Enclosure 4 to E-47060

NUHOMS®-MP197 Transportation Package Safety Analysis Report, Revision 17B Appendix A.1.4.9, Chapter A.5, and Chapter A.7 (Non-Proprietary)

NON-PROPRIETARY

NUHOMS®-MP197 TRANSPORTATION PACKAGE SAFETY ANALYSIS REPORT

Revision *17B*

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Revision Log

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Appendix A.1.4.9
NUHOMS®-69BTH DSC

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Appendix A.1.4.9
NUHOMS®-69BTH DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.9.4.

A.1.4.9.1 NUHOMS®-69BTH DSC Description

Each NUHOMS®-69BTH DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The maximum length and the outer diameter of the 69BTH DSC are approximately 197.0 inches and 69.8 inches, respectively. The shell assembly is a high integrity stainless steel welded pressure vessel that provides confinement of radioactive materials, encapsulates the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed) and provides biological shielding (in axial direction). The 69BTH DSC has double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 69BTH DSC. The top plug penetrations (siphon and vent ports) are redundantly sealed after the 69BTH DSC drying operations are complete.

The canister is designed to contain the fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails attached to the inside surface of the transport cask.

A.1.4.9.2 NUHOMS®-69BTH Fuel Basket

The basket structure is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall length and outer diameter of the basket, including the hold down ring, are approximately 178.6 inches and 68.4 inches respectively. The details of the 69BTH fuel basket is shown in the drawings in Section A.1.4.10.10 of Appendix A.1.4.10. The 69BTH basket is designed to accommodate 69 intact, or up to 24 damaged with the remainder intact BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel structural boxes. The basket is laterally supported by the basket rails and the canister shell. The aluminum basket rails are oriented parallel to the axis of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

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	4
• Maximum Number of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly	4
• 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	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.

Table A.1.4.9-2
BWR Fuel Assembly Design Characteristics for the NUHOMS®-69BTH DSC
(Part 1 of 2)

Transnuclear ID	7x7-49/0	8x8-63/1	8x8-62/2	8x8-60/4	8x8-60/1	9x9-74/2
Initial Design or Reload Fuel Designation	GE1	GE4	GE5	GE8 Type II	GE9	GE11
	GE2		GE6		GE10	GE13
	GE3		GE7			
			GE8 Type I			
			FANP 8x8-2			
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Maximum Active-Fuel Length (in)	*144*	*146*	*150*	*150*	*150*	*146*
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Number of Fuel Rods	≤ 49	≤ 63	≤ 62	≤ 60	≤ 60	≤ 74
Initial Uranium Content (kg)	≤ 198	≤ 192	≤ 192	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.738	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.640	≤ 0.566
Pellet Diameter (in)	≤ 0.487	≤ 0.416	≤ 0.411	≤ 0.411	≤ 0.411	≤ 0.376
Clad Outer Diameter (in)	≥ 0.563	≥ 0.493	≥ 0.483	≥ 0.483	≥ 0.483	≥ 0.440
Clad Thickness (in)	≥ 0.032	≥ 0.034	≥ 0.032	≥ 0.032	≥ 0.032	≥ 0.028

Transnuclear ID	10x10-92/2	7x7-49/0Z	7x7-48/1Z	8x8-60/4Z	FANP 9x9	Siemens
Initial Design or Reload Fuel Designation	GE12	ENC-IIIA	ENC-III	ENC Va	FANP 9x9-72	QFA 9x9
	GE14		ENC-IIIE	ENC Vb	FANP 9x9-79	
			ENC-IIIF		FANP 9x9-80	
					FANP 9x9-81	
Length (in) (Unirradiated)	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6	≤ 176.6
Maximum Active-Fuel Length (in)	*150*	*144*	*144*	*144*	*150*	*145.24*
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Number of Fuel Rods	≤ 92	≤ 49	≤ 48	≤ 60	≤ 81	≤ 72
Initial Uranium Content (kg)	≤ 192	≤ 198	≤ 198	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.510	≤ 0.738	≤ 0.738	≤ 0.642	≤ 0.572	≤ 0.570
Pellet Diameter (in)	≤ 0.345	≤ 0.491	≤ 0.491	≤ 0.420	≤ 0.357	≤ 0.374
Clad Outer Diameter (in)	≥ 0.404	≥ 0.570	≥ 0.570	≥ 0.501	≥ 0.424	≥ 0.433
Clad Thickness (in)	≥ 0.026	≥ 0.035	≥ 0.035	≥ 0.035	≥ 0.030	≥ 0.026

Table A.1.4.9-2
BWR Fuel Assembly Design Characteristics for the NUHOMS®-69BTH DSC
(Part 2 of 2)

Transnuclear ID	10x10-91/1	LACROSSE10	ABB-8x8	ABB-10x10-1	ABB-10x10-2
Initial Design or Reload Fuel Designation	ATRIUM-10	ALLIS CHALMERS	SVEA-64	SVEA-92	SVEA-100
	ATRIUM-10XM	EXXON/ANF		SVEA-96	
				SVEA-96 +/L	
				SVEA-OPTIMA	
				SVEA-OPTIMA 2	
Length (in) (Unirradiated)	≤ 176.6	≤ 130	≤ 176.6	≤ 176.6	≤ 176.6
Maximum Active-Fuel Length (in)	*150*	*100*	*151*	*151*	*151*
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2
Number of Fuel Rods	≤ 91	≤ 100	≤ 64	≤ 96	≤ 100
Initial Uranium Content (kg)	≤ 192	≤ 150	≤ 192	≤ 192	≤ 192
Rod Pitch (in)	≤ 0.510	≤ 0.565	≤ 0.622	≤ 0.512	≤ 0.512
Pellet Diameter (in)	≤ 0.350	≤ 0.350	≤ 0.411	≤ 0.346	≤ 0.375
Clad Outer Diameter (in)	≥ 0.395	≥ 0.394	≥ 0.462	≥ 0.378	≥ 0.443
Clad Thickness (in)	≥ 0.023	≥ 0.020	≥ 0.027	≥ 0.022	≥ 0.024

Notes:

(1) The fuel assembly fabrication documentation may be used to demonstrate compliance with these fuel assembly parameters. The fuel assembly parameters are design nominal values. The maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a fuel assembly class (or an array type).

(2) Any fuel channel average thickness up to 0.120 inch is acceptable on any of the fuel designs. Fuel channels are required for loading intact fuel assemblies with an assembly average burnup greater than 45 GWd/MTU.

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

Basket Type	Maximum Lattice Average Initial Enrichment[1] (wt. % U-235)	Minimum B10 Areal Density, gram/cm²	
		Borated Aluminum/MMC	Boral®
A	3.7	0.021	0.025
B	4.1	0.031	0.037
C	4.4	0.039	0.047
D	4.6	0.046	0.055
E	4.8	0.053	0.064
F	5.0	0.061	0.073

	Maximum Lattice Average Initial Enrichment[1][3] (wt. % U-235)			
Basket Type	Intact Assemblies	Up to 4 Damaged Assemblies[2]	5 to 8 Damaged Assemblies[2]	9 to 24 Damaged Assemblies[2]
A	3.70	3.70	3.30	2.80
B	4.10	4.10	3.60	3.00
C	4.40	4.20	3.60	3.10
D	4.60	4.40	3.70	3.20
E	4.80	4.40	3.70	3.20
F	5.00	4.80	3.90	3.40

(1) For LACROSSE10 fuel assemblies, the enrichment shall be reduced by 0.1 wt. % U-235.
(2) Allowable locations in basket per Figure A.1.4.9-1. *Enrichment limits of the damaged fuel assemblies. The enrichment limits of the complementary intact fuel assemblies are shown in the second column.*
(3) For ABB-10x10-1 fuel assemblies with a pitch greater than 0.502 inches, the enrichment shall be reduced by 0.25 wt. % U-235.

Table A.1.4.9-4

BWR Fuel Qualification Table for the NUHOMS®-69BTH DSC (Uranium Loading ≤ 198 kg) *for Heat Load Zoning Configurations 1, 2, 3, and 4*

(Minimum required years of cooling time after reactor core discharge)

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

Note: Explanatory notes and limitations regarding the use of this table follow Table A.1.4.9-5.

Table A.1.4.9-5
BWR Fuel Qualification Table for the NUHOMS®-69BTH DSC (Uranium Loading ≤ 182 kg) *for Heat Load Zoning Configurations 1, 2, 3, and 4*

(Minimum required years of cooling time after reactor core discharge)

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

Note: Explanatory notes and limitations regarding the use of this table follow Table A.1.4.9-5.

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) *for Heat Load Zoning Configurations 1, 2, 3, and 4*

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												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						Not Analyzed						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, 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
5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	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,	**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	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	14.0	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						*Not Analyzed*															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																					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	1.0	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					Not Analyzed																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

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, 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	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	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	1.0	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	1.5	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						Not Analyzed															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

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	4.0	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	4.5	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	Not Analyzed																				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

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

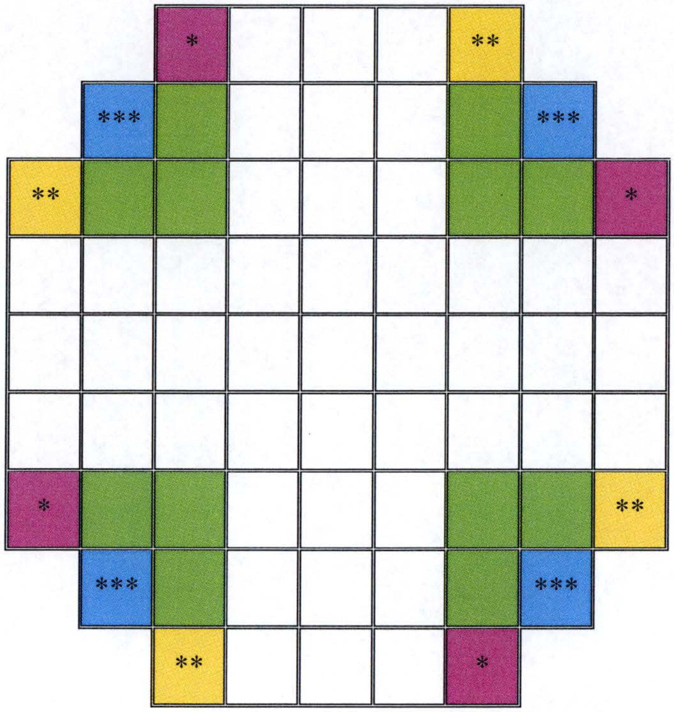
The Decay Heat (DH) in watts is expressed as:

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

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)

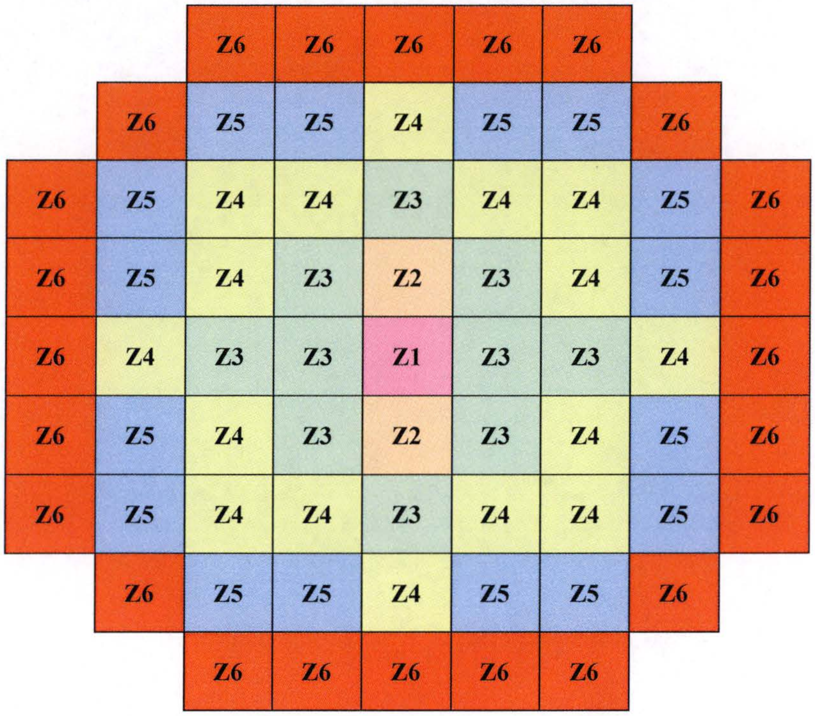
A uranium loading of 198 kg is employed in the calculation of the decay heat equation. Alternatively, the decay heat can be calculated without employing the decay heat equation, using an approved methodology with actual spent fuel parameters instead of bounding spent fuel parameters.



*	Configuration 1(1)	**	Configuration 2(1)	***	Configuration 3(1)

1	Either one of these three sets of corner locations shall only be utilized to load up to four damaged assemblies with the remaining intact in a 69BTH Basket. The maximum lattice average initial enrichment of fuel assemblies (damaged or intact transported in either magenta set of cells for configuration 1, gold set of cells for configuration 2, or blue set of cells for configuration 3) is limited to the "up to 4 damaged assemblies" column of Table A.1.4.9-3.
	Following the placement of damaged fuel assemblies in either configuration 1 or 2, the remaining gold or magenta locations shall be used to load up to 4 additional damaged assemblies, with the remaining intact in a 69BTH Basket. The maximum lattice average initial enrichment for these fuel assemblies (damaged or intact transported in gold or magenta cells available) is limited to the "5 to 8 damaged assemblies" column of Table A.1.4.9-3.
	Following the placement of eight damaged fuel assemblies in the set of corner locations marked with a "*" (shaded in magenta) and a "**" (shaded in gold), the locations shaded in green or blue in Figure shall be used to load up to sixteen additional damaged assemblies, with the remaining intact in a 69BTH Basket. The maximum lattice average initial enrichment for all 24 fuel assemblies (damaged or intact transported in these 24 locations) is limited to the "9 to 24 Damaged Assemblies" column of Table A.1.4.9-3.

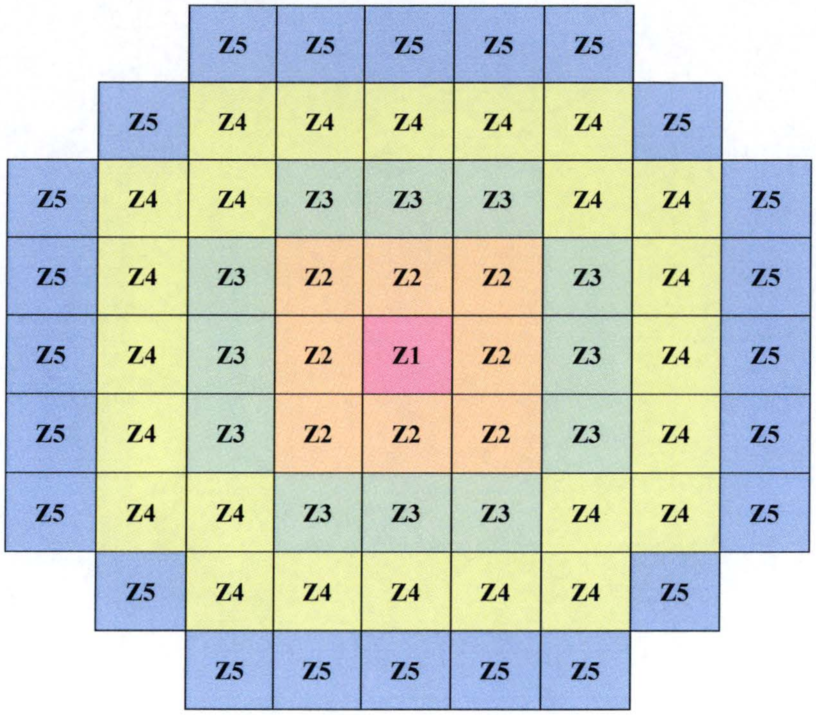
Figure A.1.4.9-1
Location of Damaged Fuel Assemblies Inside 69BTH DSC



	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5	Zone 6
Max. Decay Heat (kW/FA) (3)(4)	0.10	0.27	0.30	0.40	0.55	0.45
No. of Fuel Assemblies (1)	1	2	10	16	16	24
Max. Decay Heat per Zone (kW) (3)	0.10	0.54	3.0	6.4	8.8	10.8
Max. Decay Heat per DSC (kW)	26.0 (2) (3)					

Notes:
(1) Total number of fuel assemblies is 69 for HLZC # 1.
(2) Adjust payload to maintain the total DSC heat load within the specified limit.
(3) Reduce the maximum decay heat to 70% of the listed values for *LACROSSE10* fuel assembly. The total decay heat for *LACROSSE10* fuel assembly is 18.2 kW per DSC for HLZC No. 1.
(4) Decay heat per fuel assembly shall be determined per Table A.1.4.9-*6*.

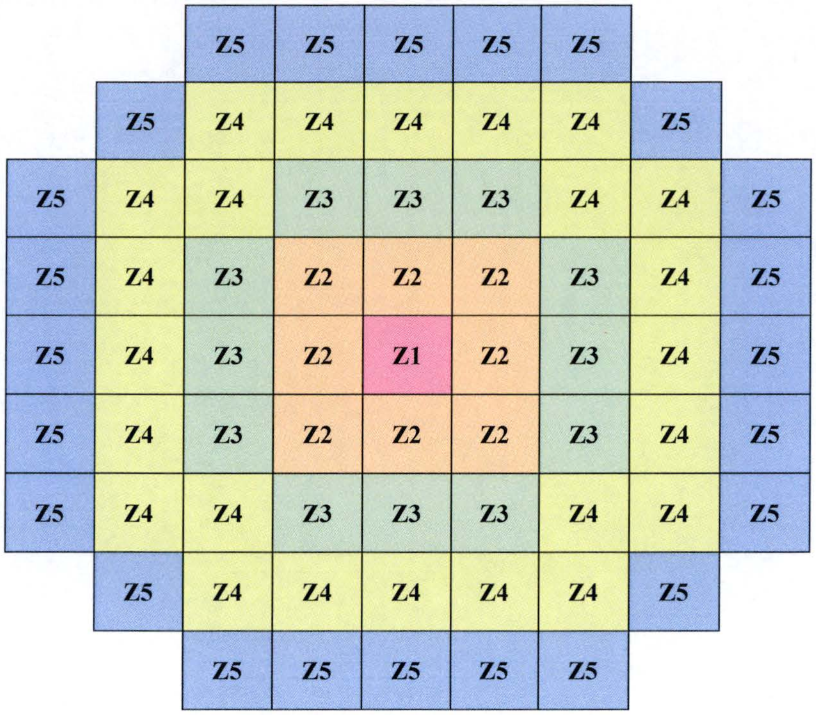
Figure A.1.4.9-2
Heat Load Zoning Configuration No. 1 for 69BTH Basket



	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5
Max. Decay Heat (kW/FA) (4)(5)	0.25	0.0 (1)	0.40	0.60	0.50
No. of Fuel Assemblies (2)	1	0	12	24	24
Max. Decay Heat per Zone (kW) (4)	0.25	0	4.8	14.4	12.0
Max. Decay Heat per DSC (kW)	26.0 (3) (4)				

Notes:
(1) Aluminum dummy assemblies replace the fuel assemblies in zone 2.
(2) Total number of fuel assemblies is 61 for HLZC # 2.
(3) Adjust payload to maintain the 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 18.2 kW per DSC for HLZC No. 2.
(5) Decay heat per fuel assembly shall be determined per Table A.1.4.9-*6*.

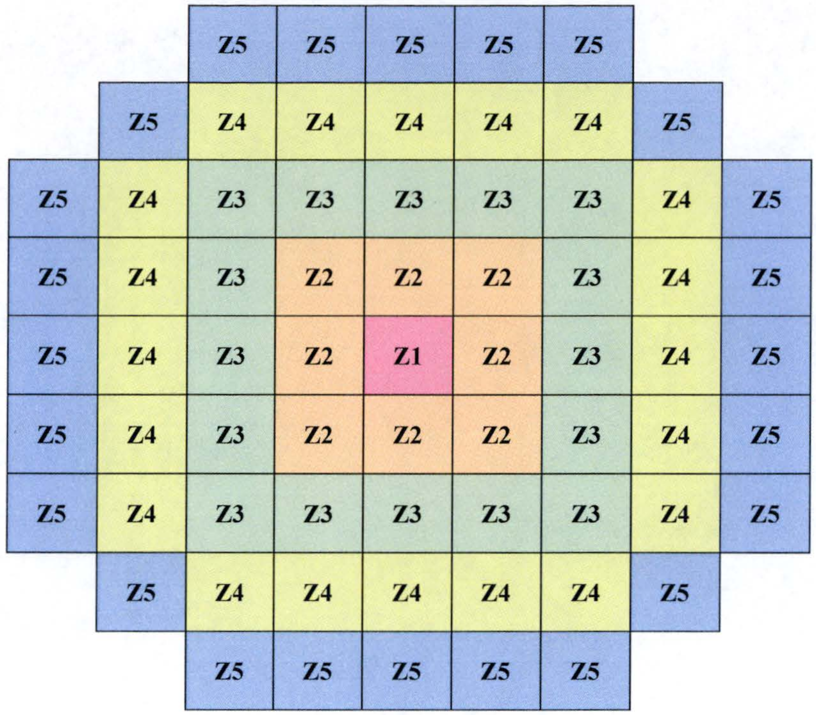
Figure A.1.4.9-3
Heat Load Zoning Configuration No. 2 for 69BTH Basket



	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5
Max. Decay Heat (kW/FA) (4)(5)	0.25	0.0 (1)	0.40	0.60	0.50
No. of Fuel Assemblies (2)	1	0	12	24	24
Max. Decay Heat per Zone (kW) (4)	0.25	0	4.8	14.4	12.0
Max. Decay Heat per DSC (kW)	29.2 (3) (4)				

Notes:
(1) Aluminum dummy assemblies replace the fuel assemblies in zone 2.
(2) Total number of fuel assemblies is 61 for HLZC # 3.
(3) Adjust payload to maintain the 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 20.4 kW per DSC for HLZC No. 3.
(5) Decay heat per fuel assembly shall be determined per Table A.1.4.9-*6*.

Figure A.1.4.9-4
Heat Load Zoning Configuration No. 3 for 69BTH Basket



	Zone 1	Zone 2	Zone 3	Zone 4	Zone 5
Max. Decay Heat (kW/FA) (4)(5)	0.0 (1)	0.45	0.0 (2)	0.70	0.60
No. of Fuel Assemblies (3)	0	8	0	20	24
Max. Decay Heat per Zone (kW) (4)	0	3.6	0	14.0	14.4
Max. Decay Heat per DSC (kW)	32.0 (4)				

Notes:
(1) The fuel compartment in zone 1 remains empty.
(2) Aluminum dummy assemblies replace the fuel assemblies in zone 3.
(3) Total number of fuel assemblies is 52 for HLZC # 4.
(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 22.4 kW per DSC for HLZC No. 4.
(5) Decay heat per fuel assembly shall be determined per Table A.1.4.9-*6*.
(6) Borated Aluminum is the only poison material allowed for HLZC #4.

Figure A.1.4.9-5
Heat Load Zoning Configuration No. 4 for 69BTH Basket

		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

		P	*P*	*P*	*P*	*P*		
	P	*I*	*I*	*I*	*I*	*I*	*P*	
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
	P	*I*	*I*	*I*	*I*	*I*	*P*	
		P	*P*	*P*	*P*	*P*		

Figure A.1.4.9-6
Peripheral and Inner Fuel Locations for the 69BTH DSC

Chapter A.5
Shielding Evaluation

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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].

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Proprietary Information Withheld Pursuant to 10 CFR 2.390

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A.5.1 Description of the Shielding Design

The MP197HB cask is designed to transport one of several NUHOMS® DSCs loaded with spent fuel assemblies or dry irradiated and/or contaminated non-fuel bearing solid materials in a radioactive waste canister (RWC) in accordance with the requirements of the 10 CFR 71. The authorized contents acceptable for transport are described in Chapter A.1, Section A.1.2.3, including appendices A.1.4.1 through A.1.4.9A. A complete list of the NUHOMS® DSCs authorized for transport is provided in Chapter A.1, Section A.1.2.3.1. Chapter A.1, Section A.1.2.3.2 (also in Appendix A.1.4.9A) provides a description of the irradiated and/or contaminated non-fuel bearing solid materials authorized for transport in the RWC as well as its respective physical dimensions.

Radiological sources used for the calculation of the dose rates presented in this chapter are determined through ranking using the response function methodology to develop the fuel qualification tables (FQT). Response function results are compared with direct MCNP analysis using a discrete MP197HB transportation package model as described in Section A.5.4.1.2.3.

By definition of the FQTs, the minimum cooling times are determined so that the maximum NCT dose rates for intact fuel at 2 m from the side of the vehicle are $\leq$ 8.2 mrem/hr. For fuel in the peripheral basket locations, additional cooling time is needed for some burnup, enrichment, and cooling time (BECT) combinations due to fuel reconfiguration, as defined using the methodology in Section A.5.4.1.3.3. Further discussion of the fuel qualification methodology is contained in Section A.5.4.1.3 and FQT results are discussed in Section A.5.5.2.

A.5.1.1 Package Design Features

Shielding for the MP197HB transportation package is provided mainly by the cask body. Shielding against gamma radiation is provided by the lead and stainless steel shells that comprise the cask wall. For the neutron shielding, a borated VYAL-B resin compound surrounds the cask body radially. Gamma shielding in the cask ends is provided by the steel top and bottom assemblies of the transportation cask and axial ends of the DSCs. Additional shielding is provided by the steel outer shell surrounding the resin layer, the steel and aluminum structure of the fuel basket and optional heat dissipation fins surrounding the cask side between impact limiters.

For transport, wood filled impact limiters are installed on either end of the cask and provide additional shielding for the ends and some radial shielding for the areas at either end of the radial neutron shield.

Important-to-shielding dimensions are shown in Table A.5-4.

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.

Generation of the source terms is a two-step process. First, two-dimensional TRITON/T-DEPL models are developed for the GE 7x7 and B&W 15x15 fuel assembly types. These computationally intensive models are used to generate burnup-dependent data libraries. The 44GROUPNDF5 cross section library *is* used *in the* TRITON/T-DEPL depletion calculations. *The TRITON/T-DEPL input parameters are summarized in Table A.5-8. Second, the source terms are computed using ORIGEN-ARP and utilize the data libraries developed above. The ORIGEN-ARP input parameters for the design basis sources are summarized in Table A.5-50.*

A.5.2.1 Gamma Source

The gamma sources used in the analysis are described in the following sections. The gamma radiation spectrum is presented *with* an 18 energy group structure consistent with the SCALE 27n-18g cross section library energy grouping structure. The lower boundary energy range in this library is 0.05 MeV. It corresponds to Group 45. The upper energy range is 8.00 to 10.00 MeV. It corresponds to Group 28. The conversion of the source spectra is performed directly through the ORIGEN-S code. The gamma source for the fuel assembly hardware is primarily from the activation of cobalt. This activation contributes primarily to energy Groups 36 and 37 of the SCALE 27n-18g library

A.5.2.1.1 24PTH

Partial length shielding assemblies (PLSAs) are only authorized for the 24PTH DSC and are Westinghouse 15x15 design fuel assemblies that consist mostly of stainless steel. They are restricted to a maximum burnup of 40 GW*d*/MTU and a maximum MTU loading of 0.330 as shown on Chapter A.1, Appendix A.1.4.3. Therefore, they are bounded by the design basis B&W 15x15 fuel assemblies.

A.5.2.1.2 37PTH

The NCT PWR design basis fuel assembly gamma source in the 37PTH DSC in the MP197HB transportation package is shown in Table A.5-36. This source is for intact fuel and 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. This table includes the gamma contribution of all four homogenized fuel assembly regions.

The bounding PWR design basis gamma 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, and 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.

It is demonstrated in Section A.5.4.1.3.3 that fuel placed in the peripheral zone requires additional cooling time for some BECT combinations in order to meet the HAC dose rate limits. For a burnup of 62 GWd/MTU and an initial enrichment of 3.4 wt. %, Table A.5-53 indicates that 9.0 years of additional cooling time is needed in the peripheral zone. Because the inner zone contains fuel with a cooling time of 21.7 years, the minimum cooling time in the outer zone is then 21.7+9.0 = 29.7 years, which is conservatively rounded down to 29.5 years.

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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.5.2.1.4 Control Components

The spent fuel payload consists of various DSCs with PWR fuel assemblies and associated control components (CCs) and is specified in Chapter A.1, Appendix A.1.4.1 through A.1.4.6. For the PWR fuel assemblies, the various authorized CCs are listed in the above appendices. These include PWR burnable poison rod assemblies *(BPRAs)*, thimble plug assemblies, control rod assemblies, control rod cluster assemblies, axial power shaping rods, orifice rod assemblies, vibration suppression inserts, neutron source assemblies, and neutron sources. The CCs are typically solid or hollow rods of stainless steel or Zircaloy containing neutron absorbing or neutron source materials. Typically, the source term from these CCs is dominated by the Co-60 spectrum. Therefore, a separate material composition and irradiation history is not necessary for characterizing all of these CCs.

Radiological source in Table A.5-18 bounds any CC authorized for loading. The source in this table is referred to as design basis CC source. Guidelines for adjustment of FQT cooling times due to presence of DB CC sources are provided in Section *A.5.5.2.1*.

DB PWR FA *BPRAs* with burnup between 36,000 MWd/MTU and 45,000 MWd/MTU are bounded by the design basis CC source after 8 years decay. All other BPRAs irradiated between 36,000 MWd/MTU and 45,000 MWd/MTU would require 13 years of decay to be bounded by the design basis CC source. All other CCs would need to be examined on a case-by-case basis.

Combinations of radiological sources due DB PWR assembly and the DB CC source result in bounding dose rates when evaluating shielding performance of MP197HB transportation package loaded with DSCs containing PWR FAs with DB CC sources.

A.5.2.1.5 RWC

The NUHOMS®-MP197HB is designed to transport a payload of *56.0* tons of dry irradiated and/or contaminated non-fuel bearing solid materials in the RWC. The safety analysis of the cask takes no credit for the containment provided by the RWC. The quantity of radioactive material is limited to a maximum of 8,182 A_2 (90,000 Ci of Co-60). A list of typical components and their associated activities is shown in Section A.1.4.9A.3.

Co-60 emits two photons per disintegration, one at 1.17 *MeV* and one at 1.33 MeV. *Because Co-60 emits highly energetic photons, 90,000 Ci Co-60 bounds any potential non-fuel bearing solid material for the purposes of dose rate calculations.*

The decay heat load of the radioactive material is expected to be less than 5 kW, which is well below the 26 kW limit for the cask.

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

strength. The axial profile in the MCNP model provides the sampling basis in the source description to properly model the distribution. A methodological development of the burnup dependent peaking factors is discussed in Section A.5.4.1.1.5.

A.5.2.3.1 PWR

A.5.2.3.1.1 Gamma

For the intact fuel models, the burnup profile is obtained from Table 2 of NUREG/CR-6801 [16] for burnups greater than 46 GWd/MTU. This burnup profile is replicated in Table A.5-11. The gamma axial source distribution is assumed to be the same as the axial burnup profile. The gamma axial source distribution is also provided normalized to a total of 1.000. When the distribution is normalized to a total of 1.000, the relative number of particles in each zone is represented. For example, in Zone 1 the distribution is 0.0318; therefore, 3.18% of the fuel gamma source is in Zone 1. The axial source distribution is applied to the active fuel in the MCNP models.

For the reconfigured fuel models, the burnup profile is obtained from Table 20 of ORNL/TM-12793 [26] for burnups between 44 and 55 GWd/MTU. This data is replicated in Table A.5-11a. The gamma axial source distribution is assumed to be the same as the axial burnup profile. The gamma axial source distribution is presented normalized to a total of 1.000. The axial source distribution is applied to the active fuel in the MCNP models.

A.5.2.3.1.2 Neutron

The neutron source is proportional to the fourth power of the axial burnup profile, as indicated in NUREG/CR-6802 [22]. The axial burnup profiles are the same as those utilized in Section A.5.2.3.1.1, Gamma. The non-normalized neutron axial source distribution is developed by raising the burnup profile to the fourth power. When the distribution is normalized to a total of 1.000, the relative number of particles in each zone is represented. This normalized profile is also provided. The neutron axial source distributions for intact and reconfigured fuel are provided in Table A.5-11 and Table A.5-11a, respectively. These axial source distributions are applied to the active fuel in the MCNP models.

A different burnup profile is used for reconfigured fuel primarily due to the effect on the neutron axial source distribution. The number of neutrons in the zone closest to the end of the neutron shield (Zone 1) becomes important for reconfigured fuel because the neutron shield stops at the impact limiter and the fuel is compressed in this direction. Comparing the two normalized PWR axial neutron source distributions (compare Table A.5-11 with Table A.5-11a), 0.52% of the neutrons are in Zone 1 in the intact fuel distribution, while 0.8% of the neutrons are in Zone 1 in the reconfigured fuel distribution.

A.5.2.3.2 BWR

A.5.2.3.2.1 Gamma

For the intact fuel models, the axial burnup profile is based on a representative fuel assembly with an average burnup of 40 GWd/MTU. The burnup profile is provided in Table A.5-10 and is divided into 12 axial zones. Rather than simply assume the gamma axial source distribution follows the axial burnup profile, an ORIGEN run is developed for each axial zone, taking account the burnup, water density, and temperature of each zone. The axial gamma source distribution is then derived from the explicit gamma sources from the ORIGEN outputs. The explicitly derived axial source distribution is provided in Table A.5-10. It may be observed that the axial source distribution is almost identical to the axial burnup profile. The gamma axial source distribution is also provided normalized to a total of 1.000. When the distribution is normalized to a total of 1.000, the relative number of particles in each zone is represented. For example, in Zone 1 the distribution is 0.0113; therefore, 1.13% of the fuel gamma source is in Zone 1. The axial source distribution is applied to the active fuel in the MCNP models.

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A.5.2.3.2.2 Neutron

The neutron source is proportional to the fourth power of the axial burnup profile, as indicated in NUREG/CR-6802 [22]. The non-normalized neutron axial source distribution for intact fuel is provided in Table A.5-10 and is developed by raising the gamma axial source distribution to the fourth power (the gamma axial source distribution is nearly identical to the axial burnup profile). The non-normalized neutron axial source distribution for reconfigured fuel is provided in Table A.5-10a and is also developed by raising the axial burnup profile to the fourth power. When the distribution is normalized to a total of 1.000, the relative number of particles in each zone is represented.

A different profile is used for reconfigured fuel primarily due to the effect on the neutron axial source distribution. The number of neutrons in the zone closest to the end of the neutron shield (Zone 1) becomes important for reconfigured fuel because the neutron shield stops at the impact limiter and the fuel is compressed in this direction. Comparing the two normalized BWR axial neutron source distributions (compare Table A.5-10 with Table A.5-10a), 0.009% of the neutrons are in Zone 1 in the intact fuel distribution, while 0.7% of the neutrons are in Zone 1 in the redistributed fuel distribution. This represents a significant shift in the neutron distribution, resulting in conservative dose rates for reconfigured fuel.

A.5.2.4 Axial Blankets

Axial blankets in the fuel are authorized provided the minimum initial enrichment of U-235 in the blanket is 0.7 wt. %. Further, to account for the uncertainty associated with the depletion of

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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A.5.3 Model Specification

A.5.3.1 Configuration of Source and Shielding

The 3-D Monte Carlo computer code MCNP is used for calculating response functions and, the gamma and neutron radiation dose rates for the bounding shielding analysis of the cask [1]. This section provides details of the geometry, material, source term configurations, physics and tallies description employed in the shielding models to determine the dose rates and response functions used for qualification of fuel assemblies for transportation based solely on the dose rate limits.

The model geometry of the shielding configuration can be viewed on Figure A.5-1 through Figure A.5-9*, and Figure A.5-23 through Figure A.5-25*. Thicknesses of the major shielding components of the cask and 69BTH DSC are summarized in Table *A.5-4*.

Variance reduction was accomplished by means of importance zoning in all MCNP models. The importance function was created to keep balance of the particles (per volume) throughout the problem geometry. The process used to do this was an iterative approach starting with basic attenuation factors for the shielding materials. The neutron importance function developed was also applied to the secondary gammas.

The cask is secured in a horizontal position on a skid attached to a railcar or other trailer with a deck or floor during transportation. However, the package and its contents are modeled as a stand alone entity, without any surroundings in computational models for bounding shielding evaluation and determination of the response functions. Therefore, effect of transportation equipment on dose rate distributions below the cask (which can be especially important for the close, less than 2 meters distances) is conservatively not accounted for in the current analysis.

Sections *A.5.3.1.1* and *A.5.3.1.2* describe the shielding model developed for the MP197HB under NCT and HAC, respectively. Described models were used to calculate the axial and radial dose rates in the bounding shielding evaluation. Similar models are used for calculating response functions, with the differences in the description of DSC basket compartments, fuel regions, shielding materials densities and the thickness on ends of the cask, and axial burn-up profile variation (BWR vs. PWR) of the radiological source. Such a difference is due to the fact that the cask is designed for the transportation of the various DSCs designed for BWR and PWR FAs. Presented description of NCT model is applicable to the models employed for calculating response functions used for transportation qualification of BWR assemblies with the difference in number and arrangement of fuel compartments and aluminum transition rails in DSCs. The PWR model is very similar with exceptions for burn-up profile, fuel region materials densities, and composition.

A.5.3.1.1 NCT

The geometry of NCT model for the bounding shielding evaluation is a complete three dimensional simulation of the MP197HB transportation package loaded with 69BTH DSC containing design basis BWR assemblies. The cask, the DSC and its contents are modeled with a discrete representation of the basket and fuel structure. Each fuel assembly is divided into four axial zones. The bottom zone represents the lower end fittings, the middle zone represents the active fuel region and the upper zones represent the plenum and upper end fittings, respectively. The modeled active fuel length is 144 inches for intact fuel and the plenum length is 12.93

inches. The modeled bottom end fitting and top end fitting lengths are 6.65 inches and 12.62 inches, respectively. The fuel, end fittings and plenum are homogenized within each assembly envelope and the axial length of their respective zones. All of the above is applicable to MCNP models used for calculation of the response functions with the exception for axial extents of fuel regions, which are different in the models corresponding to the cask with DSCs containing PWR fuel.

A fuel basket assembly is designed to locate and support fuel assemblies. For NUHOMS® 69BTH, the basket structure consists of welded stainless steel tubes (fuel compartments) separated by aluminum-poison plates. Fuel compartments are arranged in 2 arrays (full 3×3 fuel compartment array and partial 3x3 fuel compartment array) surrounded by stainless steel wrap. These compartment arrays are separated by 0.375" thick aluminum plates. Solid aluminum transition rails centers fuel compartments clusters inside of the DSC. Fuel compartments are modeled as square stainless steel tubes separated by aluminum sheets. Thickness of the compartments and aluminum plates in MCNP model is 0.20" and 0.12", respectively. This description also applies for the MCNP model used for calculation of response function related to MP197HB\69BTH DSC shielding configuration. This is a conservative modeling approach for representing DSC fuel compartments in the shielding calculations.

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External fins for excessive heat dissipation on outer surface of the cask are not modeled. MCNP simulation suggests that it may result in up to 15% (depending on burn-up\enrichment combination from FQT) of conservatism in calculated dose rates.

The borated neutron shielding material (VYAL-B) is embedded into 0.12" thick aluminum boxes. There are 60 such boxes around the side of the cask perimeter between impact limiters. Sides of the boxes adjacent to 2.50" thick cask Outer Shell and 0.375" Shield Shell are modeled as 0.125" thick aluminum cylindrical shells. The neutron shielding resin and the boxes are modeled explicitly. Information regarding the composition of the neutron shielding material is discussed in Section A.5.3.2.1.

Trunnion plugs are assumed to be made from the same neutron shielding material that is on the side of the cask. The same amount of the neutron shielding material of the plugs is assumed lost during HAC on the cask side. The plug is encased with 0.0625" thick steel shell. The shell portion at the bottom of the plug is placed on top of the plug in MCNP models. This preserves the total thickness of the steel on top and the bottom of the plug. Assumed configuration of the trunnion plugs in MCNP models is shown on Figure A.5-7.

Geometry of the grapple ring cut-out on the cask bottom and shear key cut out on the side in MCNP models are shown on sketches of Figure A.5-8 and Figure A.5-9.

The impact limiters are modeled as wood surrounded by a 0.25 in. thick steel shell. The interior steel gussets are conservatively neglected. Wood thickness between end of the impact limiters and the cask ends is 26.25" in MCNP models. The outer diameter is 125.53".

The fuel pins and fuel assembly hardware (end fittings and plenum materials) are homogenized within each assembly envelope and the axial length of their respective axial zones in MCNP models. This is a conservative modeling approach for representing fuel regions in the shielding calculations.

A.5.3.1.2 HAC

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Pages A.5-12c through A.5-12d are proprietary information withheld pursuant to 10 CFR 2.390.

A.5.3.1.3 RWC

The geometry of the RWC and the volume occupied by the irradiated\contaminated hardware are specified in MCNP models using the following assumptions. The canister *is* modeled as a carbon steel cylinder with 70.50" diameter and 189.19" height. The cylinder is centered at the cask axis and it is 2.71" from the cask bottom plug. The thickness of the cylindrical shell on side of the canister is 1.75". Thickness of shield plugs on bottom and top of the canister is 5.75" and 7.00", respectively, as specified in Appendix A.1.4.9A. Radioactive waste occupies only a portion of the inner volume of the canister. It is assumed that the waste is distributed within a cylindrical volume with 66.0" diameter and 168" height. The bottom of that cylindrical region is in contact with the bottom plug of the canister. The rest of the inner volume of the canister is occupied with air.

A.5.3.1.4 Tallies

Bounding dose rates are computed at various distances from MP197HB transportation package. Mesh tallies calculate neutron, primary and secondary gamma radiation dose rates and distributions at various distances from the side and ends of the package. Cylindrical and rectangular mesh types are used. *Dose rate location terminology is shown on Figure A.5-10.*

Locations of mesh nodes are defined either in cylindrical or rectangular (Cartesian) coordinate systems. The Z-axis of the rectangular coordinate system is along the cask axis. The X-axis is on an imaginary plane through the cask axis and trunnions. It is perpendicular to the cask axis. The XZ plane is a horizontal plane and the Y axis runs in vertical elevation when the cask is in the transportation position. Rectangular (X-Y) mesh tallies are used to calculate spatial distributions at distances less than or equal to two meters from the cask ends. Size of the mesh unit segment is 30x30 cm. The central node of the grid is symmetric around the cask axis. The mesh grid extends up to 12 feet from the cask axis in X and Y directions.

The cylindrical (angular-axial) mesh is used for determining the dose rate distribution along the cask side between the ends of impact limiters and at various radial distances from the cask side. The dose rate around the cask is the highest along the cask side, and the cylindrical (angular-axial) mesh tally along the side of the cask between ends of impact limiters was also employed for determining response functions used for fuel assemblies qualification for transportation purpose. The Z-axis of the cylindrical coordinate system coincides with the cask axis. The axial coordinates of the mesh nodes are measured from the end of the bottom impact limiter. The axial distance between nodes of the cylindrical mesh between impact limiters is 32 cm, except for over the shear key cut-out, where the axial distance between nodes is 22 cm, and area over impact limiters. Axial spacing between nodes of the mesh grid over the impact limiters is 33.5 cm. The

angular coordinate is measured in counter-clockwise direction from an imaginary plane through the cask axis and the shear key, which is the YZ plane in the rectangular coordinate system. For the dose rates at less than or equal to two meters from the cask side, segmentation of the angular coordinate is performed to keep an arch length on the cylindrical grid constant (~30 cm).

A.5.3.1.5 Physics Specification

Upper energy limit for detailed photon physics treatment during MCNP calculation is set to 20.0 MeV. Photons with energy less than 0.001 MeV are cut off. This covers energy spectrum from the fuel assemblies in storage or qualified for transportation. Physics photon treatment accounts for coherent scattering and Doppler energy broadening. It does not account for bremmstrahlung and photonuclear collisions photons because dose rates are negligible from gamma radiation sources at energy less than 0.8 MeV and greater than 5.0 MeV, respectively. MCNP default parameters for neutron transport physics are used: no lower cut off or upper limit for the neutron source energy, neutrons are treated with implicit capture, *and* no delayed neutrons since the fission is turned off by using *nonu* card in MCNP input decks for neutron transport. *Subcritical multiplication is accounted in the neutron and (n,γ) sources as discussed in Section A.5.2.2 and Section A.5.4.1.1.4.*

A.5.3.2 *Material Properties*

The following materials were used in the development of the models of the MP197HB transportation package and its associated authorized conents for the purpose of evaluating the external dose rates.

A.5.3.2.1 VYAL-B

The borated neutron shielding material (VYAL-B) is a vinylester resin mixed with alumina hydrate and zinc borate which is added for their fire retardant properties. The elemental composition of the VYAL-B resin is shown in Table A.5–5. Additional notes regarding the use of this material are discussed in Section A.5.5.1.

A.5.3.2.2 RWC Assumed Composition

It is assumed the elemental composition of the smeared material *in the RWC* is identical to that of carbon steel. The density of smeared material is 1.0 g/cc in MCNP models. The material modeling consideration employed reasonably represents irradiated non-fuel hardware.

A.5.3.2.3 Homogenized Assembly Materials

The fuel assembly is homogenized into four source regions as described in Section A.5.4.1.1.1. The resulting composition and densities of these regions are shown in Table A.5-12 for the BWR and PWR DBFA described in Section A.5.2.

A.5.3.2.4 Shielding Materials

The elemental composition and densities of the materials used for the package evaluation beyond the neutron shielding and fuel assembly are shown in Table A.5-13.

A.5.4 Evaluation

A.5.4.1 Methods

The following sections describe individual methods used in the evaluation of the MP197HB transportation package.

A.5.4.1.1 Source Methodology

Developing the source requires the use of several different methods and programs to evaluate the authorized contents are acceptable for use and the MP197HB transportation package dose rates are below the regulatory limits as shown in Section A.5.1.3.1.

For the purposes of this section, design basis source may refer to the bounding PWR or BWR assemblies, as the method to generate the sources is equivalent.

A.5.4.1.1.1 Source Homogenization

For the dose calculation around the MP197HB, the source is divided into four separate regions: fuel, plenum, top end fitting, and bottom end fitting for primary gammas. For neutrons, the source is only the active fuel region. The densities of the homogenized regions are calculated by summing all of the material in the region and dividing by the volume. This volume spans the volume of a cuboid that spans the outer envelope of the fuel assembly. The spacing between the fuel pins or other components of the fuel assembly is included in the homogenized volume.

Homogenized assembly region compositions for the BWR *and PWR* design basis fuel assembl*ies* are described in Section A.5.3.2.3.

A.5.4.1.1.2 Burned Fuel Isotopic Composition

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A.5.4.1.1.3 TRITON validation

The TRITON/T-DEPL sequence of SCALE has been validated against publicly available information to assess the uncertainties associated with the source terms.

Almost 100% of the gamma spectrum from light elements is in the range of 0.70 to 1.33 MeV which corresponds exactly to the two most prominent lines of Co-60. Shortly after discharge the emission at higher energies is dominated by actinides. This is true for energies >4 MeV at all cooling times and energy above 3.5 MeV for cooling times greater than 10 years [17]. As for fission products, the main contributors after six years with a fraction greater than 5% in the range of 0.01 to 0.90 MeV are: Sr-90, Y-90, Rh-106 (Ru-106), Cs-137, Pr-144 (Ce-144), Eu-154, and Eu-155. Contributions from Y-90, Rh-106, Cs-137, Pr-144, and Eu-154 are dominant in the range of 0.90 to 1.50 MeV. Rh-106 (Ru-106), Sm-147, Ce-142 and Pr-144 (Ce-144) are the strongest emitters at energies greater than 2.0 MeV. The accuracy of the gamma spectrum is dependent upon the energy. Photon rates computed for fission products tend to be more accurate than those for actinides because the calculation of their inventory has less uncertainty [17].

At discharge, the neutron source is almost equally produced from Cm-242 and Cm-244. The other strong contributor is Cf-252, which is approximately 1/10 of the Cm intensity, but its share vanishes after 6 years of cooling time because the half-life of Cf-252 is 2.65 years. The half-lives of Cm-242 and Cm-244 are 163 days and 18 years, respectively. Contributions from the next strongest emitters, Pu-238 and Pu-240, are lower by a factor of 1000 and 100, respectively, relative to Cm-244. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, Cm-244 represents more than 90% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein.

NUREG/CR-6700 identifies the nuclide characteristics associated with high-burnup fuels greater than 45 GWd/MTU up to 70 GWd/MTU. In particular, NUREG/CR-6700 presents the rankings of the nuclides important for decay heat and radiation sources calculation for shielding analysis.

Section 4.1 of NUREG/CR-6700 identifies that approximately 90% of the decay heat from a 5 wt% - 70 GWd/MTU fuel assembly, after five years cooling time, is due to Cs-134, Ba-137m,

Y-90, Cm-244 (ranging from an 18% down to a 16% contribution for each of these isotopes) and Pu 238, Rh-106, Cs-137 and Sr-90 (from about 8% down to less than 4% contribution for each of these isotopes). Note that Ba-137m and Y-90 are decay products of Cs-137 and Sr-90, respectively. For a 5 wt% - 70 GWd/MTU fuel assembly at 100 years cooling time, 90% of the decay heat is due to Am-241 (31%), Pu-238 (28%), Ba-137m/Cs-137 (17%) and Y-90/Sr-90 (14%).

Section 5 of NUREG/CR-6700 provides fractional contributions of individual radionuclides to the radiation dose rates from three different transport/storage designs: carbon steel transport cask, lead transfer cask, and concrete storage cask. The contributions of dominant radionuclides to dose rates for the three designs considering 5 wt% - 70 GWd/MTU BWR fuels at five years and 100 years cooling time are summarized below. The shielding rankings shown are from Table 12 of NUREG/CR-6700.

Shielding Rankings for Dominant Radionuclides - BWR Fuel

	5 wt% - 70 GWd/MTU BWR Fuel		
	Carbon Steel Design	Lead Design	Concrete Design
	Five Years Cooling Time		
Ba-137m	2.7%	(1)	11.6%
Cm-244	43.9%	45.8%	10.9%
Cs-134	19.1%	18.7%	34.9%
Co-60	10.7%	12.5%	15.1%
Eu-154	8.1%	8.2%	13.5%
Pr-144	6.1%	5.5%	5.4%
Rh-106	6.0%	5.6%	6.4%
total	96.6%	96.3%	97.8%
	100 Years Cooling Time		
Am-241	3.8%	4.8%	(1)
Ba-137m	12.2%	(1)	66.7%
Cm-244	46.2%	51.7%	14.8%
Cm-246	25.0%	28.0%	8.1%
Pu-238	4.0%	4.9%	(1)
Pu-240	3.0%	3.4%	(1)
Pu-242	1.6%	1.8%	(1)
Y-90	3.0%	3.3%	5.8%
total	98.8%	97.9%	95.4%

(1) Not reported, contribution less than 1.5%

Note that one of the major shifts in nuclide importance associated with high-burnup fuel is the increase in contribution of Cm-244 to decay heat and shielding radiation.

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

The benchmark evaluation is performed using publicly available data provided in reference [19] for fuel samples obtained from the Calvert Cliffs, TMI, and Takahama reactors, reference [21] for high burnup fuel obtained from the Vandellos reactor, and reference [20] for fuel samples obtained from Gosgen and GKN II reactors. The burnup and enrichment combinations evaluated are representative of the combinations analyzed for fuel transport in the MP197HB. A summary of the experimental samples utilized in the benchmark evaluation is provided in Table A.5-41a.

Proprietary Information on Pages A.5-16b through A.5-19
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A.5.4.1.1.6 Flux Factors

Fuel assembly hardware is also included in the source term calculation. The hardware source term is primarily due to Co-60. The fuel assembly hardware material masses are provided in Table A.5-6 and Table A.5-7 for PWR and BWR fuel, respectively, and the compositions used to compute the light element masses are provided in Table A.5-9. To account for the reduced neutron flux in the non-active fuel regions of the homogenized assembly, flux factors are employed to scale the light element masses of those regions in the ORIGEN-ARP input. The masses for the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by 0.1, 0.2, and 0.15, respectively [15] in the BWR FA model. The PWR FA model uses the same flux factors for the top fitting and plenum, and 0.2 for the bottom fitting [15].

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Pages A.5-20a through A.5-27 are proprietary information withheld pursuant to 10 CFR 2.390.

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A.5.4.1.3 Fuel Qualification Methodology

The fuel qualification methodology is a process by which authorized contents are verified to be bounded by their respective design basis. The method ties together many different components in a way that efficiently ranks sources based on their corresponding dose rate at the most critical location. The critical location is defined to be the location where, if the critical location meets the regulatory dose rate limits, all other regulatory dose rate limits would be met. For the purposes of the shielding analysis of the MP197HB transportation package, the critical location is at the active fuel midplane, two meters from the projected vertical side surface of the vehicle. This is also referred to as the *location of interest* in the current chapter. For the purposes of fuel qualification, the output from the method generates equivalent sources that result in the same dose rate at any location along the package.

For discussion, the methodology can be split into two parts. As shown in the response function methodology, Section A.5.4.1.2, and specifically, equation (A.5.25*a, b, and c*), both the source and the set of linear, energy-dependent response functions must be known in order to determine the dose rate using this method.

A.5.4.1.3.1 Response Function

The MCNP model for NCT was utilized to calculate a response function at 2 meters from the transportation vehicle. The MCNP model specification regarding axial *burnup* profile variation along in-core axial region of fuel and correction to neutron source strength due to the *burnup* profile variation, treatment of neutron subcritical multiplication, described earlier in Section A.5.2.1 *through* Section A.5.2.3 *are* applicable. MCNP model description in Section A.5.3 regarding radial shielding configuration of the cask and 69BTH DSC, physics specification is applicable.

Considerations and a specification of cylindrical (with *"angular-axial"* segmentation) mesh tallies set up along side of the cask between ends of impact limiters at 2 meters radial distance from surface of impact limiters provided in Section *A.5.3.1.4* is applicable. Fine axial and angular segmentation with 20 axial and 71 angular segments are used, respectively. Size, (*axial length* × *arc length*), of the cylindrical mesh tally grid segments is approximately 34x32 cm. and 38x32 cm over the cask side between impact limiters and over impact limiters, respectively.

Fuel assemblies are grouped by certain radial zones in a description of the radiological sources in MCNP models for calculating response functions. The zoning scheme is very similar to that employed for definition of heat loading zones in DSCs loading options provided in Section A.1.4. There are three and four radial zones for DSCs containing PWR and BWR FAs, respectively. The inner zone encompasses fuel assemblies in 4 central compartments of 24PT4, 24PTH, 32PT, 32PTH and 32PTH1 DSCs and 9 central compartments of 37PTH DSC and DSCs containing BWR FAs. Such a definition of response functions provides flexibility in increasing the overall radiation capacity of the system. Results indicate that more than 93% of the radial dose rates around the cask are due to fuel assemblies in outer radial zones. Therefore, it would be possible to increase the source terms for the fuel assemblies located in the inner zone by a factor of 2.0 and still do not significantly increase the overall dose rate distribution around the cask due to self-shielding provided by the surrounding assemblies.

For each radial zone, response function entries are calculated for neutron, secondary gamma and the following 4 primary gamma radiation energy groups:

- 1.00 – 1.33 MeV,
- 1.33 – 1.66 MeV,
- 2.00 – 2.50 MeV,
- 2.50 – 3.00 MeV.

Response functions are calculated with MCNP for each cask\DSC shielding configuration. Entries of the response functions are essentially source to dose conversion factors as a function of energy (for gamma). The "conversion factors" corresponding to neutron radiation source are multiplied by "*peaking*" factors to account for the axial *burnup* profile of the fuel and subcritical *neutron* multiplication to obtain the response functions that are used for the fuel qualification evaluation. For neutrons and associated (n,γ) radiation, since a bounding energy spectrum is used, the response function calculated is just a total source to dose factor.

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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Pages A.5-29b through A.5-29f are proprietary information withheld pursuant to 10 CFR 2.390.

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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

these fluxes to dose rates using flux-to-dose rate conversion factors as described in Section A.5.4.2. Secondary gammas resulting from neutron capture are also tallied.

Separate models are developed for neutron and gamma source terms. Simple Russian roulette is used as a variance reduction technique for most tallies and geometric based splitting.

A loaded NUHOMS®-24P in the HSM Model 80 loaded with B&W 15x15 Mark B fuel was compared against an MCNP model of the same. The MCNP model was developed to calculate dose rates at the locations where the dose rates were measured on the real system. The results of this comparison are shown in Table A.5-49 as reproduced from Chapter M.5 of the Standardized NUHOMS® System UFSAR [23]. This validation was previously employed to qualify the MCNP methodology for the Standardized NUHOMS® System. The results show that MCNP conservatively predicts total dose rates compared to the measured data. Some conservatism in the methodology used to calculate the source terms still exists and likely contributed to the general over-prediction of the calculated dose rates when compared to the measured dose rates.

MCNP5 was also benchmarked using data from the TN-68 storage system [24]. This is performed by comparing measured dose rates to calculated dose rates at the

surface of the TN-68 cask at the approximate mid-plane of the active fuel. The results of these comparisons are shown in Table A.5-48 and confirm that the MCNP dose rate calculation methodology employed does not add any adverse uncertainty.

The fuel qualification methodology, as described in Section A.5.4.1.3 is also used to generate dose rates for specific configurations as validated by the response function methodology, described in Section A.5.4.1.2. Results of the fuel qualification methodology are discussed in Section A.5.5.2.

A.5.4.2 Flux-to-Dose-Rate Conversion

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.5.4.3 Radiation Levels

A.5.4.3.1 Shear Key

As a result of the shear key cut-out on the side of the cask, the shielding properties in this area vary significantly. Depending on the neutron or gamma radiation source contribution to the total dose rate, the position of the maximum dose rate along the cask side may vary depending not only on axial but also the angular location. The transport cask shear key faces down during the transport. Neutron radiation streaming through the shear key cut-out of the neutron shielding on side of the cask will be within a solid angle not encompassing the accessible area near the transportation package. Therefore, the maximum dose rates determined with cylindrical mesh tallies around the cask side and reported in this chapter are at angular coordinates marked on Figure A.5-6 with P_x, where P_x corresponds to some angular coordinate between 0.25 and 0.75 rotations (including $P_{0.25}$ and $P_{0.75}$) if measuring counter clockwise from an imaginary plane through the cask axis and the shear key (which is designated with P_o on Figure A.5-6). The similar applies to interpretation of dose rates obtained with the response functions during qualification of assemblies for transportation. Due to symmetry of the dose rate distribution, dose rates angular coordinate between 0.25 and 0.50 rotations (including $P_{0.25}$ and $P_{0.50}$) were considered during the qualification.

There is noticeable neutron radiation streaming through the transport cask shear key. On the other hand, when the cask is at a transportation condition the shear key faces down. There is also additional shielding from a trailer or railcar skid and deck. It is observed from the analysis of the shielding configuration with a metal cask on a trailer platform that radiation from bottom portion of the cask side, scattering from concrete surface on the ground is not significant at distances greater than 1 meter from outside boundary of the transportation package. Therefore it is justified to use mesh tally segments located at top quadrants of the cylindrical mesh when considering dose rates along the cask between ends of impact limiters. Since the radiological source term distribution inside of the cask is symmetric in the computation models, only segments within top quadrant(s) of the cylindrical mesh tallies are considered to locate the maximum.

When the cask is in the vertical position the shear key cut-out is closed with the shear key plug made of steel and the same neutron shielding material as on the cask side. Prior to being rotated to the horizontal position and placed on the transport platform, the shear key plug is removed. When the cask is positioned horizontally on the transporter the shear key cut out fits over the steel shear key on the transporter platform. When the cask is removed from the transport platform, the shear key plug is reinstalled.

A.5.4.3.2 NCT Fuel Reconfiguration

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Pages A.5-32 through A.5-32b are proprietary information withheld pursuant to 10 CFR 2.390.

A.5.5 Appendix

A.5.5.1 VYAL-B Mixing and Installation

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A.5.5.2 Fuel Qualification Results

An evaluation was performed using the methodologies described in Section A.5.4 as appropriate for fuel assembly parameters of burnup, percent initial enrichment U-235 and cooling time that would result in normal conditions of transport dose rate at 2.0 meters from the *vehicle* side not exceeding 8.2 mrem/hr. The results are expressed in tabular format showing the minimum required cooling times as a function of enrichment and burnup. Such tables are referred to as Fuel Qualification Tables (FQTs).

After the cooling times to meet 8.2 mrem/hr dose rate requirement were determined, they were rounded up to the nearest 0.5 year in the final transportation FQTs. For example 5.1 is rounded up to 5.5, 8.8 is rounded up to 9.0, etc.

Due to shielding properties of the cask and concentration of most of the radiological source strength in in-core region, dose rate values along the cask side between impact limiters are larger than dose rates at the same distances from ends of impact limiters, regardless of the fuel assembly types loaded, conditions of transport (NCT or HAC) and DSC types. Therefore, in general, if dose rates along the side of the package comply with conditions of transport dose rate restrictions specified at the end of Section A.5.1.3.1, it indicates that shielding performance of the cask meets the regulatory requirements.

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 matrix shown in Table *A.5-33* provides additional cooling times at selected cooling times of CCs. The additional times shown are bounding even when all the fuel outer peripheral compartments contain CCs.
- For PWR fuel assemblies containing CCs, no additional cooling time is needed when they are loaded with cooling times greater than or equal to 15 years.

Note: the guidelines are based **only** on shielding performance evaluation.

A.5.5.2.2 Fuel Qualification Adjustment for Assemblies with Damaged or Reconstituted Rods

The following guidelines should be applied when fuel assemblies with reconstituted rods are considered for transportation in the MP197HB Transportation Cask:

- The maximum number of reconstituted fuel assemblies that can be loaded per DSC is equal to number of assemblies per DSC.
- PWR fuel assemblies may contain up to 10 rods and BWR fuel assemblies may contain up to four rods that are reconstituted with stainless steel that is irradiated.
- There is no limit on a number of rods reconstituted with un-irradiated stainless steel or Zircaloy or low enriched (lower than of an original, un-reconstituted, FA), natural uranium, UO_2 or other non-fuel material.
- There is no effect on the source terms due to the position of the reconstituted rods in the fuel rod array.
- Additional cooling time is required for assemblies in outer peripheral fuel compartments only. *The peripheral fuel compartments are shown in Figures A.5-28 through A.5-29c.*
- For cooling times greater than or equal to 10 years no extra cooling is required and the FQT cooling times for un-reconstituted FAs are also applicable for reconstituted FAs.
- One more year of cooling time is required for reconstituted fuel assemblies cooled for less than 10 years and located in outer peripheral fuel compartments.
- The maximum number of fuel assemblies with irradiated stainless steel reconstituted rods is restricted to four (Chapter A.1, Appendix A.1.4.1 through A.1.4.9) thereby ensuring that the analysis is conservative.

Note: the guidelines are based **only** on shielding performance evaluation.

A.5.5.3 Decay Heat Restrictions

The various DCS payloads allowed for transportation in the MP197HB cask have individual decay heat load limitations. There are numerous DSC loading options to meet those restrictions.

Decay heat values for various discrete decay heat power levels per fuel assembly can be obtained using the decay heat equations.

The decay heat equation (DHE) is established to calculate the decay heat of spent fuel assemblies as a function of enriched, burnup, and cooling time. Two DHEs are established: for BWR (maximum of 0.198 MTU) and PWR (maximum of 0.490 MTU) fuel assemblies. Decay heat data sets for various burn-up, enrichment and cooling time combinations are generated using DB BWR and DB PWR FA SAS2H\ORIGEN-S models *[17]*. A non linear regression analysis was performed on those data sets to obtain a fit of BWR and PWR decay heat data as a function of burnup, enrichment and cooling time. An uncertainty of 10 and 20 watts should be added to the calculated with the DHE decay heat values for BWR and PWR FAs when using the equations, respectively. These values are 2.5 and 2.3 times the standard deviations from the corresponding regression calculations. These uncertainties ensure that the calculated decay heat values from the DHEs are bounding for the DB BWR and PWR FAs. *T*he DHEs are established on data relevant to DB FAs. *As a result,* the equations predict conservative decay heat values for all other FA designs.

Thermal power was validated in reference [18]. The results show that the decay heat predicted by ORIGEN-S is within the range of uncertainty of the measurements in the reference. Specifically, the mean error was shown to be 0.7 W with an uncertainty of 17.6 W. Additionally, very similar calculated results were obtained using SAS2H and TRITON physics models. The difference was found to be less than 2% between the sequences.

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.

A.5.6 References

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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.5-2
Summary of MP197HB HAC Dose Rates
(Exclusive Use Package for Transportation)

	37PTH DSC in the MP197HB Transportation Package		
	1 m from Package Surface (mrem/hr), Limit = 1000 mrem/hr		
	Top(1)	**Side(1)**	**Bottom(2)**
Gamma	0.44 ± 0.09	5.99± 0.48	3.23 ± 0.10
Neutron	70.73± 1.41	843.57 ± 8.44	266.98 ± 2.67
(n,g)	0.20 ± 0.02	1.98 ± 0.06	1.00 ± 0.01
Total	71.36 ± 1.43	851.53 ± 8.52	271.22 ± 2.71

	69BTH DSC in the MP197HB Transportation Package		
	1 m from Package Surface (mrem/hr), Limit = 1000 mrem/hr		
	Top(1)	**Side(1)**	**Bottom(2)**
Gamma	0.25 ± 0.08	2.78 ± 0.31	3.66 ± 0.07
Neutron	61.78 ± 1.24	844.39 ± 8.44	333.65 ± 3.34
(n,g)	0.21 ± 0.02	2.37 ± 0.07	1.25 ± 0.01
Total	62.24 ± 1.24	849.54 ± 8.50	338.56 ± 3.39

Notes:
1. From Configuration 1 model (intact fuel).
2. From Configuration 3 model (reconfigured fuel, *50% void fraction,* flat axial source distribution.)

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.5-3
Not used

Table A.5-*4*
MP197HB Cask/69BTH DSC Shielding Materials

69BTH DSC shielding	**Materials**	**Thickness, inches**
Outer Side Shell	Stainless Steel(1)	0.5
Bottom Shield Plug/Covers	Stainless Steel(1)	7.25(2)
Top Shield Plugs/Covers	Stainless Steel(1)	9.75(2)
MP197HB shielding	**Materials**	**Thickness, inches**
Inner Shell	Carbon Steel	1.25
Gamma Shield	Lead	3.00
Outer Shell	Carbon Steel	2.75
Neutron Shield	VYAL-B	6.0(3)
Neutron Shield Shell	Carbon Steel	0.375
Cask Bottom	Carbon Steel	6.5" 3" if below grapple ring cut-out
Cask Lid	Carbon Steel	4.5
Impact Limiters Outer Shell	Stainless Steel(1)	0.25
Impact Limiters	Wood	26.25(4) –axial Ø125.53"

Notes:
(1) Modeled as carbon steel.
(2) This is a combined thickness of shield plugs and cover plates.
(3) Neutron shielding material VYAL-B resin (composition shown below) is inside of 0.125" thick neutron shield box.
(4) This is measured from ends of the cask to ends of Impact Limiters.

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Table A.5-*6*
DB PWR Fuel Assembly Material Mass

Fuel Assembly Region, length	Fuel Assembly Part	Material	Standard Mass (kg)	Mass of Reconstituted Fuel Assembly (kg)
Top Nozzle, 6.23 in.	Top Nozzle/Misc. Steel	SS-304	9.2	9.2
	Hold Down Spring	Inconel-718	1.8	1.8
Plenum, 8.73 in.	Upper Spring	Inconel-718	4.3	4.3
	Upper End Cap	Zircaloy-4	1.0	1.0
	Encompassing Clad.	Zircaloy-4	5.8	5.5
	Upper End Grid	Inconel-718	1.1	1.1
	Stainless Steel Rods	SS304	n/a	1.7
In-core Region, 142.29 in.	Fuel Stack	Uranium	490(1)	466(1)
	Encompassing Clad.	Zircaloy-4	101.1	96.2
	Encompassing Guide Tube	Zircaloy-4	6.3	6.3
	Spacer Grids	Inconel-718	5.0	5.0
	Grid Supports	Zircaloy-4	0.64	0.64
	Stainless Steel Rods	SS304	n/a	27.2
Bottom Nozzle, 8.38 in.	Lower End Plug	Zircaloy-4	8.9	8.5
	Encompassing Guide Tube	Zircaloy-4	0.1	0.1
	Lower Guide Tube Plugs	Zircaloy-4	1.4	1.4
	Lower End Fitting	SS 304	8.2	8.2
	Lower End Grid	Inconel-718	1.1	1.1
	Stainless Steel Rods	SS304	n/a	0.5

(1) wt. of UO_2 = 555.93 kg (Standard Mass) and 528.70 kg (Reconstituted)

Table A.5-7
DB BWR Fuel Assembly Material Mass

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

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

Table A.5-8
TRITON Input Parameters

(Part 1 of 2)

GE 7x7	
Parameter	***Value***
Assembly Pitch (cm)	*15.24*
Number of Fuel Rods	*49*
Fuel Cell Pitch (cm)	*1.87452*
Pellet OR (cm)	*0.61849*
Fuel Rod Cladding IR (cm)	*0.63373*
Fuel Rod Cladding OR (cm)	*0.71501*
Cladding Thickness (cm)	*0.08128*
Fuel Rod Cladding Material	*Zirc-4*
Square Channel Inside Dimension (cm)	*13.40612*
Channel Thickness (cm)	*0.2032*
Channel Material	*Zirc-4*
Specific Power (MW/MTU)	*40.0*
Average Fuel Temperature (K)	*850.0*
Average Cladding Temperature (K)	*620.0*
Average Moderator Temperature (K)	*560.0*
Average Moderator Density (g/cm^3)	*0.20, 0.43, and 0.80*

(Part 2 of 2)

B&W 15x15	
Parameter	***Value***
Assembly Pitch (cm)	*21.79690*
Number of Fuel Rods	*208*
Guide Tubes per Assembly	*17*
Fuel Cell Pitch (cm)	*1.44272*
Pellet OR (cm)	*0.473329*
Fuel Rod Cladding IR (cm)	*0.478790*
Fuel Rod Cladding OR (cm)	*0.546100*
Cladding Thickness (cm)	*0.067310*
Fuel Rod Cladding Material	*Zirc-4*
Guide Tube IR (cm)	*0.635000*
Guide Tube OR (cm)	*0.670560*
Guide Tube Thickness (cm)	*0.035560*
Guide Tube Material	*Zirc-4*
Specific Power (MW/MTU)	*40.0*
Average Fuel Temperature (K)	*950.0*
Average Cladding Temperature (K)	*640.0*
Average Moderator Temperature (K)	*575.0*
Average Moderator Density (g/cm^3)	*0.7232*

Table A.5-9
Material Compositions for Fuel Assembly Hardware Materials

Element	Atomic Number	Material Composition, grams per kg of material				
		Zircaloy-4	Inconel-718	Inconel X-750	Stainless Steel 304	UO_2 Fuel (per kg U)
H	1	1.30E-02	-	-	-	-
Li	3	-	-	-	-	1.00E-03
B	5	3.30E-04	-	-	-	1.00E-03
C	6	1.20E-01	4.00E-01	3.99E-01	8.00E-01	8.94E-02
N	7	8.00E-02	1.30E+00	1.30E+00	1.30E+00	2.50E-02
O	8	9.50E-01	-	-	-	1.34E+02
F	9	-	-	-	-	1.07E-02
Na	11	-	-	-	-	1.50E-02
Mg	12	-	-	-	-	2.00E-03
Al	13	2.40E-02	5.99E+00	7.98E+00	-	1.67E-02
Si	14	-	2.00E+00	2.99E+00	1.00E+01	1.21E-02
P	15	-	-	-	4.50E-01	3.50E-02
S	16	3.50E-02	7.00E-02	7.00E-02	3.00E-01	-
Cl	17	-	-	-	-	5.30E-03
Ca	20	-	-	-	-	2.00E-03
Ti	22	2.00E-02	7.99E+00	2.49E+01	-	1.00E-03
V	23	2.00E-02	-	-	-	3.00E-03
Cr	24	1.25E+00	1.90E+02	1.50E+02	1.90E+02	4.00E-03
Mn	25	2.00E-02	2.00E+00	6.98E+00	2.00E+01	1.70E-03
Fe	26	2.25E+00	1.80E+02	6.78E+01	6.88E+02	1.80E-02
Co	27	1.00E-02	4.69E+00	6.49E+00	8.00E-01	1.00E-03
Ni	28	2.00E-02	5.20E+02	7.22E+02	8.92E+01	2.40E-02
Cu	29	2.00E-02	9.99E-01	4.99E-01	-	1.00E-03
Zn	30	-	-	-	-	4.03E-02
Zr	40	9.79E+02	-	-	-	-
Nb	41	-	5.55E+01	8.98E+00	-	-
Mo	42	-	3.00E+01	-	-	1.00E-02
Ag	47	-	-	-	-	1.00E-04
Cd	48	2.50E-04	-	-	-	2.50E-02
In	49	-	-	-	-	2.00E-03
Sn	50	1.60E+01	-	-	-	4.00E-03
Gd	64	-	-	-	-	2.50E-03
Hf	72	7.80E-02	-	-	-	-
W	74	2.00E-02	-	-	-	2.00E-03
Pb	82	-	-	-	-	1.00E-03
U	92	2.00E-04	-	-	-	1.00E+03

Table A.5-10
BWR Axial *Source Distributions for Intact Fuel*

Zone	*Fraction of Core Height*	*Fractional Width of Zone*	*Axial Burnup Profile for 40 GWd/MTU*	*Gamma Axial Source Distribution (normalized to avg = 1.0)*	*Gamma Axial Source Distribution (normalized to sum = 1.0)*	*Neutron Axial Source Distribution (not normalized)*	*Neutron Axial Source Distribution (normalized to sum = 1.0)*
1	*0.05*	*0.05*	*0.2357*	*0.2256*	*0.0113*	*0.0026*	*9.181E-05*
2	*0.1*	*0.05*	*0.7746*	*0.7674*	*0.0384*	*0.3468*	*0.0123*
3	*0.2*	*0.1*	*1.0750*	*1.0854*	*0.1085*	*1.3879*	*0.0984*
4	*0.3*	*0.1*	*1.1836*	*1.2027*	*0.1203*	*2.0923*	*0.1483*
5	*0.4*	*0.1*	*1.2000*	*1.2223*	*0.1222*	*2.2321*	*0.1582*
6	*0.5*	*0.1*	*1.2000*	*1.2244*	*0.1224*	*2.2475*	*0.1593*
7	*0.6*	*0.1*	*1.1912*	*1.2164*	*0.1216*	*2.1893*	*0.1552*
8	*0.7*	*0.1*	*1.1515*	*1.1227*	*0.1123*	*1.5887*	*0.1126*
9	*0.8*	*0.1*	*1.0766*	*1.0964*	*0.1096*	*1.4450*	*0.1024*
10	*0.9*	*0.1*	*0.8973*	*0.9053*	*0.0905*	*0.6717*	*0.0476*
11	*0.95*	*0.05*	*0.6330*	*0.6255*	*0.0313*	*0.1531*	*0.0054*
12	*1.0*	*0.05*	*0.2410*	*0.2303*	*0.0115*	*0.0028*	*9.970E-05*
Total	-	-	-	-	*1.000*	-	*1.000*

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Table A.5-11
PWR Axial *Source Distributions for Intact Fuel*

Zone	*Active Fuel Zone Center (% of Height)*	*Fractional width of Zone*	*Axial Burnup Profile for >46 GWd/MTU[16]*	*Gamma Axial Source Distribution (normalized to sum = 1.0)*	*Neutron Axial Source Distribution (not normalized)*	*Neutron Axial Source Distribution (normalized to sum = 1.0)*
1	*2.78*	*0.056*	*0.573*	*0.0318*	*0.108*	*0.0052*
2	*8.33*	*0.056*	*0.917*	*0.0509*	*0.707*	*0.0341*
3	*13.89*	*0.056*	*1.066*	*0.0592*	*1.291*	*0.0623*
4	*19.44*	*0.056*	*1.106*	*0.0614*	*1.496*	*0.0721*
5	*25.00*	*0.056*	*1.114*	*0.0619*	*1.540*	*0.0743*
6	*30.56*	*0.056*	*1.111*	*0.0617*	*1.524*	*0.0735*
7	*36.11*	*0.056*	*1.106*	*0.0614*	*1.496*	*0.0721*
8	*41.69*	*0.056*	*1.101*	*0.0612*	*1.469*	*0.0708*
9	*47.22*	*0.056*	*1.097*	*0.0609*	*1.448*	*0.0698*
10	*52.78*	*0.056*	*1.093*	*0.0607*	*1.427*	*0.0688*
11	*58.33*	*0.056*	*1.089*	*0.0605*	*1.406*	*0.0678*
12	*63.89*	*0.056*	*1.086*	*0.0603*	*1.391*	*0.0671*
13	*69.44*	*0.056*	*1.081*	*0.0601*	*1.366*	*0.0659*
14	*75.00*	*0.056*	*1.073*	*0.0596*	*1.326*	*0.0639*
15	*80.56*	*0.056*	*1.051*	*0.0584*	*1.220*	*0.0588*
16	*86.11*	*0.056*	*0.993*	*0.0552*	*0.972*	*0.0469*
17	*91.67*	*0.056*	*0.832*	*0.0462*	*0.479*	*0.0231*
18	*97.22*	*0.056*	*0.512*	*0.0284*	*0.069*	*0.0033*
Total	-	-	-	*1.000*	-	*1.000*

Table A.5-11a
PWR Axial Source Distributions for Reconfigured Fuel

Zone	*Fraction of Core Length from Bottom Nozzle (%)*	*Axial Burnup Profile for 44-55 GWd/MTU [26]*	*Gamma Axial Source Distribution (normalized to sum = 1.0)*	*Neutron Axial Source Distribution (not normalized)*	*Neutron Axial Source Distribution (normalized to sum = 1.0)*
1	5	0.655	0.033	0.183	0.008
2	10	0.911	0.046	0.687	0.032
3	15	1.009	0.050	1.037	0.048
4	20	1.041	0.052	1.175	0.054
5	25	1.069	0.053	1.308	0.060
6	30	1.072	0.054	1.322	0.061
7	35	1.072	0.054	1.322	0.061
8	40	1.071	0.054	1.318	0.061
9	45	1.070	0.054	1.313	0.060
10	50	1.069	0.053	1.308	0.060
11	55	1.069	0.053	1.308	0.060
12	60	1.068	0.053	1.303	0.060
13	65	1.068	0.053	1.303	0.060
14	70	1.069	0.053	1.308	0.060
15	75	1.068	0.053	1.303	0.060
16	80	1.066	0.053	1.293	0.060
17	85	1.041	0.052	1.175	0.054
18	90	0.994	0.050	0.976	0.045
19	95	0.879	0.044	0.595	0.027
20	100	0.639	0.032	0.165	0.008
Total	-	-	1.000	-	1.000

Table A.5-12
Fuel Assembly Materials for MCNP
(Part 1 of *4*)

BWR Assembly Region Material Densities					
Element/Isotope	**Atomic Number**	**Density (g/cm^3)**			
		Bottom Fitting	**Fuel**	**Plenum**	**Top Fitting**
C	6	1.166E-3	6.759E-6	1.340E-4	2.784E-4
O	8	-	3.806E-1	-	-
Si	14	1.496E-2	2.599E-4	1.675E-3	5.237E-3
P	15	6.558E-4	3.802E-6	7.535E-5	1.566E-4
Ti	22	3.876E-4	1.754E-4	-	1.757E-3
Cr	24	2.796E-1	3.390E-3	3.260E-2	7.687E-2
Mn	25	2.915E-2	1.690E-4	3.349E-3	6.961E-3
Fe	26	9.984E-1	7.806E-3	1.161E-1	2.433E-1
Ni	28	1.498E-1	5.925E-3	1.591E-2	8.435E-2
Zr	40	3.838E-1	7.195E-1	7.660E-1	2.022E-1
Sn	50	5.665E-3	1.062E-2	1.131E-2	2.985E-3
Hf	72	3.907E-5	7.325E-5	7.799E-5	2.059E-5
U-234	92	-	1.008E-3	-	-
U-235	92	-	1.132E-1	-	-
U-236	92	-	5.209E-4	-	-
U-238	92	-	2.716E+0	-	-
Total		1.864	3.960	0.947	0.624

Table A.5-12
Fuel Assembly Materials for MCNP

(Part 2 of 4)

BWR Assembly Region Material Densities					
Element/Isotope	Atomic Number	Composition By Weight Fraction			
		Bottom Fitting	Fuel	Plenum	Top Fitting
C	6	0.00063	0.000002	0.00014	0.00045
O	8	-	0.096131	-	-
Si	14	0.00803	0.000066	0.00177	0.00839
P	15	0.00035	0.000001	0.00008	0.00025
Ti	22	0.00021	0.000044	-	0.00281
Cr	24	0.15004	0.000856	0.03441	0.12316
Mn	25	0.01564	0.000043	0.00354	0.01115
Fe	26	0.53574	0.001971	0.12259	0.38983
Ni	28	0.08037	0.001496	0.01679	0.13514
Zr	40	0.20594	0.181709	0.80866	0.32400
Sn	50	0.00304	0.002682	0.01194	0.00478
Hf	72	0.00002	0.000018	0.00008	0.00003
U-234	92	-	0.000255	-	-
U-235	92	-	0.028599	-	-
U-236	92	-	0.000132	-	-
U-238	92	-	0.685995	-	-
Gram Density (g/cc)		1.864	3.960	0.947	0.624

Table A.5-12
Fuel Assembly Materials for MCNP

(Part 3 of 4)

PWR Assembly Region Material Densities					
		Number Density (atom/b-cm)			
Element	Atomic Number	Bottom End Fitting	Fuel	Plenum	Top End Fitting
O	8	-	1.47E-02	-	-
Al	13	1.43E-05	3.93E-06	6.95E-05	3.24E-05
Ti	22	1.07E-05	2.95E-06	5.22E-05	2.43E-05
Cr	24	2.05E-03	7.19E-05	1.15E-03	3.25E-03
Mn	25	1.79E-04	-	-	2.71E-04
Fe	26	6.48E-03	9.19E-05	1.40E-03	9.97E-03
Ni	28	1.31E-03	1.56E-04	2.76E-03	2.42E-03
Zr	40	6.77E-03	4.12E-03	4.23E-03	-
Mo	42	2.01E-05	5.53E-06	9.77E-05	4.56E-05
Sn	50	8.49E-05	5.16E-05	5.30E-05	-
U-235	92	-	3.69E-04	-	-
U-238	92	-	6.93E-03	-	-
Total		1.69E-02	2.65E-02	9.81E-03	1.60E-02

Table A.5-12
Fuel Assembly Materials for MCNP

(Part 4 of 4)

PWR Assembly Region Material Densities				
Element	**Bottom Region**	**In-Core Region**	**Plenum Region**	**Top Region**
H	-	-	-	-
B-10	-	-	-	-
C	-	-	-	-
O	-	3.89E-01	-	-
Al	6.40E-04	1.76E-04	3.11E-03	1.45E-03
Ti	8.54E-04	2.35E-04	4.15E-03	1.94E-03
Cr	1.77E-01	6.21E-03	9.93E-02	2.80E-01
Mn	1.63E-02	-	-	2.47E-02
Fe	6.01E-01	8.52E-03	1.30E-01	9.25E-01
Ni	1.28E-01	1.52E-02	2.69E-01	2.35E-01
Zr	1.03E+00	6.24E-01	6.41E-01	-
Mo	3.20E-03	8.80E-04	1.56E-02	7.26E-03
Sn	1.67E-02	1.02E-02	1.04E-02	-
U-235	-	1.44E-01	-	-
U-238	-	2.74E+00	-	-
Total	1.969	3.938	1.172	1.476

Table A.5-13
Package Materials Input for MCNP

Element	Atomic Number, Z	Weight fraction, (%)					
		Air	Lead	Carbon Steel	Stainless Steel	Aluminum	Wood
H	1						6.218
C	6	0.01		1	2		44.450
N	7	75.52					
O	8	23.18					49.333
Al	13					100	
Ar	18	1.29					
Cr	24				19		
Fe	26			99	69.5		
Ni	28				9.5		
Pb	82		100				
Density, g/cm^3		1.127E-3	11.34	7.82	7.92	2.72(1)	0.125

Notes:
(1) Aluminum density is 2.702 g/cc. This discrepancy does not affect dose rates significantly.
(2) For VYAL-B, see Table A.5-5.

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Table A.5-16
Flux-to-Dose Rate Conversion Factors for Gamma

Photon Energy (MeV)	**Conversion Factor (mrem/hr)/(γ/cm^2-s)**
0.01	3.96E-03
0.03	5.82E-04
0.05	2.90E-04
0.07	2.58E-04
0.1	2.83E-04
0.15	3.79E-04
0.2	5.01E-04
0.25	6.31E-04
0.3	7.59E-04
0.35	8.78E-04
0.4	9.85E-04
0.45	1.08E-03
0.5	1.17E-03
0.55	1.27E-03
0.6	1.36E-03
0.65	1.44E-03
0.7	1.52E-03
0.8	1.68E-03
1	1.98E-03
1.4	2.51E-03
1.8	2.99E-03
2.2	3.42E-03
2.6	3.82E-03
2.8	4.01E-03
3.25	4.41E-03
3.75	4.83E-03
4.25	5.23E-03
4.75	5.60E-03
5	5.80E-03
5.25	6.01E-03
5.75	6.37E-03
6.25	6.74E-03
6.75	7.11E-03
7.5	7.66E-03
9	8.77E-03
11	1.03E-02
13	1.18E-02
15	1.33E-02

Table A.5-*17*
Flux-Dose-Rate Conversion Factors for Neutron

Neutron Energy (MeV)	**Conversion Factor (mrem/hr)/(n/cm^2-s)**
2.50E-08	3.67E-03
1.0E -07	3.67E-03
1.00E-06	4.46E-03
1.00E-05	4.54E-03
1.00E-04	4.18E-03
1.00E-03	3.76E-03
1.00E-02	3.56E-03
1.00E-01	2.17E-02
5.00E-01	9.26E-02
1	1.32E-01
2.5	1.25E-01
5	1.56E-01
7	1.47E-01
10	1.47E-01
14	2.08E-01
20	2.27E-01

Table A.5-*18*
CC Radiological Source

E_{upper} (MeV)	E_{mean} (MeV)	**Top Region γ/s/CC**	**Plenum Region γ/s/CC**	**Fuel Region γ/s/CC**
0.05	0.025	8.484E+10	1.382E+11	1.170E+12
0.1	0.075	8.191E+09	5.176E+09	1.142E+11
0.2	0.15	1.217E+09	3.534E+09	1.697E+10
0.3	0.25	2.060E+08	3.482E+09	2.872E+09
0.4	0.35	7.014E+07	2.431E+10	9.776E+08
0.6	0.5	3.235E+07	3.854E+10	4.508E+08
0.8	0.7	4.155E+09	1.783E+10	2.835E+09
1	0.9	6.951E+09	4.180E+09	4.699E+09
1.33	1.165	1.690E+12	9.340E+11	2.356E+13
1.66	1.495	7.138E+11	3.946E+11	9.953E+12
2	1.83	9.577E+01	8.812E+01	9.153E-05
2.5	2.25	1.274E+07	7.040E+06	1.775E+08
3	2.75	3.942E+04	2.179E+04	5.495E+05
4	3.5	3.947E-15	6.520E-14	7.266E-18
Total		2.509E+12	1.564E+12	3.483E+13

Pages A.5-59 through A.5-72 are proprietary information withheld pursuant to 10 CFR 2.390

Table A.5-*33*
Matrix Showing Additional to the Transportation FQTs Cooling Times
for Selected Cooling Times of CCs

Cooling Time of CCs, years	**Additional Cooling Time in Years for FAs with Burn-Up**	
	<=40 GWD/MTU	**>40 GWD/MTU**
3.0	0.8	2.5
5.0	0.6	2.0
10.0	0.3	1.0
15.0	0.15	0.5

Notes: To find additional to the transportation FQT cooling times when the cooling times of Control Components (CCs) are not shown in the first column of the table use the nearest higher CC cooling time available in the table. For example, if cooling time of CCs is 8.0 years use data in the table relevant to 10.0 years cooled CCs.

Table A.5-34
Summary *of* Maximum Dose Rate*s* of the Cask Containing *the Radioactive* Waste Canister

Radial Distance from Side of ILs or Body (1), m	Normal Conditions of Transport (NCT)		Hypothetical Accident Condition	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
Shield Shell	64.0	0.01	103.0	0.01
Package Side Perimeter	30.2	0.01	103.0	0.01
0	24.1	0.01	74.3	0.01
1	12.7	0.005	49.1	0.01
2	8.33	0.01	31.2	0.01
2.7	6.4	0.01	20.9	0.01

(1) HAC dose rates for distances equal to 1 and 2 meters correspond to radial distances measured from the cask body (Shield Shell), not from side of Impact Limiters.

Axial Distance from IL, m	Bottom at NCT		Top at NCT	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	72.2	0.002	37.1	0.002
1	18.1	0.002	14.9	0.002
2	8.29	0.003	7.7	0.002
2.7(2)	5.4	0.004	5.2	0.003
4.3	2.7	0.006	2.7	0.005
5.8	1.6	0.007	1.7	0.006
7.3	1.1	0.003	1.1	0.007

Axial Distance from IL, m	Bottom at HAC		Top at HAC	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	198.7	0.003	68.5	0.003
1	37.3	0.003	28.4	0.003
2	15.7	0.005	14.1	0.004
3.0(2)	8.4	0.01	8.1	0.005
4.6	4.3	0.01	4.3	0.007
6.1	2.6	0.01	2.7	0.010
7.6	1.7	0.01	1.8	0.004

(2) Dose rates at distances greater than 2 meters from ends of the Impact Limiters correspond to edges of 40, 50, 60 feet long transportation "platform," respectively.

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.5-36
Bounding NCT Source for the 37PTH DSC in the MP197HB Transportation Package, *Intact Fuel*

NCT Representative Radiological Source at 490 kgU/FA: 60 GWD/MTU, 3.9 wt. %, after 15.7 years of cooling. Generates 1.302 kWt/FA of decay heat. Results in 8.20 mrem/hr NCT total dose rate (~77.9 % is due to Neutron Source)						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	In-core	Plenum	Top Nozzle
0.00e+00	to	5.00e-02	7.480E+10	1.061E+15	1.571E+11	4.590E+10
5.00e-02	to	1.00e-01	1.253E+10	2.932E+14	2.760E+10	8.664E+09
1.00e-01	to	2.00e-01	4.017E+09	2.059E+14	7.314E+09	2.095E+09
2.00e-01	to	3.00e-01	2.221E+08	6.129E+13	3.816E+08	1.049E+08
3.00e-01	to	4.00e-01	4.323E+08	3.961E+13	5.879E+08	1.366E+08
4.00e-01	to	6.00e-01	5.322E+09	6.678E+13	3.486E+09	9.999E+06
6.00e-01	to	8.00e-01	4.298E+09	2.006E+15	9.561E+09	1.295E+09
8.00e-01	to	1.00e+00	1.656E+09	4.291E+13	7.839E+09	1.372E+09
1.00e+00	to	1.33e+00	3.632E+12	8.431E+13	8.018E+12	2.523E+12
1.33e+00	to	1.66e+00	1.026E+12	1.560E+13	2.264E+12	7.126E+11
1.66e+00	to	2.00e+00	7.940E+01	1.034E+11	5.176E+01	1.232E-02
2.00e+00	to	2.50e+00	2.454E+07	6.388E+09	5.418E+07	1.705E+07
2.50e+00	to	3.00e+00	2.097E+04	5.143E+08	4.629E+04	1.457E+04
3.00e+00	to	4.00e+00	1.272E-05	8.797E+07	6.509E-05	1.047E-05
4.00e+00	to	5.00e+00	3.990E-34	2.666E+07	1.996E-34	0.0
5.00e+00	to	6.50e+00	1.150E-34	1.070E+07	5.750E-35	0.0
6.50e+00	to	8.00e+00	1.462E-35	2.099E+06	7.314E-36	0.0
8.00e+00	to	1.00e+01	1.951E-36	4.456E+05	9.761E-37	0.0
Total Gamma, g/(sec*FA)			4.761E+12	3.876E+15	1.050E+13	3.296E+12
(1)Total Neutrons, n/(sec*FA)			*7.705E+8 (raw)* *1.212E+9 (adjusted)*			

(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 the burnup profile in the active fuel region, where k_{eff} = 0.2678 and bpf = 1.152.*

Table A.5-37
Bounding NCT Source for the 69BTH DSC in the MP197HB Transportation Package, *Intact Fuel*

NCT Representative Radiological Source at *198kgU*/FA: 59 GW*d*/MTU, 2.8 wt. %, after 13.7 years of cooling. Generates 0.435 kWt/FA of decay heat. Results in 8.20 mrem/hr NCT total dose rate (~87.6% is due to Neutron Source).						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	In-core	Plenum	Top Nozzle
0.00e+00	to	5.00e-02	3.310E+10	4.226E+14	1.831E+10	2.531E+10
5.00e-02	to	1.00e-01	6.247E+09	1.125E+14	1.978E+09	4.784E+09
1.00e-01	to	2.00e-01	1.660E+09	8.072E+13	1.401E+09	1.262E+09
2.00e-01	to	3.00e-01	8.565E+07	2.375E+13	8.963E+07	6.516E+07
3.00e-01	to	4.00e-01	1.333E+08	1.525E+13	2.504E+08	1.007E+08
4.00e-01	to	6.00e-01	7.952E+08	4.151E+13	4.962E+09	5.776E+08
6.00e-01	to	8.00e-01	4.271E+08	8.559E+14	2.588E+09	3.592E+08
8.00e-01	to	1.00e+00	1.421E+08	2.386E+13	4.260E+07	1.367E+08
1.00e+00	to	1.33e+00	1.817E+12	3.177E+13	5.593E+11	1.394E+12
1.33e+00	to	1.66e+00	5.132E+11	5.939E+12	1.579E+11	3.935E+11
1.66e+00	to	2.00e+00	7.616E+00	4.140E+10	5.025E+01	5.785E+00
2.00e+00	to	2.50e+00	1.228E+07	4.085E+09	3.779E+06	9.416E+06
2.50e+00	to	3.00e+00	1.049E+04	3.380E+08	3.229E+03	8.045E+03
3.00e+00	to	4.00e+00	1.297E-06	6.556E+07	9.489E-12	1.061E-05
4.00e+00	to	5.00e+00	0.0	1.595E+07	0.0	0.0
5.00e+00	to	6.50e+00	0.0	6.403E+06	0.0	0.0
6.50e+00	to	8.00e+00	0.0	1.256E+06	0.0	0.0
8.00e+00	to	1.00e+01	0.0	2.667E+05	0.0	0.0
Total Gamma, g/(sec*FA)			2.373E+12	1.614E+15	7.468E+11	1.820E+12
(1)Total Neutrons, n/(sec*FA)			4.636E+8 *(raw)* 6.769E+8 *(adjusted)*			

(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 the burnup profile in the active fuel region, where k_{eff} = 0.1699 and bpf = 1.212.*

Table A.5-38

Bounding NCT and HAC Source for the 37PTH DSC, Reconfigured Fuel, 21 Inner Compartments

62 GWd/MTU, 3.4 wt. %, after 20.7 years of cooling.						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	In-core	Plenum	Top Nozzle
0.00e+00	to	5.00e-02	4.190E+10	9.278E+14	9.356E+10	2.741E+10
5.00e-02	to	1.00e-01	7.226E+09	2.580E+14	1.596E+10	5.008E+09
1.00e-01	to	2.00e-01	2.049E+09	1.726E+14	4.060E+09	1.211E+09
2.00e-01	to	3.00e-01	1.097E+08	5.205E+13	2.108E+08	6.122E+07
3.00e-01	to	4.00e-01	1.861E+08	3.394E+13	2.991E+08	7.931E+07
4.00e-01	to	6.00e-01	1.630E+09	3.412E+13	1.075E+09	6.543E+06
6.00e-01	to	8.00e-01	2.503E+09	1.814E+15	8.966E+09	1.404E+09
8.00e-01	to	1.00e+00	1.690E+09	2.333E+13	8.322E+09	1.418E+09
1.00e+00	to	1.33e+00	2.097E+12	5.405E+13	4.626E+12	1.457E+12
1.33e+00	to	1.66e+00	5.921E+11	9.141E+12	1.306E+12	4.113E+11
1.66e+00	to	2.00e+00	8.430E+01	8.750E+10	5.495E+01	1.278E-02
2.00e+00	to	2.50e+00	1.417E+07	4.707E+09	3.126E+07	9.842E+06
2.50e+00	to	3.00e+00	1.210E+04	4.287E+08	2.671E+04	8.409E+03
3.00e+00	to	4.00e+00	1.335E-05	8.973E+07	6.834E-05	1.100E-05
4.00e+00	to	5.00e+00	1.215E-33	3.018E+07	6.079E-34	0.0
5.00e+00	to	6.50e+00	3.502E-34	1.211E+07	1.752E-34	0.0
6.50e+00	to	8.00e+00	4.454E-35	2.376E+06	2.228E-35	0.0
8.00e+00	to	1.00e+01	5.944E-36	5.045E+05	2.973E-36	0.0
Total Gamma, g/(sec*FA)			2.746E+12	3.379E+15	6.065E+12	1.904E+12
[(1)]Total Neutrons, n/(sec*FA)			8.752E+08 (raw) 1.345E+09 (adjusted intact) 1.403E+09 (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=1.152, k_{eff}=0.2505 if fuel is intact and k_{eff}=0.2812 if fuel is reconfigured. k_{eff} for reconfigured fuel includes an additional 12.25% increase to account for fuel compression.*

Table A.5-38a
Bounding NCT and HAC Source for the 37PTH DSC, Reconfigured Fuel, 16 Peripheral Compartments

62 GWd/MTU, 3.4 wt. %, after 29.5 years of cooling.						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	In-core	Plenum	Top Nozzle
0.00e+00	to	5.00e-02	1.405E+10	7.457E+14	3.435E+10	9.709E+09
5.00e-02	to	1.00e-01	2.282E+09	2.148E+14	5.081E+09	1.586E+09
1.00e-01	to	2.00e-01	5.854E+08	1.325E+14	1.260E+09	3.849E+08
2.00e-01	to	3.00e-01	3.130E+07	4.079E+13	6.798E+07	2.023E+07
3.00e-01	to	4.00e-01	4.449E+07	2.721E+13	8.651E+07	2.555E+07
4.00e-01	to	6.00e-01	1.780E+08	2.064E+13	1.229E+08	3.145E+06
6.00e-01	to	8.00e-01	1.746E+09	1.470E+15	8.468E+09	1.403E+09
8.00e-01	to	1.00e+00	1.625E+09	1.148E+13	8.178E+09	1.373E+09
1.00e+00	to	1.33e+00	6.589E+11	2.326E+13	1.454E+12	4.577E+11
1.33e+00	to	1.66e+00	1.861E+11	3.422E+12	4.105E+11	1.293E+11
1.66e+00	to	2.00e+00	6.787E+01	6.964E+10	4.424E+01	1.035E-02
2.00e+00	to	2.50e+00	4.452E+06	3.677E+09	9.823E+06	3.093E+06
2.50e+00	to	3.00e+00	3.804E+03	3.753E+08	8.393E+03	2.642E+03
3.00e+00	to	4.00e+00	1.109E-05	6.447E+07	5.677E-05	9.137E-06
4.00e+00	to	5.00e+00	1.215E-33	2.176E+07	6.079E-34	0.0
5.00e+00	to	6.50e+00	3.502E-34	8.734E+06	1.752E-34	0.0
6.50e+00	to	8.00e+00	4.454E-35	1.713E+06	2.228E-35	0.0
8.00e+00	to	1.00e+01	5.944E-36	3.638E+05	2.973E-36	0.0
Total Gamma, g/(sec*FA)			8.655E+11	2.690E+15	1.922E+12	6.015E+11
(1)Total Neutrons, n/(sec*FA)			6.324E+08 (raw) 9.720E+08 (adjusted intact) 1.014E+09 (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 burnup profile in active fuel region, where bpf=1.152, k_{eff}=0.2505 if fuel is intact and k_{eff}=0.2812 if fuel is reconfigured. k_{eff} for reconfigured fuel includes an additional 12.25% increase to account for fuel compression.*

Table A.5-39
Bounding NCT and HAC Source for the 69BTH DSC, Reconfigured Fuel, 45 Inner Compartments

62 GWd/MTU, 2.6 wt. %, after 18.5 years of cooling						
E_{min}, *MeV*	*to*	E_{max}, *MeV*	*Bottom Nozzle*	*In-core*	*Plenum*	*Top Nozzle*
0.00e+00	to	5.00e-02	1.916E+10	3.783E+14	8.447E+09	1.456E+10
5.00e-02	to	1.00e-01	3.637E+09	1.013E+14	1.130E+09	2.768E+09
1.00e-01	to	2.00e-01	9.263E+08	6.952E+13	5.678E+08	7.024E+08
2.00e-01	to	3.00e-01	4.731E+07	2.081E+13	3.469E+07	3.586E+07
3.00e-01	to	4.00e-01	6.872E+07	1.350E+13	8.793E+07	5.186E+07
4.00e-01	to	6.00e-01	2.571E+08	1.724E+13	1.586E+09	1.862E+08
6.00e-01	to	8.00e-01	1.480E+08	7.808E+14	8.274E+08	1.599E+08
8.00e-01	to	1.00e+00	6.221E+07	1.109E+13	1.490E+07	9.737E+07
1.00e+00	to	1.33e+00	1.059E+12	2.067E+13	3.239E+11	8.068E+11
1.33e+00	to	1.66e+00	2.992E+11	3.359E+12	9.148E+10	2.278E+11
1.66e+00	to	2.00e+00	8.137E+00	3.481E+10	5.299E+01	6.100E+00
2.00e+00	to	2.50e+00	7.158E+06	1.939E+09	2.189E+06	5.452E+06
2.50e+00	to	3.00e+00	6.116E+03	1.566E+08	1.870E+03	4.658E+03
3.00e+00	to	4.00e+00	1.374E-06	5.103E+07	1.198E-11	1.119E-05
4.00e+00	to	5.00e+00	0.0	1.697E+07	2.314E-33	0.0
5.00e+00	to	6.50e+00	0.0	6.811E+06	6.668E-34	0.0
6.50e+00	to	8.00e+00	0.0	1.336E+06	8.481E-35	0.0
8.00e+00	to	1.00e+01	0.0	2.837E+05	1.132E-35	0.0
Total Gamma, g/(sec*FA)			1.383E+12	1.417E+15	4.281E+11	1.053E+12
(1)Total Neutrons, n/(sec*FA)			4.958E+08 (raw) 7.201E+08 (adjusted intact) 7.380E+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 to account for fuel compression.*

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

62 GWd/MTU, 2.6 wt. %, after 35.0 years of cooling						
E_{min}, MeV	*to*	*E_{max}, MeV*	*Bottom Nozzle*	*In-core*	*Plenum*	*Top Nozzle*
0.00e+00	*to*	*5.00e-02*	*2.534E+09*	*2.507E+14*	*8.099E+08*	*1.894E+09*
5.00e-02	*to*	*1.00e-01*	*4.166E+08*	*7.103E+13*	*1.278E+08*	*3.170E+08*
1.00e-01	*to*	*2.00e-01*	*1.011E+08*	*4.284E+13*	*3.525E+07*	*7.712E+07*
2.00e-01	*to*	*3.00e-01*	*5.433E+06*	*1.328E+13*	*1.956E+06*	*4.110E+06*
3.00e-01	*to*	*4.00e-01*	*7.162E+06*	*8.920E+12*	*3.183E+06*	*5.389E+06*
4.00e-01	*to*	*6.00e-01*	*5.200E+06*	*6.427E+12*	*2.437E+07*	*3.690E+06*
6.00e-01	*to*	*8.00e-01*	*1.708E+07*	*5.261E+14*	*1.277E+07*	*6.506E+07*
8.00e-01	*to*	*1.00e+00*	*1.934E+07*	*2.893E+12*	*1.730E+06*	*6.510E+07*
1.00e+00	*to*	*1.33e+00*	*1.209E+11*	*4.682E+12*	*3.697E+10*	*9.208E+10*
1.33e+00	*to*	*1.66e+00*	*3.414E+10*	*5.966E+11*	*1.044E+10*	*2.600E+10*
1.66e+00	*to*	*2.00e+00*	*5.420E+00*	*2.270E+10*	*3.529E+01*	*4.064E+00*
2.00e+00	*to*	*2.50e+00*	*8.169E+05*	*1.206E+09*	*2.498E+05*	*6.222E+05*
2.50e+00	*to*	*3.00e+00*	*6.980E+02*	*1.115E+08*	*2.134E+02*	*5.316E+02*
3.00e+00	*to*	*4.00e+00*	*9.708E-07*	*2.735E+07*	*8.459E-12*	*7.901E-06*
4.00e+00	*to*	*5.00e+00*	*0.0*	*9.232E+06*	*2.314E-33*	*0.0*
5.00e+00	*to*	*6.50e+00*	*0.0*	*3.705E+06*	*6.668E-34*	*0.0*
6.50e+00	*to*	*8.00e+00*	*0.0*	*7.268E+05*	*8.481E-35*	*0.0*
8.00e+00	*to*	*1.00e+01*	*0.0*	*1.543E+05*	*1.132E-35*	*0.0*
*Total Gamma, g/(sec*FA)*			*1.581E+11*	*9.275E+14*	*4.843E+10*	*1.205E+11*
*(1)Total Neutrons, n/(sec*FA)*			*2.699E+08 (raw)* *3.920E+08 (adjusted intact)* *4.018E+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 to account for fuel compression.*

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.*

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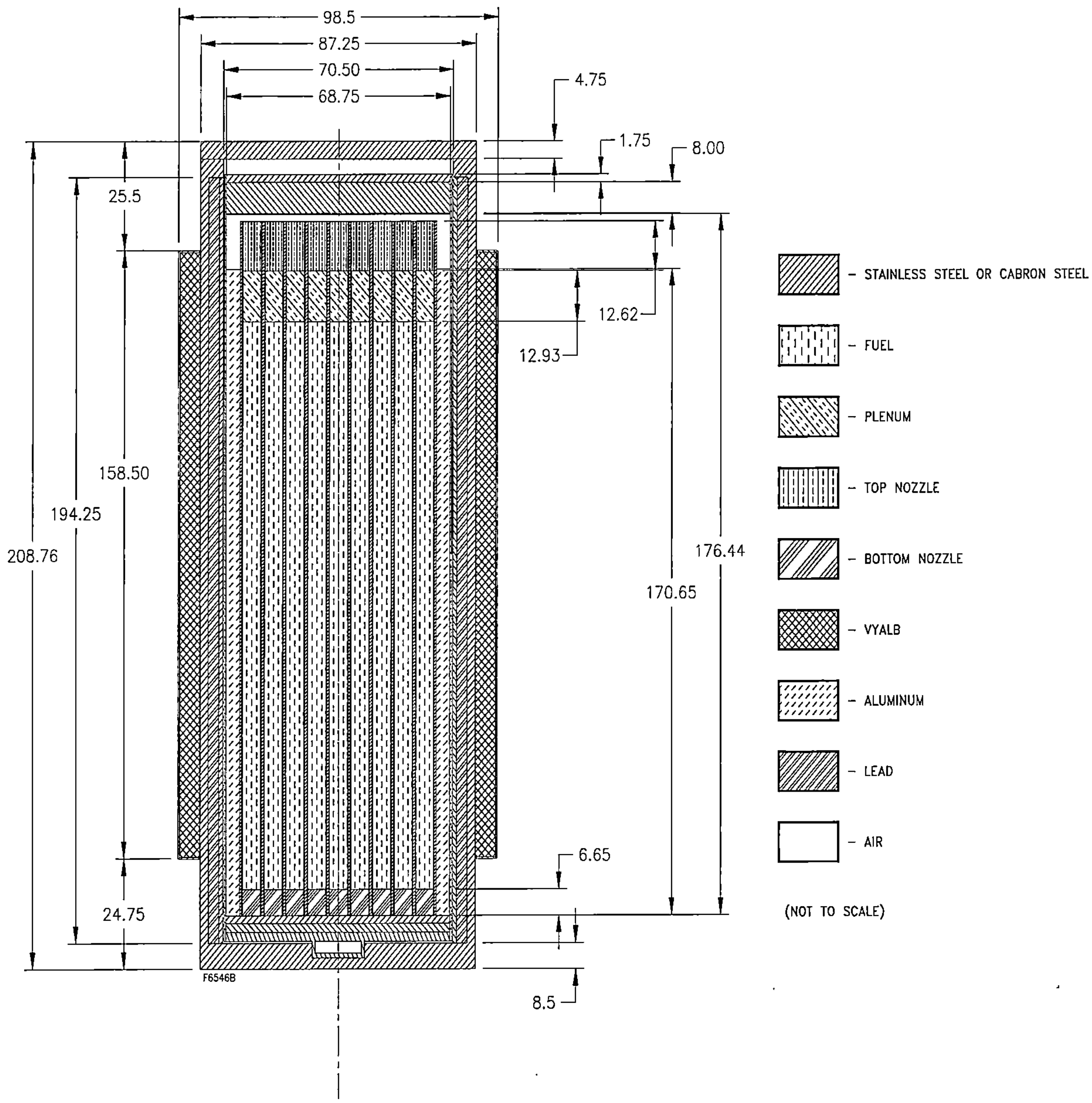

Figure A.5-1
MP197HB Transport Cask with 69BTH DSC Model, Axial View

Note: All dimensions are in inches. Impact Limiters are not shown.

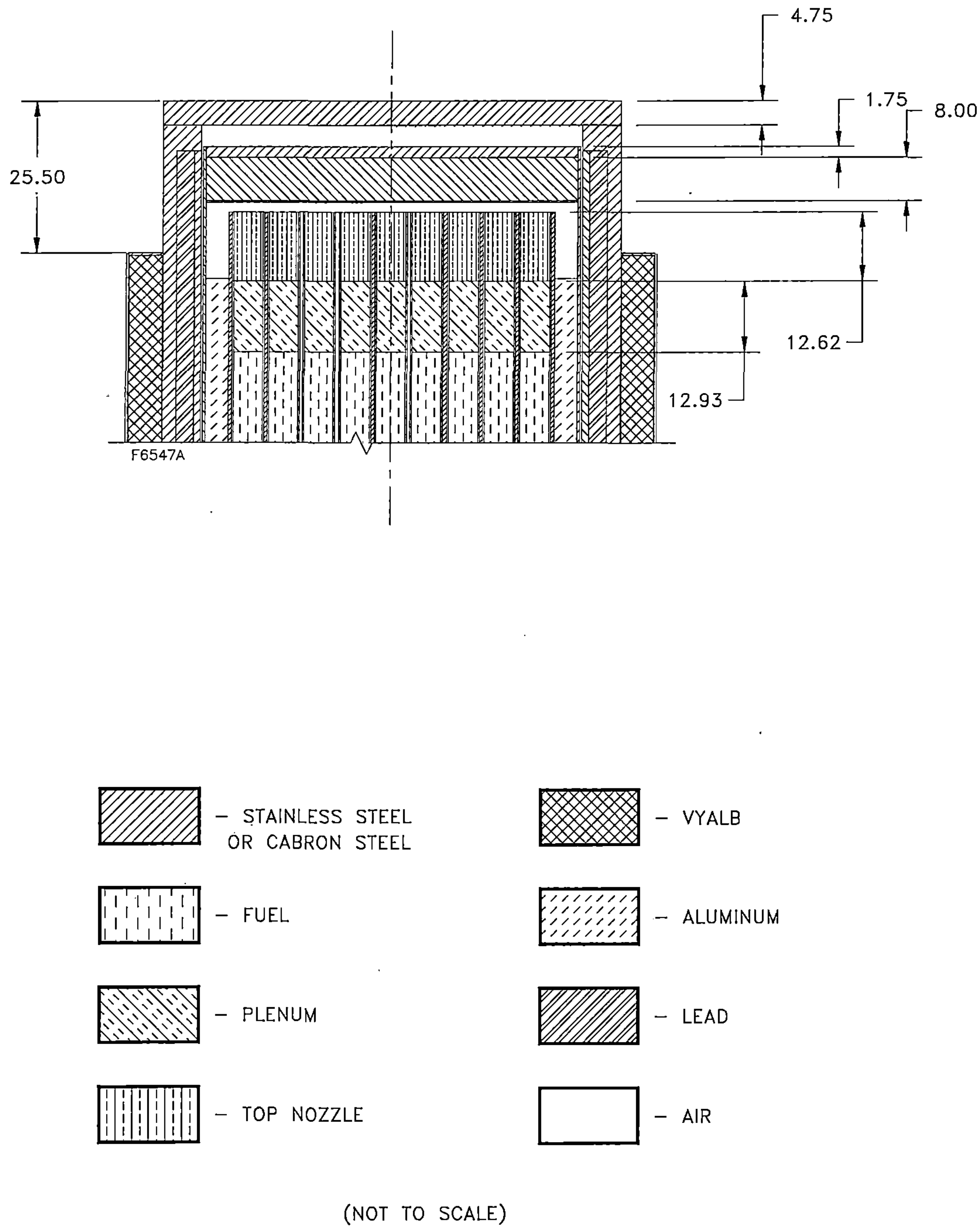

Figure A.5-2
MP197HB Transport Cask with 69BTH DSC Model, Top View Showing Cask Lid with Gap, Top Nozzle, and Plenum

Note: All dimensions are in inches. Top Impact Limiter is not shown.

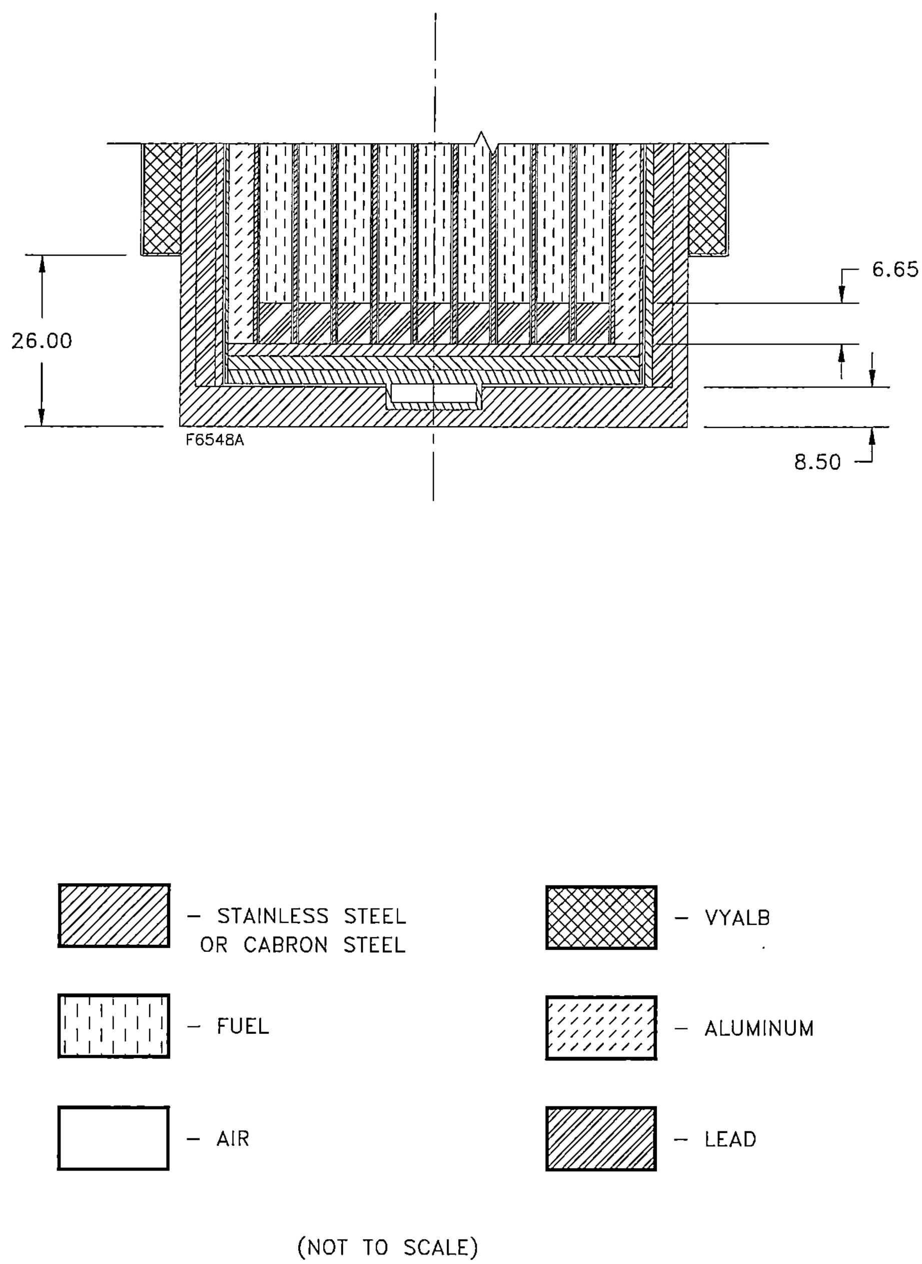

Figure A.5-3
MP197HB Transport Cask with 69BTH DSC Model, Bottom View Showing Cask Bottom and Bottom Nozzle

Note: All dimensions are in inches. Bottom Impact Limiter is not shown.

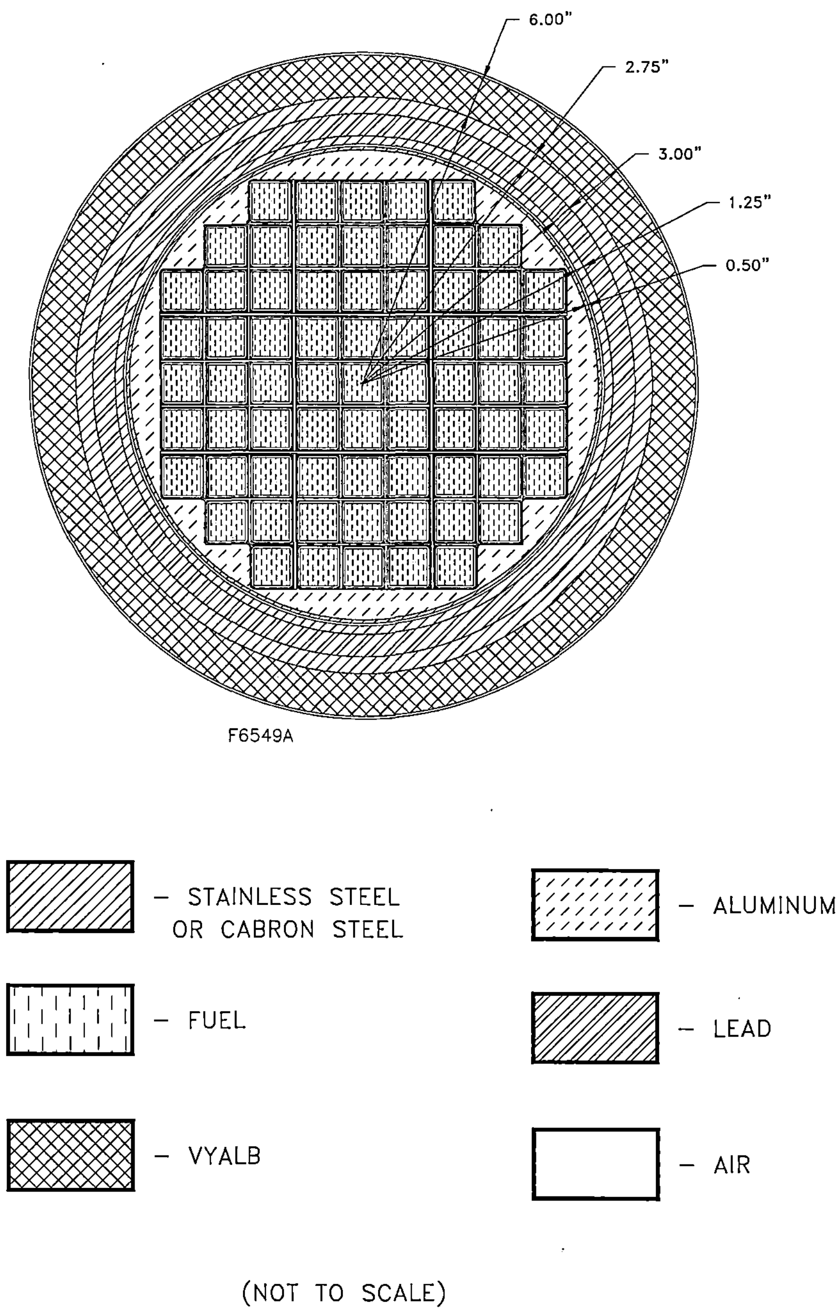

Figure A.5-4
MP197HB within 69BTH DSC Model, Cross Section View

Note: All dimensions are in inches

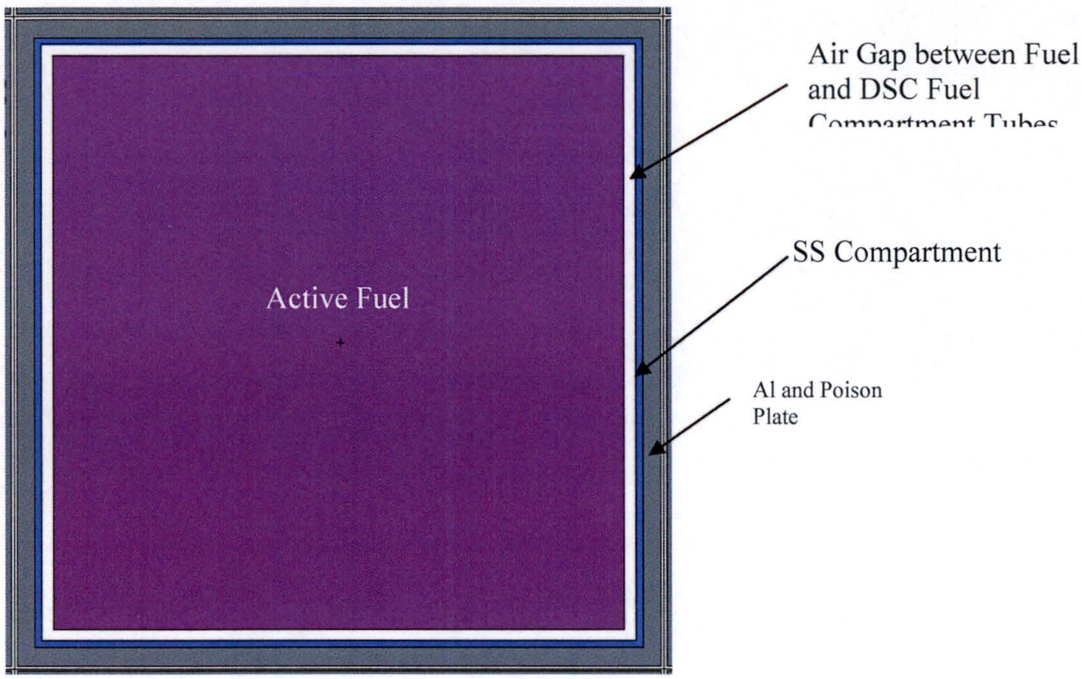

Figure A.5-5
Details of DSC Basket with Fuel Lattice Unit Cell in MCNP Model

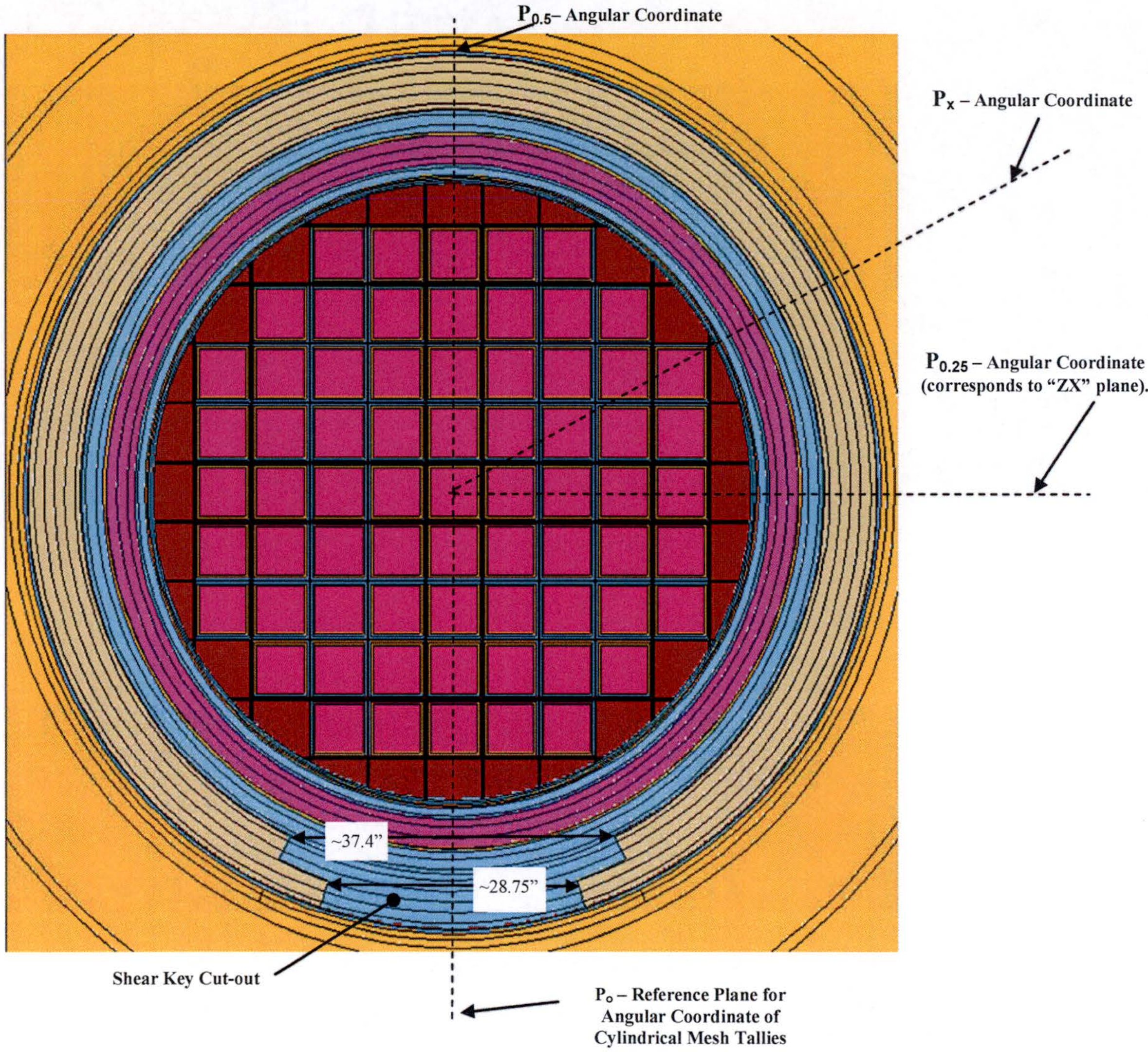

Figure A.5-6
MP197HB Transport Cask within 69 BTH DSC: MCNP Model Cut-through XY Plane (Z=25.12 cm), Normal Condition

Note: All dimensions are in inches

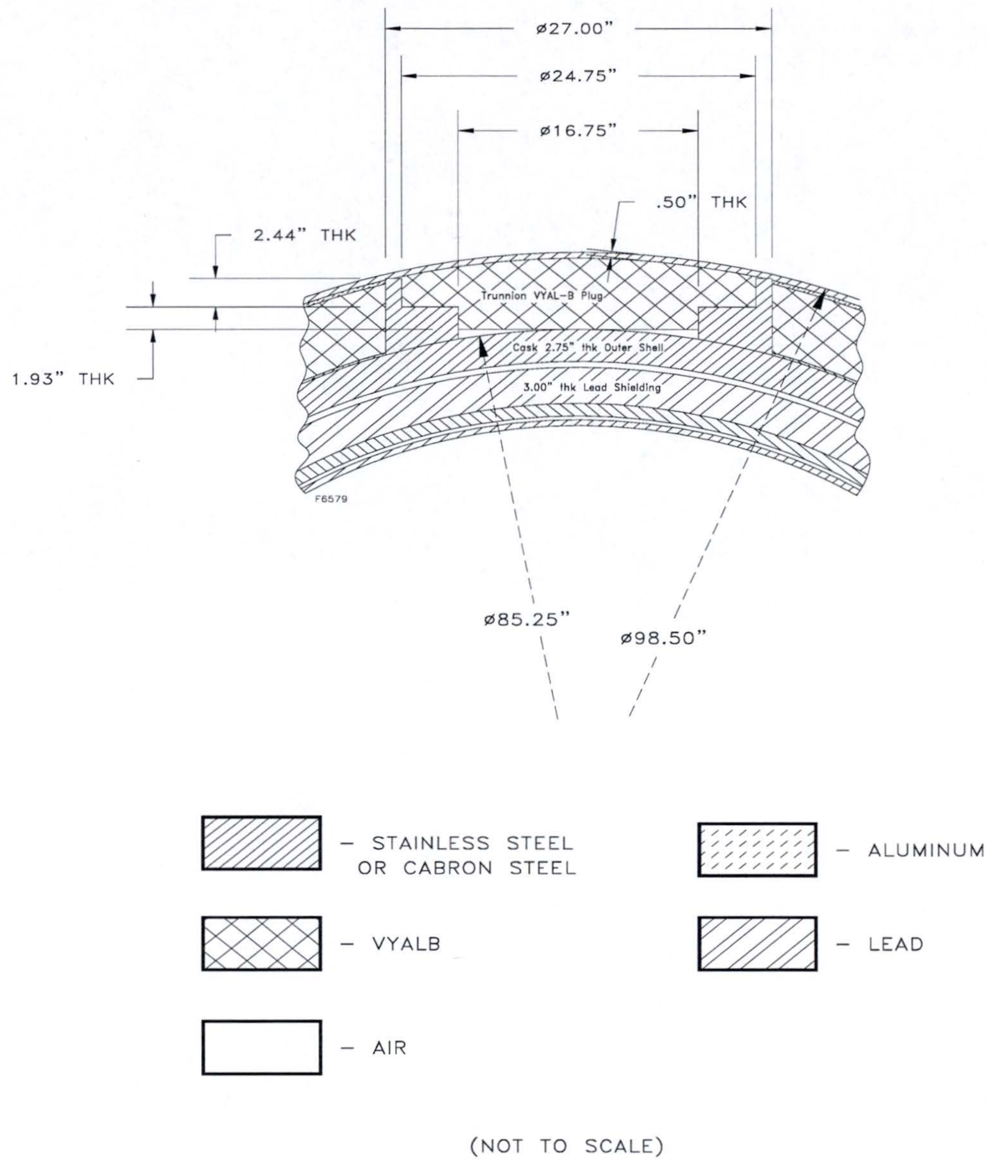

Figure A.5-7
MP197HB Bottom and Top Trunnion Attachment Block and Plug Geometry, Normal Condition

Note: All dimensions are in inches

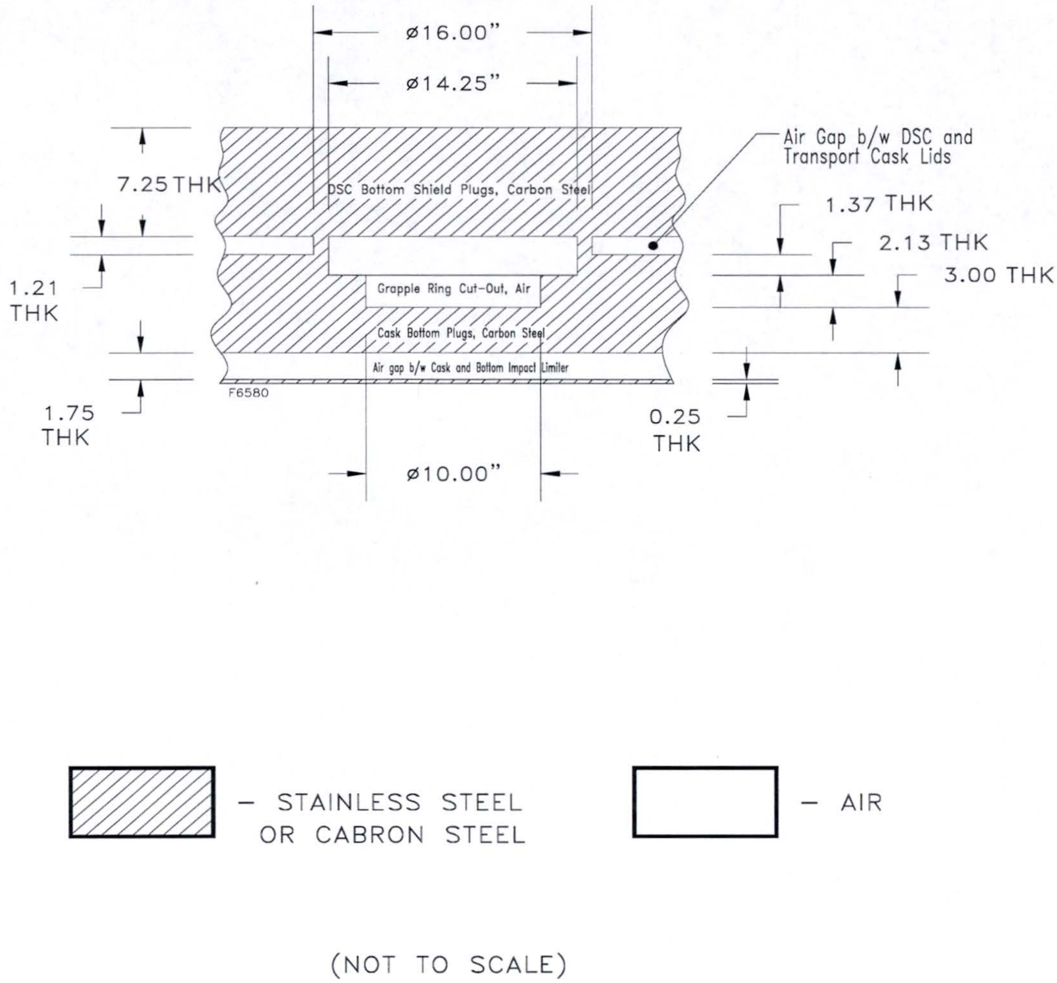

Figure A.5-8
69BTH DSC and MP197HB Bottom Plugs with Grapple Ring Cut-Out

Note: All dimensions are in inches

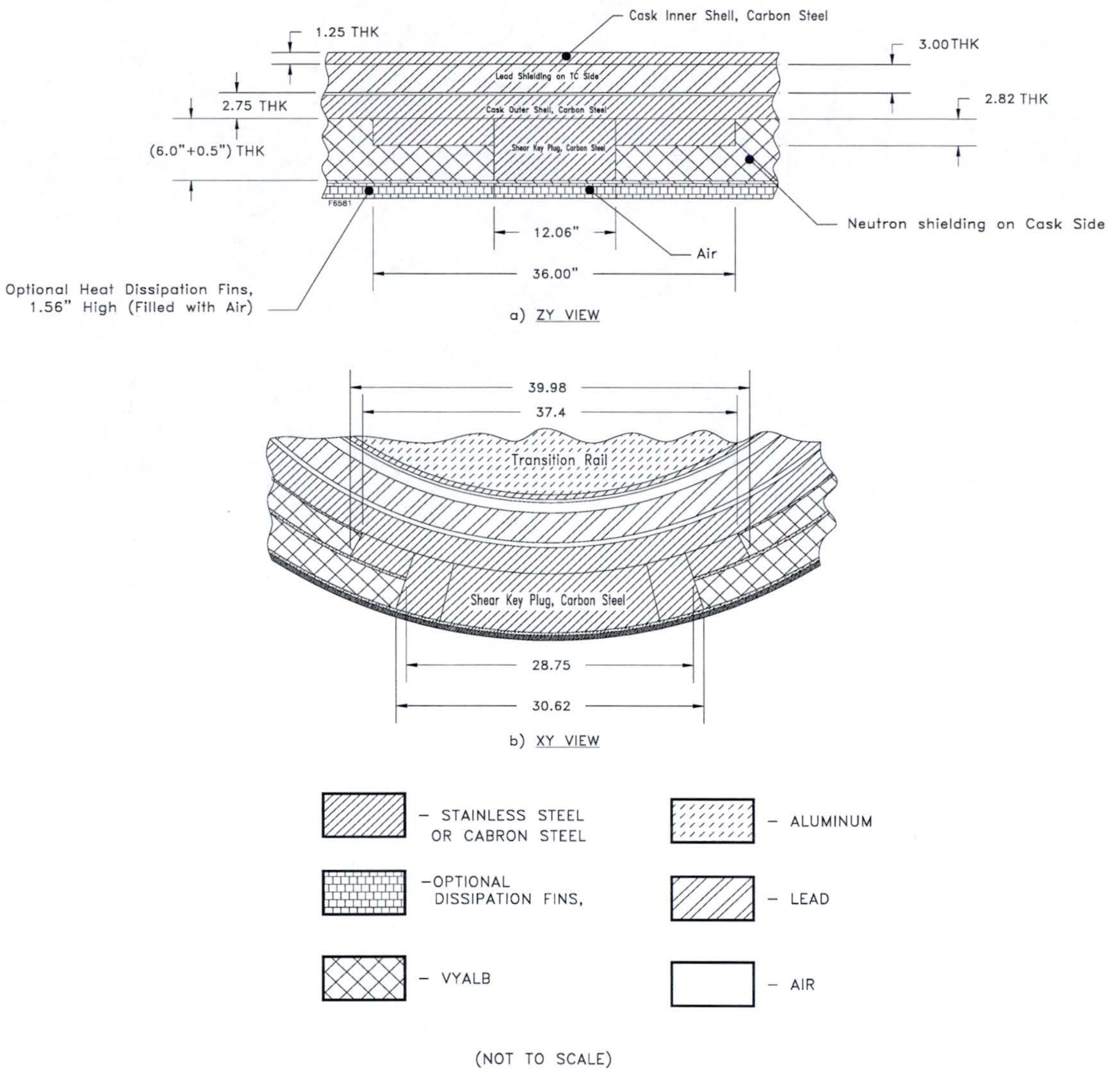

Figure A.5-9
MP197HB Shear Key

Note: 1) All dimensions are in inches
2) Two 0.125"aluminum shells from aluminum boxes encasing neutron shielding material, 0.375"cask outer shield shell as well as 0.375" air shell between lead shielding on cask side and cask outer shell are not seen.

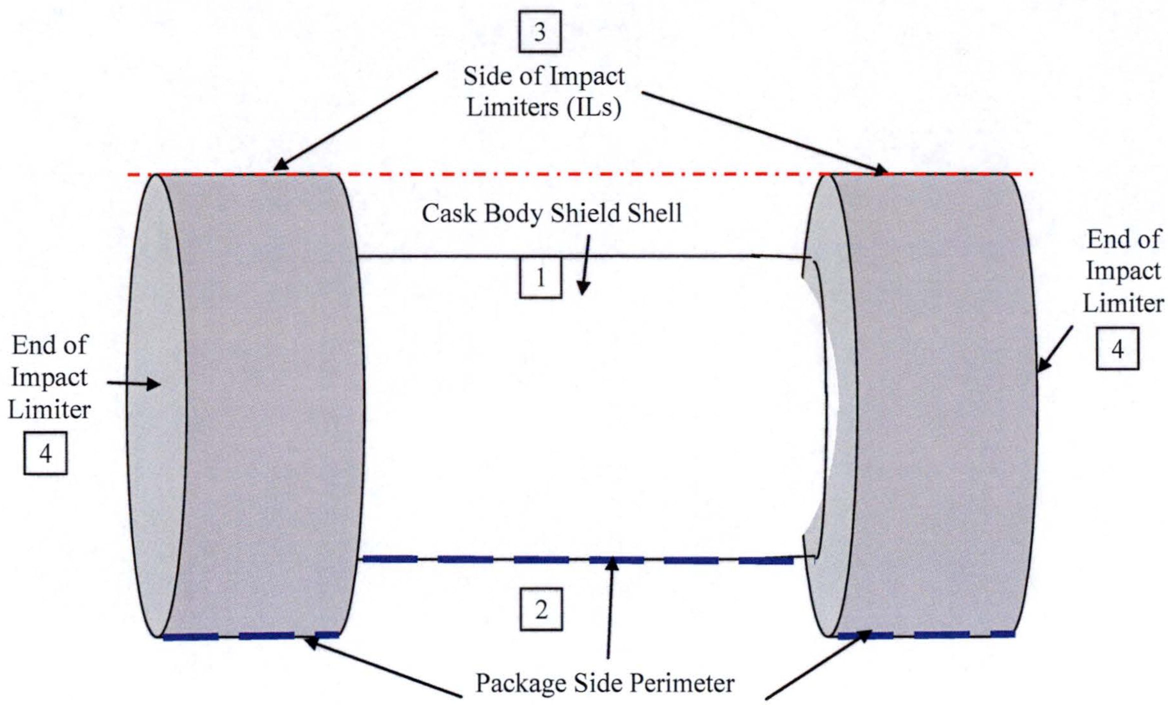

Figure A.5-10
Dose Rate Location Terminology

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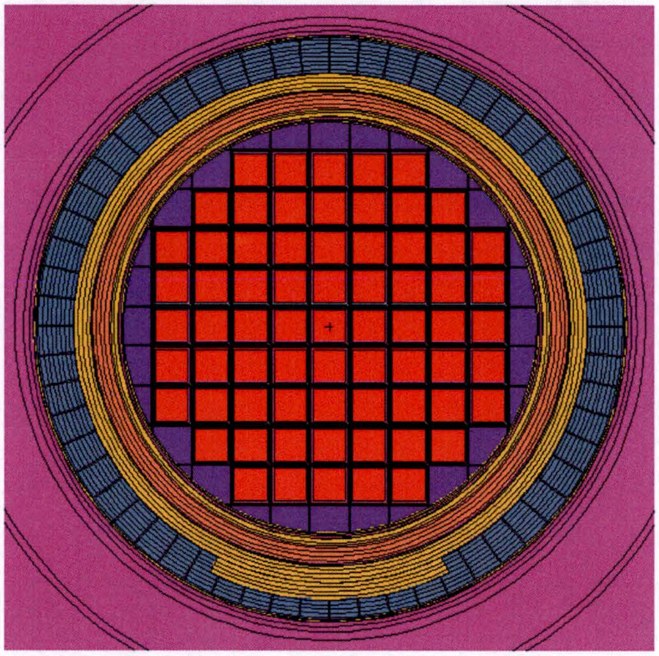
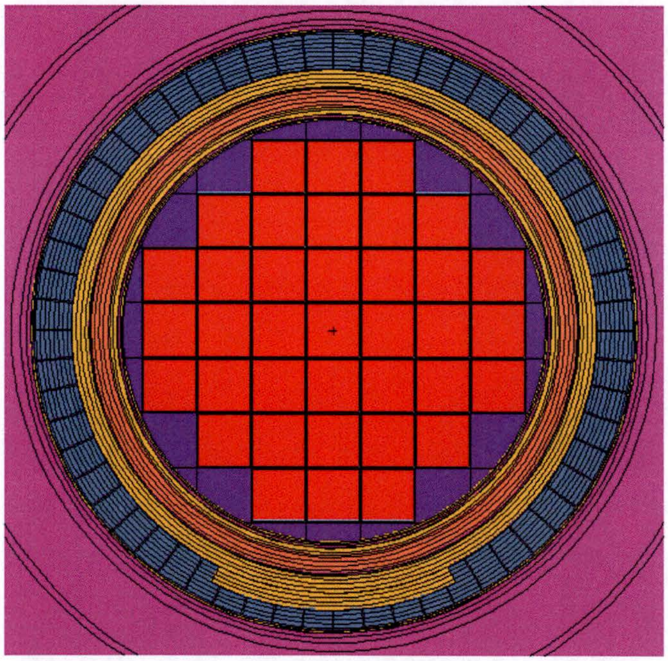


69BTH DSC



37PTH DSC

Figure A.5-23
NCT MCNP Models (x-y View)

Intact Fuel Reconfigured Fuel

Figure A.5-24
NCT 69BTH DSC MCNP Models (x-z View)

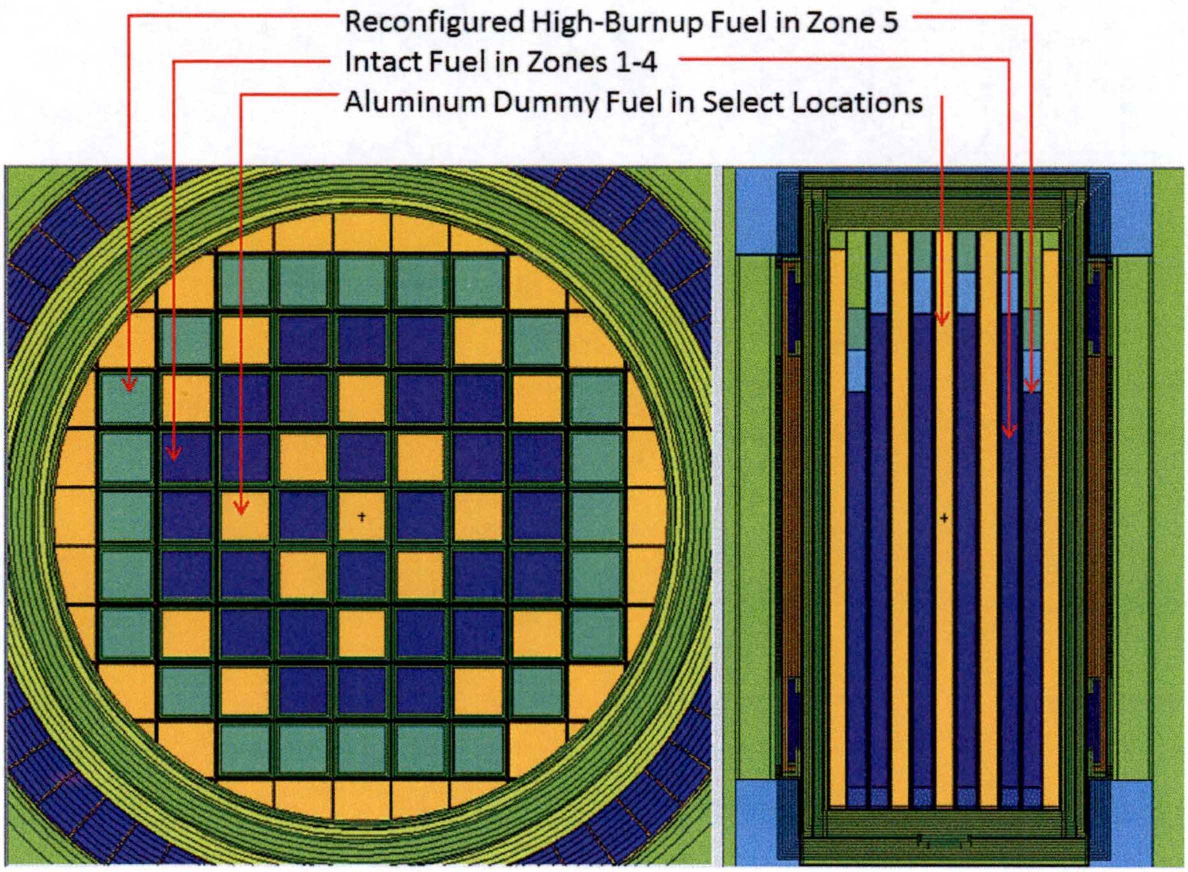
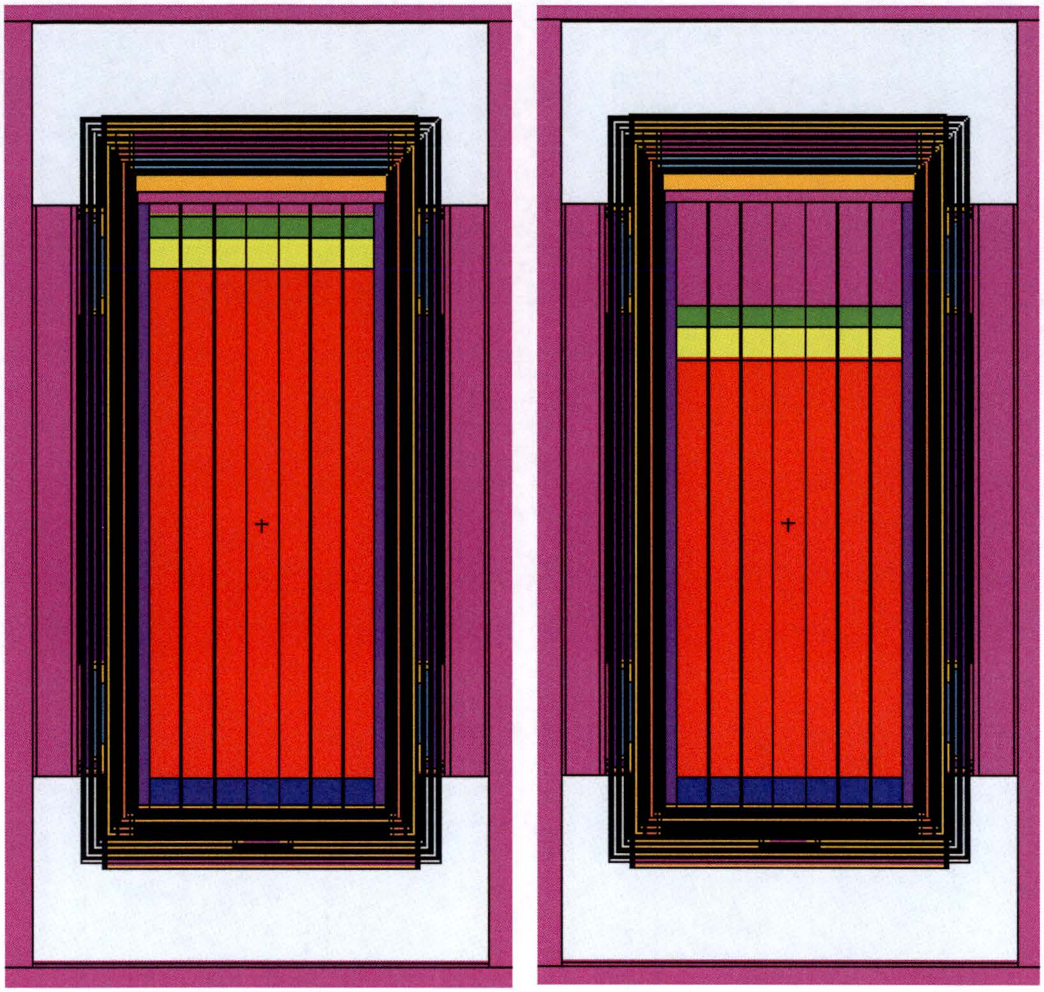


Figure A.5-24a
NCT 69BTH DSC MCNP Response Function (RF) Model Case for Heat Load Zone Configuration No. 8 with High-Burnup Reconfigured Fuel in Zone 5
(x-y and x-z views)



Intact Fuel Reconfigured Fuel

Figure A.5-25
NCT 37PTH DSC MCNP Models (x-z View)

69BTH DSC 37PTH DSC

> Note: All neutron shielding VYAL-B material, the aluminum boxes, the neutron shield steel skin, impact limiter wood, and impact limiter steel shell are modeled as air.

Figure A.5-26
HAC MCNP Models for Reconfigured Fuel (x-z View)

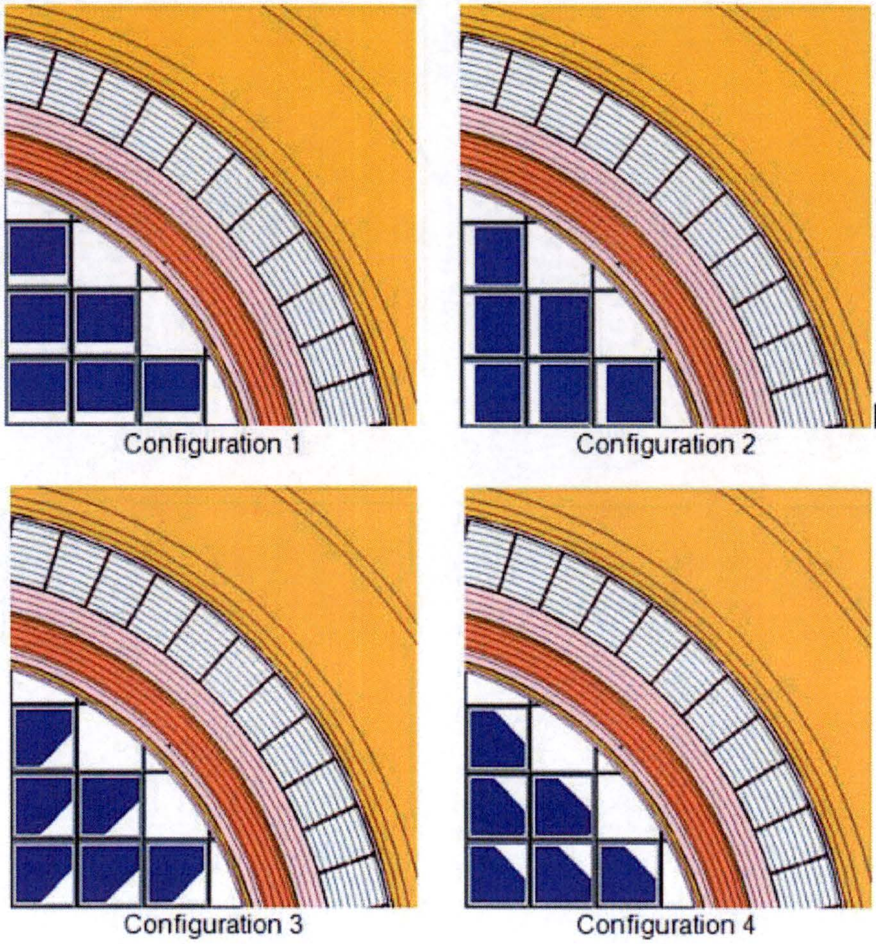


Figure A.5-27
NCT Configurations for *Radial* Fuel Reconfiguration (x-y View)

		P	*P*		
	P	*I*	*I*	*P*	
P	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*P*
	P	*I*	*I*	*P*	
		P	*P*		

Figure A.5-28
Peripheral and Inner Fuel Locations for the 24PT4 and 24PTH DSCs

	P	*P*	*P*	*P*	
P	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*P*
	P	*P*	*P*	*P*	

Figure A.5-29
Peripheral and Inner Fuel Locations for the 32PT, 32PTH, and 32PTH1 DSCs

		P	*P*	*P*		
	P	*I*	*I*	*I*	*P*	
P	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*P*
	P	*I*	*I*	*I*	*P*	
		P	*P*	*P*		

Figure A.5-29a
Peripheral and Inner Fuel Locations for the 37PTH DSC

			P	*P*	*P*			
	P	*P*	*I*	*I*	*I*	*P*	*P*	
	P	*I*	*I*	*I*	*I*	*I*	*P*	
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
	P	*I*	*I*	*I*	*I*	*I*	*P*	
	P	*P*	*I*	*I*	*I*	*P*	*P*	
			P	*P*	*P*			

Figure A.5-29b
Peripheral and Inner Fuel Locations for the 61BT and 61BTH DSCs

		P	*P*	*P*	*P*	*P*		
	P	*I*	*I*	*I*	*I*	*I*	*P*	
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
P	*I*	*I*	*I*	*I*	*I*	*I*	*I*	*P*
	P	*I*	*I*	*I*	*I*	*I*	*P*	
		P	*P*	*P*	*P*	*P*		

Figure A.5-29c
Peripheral and Inner Fuel Locations for the 69BTH DSC

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Proprietary Information Withheld Pursuant to 10 CFR 2.390

Figure A.5-38
Correction Factors vs. Burnup for Each Reactor–Cs-137

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Figure A.5-39
Cooling Time Comparison–Trending Evaluation

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Chapter A.7
Package Operations

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LIST OF FIGURES

Chapter A.7
Package Operations

NOTE: References in this chapter are shown as [1], [2], etc., and refer to the reference list in Section A.7.5. A glossary of terms used in this chapter is provided in Section A.7.6.

This chapter contains NUHOMS®-MP197HB cask loading and unloading procedures that are intended to show the general approach to cask operational activities. The procedures in this chapter are intended to show the types of operations that will be performed and are not intended to be limiting. Site specific conditions and requirements may require the use of different equipment and ordering of steps to accomplish the same objectives or acceptance criteria which must be met to ensure the integrity of the package.

A separate operations manual (OM) will be prepared for the NUHOMS®-MP197HB cask to describe the operational steps in greater detail. The OM, along with the information in this chapter, will be used to prepare the site-specific procedures that will address the particular operational considerations related to the cask.

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A.7.1 NUHOMS®-MP197HB Package Loading

The use of the NUHOMS®-MP197HB cask to transport fuel offsite involves (1) preparation of the cask for use, (2) verification that the fuel assemblies loaded in the dry shielded canister (DSC) meet the criteria set forth in this document, and (3) installation of a DSC into the cask. Also included herein are procedures to prepare and load fuel in an empty DSC contained in a NUHOMS®-MP197HB cask and to close the DSC.

The use of the NUHOMS®-MP197HB cask to transport dry irradiated and/or contaminated non-fuel bearing solid materials in radioactive waste canisters (RWCs) involves (1) preparation of the cask for use, (2) verification that the waste to be loaded meet the criteria set forth in this document, and (3) loading of the RWC and waste into the cask.

Offsite transport involves (1) preparation of the cask for transport, (2) assembly verification leakage-rate testing of the packaging containment boundary, (3) placement of the cask onto a transportation vehicle, and (4) installation of the impact limiters.

During shipment, the packaging contains any one of the DSCs as described in Chapter A.1, Appendices A.1.4.1 through A.1.4.9 or an RWC as described in Appendix A.1.4.9A. Type and

form of the content and their maximum quantity to be loaded in any of the nine DSCs are specified in Table A.7-2a. Type and form of the content and their maximum quantity to be loaded in an RWC are specified in Table A.7-2b. Procedures are provided in this section for (1) transport of the cask/DSC/RWC directly from the plant spent fuel pool and (2) transport of a DSC/RWC which was previously stored in a NUHOMS® horizontal storage module (HSM). Section A.7.7 contains an appendix for each DSC model detailing its loading procedures. Table A.7-3 lists these appendices.

A.7.1.1 NUHOMS®-MP197HB Cask Preparation for Loading

Procedures for preparing the cask for use after receipt at the loading site are provided in this section and are applicable for shipment of DSCs loaded with fuel or of RWCs loaded with dry irradiated and/or contaminated non-fuel bearing solid materials.

1. Remove the impact limiters from the cask.
2. Prior to removing the lid, sample the cask cavity atmosphere.
3. Remove the transportation skid personnel barrier and tie down assembly.
4. Take contamination smears on the outside surfaces of the cask. If necessary, decontaminate the cask.
5. O-ring seals (metallic or elastomer) shall be discarded after each use.
6. Install the front and rear trunnions, if required. Install the trunnion bolts and torque them to 1000-1100 ft-lbs following the torquing sequence shown in Figure A.7-1.
7. Lift the cask and place it on the onsite transfer trailer or upending frame, or lift the cask/transport skid and place them in the appropriate location.
8. NOT USED.
9. NOT USED.
10. If transporting any of the smaller diameter DSC models (NUHOMS®-24PT4, 32PT, 24PTH, 61BT, or 61BTH) or an RWC, verify that the MP197HB cask has been fitted with an internal aluminum sleeve (Refer to Drawing MP197HB-71-1014 provided in Chapter A.1, Appendix A.1.4.10.1). This step, if required, can be performed at any time prior to placing the DSC or RWC in the cask.
11. If transporting a NUHOMS®-69BTH DSC with heat load greater than 26 kW, verify that the removable external aluminum fins are available to be fitted to the cask after the cask is closed (Refer to Drawing MP197HB-71-1011 provided in Appendix A.1.4.10.1). Note that fins are not required to meet the 10 CFR 71 requirements and are optional.
12. For a specific DSC model to be loaded inside the MP197HB cask, verify the canister/basket type (A, B, C, D, *E, or F* as applicable) is appropriate for the fuel to be transported.
13. The candidate intact, damaged and failed fuel assemblies to be transported in a specific DSC model must be evaluated (by plant records or other means) to verify that they meet the criteria of the applicable fuel specification as listed in Table A.7-2*a*.
14. For the transportation of fuel within the NUHOMS®-32PT, 24PTH, 32PTH, 32PTH1, or 37PTH DSCs where burnup credit is employed for criticality safety, additional administrative controls to prevent misloading are also outlined in the applicable appendices of this chapter.

A.7.1.2 NUHOMS®-MP197HB Cask Wet Loading

NOTE: The wet loading procedure described in this section is applicable only when using the MP197HB cask for loading fuel from a spent fuel pool into any one of the DSCs listed in Chapter A.1 or for loading irradiated waste into a RWC. This section also provides steps for closure of the DSC/RWC.

Site specific conditions and requirements may require the use of different equipment and ordering of steps than those described below to accomplish the same objectives or acceptance criteria which must be met to ensure the integrity of the package.

1. Prior to being placed in service, the cask is to be cleaned or decontaminated as necessary.
2. NOT USED.
3. Remove the ram access closure plate, inspect the sealing surfaces, replace the old seals with new seals, lubricate and re-install the ram access closure plate.
4. NOT USED.
5. Engage the cask front trunnions with the lifting yoke using the plant crane, rotate the cask to a vertical orientation, lift the cask from the onsite transfer skid, and place the cask in the plant designated preparation area.
6. Install the shear key plug assembly.
7. If the cask lid has not already been removed, remove the bolts from the cask lid and lift the lid from the cask.
8. Discard the used lid O-rings.
9. NOT USED.
10. If loading any one of the smaller diameter DSC models (NUHOMS®-24PT4, 32PT, 24PTH, 61BT, or 61BTH) or RWCs from the MP197HB cask, install an unloading flange.. Depending on the DSC model being loaded, verify that a DSC bottom spacer of appropriate height is placed at the bottom of the cask. The height of the DSC bottom spacer required for each type of DSC is listed in Table A.7-1.
11. Place an empty DSC in the cask.
12. If damaged fuel is to be loaded in the DSC, place bottom end caps into the cell locations that are to receive damaged fuel. For the NUHOMS®-24PT4 DSC only, verify that the failed fuel cans, required for loading damaged fuel assemblies if used, have replaced the guide sleeves at the locations specified for the specific configurations of the 24PT4 DSC basket.
13. If failed fuel is to be loaded in the DSC (24PTH or 61BTH DSCs only), put the appropriate empty failed fuel cans in the appropriate locations.

13a. If fuel and basket spacers are required, the height of the fuel and basket spacers required for each type of DSC is listed in Table A.7-1.

14. Fill the cask/DSC annulus with water. Install the annulus seal.
15. Fill the DSC cavity with water. For the NUHOMS®-32PT, 24PTH, 32PTH, 32PTH1, and 37PTH DSCs, a minimum soluble boron concentration is required during loading and unloading operations.

A.7.1.2.1 DSC/RWC Wet Loading

The procedures for loading, vacuum drying, and sealing the DSC/RWC are described in detail in Appendices A.7.7.1 through A.7.7.10 as listed in Table A.7-3.

Following the completion of the wet loading activities described in a specific appendix listed in Table A.7-3, the MP197HB cask is prepared for downending as described in the next section.

A.7.1.2.2 Preparing the NUHOMS®-MP197HB Cask for Downending

1. Discard and install new drain port seals.
2. If transporting any one of the smaller diameter DSC models (NUHOMS®-24PT4, 32PT, 24PTH, 61BT, or 61BTH) or RWC, place a cask spacer ring at the top of the aluminum sleeve as shown in Drawing MP197HB-71-1014, Chapter A.1, Appendix A.1.4.10.1.
3. Verify that the lid O-ring seals are new.
4. NOT USED.
5. Install shims, if required.

5a. Install the DSC top spacer if required. The appropriate height of the DSC top spacer required for each type of DSC is listed in Table A.7-1.

6. Install the cask lid. Follow the torquing sequence shown in Figure A.7-1. Torque to between 950 and 1040 ft-lbs.
7. Install new cask vent port seals.
8. Install new cask test port seals.
9. Evacuate the cavity between the cask and the DSC and backfill with helium.
10. Perform the assembly verification leakage test following the procedure given in Section A.7.4.1.

A.7.1.2.3 NUHOMS®-MP197HB Cask Downending

NOTE: Alternate procedures may be developed for plants with unique requirements.

1. Remove the shear key plug assembly from the cask.
2. Lift the cask over the onsite transfer skid on the transfer trailer.
3. NOT USED.
4. Position the cask rear trunnions onto the onsite transfer skid pillow blocks.
5. Downend the cask and secure it to the skid.
6. NOT USED.
7. NOT USED.
8. NOT USED.
9. Prepare the cask for transportation in accordance with the procedure described in Section A.7.1.4.

A.7.1.3 NUHOMS®-MP197HB Cask Dry Loading (Transferring a Loaded DSC or RWC from an Overpack into an MP197HB Cask)

A number of NUHOMS® DSCs are currently being used for onsite storage of spent fuel inside the NUHOMS® horizontal storage modules (HSMs) or the advanced horizontal storage modules (AHSMs) under the provisions of 10 CFR 72.

This section summarizes the steps for transferring a previously loaded DSC under a 10 CFR *Part* 72 license from the HSM or AHSM (generally referred here as HSM) to the MP197HB cask for transportation. Depending on the most recent use of the cask, several of the initial steps listed below may not be necessary.

An RWC may be stored in an HSM, AHSM or other allowed overpack on the plant site. When the MP197HB cask is dry loaded with an RWC, operational steps similar to dry loading a DSC from an HSM into the MP197HB cask should be used depending on the storage overpack.

CAUTION:

Before initiating any steps described in this section:

- The licensee shall perform an audit of spent fuel pool records from the time of canister loading for the identification of the loaded fuel assemblies, and
- The licensee shall compare the irradiation parameters of the loaded contents against those shown in Table A.6-17 to ensure compliance with the isotopic depletion analysis.

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Page A.7-5a through A.7-5c are proprietary information withheld pursuant to 10 CFR 2.390.

3. If loading any one of the smaller diameter DSC models (NUHOMS®-24PT4, 32PT, 24PTH, 61BT, or 61BTH) in the NUHOMS®-MP197HB cask, install an unloading flange. Verify that a DSC bottom spacer of appropriate height is placed at the bottom of the cask. The height of the DSC bottom spacer required for each type of DSC is listed in Table A.7-1.
4. Remove the ram access closure plate and the lid.
5. NOT USED.
6. Bring the onsite transfer trailer and the NUHOMS®-MP197HB cask to the ISFSI site.
7. Remove the HSM door and the DSC seismic restraint assembly from the HSM.
8. NOT USED.
9. Align and dock the cask with the HSM.
10. Install the cask/HSM restraints.
11. Align the hydraulic ram cylinder in the ram trunnion support assembly.
12. Extend the ram hydraulic cylinder and engage the grapple ring.
13. NOT USED.
14. Retract the ram hydraulic cylinder until the DSC is fully retracted into the cask.
15. NOT USED.
16. Disengage the hydraulic ram from the grapple ring and remove from the cask ram access area.
17. Install the cask ram access closure plate (with new seals) following the torquing sequence shown in Figure A.7-1, and torque as specified in Drawing MP197HB-71-1002, Chapter A.1, Appendix A.1.4.10.1.
18. Remove the cask/HSM restraints.
19. Move the cask to the transfer position.
20. NOT USED.
21. NOT USED.
22. If transporting any one of the smaller diameter DSC models (NUHOMS®-24PT4, 32PT, 24PTH, 61BT, or 61BTH), place a cask spacer ring at the top of the aluminum sleeve as shown in Drawing MP197HB-71-1014, Chapter A.1, Appendix A.1.4.10.1.
23. Install shims, if required.

23a. Install the DSC top spacer as specified in Table A.7-1.

24. Install the cask lid following the torquing sequence shown in Figure A.7-1. Torque to between 950 and 1040 ft-lbs.
25. NOT USED.
26. Install new cask test port seals.
27. When required, perform helium leak detection testing of the cask cavity to verify the integrity of a loaded DSC in accordance with the procedure given in *Part 3 of* [16] TN E-33299, "Evaluation Procedure to Verify DSC Acceptance for Transport."

28. Remove the shear key plug assembly from the cask.
29. Perform the assembly verification leakage test following the procedure given in Section A.7.4.1.
30. Prepare the cask for transportation in accordance with the procedure described in Section A.7.1.4.

A.7.1.4 NUHOMS®-MP197HB Cask Preparation for Transport

Once the NUHOMS®-MP197HB cask has been loaded using either the wet loading procedure described in Section A.7.1.2 or the dry loading procedure described in Section A.7.1.3 above, the following tasks are performed to prepare the cask for transportation. The cask is assumed to be seated horizontally in the onsite transfer skid. Alternate procedures may be developed for plants with unique requirements.

1. Verify that the cask surface removable contamination levels meet the requirements of 49 CFR 173.443 [2] and 10 CFR 71.87 [3].
2. Verify that the assembly verification leakage rate testing specified in Section A.7.4.1 has been performed. This test must be performed within 12 months prior to the shipment.
3. If the packaging contains high burnup fuel assemblies, perform a Radiation Survey (both neutron and gamma) and a Thermal Survey of the cask loaded with the contents to evaluate the axial radiation and thermal source distributions prior to transportation.

   These surveys shall be performed using the same quality assurance requirements to comply with 10 CFR 71.43 and 10 CFR 71.47 [3] in accordance with Regulatory Guide 1.21 [19]. A record of the survey results showing the location of the survey points, type and model of the instruments shall be prepared for delivery to the packaging recipient. The location of the survey points should be labeled legibly on the cask outer surface for recipient use. An example showing survey points on a Type B cask for transportation of non-fuel bearing solid material is provided in Figure A.7-6.

A.7.1.4.1 Placing the NUHOMS®-MP197HB Cask onto the Conveyance

The procedure for placement of the cask on the conveyance is given in this section. If the cask is already on the transportation skid, rig the cask/skid, lift and place on the conveyance, then skip to Step 8.

1. Bring the cask and onsite transfer trailer to the conveyance.
2. NOT USED.
3. NOT USED.
4. NOT USED.
5. Place the cask onto the transportation skid.
6. Remove the cask upper and lower trunnions and install the trunnion plugs.
7. NOT USED.
8. If necessary, install the optional external aluminum fins.

9. Install the transportation skid tie-down straps.
10. Install the impact limiters on the cask and torque the attachment bolts in accordance with the drawings in Chapter A.1, Appendix A.1.4.10.1.
11. Remove the impact limiter hoist rings and replace them with hex bolts.
12. Install the cask tamperproof seals.
13. Install the transportation skid personnel barrier.
14. Perform a final radiation survey to assure the cask radiation levels do not exceed 49 CFR 173.441 [2] and 10 CFR 71.47 [3] requirements.
15. Verify that the temperature on all accessible surfaces is < 185°F.
16. Prepare the final shipping documentation and release the loaded cask for shipment.

A.7.2 NUHOMS®-MP197HB Package Unloading

Unloading the NUHOMS®-MP197HB cask after transport involves removing the cask from the conveyance and removing the DSC/RWC from the cask. The cask is designed to allow the DSC/RWC to be unloaded from the cask into a NUHOMS® staging module, hot cell or other suitable overpack, and provisions exist to allow wet unloading into a fuel pool. The necessary procedures for these tasks are essentially the reverse of those described in Section A.7.1.

A.7.2.1 Receipt of Loaded NUHOMS®-MP197HB Package from Carrier

Procedures for receiving the loaded cask after shipment are described in this section. Procedures for receiving an empty cask are provided in Section A.7.1.1.

1. Verify that the tamperproof seals are intact.
2. Remove the tamperproof seals.
3. Remove the hex bolts from the impact limiters and replace them with the impact limiter hoist rings provided.
4. Remove the impact limiters from the cask.
5. Remove the transportation skid personnel barrier and tie-down straps.
6. Remove the external aluminum fins, if present.
7. Take contamination smears on the outside surfaces of the cask. If necessary, decontaminate the cask.
8. Install the front and rear trunnions and torque the bolts to 1000-1100 ft-lb for double shoulder trunnions and 800-900 ft-lb for single shoulder trunnions following the *torqueing* sequence shown in Figure A.7-1.
9. If the packaging contains high burnup fuel assemblies, perform a Radiation Survey (both neutron and gamma) and a Thermal Survey of the cask loaded with the contents to evaluate the axial radiation and thermal source distributions. *These surveys shall be performed on the survey locations identified in Section A.7.1.4, Step 3 using the same quality assurance requirements. It is recommended to use the same type and model of the*

instruments that were used in the surveys prior to transportation. Compare the results of the above surveys to the results of the surveys performed prior to transportation required in Section A.7.1.4.

The shifts of the peak temperature or peak dose rates, particularly the neutron dose rate, toward the cask ends are the criteria to indicate that potentially a fuel reconfiguration occurred during transport.

The information obtained from these inspections can be employed *as indicators* to determine the *ALARA* requirements of the unloading operations.

10. Lift the cask from the conveyance. Place cask onto the onsite transfer trailer or other location.
11. Prior to unloading of high burnup fuel assemblies, perform the operations outlined in Table A.7-5.
12. Transfer the cask to a staging module, fuel pool, dry cell or storage overpack and unload using the procedures described in the following sections.

A.7.2.2 Removal of Contents from NUHOMS®-MP197HB Cask

For unloading of high burnup fuel assemblies, proceed with the following operations only after performing the operations outlined in Table A.7-5.

A.7.2.2.1 Unloading the NUHOMS®-MP197HB Cask to a Suitable Overpack

The procedure for unloading a DSC/RWC from the cask into an HSM or other authorized overpack is summarized in this section. This procedure is typical of NUHOMS® ISFSIs. Alternate procedures may be developed for plants with unique requirements.

1. Verify that the MP197HB cask *has been prepared as described* in Section A.7.1.1.
2. If the shear key plug assembly is not in place, install the shear key plug assembly.
3. Position the onsite transfer trailer in front of the module face.
4. Sample the cask cavity atmosphere through the vent port. Flush the cask interior gases if necessary.
5. Remove the cask ram *access* closure plate.
6. NOT USED.
7. Remove the HSM/overpack door.
8. Align the cask with the HSM/overpack.
9. Remove the cask lid.
10. If unloading any one of the smaller diameter DSC models (NUHOMS®-24PT4, 32PT, 24PTH, 61BT, or 61BTH) or *the* RWC from the MP197HB cask, *remove the cask spacer ring, and* install an unloading flange.
11. Dock the cask with the HSM/overpack and install the cask/HSM restraints.
12. NOT USED.
13. Extend the ram hydraulic cylinder and engage the grapple ring.
14. NOT USED.
15. Using the ram hydraulic cylinder move the DSC/RWC into the HSM/overpack.
16. NOT USED.
17. NOT USED.
18. Remove the cask/HSM restraints and move the cask away from the HSM/overpack.
19. Install the cask lid and cask ram *access* closure plate, if required.
20. Install the HSM/overpack door and seismic restraint, as applicable.
21. NOT USED.
22. NOT USED.

A.7.2.2.2 Unloading the NUHOMS®-MP197HB Cask to a Fuel Pool

The procedure for unloading the cask and DSC/RWC to a fuel pool is summarized in this section. Site specific conditions and requirements may require the use of different equipment and ordering of steps than those descried below to accomplish the same objectives or acceptance criteria which must be met to ensure the integrity of the package. Note that the NUHOMS®-MP197HB cask or an alternate suitable cask may be used for onsite movements of the DSC/RWC.

1. Verify that the NUHOMS®-MP197HB cask *has been prepared as described* in Section A.7.1.1.
2. Place the cask in the fuel receiving area.

3. NOT USED.
4. Rotate the cask to a vertical orientation and place the cask in the decon pit.
5. If the shear key plug assembly is not already in place, install the shear key plug assembly.
6. Sample the cask cavity atmosphere. Flush the cask interior gases if necessary.
7. Remove the lid from the cask.
8. NOT USED.
9. If the cask contains any one of the smaller diameter DSC models (NUHOMS®-24PT4, 32PT, 24PTH, 61BT, or 61BTH) or an RWC, remove the cask spacer ring at the top of the aluminum sleeve as shown in Drawing MP197HB-71-1014, Appendix A.1.4.10.1.
10. Fill the cask/DSC or cask/RWC annulus with water and install the cask/DSC or cask/RWC annulus seal.

After completion of the preparatory steps described above, follow the specific DSC unloading procedure as described in one of Appendices A.7.7.1 through A.7.7.9 as listed in Table A.7-4.

Section A.7.2.2.4 describes the procedures used for unloading of a NUHOMS®-MP197HB cask with an RWC.

A.7.2.2.3 Unloading the NUHOMS®-MP197HB Cask to a Dry Cell

The procedure for handling a DSC in a dry cell is highly dependent on the design of the dry cell and on the intended future use of the DSC. The procedure described below is intended to show the type of operations that will be performed and is not intended to be limiting.

1. Tow the onsite transfer trailer to the hot cell area.
2. NOT USED.
3. Using the cask lifting yoke, place the cask in the appropriate handling area.
4. Sample the cask cavity atmosphere. Flush the cask interior gases if necessary.
5. Install the shear key plug assembly, if required.
6. Remove the lid from the cask.
7. NOT USED.
8. Transfer the cask to the unloading area.
9. Remove the contents from the cask.
10. Decontaminate the cask as necessary.
11. NOT USED.

A.7.2.2.4 Horizontal Unloading of an RWC from the NUHOMS®-MP197HB Cask

This procedure is for handling a NUHOMS®-MP197HB cask with an RWC at a disposal site. The procedure described below is intended to show the type of operations that will be performed and is not intended to be limiting.

1. NOT USED.
2. Lift the cask and transfer it onto an unloading cradle.
3. NOT USED.
4. NOT USED.
5. NOT USED.
6. Remove the lid from the cask.
7. Install sealing surface protection, as appropriate.
8. Attach liner or waste removal tools.
9. Unload the cask contents into the disposal area.

A.7.3 Preparation of Empty Package for Transport

Previously used and empty NUHOMS®-MP197HB casks shall be prepared for transport per the requirements of 49 CFR 173.427 [2].

A.7.4 Other Operations

A.7.4.1 *Assembly Verification* Leakage Testing of the *NUHOMS®-MP197HB Cask* Containment Boundary

The procedure for *assembly verification* leakage testing of the cask containment boundary prior to shipment is given in this section. Assembly verification leakage testing shall conform to the requirements of ANSI N14.5 [1] or ISO -12807 [11]. A flow chart of the assembly verification leakage test is provided in Figure A.7-2. The order in which the leakage tests of the various seals are performed may vary. If more than one leakage detector is available then more than one seal may be tested at a time. Personnel performing the leakage test shall be specifically trained in leakage testing in accordance with SNT-TC-1A [7].

1. Remove the cask vent port plug.
2. Install the cask port tool in the cask vent port.
3. Open the cask vent port.
4. Attach a suitable vacuum pump to the cask port tool.
5. Reduce the cask cavity pressure to below 1.0 psia.
6. NOT USED.
7. Fill the cask cavity with helium to atmospheric pressure.
8. Close the vent port bolt.

9. Remove the helium-saturated cask port tool and install a clean (helium free) cask port tool.
10. Connect a mass spectrometer leak detector to the cask port tool.
11. Evacuate the vent port until the vacuum is sufficient to operate the leakage detection equipment.
12. Perform the leakage test. If the leakage rate is greater than $1x10^{-7}$ ref·cm^3/s, repair or replace the vent port bolt and/or seal, as required, and retest.

    NOTE: Upon removing the vent port bolt, it will be necessary to reduce the cask cavity pressure below 1.0 psia and refill with helium through the vent port.
13. Remove the leakage detection equipment.
14. Remove the cask port tool and replace the vent port plug.
15. Remove the lid test port plug.
16. Install the cask port tool in the lid test port.
17. Open the lid test port.
18. Connect the vacuum pump to the cask port tool.
19. Connect the leakage detector to the cask port tool.
20. Evacuate the lid test port until the vacuum is sufficient to operate the leakage detection equipment per the manufacturer's *directions*.
21. Perform the helium leakage test. If the leakage rate is greater than $1x10^{-7}$ ref·cm^3/s, repair or replace the cask lid or the cask lid O-ring seals, as required, and retest.

    NOTE: Upon removing and reinstalling the cask lid, it will be necessary to reduce the cask cavity pressure below 1.0 psia and refill with helium through the vent port. The vent port assembly verification leakage test must also be retested as described above.
22. Remove the leakage detection equipment.
23. Tighten the lid test port *bolt* in accordance with Drawing MP197HB-71-1002 in Chapter A.1, Appendix A.1.4.10.1. Remove the cask port tool from the lid test port and replace the lid test port plug.
24. Remove the cask drain port plug.
25. Install the cask port tool in the cask drain port.
26. Verify that the cask drain port is closed.
27. Connect the vacuum pump to the cask port tool.
28. Connect the leakage detector to the cask port tool.
29. Evacuate the drain port until the vacuum is sufficient to operate the leakage detection equipment.
30. Perform the leakage test. If the leakage rate is greater than $1x10^{-7}$ ref·cm^3/s, repair or replace the drain port bolt and/or seal, as required, and retest.

NOTE: Upon removing the drain port bolt, it will be necessary to reduce the cask cavity pressure below 1.0 psia and refill with helium through the vent port. The vent port assembly verification test must also be retested as described above.

31. Remove the leakage detection equipment.
32. Tighten the drain port bolt in accordance with Drawing MP197HB-71-1002 in Chapter A.1, Appendix A.1.4.10.1. Remove the cask port tool from the cask drain port and replace the drain port plug.
33. Remove the bottom test port plug.
34. Install the cask port tool in the bottom test port.
35. Open the bottom test port.
36. Connect the vacuum pump to the cask port tool.
37. Connect the leakage detector to the cask port tool.
38. Evacuate the bottom test port until the vacuum is sufficient to operate the leakage detection equipment *per the manufacturer's directions.*
39. Perform the helium leakage test. If the leakage rate is greater than $1x10^{-7}$ ref·cm^3/s, repair or replace the cask ram access closure plate or the cask ram access closure plate O-ring seals, as required, and retest.

    NOTE: Upon removing the cask ram access closure plate, it will be necessary to reduce the cask cavity pressure below 1.0 psia and refill with helium through the vent port. The vent port assembly verification test must also be retested as described above.

40. Remove the leakage detection equipment.
41. Tighten the bottom test port bolt in accordance with Drawing MP197HB-71-1002 in Chapter A.1, Appendix A.1.4.10.1. Remove the cask port tool from the bottom test port and replace the bottom test port plug.

This concludes the assembly verification leakage test procedure.

A.7.5 References

1. ANSI N14.5-1997, "American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment," American National Standards Institute, Inc., New York, 1997.
2. Title 49, Code of Federal Regulations, Part 173 (49 CFR 173), "Shippers - General Requirements for Shipments and Packaging."
3. Title 10, Code of Federal Regulations, Part 71 (10 CFR 71), "Packaging and Transportation of Radioactive Material."
4. U.S. Nuclear Regulatory Commission, Office of the Nuclear Material Safety and Safeguards, "Safety Evaluation of VECTRA Technologies' Response to Nuclear Regulatory Commission Bulletin 96-04 for the NUHOMS®-24P and NUHOMS®-7P."
5. U.S. Nuclear Regulatory Commission Bulletin 96-04, "Chemical, Galvanic or Other Reactions in Spent Fuel Storage and Transportation Casks," July 5, 1996.
6. Not Used.
7. SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing."
8. Updated Final Safety Analysis Report for The Standardized Advanced NUHOMS® Horizontal Modular Storage System For Irradiated Nuclear Fuel (CoC 1029) Revision 3.
9. Not used.
10. Not used.
11. ISO-12807, "Safety Transport of Radioactive Materials – Leakage Testing on Packages," First Edition, 1996.
12. NUREG-1927, March 2011, "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance," United States Nuclear Regulatory Commission.
13. EPRI Report No. 1013524, "Climatic Corrosion Considerations for Independent Spent Fuel Storage Installations in Marine Environments," Electric Power Research Institute, June 2006.
14. *"Repairing SCC of type 316 SS vessels" Materials Performance, September 2007, NACE International (p. 80).*
15. NUREG/CR-7030, "Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments," Page 47, Nuclear Regulatory Commission, October 2010.
16. E-33299, "Evaluation Procedure to Verify DSC Acceptance for Transport," Transnuclear, August 2012, *Revision 1.*
17. *Catherine Houska "Deicing Salt – Recognizing The Corrosion Threat" TMR Consulting, Pittsburg, PA http://www.imoa.info/_files/pdf/DeicingSalt.pdf*
18. *Greg Oberson, Darrel Dunn, Todd Mintz, Xihua He, Roberto Pabalan and Larry miller, "US NRC-Sponsored Research on Stress Corrosion Cracking Susceptibility of Dry Storage Canister Materials in Marine Environments – 13344" WM2013 Conference, February 24-28, 2013, Phoenix, Arizona USA, US NRC ADAMS, ML13029A490*

19. *Regulatory Guide, 1.21, Revision 2, June 2009, "Measuring, Evaluation, and Reporting Radioactive Material in Liquid Gaseous Effluents and Solid Waste," United States Nuclear Regulatory Commission.*

20. *NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel Final Report," March 2000.*

A.7.6 Glossary

The terms used in the above procedures are defined below.

annulus seal: Seal placed between the cask and DSC/RWC during operations in the fuel pool.

cask lifting yoke: Passive, open hook lifting yoke used for vertical lifts of the cask.

cask/HSM restraints: Provides the load path between the cask and HSM during DSC transfer operation.

conveyance: Any suitable conveyance such as a railcar, heavy haul trailer, barge, ship, etc.

horizontal storage module (HSM): Concrete shielded structure used for onsite storage of DSCs. HSM references herein refer to all models of HSM (e.g., HSM Model 80, Model 102, Model 152, Model 202, HSM-H, HSM-HS, AHSM, etc.) HSM also includes any other overpack authorized to accept a DSC or RWC via a horizontal transfer.

hydraulic ram: Hydraulic cylinder used to insert/withdraw DSCs to/from HSMs.

onsite transfer skid: Skid present on the onsite transfer trailer used to support the cask during onsite movements. Note in some cases the transportation skid may function as the onsite transfer skid.

onsite transfer trailer: A trailer used for onsite movements of the cask.

ram trunnion support assembly: Frame attached to the skid which provides an anchor for the hydraulic ram during DSC insertion and retrieval.

skid positioning system: Hydraulically operated alignment system that provides the interface between the onsite transfer trailer and the onsite transfer skid. It is used to align the skid (and cask) with the HSM prior to transfer.

A.7.7 Appendices

A.7.7.1 NUHOMS®-24PT4 DSC Wet Loading and Unloading

A.7.7.2 NUHOMS®-32PT DSC Wet Loading and Unloading

A.7.7.3 NUHOMS®-24PTH DSC Wet Loading and Unloading

A.7.7.4 NUHOMS®-32PTH DSC Wet Loading and Unloading

A.7.7.5 NUHOMS®-32PTH1 DSC Wet Loading and Unloading

A.7.7.6 NUHOMS®-37PTH DSC Wet Loading and Unloading

A.7.7.7 NUHOMS®-61BT DSC Wet Loading and Unloading

A.7.7.8 NUHOMS®-61BTH DSC Wet Loading and Unloading

A.7.7.9 NUHOMS®-69BTH DSC Wet Loading and Unloading

A.7.7.10 RWC Wet Loading

Table A.7-1
DSC, Fuel, and Basket Spacer Nominal Heights for Each Type of DSC

(All dimensions are in inches)

Canister Type	61BT	61BTH		69BTH	24PTH			24PT4	32PT				32PTH	32PTH	32PTH1			37PTH		RWC
		Type 1	Type 2		S	L	S-LC		S-100	S-125	L-100	L-125		Type 1	S	M	L	S	M	
DSC bottom spacer height(1)	2.20	2.20	2.20	1.24	11.7	5.7	11.7	2.2	11.7	11.7	5.7	5.7	12.5	5.25	12.5	5.25	N/A	16.25	9.0	11.75
DSC top spacer height	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)
Fuel spacer height	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	(2)(4)	N/A
Basket spacer height	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	(3)(4)	N/A

(1) DSC top and bottom spacers can be combined to one spacer. If one spacer is used, it can be installed either on top or bottom of the DSC. The height of the spacer is to be determined such that the gap between the cask and DSC is below 0.5" *for normal transport conditions.*

(2) Fuel spacer can be installed either on top or bottom of the fuel assembly. The height of the *fuel* spacer to be determined using the formula specified in Appendix A.2.13.14, Table A.2.13.14-2 *such that the gap between the fuel assemblies and the DSC is below 1.5" for normal transport conditions.*

(3) Basket spacer can be installed either on top or bottom of the basket. The height of the *basket* spacer is to be determined such that the gap between the basket and the DSC is below 0.815" *for normal transport conditions.*

(4) Fuel and basket spacers can be combined in one spacer.

Table A.7-2*a*
Applicable Fuel Specification for Various DSCs

DSC MODEL	Applicable Fuel Specification from Chapter A.1
NUHOMS®-24PT4	Tables A.1.4.1-1 and A.1.4.1-2
NUHOMS®-32PT	Table A.1.4.2-2
NUHOMS®-24PTH	Table A.1.4.3-2
NUHOMS®-32PTH	Table A.1.4.4-2
NUHOMS®-32PTH1	Table A.1.4.5-2
NUHOMS®-37PTH	Table A.1.4.6-2
NUHOMS®-61BT	Table A.1.4.7-2
NUHOMS®-61BTH	Table A.1.4.8-2
NUHOMS®-69BTH	Table A.1.4.9-1

Table A.7-*2b*
Applicable Content Specification for RWC

Type and Form of Material	*The NUHOMS®-MP197HB packaging is designed for shipment of various types of irradiated and contaminated reactor hardware. The payload will vary from shipment to shipment. Typical composition of the payload consists of the following components either individually or in combinations:* *1. BWR Control Rod Blades* *2. BWR Local Power Range Monitors (LPRMs)* *3. BWR Fuel Channels* *4. BWR Poison Curtains* *5. PWR Burnable Poison Rod Assemblies (BPRAs)* *6. PWR and BWR Reactor Vessel and Internals*
Decay Heat load	*≤ 5 kW*
Loading	*Components with high specific activity are generally placed near the center of the RWC. For each shipment, the RWC is normally filled to capacity, which prevents shifting of the contents during transport. If the RWC is not full, appropriate component spacers or shoring is used to prevent significant movement of the contents.*
Maximum Quantity of Material per Package	*(a) The quantity of radioactive material is limited to a maximum of 8,182 A_2. The radioactive material is primarily in the form of neutron activated metals, or metal oxides in solid form. Surface contamination may also be present on the irradiated components. When a wet load procedure (i.e., in-pool) is followed for cask loading, the cask cavity and RWC are drained and dried to ensure that there are no free liquids in the package during transport.* *(b) The NUHOMS®-MP197HB packaging is designed to transport a payload of up to 56.0 tons of dry irradiated and/or contaminated non-fuel bearing solid materials in the RWC.* *(c) The maximum quantity of non-fuel bearing radioactive material loaded into a package shall not exceed 8,182 A_2 (90,000 Ci of Co-60).*

Table A.7-3
Appendices Containing Loading Procedures for Various DSCs

DSC Model	Appendix
NUHOMS®-24PT4	A.7.7.1
NUHOMS®-32PT	A.7.7.2
NUHOMS®-24PTH	A.7.7.3
NUHOMS®-32PTH	A.7.7.4
NUHOMS®-32PTH1	A.7.7.5
NUHOMS®-37PTH	A.7.7.6
NUHOMS®-61BT	A.7.7.7
NUHOMS®-61BTH	A.7.7.8
NUHOMS®-69BTH	A.7.7.9
RWC	A.7.7.10

Table A.7-4
Appendices Containing Unloading Procedures for Various DSCs

DSC Model	Appendix
NUHOMS®-24PT4	A.7.7.1, Section A.7.7.1.4
NUHOMS®-32PT	A.7.7.2, Section A.7.7.2.4
NUHOMS®-24PTH	A.7.7.3, Section A.7.7.3.4
NUHOMS®-32PTH	A.7.7.4, Section A.7.7.4.4
NUHOMS®-32PTH1	A.7.7.5, Section A.7.7.5.4
NUHOMS®-37PTH	A.7.7.6, Section A.7.7.6.4
NUHOMS®-61BT	A.7.7.7, Section A.7.7.7.4
NUHOMS®-61BTH	A.7.7.8, Section A.7.7.8.4
NUHOMS®-69BTH	A.7.7.9, Section A.7.7.9.4

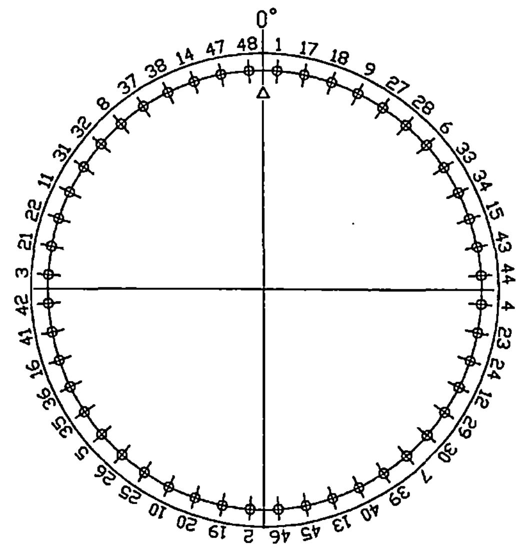
Table A.7-5

The Unloading Procedure which Shall Be Part of the User's Operating Procedures

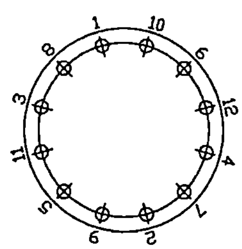
NOTE: In the event that breach and/or reconfiguration of fuel are confirmed during unloading of high burnup fuel assemblies, a written report should be generated in accordance with 10 CFR 71.95. *The report shall include the required corrective action(s). The corrective action(s) shall be implemented prior to resumption of transports.*

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Proprietary Information Withheld Pursuant to 10 CFR 2.390



MP197HB Cask Lid



Trunnion and Ram *Access* Closure Plate

Figure A.7-1
Torquing Patterns

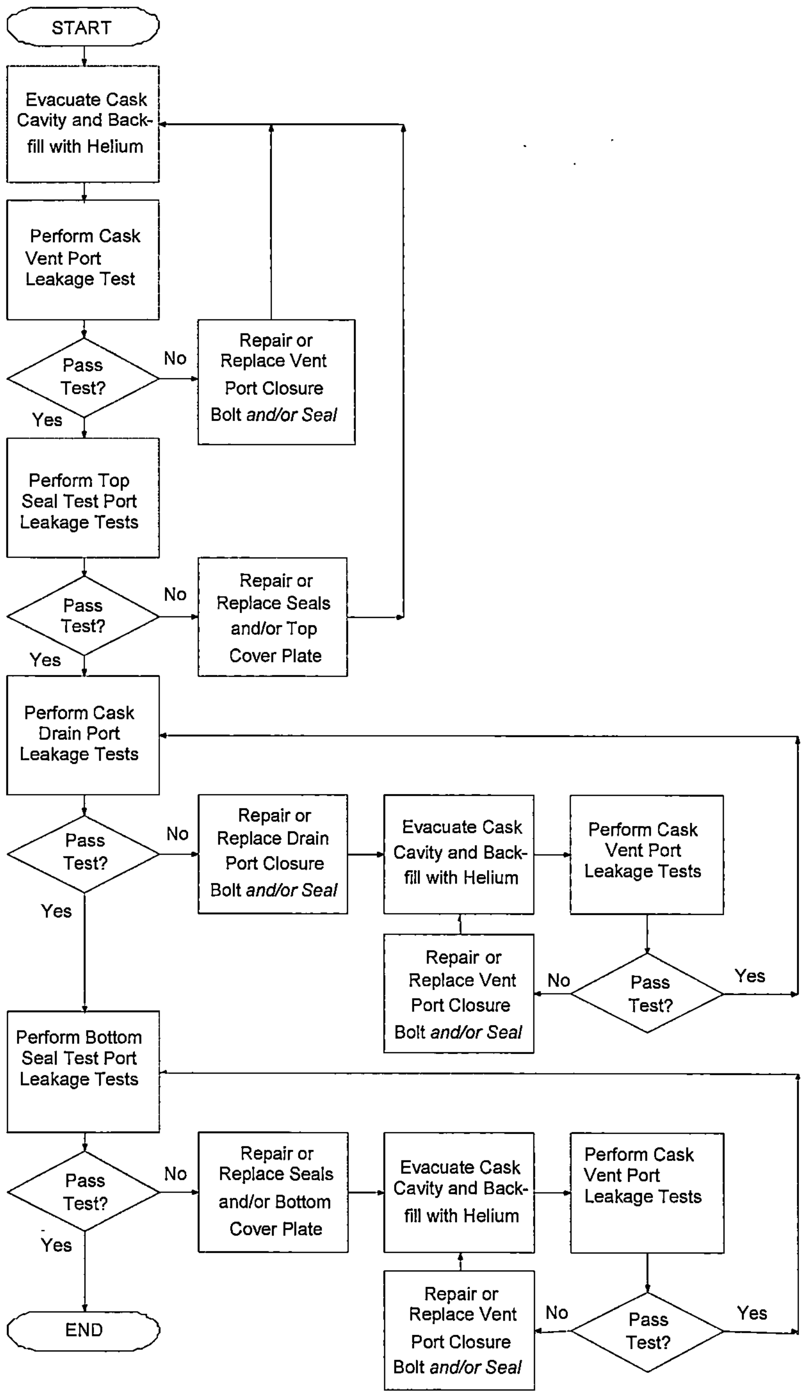


Figure A.7-2
Assembly Verification Leakage Test

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Figure A.7-3
DSC Evaluation for Transport Flowchart

Proprietary Information Withheld Pursuant to 10 CFR 2.390

Figure A.7-4
Example of Survey Point Locations on a Transport Cask