2.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
26	23.5	11.5	4.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
27	26.5	15.0	8.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
28	29.5	18.5	11.5	2.5	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
29	32.5	21.5	15.0	4.5	3.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
30	35.0	24.5	18.0	7.5	5.5	4.0	2.5	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
31				10.5	8.5	6.5	4.5	3.0	2.5	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
32				13.5	11.5	9.5	7.5	5.5	4.0	3.0	2.5	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
33				16.5	14.5	12.5	10.5	8.5	7.0	5.0	3.5	2.5	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
34				19.5	17.5	15.5	13.5	11.5	10.0	8.0	6.0	4.5	3.5	2.5	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
35				22.0	20.5	18.5	16.5	14.5	13.0	11.0	9.0	7.5	6.0	4.5	3.5	2.5	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
36				25.0	23.0	21.0	19.5	17.5	15.5	14.0	12.0	10.5	8.5	7.0	5.5	4.5	3.5	2.5	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
37				27.5	25.5	24.0	22.0	20.0	18.5	16.5	15.0	13.0	11.5	10.0	8.5	7.0	5.5	4.0	3.5	2.5	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
38				30.0	28.0	26.5	24.5	23.0	21.0	19.5	17.5	16.0	14.5	13.0	11.0	9.5	8.0	6.5	5.5	4.0	3.5	2.5	2.5	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
39				32.5	30.5	29.0	27.0	25.5	23.5	22.0	20.5	18.5	17.0	15.5	14.0	12.5	11.0	9.5	8.0	6.5	5.5	4.0	3.5	3.0	2.5	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5
40										24.5	23.0	21.5	19.5	18.0	16.5	15.0	13.5	12.0	10.5	9.5	8.0	6.5	5.5	4.0	3.5	3.0	2.5	2.0	2.0	2.0	2.0	2.0	1.5	1.5
41										27.0	25.5	24.0	22.0	20.5	19.0	17.5	16.5	15.0	13.5	12.0	10.5	9.0	8.0	6.5	5.5	4.5	3.5	3.0	2.5	2.0	2.0	2.0	2.0	2.0
42										29.5	28.0	26.5	25.0	23.5	22.0	20.5	19.0	17.5	16.0	14.5	13.5	12.0	10.5	9.5	8.0	7.0	5.5	4.5	3.5	3.0	2.5	2.5	2.0	2.0
43										32.0	30.5	29.0	27.5	26.0	24.5	23.0	21.5	20.0	18.5	17.5	16.0	14.5	13.5	12.0	10.5	9.5	8.0	7.0	6.0	5.0	4.0	3.5	3.0	2.5
44				Not Analyzed						34.5	32.5	31.0	30.0	28.5	27.0	25.5	24.0	22.5	21.0	20.0	18.5	17.0	16.0	14.5	13.5	12.0	11.0	9.5	8.5	7.0	6.0	5.0	4.0	3.5
45										36.5	35.0	33.5	32.0	30.5	29.5	28.0	26.5	25.0	23.5	22.5	21.0	19.5	18.5	17.0	16.0	14.5	13.5	12.0	11.0	10.0	8.5	7.5	6.5	5.5
46																																		
70																																		
Enr. wt.%	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Table A.1.4.9-5b
BWR Fuel Qualification Table for NUHOMS®-69BTH DSC using Type "F" Poison in HLZC 8: Zone 2 with 188 kgU/FA
(Minimum required years of cooling time after reactor core discharge for fuel with 188 kgU per FA)
(Part 2 of 5)

BU, GWd/MTU	Assembly Average Initial Enrichment (wt.% U-235)																																	
	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
20	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
30	3.5	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
31				1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
35				2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0
39				3.0	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
40										2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
42										2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
44										3.5	3.0	3.0	2.5	2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5
45										4.0	3.5	3.0	3.0	3.0	2.5	2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	1.5
46										8.0	7.0	6.0	5.0	4.5	4.0	3.5	3.0	3.0	3.0	2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
47										10.0	8.5	7.5	6.5	5.5	4.5	4.0	3.5	3.5	3.0	3.0	3.0	2.5	2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
48										12.0	10.5	9.5	8.0	7.0	6.0	5.0	4.5	4.0	3.5	3.5	3.0	3.0	3.0	2.5	2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0
49										14.0	12.5	11.5	10.0	9.0	7.5	6.5	5.5	5.0	4.5	4.0	3.5	3.5	3.0	3.0	3.0	2.5	2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0
50										16.0	14.5	13.0	12.0	10.5	9.5	8.5	7.0	6.0	5.5	4.5	4.0	4.0	3.5	3.0	3.0	3.0	3.0	2.5	2.5	2.5	2.5	2.5	2.5	2.0
51										17.5	16.5	15.0	14.0	12.5	11.5	10.0	9.0	8.0	7.0	6.0	5.0	4.5	4.0	3.5	3.5	3.0	3.0	3.0	3.0	2.5	2.5	2.5	2.5	2.5
52										19.5	18.0	17.0	15.5	14.5	13.5	12.0	11.0	9.5	8.5	7.5	6.5	5.5	5.0	4.5	4.0	3.5	3.5	3.0	3.0	3.0	3.0	2.5	2.5	2.5
53										21.0	20.0	18.5	17.5	16.5	15.0	14.0	13.0	11.5	10.5	9.5	8.0	7.0	6.0	5.5	5.0	4.5	4.0	3.5	3.5	3.0	3.0	3.0	3.0	2.5
54										23.0	21.5	20.5	19.5	18.0	17.0	16.0	14.5	13.5	12.5	11.0	10.0	9.0	8.0	7.0	6.0	5.5	4.5	4.5	4.0	3.5	3.5	3.0	3.0	3.0
55										24.5	23.5	22.0	21.0	20.0	18.5	17.5	16.5	15.0	14.0	13.0	12.0	10.5	9.5	8.5	7.5	6.5	6.0	5.0	4.5	4.0	4.0	3.5	3.5	3.0
56										26.0	25.0	24.0	22.5	21.5	20.5	19.5	18.0	17.0	16.0	14.5	13.5	12.5	11.5	10.5	9.5	8.0	7.5	6.5	5.5	5.0	4.5	4.0	4.0	3.5
57										28.0	26.5	25.5	24.5	23.0	22.0	21.0	20.0	18.5	17.5	16.5	15.5	14.5	13.0	12.0	11.0	10.0	9.0	8.0	7.0	6.0	5.5	5.0	4.5	4.0
58										29.5	28.0	27.0	26.0	25.0	23.5	22.5	21.5	20.5	19.5	18.0	17.0	16.0	15.0	14.5	13.0	11.5	10.5	9.5	8.5	7.5	7.0	6.0	5.5	5.0
59										31.0	29.5	28.5	27.5	26.5	25.5	24.0	23.0	22.0	21.0	20.0	19.0	17.5	16.5	15.5	14.5	13.5	12.5	11.5	10.5	9.5	8.5	7.5	6.5	6.0
60										32.5	31.0	30.0	29.0	28.0	27.0	26.0	24.5	23.5	22.5	21.5	20.5	19.5	18.5	17.0	16.0	15.0	14.0	13.0	12.0	11.0	10.0	9.0	8.0	7.5
61										34.0	32.5	31.5	30.5	29.5	28.5	27.5	26.5	25.5	24.0	23.0	22.0	21.0	20.0	19.0	18.0	17.0	16.0	15.0	14.0	13.0	12.0	11.0	10.0	9.0
62										35.5	34.0	33.0	32.0	31.0	30.0	29.0	28.0	27.0	26.0	25.0	24.0	22.5	21.5	20.5	19.5	18.5	17.5	16.5	15.5	14.5	13.5	12.5	11.5	10.5
63																					25.5	24.5	23.5	22.5	21.5	20.5	19.0	18.0	17.5	16.5	15.5	14.5	13.5	12.5
64																					27.0	26.0	25.0	24.0	23.0	22.0	21.0	20.0	19.0	18.0	17.0	16.0	15.0	14.0
65				Not Analyzed																	28.5	27.5	26.5	25.5	24.5	23.5	22.5	21.5	20.5	19.5	18.5	17.5	16.5	16.0
66																					30.0	29.0	28.0	27.0	26.0	25.0	24.0	23.0	22.0	21.0	20.5	19.5	18.5	17.5
67																					31.5	30.5	29.5	28.5	27.5	26.5	25.5	24.5	24.0	23.0	22.0	21.0	20.0	19.0
68																					33.0	32.0	31.0	30.0	29.0	28.0	27.0	26.5	25.5	24.5	23.5	22.5	21.5	20.5
69																					34.5	33.5	32.5	31.5	30.5	29.5	28.5	28.0	27.0	26.0	25.0	24.0	23.0	22.5
70																					36.0	35.0	34.0	33.0	32.0	31.0	30.0	29.5	28.5	27.5	26.5	25.5	24.5	24.0
Enr. wt.%	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Table A.1.4.9-5b
BWR Fuel Qualification Table for NUHOMS®-69BTH DSC using Type "F" Poison in HLZC 8: Zone 3 with 188 kgU/FA
(Minimum required years of cooling time after reactor core discharge for fuel with 188 kgU per FA)
(Part 3 of 5)

BU,	Assembly Average Initial Enrichment (wt.% U-235)																																		
GWd/MTU	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
20	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
30	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
31				0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
35				0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
39				0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
40										0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
42										0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
44										1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
45										1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
46										1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
47										1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
48										1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
49										2.0	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
50										2.0	2.0	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5
51										2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5
52										3.0	2.5	2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
53										4.0	3.5	3.0	2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
54										5.0	4.0	3.5	3.0	2.5	2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
55										6.5	5.5	4.5	4.0	3.0	3.0	2.5	2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
56										7.5	6.5	5.5	5.0	4.0	3.5	3.0	2.5	2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0
57										9.0	8.0	7.0	6.0	5.0	4.5	4.0	3.5	3.0	2.5	2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0
58										10.5	9.5	8.5	7.5	6.5	5.5	5.0	4.0	3.5	3.0	2.5	2.5	2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0
59										12.0	11.0	10.0	9.0	8.0	7.0	6.0	5.0	4.5	4.0	3.5	3.0	2.5	2.5	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5
60										13.5	12.5	11.5	10.5	9.5	8.5	7.5	6.5	5.5	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5
61										15.0	14.0	13.0	12.0	11.0	10.0	9.0	8.0	7.0	6.0	5.0	4.5	4.0	3.5	3.0	2.5	2.5	2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5
62										16.5	15.0	14.0	13.5	12.5	11.5	10.5	9.5	8.5	7.5	6.5	5.5	5.0	4.0	3.5	3.0	3.0	2.5	2.5	2.0	2.0	2.0	2.0	1.5	1.5
63																					7.0	6.0	5.5	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.0	2.0	2.0	2.0
64																					8.5	7.5	6.5	5.5	5.0	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.0	2.0
65																					10.0	9.0	8.0	7.0	6.0	5.5	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.5
66																					11.0	10.5	9.5	8.5	7.5	6.5	6.0	5.0	4.5	4.0	3.5	3.0	3.0	2.5
67																					12.5	11.5	11.0	10.0	9.0	8.0	7.0	6.5	5.5	5.0	4.5	4.0	3.5	3.0
68																					14.0	13.0	12.0	11.5	10.5	9.5	8.5	7.5	7.0	6.0	5.5	4.5	4.0	3.5
69																					15.5	14.5	13.5	12.5	12.0	11.0	10.0	9.0	8.5	7.5	6.5	6.0	5.0	4.5
70																					17.0	16.0	15.0	14.0	13.0	12.5	11.5	10.5	9.5	9.0	8.0	7.0	6.5	5.5
Enr. wt.%	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Not Analyzed

Table A.1.4.9-5b
BWR Fuel Qualification Table for NUHOMS®-69BTH DSC using Type "F" Poison in HLZC 8: Zone 4 with 188 kgU/FA
(Minimum required years of cooling time after reactor core discharge for fuel with 188 kgU per FA)
(Part 4 of 5)

BU,	Assembly Average Initial Enrichment (wt.% U-235)																																	
GWd/MTU	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
20	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
30	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
31				0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
35				0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
39				1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
40										0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
42										1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
44										1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
45										1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
46										1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
47										1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5	0.5	0.5	0.5
48										1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	0.5	0.5	0.5	0.5
49										1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
50										2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
51										2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
52										2.5	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
53										2.5	2.5	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
54										3.0	2.5	2.5	2.5	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
55										3.5	3.0	3.0	2.5	2.5	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0	1.0	1.0
56										4.0	3.5	3.0	3.0	2.5	2.5	2.5	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0	1.0
57										5.0	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.0	1.0	1.0	1.0
58										6.0	5.5	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
59										7.5	6.5	5.5	5.0	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5
60										8.5	7.5	7.0	6.0	5.0	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5	1.5	1.5
61										10.0	9.0	8.0	7.0	6.5	5.5	5.0	4.5	4.0	3.5	3.0	3.0	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	1.5	1.5	1.5	1.5
62										11.5	10.5	9.5	8.5	7.5	6.5	6.0	5.0	4.5	4.0	3.5	3.5	3.0	3.0	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0	1.5	1.5
63																					4.0	3.5	3.0	3.0	2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0	2.0	2.0
64																					4.5	4.0	3.5	3.5	3.0	3.0	2.5	2.5	2.5	2.5	2.0	2.0	2.0	2.0
65																					5.5	5.0	4.5	4.0	3.5	3.0	3.0	3.0	2.5	2.5	2.5	2.0	2.0	2.0
66																					6.5	6.0	5.0	4.5	4.0	4.0	3.5	3.0	3.0	3.0	2.5	2.5	2.5	2.0
67																					8.0	7.0	6.5	5.5	5.0	4.5	4.0	3.5	3.5	3.0	3.0	2.5	2.5	2.5
68																					9.0	8.5	7.5	6.5	6.0	5.5	4.5	4.5	4.0	3.5	3.5	3.0	3.0	2.5
69																					10.5	9.5	9.0	8.0	7.0	6.5	5.5	5.0	4.5	4.0	3.5	3.5	3.0	3.0
70																					12.0	11.0	10.0	9.0	8.5	7.5	7.0	6.0	5.5	5.0	4.5	4.0	3.5	3.5
Enr. wt.%	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Not Analyzed

Table A.1.4.9-5b
BWR Fuel Qualification Table for NUHOMS®-69BTH DSC using Type "F" Poison in HLZC 8: Zone 5 with 188 kgU/FA
(Minimum required years of cooling time after reactor core discharge for fuel with 188 kgU per FA)
(Part 5 of 5)

BU, GWd/MTU	Assembly Average Initial Enrichment (wt.% U-235)																																	
	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0
10	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0	2.0
20	3.0	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5	2.5
30	4.0	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
31				3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	2.5
35				3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
39				4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
40										3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0	3.0
42										4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.0	3.0
44										4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
45										4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
46										4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
47										4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
48										5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5	3.5	3.5	3.5
49										5.5	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5	3.5	3.5	3.5
50										6.0	5.5	5.5	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	3.5	3.5
51										6.5	6.0	5.5	5.5	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
52										7.0	6.5	6.0	6.0	5.5	5.5	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
53										8.0	7.0	6.5	6.5	6.0	5.5	5.5	5.5	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0
54										9.0	8.0	7.5	7.0	6.5	6.0	6.0	5.5	5.5	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0	4.0	4.0
55										10.0	9.0	8.5	8.0	7.0	7.0	6.5	6.0	6.0	5.5	5.5	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.0	4.0	4.0	4.0
56										11.0	10.5	9.5	9.0	8.0	7.5	7.0	6.5	6.0	6.0	5.5	5.5	5.5	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5	4.5	4.0
57										12.5	11.5	10.5	10.0	9.0	8.5	8.0	7.5	7.0	6.5	6.0	6.0	5.5	5.5	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5	4.5	4.5
58										13.5	13.0	12.0	11.0	10.0	9.5	8.5	8.0	7.5	7.0	6.5	6.5	6.0	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	4.5	4.5	4.5	4.5
59										15.0	14.0	13.0	12.0	11.5	10.5	10.0	9.0	8.5	8.0	7.5	7.0	6.5	6.0	6.0	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	4.5	4.5
60										16.5	15.5	14.5	13.5	12.5	12.0	11.0	10.0	9.5	8.5	8.0	7.5	7.0	6.5	6.5	6.0	6.0	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0
61										17.5	16.5	15.5	15.0	14.0	13.0	12.0	11.5	10.5	10.0	9.0	8.5	8.0	7.5	7.0	6.5	6.5	6.0	6.0	5.5	5.5	5.5	5.0	5.0	5.0
62										19.0	18.0	17.0	16.0	15.0	14.5	13.5	12.5	12.0	11.0	10.0	9.5	9.0	8.0	7.5	7.0	7.0	6.5	6.0	6.0	6.0	5.5	5.5	5.5	5.0
63																					10.5	10.0	9.0	8.5	8.0	7.5	7.0	6.5	6.5	6.0	6.0	5.5	5.5	5.5
64																					12.0	11.0	10.5	9.5	9.0	8.5	8.0	7.5	7.0	6.5	6.5	6.0	6.0	5.5
65																					13.0	12.5	11.5	10.5	10.0	9.5	9.0	8.0	7.5	7.5	7.0	6.5	6.5	6.0
66																					14.5	13.5	12.5	12.0	11.0	10.5	10.0	9.0	8.5	8.0	7.5	7.0	7.0	6.5
67																					15.5	15.0	14.0	13.0	12.5	11.5	11.0	10.0	9.5	9.0	8.5	8.0	7.5	7.0
68																					17.0	16.0	15.0	14.5	13.5	13.0	12.0	11.5	10.5	10.0	9.5	9.0	8.5	8.0
69																					18.0	17.5	16.5	15.5	15.0	14.0	13.5	12.5	12.0	11.0	10.5	10.0	9.0	8.5
70																					19.5	18.5	17.5	17.0	16.0	15.5	14.5	13.5	13.0	12.0	11.5	11.0	10.0	9.5
Enr. wt.%	0.7	1.2	1.5	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8	4.9	5.0

Not Analyzed

Notes for using Table A.1.4.9-5b, parts 1 through 5:

- *BU = Assembly Average burnup.*
- *For fuel assemblies with natural uranium blankets greater than 8 inches at the top and/or bottom end, BU=Maximum Planar Average Burnup.*
- *Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Fuel with a lattice average initial enrichment of less than 0.7 (or less than the minimum provided above for each burnup) or greater than 5.0 wt. % U-235 is unacceptable for transportation.*
- *Fuel with an assembly average burnup greater than 70 GWd/MTU is unacceptable for transportation.*
- *For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 for fuel assemblies in the 24 peripheral locations of the canister with cooling times less than 10 years. No adjustment of cooling time is required for fuel assemblies in other locations or for those that have cooled for more than 10 years.*
- *Example: An assembly is to be loaded into 69BTH using HLZC 8: The initial enrichment is 4.85 wt. % U-235, the Uranium loading is 188 kg, and the burnup is 41.5 GWD/MTU. Round 4.85 wt. % U-235 down to 4.8 wt. % U-235, round 41.5 GWD/MTU up to 42 GWD/MTU.*
    - *The FA is acceptable for transport in heat loading zone 1 according to Table A.1.4.9-5b, part 1 after 2.5 years of cooling time (other considerations not withstanding).*
    - *The FA is acceptable for transport in heat loading zone 2 according to Table A.1.4.9-5b, part 2 after 1.5 years of cooling time (other considerations not withstanding).*
    - *The FA is acceptable for transport in heat loading zone 3 according to Table A.1.4.9-5b, part 3 after 0.5 years of cooling time (other considerations not withstanding).*
    - *The FA is acceptable for transport in heat loading zone 4 according to Table A.1.4.9-5b, part 4 after 0.5 years of cooling time (other considerations not withstanding).*
    - *The FA is acceptable for transport in heat loading zone 5 according to Table A.1.4.9-5b, part 5 after 3.5 years of cooling time (other considerations not withstanding).*

Table A.1.4.9-6
BWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -59.1 + 23.4*X1 - 21.1*X2 + 0.280*X1^2 - 3.52*X1*X2 + 12.4*X2^2$$
$$DH = F1*Exp(\{[1-(1.2/X3)]* -0.720\}*[(X3-4.5)^{0.157}]*[(X2/X1)^{-0.132}]) + 10$$

where,

F1 Intermediate Function
X1 Assembly Burnup in GWD/MTU
X2 Initial Enrichment in wt. % U-235
X3 Cooling Time in Years (minimum *5* years)

A uranium loading of 198 kg is employed in the calculation of the decay heat equation. Alternatively, the decay heat can be calculated without employing the decay heat equation, using an approved methodology with actual spent fuel parameters instead of bounding spent fuel parameters.

		Z5	Z5	Z5	Z5	Z5		
	Z5	- (1)	Z3	Z2	Z3	-	Z5	
Z5	-	Z4	Z3	-	Z3	Z4	-	Z5
Z5	Z3	Z3	-	Z1	-	Z3	Z3	Z5
Z5	Z2	-	Z1	-	Z1	-	Z2	Z5
Z5	Z3	Z3	-	Z1	-	Z3	Z3	Z5
Z5	-	Z4	Z3	-	Z3	Z4	-	Z5
	Z5	-	Z3	Z2	Z3	-	Z5	
		Z5	Z5	Z5	Z5	Z5		

	Z1 (Zone 1)	Z2 (Zone 2)	Z3 (Zone 3)	Z4 (Zone 4)	Z5 (Zone 5)
Max. Decay Heat (kW/FA) (4)(5)	0.13	0.31	0.76	0.95	0.52
No. of Fuel Assemblies(2)	4	4	16	4	24
Max. Decay Heat per DSC (kW)	30.20 (3)(4)				

Notes:
(1) *Aluminum dummy assembly (solid aluminum block) is required in the fuel compartments designated with a dash (-).*
(2) *Total number of fuel assemblies is 52 for HLZC No. 8.*
(3) *Adjust payload to maintain total DSC heat load within the specified limit.*
(4) *Reduce the maximum decay heat to 70% of the listed values for LACROSSE10 Fuel assembly. The total decay heat for LACROSSE10 fuel assembly is 21.1 kW per DSC for HLZC No. 8.*
(5) *Type "F" borated aluminum is the only poison material authorized to use with HLZC No. 8.*
(6) *This figure is also shown in Section A.3.3.1.4, titled "Heat Load Zoning Configuration No. 8 for 69BTH Basket."*
(7) *The decay heat per fuel assembly may be determined using an approved methodology (e.g., ORIGEN-ARP) with actual spent fuel parameters. Alternately, for cooling times > 5 years, the decay heat equation (Table A.1.4.9-6) may be used.*

Figure A.1.4.9-5a
Heat Load Zoning Configuration No. 8 for 69BTH DSC Basket

Appendix A.1.4.10
NUHOMS®-MP197HB SAR Drawings

The following drawings for the NUHOMS®-MP197HB Cask are included in Section A.1.4.10.1.

Drawing Number	Title
MP197HB-71-1001 Rev 3	NUHOMS®-MP197HB Packaging Transport Configuration (2 sheets)
MP197HB-71-1002 Rev 6	NUHOMS®-MP197HB Packaging Parts List (2 sheets)
MP197HB-71-1003 Rev 2	NUHOMS®-MP197HB Packaging General Arrangement (1 sheet)
MP197HB-71-1004 Rev 4	NUHOMS®-MP197HB Packaging Cask Body Assembly (1 sheet)
MP197HB-71-1005 Rev *5*	NUHOMS®-MP197HB Packaging Cask Body Details (3 sheets)
MP197HB-71-1006 Rev 2	NUHOMS®-MP197HB Packaging Lid Assembly and Details (1 sheet)
MP197HB-71-1007 Rev 0	NUHOMS®-MP197HB Packaging Regulatory Plate (1 sheet)
MP197HB-71-1008 Rev 1	NUHOMS®-MP197HB Packaging Impact Limiter Assembly (1 sheet)
MP197HB-71-1009 Rev 1	NUHOMS®-MP197HB Packaging Impact Limiter Details (1 sheet)
MP197HB-71-1011 Rev 0	NUHOMS®-MP197HB Packaging Transport Configuration Outer Sleeve With Fins Option (1 sheet)
MP197HB-71-1014 Rev 1	NUHOMS®-MP197HB Packaging Internal Sleeve Design (2 sheets)

The following drawings for the NUHOMS® 24PT4 DSC are included in Section A.1.4.10.2.

Drawing Number	Title
NUH24PT4-71-1001 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Basket Assembly (5 sheets)
NUH24PT4-71-1002 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Main Assembly (8 sheets)
NUH24PT4-71-1003 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Failed Fuel Can (4 sheets)

The following drawings for the NUHOMS® 69BTH DSC are included in Section A.1.4.10.10.

Drawing Number	Title
NUH69BTH-71-1001 Rev 3	NUHOMS® 69BTH Transportable Canister For BWR Fuel Main Assembly (4 sheets)
NUH69BTH-71-1002 Rev 3	NUHOMS® 69BTH Transportable Canister For BWR Fuel Basket – Shell Assembly (4 sheets)
NUH69BTH-71-1003 Rev 3	NUHOMS® 69BTH Transportable Canister For BWR Fuel Shell Assembly (4 sheets)
NUH69BTH-71-1004 Rev *6*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Alternate Top Closure (*7* sheets)
NUH69BTH-71-1011 Rev 3	NUHOMS® 69BTH Transportable Canister For BWR Fuel Basket Assembly (5 sheets)
NUH69BTH-71-1012 Rev 4	NUHOMS® 69BTH Transportable Canister For BWR Fuel Transition Rail Assembly And Details (6 sheets)
NUH69BTH-71-1013 Rev 4	NUHOMS® 69BTH Transportable Canister For BWR Fuel Holddown Ring Assembly (2 sheets)
NUH69BTH-71-1014 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Damaged Fuel Modification (1 sheet)
NUH69BTH-71-1015 Rev 2	NUHOMS® 69BTH Transportable Canister For BWR Fuel Damaged Fuel End Caps (1 sheet)

The following drawings for the Radioactive Waste Canister are included in Section A.1.4.10.11.

Drawing Number	Title
NUHRWC-71-1001 Rev 1	NUHOMS® System RWC Canister - Welded Top Shield Plug Design Main Assembly (5 sheets)
NUHRWC-71-1002 Rev 1	NUHOMS® System RWC Canister - Welded Top Shield Plug Design Inner Liner (3 sheets)
NUHRWC-71-1003 Rev 0	NUHOMS® System RWC Canister - Bolted Top Shield Plug Design Main Assembly (4 sheets)

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
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0	FIRST ISSUE	03/26/09

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269

AREVA

SAFETY ANALYSIS REPORT

NUHOMS®MP197HB PACKAGING
CASK BODY DETAILS

DRAWING NO.	SCALE	SHEET
MP197HB-71-1005	NONE	1 OF 3

DRAWING NO. MP197HB-71-1005 SHEET 1 OF 3

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
MP197HB-71-1005	2 OF 3	5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
MP197HB-71-1005	3 OF 3	5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269

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SAFETY ANALYSIS REPORT
NUHOMS® 69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
ALTERNATE TOP CLOSURE

DRAWING NO.	SCALE	SHEET
NUH69BTH-71-1004	NONE	1 OF 7

DRAWING NO. NUH69BTH-71-1004 SHEET 1 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	4 OF 7	6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	5 OF 7	6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	6 OF 7	6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	7 OF 7	6

Chapter A.3
Thermal Evaluation

TABLE OF CONTENTS

A.3.1 Description of Thermal Design Criteria A.3-1a
A.3.1.1 Design Features A.3-3
A.3.1.2 Content's Decay Heat A.3-6
A.3.1.3 Summary Tables of Temperatures A.3-7
A.3.1.4 Summary Tables of Maximum Pressures A.3-8
A.3.2 Material Properties and Component Specifications A.3-8
A.3.2.1 Material Properties A.3-8
A.3.2.2 Component Specifications A.3-27
A.3.3 Thermal Evaluation under Normal Conditions of Transport A.3-29
A.3.3.1 Thermal Models A.3-29
A.3.3.2 Heat and Cold A.3-72
A.3.3.3 Maximum Normal Operating Pressure A.3-74
A.3.3.4 Thermal Evaluation for Loading/Unloading Operations A.3-84
A.3.4 Thermal Evaluation under Hypothetical Accident Conditions A.3-86
A.3.4.1 Initial Conditions A.3-86
A.3.4.2 Fire Test Conditions A.3-87
A.3.4.3 Maximum Temperatures and Pressure A.3-95
A.3.4.4 Maximum Thermal Stresses A.3-97
A.3.4.5 Accident Conditions for Fissile Material Packages for Air Transport A.3-97
A.3.5 References A.3-98
A.3.6 Appendices A.3-101
A.3.6.1 Thermal Evaluation of MP197HB TC and 69BTH DSC for HLZC # 8 for NCT and HAC A.3-102
A.3.6.2 Mesh Sensitivity A.3-103
A.3.6.3 Justification for Bounding Temperature Profiles A.3-104
A.3.6.4 Acceptance Criteria for Coating Damages for MP197HB TC A.3-106
A.3.6.5 Effective Thermal Properties of the Fuel Assemblies A.3-113
A.3.6.6 Thermal Analysis of 24PTHF and 61BTHF DSCs in the MP197HB TC A.3-120
A.3.6.7 Justification of Hot Gaps A.3-127
A.3.6.8 Sensitivity Study for Effects of Fire Emissivity A.3-140
A.3.6.9 Sensitivity Study for Effects of High Burnup Damaged Fuel Assemblies A.3-144
A.3.6.10 Sensitivity Analysis for HAC using Coupled Model A.3-145
A.3.6.11 Thermal Evaluation of MP197HB Transport Cask with High Burnup Fuel Assemblies A.3-147

LIST OF TABLES

Table A.3-1 DSC Shell Nominal Dimensions *A.3-152*
Table A.3-2 Peaking Factors for Fuel Assemblies in the 69BTH DSC Model *A.3-153*
Table A.3-3 Effective Density for the 69BTH Basket *A.3-154*
Table A.3-4 Effective Specific Heat for the 69BTH Basket *A.3-155*
Table A.3-5 Effective Axial Conductivity for the 69BTH Basket *A.3-156*
Table A.3-6 Effective Radial Conductivity for the 69BTH Basket *A.3-157*
Table A.3-7 Peaking Factors for Fuel Assemblies in the 37PTH DSC Model *A.3-158*
Table A.3-8 Maximum TC Component and DSC Shell Temperatures for NCT (100°F and Insolation), BWR DSCs *A.3-159*
Table A.3-9 Maximum TC Component and DSC Shell Temperatures for NCT (100°F and Insolation), PWR DSCs *A.3-160*
Table A.3-10 Maximum Fuel Cladding and Basket Component Temperatures for NCT *A.3-161*
Table A.3-11 Maximum/Minimum Component Temperatures for NCT *A.3-162*
Table A.3-12 Maximum TC Component and 69BTH DSC Temperatures for Cold NCT (32 kW, No insolation) *A.3-164*
Table A.3-13 Maximum TC Component and 37PTH DSC Temperatures for Cold NCT (22 kW, No insolation) *A.3-165*
Table A.3-14 Average Component Temperatures for NCT and HAC *A.3-166*
Table A.3-15 List of the Gaps and Thermal Properties for HAC Analysis *A.3-167*
Table A.3-16 Maximum Temperatures of TC Components / DSC Shell for HAC *A.3-168*
Table A.3-17 Maximum DSC Shell Temperatures for Steady-State Cool-Down *A.3-169*
Table A.3-18 Maximum Fuel Cladding and Basket Component Temperatures for HAC *A.3-170*
Table A.3-19 Summary of the Maximum Temperatures for HAC *A.3-171*
Table A.3-20 MP197HB TC Cavity Pressure for NCT *A.3-173*
Table A.3-21 Volume and Average Helium Temperatures in 69BTH and 37PTH DSCs for NCT and HAC *A.3-174*
Table A.3-22 Maximum Internal Pressures of 69BTH and 37PTH DSCs in MP197HB TC *A.3-175*
Table A.3-23 Maximum Internal Pressure in DSCs for Transport in MP197HB TC *A.3-176*
Table A.3-24 Comparison of Maximum DSC Shell Temperatures *A.3-177*
Table A.3-25 MP197HB Transport Cask Thermal Evaluation for Reconfigured Fuel Assemblies *A.3-178*
Table A.3-26 Maximum Temperatures of 69BTH DSC/ MP197HB Transport Cask Components (Reconfigured Fuel at Top, 32 kW with Fins, Load Case # F1O & F1A) *A.3-179*
Table A.3-27 Maximum Temperatures of 69BTH DSC/ MP197HB Transport Cask Components (Reconfigured Fuel at Bottom, 32 kW with Fins, Load Case # F2O & F2A) *A.3-180*
Table A.3-28 Maximum Temperatures of 69BTH DSC/ MP197HB Transport Cask Components (Reconfigured Fuel at Top, 26 kW, Load Case # U1O & U1A) *A.3-181*

Table A.3-29 Maximum Temperatures of 69BTH DSC/ MP197HB Transport Cask Components (Reconfigured Fuel at Bottom, 26 kW, Load Case # U2O & U2A).....*A.3-182*
Table A.3-30 Comparison of Maximum Seal Temperatures.....*A.3-182a*
Table A.3-31 Grid Sizes and Refinement Factors.....*A.3-182b*
Table A.3-32 Temperature of Neutron Shield Resin.....*A.3-182b*
Table A.3-33 Discretization Error of the Neutron Shield Resin within MP197HB TC.....*A.3-182c*

LIST OF FIGURES

Figure A.3-1 Location of DSC within the MP197HB TC *A.3-183*
Figure A.3-2 Finite Element Model of the MP197HB with the 69BTH DSC *A.3-184*
Figure A.3-3 Components of the MP197HB TC Model *A.3-185*
Figure A.3-4 Gaps in the MP197HB TC Model *A.3-186*
Figure A.3-5 Gaps in the MP197HB TC Cross Section *A.3-187*
Figure A.3-6 Typical Decay Heat and Insolance Boundary Conditions *A.3-188*
Figure A.3-7 Typical Convection and Radiation Boundary Conditions *A.3-189*
Figure A.3-8 Typical Convection and Radiation Boundary Conditions – Details *A.3-190*
Figure A.3-9 Schematic View of Cask and Personnel Barrier *A.3-191*
Figure A.3-10 Sub-Model of the External Fins for MP197HB TC *A.3-192*
Figure A.3-11 Sub-Model for Helium Gap Effective Conductivity Calculation *A.3-193*
Figure A.3-12 Assumed Geometry of Internal Sleeve in Thermal Model *A.3-194*
Figure A.3-13 Finite Element Model of 69BTH DSC/Basket *A.3-195*
Figure A.3-14 69BTH DSC/Basket – Cross Section *A.3-196*
Figure A.3-15 69BTH DSC/Basket – Gaps between Rail Sections at Cross Section *A.3-197*
Figure A.3-16 69BTH DSC/Basket – Gaps between Basket Plates at Cross Section *A.3-198*
Figure A.3-17 69BTH DSC/Basket – Axial Gaps *A.3-199*
Figure A.3-18 Typical Boundary Conditions for the 69BTH Basket *A.3-200*
Figure A.3-19 Thermal Resistances for Aluminum Dummy Assembly *A.3-201*
Figure A.3-20 Basket Slice Models *A.3-202*
Figure A.3-21 Typical Boundary Conditions for the Basket Slice Model *A.3-203*
Figure A.3-22 Schematic View of Top Grid Assembly (Hold-Down Ring) *A.3-204*
Figure A.3-23 Finite Element Model of the 37PTH DSC/Basket *A.3-205*
Figure A.3-24 37PTH DSC/Basket – Cross Section and Details *A.3-206*
Figure A.3-25 37PTH DSC/Basket – Gaps between Rail Sections *A.3-207*
Figure A.3-26 37PTH DSC/Basket – Gaps between Basket Plates at Cross Section *A.3-208*
Figure A.3-27 37PTH DSC/Basket – Axial Gaps *A.3-209*
Figure A.3-28 Typical Boundary Conditions for the 37PTH Basket *A.3-210*
Figure A.3-29 Temperature Profiles for MP197HB Transport Cask NCT, 100 °F, Insolation, 69BTH DSC, 32 kW *A.3-211*
Figure A.3-30 Temperature Profiles for MP197HB Transport Cask NCT, 100 °F, Insolation, 24PTH DSC, 26 kW *A.3-212*
Figure A.3-31 Typical Temperature Distributions for 69BTH Basket (NCT @ 100 °F, HLZC#1, 26kW) *A.3-213*
Figure A.3-32 Typical Temperature Distributions for 69BTH Basket (NCT @ 100 °F, HLZC#4, 32kW) *A.3-214*
Figure A.3-33 Temperature Distributions for Fuel Assemblies in 69BTH Basket (NCT @ 100 °F) *A.3-215*
Figure A.3-34 Typical Temperature Distributions for 37PTH Basket (NCT @ 100 °F, 22 kW) *A.3-216*

Figure A.3-35 Impact Limiter Crush Areas for MP197HB (Side, End, and Corner Drop) *A.3-217*
Figure A.3-36 Impact Limiter Crush Areas for MP197HB (Slap Down) *A.3-218*
Figure A.3-37 Finite Element Model of TC with Deformed Impact Limiters *A.3-219*
Figure A.3-38 Typical Boundary Conditions during Fire Conditions *A.3-220*
Figure A.3-39 Typical Boundary Conditions for Smoldering/Cool-Down Periods *A.3-221*
Figure A.3-40 Time-Temperature Histories for TC/DSC Shell HAC, with External Fins (Melted after Fire), 69BTH DSC, 32 kW *A.3-222*
Figure A.3-41 Time-Temperature Histories for TC/DSC Shell HAC, w/o Fins, 69BTH DSC, 26 kW *A.3-223*
Figure A.3-42 Time-Temperature Histories for TC/DSC Shell HAC, w/o Fins, 24PTH DSC, 26 kW *A.3-224*
Figure A.3-43 Temperature Profiles for MP197HB Transport Cask HAC, 69BTH DSC, 32 kW, with External Fins (Melted after Fire) *A.3-225*
Figure A.3-44 Temperature Profiles for MP197HB Transport Cask HAC, 24PTH DSC, 26 kW, without External Fins *A.3-226*
Figure A.3-45 Typical Temperature Distributions for 69BTH DSC (HAC Post Fire Cool-Down, HLZC#1, 26kW) *A.3-227*
Figure A.3-46 Typical Temperature Distributions for 69BTH DSC (HAC Post Fire Cool-Down, HLZC#4, 32kW) *A.3-228*
Figure A.3-47 Temperature Distributions for Fuel Assemblies in 69BTH DSC (HAC Post Fire Cool-Down) *A.3-229*
Figure A.3-48 Typical Temperature Distributions for 37PTH DSC (HAC Post Fire Cool-Down, 22 kW, Boral Plates) *A.3-230*
Figure A.3-49 Location of Damaged/Failed Fuel inside 24PTHF DSC *A.3-231*
Figure A.3-50 Location of Damaged/Failed Fuel inside 61BTHF DSC *A.3-232*
Figure A.3-51 Comparison of DSC Shell Temperature Profiles for 61BTH DSC *A.3-233*
Figure A.3-52 Comparison of DSC Shell Temperature Profiles for 24PTH DSC *A.3-234*
Figure A.3-53 Coupled Model of MP197HB TC and 69BTH DSC *A.3-235*
Figure A.3-54 Temperature Time Histories for Coupled Model, 69BTH DSC Components *A.3-236*
Figure A.3-55 Temperature Time Histories for Coupled Model, MP197HB TC Components *A.3-237*
Figure A.3-56 Temperature Time History for MP197HB Transport Cask HAC, 69BTH DSC, 32 kW, with External Fins (Load Case F1A) *A.3-238*
Figure A.3-57 Temperature Time History for MP197HB Transport Cask HAC, 69BTH DSC, 32 kW, with External Fins (Load Case F2A) *A.3-239*
Figure A.3-58 Temperature Time History for MP197HB Transport Cask HAC, 69BTH DSC, 26 kW, without External Fins (Load Case U1A) *A.3-240*
Figure A.3-59 Temperature Time History for MP197HB Transport Cask HAC, 69BTH DSC, 26 kW, without External Fins (Load Case U2A) *A.3-241*
Figure A.3-60 Temperature vs. Number of Elements *A.3-242*

Chapter A.3
Thermal Evaluation

NOTE: References in this Chapter are shown as [1], [2], etc. and refer to the reference list in Section A.3.5.

This chapter presents the thermal evaluations which demonstrate that the MP197HB transport cask (TC) meets thermal requirements of 10 CFR 71 [6] for transportation of BWR and PWR spent fuel assemblies (FA) within the following DSCs: 69BTH, 61BTH Type 1 and Type 2, 61BT, 37PTH, 32PTH, 32PTH Type 1, 32PTH1 Type 1 and Type 2, 32PT, 24PTH-S, 24PTH-L, 24PTH-S-LC, 24PT4, 61BTHF, and 24PTHF.

The maximum heat load per DSC allowed for transportation in MP197HB TC varies for different DSC types from 18.3 kW to 32 kW. The table below summarizes the maximum heat load per DSC for transportation in comparison with maximums allowed for storage.

Maximum Heat Load per DSC

DSC type	Max. Heat Load for Transport (kW)	Max. Heat Load for Storage (kW)
69BTH	26.0	*26.0 [3]*
	29.2	*29.2 [3]*
	30.2	*N/A*
	32.0	*32.0 [3]*
61BTH Type 1[(1)]	22.0	22.0 [3]
61BTH Type 2[(1)]	24.0	31.2 [3]
61BT	18.3	18.3 [1]
37PTH	22.0	N/A
32PTH/32PTH Type 1	26.0	34.8 [4]
32PTH1 Type 1	26.0	40.8 [3]
32PTH1 Type 2	24.0	31.2 [3]
32PT	24.0	24.0 [1]
24PTH-S or –L (w/ Al inserts)[(1)]	26.0	40.8 [1]
24PTH-S or –L (w/o Al inserts)[(1)]	26.0	31.2 [1]
24PTH-S-LC	24.0	24.0 [1]
24PT4	24.0	24.0 [2]

Note:
[(1)] See Section A.3.1.2 for heat loads of 61BTHF and 24PTHF DSCs.

For all DSC types, this evaluation demonstrates that DSC component temperatures are within material temperature limits and fuel cladding temperatures meet the thermal requirements of ISG-11 [7].

[

]

The NCT ambient temperature range is –20°F to 100°F (-29°C to 38°C) per 10 CFR 71.71(b) [6]. In general, all the thermal criteria are associated with maximum temperature limits and not minimum temperatures. All materials can be subjected to the minimum environment temperature of -40°F (-40°C) without adverse effects as required by 10 CFR 71.71 (c)(2) [6].

Thermal performance of MP197HB TC is evaluated based on finite element analyses using ANSYS computer code [27].

The following thermal design criteria are established for 69BTH and 37PTH DSCs to satisfy the functional and safety requirements.

- Maximum fuel cladding temperature limits of 752°F (400°C) for NCT and 1,058°F (570°C) for HAC are considered for the fuel assemblies with an inert cover gas as concluded in ISG-11 [7].
- Based on ISG-11 [7], the fuel cladding temperature is limited to 400°C (752°F) for short term operation such as vacuum drying.
- The maximum DSC cavity internal design pressures are summarized below:

Operating condition	69BTH DSC	37PTH DSC
Normal Conditions of Transport (NCT) (3% rods ruptured)	15 psig	15 psig
Hypothetical Accident Conditions (HAC) (100% rods ruptured)	*120* psig	140 psig

The thermal performance of 69BTH and 37PTH DSCs is evaluated based on finite element analyses using ANSYS computer code [27].

A.3.1.1 Design Features

A.3.1.1.1 MP197HB TC

The MP197HB transport cask includes optional features such as an aluminum internal sleeve to accommodate DSC types with outer diameters smaller than 69.75" and an aluminum shell with external circular fins. The external circular fins are used for heat loads greater than 26 kW. The TC design features for different DSC types considered for transportation in MP197HB TC are listed in the table below.

MP197HB TC Design Features

DSC Type	DSC OD (in.)	TC Sleeve	TC External Fins	Max. DSC Heat Load for Transport (kW)
69BTH	69.75	No	No	26.0
			Yes	29.2
			Yes	*30.2*
			Yes	32.0
61BTH Type 1 (1)	67.25	Yes	No	22.0
61BTH Type 2 (1)	67.25	Yes	No	24.0
61BT	67.25	Yes	No	18.3
37PTH	69.75	No	No	22.0
32PTH/32PTH Type 1	69.75	No	No	26.0
32PTH1 Type 1	69.75	No	No	26.0
32PTH1 Type 2	69.75	No	No	24.0
32PT	67.19	Yes	No	24.0
24PTH-S or -L (w/ Al inserts)(1)	67.19	Yes	No	26.0
24PTH-S or –L (w/o Al inserts)(1)	67.19	Yes	No	26.0
24PTH-S-LC (1)	67.19	Yes	No	24.0 (2)
24PT4	67.19	Yes	No	24.0

Notes:
(1) 61BTHF and 24PTHF DSCs have the same dimensions and use the same MP197HB features as DSC types 61BTH and 24PTH, respectively.
(2) The 24PTH-S-LC DSC is allowed for 24 kW heat load. The analyis assumes a heat load of 26 kW for conservatism.

The MP197HB TC consists of multiple shells which conduct the decay heat to the cask outer surface. The other thermal design feature of the cask is the conduction path created by the aluminum boxes that contain the neutron shielding material as described in Chapter A.5. The neutron shielding material is provided by a resin compound cast into long slender aluminum boxes placed around the gamma shield shell and enclosed within a steel shell (shield shell). The aluminum boxes are designed to fit tightly against the steel shell surfaces, thus improving the heat transfer across the neutron shield.

Heat dissipates from the packaging outer surfaces via natural convection and radiation. The outer surface of the shield shell is painted white to enhance the thermal radiation exchange with ambient.

The design of the steel-encased wood impact limiters is described in Chapter A.1, Section A.1.2. These components are included in the thermal analysis because of their contribution as a thermal insulator. The impact limiters provide protection to the lid and bottom regions from the external heat input due to fire during the HAC thermal event.

A personnel barrier prevents access to the outer surfaces of the cask body. The barrier, which consists of a stainless steel mesh attached to stainless steel tubing, encloses the cask body between the impact limiters, and have an open area fraction of approximately 80%.
The gaps considered in the thermal model of MP197HB TC are described in Section A.3.3.1.1.

The decay heat from the canister is carried to the inner shell of the MP197HB cask via conduction, convection, and radiation. Convection heat transfer in the annulus between the DSC outer shell and the cask inner shell is not considered in the thermal evaluation for conservatism.

A.3.1.1.4 Other DSC Types

The design features of DSC types 61BT, 32PT, and 24PTH are described in [1], Appendices K, M, and P, respectively. The design features of 24PT4 DSC are described in [2], Appendix A. The design features of DSC types 61BTH and 32PTH1 are described in Appendices T and U, respectively, associated with Amendment 10 to Part 72 CoC 1004 for the Standardized NUHOMS® System [3]. The design features of 32PTH and 32PTH Type 1 DSCs are described in [4].

A.3.1.2 Content's Decay Heat

The thermal analysis for MP197HB TC loaded with existing DSCs currently qualified for storage along with 69BTH and 37PTH DSCs is based on a range of maximum total heat load of 18.3 kW to 32 kW per DSC.

The maximum decay heat loads of 26.0, 29.2, *30.2,* and 32.0 kW are considered for transport of 69BTH DSC in combination with various design features of MP197HB TC listed in Section A.3.1.1.1.

The maximum decay heat loads for transport of DSC types 61BTH Type 1, 61BT, 32PT, and 24PT4 are equal to the maximum heat loads for storage conditions i.e., 22.0 kW, 18.3 kW, 24.0 kW, and 24.0 kW, respectively as shown in Section A.3.1.1.1 based on [3], [1], and [2].

The maximum decay heat load for transport of DSC types 32PTH, 32PTH Type 1, 32PTH1 Type 1, 24PTH-S, and 24PTH-L are considered equal to 26.0 kW.

The maximum decay heat load for 24PTH-S-LC is 24 kW [1]. Therefore, the results of 24PTH-S or -L (without aluminum inserts) DSC shell temperatures with 26 kW heat load is bounding for use with the 24PTH-S-LC DSC with 24 kW heat load.

The maximum decay heat load for transport of DSC types 61BTH Type 2 and 32PTH1 Type 2 are considered equal to 24.0 kW in order to maintain the maximum DSC shell temperature below the values reported for normal transfer conditions in Appendix T.4 and Appendix U.4 of the SAR associated with Amendment 10 to Part 72 CoC 1004 for the Standardized NUHOMS® System [3].

The maximum decay heat load for 37PTH DSC is equal to 22.0 kW for transport operations.

61BTHF and 24PTHF DSCs contain failed fuel and have lower heat loads than 61BTH and 24PTH DSCs, respectively. Nevertheless, it is considered in the thermal analysis that the heat loads for 61BTHF and 24PTHF DSCs are the same as those for 61BTH and 24PTH DSCs, respectively.

The MP197HB TC is designed to transport a payload of up to 56.0 tons of dry irradiated and/or contaminated non-fuel bearing solid materials in secondary containers. The decay heat load of the radioactive material is limited to 5 kW, which is well below the heat loads specified for the cask loaded with DSCs.

The permitted heat load zoning configurations (HLZC) for all DSCs are listed in Chapter A.1, Appendix 1.4.1 through Appendix 1.4.9. *Sections A.3.3 and A.3.4 present the thermal evaluation of the 69BTH DSC for HLZC # 1 through 4. Section A.3.6.1 presents the thermal evaluation of the 69BTH DSC for HLZC # 8.*These design basis HLZCs are symmetrical and show maximum allowable heat load per FA and per DSC, which result in bounding maximum fuel cladding and DSC component temperatures. Possible asymmetry in HLZC (within specified FA and DSC limits) means reduction of heat load in a particular FA resulting in reduction of local and maximum temperatures of fuel cladding and DSC components.

A peaking factor is considered along the active fuel length for calculation of the decay heat profile of the fuel assemblies as described in Section A.3.3.1.4 and Section A.3.3.1.6 for 69BTH and 37PTH DSCs, respectively.

[

]

A description of the detailed analyses of MP197HB TC, 69BTH and 37PTH DSCs for NCT and HAC is provided in Section A.3.3 and Section A.3.4, respectively. These analyses demonstrate that the maximum temperatures and pressures for other DSC types under transport conditions of 10 CFR 71 are bounded by the evaluations provided in [1], [2], [3], and [4] under storage/transfer conditions of 10 CFR 72.

Summaries of the maximum temperatures and pressures are provided in Section A.3.1.3 and Section A.3.1.4, respectively. The thermal evaluation concludes that for the maximum heat loads listed in Section A.3.1.1.1 all design criteria listed in Section A.3.1 are satisfied.

A.3.1.3 Summary Tables of Temperatures

The maximum and minimum MP197HB TC and DSC component temperatures for NCT are summarized in Table A.3–11. The component temperatures remain within the allowable range for NCT.

The maximum MP197HB TC and DSC component temperatures for cold conditions at -20°F and -40°F ambient without insolance are presented in Table A.3–12 and Table A.3–13. These temperatures are used for the structural evaluation of 69BTH and 37PTH DSCs.

69BTH basket with HLZC # 4 *and HLZC # 8*, only borated aluminum can be used as poison material. The HLZCs for 69BTH are described in Section A.3.3.1.4. *Sections A.3.3 and A.3.4 present the thermal evaluation of the 69BTH DSC for HLZC # 1 through 4. Section A.3.6.1 presents the thermal evaluation of the 69BTH DSC for HLZC # 8.*

A.3.2.2.3 37PTH DSC

The 37PTH DSC design allows the use of different neutron absorber materials. Boral plates paired with Al1100 plates or single plates of metal matrix composite (MMC) or borated aluminum can be used as poison materials in 37PTH basket with 22 kW heat load.

The following table summarizes the poison material configuration allowed for 69BTH and 37PTH basket designs:

Heat Load Zoning Configurations for 69BTH and 37PTH DSCs in MP197HB

	DSC type	HLZC	Poison Material	Max. Heat Load (kW)
1	69BTH	1 or 2	Boral/MMC/ Borated Aluminum	26.0
2	69BTH	3	Boral/MMC/ Borated Aluminum	29.2
3	69BTH	4	Borated Aluminum	32.0
4	*69BTH*	*8*	*Borated Aluminum*	*30.2*
5	37PTH	---	Boral Paired Al/ Single MMC/Borated Al	22

The neutron absorber materials in 69BTH and 37PTH baskets are subjected to the following minimum thermal conductivity, which are used in the canister thermal analyses.

Minimum Conductivity of Poison Material

Boral Core Matrix			
Temperature (°F)	Conductivity (W/cm-K) [26]	Conductivity (Btu/hr-in-°F)	90% Conductivity(1) (Btu/hr-in-°F)
100	0.859	4.136	3.723
500	0.768	3.698	3.328
Metal Matrix Composite (MMC)			
Temperature (°F)	Conductivity (Btu/min-in-°F),	Conductivity (Btu/hr-in-°F)	
All temperatures	0.0964 [3]	5.78	
Borated Aluminum			
Temperature (°F)	Conductivity (Btu/min-in-°F) [3]	Conductivity (Btu/hr-in-°F)	
68	0.123	7.38	
212	0.132	7.92	
392	0.141	8.46	
482	0.145	8.70	

Note: (1) The conductivity of Boral core is reduced by 10% and used in thermal analysis

fuel assembly. Substitution of these values in the above equation gives the reduction of the heat load for fuel assembly LaCrosse.

$$q_{LaCrosse} = q_{boundingFA} \frac{85}{144} \times 1.199 = 0.708\ q_{bundingFA}$$

The heat load for LaCrosse fuel assembly should be reduced to 70% of the heat load for bounding fuel assembly to maintain the 69BTH DSC temperatures at the same level calculated for the bounding fuel assembly.

Axial Heat Flux Profile

In-core data from an operating BWR facility forms the basis for the evaluation. The data provides the burnup and moderator density for 25 axial locations along the length of a fuel assembly. Five fuel assemblies at different locations in the reactor core are utilized to generate a burnup (peaking factor) distribution for the assembly. The resulting axial heat flux profile is shown in Figure T.4-37 associated with Amendment 10 to Part 72 CoC 1004 for the Standardized NUHOMS® System [3].

The heat flux profile shown in Figure T.4-37 [3] is used in the thermal model of the Transnuclear, Inc. TN-68 dry transport/storage cask [36], which can accommodate BWR spent fuel with a maximum bundle average burnup of 45,000 MWd/MTU. The maximum assembly average burnup allowed in 69BTH DSCs is *70*,000 MWd/MTU, which is considerably higher than 45,000 MWd/MTU for TN-68. Reference [12] shows that at a higher burnup, the heat flux shape tends to flatten with a reduction in the maximum axial peaking factor in the middle region, and the flux shape becomes more pronounced in the fuel end regions. The reduction of the maximum axial peaking factor in a more flattened heat flux shape will result in lower fuel cladding temperatures. Therefore, the application of a heat flux shape for a lower burnup spent fuel (45,000 MWd/MTU) on a higher burnup spent fuel (*70*,000 MWd/MTD) is conservative.

Axial decay heat profile for BWR fuel assemblies is the same as the one described in Appendix T, Section T.4.6.4 associated with Amendment 10 to Part 72 CoC 1004 for the Standardized NUHOMS® System [3]. The peaking factors from Appendix T, Section T.4.6.4 [3] are converted to match the regions defined for the fuel assemblies in 69BTH DSC/basket model. These peaking factors are listed in Table A.3–2 and are shown in the following figure.

		Z5	Z5	Z5	Z5	Z5		
	Z5	- (1)	Z3	Z2	Z3	-	Z5	
Z5	-	Z4	Z3	-	Z3	Z4	-	Z5
Z5	Z3	Z3	-	Z1	-	Z3	Z3	Z5
Z5	Z2	-	Z1	-	Z1	-	Z2	Z5
Z5	Z3	Z3	-	Z1	-	Z3	Z3	Z5
Z5	-	Z4	Z3	-	Z3	Z4	-	Z5
	Z5	-	Z3	Z2	Z3	-	Z5	
		Z5	Z5	Z5	Z5	Z5		

	Z1 (Zone 1)	Z2 (Zone 2)	Z3 (Zone 3)	Z4 (Zone 4)	Z5 (Zone 5)
Max. Decay Heat (kW/FA) (4)(5)	0.13	0.31	0.76	0.95	0.52
No. of Fuel Assemblies(2)	4	4	16	4	24
Max. Decay Heat per DSC (kW)	30.20 (3)(4)				

Notes:
(1) Aluminum dummy assembly (solid aluminum block) is required in the fuel compartments designated with a dash (-).
(2) Total number of fuel assemblies is 52 for HLZC No. 8.
(3) Adjust payload to maintain total DSC heat load within the specified limit.
(4) Reduce the maximum decay heat to 70% of the listed values for LACROSSE10 Fuel assembly. The total decay heat for LACROSSE10 fuel assembly is 21.1 kW per DSC for HLZC No. 8.
(5) Type "F" borated aluminum is the only poison material authorized to use with HLZC No. 8.
(6) This figure is also shown as Figure A.1.4.9-5a, "Heat Load Zoning Configuration No. 8 for 69BTH Basket."

Heat Load Zoning Configuration No. 8 for 69BTH Basket

where,
k_{poison} = conductivity of poison plate or conductivity of core material for Boral (Btu/hr-in-°F)
t_{poison} = thickness of poison plate or thickness of core material for Boral (in)
k_{Al} = conductivity of Al 1100 (Btu/hr-in-°F)
t_{Al} = thickness of aluminum plate or aluminum clad for Boral (in)
t_{model} = thickness paired Al/Poison plates in the model (in)

For conservatism, the conductivity of Boral core is reduced by 10% for calculation of effective conductivities.

The calculated effective conductivity values for paired aluminum and poison plates are listed in Section A.3.2.1 material # 17 to 19.

Borated aluminum plates can be used for all HLZCs in 69BTH basket. Boral or MMC plates paired with aluminum 1100 plates can be used for HLZC # 1, # 2, and # 3 but shall not be used for HLZC # 4 *and # 8*.

A comparison between material # 17 and 18 in Section A.3.2.1 shows that the effective conductivities for paired aluminum and MMC plates are higher than those for paired aluminum and Boral plates for the entire temperature range. Therefore, the effective conductivities of paired aluminum and Boral plates are considered to bound the maximum component temperatures for HLZC # 1, # 2, and # 3.

The effective conductivities for paired aluminum and borated aluminum plates are used for HLZC # 4 *and # 8*.

2) Effective Conductivity for Dummy Aluminum Assemblies

Aluminum dummy assemblies replace the fuel assemblies in assigned compartments defined in the 69BTH basket for HLZC # 2 through # 4, *and # 8* as shown in Section A.3.3.1.4.

The dummy assemblies are considered as aluminum blocks with a cross section of 5.875" × 5.875" and a length equal to BWR fuel assemblies. A uniform gap of 0.0625" is considered around the cross section of the dummy assembly within the fuel compartment.

The effective conductivity in transverse direction is a combination of serial and parallel thermal resistances shown in the following Figure A.3–19. The transverse effective conductivity for dummy assembly is calculated as follows.

$$k_{eff,tr,dummy} = \frac{1}{R_{eff,tr,dummy}}$$

with

$$R_{eff,tr,dummy} = 2R_{th,He1} + \frac{1}{\left(\frac{2}{R_{th,He2}} + \frac{1}{R_{th,Al}}\right)}$$

Due to low heat load of the secondary containers and expected low amount of fission gas release, the internal pressure of the cask loaded with the DSCs is bounding for the cask loaded with the secondary containers.

A.3.3.3.2 69BTH DSC Operating Pressure

The maximum internal pressure for the 69BTH DSC within MP197HB TC for NCT is determined based on the maximum allowable heat load for each HLZC discussed in Section A.3.3.1.4 and maximum assembly average burnup of *70* GWD/MTU. The limiting fuel assembly type considered in this evaluation is the FANP 9x9-2 assembly.

The calculations account for the canister free volume, the quantities of canister backfill gas, fuel rod fill gas, and fission products and the average canister cavity gas temperature. The internal canister pressures are then calculated using ideal gas law.

$$P_{can} = \frac{n_{total}\, R\, T_{avg,can}}{V_{free,can}}$$

Where:
n_{total} = Total number of moles of gases within canister cavity (lb-moles),
R = Universal gas constant (10.73 psia-ft^3/lb-moles-°R),
$T_{avg,can}$ = DSC cavity average temperature (°R),
$V_{free,can}$ = DSC free volume (ft^3), and
P_{can} = DSC internal pressure (psia).

The following assumptions and conservatisms are considered in calculating the maximum internal pressures within the 69BTH DSC:

- The DSC internal pressure is calculated below for the most limiting normal and accident cases for the 69BTH DSC. For these cases, 3% and 100% of the fuel rods are assumed to rupture, 100% of the fuel rod helium fill gas and 30% of the fission gases are assumed to be released into the DSC cavity [9]. The methodology for DSC internal pressure calculations is in accordance with the methodology used for the 61BTH DSC associated with Amendment 10 to Part 72 CoC 1004 for the Standardized NUHOMS® System [3].

- *The amount of fission gases produced by the "generic" BWR fuel assembly (FA) is calculated in Section A.5.5.4.*

- The DSC cavity average helium temperatures calculated *for* 69BTH DSC in Sections A.3.3.*2* and A.3.4 for limiting FANP 9x9-2 fuel assembly, are assumed to bound average helium temperatures for the rest of 69BTH fuel assemblies (FA).

- Similar to the 61BTH DSC, the 69BTH DSC is assumed to be backfilled with helium to a pressure of 3.5 (2.5 ± 1.0) psig after vacuum drying [3].

- An average temperature of 212°F equal to water boiling temperature is considered for the DSC shell during backfill operation.

- The initial temperature of fuel rod plenum fill gas is assumed to be at room temperature (70°F or 530°R). This is a reasonable assumption since the process takes place in a controlled environment.
- The highest assembly average burnup of *70* GWD/MTU proposed for the 69BTH DSC is assumed for the pressure calculation. Maximum burnup creates a bounding case for the amount of fission gases produced in the fuel rod during reactor operation.
- The highest weight and volume of fuel assembly was conservatively used because it displaces the most of free DSC cavity volume.
- The bounding (maximum) plenum volume among all fuel assemblies V_{pl}=2.136 in^3 (For SVEA FAs) is conservatively used in this calculation.
- The bounding (maximum) initial rod pressure ($p_{initial}$) of 160 psia (For SVEA FAs) is used.
- Only 98% of the DSC free cavity volume is used for pressure calculation.

1) Free DSC Cavity Volume

The volume of helium in the DSC cavity is calculated based on 69BTH weight calculations provided in Chapter A.2. The 69BTH DSC cavity free volume is calculated as DSC cavity volume minus basket, dummy assemblies (DA) and fuel assemblies' volume. The free volume for helium in the DSC is calculated as,

$$V_{He_DSC} = V_{He_DSC_w/o_FA} - V_{FAs} - V_{DAs}$$

Where,

V_{He_DSC} = Free helium volume in DSC

$V_{He_DSC_w/o_FA}$ = Helium volume outside fuel compartments in DSC cavity

V_{FAs} = Helium volume within compartment containing fuel assemblies

V_{DAs} = Helium volume within compartment containing dummy assemblies

The 69BTH DSC free helium volume and DSC cavity volumes used for pressure calculations are summarized in Table A.3–21 and Table A.3–22, respectively.

2) Average Helium Temperature

The average helium temperature in the DSC cavity is calculated based on DSC thermal analysis presented in Section A.3.3.1.4 and weight data provided in Chapter A.2 as volumetric average:

$$T_{He_DSC} = (V_{He_in_FA_act_fuel} \cdot T_{He_in_FA_act_fuel} + V_{He_in_DA_act_fuel} \cdot T_{He_in_DA_act_fuel} + V_{He_DSC_out_FA/DA_act_fuel} \cdot T_{He_DSC_out_FA/DA_act_fuel}) / V_{He_DSC}.$$

4) Quantity of Fill Gas in Fuel Rod

The volume of the helium fill gas in a fuel rod for the limiting (by number of fuel rods) SVEA fuel assemblies, at cold, unirradiated conditions is assumed to be 2.136 in^3 (maximum among all 69BTH fuels). There are at most 100 fuel rods in an SVEA fuel assembly with a maximum fill pressure of 145.3 psig (160 psia).

The fill gas is assumed to be at room temperature (70°F or 530°R). Per the ideal gas law, the quantity of fuel rod fill gas in 69 assemblies is:

$$n_{he} = \frac{(160\,psia)(6894.8\,Pa/psi)(69 \cdot 100 \cdot 2.136\,in^3)(1.6387 \cdot 10^{-5}\,m^3/in^3)}{(8.314\,J/(mol \cdot K))(530\,°R)(5/9\,K/°R)},$$

$$n_{he} = 108.91\,g-moles.$$

Based on 10 CFR 71 [6], the maximum fraction of the fuel pins that are assumed to rupture and release their charge gas for normal and accident conditions is 3% and 100%, respectively. Table A.3–22 summarizes the amount of helium fill gas released for the 69BTH DSC for NCT and HAC.

5) Quantity of Gases released as a Result of Irradiation

For the 69BTH DSC, the amount of fission gases produced by the "generic" BWR fuel assembly (FA) is 23.0 g-moles calculated in Section A.5.5.4. The total gas moles (per fuel assembly) are then multiplied by the 30% release fraction to obtain the number of moles contributing to 69BTH DSC cavity gas pressure.

The amount of gas released from one assembly due to irradiation is:

$n_{ig\,1\,FA}$ = *23.0 g-moles * 0.3 = 6.9 g-moles*

Table A.3–22 presents the amount of gas released into the DSC cavity by fuel assemblies n_{ig} for normal and accident conditions assuming a 30% gas release from the fuel pellets [9] and a 3% and 100% rod rupture percentage, respectively.

6) Maximum Normal Operating Pressure Calculation

Calculation of the maximum pressure in the 69BTH DSC HLZC #1 for normal conditions of transport is shown below. In accordance with [9], 3% of the fuel rods are assumed to be ruptured. The total amount of gas in the DSC cavity is therefore:

$$n_{DSC-NCT} = n_{he\ initial} + n_{he\ fuel\ rod\ release} + n_{ig},$$

$$n_{DSC-NCT} = 134.39 + 3.27 + 14.28 = 151.94\,g-moles$$

Ruptured pins will vent plenum gas until it comes into equilibrium with the DSC pressure; therefore, the plenum volume within the ruptured pins can be included in the total DSC internal volume. For a 3% pin rupture the additional volume is therefore:

$$V_{pin\ plenum} = 100\, pins/assy \cdot 2.136\, in^3/pin \cdot 69\, assy/basket \cdot 0.03 = 442.15\ \ in^3.$$

The maximum normal operating pressure (MNOP) for this configuration is then,

$$P_{DSC-NCT} = \frac{\left(1.4504 \cdot 10^{-4}\ \frac{psia}{Pa}\right)(151.94\, g-moles)(8.314\, J/(mol \cdot K))(984\, °R)(5/9\, K/°R)}{(258413\, in^3 + 442.15\, in^3)(1.6387 \cdot 10^{-5}\, m^3/in^3)},$$

$$P_{DSC-NCT} = 23.61\, psia\, (8.91\, psig).$$

The maximum pressures are summarized in Table A.3–22. As seen from Table A.3–22, the maximum internal pressures in the 69BTH DSC calculated based on thermal conditions are lower than the design pressures considered for the structural evaluation.

A.3.3.3.3 37PTH DSC Operating Pressure

The maximum internal pressure for the 37PTH DSC within the MP197HB TC for NCT is determined based on the maximum allowable heat load of 22 kW discussed in Section A.3.3.1.6 and maximum assembly average burnup of 62 GWD/MTU. Although BW 15x15 fuel assembly is not allowed for transport in the 37PTH DSC, it is considered as the limiting fuel assembly type in this evaluation for conservatism.

Calculation of Maximum Normal Operating Pressure inside the 37PTH DSC follows the same methodology as described for 69BTH DSC in Section A.3.3.3.2.

The following assumptions and conservatisms are considered in calculating the maximum internal pressures within the 37PTH DSC:

a) The DSC internal pressure is calculated for the most limiting normal, and accident cases. For these cases, 3% and 100% of the fuel rods are assumed to rupture, 100% of the fuel rod helium fill gas and 30% of the fission gases are assumed to be released into the DSC cavity [9]. The methodology for DSC internal pressure calculations is in accordance with the methodology used for the 32PTH1 associated with Amendment 10 to Part 72 CoC 1004 for the Standardized NUHOMS® System [3].

b) The following fuel assemblies are considered bounding for internal DSC pressure analyses performed for 32PTH1, 32PTH, 24PTH, and 24PT4 DSCs:
WE 15x15 - for 32PTH DSC [4],
BW 15x15 - for 24PTH DSC [1],
CE 16x16 – for 24PT4 DSC [2].
Since BW15x15 is bounding by amount of irradiation gases [30], it is selected for this analysis.

c) The 37PTH DSC is assumed to be backfilled with helium to a pressure of 2.5 ± 1.0 psig after vacuum drying similar to the 32PTH1 DSC [3].

ruptured for accident cases is 100%. The total quantity of moles released to 69BTH DSC cavity is, therefore,

$$n_{DSC-HAC} = n_{he\ initial} + n_{he\ fuel\ rod\ release,HAC} + n_{ig,HAC}$$

$$n_{DSC-HAC} = 134.39 + 108.91 + 476.10 = 719.40\, g - moles$$

The maximum accident operating pressure for this configuration is then,

$$P_{DSC-HAC} = \frac{\left(1.4504 \cdot 10^{-4} \frac{psia}{Pa}\right)(719.40\, g - moles)(8.314\, J/(mol \cdot K))(992\,°R)(5/9\, K/°R)}{(258415\, in^3 + 14738.40\, in^3)(1.6387 \cdot 10^{-5}\, m^3/in^3)}$$

$$P_{DSC-HAC} = 106.77\, psia\ (92.07\, psig).$$

The maximum internal pressures inside 37PTH and 69BTH DSCs are summarized in Table A.3–22. The maximum internal pressures are 102.64 psig and *96.16* psig for the 37PTH DSC and 69BTH DSC under HAC, respectively. These maximum calculated internal DSC pressures are below the design pressure*s* of *120 psig and* 140 psig specified in Section A.3.1 for HAC *for* the 69BTH and 37PTH DSCs, *respectively.*

Based on discussions in Section A.3.3.3.4, the maximum internal pressures for DSCs analyzed for accident storage/transfer conditions under 10 CFR 72 requirements remain bounding for transport conditions under HAC. The evaluated DSC internal pressures for DSCs to be transported within MP197HB TC are summarized in Table A.3–23. As seen in Table A.3–23, the maximum calculated internal pressures under HAC remains below the corresponding design pressures considered for the structural evaluations for HAC for all DSC types.

A.3.4.4 Maximum Thermal Stresses

Thermal stresses for the MP197HB TC loaded with DSCs are discussed in Chapter A.2.

A.3.4.5 Accident Conditions for Fissile Material Packages for Air Transport

The MP197HB TC is not designed for air transportation. Therefore, the accident conditions for air transport are irrelevant.

A.3.5 References

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A.3.6.1 Thermal Evaluation of MP197HB TC and 69BTH DSC for HLZC # 8 for NCT and HAC

This section presents the thermal evaluation of the 69BTH DSC with HLZC # 8 during the transport in the MP197HB TC. As shown in Figure A.1.4.9-5a, the maximum allowable heat load for HLZC # 8 is 30.2 kW and is bounded by the maximum heat load of 32 kW for HLZC # 4 shown in Figure A.1.4.9-5. Therefore, the thermal evaluation is limited to a sensitivity evaluation of the 69BTH DSC in MP197HB TC for the hot NCT.

The sensitivity study of the MP197HB TC loaded with the 69BTH DSC with HLZC # 8 for hot NCT is based on the methodology and thermal models presented in Section A.3.3. For the MP197HB TC, the thermal model described in Section A.3.3.1.1 with external fins is re-evaluated by modifying the heat load to 30.2 kW. Similarly, the thermal model for the 69BTH DSC in Section A.3.3.1.4 for HLZC # 4 is re-evaluated by modifying the heat generation rates for HLZC # 8 and the DSC shell temperature profile from the MP197HB TC thermal model with 30.2 kW heat load. No other changes are considered in these thermal models.

Normal Conditions of Transport

The maximum component temperatures for the MP197HB TC model with 30.2 kW heat load (HLZC # 8) during hot NCT are listed in table below.

Maximum Component Temperatures of MP197HB TC / 69BTH DSC Shell for Hot NCT (100 °F and Insolation)

	HLZC # 4 (1)	HLZC # 8
Heat Load	32 kW	30.2 kW
Component	T_{max} (°F)	
DSC shell	484	467
DSC shell @ mid-length (2)	470	454
Cask inner shell	367	354
Gamma shield	366	352
Outer shell	352	339
Shield shell	305	295
Finned Shell	229	223
Cask lid	267	259
Cask bottom plate	353	341
Neutron Shield Resin (3)	290	281
Trunnion Plug Resin (3)	277	268
Cask lid seal	268	260
Ram plate seal	352	339
Drain port seal @ bottom	351	339
Test seal @ bottom	349	337
Vent & test seal @ top	267	259
Wood in Impact limiter	302	292

(1) *The temperature data are from Table A.3-8 for HLZC # 4.*

(2) *This value is the maximum DSC shell temperature in the region where the fuel assemblies have the maximum peaking factor.*

(3) *This temperature is the volumetric, average temperature of the elements located at hottest cross section of the neutron shield resin / the trunnion plug resin.*

The average temperature of helium within the MP197HB TC cavity for hot NCT is listed in the table below.

Average Helium Temperatures of MP197HB TC for Hot NCT (100 °F and Insolation)

	HLZC # 4 (1)	HLZC # 8
Heat Load	32 kW	30.2 kW
Component	T_{avg} (°F)	
Helium in TC cavity	339	328

(1) The temperature is from Table A.3-14 for HLZC # 4.

Based on the comparisons in the above two tables, both the maximum and average temperatures of the MP197HB TC components for HLZC # 8 are bounded by those for HLZC # 4. Therefore, the MP197HB TC components are within temperature limits for their respective materials, and perform their intended safety function within the operating range for hot NCT of the 69BTH DSC with HLZC # 8.

The maximum fuel cladding and component temperatures for the 69BTH basket with HLZC # 8 are compared with those for HLZC # 4 for hot NCT in the table below.

Maximum Fuel Assemblies and 69BTH Basket Component Temperatures for Hot NCT

	HLZC # 4 (1)	HLZC # 8
Heat Load	32 kW	30.2 kW
Component	T_{max} (°F)	
Fuel Cladding	650	642
Basket (compartment)	612	594
Al / Poison Plate	612	593
Basket Rails	507	489

(1) The temperature data are from Table A.3-10 for HLZC # 4.

As shown in the table above, the maximum fuel cladding and 69BTH DSC component temperatures for HLZC # 8 are bounded by those for the HLZC # 4.

The average temperatures of fuel assemblies and helium within the DSC cavities are listed in the table below.

Average Fuel/Dummy Assemblies and 69BTH DSC Helium Temperatures for Hot NCT

HLZC	HLZC # 4 (1)	HLZC # 8
Heat Load	32 kW	30.2 kW
Component	T_{avg} (°F)	
Fuel Assemblies	547	533
Dummy Assemblies	558	534
Helium Elements (2)	432	413

(1) The temperature data are from Table A.3-14 and Table A.3-21 for HLZC # 4.

(2) This value is the volumetric, average temperature of the elements with helium properties in the basket model. In addition to the gaps, helium properties are considered for the elements within the fuel compartments located beyond the active fuel length and for the empty compartments

Based on the comparisons in the above two tables, both the maximum and average temperatures of fuel cladding and 69BTH DSC components for HLZC # 8 are bounded by those for HLZC # 4. Therefore, the 69BTH DSC components are within temperature limits for their respective materials, and perform their intended safety function within the operating range for hot NCT of the 69BTH DSC in the MP197HB TC with HLZC # 8.

Hypothetical Accident Conditions

As concluded above, all the temperatures of the 69BTH DSC with HLZC # 8 during hot NCT in the MP197HB TC are bounded by those computed for HLZC # 4. A similar behavior will also be observed during HAC since the thermal model of the 69BTH DSC with HLZC # 8 during HAC in the MP197HB TC is identical to that for HLZC # 4, except for the heat load in each zone. Therefore, the maximum temperatures of the 69BTH DSC with HLZC # 4 (32 kW) during HAC in the MP197HB TC remain bounding for HLZC # 8.

Internal Pressure in 69BTH DSC

As shown in the last table above, all the average temperatures within the 69BTH DSC with HLZC # 8 are lower than those computed for HLZC # 4. Therefore, the average helium temperature within the 69BTH DSC for HLZC # 4 remains bounding.

In addition to the average temperatures in the 69BTH DSC, a review of the HLZC # 4 and 8 in Figures A.1.4.9-5 and A.1.4.9-5a, respectively, shows that both configurations allow for the same number of fuel assemblies, i.e., 52. This ensures that the quantities of fill gases in fuel rods and fission gases for HLZC # 8 are the same as those for HLZC # 4. However, HLZC # 8 has one additional dummy fuel assembly in the center compartment; whereas, it remains empty in HLZC # 4 (identified as Zone 1 in Figure A.1.4.9-5). This small change does not have any significant impact on the internal pressure for HLZC # 8, as the small reduction in the DSC free volume due to the addition of one dummy fuel assembly will be offset by the lower initial helium back fill gas and the lower average temperatures discussed above.

Therefore, the internal pressures for the 69BTH DSC with HLZC # 4 for NCT and HAC in the MP197HB transport cask are still applicable for HLZC # 8.

Table A.3-22
Maximum Internal Pressures of 69BTH and 37PTH DSCs in MP197HB TC

Maximum Pressures for Normal Conditions of Transport

DSC	HLZC #	DSC cavity volume	Helium fill amount	Plenum volume	Plenum helium amount	CC gas amount	Fission products amount	Total gas amount	$T_{av\ He}$	Calculated Pressure	Design Pressure
		in^3	g-moles	in^3	g-moles	g-moles	g-moles	g-moles	°F	psig	psig
69BTH	1	258,413	134.4	*442*	3.3	0.0	*14.28*	*151.94*	524	*8.91*	15
	2	228,101	118.6	*391*	2.9	0.0	*12.63*	*134.14*	517	*8.76*	
	3	228,101	118.6	*391*	2.9	0.0	*12.63*	*134.14*	527	*8.98*	
	4	199,970	104.0	*333*	2.5	0.0	10.67	*117.22*	538	***9.17***	
37PTH	37 FAs w/CC	257,646	138.1	*306*	17.0	2.5	0.0	157.6	500	9.28	15

Maximum Pressures for Hypothetical Accident Conditions

DSC	HLZC #	DSC cavity volume	Helium fill amount	Plenum volume	Plenum helium amount	CC gas amount	Fission products amount	Total gas amount	$T_{av\ He}$	Calculated Pressure	Design Pressure
		in^3	g-moles	in^3	g-moles	g-moles	g-moles	g-moles	°F	psig	psig
69BTH	1	258,413	134.4	*14,738*	108.9	0.0	*476.40*	*719.40*	532	*92.07*	*120*
	2	228,101	118.6	*13,030*	96.3	0.0	*420.90*	*635.80*	525	*91.47*	
	3	228,101	118.6	*13,030*	96.3	0.0	*420.90*	*635.80*	568	***96.16***	
	4	199,970	104.0	*11,107*	82.1	0.0	*358.80*	*544.87*	581	*95.13*	
37PTH	37 FAs w/CC	257,646	138.1	*10,205*	568.1	82.9	0.0	789.2	514	102.64	140

Table A.3-23
Maximum Internal Pressure in DSCs for Transport in MP197HB TC

DSC	Operating conditions	Design Pressure psig	Calculated Pressure psig
69BTH	NCT	15	*9.17*
	HAC	*120*	*96.16*
61BTH/61BTHF Type 1	NCT	10	8.10
	HAC	65	56.10
61BTH Type 2	NCT	15	8.60
	HAC	120	68.70
61BT	NCT	10	9.56
	HAC	65	46.00
37PTH	NCT	15	9.28
	HAC	140	102.64
32PTH/32PTH Type 1	NCT	15	7.47
	HAC	120	91.00
32PTH1 Type 1/Type 2	NCT	15	10.74
	HAC	140	126.34
32PT	NCT	15	*8.09*
	HAC	*125*	*113.20*
24PTH/24PTHF (all types)	NCT	15	*8.66*
	HAC	120	*102.10*
24PT4	NCT	20	18.57
	HAC	100	*80.80*

Chapter A.5
Shielding Evaluation

TABLE OF CONTENTS

A.5.1 Description of the Shielding Design A.5-1b
A.5.1.1 Package Design Features A.5-1b
A.5.1.2 Codes and Standards A.5-2
A.5.1.3 Summary Table of Maximum Radiation Levels A.5-2
A.5.2 Source Specification A.5-3
A.5.2.1 Gamma Source A.5-4
A.5.2.2 Neutron Source A.5-7
A.5.2.3 Axial Source Distribution A.5-8
A.5.2.4 Axial Blankets A.5-10
A.5.2.5 HAC Sources A.5-11
A.5.3 Model Specification A.5-12
A.5.3.1 Configuration of Source and Shielding A.5-12
A.5.3.2 Material Properties A.5-14
A.5.4 Evaluation A.5-15
A.5.4.1 Methods A.5-15
A.5.4.2 Flux-to-Dose-Rate Conversion A.5-30a
A.5.4.3 Radiation Levels A.5-31
A.5.5 Appendix A.5-33
A.5.5.1 VYAL-B Mixing and Installation A.5-33
A.5.5.2 Fuel Qualification Results A.5-33
A.5.5.3 Decay Heat Restrictions A.5-35
A.5.5.4 Fission gas for BWR fuel at 70 GWd/MTU A.5-38a
A.5.6 References A.5-39

LIST OF TABLES

Table A.5-1 Summary of MP197HB NCT Dose Rates for Intact Fuel A.5-41
Table A.5-1a Summary of MP197HB NCT Dose Rates for Reconfigured Fuel, 50% Void Fraction..... A.5-41a
Table A.5-1b Summary of MP197HB NCT Dose Rates for Reconfigured Fuel, 50% Void Fraction, Additional Results A.5-41b
Table A.5-1c Summary of MP197HB NCT Dose Rates for Reconfigured Fuel, Reconfiguration Compliant with Enclosure 2 of the US NRC RIS 2015-XX [28] A.5-41c
Table A.5-2 Summary of MP197HB HAC Dose Rates A.5-42
Table A.5-2a Summary of MP197HB HAC Dose Rates for Reconfigured Fuel A.5-42a
Table A.5-3 Not used A.5-43
Table A.5-4 MP197HB Cask/69BTH DSC Shielding Materials A.5-44
Table A.5-5 Composition of VYAL-B A.5-45
Table A.5-6 DB PWR Fuel Assembly Material Mass A.5-46
Table A.5-7 DB BWR Fuel Assembly Material Mass A.5-47
Table A.5-8 TRITON Input Parameters A.5-48
Table A.5-9 Material Compositions for Fuel Assembly Hardware Materials A.5-49
Table A.5-10 BWR Axial Source Distributions for Intact Fuel A.5-50
Table A.5-10a BWR Axial Source Distributions for Reconfigured Fuel A.5-50
Table A.5-11 PWR Axial Source Distributions for Intact Fuel A.5-51
Table A.5-11a PWR Axial Source Distributions for Reconfigured Fuel A.5-51
Table A.5-12 Fuel Assembly Materials for MCNP A.5-52
Table A.5-13 Package Materials Input for MCNP A.5-54
Table A.5-14 Fuel Compositions for Burned Fuel Sensitivity Study A.5-55
Table A.5-15 Burned Fuel Sensitivity Study Dose Rate Results A.5-55
Table A.5-16 Flux-to-Dose Rate Conversion Factors for Gamma A.5-56
Table A.5-17 Flux-Dose-Rate Conversion Factors for Neutron A.5-57
Table A.5-18 CC Radiological Source A.5-58
Table A.5-19 Dry k_{eff} Results for the PWR DBFA A.5-59
Table A.5-20 Dry k_{eff} Results for the BWR DBFA A.5-60
Table A.5-21 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 24PT4 DSC A.5-61
Table A.5-22 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 24PTH DSC A.5-62
Table A.5-23 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 32PT DSC A.5-63
Table A.5-24 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 32PTH DSC A.5-64
Table A.5-25 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 32PTH1 DSC A.5-65
Table A.5-26 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 37PTH DSC A.5-66

Table A.5-27 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 61BT DSC A.5-67
Table A.5-28 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 61BTH Type 1 DSC A.5-68
Table A.5-29 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 61BTH Type 2 DSC A.5-69
Table A.5-30 Bounding Response Function NCT Dose Rates at 2 Meters from Surface of the Cask Containing 69BTH DSC A.5-70
Table A.5-31 PWR Applied keff A.5-71
Table A.5-32 BWR Applied keff A.5-72
Table A.5-33 Matrix Showing Additional to the Transportation FQTs Cooling Times for Selected Cooling Times of CCs A.5-73
Table A.5-34 Summary of Maximum Dose Rates of the Cask Containing the Radioactive Waste Canister A.5-74
Table A.5-35 *Amount of Moles of Gases released as Result of Irradiation for GE7x7 BWR Fuel Assembly at 0.198 MTU, 3.7 wt % U-235, and 70 GWd/MTU* A.5-75
Table A.5-36 Bounding NCT Source for the 37PTH DSC in the MP197HB Transportation Package, Intact Fuel A.5-76
Table A.5-37 Bounding NCT Source for the 69BTH DSC in the MP197HB Transportation Package, Intact Fuel A.5-77
Table A.5-38 Bounding NCT and HAC Source for the 37PTH DSC, Reconfigured Fuel, 21 Inner Compartments A.5-78
Table A.5-38a Bounding NCT and HAC Source for the 37PTH DSC, Reconfigured Fuel, 16 Peripheral Compartments A.5-78
Table A.5-39 Bounding NCT and HAC Source for the 69BTH DSC, Reconfigured Fuel, 45 Inner Compartments A.5-79
Table A.5-39a Bounding NCT and HAC Source for the 69BTH DSC, Reconfigured Fuel, 24 Peripheral Compartments A.5-79
Table A.5-39b Bounding NCT and HAC Source for the 69BTH DSC, Reconfigured Fuel, 24 Peripheral Compartments A.5-79a
Table A.5-40 Photon Dose Rates Computed with C/M Average minus Standard Deviation A.5-80
Table A.5-40a Neutron Dose Rates Computed with Minimum C/M Ratios A.5-80aa
Table A.5-40b Cm-244 Samples Available for Low Burnup Fuel A.5-80ab
Table A.5-40c Cm-244 Samples Available for High Burnup Fuel A.5-80ac
Table A.5-41 Number of Samples Used for Each Isotope A.5-80a
Table A.5-41a Summary of Experimental Samples as a Function of Burnup Range A.5-80a
Table A.5-42 C/M Results from NUREG/CR-7012 ..... A.5-80b
Table A.5-43 C/M Results from NUREG/CR-7013 ..... A.5-80c
Table A.5-44 C/M Results from NUREG/CR-6968 ..... A.5-80d
Table A.5-45 Experimental Error Applied to the Isotopic C/M Data A.5-80f
Table A.5-46 Dose Rate Contribution for Isotopes of Interest (mrem/hr) A.5-80g
Table A.5-47 Not used A.5-80h
Table A.5-48 Selected TN-68 Dose Rates A.5-80i

Table A.5-49 MCNP Benchmark Results for HSM Model 80 ..... A.5-80j

Table A.5-50 ORIGEN-ARP Input Parameters for the Design Basis Sources A.5-80k

Table A.5-51 D_{NCI} Neutron Dose Rate Values for Various Active Fuel Compressions A.5-80l

Table A.5-52 "B" Parameters to Determine Additional Cooling Time for Reconfigured BWR Fuel (years) A.5-80m

Table A.5-53 "B" Parameters to Determine Additional Cooling Time for Reconfigured PWR Fuel (years) A.5-80n

Table A.5-54 Key Parameters of Compressed Fuel for 69BTH DSC Fuel Reconfiguration A.5-80o

Table A.5-55 NCT Reconfigured Fuel Dose Rates for MP197HB Containing 69BTH DSC with Compressed Fuel Configuration, 41% Void Fraction by Volume (30% Fuel Compaction) A.5-80p

Table A.5-56 NCT Reconfigured Fuel Dose Rates for MP197HB Containing 69BTH DSC with Compressed Fuel Configuration, 17% Void Fraction by Volume (50% Fuel Compaction) A.5-80q

Table A.5-57 HAC 1 Meter Dose Rates of MP197HB Containing 69BTH DSC with Compressed Fuel Configuration, 41% Void Fraction by Volume (30% Fuel Compaction) A.5-80r

Table A.5-58 HAC 1 Meter Dose Rates of MP197HB Containing 69BTH DSC with Compressed Fuel Configuration, 17% Void Fraction by Volume (50% Fuel Compaction) A.5-80s

Table A.5-59 69BTH NCT 17% Compressed Fuel – Neutron and Secondary Gamma – Burned Fuel Composition and MCNP Fission Neutron Multiplication Evaluation – Design Basis Source 2.6 wt. % U-235 / 62 GWd/MTU A.5-80t

Table A.5-60 69BTH NCT 17% Compressed Fuel – Neutron and Secondary Gamma – Burned Fuel Composition and MCNP Fission Neutron Multiplication Evaluation – High Enriched and High Burnup Fuel (5.0 wt. % U-235 / 70 GWd/MTU) A.5-80t

LIST OF FIGURES

Figure A.5-1 MP197HB Transport Cask with 69BTH DSC Model, Axial View A.5-81
Figure A.5-2 MP197HB Transport Cask with 69BTH DSC Model, Top View Showing Cask Lid with Gap, Top Nozzle, and Plenum A.5-82
Figure A.5-3 MP197HB Transport Cask with 69BTH DSC Model, Bottom View Showing Cask Bottom and Bottom Nozzle A.5-83
Figure A.5-4 MP197HB within 69BTH DSC Model, Cross Section View A.5-84
Figure A.5-5 Details of DSC Basket with Fuel Lattice Unit Cell in MCNP Model A.5-85
Figure A.5-6 MP197HB Transport Cask within 69 BTH DSC: MCNP Model Cut-through XY Plane (Z=25.12 cm), Normal Condition A.5-86
Figure A.5-7 MP197HB Bottom and Top Trunnion Attachment Block and Plug Geometry, Normal Condition A.5-87
Figure A.5-8 69BTH DSC and MP197HB Bottom Plugs with Grapple Ring Cut-Out A.5-88
Figure A.5-9 MP197HB Shear Key A.5-89
Figure A.5-10 Dose Rate Location Terminology A.5-90
Figure A.5-11 24PT4 DSC Radial Zones A.5-91
Figure A.5-12 24PTH and 24PTHF DSC Radial Zones A.5-91
Figure A.5-13 32PT, 32PTH, and 32PTH1 DSC Radial Zones A.5-92
Figure A.5-14 37PTH DSC Radial Zones A.5-92
Figure A.5-15 61BT, 61BTH (Type 1 and Type 2) and 61BTHF DSC Radial Zones A.5-93
Figure A.5-16 69BTH DSC Radial Zones A.5-94
Figure A.5-17 MCNP vs. Response Function for 37PTH DSC A.5-95
Figure A.5-18 MCNP vs. Response Function for 69BTH DSC A.5-96
Figure A.5-19 Peaking Factors for BWR fuel A.5-97
Figure A.5-20 Peaking Factors for PWR fuel A.5-98
Figure A.5-21 37PTH DSC Neutron Fraction of Total Dose Rate A.5-99
Figure A.5-22 69BTH DSC Neutron Fraction of Total Dose Rate A.5-100
Figure A.5-23 NCT MCNP Models (x-y View) A.5-101
Figure A.5-24 NCT 69BTH DSC MCNP Models (x-z View) A.5-102
Figure A.5-24a NCT 69BTH DSC MCNP Model for Heat Load Zone Configuration No. 8 with High-Burnup Reconfigured Fuel in Zone 5 (x-y and x-z views) *A.5-102a*
Figure A.5-25 NCT 37PTH DSC MCNP Models (x-z View) A.5-103
Figure A.5-26 HAC MCNP Models for Reconfigured Fuel (x-z View) A.5-104
Figure A.5-27 NCT Configurations for *Radial* Fuel Reconfiguration (x-y View) A.5-105
Figure A.5-28 Peripheral and Inner Fuel Locations for the 24PT4 and 24PTH DSCs A.5-106
Figure A.5-30 Correction Factors vs. Burnup–Cm-244 ..... A.5-108
Figure A.5-31 Correction Factors vs. Burnup–Cs-134 ..... A.5-108

Chapter A.5
Shielding Evaluation

NOTE: References in this chapter are shown as [1], [2], etc. and refer to the reference list in Section A.5.6.

This chapter describes the shielding evaluation of the NUHOMS® MP197HB transportation package. The MCNP computer program is used to calculate the dose rates using a detailed three-dimensional model [1]. The source terms are generated with TRITON/ORIGEN-ARP sequence of SCALE [2]. The dose rates are in compliance with the applicable requirements of 10 CFR Part 71 for exclusive-use transportation in an open transport vehicle [3].

[

]

A full discussion and description of the models used in the shielding evaluation is contained in Section A.5.3.1.

Material properties used in the evaluation of the MP197HB transportation package are described in Section A.5.3.2.

A.5.1.2 Codes and Standards

ANSI/ANS-6.1.1-1977 flux-to-dose rate conversion factors are utilized for both gamma and neutron radiation [4]. These factors are provided in Table A.5-16 and Table A.5-17, respectively.

A.5.1.3 Summary Table of Maximum Radiation Levels

A.5.1.3.1 Regulatory Limits

The dose rates limits for the transportation of the MP197HB package, in an open vehicle, are obtained from 10 CFR 71.47(b) and 10 CFR 71.51(a)(2) and are listed as follows [3].

- Dose rate at any point on the external surface of the package under normal conditions is 200 mrem/hr (maximum).
- Dose rate at any point on the vertical planes projected from the outer edges, including the top and underside, under normal conditions is 200 mrem/hr (maximum).
- Dose rate at any point 2 m from the vertical planes projected from the outer edges, excluding the top and underside, under normal conditions is 10 mrem/hr (maximum).
- External dose rate at any point 1 m from the surface of the package under hypothetical accident conditions is 1000 mrem/hr (maximum).

A.5.1.3.2 Maxima

Normal conditions of transport (NCT) dose rates are computed for exclusive-use transport in an open vehicle. For the purposes of the shielding evaluation, the package surfaces, as shown in drawing MP197HB-71-1001, are conservatively assumed to be the vehicle surfaces. The package is assumed to be as wide as the open vehicle. Either end of the vehicle is assumed to be at the end of the impact limiters. The underside (floor) and top of the vehicle are assumed to correspond to the radius of the impact limiters. The package is assumed to be under the control of a private carrier and any operator in an occupied position will wear a dosimeter and be subject to a dose program to satisfy the requirements of 10 CFR 20.1502.

The maximum radiation dose rates for intact fuel during NCT are provided in Table A.5-1. The 37PTH DSC dose rates bound all PWR-type DSCs, and the 69BTH DSC dose rates bound all BWR-type DSCs. Dose rates are computed to bound all authorized burnup and enrichment combinations defined in the fuel qualification tables in Chapter A.1.

The maximum radiation dose rates for reconfigured fuel during NCT are provided in Table A.5-1a *and Table A.5-1b.* **[**

] In the reconfigured fuel models *for Table A.5-1a and Table A.5-1b,* the active fuel length is compressed until 50% void volume is reached. Fifty percent void volume is considered the design basis reconfiguration. Fuel reconfiguration increases the dose rates compared to the intact fuel models. To reduce the dose rates, in the reconfigured fuel models a

two-zone loading approach is utilized. Fuel in the inner zone is governed by the FQT cooling times, while fuel in the peripheral zone is cooled for a longer time. The FQT methodologies in the appendices of Chapter A.1 are modified to reflect the two-zone loading requirement.

Hypothetical accident condition (HAC) dose rates are calculated 1 m from the surface of the cask body. No credit is taken for the neutron shield or impact limiters. Both intact and reconfigured fuel are considered. The maximum radiation dose rates for HAC are shown in Table A.5-2 **[** **]** *and Table A.5-2a* **[** **]**

Dose rates for RWC are provided in Table A.5-34. This table contains both NCT and HAC dose rate results. Compared to spent fuel, RWC dose rates are low.

A.5.2 Source Specification

There are five principal sources of radiation associated with transport of spent nuclear fuel that are of concern for radiation protection.

1. Primary gamma radiation from spent fuel.
2. Primary neutron radiation from spent fuel (both alpha-n reactions and spontaneous fission).
3. Gamma radiation from activated fuel structural materials and fuel inserts.
4. Capture gamma radiation produced by attenuation of neutrons by shielding material of the cask.
5. Neutrons produced by sub-critical multiplication in the fuel.

There are three source configurations used in the evaluation of the shielding performance of the MP197HB transportation package. These configurations are selected because of their respective bounding parameters on all authorized contents. The bounding configurations are as follows:

- 8,182 A_2 (90,000 Ci of Co-60) in the RWC,
- 69 GE-2,3 7x7 Type G2A BWR spent fuel assemblies in the 69BTH DSC, and
- 37 B&W 15x15 Mark B-10 PWR spent fuel assemblies in the 37PTH DSC.

For the spent fuel assemblies listed, design basis sources which encompass the allowable burnup and enrichment combinations for the authorized contents are developed in the following subsections. The spent fuel assembly types are selected as bounding mainly because their respective initial uranium loading bounds all others.

For the 37PTH and 69BTH DSCs in the MP197HB transportation package, design basis BWR and PWR spent fuel sources are developed based on a bounding assembly average burnup, initial enrichment, and cooling time. These parameters are selected based on the fuel qualification method discussed in Section A.5.4.1.3. The B&W 15x15 Mark B10 and the GE-2, 3 7x7 Type G2A fuel assemblies contain the maximum heavy metal weight for their type, nearly 490 and 198 kgU, respectively. They result in bounding neutron and gamma source terms for PWR and BWR type of assemblies, respectively. Therefore, B&W 15x15 Mark B10 and the GE 2, 3 7x7 are evaluated as the design basis (DB) PWR and BWR fuel assembly (FA) in the shielding evaluation of MP197HB transportation package, respectively.

The fuel assembly hardware for the GE 7x7 fuel assembly is bounding as it contains the maximum amount of steel and inconel than any other BWR fuel design. For the PWR fuel assembly designs with cooling times greater than 15 years, the contribution from the fuel assembly hardware becomes less important due to substantial decay of the Co-60 source from hardware irradiation.

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A.5.2.1.3 69BTH

The spent fuel payload consists of various DSCs with BWR fuel assemblies with or without channels and is specified in Chapter A.1, Appendix A.1.4.7 through A.1.4.9. The source term calculations for DSCs with the BWR fuel payload include the contribution from the channel (Table A.5-8) while the shielding calculations do not take credit for them. This represents conservatism in the gamma dose rate calculations by approximately 15% for fuel assemblies with channels (typically the most representative of all loaded BWR fuel assemblies).

The NCT BWR design basis gamma source for intact fuel is shown in Table A.5-37. This source is calculated at a burnup of 59 GWd/MTU, an initial enrichment of 2.8 wt. % U-235, and a cooling time of 13.7 years. This table includes the gamma contribution of all four homogenized fuel assembly regions.

The bounding BWR design basis gamma source terms for reconfigured fuel are shown in Table A.5-39 and Table A.5-39a for the inner and peripheral zones, respectively. These source terms are used in both NCT and HAC calculations that utilize reconfigured fuel. The inner zone consists of the inner 45 fuel assemblies, and the peripheral zone consists of the outer 24 fuel assemblies (see Figure A.5-29c). Both sources are calculated at a burnup of 62 GWd/MTU and an initial enrichment of 2.6 wt. %. Fuel with a cooling time of 18.5 years is placed in the inner zone, while fuel with a cooling time of 35.0 years is placed in the peripheral zone.

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It is demonstrated in Section A.5.4.1.3.3 that *reconfigured* fuel placed in the peripheral zone requires additional cooling time for some BECT combinations in order to meet the NCT and HAC dose rate limits. For a burnup of 62 GWd/MTU and an initial enrichment of 2.6 wt.%, Table A.5-52 indicates that *10.5* years of additional cooling time is needed in the peripheral zone. Because the inner zone contains fuel with a cooling time of 18.5 years, the minimum cooling time in the outer zone is then 18.5+*10.5* = *29.0* years.

A.5.2.1.6 Reconstituted Fuel Assemblies

Reconstituted fuel assemblies are those where one or more fuel rods are replaced with non-fuel rods that displace the same amount of moderator in the active fuel region. Table A.5-6 of the SAR provides material details of a reconstituted fuel assembly where the rods are replaced with solid stainless steel after one cycle of irradiation. This assembly undergoes two additional cycles of irradiation where the source terms of the original and reconstituted fuel assemblies are compared. The summary of these evaluations is discussed in Section A.5.5.2.

A.5.2.2 Neutron Source

The neutron source energy distribution used in the shielding analysis is based on the Cm-244 Watt fission spectrum in the MCNP models. Cm-244 spontaneous fission represents approximately 90% of the neutron source in the active fuel region of the used nuclear fuel. It can be expressed in the following relationship:

$$p(E) \sim \exp\left(\frac{-E}{a}\right)\sinh\sqrt{bE} \ ,$$

where $a = 0.906\,\text{MeV}$ and $b = 3.8486\,(\text{MeV})^{-1}$ [1].

The strength of the neutron sources used in the shielding analysis was calculated using ORIGEN-ARP.

Sub-critical multiplication of the neutron and (n,γ) source terms was developed according to the methodology in Section A.5.4.1.1.4. Neutron peaking factors to account for the burnup dependent, neutron axial profile were developed and discussed in Section A.5.4.1.1.5.

A.5.2.2.1 37PTH

The intact fuel PWR NCT neutron source in the 37PTH DSC is provided in Table A.5-36. This source is calculated at a burnup of 60 GWd/MTU, an initial enrichment of 3.9 wt. % U-235, and a cooling time of 15.7 years. The k_{eff} value used to account for subcritical neutron multiplication is 0.2678 and the peaking factor employed is 1.152.

In calculations with reconfigured fuel, the neutron source term is dominant. The source in the inner zone corresponds to the largest FQT neutron source. The PWR design basis neutron source terms for reconfigured fuel are shown in Table A.5-38 and Table A.5-38a for the inner and peripheral zones, respectively. These source terms are used in both NCT and HAC calculations that utilize reconfigured fuel. The inner zone consists of the inner 21 fuel assemblies, while the peripheral zone consists of the outer 16 fuel assemblies (see Figure A.5-29a). Both sources are calculated at a burnup of 62 GWd/MTU and an initial enrichment of 3.4 wt. %. Fuel with a cooling time of 20.7 years is placed in the inner zone, while fuel with a cooling time of 29.5 years is placed in the peripheral zone. The k_{eff} value used to account for subcritical neutron multiplication is 0.2812 and the peaking factor employed is 1.152.

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A.5.2.2.2 69BTH

The intact fuel BWR NCT neutron source in the 69BTH DSC is provided in Table A.5-37. This source is calculated at a burnup of 59 GWd/MTU, an initial enrichment of 2.8 wt. % U-235, and a cooling time of 13.7 years. The k_{eff} value used to account for subcritical neutron multiplication is 0.1699 and the peaking factor employed is 1.212.

In calculations with reconfigured fuel, the neutron source term is dominant. The source in the inner zone corresponds to the largest FQT neutron source. The BWR design basis neutron source terms for reconfigured fuel are shown in Table A.5-39 and Table A.5-39a for the inner and peripheral zones, respectively. These source terms are used in both NCT and HAC calculations that utilize reconfigured fuel. The inner zone consists of the inner 45 fuel assemblies, while the peripheral zone consists of the outer 24 fuel assemblies (see Figure A.5-29c). Both sources are calculated at a burnup of 62 GWd/MTU and an initial enrichment of 2.6 wt. %. Fuel with a cooling time of 18.5 years is placed in the inner zone, while fuel with a cooling time of 35.0 years is placed in the peripheral zone. [

] The k_{eff} value used to account for subcritical neutron multiplication is 0.1858 and the peaking factor employed is 1.212.

A.5.2.3 Axial Source Distribution

Source terms are developed for configurations with spent fuel assemblies in two dimensions using the ORIGEN-ARP sequence in SCALE. Axial profiles and peaking factors are employed to transform the source into three dimensions. The peaking factor multiplied by the source strength calculated in two dimensions is applied to the three dimensional MCNP model as the total source

fuel assemblies with large (>5% active fuel length) natural uranium blankets, the fuel qualification is based on the maximum burnup instead of the assembly average burnup.

A.5.2.5 HAC Sources

There are several types of DSCs considered for loading in the transportation cask. The FQTs for each payload are defined so that the total NCT dose rate for intact fuel does not exceed the specified dose rate limit of 8.2 mrem/hr. Therefore, any source from any FQT resulting in the specified dose rate limit of 8.2 mrem/hr can be used to perform the intact fuel NCT shielding analysis. However, the bounding fractions of the total NCT dose rate at the location of interest (2 m from the vehicle) due to the neutron source are different for different BECT combinations, as shown in Figure A.5-21 for the 37PTH DSC and Figure A.5-22 for the 69BTH DSC.

Therefore, for the case when neutron shielding is completely lost or severely degraded during HAC, one would have to maximize D^n for Equation (A.5.29), not just from any FQT but from the FQT having the largest D^n. Among all the radiological sources from all the BECTs in all the FQTs, the radiological source corresponding to 62 GWd/MTU, 2.6 wt. %, 18.5 years cooling in the FQT for the 69BTH DSC payload results in the largest neutron fraction.

This source is appropriate both for HAC analysis and NCT reconfiguration analysis because the neutron component also becomes dominant in the NCT reconfiguration analysis. To reduce the dose rates it is necessary to use a zoned loading configuration. The source described above is used in the inner 45 compartments of the 69BTH DSC, while fuel with the same enrichment and burnup cooled to 35.0 years is placed in the 24 peripheral locations. The 69BTH HAC source terms are provided in Table A.5-39 and Table A.5-39a.

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For the 37PTH DSC, the bounding neutron source occurs at a burnup of 62 GWd/MTU, an initial enrichment of 3.4 wt. % U-235, and a cooling time of 20.7 years. As with the 69BTH DSC, a two-zoned loading is utilized to reduce the dose rates. The inner 21 compartments utilize the maximum neutron source, and the peripheral 16 compartments utilize fuel with the same enrichment and burnup cooled to 29.5 years. The 37PTH HAC source terms are provided in Table A.5-38 and Table A.5-38a.

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Alternatively, the response functions can be thought of as a 2D matrix which converts a source vector to a dose vector. The components of the dose vector are the dose rate values at the various angular and axial locations around the transportation package. The fuel qualification methodology is to select sources such that the largest component of the dose rate vector does not exceed 8.2 mrem/hr.

Results of the fuel qualification methodology are shown in Section A.5.5.2 for the MP197HB transportation package.

A.5.4.1.3.2 Source

The source portion of the fuel qualification method first requires that the proposed contents be known. Specifically, all the physical parameters of the fuel assembly, included isotopic composition of all the materials, should be well understood. The design basis assembly is depleted using the TRITON/T-DEPL sequence of SCALE to create an ORIGEN-ARP library for that specific assembly type. As the assembly is burned to generate the source, operational parameters such as assembly specific power and duration of cycle are controlled to bound the source as well as get the appropriate burnup.

The result of the application of this portion of the fuel qualification method is a primary gamma source for the four homogenized burned assembly regions as described in Section A.5.4.1.1.1, a primary neutron source, and a (n,γ) source.

A.5.4.1.3.3 Additional Cooling Time for Reconfigured Fuel

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Withheld Pursuant to 10 CFR 2.390

A.5.4.1.3.4 Cooling Time Derivation for 69BTH DSC using Heat Load Zone Configuration 8

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A series of verification cases are run for both NCT and HAC configurations using Heat Load Zone Configuration No. 8 to ensure that FQT based cooling times comply with NCT and HAC dose rate limits.

A.5.4.1.4 Dose Calculation Methodology

MCNP5 v1.40 is used for the shielding analysis [1]. MCNP5 is a standard, well-accepted shielding program utilized to compute dose rates for radiation protection. A three-dimensional model is developed that captures all of the relevant design parameters of the package. Dose rates are calculated by tallying the neutron and gamma fluxes over surfaces of interest and converting

The transportation FQTs are, therefore, a set of acceptable combinations of burnup, enrichment, and cooling times such that the expected NCT dose rates at two meters from the cask are the same, regardless of the canister type contained inside of the cask. Therefore, no bounding set of NCT source terms are generated – rather all the source terms are expected to result in similar dose rates – bounded only by the maximum dose rate limits.

The transportation FQT dose rates for all bounding authorized contents are shown in Table A.5-21 through Table A.5-30 for the authorized DSCs in the MP197HB transportation package as described in Section A.1.2.3 and Appendices A.1.4.1 through A.1.4.9. *The transportation FQT for the 69BTH using Heat Load Zone Configuration 8 is shown in Table A.1.4.9-5b of Section A.1.4.9.*

A.5.5.2.1 Fuel Qualification Adjustment for Control Components

The spent fuel assemblies transported in the cask may contain irradiated fuel inserts (BPRA, TPA, etc.). They are referred to as control components (CCs). Control components are also allowed to be stored or transported along with fuel assemblies. CCs include burnable poison rod assemblies (BPRAs), thimble plugs, neutron sources, control rods, etc. It is assumed in the assemblies' qualification that they are not necessarily bound to any specific fuel assembly and can be removed from the assembly in which they were generated. It is expected that, for example, a 3 year cooled assembly can be mixed with a 13 year cooled CC or 17 years cooled FA can be mixed with 15 years cooled CCs.

From the shielding stand point, it is permissible to place CCs in any number of the fuel compartments not in a periphery, in any DSC Transport Cask shielding configuration. CCs in those compartments may affect a shape of the dose rates distribution near the transportation package but they will have a negligible effect on the dose rate maximums at the location of interest. When the CC sources are in outer peripheral DSCs' fuel compartments, the dose rate increase is entirely due radiation in 1.00 to 1.66 MeV energy range.

The following guidelines should be applied when fuel assemblies with CCs are considered for transportation in MP197HB transportation cask.

- CCs include burnable poison rod assemblies (BPRAs), thimble plugs, neutron sources, control rods, etc.
- CCs are not necessarily bounded to any specific fuel assembly and can be removed from the assembly in which they were generated. It is expected that, for example, a 3 year cooled assembly can be mixed with a 13 year cooled CC or 17 years cooled FA can be mixed with 15 years cooled CC, etc.
- The maximum number of fuel assemblies with CCs that can be loaded per DSC is equal to the number of assemblies per DSC.
- Additional cooling time is required for assemblies only in peripheral locations of the DSCs. There are 12 peripheral compartments in DSCs with 24 FA locations and 16 peripheral compartments in DSCs with 32 or 37 locations. Number of peripheral fuel compartments in DSCs containing BWR FAs is 24. The peripheral fuel compartments are shown in Figures A.5-28 through A.5-29c.
- No additional cooling time is required for assemblies with CCs in 24PT4 DSC because the transportation FQT already accounts for presence of 24PT4 design basis CC sources.

A.5.5.3.1 BWR Decay Heat Equation

A nonlinear regression analysis was performed to obtain a fit of the BWR decay heat data as a function of burnup, initial enrichment, and cooling time. A very good fit was obtained based on an iterative evaluation using a 9-parameter model. The functional form is expressed below.

The decay heat (DH) in watts is expressed as:

$$DH_{BWR} = F_1 \exp\left[G\left(1-\frac{1.2}{x_3}\right)(x_3-4.5)^H\left(\frac{x_2}{x_1}\right)^I\right]+10,$$

where,

- $F_1 = A + Bx_1 + Cx_2 + Dx_1^2 + Ex_1x_2 + Fx_2^2$,
- x_1 equals the assembly average burnup in GWd/MTU,
- x_2 equals the initial enrichment in wt. % U-235,
- x_3 equals the cooling time in years,
- $A = -59.1$,
- $B = 23.4$,
- $C = -21.1$,
- $D = 0.280$,
- $E = -3.52$,
- $F = 12.4$,
- $G = -0.720$,
- $H = 0.157$, and
- $I = -0.132$.

The calculation uncertainty is 10 watts. It is added to the equation above as the last term. The minimum cooling time for *decay* heat calculation is 5 years.

Alternatively, the decay heat can be calculated without employing the decay heat equation, using an approved methodology with actual spent fuel parameters instead of bounding spent fuel parameters.

A.5.5.3.2 PWR Decay Heat Equation

A nonlinear regression analysis was performed to obtain a fit of the PWR decay heat data as a function of burnup, initial enrichment, and cooling time. A very good fit was obtained based on an iterative evaluation using a 9-parameter model. The functional form is expressed below.

The decay heat (DH) in watts is expressed as:

$$DH_{PWR} = F_1 \exp\left[G\left(1-\frac{1.8}{x_3}\right)(x_3-4.5)^H\left(\frac{x_2}{x_1}\right)^I\right]+20,$$

where,

- $F_1 = A + Bx_1 + Cx_2 + Dx_1^2 + Ex_1x_2 + Fx_2^2$,
- x_1 equals the assembly average burnup in GWd/MTU,
- x_2 equals the initial enrichment in wt. % U-235,
- x_3 equals the cooling time in years,
- $A = -44.8$,
- $B = 41.6$,
- $C = -37.1$,
- $D = 0.611$,
- $E = -6.80$,
- $F = 24.0$,
- $G = -0.575$,
- $H = 0.169$, and
- $I = -0.147$.

The calculation uncertainty is 20 watts. It is added to the equation above as the last term. The minimum cooling time for *decay* heat calculation is 5 years.

Alternatively, the decay heat can be calculated without employing the decay heat equation, using an approved methodology with actual spent fuel parameters instead of bounding spent fuel parameters.

A.5.5.4 Fission gas for BWR fuel at 70 GWd/MTU

The bounding quantity of gases released as a result of irradiation for the "generic" GE 7x7 BWR fuel assembly is evaluated.

Since the amount of released fission gases is conservative on the basis of lowest enrichment for a given burnup, the quantity of gases released is evaluated for a burnup and enrichment combination of 70 GWd/MTU and 3.70 wt% initial U-235, per BWR FQT in Tables A.1.4.9-4 to A.1.4.9-5a.

The total bounding amount of moles of gases released as a result of irradiation for one fuel assembly is presented in Table A.5-35; data presented in the table corresponds to 3.0 and 5.0 years of cooling time and 0.198 MTU.

The amount of gas produced for a "generic" BWR fuel assembly, 0.198 MTU, at 70 GWd/MTU, 3.70 wt% initial U-235 is 23.0 g-moles.

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19. U.S. Nuclear Regulatory Commission, "Analysis of Experimental Data for High Burnup PWR Spent Fuel Isotopic Validation—Calvert Cliffs, Takahama, and Three Mile Island Reactors," NUREG/CR-6968, Published February 2010, ORNL-TM-2008-71.

20. U.S. Nuclear Regulatory Commission, "Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel," NUREG/CR-7012, Published January 2011, ORNL-TM-2010-41.

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22. NUREG/CR-6802 (ORNL/TM-2002/31), May 2003, "Recommendations for Shielding Evaluations for Transport and Storage Packages."

23. Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUH-003, Rev. 12.

24. Transnuclear, Inc., "TN-68 Dry Storage Cask, Updated Final Safety Analysis Report," Revision 6. Certificate of Compliance No. 1027.

25. NOK Document "Technical Specification for the Supply of Transportable Casks for the Storage of Kernkraftwerk Leibstardt (KKL) Spent Fuel in ZWILAG." TS 07/01, Rev. 1.

26. M. D. DeHart, "Sensitivity and Parametric Evaluations of Significant Aspects of Burn-up Credit for PWR Spent Fuel Packages," ORNL/TM-12973, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, May 1996.

27. NUREG/CR-6835, September 2003, "Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks."

28. U.S. Nuclear Regulatory Commission, "NRC Draft Regulatory Issue Summary 2015-XX Considerations In Licensing High Burnup Spent Fuel in Dry Storage and Transportation.", Regulations.gov. Document ID: NRC-2015-0047-0002. Accessed April 27, 2016. URL: https://www.regulations.gov/document?D=NRC-2015-0047-0002 (NRC ADAMS Accession Number, ML14175A205).

Table A.5-1
Summary of MP197HB NCT Dose Rates for Intact Fuel

(Exclusive Use Package for Transportation)

37PTH DSC in the MP197HB Transportation Package			
Vehicle (Package) Surface[2] (mrem/hr), Limit = 200 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	1.00 ± 0.056	24.5 ± 0.74	2.29 ± 0.025
Neutron	3.79 ± 0.10	*45.37 ± 1.01*	9.99 ± 0.13
(n,g)	1.54 ± 0.023	9.82 ± 0.09*8*	4.29 ± 0.035
Total[1]	6.33 ± 0.12	*53.02 ± 1.02*	16.6 ± 0.13
2 m from Vehicle Surface (mrem/hr), Limit = 10 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.243 ± 0.020	2.35 ± 0.071	0.335 ± 0.0041
Neutron	0.773 ± 0.019	4.75 ± 0.067	1.54 ± 0.022
(n,g)	0.177 ± 0.0044	1.74 ± 0.016	0.488 ± 0.0044
Total[1]	1.06 ± 0.032	8.25 ± 0.085	2.36 ± 0.023

69BTH DSC in the MP197HB Transportation Package			
Vehicle (Package) Surface[2] (mrem/hr), Limit = 200 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	1.67 ± 0.15	*19.52 ± 0.56*	8.25 ± 0.073
Neutron	1.55 ± 0.037	*37.49 ± 0.65*	9.82 ± 0.11
(n,g)	0.731 ± 0.010	8.29 ± 0.*06*	4.15 ± 0.025
Total[1]	3.62 ± 0.041	*46.10 ± 0.72*	22.2 ± 0.013
2 m from Vehicle Surface (mrem/hr), Limit = 10 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.344 ± 0.0040	3.86 ± 0.11	0.972 ± 0.013
Neutron	0.504 ± 0.0092	3.36 ± 0.045	1.47 ± 0.023
(n,g)	0.119 ± 0.0027	1.35 ± 0.011	0.464 ± 0.0030
Total[1]	0.917 ± 0.037	8.16 ± 0.10	2.91 ± 0.026

1. The total is not necessarily the sum of the corresponding dose rates listed. The total is the greatest total dose rate at any location, while the components of the dose rates are the respective maxima and may occur at different locations.
2. Dose rates are computed on the surfaces of the impact limiters and on the surface of the neutron shield. These package surface dose rates are conservatively reported as vehicle surface dose rates.

Table A.5-1a
Summary of MP197HB NCT Dose Rates for Reconfigured Fuel, 50% Void Fraction
(Exclusive Use Package for Transportation)

37PTH DSC in the MP197HB Transportation Package			
Vehicle (Package) Surface[1] (mrem/hr), Limit = 200 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.154 ± 0.015	*0.844 ± 0.066*	1.30 ± 0.012
Neutron	3.77 ± 0.11	*80.49 ± 1.37*	24.0 ± 0.22
(n,g)	1.53± 0.024	*7.54 ± 0.07*	8.78 ± 0.051
Total	5.46 ± 0.11	*88.88 ± 1.38*	34.1 ± 0.22
2 m from Vehicle Surface (mrem/hr), Limit = 10 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.143 ± 0.00033	0.387 ± 0.015	0.163 ± 0.0020
Neutron	0.748 ± 0.015	5.54 ± 0.067	3.40 ± 0.036
(n,g)	0.133 ± 0.0026	1.74 ± 0.016	0.947 ± 0.0060
Total	0.895 ± 0.015	7.44 ± 0.082	4.51 ± 0.036

69TH DSC in the MP197HB Transportation Package			
Vehicle (Package) Surface[1] (mrem/hr), Limit = 200 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.0233 ± 0.00098	*0.213 ± 0.020*	2.68 ± 0.027
Neutron	1.69 ± 0.058	*107.35 ± 1.69*	43.0 ± 0.33
(n,g)	0.303 ± 0.0073	*11.17 ± 0.10*	15.3 ± 0.072
Total	2.01 ± 0.058	*118.74 ± 1.70*	61.0 ± 0.034
2 m from Vehicle Surface (mrem/hr), Limit = 10 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.00602 ± 0.0011	0.126 ± 0.0083	0.280 ± 0.0049
Neutron	0.701 ± 0.014	6.64 ± 0.011	5.95 ± 0.056
(n,g)	0.110 ± 0.0024	1.74 ± 0.016	1.62 ± 0.0084
Total	0.817± 0.015	8.50 ± 0.11	7.85 ± 0.057

1. Dose rates are computed on the surfaces of the impact limiters and on the surface of the neutron shield. These package surface dose rates are conservatively reported as vehicle surface dose rates.

Table A.5-1b
Summary of MP197HB NCT Dose Rates for Reconfigured Fuel, 50% Void Fraction, Additional Results
(Exclusive Use Package for Transportation)

[ ]

69TH DSC in the MP197HB Transportation Package			
Vehicle (Package) Surface(1) (mrem/hr), Limit = 200 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.065 ± 0.023	1.596 ± 0.072	2.753 ± 0.020
Neutron	1.727 ± 0.008	113.051 ± 0.453	39.964 ± 0.088
(n,g)	0.445 ± 0.003	10.379 ± 0.026	13.674 ± 0.019
Total(2)	2.025 ± 0.011	123.770 ± 0.454	56.391 ± 0.092
2 m from Vehicle Surface (mrem/hr), Limit = 10 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.014 ± 0.004	0.306 ± 0.014	0.288 ± 0.003
Neutron	0.697 ± 0.003	6.727 ± 0.031	5.597 ± 0.014
(n,g)	0.126 ± 0.001	1.671 ± 0.004	1.453 ± 0.002
Total(2)	0.802 ± 0.004	8.533 ± 0.031	7.338 ± 0.015

1. *Dose rates are computed on the surfaces of the impact limiters and on the surface of the neutron shield. These package surface dose rates are conservatively reported as vehicle surface dose rates.*
2. *Spatial locations of maximums of components of the total dose rate are generally different. Because of this, the maximum of the total dose rate is generally not equal to the sum of maximums of the components.*

Table A.5-1c
Summary of MP197HB NCT Dose Rates for Reconfigured Fuel, Reconfiguration Compliant with Enclosure 2 of the US NRC RIS 2015-XX [28]

(Exclusive Use Package for Transportation)

[ ]

69TH DSC in the MP197HB Transportation Package			
Vehicle (Package) Surface(1) (mrem/hr), Limit = 200 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.365 ± 0.002	2.875 ± 0.089	2.742 ± 0.016
Neutron	2.933 ± 0.018	107.463 ± 0.432	39.740 ± 0.078
(n,g)	1.305 ± 0.005	9.757 ± 0.024	13.233 ± 0.016
Total(2)	4.602 ± 0.019	117.526 ± 0.443	55.716 ± 0.081
2 m from Vehicle Surface (mrem/hr), Limit = 10 mrem/hr			
	Top End	**Side**	**Bottom End**
Gamma	0.076 ± 0.001	0.420 ± 0.019	0.284 ± 0.002
Neutron	0.794 ± 0.004	6.730 ± 0.029	5.572 ± 0.013
(n,g)	0.169 ± 0.001	1.771 ± 0.004	1.401 ± 0.002
Total(2)	0.968 ± 0.004	8.648 ± 0.031	7.257 ± 0.014

1. *Dose rates are computed on the surfaces of the impact limiters and on the surface of the neutron shield. These package surface dose rates are conservatively reported as vehicle surface dose rates.*
2. *Spatial locations of maximums of components of the total dose rate are generally different. Because of this, the maximum of the total dose rate is generally not equal to the sum of maximums of the components.*

Table A.5-2a
Summary of MP197HB HAC Dose Rates for Reconfigured Fuel

(Exclusive Use Package for Transportation)

[ ]

69TH DSC in the MP197HB Transportation Package			
1 m from Package Surface (mrem/hr), Limit = 1000 mrem/hr			
	Top End(2)	**Side(1,2)**	**Bottom End(3)**
Gamma	0.759 ± 0.211	4.542 ± 0.455	3.717 ± 0.063
Neutron	65.074 ± 1.405	825.304 ± 5.934	318.511 ± 2.589
(n,g)	0.174 ± 0.014	1.907 ± 0.047	1.102 ± 0.008
Total(4 (5))	65.519 ± 1.406	830.920 ± 5.942	323.330 ± 2.590

Notes:

1. *Dose rates are at radial distance measured from the location corresponding to the shell of the neutron shielding on side of the cask.*
2. *From the cask containing not reconfigured fuel.*
3. *From the cask containing reconfigured fuel and having "flat" axial distribution of radiological sources in active regions of the fuel assemblies.*
4. *Spatial locations of maximums of components of the total dose rate are generally different. Because of this, the maximum of the total dose rate is generally not equal to the sum of maximums of the components.*
5. *Dose rates bound dose rates calculated with fuel reconfiguration approach in compliance with recommendations in Enclosure 2 of the US NRC RIS 2015-XX [28]*

Table A.5-35

Amount of Moles of Gases released as Result of Irradiation for GE7x7 BWR Fuel Assembly at 0.198 MTU, 3.7 wt % U-235, and 70 GWd/MTU

Nuclide	Atomic Number	Initial Amount of Fission Gases		Amount of Fission Gases at 3 Years after Discharge		Amount of Fission Gases at 5 Years after Discharge	
		grams	gram-moles	grams	gram-moles	grams	gram-moles
Due to "Light" Elements							
H	1	1.23E+00	1.23E+00	1.23E+00	1.23E+00	1.23E+00	1.23E+00
H	2	7.48E-03	3.74E-03	7.48E-03	3.74E-03	7.48E-03	3.74E-03
H	3	6.48E-03	2.16E-03	5.47E-03	1.82E-03	4.89E-03	1.63E-03
He	3	2.40E-05	8.01E-06	1.03E-03	3.43E-04	1.61E-03	5.37E-04
He	4	1.81E+00	4.53E-01	1.81E+00	4.53E-01	1.81E+00	4.53E-01
N	14	1.55E+01	1.10E+00	1.55E+01	1.10E+00	1.55E+01	1.10E+00
N	15	7.13E-02	4.75E-03	7.13E-02	4.75E-03	7.13E-02	4.75E-03
F	19	2.40E+00	1.26E-01	2.40E+00	1.26E-01	2.40E+00	1.26E-01
Ne	20	2.83E-04	1.41E-05	2.83E-04	1.41E-05	2.83E-04	1.41E-05
Ne	21	3.48E-05	1.66E-06	3.48E-05	1.66E-06	3.48E-05	1.66E-06
Ne	22	1.36E-05	6.18E-07	1.36E-05	6.18E-07	1.36E-05	6.18E-07
Cl	35	6.75E-01	1.93E-02	6.75E-01	1.93E-02	6.75E-01	1.93E-02
Cl	36	1.62E-01	4.49E-03	1.62E-01	4.49E-03	1.62E-01	4.49E-03
Cl	37	2.87E-01	7.75E-03	2.87E-01	7.75E-03	2.87E-01	7.75E-03
Ar	36	4.55E-07	1.26E-08	1.55E-06	4.31E-08	2.28E-06	6.34E-08
Ar	38	6.48E-04	1.70E-05	6.48E-04	1.70E-05	6.48E-04	1.70E-05
Ar	39	6.80E-07	1.74E-08	6.75E-07	1.73E-08	6.72E-07	1.72E-08
Ar	40	1.16E-05	2.90E-07	1.16E-05	2.90E-07	1.16E-05	2.90E-07
I	127	2.55E-04	2.00E-06	2.72E-04	2.14E-06	2.72E-04	2.14E-06
Xe	128	1.96E-05	1.53E-07	1.96E-05	1.53E-07	1.96E-05	1.53E-07
Xe	129	1.78E-07	1.38E-09	1.79E-07	1.39E-09	1.79E-07	1.39E-09
Due to Actinides							
He	4	1.87E-01	4.68E-02	4.76E-01	1.19E-01	6.02E-01	1.50E-01
Due to Fission Products							
H	3	2.66E-02	8.87E-03	2.25E-02	7.49E-03	2.01E-02	6.69E-03
Br	79	9.69E-06	1.23E-07	2.24E-05	2.84E-07	3.09E-05	3.91E-07
Kr	80	2.87E-05	3.59E-07	2.87E-05	3.59E-07	2.87E-05	3.59E-07
Br	81	7.77E+00	9.59E-02	7.77E+00	9.59E-02	7.77E+00	9.59E-02
Kr	81	7.89E-06	9.74E-08	7.89E-06	9.74E-08	7.89E-06	9.74E-08
Kr	82	4.88E-01	5.95E-03	4.91E-01	5.99E-03	4.91E-01	5.99E-03
Kr	83	1.04E+01	1.25E-01	1.04E+01	1.25E-01	1.04E+01	1.25E-01
Kr	84	4.75E+01	5.66E-01	4.75E+01	5.66E-01	4.75E+01	5.66E-01
Kr	85	8.96E+00	1.05E-01	7.38E+00	8.69E-02	6.49E+00	7.63E-02
Kr	86	6.54E+01	7.61E-01	6.54E+01	7.61E-01	6.54E+01	7.61E-01
I	127	1.88E+01	1.48E-01	1.97E+01	1.55E-01	1.97E+01	1.55E-01
Xe	128	2.99E+00	2.34E-02	2.99E+00	2.34E-02	2.99E+00	2.34E-02
I	129	6.57E+01	5.09E-01	6.66E+01	5.16E-01	6.66E+01	5.16E-01
Xe	129	4.28E-02	3.31E-04	4.41E-02	3.42E-04	4.41E-02	3.42E-04
Xe	130	6.92E+00	5.33E-02	6.94E+00	5.34E-02	6.94E+00	5.34E-02
Xe	131	1.19E+02	9.05E-01	1.22E+02	9.33E-01	1.22E+02	9.33E-01
Xe	132	5.25E+02	3.98E+00	5.27E+02	3.99E+00	5.27E+02	3.99E+00
Xe	134	6.36E+02	4.75E+00	6.37E+02	4.75E+00	6.37E+02	4.75E+00
Xe	136	1.06E+03	7.82E+00	1.06E+03	7.82E+00	1.06E+03	7.82E+00
Total Amounts Due To Different Groups							
Total "Light" Elements		2.21E+01	2.95E+00	2.21E+01	2.95E+00	2.21E+01	2.95E+00
Actinides		1.87E-01	4.68E-02	4.76E-01	1.19E-01	6.02E-01	1.50E-01
Fission products		2.58E+03	1.98E+01	2.58E+03	1.99E+01	2.58E+03	1.99E+01
Total		2.60E+03	2.28E+01	2.61E+03	2.30E+01	2.61E+03	2.30E+01
Relative Amount of Different Groups from Total, %							
Total "Light" Elements		1%	13%	1%	13%	1%	13%
Actinides		0%	0%	0%	1%	0%	1%
Fission products		99%	87%	99%	87%	99%	86%
Total		100%	100%	100%	100%	100%	100%

Table A.5-39b
Bounding NCT and HAC Source for the 69BTH DSC, Reconfigured Fuel, 24 Peripheral Compartments:

[ ]

62 GWd/MTU, 2.6 wt. %, after 29.0 years of cooling						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	In-core	Plenum	Top Nozzle
0.00e+00	to	5.00e-02	5.075E+09	2.898E+14	1.728E+09	3.835E+09
5.00e-02	to	1.00e-01	9.146E+08	8.044E+13	2.810E+08	6.961E+08
1.00e-01	to	2.00e-01	2.239E+08	5.062E+13	8.827E+07	1.705E+08
2.00e-01	to	3.00e-01	1.158E+07	1.555E+13	4.938E+06	8.784E+06
3.00e-01	to	4.00e-01	1.559E+07	1.035E+13	9.402E+06	1.179E+07
4.00e-01	to	6.00e-01	1.943E+07	7.871E+12	1.107E+08	1.401E+07
6.00e-01	to	8.00e-01	2.441E+07	6.053E+14	5.777E+07	7.039E+07
8.00e-01	to	1.00e+00	2.581E+07	4.238E+12	3.710E+06	7.004E+07
1.00e+00	to	1.33e+00	2.662E+11	7.793E+12	8.140E+10	2.027E+11
1.33e+00	to	1.66e+00	7.517E+10	1.055E+12	2.299E+10	5.725E+10
1.66e+00	to	2.00e+00	6.283E+00	2.644E+10	4.091E+01	4.711E+00
2.00e+00	to	2.50e+00	1.799E+06	1.409E+09	5.500E+05	1.370E+06
2.50e+00	to	3.00e+00	1.537E+03	1.240E+08	4.699E+02	1.170E+03
3.00e+00	to	4.00e+00	1.102E-06	3.405E+07	9.599E-12	8.966E-06
4.00e+00	to	5.00e+00	0.0	1.150E+07	2.314E-33	0.0
5.00e+00	to	6.50e+00	0.0	4.614E+06	6.668E-34	0.0
6.50e+00	to	8.00e+00	0.0	9.051E+05	8.481E-35	0.0
8.00e+00	to	1.00e+01	0.0	1.922E+05	1.132E-35	0.0
Total Gamma, g/(sec*FA)			3.477E+11	1.073E+15	1.067E+11	2.649E+11
(1)Total Neutrons, n/(sec*FA)			3.359E+08 (raw) 4.878E+08 (adjusted intact) 5.000E+08 (adjusted reconfigured)			

(1) *The "raw" source is calculated with ORIGEN-ARP. The adjusted source is the raw source multiplied by bpf/(1-k_{eff}) to account for subcritical neutron multiplication and an axial variation of burn-up profile in active fuel region, where bpf=1.212, k_{eff}=0.1655 if fuel is intact and k_{eff}=0.1655*(1+0.1225)=0.1858 if fuel is reconfigured. k_{eff} for reconfigured fuel includes an additional 12.25% increase.*

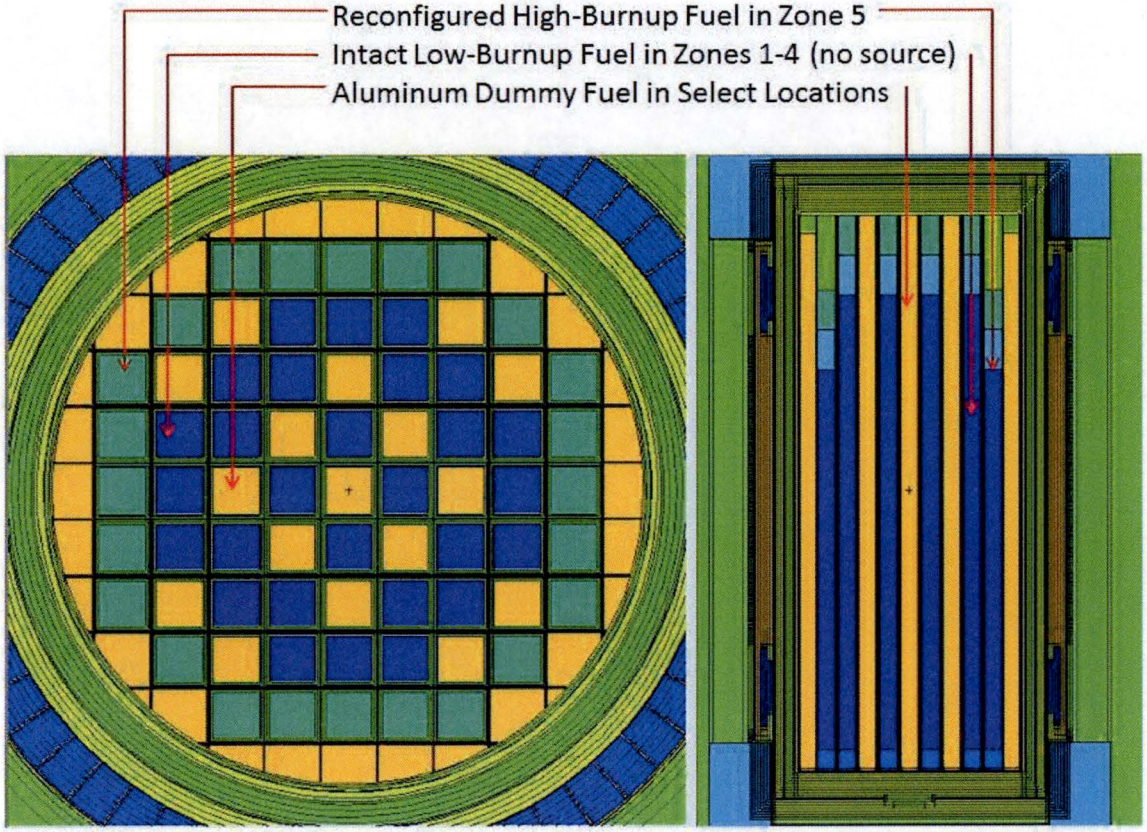

Figure A.5-24a
NCT 69BTH DSC MCNP Model for Heat Load Zone Configuration No. 8 with High-Burnup Reconfigured Fuel in Zone 5
(x-y and x-z views)

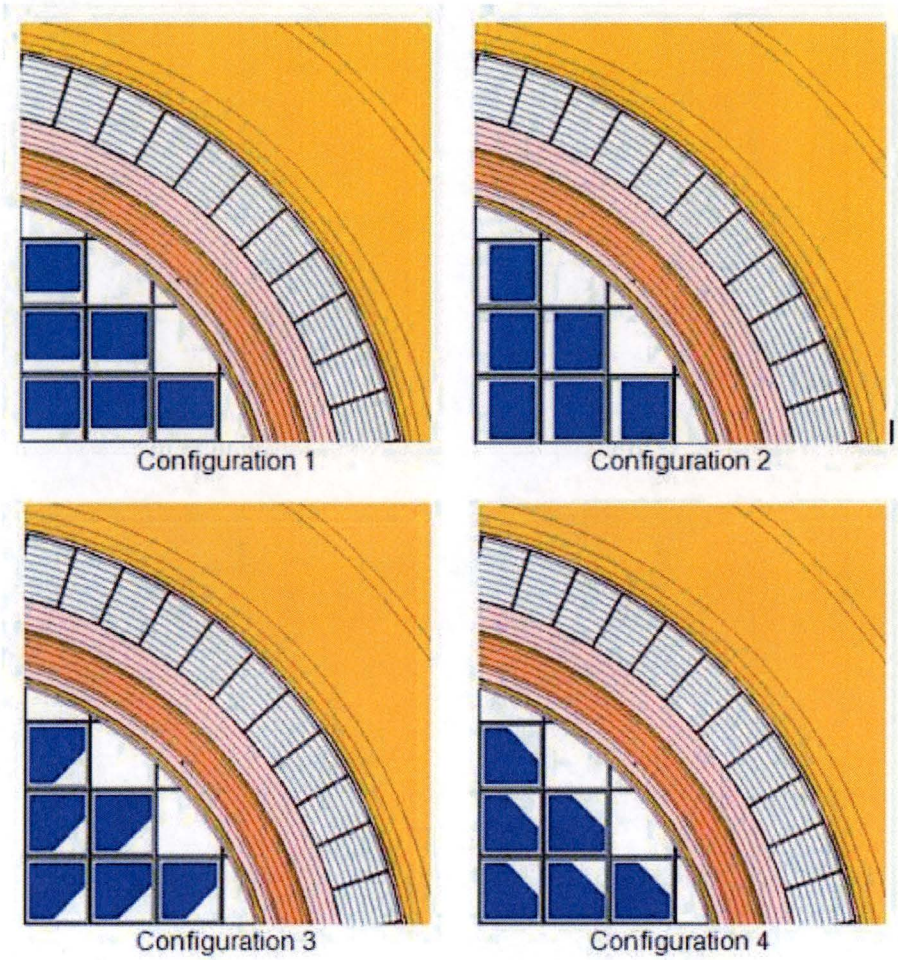

Figure A.5-27
NCT Configurations for *Radial* Fuel Reconfiguration (x-y View)